<table>
<tr><td colspan="12">Facility: Browns Ferry</td><td colspan="6">Date of Exam: May 17, 2021</td></tr>
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<tr><th>K1</th><th>K2</th><th>K3</th><th>K4</th><th>K5</th><th>K6</th><th>A1</th><th>A2</th><th>A3</th><th>A4</th><th>G*</th><th>Total</th><th colspan="2">A2</th><th colspan="2">G*</th><th>Total</th></tr>
<tr><td rowspan="3">1. Emergency and Abnormal Plant Evolutions</td><td>1</td><td>3</td><td>3</td><td>4</td><td colspan="3" rowspan="3">N/A</td><td>4</td><td>3</td><td colspan="2" rowspan="3">N/A</td><td>3</td><td>20</td><td colspan="2">4</td><td colspan="2">3</td><td>7</td></tr>
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<tr><td>Tier Totals</td><td>4</td><td>5</td><td>5</td><td>5</td><td>4</td><td>4</td><td>27</td><td colspan="2">6</td><td colspan="2">4</td><td>10</td></tr>
<tr><td rowspan="3">2. Plant Systems</td><td>1</td><td>3</td><td>2</td><td>2</td><td>2</td><td>2</td><td>1</td><td>3</td><td>2</td><td>3</td><td>3</td><td>3</td><td>26</td><td colspan="2">3</td><td colspan="2">2</td><td>5</td></tr>
<tr><td>2</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>2</td><td>1</td><td>12</td><td>0</td><td>2</td><td colspan="2">1</td><td>3</td></tr>
<tr><td>Tier Totals</td><td>4</td><td>3</td><td>3</td><td>3</td><td>3</td><td>2</td><td>4</td><td>3</td><td>4</td><td>5</td><td>4</td><td>38</td><td colspan="2">5</td><td colspan="2">3</td><td>8</td></tr>
<tr><td colspan="5" rowspan="2">3. Generic Knowledge and Abilities Categories</td><td colspan="2">1</td><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">4</td><td rowspan="2">10</td><td>1</td><td>2</td><td>3</td><td>4</td><td rowspan="2">7</td></tr>
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</table>

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 outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)

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 outline. Systems or evolutions that do not apply at the facility should be deleted with justification. 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. 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' 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 a category 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.

G* Generic K/As

* These systems/evolutions must be included as part of the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan. They are not required to be included when using earlier revisions of the K/A catalog.

** These systems/evolutions may be eliminated from the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan.

ES-401 BWR Examination Outline Form ES-401-1
Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/SRO)

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4					X	X	2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. AA2.03 Actual core flow	4.6 3.3	
295003 (APE 3) Partial or Complete Loss of AC Power / 6				X			AA1.03 Systems necessary to assure safe plant shutdown	4.4	
295004 (APE 4) Partial or Total Loss of DC Power / 6			X		X		AK3.02 Ground isolation/fault determination AA2.03 Battery voltage	2.9 2.9	
295005 (APE 5) Main Turbine Generator Trip / 3			X				AK3.05 Extraction steam/moisture separator isolations	2.5	
295006 (APE 6) Scram / 1					X	X	2.4.1 Knowledge of EOP entry conditions and immediate action steps. AA2.01 Reactor power	4.6 4.6	
295016 (APE 16) Control Room Abandonment / 7					X		AA2.05 Drywell pressure	3.8	
295018 (APE 18) Partial or Complete Loss of CCW / 8	X						AK1.01 Effects on component/system operations	3.5	
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8				X			AA1.02 Instrument air system valves: Plant-Specific	3.3	
295021 (APE 21) Loss of Shutdown Cooling / 4	X						AK1.01 Decay heat	3.6	
295023 (APE 23) Refueling Accidents / 8		X					AK2.03 Radiation monitoring equipment	3.4	
295024 High Drywell Pressure / 5			X			X	EK3.06 Reactor SCRAM 2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	4.0 4.2	
295025 (EPE 2) High Reactor Pressure / 3		X					EK2.08 Reactor/turbine pressure regulating system: Plant-Specific	3.7	
295026 (EPE 3) Suppression Pool High Water Temperature / 5					X	X	EA2.02 Suppression pool level 2.4.18 Knowledge of the specific bases for EOPs.	3.8 4.0	
295027 (EPE 4) High Containment Temperature (Mark III Containment Only) / 5							N/A for BFN		
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5			X		X		EK3.05 Reactor SCRAM EA2.01 Drywell temperature	3.6 4.1	
295030 (EPE 7) Low Suppression Pool Water Level / 5						X	2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems.	3.9	
295031 (EPE 8) Reactor Low Water Level / 2					X		EA2.01 Reactor water level	4.6	

295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1				X		X	EA1.04 SBLC 2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	4.5 4.2	
295038 (EPE 15) High Offsite Radioactivity Release Rate / 9				X			EA1.05 Post accident sample system (PASS): Plant-Specific	3.0	
600000 (APE 24) Plant Fire On Site / 8	X						AK1.01 Fire Classifications by type	2.5	
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6		X					AK2.02 Breakers, relays	3.1	
K/A Category Totals:							Group Point Total:		20/7

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/SRO)		Form ES-401-1
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295002 (APE 2) Loss of Main Condenser Vacuum / 3	X						AK1.04 Increased off-gas flow	3.0	
295007 (APE 7) High Reactor Pressure / 3									
295008 (APE 8) High Reactor Water Level / 2									
295009 (APE 9) Low Reactor Water Level / 2		X					AK2.04 Reactor water cleanup	2.6	
295010 (APE 10) High Drywell Pressure / 5					X		AA2.01 Leak rates	3.8	
295011 (APE 11) High Containment Temperature (Mark III Containment only) / 5									
295012 (APE 12) High Drywell Temperature / 5									
295013 (APE 13) High Suppression Pool Temperature. / 5									
295014 (APE 14) Inadvertent Reactivity Addition / 1									
295015 (APE 15) Incomplete Scram / 1					X		AA2.02 Control rod position	4.2	
295017 (APE 17) Abnormal Offsite Release Rate / 9					X		AA2.03 Radiation levels: Plant-Specific	3.1	
295020 (APE 20) Inadvertent Containment Isolation / 5 & 7				X			AA1.02 Drywell ventilation/cooling system	3.2	
295022 (APE 22) Loss of Control Rod Drive Pumps / 1									
295029 (EPE 6) High Suppression Pool Water Level / 5						X	2.2.22 Knowledge of limiting conditions for operations and safety limits.	4.7	
295032 (EPE 9) High Secondary Containment Area Temperature / 5			X				EK3.01 Emergency/normal depressurization	3.5	
295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9									
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9						X	2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	4.2	
295035 (EPE 12) Secondary Containment High Differential Pressure / 5									
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5		X					EK2.01 Secondary containment equipment and floor drain system	3.1	
500000 (EPE 16) High Containment Hydrogen Concentration / 5									
K/A Category Point Totals:							Group Point Total:		7/3

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode									X			A3.01 Valve operation	3.8	
205000 (SF4 SCS) Shutdown Cooling							X					A1.03 Recirculation loop temperatures	3.3	
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection					X							K5.06 Turbine speed measurement: BWR-2,3,4	2.6	
207000 (SF4 IC) Isolation (Emergency) Condenser												N/A for BFN		
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray								X			X	2.4.31 Knowledge of annunciator alarms, indications, or response procedures. A2.06 Inadequate system flow	4.2 3.2	
209002 (SF2, SF4 HPCS) High-Pressure Core Spray												N/A for BFN		
211000 (SF1 SLCS) Standby Liquid Control	X										X	K1.03 Plant air systems: Plant-Specific 2.2.40 Ability to apply Technical Specifications for a system.	2.5 4.7	
212000 (SF7 RPS) Reactor Protection			X									K3.05 RPS logic channels	3.7	
215003 (SF7 IRM) Intermediate-Range Monitor		X							X			A3.03 RPS status K2.01 IRM channels/detectors	3.7 2.5	
215004 (SF7 SRMS) Source-Range Monitor	X											K1.06 Reactor vessel	2.8	
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor						X						K6.04 Trip units	3.1	
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling		X						X				K2.02 RCIC initiation signals (logic) A2.07 Loss of lube oil	2.8 3.1	
218000 (SF3 ADS) Automatic Depressurization								X				A2.02 Large break LOCA	3.5	
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff										X	X	A4.01 Valve closures 2.1.32 Ability to explain and apply system limits and precautions.	3.6 3.8	
239002 (SF3 SRV) Safety Relief Valves											X X	**2.2.44 Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions.** **2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.**	4.2 4.6	
259002 (SF2 RWLCS) Reactor Water Level Control	X		X									K1.11 Drywell pressure: FWCI/HPCI K3.06 Main turbine	3.0 2.8	
261000 (SF9 SGTS) Standby Gas Treatment								X		X		A4.01 Off-site release levels: Plant-Specific A2.12 High fuel pool ventilation radiation: Plant-Specific	3.2 3.4	
262001 (SF6 AC) AC Electrical Distribution				X								K4.04 Protective relaying	2.8	
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)							X					A1.02 Motor generator outputs	2.5	

263000 (SF6 DC) DC Electrical Distribution				X			X					A1.01 Battery charging/discharging rate K4.01 Manual/ automatic transfers of control: Plant-Specific	2.5 3.1	
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG					X					X		A4.02 Synchroscope K5.05 Paralleling A.C. power sources	3.4 3.4	
300000 (SF8 IA) Instrument Air								X				A2.01 Air dryer and filter malfunctions	2.9	
400000 (SF8 CCS) Component Cooling Water									X			A3.01 Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS	3.0	
510000 (SF4 SWS*) Service Water (Normal and Emergency)												N/A- Sample plan generated using Rev.2 Supp. 1 of NUREG-123		
K/A Category Point Totals:												Group Point Total:		26/5

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201001 (SF1 CRDH) CRD Hydraulic														
201002 (SF1 RMCS) Reactor Manual Control										X		A4.05 Rod select matrix	3.1	
201003 (SF1 CRDM) Control Rod and Drive Mechanism											X	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.7	
201004 (SF7 RSCS) Rod Sequence Control														
201005 (SF1, SF7 RCIS) Rod Control and Information														
201006 (SF7 RWMS) Rod Worth Minimizer											X	2.4.6 Knowledge of EOP mitigation strategies.	3.7	
202001 (SF1, SF4 RS) Recirculation		X										K2.01 Recirculation pumps: Plant-Specific	3.2	
202002 (SF1 RSCTL) Recirculation Flow Control						X						K6.02 D.C. power	2.6	
204000 (SF2 RWCU) Reactor Water Cleanup														
214000 (SF7 RPIS) Rod Position Information														
215001 (SF7 TIP) Traversing In-Core Probe	X											K1.10 Area radiation monitoring system: (Not-BWR1)	2.6	
215002 (SF7 RBMS) Rod Block Monitor										X		A4.02 RBM back panel switches, meters and indicating lights: BWR-3,4,5	2.9	
216000 (SF7 NBI) Nuclear Boiler Instrumentation														
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode														
223001 (SF5 PCS) Primary Containment and Auxiliaries			X									K3.01 Secondary containment	3.6	
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode								X				A2.08 Pump seal failure	2.5	
230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Spray Mode														
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup														
234000 (SF8 FH) Fuel-Handling Equipment				X								K4.02 Prevention of control rod movement during core alterations	3.3	
239001 (SF3, SF4 MRSS) Main and Reheat Steam														
239003 (SF9 MSVLCS) Main Steam Isolation Valve Leakage Control														
241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating							X					A1.23 Main turbine vibration	2.8	
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary														
256000 (SF2 CDS) Condensate														
259001 (SF2 FWS) Feedwater								X				A2.06 Loss of A.C. electrical power	3.2	
268000 (SF9 RW) Radwaste														
271000 (SF9 OG) Offgas								X				A2.12 Recombiner high temperature	2.9	
272000 (SF7, SF9 RMS) Radiation Monitoring														
286000 (SF8 FPS) Fire Protection														
288000 (SF9 PVS) Plant Ventilation														
290001 (SF5 SC) Secondary Containment														
290003 (SF9 CRV) Control Room Ventilation									X			A3.01 Initiation/reconfiguration	3.3	
290002 (SF4 RVI) Reactor Vessel Internals					X							K5.03 Burnable poisons	2.7	
51001 (SF8 CWS*) Circulating Water												N/A- Sample plan generated using Rev.2 Supp. 1 of NUREG-123		

K/A Category Point Totals:												Group Point Total:		12/3

Facility: Browns Ferry		Date of Exam: May 2021				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements.	3.8			
	2.1.29	Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	4.1			
	2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.			4.7	
	2.1.41	Knowledge of the refueling process.			3.7	
	2.1.					
	2.1.					
	Subtotal			2		2
2. Equipment Control	2.2.35	Ability to determine Technical Specification Mode of Operation.	3.6			
	2.2.41	Ability to obtain and interpret station electrical and mechanical drawings.	3.5			
	2.2.12	Knowledge of surveillance procedures.	3.7			
	2.2.14	Knowledge of the process for controlling equipment configuration or status.			4.3	
	2.2.6	Knowledge of the process for making changes to procedures.			3.6	
	2.2.					
	Subtotal			3		2
3. Radiation Control	2.3.13	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	3.4			
	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2			
	2.3.5	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9			
	2.3.11	Ability to control radiation releases.			4.3	
	2.3.					
	2.3.					
	Subtotal			3		1
4. Emergency Procedures/Plan	2.4.27	Knowledge of “fire in the plant” procedures.	3.4			
	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.0			
	2.4.40	2.4.40 Knowledge of SRO responsibilities in emergency plan implementation.			4.5	

	2.4.43	Knowledge of emergency communications systems and techniques.			3.8	
	2.4.					
	2.4.					
	Subtotal					
Tier 3 Point Total				10		7

Facility Browns Ferry NPP	Date of Examination: 5/17/21
Examination Level: RO ☒ SRO ☐	Operating Test Number: 21-04

Administrative Topic (see Note)	Type Code*	Describe activity to be performed	
Conduct of Operations	R, M	JPM 516	Determine Control Rod Withdrawal Requirements
		K/A 2.1.37 (RO 4.3)	Knowledge of procedures, guidelines, or limitations associated with reactivity management.
Conduct of Operations	R, D	JPM 745	Place an RPS Channel in Trip
		K/A 2.1.25 (RO 3.9)	Ability to interpret reference materials, such as graphs, curves, tables, etc.
Equipment Control	R, D	JPM 510	Evaluate Recombiner Performance
		K/A 2.2.44 (RO 4.2)	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.
Radiation Control	R, D	JPM 682	Review a Radiological Work Permit (RWP)
		K/A 2.3.7 (RO 3.5)	Ability to comply with radiation work permit requirements during normal or abnormal conditions.
Emergency Plan	N/A	N/A	

NOTE: All items (five total) are required for SROs. RO applicants require only four items unless they are retaking only the administrative topics (which would require all five items).

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

Reactor Operator

1. Conduct of Ops - Determine Control Rod Withdrawal Requirements

Given initial and current SRM readings, determine how Control Rods should be withdrawn in accordance with GOI-100-1A, Unit Startup and Power Operation.

K/A 2.1.37: Knowledge of procedures, guidelines, or limitations associated with reactivity management. (RO 4.3)

2. Conduct of Ops – Place an RPS Channel in Trip

Given a failed Reactor High Pressure RPS instrument, determine how to place the RPS instrument channel in trip in accordance with OI-99, Reactor Protection System.

K/A 2.1.25: Ability to interpret reference materials, such as graphs, curves, tables, etc. (RO 3.9)

3. Equipment Control - Evaluate Recombiner Performance

Evaluate Off-Gas Recombiner Performance to determine if it meets Acceptance Criteria in accordance OI-66, Off-Gas System.

K/A 2.2.44. Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (RO 4.2)

4. Radiation Control – Review a Radiological Work Permit (RWP)

Given an RWP and dose rates for a task to be performed, calculate the expected dose to determine if the task can or cannot be performed in accordance with NPG-SPP-05.18, Radiation Work Permits.

K/A 2.3.7: Ability to comply with radiation work permit requirements during normal or abnormal conditions. (RO 3.5)

5. Emergency Plan – N/A

Facility Browns Ferry NPP

Date of Examination: 5/17/21

Examination Level: RO ☐ SRO ☒

Operating Test Number: 21-04

Administrative Topic (see Note)	Type Code*	Describe activity to be performed	
Conduct of Operations	R, D	JPM 678	Determine Crew Shift Staffing Requirements
		K/A 2.1.5 (SRO 3.9)	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.
Conduct of Operations	R, D	JPM 745	Place an RPS Channel in trip and determine REQUIRED ACTIONS in accordance with Technical Specifications
		K/A 2.1.25 (SRO 4.2)	Ability to interpret reference materials, such as graphs, curves, tables, etc.
Equipment Control	R, N	JPM 746	Review a completed Surveillance (SR)
		K/A 2.2.22 (SRO 4.7)	Knowledge of limiting conditions for operations and safety limits.
Radiation Control	R, D	JPM 682	Review a Radiological Work Permit
		K/A 2.3.7 (SRO 3.6)	Ability to comply with radiation work permit requirements during normal or abnormal conditions.
Emergency Plan	R, N	JPM 738	Emergency Action Level Classification
		K/A 2.4.41 (SRO 4.6)	Knowledge of the Emergency Action Level thresholds and Classifications

NOTE: All items (five total) are required for SROs. RO applicants require only four items unless they are retaking only the administrative topics (which would require all five items).

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

Senior Reactor Operator

1. **Conduct of Ops** - Determine Crew Shift Staffing Requirements

   Given a Shift Manager's Staffing Sheet, determine if Shift Staffing Requirements are met or if a callout is required in accordance with OPDP-1, Conduct of Operations Attachment 1, NPG-SPP-03.21, Nuclear Fatigue Management Program, Section 3.2.7, 2.a., and OSIL-25, TVA BFN Operations Section Instruction Letter Overtime, Leave, and Relief Policy, Attachment 2.

   K/A 2.1.5: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (SRO 3.9)

2. **Conduct of Ops** – Place an RPS Channel in trip and determine REQUIRED ACTIONS in accordance with Technical Specifications

   Given a failed Reactor High Pressure RPS instrument, determine Technical Specification 3.3.1.1, RPS Instrumentation REQUIRED ACTION and how to place the RPS instrument channel in trip in accordance with 2-OI-99, Reactor Protection System.

   K/A 2.1.25: Ability to interpret reference materials, such as graphs, curves, tables, etc. (SRO 4.2)

3. **Equipment Control** – Review a completed Surveillance (SR)

   Given a completed 3-SR-3.8.7.1, Weekly Check of Power Availability to Required AC and DC Power Distribution Subsystems, determine if the SR has been completed correctly and address Technical Specification Requirements.

   K/A 2.2.22: Knowledge of limiting conditions for operations and safety limits. (SRO 4.7)

4. **Radiation Control** – Review a Radiological Work Permit (RWP)

   Given an RWP and dose rates for a task to be performed, calculate the expected dose to determine if the task can or cannot be performed in accordance with NPG-SPP-05.18, Radiation Work Permits. Additionally, determine if the task requires additional authorization in accordance with the TVA Radiological Annual Administrative Dose Level Program.

   K/A 2.3.7: Ability to comply with radiation work permit requirements during normal or abnormal conditions. (SRO 3.6)

5. Emergency Plan – Emergency Action Level Classification

Given plant conditions, classifies an Event as an Alert (RA2), and completes the Initial Notification Form with correct information within the required time in accordance with the EPIPs.

K/A 2.4.41: Knowledge of the Emergency Action Level thresholds and classifications. (SRO 4.6)

Facility: Browns Ferry NPP Date of Examination: 5/17/21

Exam Level: ☒ RO ☐ SRO-I ☐ SRO-U Operating Test No.: 21-04

Control Room Systems:@ 8 for RO, 7 for SRO-I, 2 or 3 for SRO-U, including 1 ESF			
System / JPM Title		Type Code*	Safety Function
a. JPM 80A	Respond to a Control Rod Drift in accordance with AOI-85-5, Rod Drift In	A, D, S	1
b. JPM 18A	Inject to the Reactor in accordance with EOI Appendix-5C, Injection System Lineup-RCIC	A, D, P, L, S	2
c. JPM 743A	Alternate Generator Bus Duct Fans in accordance with OI-47, Turbine-Generator System	A, N, S	4
d. JPM 747	Purge the Drywell with the Primary Containment Purge Filter Fan in accordance with OI-64, Primary Containment System	L, N, S	5
e. JPM 631	Restore Offsite Power to a 4KV Shutdown Board at Panel 9-23 in accordance with 0-OI-82, Standby Diesel Generator (EDG) System	D, S	6
f. JPM 748	Recover from a loss of RPS in accordance with AOI-99-1, Loss of Power to One RPS Bus	N, S	7
g. JPM 602A	Respond to a Loss of Reactor Building Closed Cooling Water (RBCCW) in accordance with AOI-70-1, Loss of Reactor Building Closed Cooling Water	A, D, S	8
h. JPM 55	Emergency Vent Primary Containment in accordance with EOI Appendix-13, Emergency Venting Primary Containment	D, EN, L, S	9
In-Plant Systems:@ 3 for RO, 3 for SRO-I, 3 or 2 for SRO-U			
i. JPM 247	Perform Field Actions for a Stuck Open Main Steam Relief Valve (MSRV) per AOI-1-1, Relief Valve Stuck Open	D, E, EN	3
j. JPM 733A	Locally Start an EHPM Pump per EOI Appendix-7L, Alternate Injection System Lineup EHPM System	A, E, N, R	2
k. JPM 306	Place the Division I ECCS ATU Inverter in Service per 0-OI-57C, 208V / 120V AC Electrical System	D, L	6

@ 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
(C)ontrol room	
(D)irect from bank	≤ 9/≤ 8/≤4
(E)mergency or abnormal in-plant	≥ 1/≥ 1/≥1
(EN)gineered safety feature	≥1/≥1/≥1 (control room system)
(L)ow-Power / Shutdown	≥1/≥1/≥1
(N)ew or (M)odified from bank including 1(A)	≥ 2/≥ 2/≥1
(P)revious 2 exams	≤ 3/≤ 3/≤ 2 (randomly selected)
(R)CA	≥1/≥1/≥1
(S)imulator	

Reactor Operator

Job Performance Measures

a. JPM 80A

Title: Respond to a Control Rod Drift in accordance with AOI-85-5, Rod Drift In

Description: The candidate will perform Surveillance 3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3, Exercising a Partially Withdrawn Control Rod. During the performance of the surveillance, a Control Rod will drift in, requiring the candidate to respond in accordance with AOI-85-5, Rod Drift In. During the actions required by AOI-85-5, other Control Rods will drift in and the candidate will insert a manual Reactor SCRAM.

K/A: 201002 Reactor Manual Control System A2.02; Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rod Drift Alarm (3.2)

b. JPM 18A

Title: Inject to the Reactor in accordance with EOI Appendix-5C, Injection System Lineup - RCIC

Description: The candidate will inject to the Reactor using Reactor Core Isolation Cooling (RCIC) to maintain Reactor Water Level in accordance with EOI-Appendix 5C, Injection System Lineup – RCIC. After injection has begun, the RCIC Flow Controller will fail to operate in automatic, and the candidate will take manual control of the RCIC Flow Controller to continue injection.

K/A 217000 Reactor Core Isolation Cooling System (RCIC) A1.01; Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: RCIC Flow (3.7)

c. JPM 743A **Title:** Alternate Generator Bus Duct Fans in accordance with OI-47, Turbine-Generator System

Description: The candidate will alternate Turbine-Generator Bus Duct Cooling Fans in accordance with OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans. When the standby Bus Duct Fan is started it will trip, and no Bus Duct Fans will be able to be started. The candidate will respond in accordance with Alarm Response Procedures and insert a manual Reactor SCRAM and trip the Main Turbine.

K/A 245000 Main Turbine Generator and Auxiliary Systems A4.02: Ability to manually operate and/or monitor in the control room: Generator controls (3.1)

d. JPM 747 **Title:** Purge the Drywell with the Primary Containment Purge Filter Fan in accordance with OI-64, Primary Containment System

Description: The candidate will perform operations necessary to air purge the Drywell with the Primary Containment Purge Filter Fan for Drywell entry in accordance with OI-64, Purging the Drywell with Primary Containment Purge Filter Fan, Section 8.2.

K/A 223001 Primary Containment System and Auxiliaries A4.05: Ability to manually operate and/or monitor in the Control Room: Containment/Drywell oxygen concentration (3.6)

e. JPM 631 **Title:** Restore Offsite Power to a 4KV Shutdown Board at Panel 9-23 in accordance with 0-OI-82, Standby Diesel Generator (EDG) System

Description: The candidate will restore Offsite Power to a 4KV Shutdown Board in accordance with OI-82, Standby Diesel Generator System, Section 8.3, Restoring Offsite Power to 4-kV Shutdown Board at Panel 9-23. The candidate will parallel Offsite power to the running Diesel Generator.

K/A 262001 A.C. Electrical Distribution; A4.02 Ability to manually operate and/or monitor in the control room: Synchroscope, including understanding of running and incoming voltages (3.4)

f. JPM 748

Title: Recover from a loss of RPS in accordance with AOI-99-1, Loss of Power to One RPS Bus

Description: The candidate will perform operations required to restore systems following a loss of one RPS Bus in accordance with AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

K/A

212000 Reactor Protection System A2.01; Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RPS motor-generator set failure (3.7)

g. JPM 602A

Title: Respond to a loss of Reactor Building Closed Cooling Water (RBCCW) in accordance with AOI-70-1, Loss Reactor Building Closed Cooling Water

Description: The candidate will respond to a trip of an RBCCW pump in accordance with AOI-70-1, Loss of Reactor Building Closed Cooling Water. While performing actions in accordance with AOI-70-1, a total loss of RBCCW will occur, forcing the Operator to insert a Core Flow Runback and a manual Reactor SCRAM.

K/A

400000 Component Cooling Water System A2.01: Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Loss of CCW pump (3.3)

h. JPM 55

Title: Emergency Vent Primary Containment in accordance with EOI Appendix-13, Emergency Venting Primary Containment

Description: The candidate will perform operations required to Emergency Vent the Primary Containment in accordance with EOI-Appendix-13, Emergency Venting Primary Containment.

K/A

288000 Plant Ventilation Systems A2.01: Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High Drywell Pressure: Plant-Specific (3.3)

i. JPM 247 **Title:** Perform Field Actions for Stuck Open Main Steam Relief Valve (MSRV) in accordance with AOI-1-1, Relief Valve Stuck Open

Description: The candidate will perform field actions necessary to close a stuck Open MSRV in accordance with AOI-1-1, Relief Valve Stuck Open, Step 4.2.3[2].

K/A 239002 Relief/Safety Relief Valves A2.03; Ability to (a) predict the impacts of the following on the RELIEF/SAFETY RELIEF VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV (4.1)

j. JPM 733A **Title:** Locally Start an EHPM Pump in accordance with EOI Appendix-7L, Alternate Injection System Lineup EHPM System

Description: The candidate will perform the actions necessary in accordance with EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System to locally start an EHPM from the local control panel (LPNL-925-6000) to raise Reactor Water Level to (+)2 to (+)51 inches. The candidate will be required to take action to provide a power source to the EHPM in accordance with Attachment 1, EHPM Pump Operation from Local Control Panel LNPL-925-6000.

K/A 295009 Low Reactor Water Level AA1.02; Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor Water Level Control (4.0)

k. JPM 306 **Title:** Place the Division I ECCS ATU Inverter in Service in accordance with 0-OI-57C, 208V / 120V AC Electrical System

Description: The candidate will perform operations necessary to return the Division I ECCS Analog Trip Unit Inverter to service in accordance with 0-OI-57C, 208V/120V AC Electrical System.

K/A 263000 D.C. Electrical Distribution A3.01; Ability to monitor automatic operations of D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights (3.2)

Facility: Browns Ferry NPP

Date of Examination: 5/17/21

Exam Level: ☐ RO ☒ SRO-I ☐ SRO-U

Operating Test No.: 21-04

Control Room Systems:[@] 8 for RO, 7 for SRO-I, 2 or 3 for SRO-U, including 1 ESF

System / JPM Title		Type Code*	Safety Function
a. JPM 80A	Respond to a Control Rod Drift in accordance with AOI-85-5, Rod Drift In	A, D, S	1
b. JPM 18A	Inject to the Reactor in accordance with EOI Appendix-5C, Injection System Lineup-RCIC	A, D, P, L, S	2
c. JPM 743A	Alternate Generator Bus Duct Fans in accordance with OI-47, Turbine-Generator System	A, N, S	4
d. JPM 747	Purge the Drywell with the Primary Containment Purge Filter Fan in accordance with OI-64, Primary Containment System	L, N, S	5
e. N/A			
f. JPM 748	Recover from a loss of RPS in accordance with AOI-99-1, Loss of Power to One RPS Bus	N, S	7
g. JPM 602A	Respond to a Loss of Reactor Building Closed Cooling Water (RBCCW) in accordance with AOI-70-1, Loss of Reactor Building Closed Cooling Water	A, D, S	8
h. JPM 55	Emergency Vent Primary Containment in accordance with EOI Appendix-13, Emergency Venting Primary Containment	D, EN, L, S	9
In-Plant Systems:[@] 3 for RO, 3 for SRO-I, 3 or 2 for SRO-U			
i. JPM 247	Perform Field Actions for a Stuck Open Main Steam Relief Valve (MSRV) in accordance with AOI-1-1, Relief Valve Stuck Open	D, E, EN	3
j. JPM 733A	Locally Start an EHPM Pump in accordance with EOI Appendix-7L, Alternate Injection System Lineup EHPM System	A, E, N, R	2
k. JPM 306	Place the Division I ECCS ATU Inverter in Service in accordance with0-OI-57C, 208V / 120V AC Electrical System	D, L	6

[@] 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
(C)ontrol room	
(D)irect from bank	≤ 9/≤ 8/≤4
(E)mergency or abnormal in-plant	≥ 1/≥ 1/≥1
(EN)gineered safety feature	≥1/≥1/≥1 (control room system)
(L)ow-Power / Shutdown	≥1/≥1/≥1
(N)ew or (M)odified from bank including 1(A)	≥ 2/≥ 2/≥1
(P)revious 2 exams	≤ 3/≤ 3/≤ 2 (randomly selected)
(R)CA	≥1/≥1/≥1
(S)imulator	

Senior Reactor Operator (Instant)

Job Performance Measures

a. JPM 80A **Title:** Respond to a Control Rod Drift in accordance with AOI-85-5, Rod Drift In

Description: The candidate will perform Surveillance 3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3, Exercising a Partially Withdrawn Control Rod. During the performance of the surveillance, a Control Rod will drift in, requiring the candidate to respond in accordance with AOI-85-5, Rod Drift In. During the actions required by AOI-85-5, other Control Rods will drift in and the candidate will insert a manual Reactor SCRAM.

K/A: 201002 Reactor Manual Control System A2.02; Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rod Drift Alarm (3.3)

b. JPM 18A **Title:** Inject to the Reactor in accordance with EOI Appendix-5C, Injection System Lineup - RCIC

Description: The candidate will inject to the Reactor using Reactor Core Isolation Cooling (RCIC) to maintain Reactor Water Level in accordance with EOI-Appendix 5C, Injection System Lineup – RCIC. After injection has begun, the RCIC Flow Controller will fail to operate in automatic, and the candidate will take manual control of the RCIC Flow Controller to continue injection.

K/A 217000 Reactor Core Isolation Cooling System (RCIC) A1.01; Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: RCIC Flow (3.7)

c. JPM 743A **Title:** Alternate Generator Bus Duct Fans in accordance with OI-47, Turbine-Generator System

Description: The candidate will alternate Turbine-Generator Bus Duct Cooling Fans in accordance with OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans. When the standby Bus Duct Fan is started it will trip, and no Bus Duct Fans will be able to be started. The candidate will respond in accordance with Alarm Response Procedures and insert a manual Reactor SCRAM and trip the Main Turbine.

K/A 245000 Main Turbine Generator and Auxiliary Systems A4.02: Ability to manually operate and/or monitor in the control room: Generator controls (2.9)

d. JPM 747 **Title:** Purge the Drywell with the Primary Containment Purge Filter Fan in accordance with OI-64, Primary Containment System

Description: The candidate will perform operations necessary to air purge the Drywell with the Primary Containment Purge Filter Fan for Drywell entry in accordance with OI-64, Purging the Drywell with Primary Containment Purge Filter Fan, Section 8.2.

K/A 223001 Primary Containment System and Auxiliaries A4.05: Ability to manually operate and/or monitor in the Control Room: Containment/Drywell oxygen concentration (3.6)

e. N/A

f. JPM 748 **Title:** Recover from a loss of RPS in accordance with AOI-99-1, Loss of Power to One RPS Bus

Description: The candidate will perform operations required to restore systems following a loss of one RPS Bus in accordance with AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

K/A 212000 Reactor Protection System A2.01; Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RPS motor-generator set failure (3.9)

g. JPM 602A **Title:** Respond to a loss of Reactor Building Closed Cooling Water (RBCCW) in accordance with AOI-70-1, Loss Reactor Building Closed Cooling Water

Description: The candidate will respond to a trip of an RBCCW pump in accordance with AOI-70-1, Loss of Reactor Building Closed Cooling Water. While performing actions in accordance with AOI-70-1, a total loss of RBCCW will occur, forcing the Operator to insert a Core Flow Runback and a manual Reactor SCRAM.

K/A 400000 Component Cooling Water System A2.01: Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Loss of CCW pump (3.4)

h. JPM 55 **Title:** Emergency Vent Primary Containment in accordance with EOI Appendix-13, Emergency Venting Primary Containment

Description: The candidate will perform operations required to Emergency Vent the Primary Containment in accordance with EOI-Appendix-13, Emergency Venting Primary Containment.

K/A 288000 Plant Ventilation Systems A2.01: Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High Drywell Pressure: Plant-Specific (3.4)

i. JPM 247 **Title:** Perform Field Actions for Stuck Open Main Steam Relief Valve (MSRV) in accordance with AOI-1-1, Relief Valve Stuck Open

Description: The candidate will perform field actions necessary to close a stuck Open MSRV in accordance with AOI-1-1, Relief Valve Stuck Open, Step 4.2.3[2].

K/A 239002 Relief/Safety Relief Valves A2.03; Ability to (a) predict the impacts of the following on the RELIEF/SAFETY RELIEF VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV (4.2*)

j.	JPM 733A	**Title:** Locally Start an EHPM Pump in accordance with EOI Appendix-7L, Alternate Injection System Lineup EHPM System
		Description: The candidate will perform the actions necessary in accordance with EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System to locally start an EHPM from the local control panel (LPNL-925-6000) to raise Reactor Water Level to (+)2 to (+)51 inches. The candidate will be required to take action to provide a power source to the EHPM in accordance with Attachment 1, EHPM Pump Operation from Local Control Panel LNPL-925-6000.
	K/A	295009 Low Reactor Water Level AA1.02; Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor Water Level Control (4.0)
k.	JPM 306	**Title:** Place the Division I ECCS ATU Inverter in Service in accordance with 0-OI-57C, 208V / 120V AC Electrical System
		Description: The candidate will perform operations necessary to return the Division I ECCS Analog Trip Unit Inverter to service in accordance with 0-OI-57C, 208V/120V AC Electrical System.
	K/A	263000 D.C. Electrical Distribution A3.01; Ability to monitor automatic operations of D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights (3.2)

Facility: Browns Ferry NPP

Date of Examination: 5/17/21

Exam Level: ☐ RO ☐ SRO-I ☒ SRO-U

Operating Test No.: 21-04

Control Room Systems:@ 8 for RO, 7 for SRO-I, 2 or 3 for SRO-U, including 1 ESF

System / JPM Title		Type Code*	Safety Function
a. N/A			
b. N/A			
c. JPM 743A	Alternate Generator Bus Duct Fans in accordance with OI-47, Turbine-Generator System	A, N, S	4
d. N/A			
e. N/A			
f. N/A			
g. JPM 602A	Respond to a Loss of Reactor Building Closed Cooling Water (RBCCW) in accordance with AOI-70-1, Loss of Reactor Building Closed Cooling Water	A, D, S	8
h. JPM 55	Emergency Vent Primary Containment in accordance with EOI Appendix-13, Emergency Venting Primary Containment	D, EN, L, S	9
In-Plant Systems:@ 3 for RO, 3 for SRO-I, 3 or 2 for SRO-U			
i. N/A			
j. JPM 733A	Locally Start an EHPM Pump in accordance with EOI Appendix-7L, Alternate Injection System Lineup EHPM System	A, E, N, R	2
k. JPM 306	Place the Division I ECCS ATU Inverter in Service in accordance with 0-OI-57C, 208V / 120V AC Electrical System	D, L	6

@ 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
(C)ontrol room	
(D)irect from bank	≤ 9/≤ 8/≤4
(E)mergency or abnormal in-plant	≥ 1/≥ 1/≥1
(EN)gineered safety feature	≥1/≥1/≥1 (control room system)
(L)ow-Power / Shutdown	≥1/≥1/≥1
(N)ew or (M)odified from bank including 1(A)	≥ 2/≥ 2/≥1
(P)revious 2 exams	≤ 3/≤ 3/≤ 2 (randomly selected)
(R)CA	≥1/≥1/≥1
(S)imulator	

Senior Reactor Operator (Upgrade)

Job Performance Measures

a. N/A

b. N/A

c. JPM 743A **Title:** Alternate Generator Bus Duct Fans in accordance with OI-47, Turbine-Generator System

Description: The candidate will alternate Turbine-Generator Bus Duct Cooling Fans in accordance with OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans. When the standby Bus Duct Fan is started it will trip, and no Bus Duct Fans will be able to be started. The candidate will respond in accordance with Alarm Response Procedures and insert a manual Reactor SCRAM and trip the Main Turbine.

K/A 245000 Main Turbine Generator and Auxiliary Systems A4.02: Ability to manually operate and/or monitor in the control room: Generator controls (2.9)

d. N/A

e. N/A

f. N/A

g. JPM 602A **Title:** Respond to a loss of Reactor Building Closed Cooling Water (RBCCW) in accordance with AOI-70-1, Loss Reactor Building Closed Cooling Water

Description: The candidate will respond to a trip of an RBCCW pump in accordance with AOI-70-1, Loss of Reactor Building Closed Cooling Water. While performing actions in accordance with AOI-70-1, a total loss of RBCCW will occur, forcing the Operator to insert a Core Flow Runback and a manual Reactor SCRAM.

K/A 400000 Component Cooling Water System A2.01: Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Loss of CCW pump (3.3)

h. JPM 55 **Title:** Emergency Vent Primary Containment in accordance with EOI Appendix-13, Emergency Venting Primary Containment

Description: The candidate will perform operations required to Emergency Vent the Primary Containment in accordance with EOI-Appendix-13, Emergency Venting Primary Containment.

K/A 288000 Plant Ventilation Systems A2.01: Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High Drywell Pressure: Plant-Specific (3.4)

i. N/A

j. JPM 733A **Title:** Locally Start an EHPM Pump in accordance with EOI Appendix-7L, Alternate Injection System Lineup EHPM System

Description: The candidate will perform the actions necessary in accordance with EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System to locally start an EHPM from the local control panel (LPNL-925-6000) to raise Reactor Water Level to (+)2 to (+)51 inches. The candidate will be required to take action to provide a power source to the EHPM in accordance with Attachment 1, EHPM Pump Operation from Local Control Panel LNPL-925-6000.

K/A 295009 Low Reactor Water Level AA1.02; Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor Water Level Control (4.0)

k. JPM 306 **Title:** Place the Division I ECCS ATU Inverter in Service in accordance with 0-OI-57C, 208V / 120V AC Electrical System

Description: The candidate will perform operations necessary to return the Division I ECCS Analog Trip Unit Inverter to service in accordance with 0-OI-57C, 208V/120V AC Electrical System.

K/A 263000 D.C. Electrical Distribution A3.01; Ability to monitor automatic operations of D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights (3.2)

Examination Outline Cross-reference:	Level	RO	SRO
202001 (SF1, SF4 RS) Recirculation	Tier #	2	------
K2.01 (10CFR 55.41.7)	Group #	2	------
Knowledge of electrical power supplies to the following: • Recirculation pumps: Plant-Specific	K/A #	202001K2.01	
	Importance Rating	3.2*	------

Proposed Question: **# 1**

With respect to Recirc Pump power supplies, which **ONE** of the following completes the statement below?

2B Recirc Pump is **NORMALLY** powered by ________.

Note: Unit Station Service Transformer (USST)

A. USST 2A

B. USST 2B

C. Start Bus 2A

D. Start Bus 2B

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System, 2B Recirc Pump is normally powered by USST 2A.

B INCORRECT: Incorrect but plausible in that the Browns Ferry Electrical System is complex and often confused; normally 'B' motors are powered by a 'B' board.

C INCORRECT: Incorrect by plausible in that 4KV Start Bus 2A is an alternate power supply to the 4KV Recirc Boards.

D INCORRECT: Incorrect by plausible in that 4KV Start Bus 2B is an alternate power supply to the 4KV Recirc Boards.

RO Level Justification: Tests the candidate's knowledge of the power supplies for the Recirculation Pumps. This question is rated as memory due to strictly recalling facts related to the electrical distribution system.

Technical Reference(s): 0-OI-57A, Rev. 166 (Attach if not previously provided)
AC Distribution PIP 02-03, 7/30/19

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.007 Obj,3 (As available)

Question Source:

Bank #	BFN 1703 #56	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam 2017

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Copy of Bank Question:

QUESTION 56 rev 1

Which one of the following completes the statement below?

1B Recirc VFD is normally powered by ________.

A. USST 1A

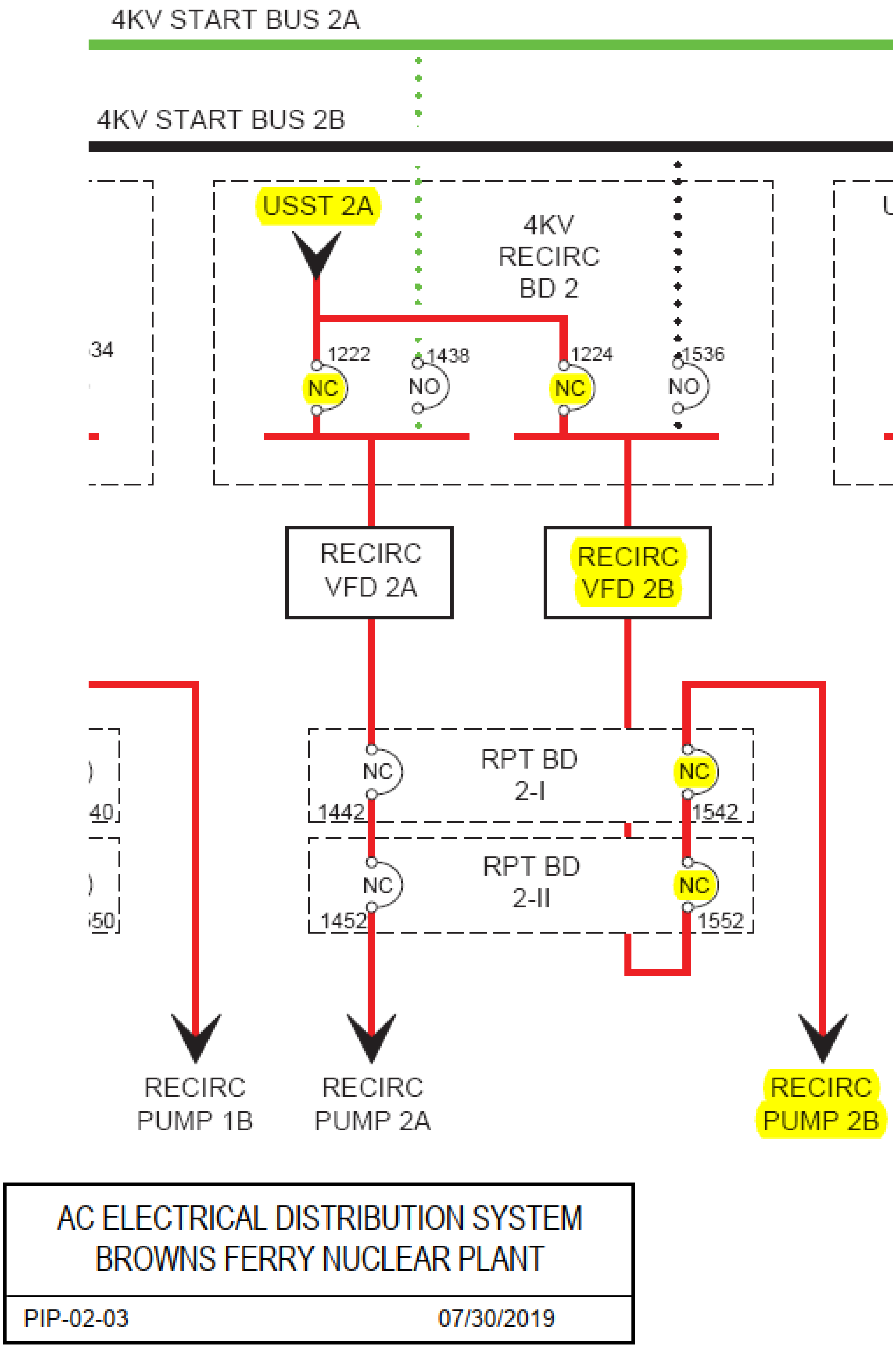
B. USST 1B

C. Start Bus 2A

D. Start Bus 2B

Answer: A

Excerpt from AC Distribution PIP 02-03:



Excerpt from 0-OI-57A:

BFN Unit 0	Switchyard and 4160V AC Electrical System	0-OI-57A Rev. 0166 Page 190 of 210

Attachment 1
(Page 3 of 7)

Auxiliary Power Supplies and Bus Transfer Schemes

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE	REMARKS
7	4kV Recirculation Pump Boards: (Unit 1,2,3)			
	A. Recirc VFD set A	Unit SS TR A (BKRs 1122,1222, 1322)	Start Bus 2A (BKRs 1436,1438, 1442)	Automatic high speed transfer from the normal to the alternate source is initiated by main generator unit trip relays. Automatic delayed transfer from the normal to the alternate source is initiated by high-speed voltage relay. (Breakers are listed in Unit 1, 2, 3 order.)
	B. Recirc VFD set B	Unit SS TR A (BKRs 1124,1224, 1324)	Start Bus 2B (BKRs 1534,1536, 1538)	

Examination Outline Cross-reference:	Level	RO	SRO
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation 1/4	Tier #	1	------
G2.4.49 (10CFR 55.41.10)	Group #	1	------
Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	K/A #	295001G2.4.49	
	Importance Rating	4.6	------

Proposed Question: **# 2**

Unit 1 is recovering from an inadvertent trip of Recirc Pump 1A, and the crew is executing the actions of 1-AOI-68-1, Recirc Pump Trip/Core Flow Decrease.

Which **ONE** of the following completes the statement below?

Given the condition above, the required action is to _________.

A. SCRAM the Reactor

B. INSERT Control Rods to less than 67.0% loadline

C. INSERT Control Rods to less than 74.0% loadline

D. COMMENCE a Reactor plant shut down and cooldown

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that 1-AOI-68-1 requires the Reactor to be SCRAMMED as an IMMEDIATE ACTION if both Recirc Pumps are tripped in MODES 1 or 2.

B **CORRECT:** *(See attached)* In accordance with 1-AOI-68-1, Recirc Pump Trip/Core Flow Decrease, if loadline is greater than 67%, then IMMEDIATELY take actions to insert Control Rods to less than 67.0% loadline. This step is right after the OPENING of RECIRC PUMP 1A DISCHARGE VALVE (as given), but inserting Control Rods can be performed at the same time by the Operator at the Controls (OATC).

C INCORRECT: Incorrect but plausible in that the previous revision of 1-AOI-68-1A (was consolidated into 1-AOI-68-1 current revision) required inserting Control Rods to less than 74% loadline.

D INCORRECT: Incorrect but plausible in that 1-AOI-68-1 states that when Recirc Pump was tripped due to a dual seal failure and loop temperature requirements cannot be maintained then, commence plant shut down and cooldown.

RO Level Justification: Tests the candidate's ability to recognize a partial loss of forced core flow circulation with a single Recirc Pump Trip with the priority to immediately perform required actions to insert Control Rods. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. Difficulty is enhanced in that the candidate must decide between actions for a Single or Dual Recirc Pump Trip.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 1-AOI-68-1, Rev. 1 (Attach if not previously provided)

1-AOI-68-1A, Rev. 5

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.007, Obj. 22 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1804 #1	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

Proposed Question: **# 1**

Unit 1 is recovering from a trip of Recirc Pump 1A and while executing the actions of 1-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease OPRMs Operable, the Balance of Plant (BOP) Operator has just reported that 1-FCV-68-3; RECIRC PUMP 1A DISCHARGE VALVE has been manually opened.

The BOP then reports that Recirc Pump 1B has tripped and Unit 1 has entered Region I of the Power to Flow Map.

Which ONE of the following completes the statement below in accordance with 1-AOI-68-1A?

Given the current plant conditions, the required action is to __________.

A. insert a manual Reactor SCRAM

B. insert Control Rods to less than 74% Loadline

C. commence a normal Reactor Shutdown / Cooldown

D. close the Discharge Valve on the outlet of Recirc Pump 1B

Proposed Answer: **A**

Excerpts from 1-AOI-68-1:

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease	1-AOI-68-1 Rev. 0001 Page 7 of 14

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] IF both Recirc Pumps are tripped in modes 1 or 2, THEN (Otherwise MARK NA)

SCRAM the Reactor.

4.2 Subsequent Actions

CAUTION

[NER/C] Failure to restart Reactor Recirculation pumps in a timely manner may result in exceeding the differential temperature limit for pump start and subsequently require plant depressurization to avoid exceeding pressure-temperature limits for the reactor vessel. [SER 93-005]

[1] IF both Recirc Pumps are tripped in modes 1 or 2, THEN (Otherwise MARK NA)

PERFORM the following:

[1.1] RESTART affected Reactor Recirculation pumps. REFERENCE 1-OI-68.

[1.2] IF the ΔT between the Rx vessel bottom head temperature and the moderator temperature precludes restart of a Recirc pump,

OR

Forced Recirculation flow CANNOT be established for any reason, THEN (Otherwise MARK NA)

[1.2.1] INITIATE a plant cooldown to prevent exceeding

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease	1-AOI-68-1 Rev. 0001 Page 8 of 14

4.2 Subsequent Actions (continued)

NOTES
1) Step 4.2[2] through Step 4.2[17.3] apply to any core flow lowering event.
2) Power To Flow Map is maintained in 0-TI-248, Station Reactor Engineer and on ICS.
3) If a cell bypasses while a recirc pump is running, a drop of ≈200 rpm will occur. The drive will remain at the speed at which it stopped. Operator action is required to raise speed following a cell bypass.
4) Single recirculation loop operation (SLO) is prohibited in the MELLLA+ operating domain. REFER TO Tech Spec 3.4.1.

[2] IF a single Recirc Pump has tripped, THEN

CLOSE tripped Recirc Pump 1A(1B) discharge valve 1-FCV-068-0003(0079).

[3] IF OPRM Upscale Trip Function is inoperable, THEN (Otherwise MARK N/A)

PERFORM 1-SR-3.3.1.1.I, Core Thermal Hydraulic Stability.

[4] [NER/C] WHEN the Recirc Pump discharge valve has been closed for at least five minutes (to prevent reverse rotation of the pump) [GE SIL-517], THEN (MARK N/A if Recirc Pump will be isolated for maintenance).

OPEN Recirc Pump 1A(1B) discharge valve, 1-FCV-068-0003(0079) as necessary to maintain Recirc Loop in thermal equilibrium.

[5] IF loadline is greater than 67.0%, THEN (Otherwise MARK N/A)

IMMEDIATELY take actions to insert control rods to less than 67.0% loadline. REFERENCE 0-TI-464, Reactivity Control Plan and 1-OI-85.

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease	1-AOI-68-1 Rev. 0001 Page 9 of 14

4.2 Subsequent Actions (continued)

NOTE

The remaining subsequent action steps apply to a single Reactor Recirc Pump trip.

[7] **MAINTAIN** operating Recirc pump flow less than 46,600 gpm. **REFERENCE** 1-OI-68.

[8] [NER/C] **WHEN** plant conditions allow, **THEN**, (Otherwise **MARK** N/A)

MAINTAIN operating jet pump loop flow greater than 41×10^6 lbm/hr (1-FI-68-46 or 1-FI-68-48). [GE SIL 517]

CAUTION

The temperature of the coolant between the dome and the idle Recirc loop should be maintained within 75°F of each other. If this limit cannot be maintained, a plant cool down should be initiated. Failure to maintain this limit and not cool down could result in hangers and/or shock suppressers exceeding their maximum travel range. [GE SIL 251, 430 and 517]

[9] **IF** Recirc Pump was tripped due to dual seal failure, **THEN** (Otherwise **MARK** N/A)

[9.1] **ENSURE** TRIPPED, RECIRC DRIVE 1A(1B) NORMAL FEEDER, 1-HS-57-17(14).

[9.2] **ENSURE** TRIPPED, RECIRC DRIVE 1A(1B) ALTERNATE FEEDER, 1-HS-57-15(12).

[9.3] **CLOSE** tripped recirc pump suction valve using, RECIRC PUMP 1A(1B) SUCTION VALVE, 1-HS-68-1(77).

[9.4] **IF** it is evident that 75°F between the dome AND the idle Recirc loop CANNOT be maintained, **THEN**

COMMENCE plant shut down and cool down. **REFERENCE** 1-GOI-100-12A.

Excerpt from 1-AOI-68-1**A** (previous revision consolidated into current revision of 1-AOI-68-1): supports Distractor (C)

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease OPRMs Operable	1-AOI-68-1A Rev. 0005 Page 8 of 13

4.2 Subsequent Actions (continued)

NOTES
1) Step 4.2[2] through Step 4.2[17.3] apply to any core flow lowering event.
2) Power To Flow Map is maintained in 0-TI-248, Station Reactor Engineer and on ICS.
3) If a cell bypasses while a recirc pump is running, a drop of ≈200 rpm will occur. The drive will remain at the speed at which it stopped. Operator action is required to raise speed following a cell bypass.

[2] **IF** a single Recirc Pump has tripped, **THEN**

CLOSE tripped Recirc Pump 1A(1B) discharge valve 1-FCV-068-0003(0079). ☐

[3] [NER/C] **WHEN** the Recirc Pump discharge valve has been closed for at least five minutes (to prevent reverse rotation of the pump) [GE SIL-517], **THEN** (N/A if Recirc Pump will be isolated for maintenance).

OPEN Recirc Pump 1A(1B) discharge valve, 1-FCV-068-0003(0079) as necessary to maintain Recirc Loop in thermal equilibrium. ☐

[4] **IF** loadline is greater than 74%, **THEN** (Otherwise N/A)

IMMEDIATELY take actions to insert control rods to less than 74% loadline AND **REFER TO** 0-TI-464, Reactivity Control Plan and 1-OI-85. ☐

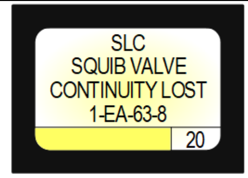
[5] **RAISE** core flow to greater than 45% in accordance with 1-OI-68. ☐

[6] **INSERT** control rods to exit regions if **NOT** already exited AND **REFER TO** 0-TI-464, Reactivity Control Plan and 1-OI-85. ☐

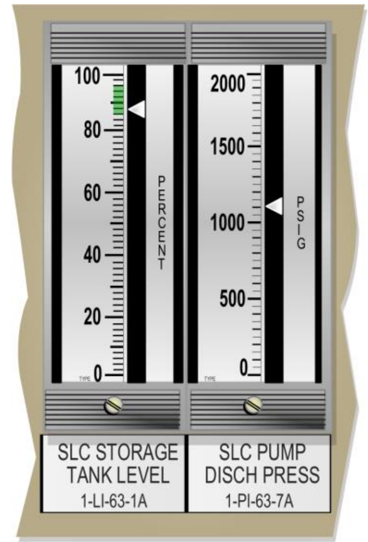
Examination Outline Cross-reference:	Level	RO	SRO
295037 (EPE 14) SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1	Tier #	1	------
EA1.04 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:	K/A #	295037EA1.04	
• SBLC			------
	Importance Rating	4.5*	

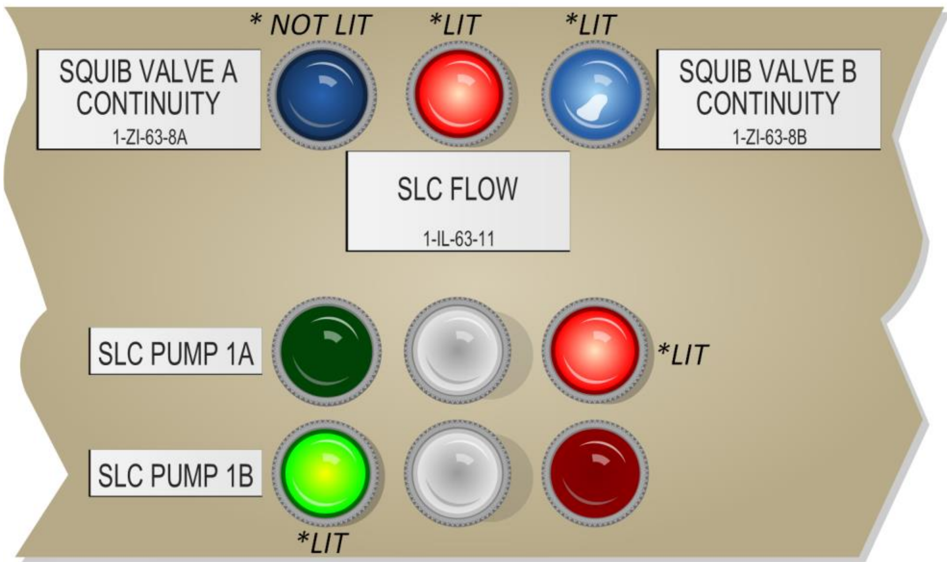
Proposed Question: **# 3**

Unit 1 Anticipated Transient Without a SCRAM (ATWS) conditions exist with the following conditions:



- 1-EOI-1A, ATWS RPV Control is in progress
- Reactor Pressure is 950 psig
- 1-EOI Appendix-3A, SLC INJECTION in progress as indicated
- SLC SQUIB VALVE CONTINUITY LOST, (1-9-5B, WINDOW 20) alarms





Given the conditions above, which **ONE** of the following completes the statement below regarding the status of the Standby Liquid Control (SLC) System?

SQUIB VALVE ____**(1)**____ has fired and SLC ____**(2)**____ injecting to the Reactor.

A. (1) A
(2) is

B. (1) A
(2) is NOT

C. (1) B
(2) is

D. (1) B
(2) is NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (*See attached*) In accordance with 1-EOI Appendix-3A, SLC INJECTION, SQUIB VALVE A has fired. This is indicated by SQUIB VALVE 'A' CONTINUITY **extinguished** blue light and the provided SLC SQUIB VALVE CONTINUITY LOST annunciator being **illuminated**. For second part, SLC is injecting to the RPV based on 1100 psig discharge pressure, SLC PUMP 1A red light is lit indicating pump is running and the SLC FLOW red light is illuminated. However, the SLC INJECTION TO REACTOR (1-9-5B, Window 14) was not provided to the candidate.

B INCORRECT**:** First part is correct (*See A*). Second part is incorrect but plausible in that up to 10 indications can exist to indicate whether SLC is or is not injecting to the RPV. The candidate could easily confuse the various light indications and/or the given annunciator requirements to be illuminated.

C INCORRECT**:** First part is incorrect but plausible if the candidate confuses squib valve firing indications with the provided SLC SQUIB VALVE CONTINUITY LOST annunciator requirements to be illuminated or the SQUIB VALVE 'B' CONTINUITY blue light indicated as LIT. Second part is correct (*See A*).

D INCORRECT**:** First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability to operate and monitor the Standby Liquid Control (SLC) System as it relates to Anticipated Transient Without SCRAM (ATWS) condtions. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 1-ARP-9-5B, Rev. 22 (Attach if not previously provided)

1-EOI Appendix-3A, Rev. 0

OPL171.039, Rev. 22

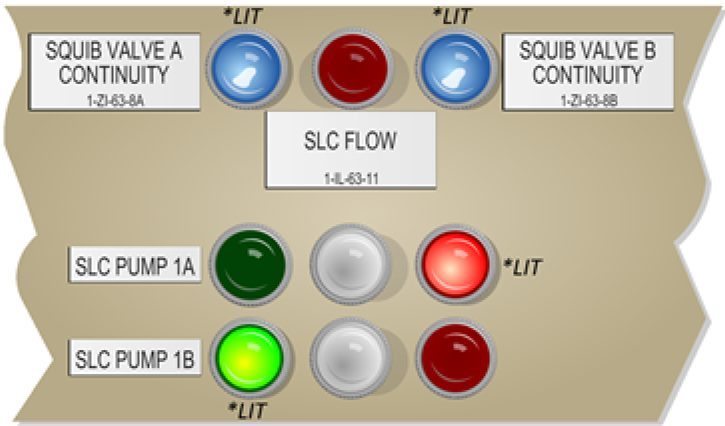
Proposed references to be provided to applicants during examination: **Panel 1-9-5, 1-LI-63-1A, SLC Storage Tank Level and 1-PI-63-7A, SLC Pump Discharge Pressure and 1-HS-63-6A, SLC Pump 1A/1B Pump/light indications and 1-ZI-63-8A/B, SLC Valve A/B Continuity lights and 1-IL-63-11, SLC Flow light indication. SLC SQUIB VALVE CONTINUITY LOST, (1-9-5B, WINDOW 20)**

Learning Objective:	OPL171.039 Obj. 4, 9, 10	(As available)	
Question Source:	Bank #		
	Modified Bank #	BFN 1804 #32	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		
Comments:			

Copy of Bank Question:

Unit 1 is executing 1-EOI-1A, ATWS RPV Control. The Unit Operator (UO) is directed to inject Standby Liquid Control (SLC) in accordance with 1-EOI Appendix-3A, SLC INJECTION.

The UO places 1-HS-63-6A, SLC PUMP 1A/1B control switch in the 'START-A' position and observes the following Panel 1-9-5 conditions:



Which ONE of the following completes the statement below?

The status of the SLC system indicates ___**(1)**___ **AND** the correct action(s) as stated in 1-EOI Appendix-3A, is to ___**(2)**___.

A. **(1)** **NEITHER** squib valve has fired
(2) start SLC Pump 1B, **AND** verify proper operation

B. **(1)** **ONE** squib valve has fired
(2) start SLC Pump 1B, **AND** verify proper operation

C. **(1)** **ONE** squib valve has fired
(2) verify proper system operation by observing the SLC tank level lowering by ~1% per minute

D. **(1)** **BOTH** squib valves have fired
(2) verify proper system operation by observing the SLC tank level lowering by ~1% per minute

Proposed Answer: **A**

Excerpt from 1-ARP-9-5B:

BFN Unit 1	Panel 9-5 1-XA-55-5B	1-ARP-9-5B Rev. 0022 Page 23 of 42

SLC
SQUIB VALVE
CONTINUITY LOST
1-EA-63-8
20

(Page 1 of 1)

Sensor/Trip Point:

1-XM-63-8A and 8B   fall 3.0 ma
rise 7.0 ma

Sensor Location: Behind Panel 1-9-5
Main Control Room

Probable Cause:
A. Cleared fuse.
B. Loss of power supply.
C. Blown Photohelic Bulb.
D. Sensor malfunction.
E. SLC pump start from the control room.
F. Corrosion buildup inside electrical connector

Automatic Action: None

Operator Action:
A. **IF** SLC has been initiated, **THEN REFER TO** 1-EOI-1 or 1-AOI-79-2. ☐
B. **IF** SLC has **NOT** been initiated, **THEN PERFORM** the following:
   1. **CHECK** blue indicating lights on Panel 1-9-5 to determine which valve ignition circuit failed. ☐
   2. **CHECK** sensor and amp meter in back of Panel 1-9-5. ☐
   3. **DISPATCH** personnel to the SLC tank, RB El. 639', to

Excerpt from 1-EOI APPENDIX-3A:

BFN UNIT 1	SLC INJECTION	1-EOI APPENDIX-3A Rev. 0 Page 1 of 2

LOCATION:	Unit 1 Control Room
ATTACHMENTS:	None (√)

1. **UNLOCK** and **PLACE** 1-HS-63-6A, SLC PUMP 1A/1B, control switch in START-A or START-B position. _____

2. **CHECK** SLC System for injection by observing the following:

   - Selected pump starts, as indicated by red light illuminated above pump control switch. _____
   - Squib valves fire, as indicated by SQUIB VALVE A and B CONTINUITY blue lights extinguished, _____
   - SLC SQUIB VALVE CONTINUITY LOST 1-EA-63-8 Annunciator in alarm on Panel 1-9-5 (1-XA-55-5B, Window 20). _____
   - 1-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure. _____
   - System flow, as indicated by 1-IL-63-11, SLC FLOW, red light illuminated on Panel 1-9-5, _____
   - SLC INJECTION FLOW TO REACTOR 1-FA-63-11, Annunciator in alarm on Panel 1-9-5 (1-XA-55-5B, Window 14). _____

3. IF .... Proper system operation CANNOT be verified,

   THEN.......... **RETURN** to Step 1 and **START** other SLC pump. _____

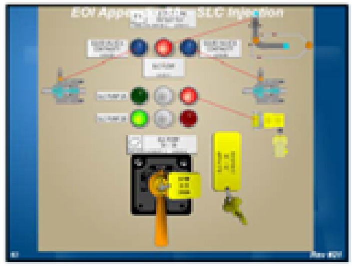
4. **VERIFY** RWCU isolation by observing the following:

   - RWCU Pumps 1A and 1B tripped _____
   - 1-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE closed _____
   - 1-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE closed _____
   - 1-FCV-69-12, RWCU RETURN ISOLATION VALVE closed. _____

5. **VERIFY** ADS inhibited. _____

6. **MONITOR** reactor power for downward trend. _____

Excerpt from OPL171.039 SLC Lesson Plan:



63

OBJ:
NLO 4.h, 7, 10.a / b / c / e / f / g
LO 4.h, 9, 12.a / b / c / e / f / g

When initiation of the SLC System is required, normal system operation is governed by EOI Appendix 3A, "SLC Injection." The SLC System is normally in a standby readiness lineup with the SLC Storage Tank aligned to the SLC Pumps. System initiation requires only that the operator place the key, located beside SLC PUMP A/B switch (HS-63-6A) at Panel 9-5, into the switch and turn it; followed by placing the switch itself into either START PUMP A or START PUMP B position. The switch will spring-return to the respective NORM AFT START position. Objective 7

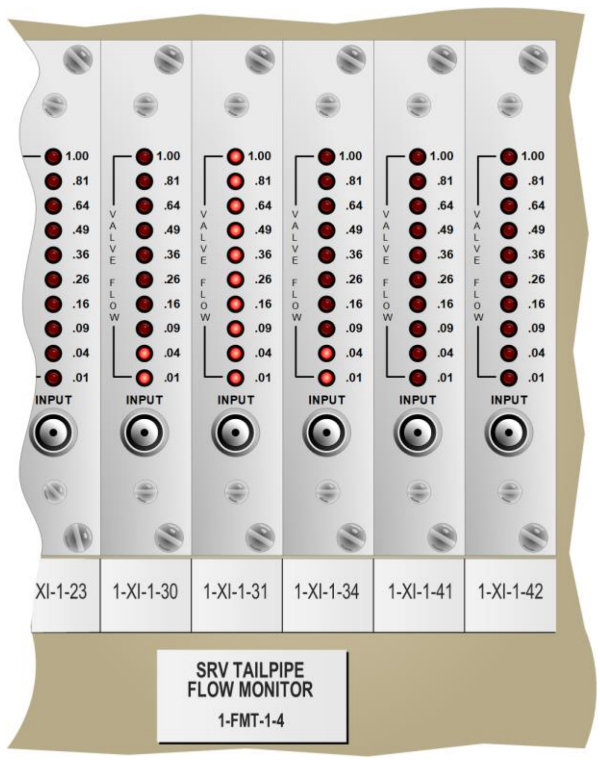
Upon placing the switch to START PUMP A or START PUMP B, the following actions take place:

- The selected SLC PUMP starts as indicated by the red "pump running" light on Panel 9-5.
- Both SQUIB VALVES fire, as indicated by both SQUIB VALVE A and B CONTINUITY blue lights extinguished and activation of Annunciator SLC SQUIB VALVE CONTINUITY LOST, (9-5B Window 20).
- SLC PUMP DISCHARGE PRESSURE indicates above RPV pressure on Panel 9-5 (approximately 1100 psig with Reactor Pressure in the normal range).
- System flow initiates, as indicated by the SLC FLOW red indicating light on Panel 9-5.
- SLC INJECTION FLOW TO REACTOR, (9-5B Window 14) Annunciator activates on Panel 9-5.
- Reactor Water Cleanup System INBOARD AND OUTBOARD SUCTION ISOLATION VALVES (FCV- 69-1 and 2) and RETURN ISOLATION VALVE (FCV-69-12) automatically close to prevent removal of the poison solution from the reactor vessel. The RWCU PUMPS also trip on interlock.

Examination Outline Cross-reference:	Level	RO	SRO
239002 (SF3 SRV) Safety Relief Valves	Tier #	2	------
G2.2.44 (10CFR 55.41.5)	Group #	1	------
Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions.	K/A #	239002G2.2.44	
	Importance Rating	4.2	------

Proposed Question: **# 4**

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



Which **ONE** of the following completes the statements below?

As a result of the above condition, indicated Main Generator Megawatts Electric (MWe) on Panel 1-9-5 will ___**(1)**___.

In accordance with the applicable Abnormal Operating Procedure (AOI), the correct Immediate Action is to reduce Reactor Power to less than or equal to ___**(2)**

.

A. (1) lower
(2) 95%

B. (1) lower
(2) 90%

C. (1) remain constant
(2) 95%

D. (1) remain constant
(2) 90%

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** The first part is correct (*See* B). The second part is incorrect but plausible in that in accordance with 1-AOI-6-1, Feedwater Heater String/Extraction Steam Isolation, Reactor Power is reduced by 5% as an immediate action. Additionally, previous plant procedures directed Reactor Power reduction of 5% when an SRV inadvertently opened, regardless of the power level at which the SRV opened.

B **CORRECT:** (*See attached*) In accordance with 1-AOI-1-1, Relief Valve Stuck Open, a symptom of an open SRV is that Main Generator Output lowers. This is due to steam being directed from the Main Steam Line before it reaches the Main Turbine. For second part, in accordance with 1-AOI-1-1, an Immediate Action for a stuck open SRV is to reduce Reactor Power to less than 90% if Reactor Power is above 90% at the time the SRV opens. The candidate is given in the conditions that Reactor Power is 100%, therefore a power reduction to ≤ 90% is required.

C INCORRECT: The first part is incorrect but plausible in that the candidate may believe that an open SRV has no effect on Main Generator loading due to the Turbine Control System. The second part is incorrect but plausible (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is correct (*See B*).

RO Level Justification: Tests the candidate's ability to interpret Control Room indications of a stuck open SRV and understand the actions required to affect plant conditions. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-AOI-1-1, Rev.5 (Attach if not previously provided)
1-AOI-6-1, Rev. 0
1-47E801-1, Rev. 27

Proposed references to be provided to applicants during examination: **1-FMT-1-4, SRV TAILPIPE FLOW MONITOR DRAWING**

Learning Objective: OPL171.074 Obj. 2 (As available)
OPL171.009, Obj 6

Question Source:			
	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.009-06 002 #361	(Note changes or attach parent)
	New		
Question History:			

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Copy of Bank Question:

361. OPL171.009-06 002

Which **ONE** of the following indicates the expected response of Main Generator Megawatts Electric (MWE) and Main Steam Line Flow in the affected Main Steam Line upon the inadvertent opening of an MSRV with the reactor at 100% power?

	Main Generator MWE	Affected Main Steam Line Steam Flow
A.	Rises	Rises
B.	Rises	Lowers
C.	Lowers	Rises
D.✓	Lowers	Lowers

Excerpts from 1-AOI-1-1:

BFN Unit 1	Relief Valve Stuck Open	1-AOI-1-1 Rev. 0005 Page 3 of 34

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a stuck open relief valve.

2.0 SYMPTOMS

A. Annunciator MAIN STEAM RELIEF VALVES OPEN 1-FA-1-1 (1-XA-55-3C, Window 25) is in alarm due to the SRV Tailpipe Flow monitor sensing flow.

B. GENERATOR LOAD recorder, 1-XR-57-57, Panel 1-9-8, indication is lowering.

C. MAIN STEAM/TURBINE STEAM FLOW, flow recorder 1-FR-46-5, Panel 1-9-5, indication is lowering.

D. SUPPRESSION POOL WATER TEMPERATURE recorder, 1-TR-64-161 and SUPPRESSION POOL WATER TEMPERATURE recorder, 1-TR-64-162, Panel 1-9-3, indication is rising.

3.0 AUTOMATIC ACTION

None

BFN Unit 1	Relief Valve Stuck Open	1-AOI-1-1 Rev. 0005 Page 4 of 34

4.0 OPERATOR ACTION

NOTE

Once a MSRV is operated, a time delay of 15 to 30 seconds can be expected before a response can be detected on 1-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE. ICS can be used to monitor the discharge tailpipe temperature, but the appropriate indications on 1-TR-1-1 must be confirmed.

4.1 Immediate Action

[1] **IDENTIFY** stuck open relief valve by

OBSERVING the following:

- SRV TAILPIPE FLOW MONITOR, 1-FMT-1-4, on Panel 1-9-3,

  OR

- MSRV DISCHARGE TAILPIPE TEMPERATURE, 1-TR-1-1 on Panel 1-9-47. ☐

[2] **IF** relief valve transient occurred while operating above 90% power, **THEN**

REDUCE reactor power to ≤90% RTP with recirc flow. (Otherwise N/A) ☐

Excerpt from 1-AOI-6-1: Supports Distractors A(2), C(2)

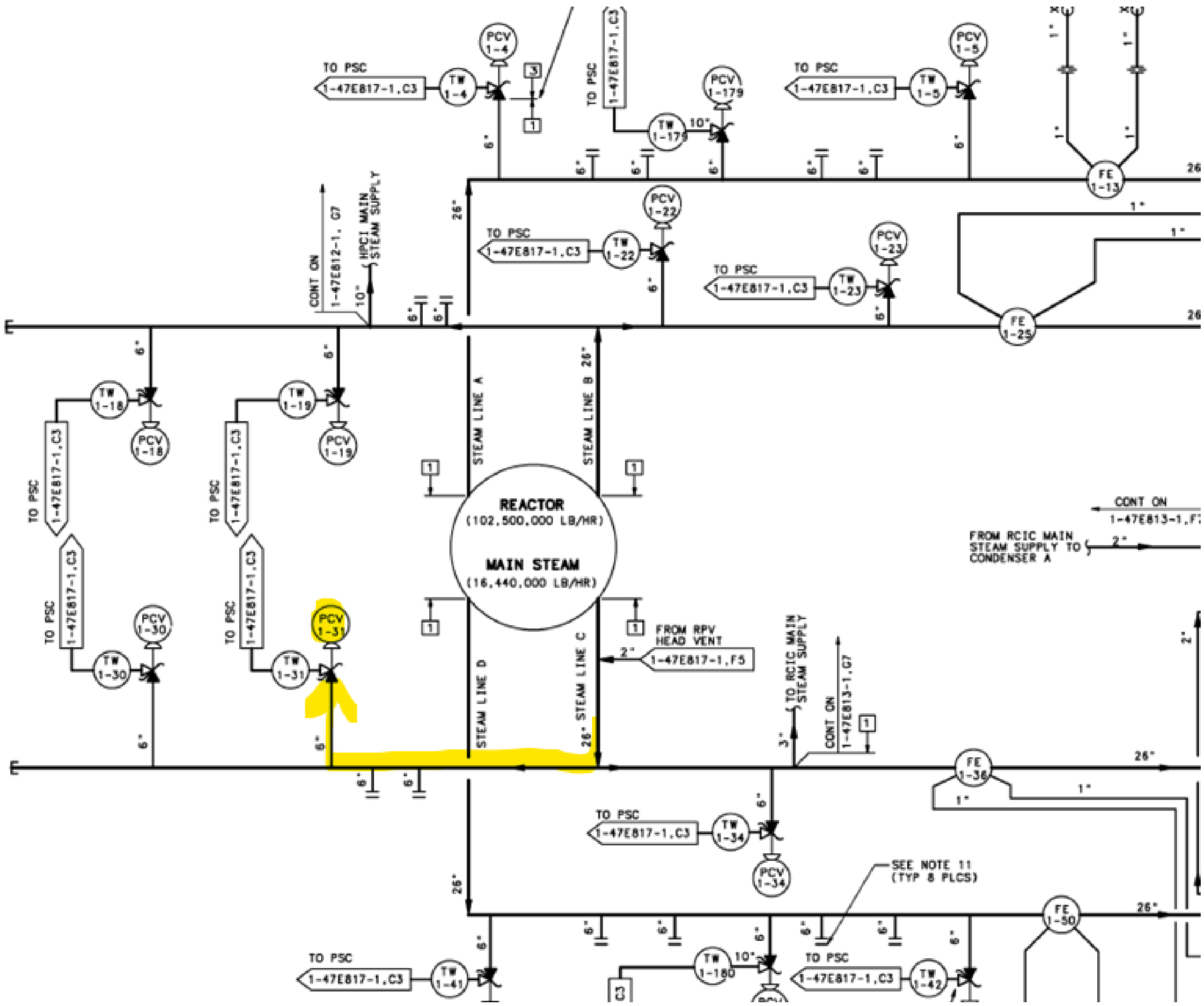
BFN Unit 1	Feedwater Heater String/Extraction Steam Isolation	1-AOI-6-1 Rev. 0000 Page 7 of 18

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **REDUCE** Core Power to greater than or equal to 5% below initial power level to maintain thermal margin.

Excerpt from 1-47E801-1: Illustrates provided SRV 1-31 location on Main Steam header



Examination Outline Cross-reference:

259002 (SF2 RWLCS) Reactor Water Level Control

K1.11 (10CFR 55.41.7)

Knowledge of the physical connections and/or cause effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following:

- Drywell pressure: FWCI/HPCI

	RO	SRO
Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	259002K1.11	
Importance Rating	3.0	------

Proposed Question: **# 5**

Unit 2 was operating at 100% RTP when an event occurred, resulting in the following conditions:

- A manual Reactor SCRAM was inserted
- Reactor Water Level lowered to (-) 30 inches
- Drywell Pressure is 2.85 psig
- HPCI automatically initiated and is injecting to the Reactor at 5300 gpm

Given the conditions above, which **ONE** of the following completes the statement below in accordance with 2-OI-73, High Pressure Coolant Injection System?

HPCI automatically initiated due to ____**(1)**____ and is injecting into ____**(2)**____ Feedwater System line.

A. **(1)** High Drywell Pressure
(2) 'B'

B. **(1)** High Drywell Pressure
(2) 'A'

C. **(1)** Low Reactor Water Level
(2) 'B'

D. **(1)** Low Reactor Water Level
(2) 'A'

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that RCIC injects into 'B' Feedwater System line. HPCI and RCIC Systems components, setpoint, isolations and initiation are often confused.

B **CORRECT**: (*See attached*) In accordance with 2-OI-73, High Pressure Coolant Injection System, the HPCI System automatically initiates on Low Reactor Water Level at (-) 45 inches and High Drywell Pressure at (+) 2.45 psig. For second part, HPCI injects into 'A' Feedwater System line.

C INCORRECT: First part is incorrect but plausible if the candidate confuses the Reactor Water Level setpoint of (+) 2 inches as an automatic initiation for HPCI. That is a specific Reactor Water Level setpoint for numerous systems to isolate from PCIS Groups 2, 3, 6 and 8 and a SCRAM setpoint. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the cause and effect relationship between Reactor Water Level Control as it relates to HPCI and High Drywell Pressure. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-73, Rev. 101 (Attach if not previously provided)
OPL171.042, Rev. 23
OPL171.040, Rev. 31

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.042 Obj. 3a (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43

Excerpt from 2-OI-73:

BFN Unit 2	High Pressure Coolant Injection System	2-OI-73 Rev. 0101 Page 12 of 97

3.3 Equipment (continued)

H. The HPCI Injection valve, 2-FCV-73-44, is a 14 inch, Crane, Class 900, flex wedge gate valve. Flex wedge valves are potentially susceptible to pressure locking. DCN 69896 has been implemented to eliminate the potential for pressure locking of 2-FCV-73-44 by drilling a 1/4" hole in the downstream side of the disc.

3.4 Initiation

A. When any of the following signals are received, the HPCI System automatically initiates:

1. Low RPV water level at -45".

2. High drywell pressure at 2.45 psig.

B. The HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, will automatically open when system flow is at or below 900 gpm (lowering) if a system initiation signal is present, and will automatically close when system flow is at or above 1255 gpm (rising) regardless of presence of initiation signal.

C. HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, will open on receipt of an initiation signal even with HPCI Auxiliary Oil pump in PULL-TO-LOCK position resulting in slowly draining CST to Suppression Chamber.

3.5 Isolation

A. When any of the following signals are received, the HPCI System automatically isolates: (REFER TO 2-AOI-64-2b, Group 4 HPCI Isolation.)

Excerpt from OPL171.042 Lesson Plan:

OPL171.042 , High Pressure Cooling Injection (HPCI), Rev# 23U2

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
c) Turbine Auxiliaries	
3. Flow path	
a) One 100% System	
b) Steam Path (1) From B Main Steam Line upstream of the flow restrictor (2) Through isolation valves (3) Through stop valve and control valves (4) Exhaust through check valve to suppression pool	Obj. ILT 1 Obj. LOR 1 Obj. NLO 10
c) Water Path (1) Normal condensate path from CST to the HPCI pump, to the A Feedwater line and into the Reactor vessel (2) Alternate suction path from suppression pool (3) Automatic swap over to suppression pool on high suppression pool level +5.25" or low CST level Elev < 552'6"	TP-1 Approx. 7000 gallons left in CST piping when auto swap occurs on low CST Level.

Excerpt from OPL171.040 Lesson Plan:

OPL171.040, Reactor Isolation Cooling (RCIC), Rev.# 31

Lesson Plan Content

Outline of Instruction	Instructor N and Method

The demonstration is based on Browns Ferry and other BWR plant performance problems. These plant performance problems are derived from the following operating experience:

- PER 00-011480: Discharge Piping Overpressurization
- NRC IN 2000-01: BWR Operational Issues During Rx Scram & Transient
- GE SIL No. 623: Peak Pump Discharge Pressure during SRs
- OE10570: Turbine Trips Due to Check Valve
- OE11133: Pump Trip on Lo Suct Press Due To Unfilled Discharge Piping

Presentation

A. General Description

1. The purpose of the RCIC System is to provide a source of high pressure coolant makeup to the Reactor vessel in case of a loss of feedwater flow. The system is used to maintain the Reactor water level and for Reactor pressure control under MSIV isolation conditions and loss of normal feedwater.

2. Safety Design Basis

   RCIC operates automatically to maintain sufficient coolant in the vessel so that the fuel will not overheat in the event of Reactor isolation and loss of feedwater flow. The system is a consequence limiting system rather than an ECCS system. ILT/NLO 1

B. The RCIC System Consists of :

1. Turbine-driven pump located in basement of Reactor Building (Elev. 519) — Obj. NLOR 1
2. Turbine is driven by steam from Main Steam Line "C" and exhausts to the suppression pool. — Obj. NLO 2; TP-1 & TP-2
3. Pump is normally lined up to take suction from the Condensate Storage Tank (CST), but can take suction from suppression pool (only done manually). — Obj. ILT 2.
4. Pump discharges to Reactor via feedwater line B — Obj. NLO 3

   a. Turbine

Examination Outline Cross-reference:	Level	RO	SRO
215004 (SF7 SRMS) Source-Range Monitor	Tier #	2	------
K1.06 (10CFR 55.41.2)	Group #	1	------
Knowledge of the physical connections and/or cause-effect relationships between the SOURCE RANGE MONITOR (SRM) SYSTEM and the following:	K/A #	215004K1.06	
• Reactor vessel	Importance Rating	2.8	------

Proposed Question: **# 6**

With regards to the Source Range Monitors (SRMs), which **ONE** of the following completes the statements below?

To fully **WITHDRAW** the SRMs from the Core, the DRIVE OUT pushbutton is required to be **(1)** .

To fully **INSERT** the SRMs into the Core, the DRIVE IN pushbutton is required to be **(2)** .

A. (1) momentarily depressed and released due to a seal-in contact
(2) momentarily depressed and released due to a seal-in contact

B. (1) momentarily depressed and released due to a seal-in contact
(2) continuously depressed and held for the entire length of travel

C. (1) continuously depressed and held for the entire length of travel
(2) momentarily depressed and released due to a seal-in contact

D. (1) continuously depressed and held for the entire length of travel
(2) continuously depressed and held for the entire length of travel

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that in accordance with 1-OI-92, Source Range Monitors, to withdraw the SRMs from the core, the procedure directs the operator to depress and hold the Drive Out Push Button. This procedure is infrequently performed, and the drive in/drive out circuitry is often confused. The candidate must know the circuitry to realize that the Drive Out Push Button is a momentary contact and does not have a seal-in contact to maintain the drive out circuit energized once released. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that the Drive In Push Button has a maintained contact in order to keep the drive in circuit energized until the SRMs have been fully inserted into the core.

C **CORRECT:** *(See attached)* In accordance with 1-OI-92, Source Range Monitors, the Drive Out Push Button must be continuously depressed in order to maintain the drive in circuit energized. For second part, the DRIVE IN pushbutton has a maintaining contact to keep the drive in circuit energized until the SRMs have been fully driven into the core.

D INCORRECT: First part is correct *(See C).* Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the SRMs and how they are driven into and out of the core. This question is rated as Memory due to the requirement to strictly recall facts related to how the SRMs are driven into and out of the core.

Technical Reference(s): 1-OI-92, Rev.10 (Attach if not previously provided)

OPL171.019, Rev. 14

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.019_Obj. 4d (As available)

Question Source:	Bank #		
	Modified Bank #	ILT Exam Bank OPL171.019-07 001 #593	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 06 27 2013

593. OPL171.019-07 001

Unit 1 is performing a reactor startup with the following:

- Reactor Power indicates 50/125 on Range 5 of all IRMs.
- Reactor Period is approximately 100 seconds and stable.
- SRM 'B' and SRM 'C' indicate 1000 cps and are partially withdrawn.
- SRM 'A' and SRM 'D' indicate 30,000 cps and are fully inserted.
- All 4 SRMs are selected.
- The DRIVE OUT pushbutton is firmly depressed, then released after 3 seconds.
- NO other operator actions are taken.

Which ONE of the following describes the expected response of the SRM detectors and Reactor Manual Control System as a result of these actions?

A. All 4 SRM detectors fully withdraw from the core.
There are no control rod insert or withdrawal blocks received as a result of these actions.

B. All 4 SRMs will fully withdraw from the core.
A control rod withdrawal block will occur when any detector indication lowers below 145 cps.

C. All 4 SRM detectors momentarily withdraw, then stop when full-in indication is lost.
A control rod withdrawal block is received when any SRM detector is not fully inserted.

D✓ All 4 SRMs will withdraw while the DRIVE OUT pushbutton is depressed, then stop when the button is released.
A control rod withdrawal block will occur when any SRM detector indication reaches 68,000 cps.

Excerpts from 1-OI-92:

BFN Unit 1	Source Range Monitors	1-OI-92 Rev. 0010 Page 15 of 20

6.3 Withdrawing SRMs (continued)

NOTES

1) SRM Count indicators SRM A 1-XI–92-7/43A(SRM C 1-XI-92-7/43C, SRM B 1-XI–92-7/43B, SRM D 1-XI–92-7/43D) will decrease when the associated SRM is withdrawn.
2) While withdrawing SRMs during startup the SRM count rate should be maintained between 10^2 cps and 10^5 cps for the SRM being withdrawn.
3) When withdrawing SRM Detectors the DRIVE OUT 1-HS-92-7C/S3 pushbutton must remain depressed until the desired Count Rate on the associated detectors are reached or the Full Out position is reached.
4) If withdrawing multiple SRM's and only one SRM detector needs to be stop, the associated CH SRM SELECT pushbutton needs to be depressed.

[3.4] **ONCE** all the detector being withdrawn has been selected, **THEN**

PERFORM the following to withdraw the SRM detectors:

[3.4.1] **WHILE MONITORING** and **MAINTAINING** the associated SRM COUNTS between 10^2 cps and 10^5 cps on the appropriate SRM Count Rate indicator,

DEPRESS and **HOLD** the DRIVE OUT 1-HS-92-7C/S3 pushbutton until the desired count rate is obtained. (see note 3 above)

[3.4.2] **IF** The SRM's fail to withdraw, **THEN**

PERFORM the following: (Otherwise N/A)

A. **RELEASE** the DRIVE OUT 1-HS-92-7C/S3 pushbutton.

B. **DEPRESS** the DRIVE IN, 1-HS-92-7C/S2 to reset the DRIVE IN Circuitry Seal In.

C. **DEPRESS** and **HOLD** the DRIVE OUT 1-HS-92-7C/S3 pushbutton until the desired count rate is obtained. (see note 3 above)

6.4 Inserting SRMs

NOTE

1) All operations are performed on Panel 1-9-5 unless specifically stated otherwise.
2) While inserting SRMs during shutdown the SRM count rate should be maintained between 10^2 cps and 10^5 cps for the SRM being inserted.
3) SRMs should be fully inserted as directed by 1-GOI-100-12 or as needed for SRM testing.
4) More than one SRM detector may be inserted at a time if needed. During a Reactor Shutdown one SRM Power Indication per channel should used to monitor power during SRM insertion.
5) All SRM's should be inserted at the same time during a Reactor Scram with all Control Rods are Fully Inserted per 1-AOI-100-1.

[1] **VERIFY** Source Range Monitor System in standby readiness. **REFER TO** Section 4.0.

[2] **REVIEW** all precautions and limitations. **REFER TO** Section 3.0.

[3] **SELECT** the desired SRM (SRM A, SRM B, SRM C, SRM D) drives by:

PERFORMING the following: (Refer to notes above.)

[3.1] **DEPRESS** SRM/IRM DETECTOR POSITION, 1-HS-92-7C/S1 pushbutton.

<u>AND</u>

OBSERVE the background light ILLUMINATES.

[3.2] **DEPRESS** the appropriate CH A (B, C, D) SRM SELECT pushbutton.

<u>AND</u>

OBSERVE the background light ILLUMINATES.

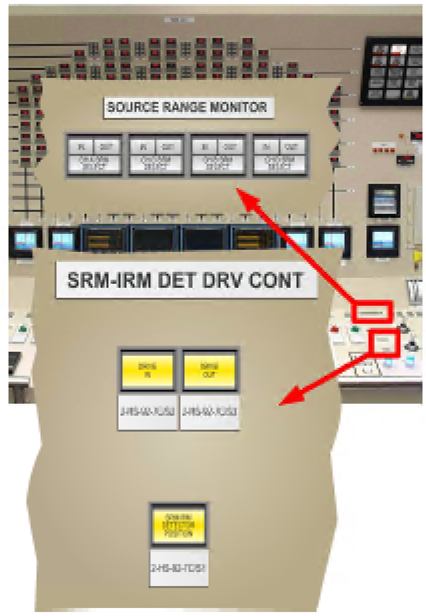
[3.3] **DEPRESS** the DRIVE IN, 1-HS-92-7C/S2 pushbutton.

<u>AND</u>

OBSERVE the background light ILLUMINATES.

Excerpt from OPL171.019 Lesson Plan:

Panel 9-5 Controls and Indications



Each SRM detector (A-D) is provided with a SELECT pushbutton on Panel 9-5 in the main control room. SRM select power is supplied by 120 VAC from I & C Bus A. When an SRM channel is selected, the bottom half of the switch lights up. Backlighting behind the upper half of the switches indicates detector position. The "In" section is illuminated when the detector is at the full in electrical stop. The "Out" section is illuminated when detector is at the full out electrical stop. If neither section is illuminated, the detector is between full in and full out.

Below the selection pushbuttons are the DRIVE IN and DRIVE OUT control pushbuttons. When the DRIVE IN pushbutton is depressed, all selected detectors will travel inward. This is a maintained contact pushbutton, and is not required to be continuously held by the operator. However, to withdraw selected SRM detectors, this pushbutton must be deselected. This motion will continue until either the full-in electrical stop is reached or until the DRIVE IN pushbutton is depressed a second time. The DRIVE OUT pushbutton is a momentary contact, and therefore must be depressed and held by the operator during outward movement. When attempting to drive out, if no movement occurs, depress the DRIVE IN to deselect the maintain contact then depress DRIVE OUT. The DRIVE IN and DRIVE OUT pushbuttons are also backlit to indicate that motion has been initiated. When driving in, the lamp will extinguish when all selected detectors reach full in. The time required to drive a detector full in to full out is approximately 3 minutes.

Examination Outline Cross-reference:	Level	RO	SRO
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff **A4.01** (10CFR 55.41.7) Ability to manually operate and/or monitor in the control room: • Valve closures	Tier #	2	------
	Group #	1	------
	K/A #	223002A4.01	
	Importance Rating	3.6	------

Proposed Question: **# 7**

The following conditions exist on Unit 2:

- A LOCA is in progress
- Drywell Pressure is 2.2 psig and slowly rising
- Reactor Water Level is (-) 10 inches and slowly lowering
- Main Steam Tunnel Temperature is 180 ºF and slowly rising

Given the conditions above, which **ONE** of the following completes the statement below?

PCIS Groups that **HAVE** received an isolation signal include Groups _________, and 8.

A. 1, 2

B. 1, 5

C. 2, 6

D. 4, 5

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that there are numerous PCIS Groups, with various isolation signals. It is plausible that any Group could be isolated by any signal, as the purpose of the PCIS Groups is to isolate systems in the event of a leak from the Reactor Coolant System or a radiation problem in Primary/Secondary Containment.

B INCORRECT**:** Incorrect but plausible (*See A*).

C **CORRECT**: (*See attached*) The given Reactor Water Level is below (+) 2 inches, PCIS Groups 2, 3, 6, and 8 isolate.

D INCORRECT**:** **I**ncorrect but plausible (*See A*).

RO Level Justification: Tests the candidate's ability to monitor the operation of PCIS Valves given isolation signals. This question is rated as C/A due to the requirement to assemble, sort, and integrate the plant conditions to predict an outcome. The candidate interprets several parameters to achieve the integrated outcome.

Technical Reference(s): OPL71.017, Rev. 21 (Attach if not previously provided)

2-AOI-64-2D, Rev. 36

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.017 Obj.1 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.017-01 002 #529 (pg 879)
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

Question Cognitive Level:

Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

529. OPL171.017-01 002

The following conditions exist on Unit 1:

- A Loss of Coolant Accident is in progress
- Drywell pressure is 2.2 psig
- Reactor water level is (-) 40 inches
- Main Steam Tunnel temperature is 175º F

Which ONE of the following sets of PCIS groups should have received automatic isolation signals?

A. Groups 1, 3, 6, 8

B. Groups 3, 4, 5, 6

C. Groups 1, 2, 5, 8

D.✓ Groups 2, 3, 6, 8

Excerpt from 1-AOI-64-2D:

BFN Unit 1	Group 6 Ventilation System Isolation	1-AOI-64-2D Rev. 0020 Page 4 of 17

1.0 PURPOSE

This procedure provides symptoms, automatic actions, and operator actions for a Group 6 Ventilation System Isolation.

2.0 SYMPTOMS

NOTES

1) PCIS Group 6 Isolation is initiated by any one of the following signals:
   - Reactor vessel water level at +2.0"
   - Drywell pressure at 2.45 psig
   - Reactor zone exhaust radiation at 72 mr/hr
   - Refuel zone exhaust radiation at 72 mr/hr
2) High Refuel Zone exhaust radiation causes only the automatic actions listed in Section 3.1.
3) Refuel Zone isolation due to Group 6 isolation initiated on Unit 2 or Unit 3.

Excerpts from OPL171.017 Lesson Plan:

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 21

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
b) Group 2	
(1) This group includes Residual Heat Removal (RHR), drywell floor/equipment drain sump, and PSC Head Tank Pump suction valves.	ILT- 2g LOR- 2g 730E927-7,8
(2) The signals which will initiate a Group 2 Isolation are: (2-730E927-13)	730E927-13
(a) RPV low level (+2"or Level 3)	NLO / NLOR- 2
(b) Drywell High Pressure (+2.45 psig)	
(c) Reactor High pressure (100 psig) (SDC) only.	

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 21

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
Figure-3, energization of relay K14 is necessary to allow the AC pilot solenoid to be re-energized.	Figure-3
5. Some Isolation signals are reset via controls other than or in addition to the switches on Panel 9-4. These are: a) Push buttons on Panel 9-3 control resetting of HPCI and RCIC Isolations.	ILT- 3e LOR- 3e Adherence to procedures
b) The H2/O2 Analyzer Group 6 Isolation valves are reset at Panels 9-54 and 9-55 (pushbuttons).	See Table 1
c) The drywell DP air compressor suction and discharge valves are reset using pushbuttons on Panel 9-3.	
d) A pushbutton on Panel 9-13 must be depressed to reset a Group 8 (TIP) Isolation.	730E927-9
e) The LPCI Inboard Injection valves (FCV-74-53/67) (Group 2) are reset using the Isolation Reset pushbuttons located on Panel 9-3.	
6. What is important to note is that no single manipulation will re-open a PCIS valve. It takes two deliberate actions (resetting the Isolation signal and opening the valve). A complete listing of PCIS Valves is in the FSAR Table 5.2.2 NOTE: Details on manipulations required to reset the various groups/Isolation valves are contained in Table 1.	ILT- 3d,e LOR- 3d,e
F. Valve Groups	
1. The Isolation valves at BFN are categorized into one of seven "groups" (1,2,3,4,5,6,& 8), based generally upon the type of system(s) Isolated and associated Isolation signals. NOTE: Group 7 was removed from PCIS long ago. These are HPCI/RCIC drain valves which close on system initiation.	NLO / NLOR- 2
2. Details regarding valves in each group are provided in Table 1. A basic description of each group is as follows:	
a) Group 1 (1) This group includes the Main Steam Isolation Valves (MSIVs), main steam line drains, and reactor water sample line Isolation valves. (2) The signals which will initiate a Group 1 Isolation are as follows: (a) RPV low-low-low level (-122" or Level 1) (b) *MSL High Flow .45 sec TD (135%) (c) *MSL Area High Temperature (189ºF) (d) *MSL Low Pressure (852 psig with the Mode switch in RUN) *(MSIVs and MSL Drains only)	ILT- 2a LOR- 2a 730E927-7,8 730E927-10 730E927-15 NLO / NLOR-2

QA Record. Non-RP - Retain in ECM (Lifetime Retention)

d) Group 4	
(1) This group provides for Isolation of the HPCI System, and includes the HPCI Steam Supply Isolation valves, and the HPCI pump torus suction Isolation valves.	ILT- 2d, 3b LOR- 2d, 3b 730E928-2,3,4
(2) The signals which will initiate a Group 4 Isolation are:	NLO / NLOR- 2
(a) HPCI Area High Temperature (185°F Pump Room/165°F Torus) All 3 units.	
(b) HPCI Steam Line High Flow at 85 psid (200% of rated) (3 sec time delay) All 3 units	
(c) HPCI Steam Line Low Pressure (110 psig) Does not seal-in.	
(d) HPCI High Pressure Between Rupture Discs (10 psig)	
(e) Remote Manual HPCI AUTO-INIT MANUAL ISOLATION pushbutton (1)(2)(3) HS-73-61, if automatic initiation signal is present.	
e) Group 5	
(1) This group provides for Isolation of the RCIC inboard and outboard steam supply Isolation valves.	ILT- 2e, 3b LOR- 2e, 3b 45E626-1,2
(2) The signals which will initiate a Group 5 Isolation are:	
(a) RCIC Area High Temperature (165°F general area and 165°F torus/pump room)	NLO / NLOR- 2
(b) RCIC Steam Line High Flow (150% after 3 sec)	

(c) All 3 units
(d) RCIC Steam Line Low Pressure (U1- 70 psig, U2- 73 psig, U3- 70 psig)
(e) RCIC High Pressure Between Rupture Discs (10 psig)
(f) Remote manual isolation (RCIC AUTO-INIT MANUAL ISOLATION pushbutton, 1-HS-71-54, depressed, only if RCIC initiation signal is present).

Examination Outline Cross-reference:	Level	RO	SRO
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6	Tier #	1	------
AK2.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following:	K/A #	700000AK2.02	
• Breakers, relays	Importance Rating	3.1	------

Proposed Question: **# 8**

Unit 2 is operating at 100% RTP when a Loss of Offsite Power occurs.

Given the conditions above, which **ONE** of the following completes the statements below?

The EDGs are required to start and tie onto their respective 4KV Shutdown Boards within **___(1)___** seconds.

The EECW Pumps will start **___(2)___** seconds after the EDGs re-energize the 4KV Shutdown Boards.

A. (1) 5
(2) 10

B. (1) 5
(2) 14

C. (1) 10
(2) 10

D. (1) 10
(2) 14

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that 5 seconds is the time that it takes for all 4KV Shutdown Board breakers (except those feeding transformers) to open. The second part is incorrect but plausible in that following a loss of Offsite Power and subsequent EDG start; there are many loads that cycle back on at different times. 10 seconds is the time requirement for EDGs to re-energize their respective 4KV Shutdown Boards.

B INCORRECT: The first part is incorrect but plausible (*See A*). The second part is correct (*See D*).

C INCORRECT: The first part is correct (*See D*). The second part is incorrect but plausible *(See A)*.

D **CORRECT:** *(See attached)* In accordance with 0-AOI-57-1A, EDGs will start and tie onto their respective 4KV Shutdown Boards within 10 seconds. For second part, in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500KV) Station Blackout, 14 seconds after the EDGs re-energize the 4160V Shutdown Boards (known as Diesel Generator Power Available - DGVA), all available Emergency Equipment Cooling Water System (EECW) pumps start.

RO Level Justification: Question tests candidate's knowledge of the effect of a loss of Offsite Power on system circuit breakers and relaying to restore power to the 4160KV Shutdown Boards and loads required for EDG operation. This question is rated as Memory due to strictly recalling facts related to breaker operation during a loss of Offsite Power.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 2-AOI-57-1A, Rev.112 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.036, Obj.16 (As available)

Question Source:	Bank #		
	Modified Bank #	ILT Exam Bank OPL171.036-08 012 #918	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

918. OPL171.036-08 013

Which one of the following completes the statements below in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV)/Station Blackout?

Following a Loss of Offsite Power (LOSP), ALL breakers on 4160V Shutdown boards except those feeding transformers, open in a **minimum** of ___(1)___ seconds.

EECW pumps start in a **minimum** of ___(2)___ seconds after the diesel generators re-energize the 4160V Shutdown Boards.

A. (1) five
(2) ten

B. (1) nine
(2) ten

C.✔ (1) five
(2) fourteen

D. (1) nine
(2) fourteen

Excerpt from 0-AOI-57-1A:

BFN Unit 0	Loss of Offsite Power (161 and 500 KV)/Station Blackout	0-AOI-57-1A Rev. 0112 Page 7 of 119

3.0 AUTOMATIC ACTIONS (continued)

S. The following automatic sequence occurs after the loss of power:

1.5 seconds:	All diesel generators start
1.8 seconds:	All electrically operated breakers on 480V Shutdown boards open
5 seconds:	All breakers on 4160V Shutdown boards except those feeding transformers open
6 seconds:	Equipment breakers on 4160V Unit boards open on undervoltage
10 seconds:	Diesel generators tie to their respective Shutdown boards

T. Fourteen (14) seconds after the diesel generators re-energize the 4160V Shutdown Boards, all available EECW pumps start.

1. The following EECW pumps start when Unit 1 or 2 Diesel Generator voltage available:

   A1 (with 0-HS-067-0088, RHRSW PUMP A1 EECW MODE SWITCH in the EECW position) and B3

   C1 (with 0-HS-067-0049, RHRSW PUMP C1 EECW MODE SWITCH in the EECW position) and D3

2. The following EECW pumps start when Unit 3 Diesel Generator voltage available:

   B1 (with 0-HS-067-0089, RHRSW PUMP B1 EECW MODE SWITCH in the EECW position) and A3

   D1 (with 0-HS-067-0048, RHRSW PUMP D1 EECW MODE SWITCH in the EECW position) and C3

U. Security Lighting Diesel Generator starts and loads.

Examination Outline Cross-reference:	Level	RO	SRO
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG	Tier #	2	------
A4.02 (10CFR 55.41.7)	Group #	1	------
Ability to manually operate and/or monitor in the control room: • Synchroscope	K/A #	264000A4.02	
	Importance Rating	3.4	------

Proposed Question: **# 9**

0-SR-3.8.1.1(D), Diesel Generator 'D' Monthly Operability Test is in progress.

Which **ONE** of the following completes the statement below?

Prior to paralleling 'D' EDG with the 'D' 4KV Shutdown Board, EDG frequency is adjusted so that the synchroscope rotates slowly in the ___**(1)**___ direction and EDG voltage is adjusted to be ___**(2)**___ Shutdown Board Voltage.

A. (1) slow
(2) above

B. (1) slow
(2) equal to

C. (1) fast
(2) above

D. (1) fast
(2) equal to

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that paralleling generators is not an intuitive process, and it is reasonable that having the synchrosope turn slowly in the slow direction is the way to properly parallel the EDG to the Shutdown Boards. Second part is incorrect but plausible in that when paralleling sources, the transfer is set up to transfer load to the incoming machine to prevent a reverse power condition, and it is reasonable to assume that setting voltage higher would accomplish transferring load to the incoming machine. However, having a voltage mismatch causes unwanted arcing in the circuit breaker, which could cause damage to the contact surfaces of the EDG output breaker.

B INCORRECT: First part is incorrect but plausible *(See A).* The second part is correct (*See D*).

C INCORRECT: First part is correct (*See D*). The second part is incorrect but plausible (*See A*).

D **CORRECT**: (*See attached*) In accordance with 0-SR-3.8.1.1(D), Diesel Generator D Monthly Operability Test, the speed of 'D' EDG is adjusted so that the synchrosope is turning one revolution in the FAST direction every 15 to 20 seconds. This is done to ensure that the EDG picks up load once the output breaker is closed and prevents a reverse power condition. For second part, in accordance with 0-SR-3.8.1.1(D), EDG voltage is adjusted to match Shutdown Board Voltage. This is done in order to minimize the voltage transient when the breaker is closed, thereby minimizing the arc drawn inside the circuit breaker.

RO Level Justification: Tests the candidate's knowledge of synchrosope operation when paralleling a Diesel Generator to a running source (4KV Shutdown Board). This question is rated as Memory due to strictly recalling facts related to paralleling procedures for the Diesel Generators.

Technical Reference(s): 0-SR-3.8.1.1(D), Rev.55 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.036 Obj. 13 (As available)

Question Source:	Bank #	Dresden 2011 #72	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2011	

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

EXAMINATION ANSWER KEY

10-1 (2011-301) NRC Exam

72 **Points: 1.00**

Per DOS 6600-01, DIESEL GENERATOR SURVEILLANCE, while synchronizing a D/G to an energized bus, the synchroscope should rotate in the ___(1)___ direction, and the INCOMING voltage should be slightly ___(2)___ than the RUNNING voltage.

A. (1) fast (clockwise)
(2) lower

B. (1) fast (clockwise)
(2) higher

C. (1) slow (counter-clockwise)
(2) lower

D. (1) slow (counter-clockwise)
(2) higher

Answer: B

Excerpt from 0-SR-3.8.1.1(D):

BFN Unit 0	Diesel Generator D Monthly Operability Test	0-SR-3.8.1.1(D) Rev. 0055 Page 52 of 94

Date ____________

6.9 Preparing Diesel Generator for Paralleling (continued)

CAUTION

If Shutdown Board D is experiencing abnormal voltage or frequency transients, or if voltage or frequency is out of the specified range, Diesel Generator D should not be paralleled with Shutdown Board D.

[4] On Panel 0-9-23-8

PREPARE the Diesel Generator to be paralleled as follows:

[4.1] **PLACE** DG D BKR 1816 SYNC, 0-25-211-D/20A to the "ON" position, if necessary. ________

[4.2] **CHECK** Shutdown Board D voltage is 3950 to 4400 VOLTS and NOT undergoing abnormal voltage transients using 4KV SD BD D VOLTS, 0-EI-211-D. ________

[4.3] **CHECK** SYSTEM SYNC FREQUENCY, 0-SI-211-CD is 59 to 61 HZ and not undergoing abnormal frequency transients. ________

[4.4] **PULL OUT** and **PLACE** DG D MODE SELECT, 0-HS-82-D/5A in PARALLELED WITH SYSTEM. ________ 1st ________ CV

[4.5] **CHECK** PARALLELED WITH SYSTEM light ILLUMINATED. ________

[4.6] **ADJUST** Diesel Generator D frequency, using DG D GOVERNOR CONTROL, 0-HS-82-D/3A to obtain a Synchroscope needle rotating one revolution every 15 to 20 seconds in the FAST direction. ________

[4.7] **ADJUST** Diesel Generator D voltage (GEN SYNC REF VOLTAGE, 0-EI-82-CD) to match Shutdown Board D voltage (SYSTEM SYNC REF VOLTAGE, 0-EI-211-CD) using DG D VOLT REGULATOR CONT, 0-HS-82-D/2A. ________

Examination Outline Cross-reference:	Level	RO	SRO
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG	Tier #	2	------
K5.05 (10CFR 55.41.5)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): • Paralleling A.C. power sources	K/A #	264000K5.05	
	Importance Rating	3.4	------

Proposed Question: **# 10**

'A' EDG has been started for testing and is being paralleled with Offsite Power in accordance with 0-OI-82, Standby Diesel Generator System.

Given the conditions above, which **ONE** of the following completes the statement below?

Prior to closing the respective EDG output breaker, the DG Mode Select Switch will be placed in the ___**(1)**___ position to prevent EDG ___**(2)**___.

A. (1) Units in Parallel
(2) overload

B. (1) Units in Parallel
(2) overspeed

C. (1) Parallel with System
(2) overload

D. (1) Parallel with System
(2) overspeed

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT**:** The first part is incorrect but plausible in that the DG Mode Switch has three positions (Single Unit, Units in Parallel, and Parallel with System), Units in Parallel is used when two EDGs are to be tied together. The second part is correct (*See C*).

B INCORRECT: The first part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that the EDGs have several trips and the failure to place the DG Mode Switch in the correct position could cause any of the conditions to trip the EDG. The DG will not have any droop (voltage or speed) when the DG Mode Switch is in the "Units in Parallel" position, so it is plausible that the EDG could overspeed.

C **CORRECT**: (*See attached*) In accordance with 0-OI-82, Standby Diesel Generator System, the DG Mode Switch is placed in the "Parallel with System" position before paralleling with Offsite Power. For second part, the DG Mode Switch is placed in the "Parallel with System" position to prevent EDG overload when in parallel with Offsite Power.

D INCORRECT: The first part is incorrect but plausible *(See C)*. The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the procedure to parallel an EDG to Offsite Power and the reason for the EDG Mode Switch position. This question is rated as Memory due to strictly recalling facts related to paralleling procedures for the Diesel Generators.

Technical Reference(s): 0-OI-82, Rev.170 (Attach if not previously provided)

OPL171.038, Rev. 23

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.009 Obj. 6, 14b (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.038-06 001 #1129	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1129. OPL171.038-06 001

Emergency Diesel Generator (DG) 3EA was started for its Monthly Load Test Surveillance.

Which ONE of the following will occur if the DG's output breaker is closed with the DG Mode Selector Switch in the **SINGLE UNIT** position?

A✔ The zero droop governor advances the fuel supply to the diesel to raise output frequency to the governor's setpoint. This will cause the Normal Feeder Breaker to trip on overload.

B. The speed regulator lowers the fuel supply to the diesel to lower output voltage to the governor's setpoint. The will cause the DG Output Breaker to trip on undervoltage.

C. The zero droop governor advances the fuel supply to the diesel to raise output frequency to the governor's setpoint. This will cause the DG to trip on overspeed.

D. The speed regulator lowers the fuel supply to the diesel to lower output voltage to the governor's setpoint. This will cause the Normal Feeder Breaker to trip on reverse power.

Excerpt from 0-OI-82:

BFN Unit 0	Standby Diesel Generator System	0-OI-82 Rev. 0170 Page 94 of 224

8.1 Parallel with System Operation at Panel 9-23 (continued)

CAUTION

Only one Unit 1 and 2 Diesel Generator at a time is allowed to be operated in parallel with system.

[6] **PULL** and **PLACE** the associated Diesel Generator mode selector switch in "PARALLELED WITH SYSTEM".

Diesel	Handswitch Name	Handswitch No.	Panel
A	DG A MODE SELECT	0-HS-82-A/5A	0-9-23-7
B	DG B MODE SELECT	0-HS-82-B/5A	0-9-23-7
C	DG C MODE SELECT	0-HS-82-C/5A	0-9-23-8
D	DG D MODE SELECT	0-HS-82-D/5A	0-9-23-8

CAUTION

Failure of the PARALLELED WITH SYSTEM light to illuminate in the following step could indicate that the DG is still in SINGLE UNIT operation and result in overload when the DG output breaker is closed.

Excerpt from OPL171.038 Lesson Plan:

OPL171.038 DIESEL GENERATORS AND AUX POWER SUPPLY Rev.#23

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (optional)
Shutdown Board. Speed (frequency) droop (decrease) from zero load KILOWATTS to full rated load KILOWATTS is zero. Voltage droop from zero load OUTGOING KILOVARS is zero	
(i) Any fast start signal or DC control power loss automatically places the mode of operation to SINGLE UNIT mode, without turning on the RED light above the SINGLE UNIT position	ILT OBJ. 8 LOR OBJ. 5 NLOR OBJ. 13 NLO OBJ. 13
(ii) Regardless of start signal type, pulling up the MODE SELECT handle and placing it in the desired position will change the mode of operation and light up the associated RED LIGHT above the MODE SELECT switch.	
(b) UNITS IN PARALLEL - Used when two diesel generators are to be tied together. Speed and voltage droop are identical to SINGLE UNIT mode. Breaker positions are sensed to determine if the diesel generator is operating in parallel with any of the other diesel generators. If so, feedback occurs between speed regulators (and voltage regulators) of the paralleled generators so that stable load sharing operation results. Any additional load placed on the boards will be split evenly between the two diesels.	
(c) PARALLEL WITH SYSTEM - Used when the diesel generator is to be parallel with offsite power. To ensure stable load sharing with grid, speed droop from zero load KILOWATTS to full rated load KILOWATTS is approximately five percent. Voltage droop from zero load OUTGOING KILOVARS to full rated load OUTGOING KILOVARS is approximately five percent.	
(3) Governor Control Switch Raises speed governor setpoint (opens throttle) to either	
(a) Increase KILOWATT loading when paralleled to offsite power (output breaker closed) OR	
(b) Increase output frequency when diesel generator is the only supply to a board.	

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.1 (10CFR 55.41.10)	Tier #	3	------
Knowledge of conduct of operations requirements.	Group #	------	------
	K/A #	G2.1.1	
	Importance Rating	3.8	------

Proposed Question: **# 11**

Unit 1 is operating at 100% RTP.

In accordance with OPDP-1, Conduct of Operations, which **ONE** of the following completes the statement below?

The OATC is required to conduct a panel walkdown a **MINIMUM** of once every __________.

A. hour

B. 2 hours

C. 4 hours

D. 6 hours

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (*See attached*) In accordance with OPDP-1, a Licensed Operator is required to conduct a walkdown of Reactor Controls Area Panels approximately once per hour to ensure indications are within established bands. The term “approximately” is used to indicate that a 15 minute grace period may be applied to the one hour requirement in accordance with 0-GOI-300-1, Operator Round Logs, Attachment 16, Operator at the Controls Duty Station Checklist.

B INCORRECT: Incorrect but plausible in that each Unit performs SR-2, Instrument Checks and Observations for Core Thermal Power every 2 hours.

C INCORRECT: Incorrect but plausible in that each Unit performs SR-2, Instrument Checks and Observations for Drywell Leakage every 4 hours.

D INCORRECT**:** Incorrect but plausible in that OPDP-1 requires that the panels outside the Reactor Controls Area be walked down twice a shift.

RO Level Justification: Tests the candidate's knowledge conduct of operations requirements to conduct board walkdowns in accordance with OPDP-1. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): OPDP-1, Rev. 46 (Attach if not previously provided)
0-GOI-300-1, Rev. 212
1-SR-2, Rev. 38

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.071 Obj. 3L (As available)

Question Source:	Bank #	BFN 1510 #66	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2015	

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTION 66 Rev 0

What is the frequency of panel walk downs in accordance with OPDP-1, Conduct of Operations?

The Unit Operator is to perform a panel walk down a minimum of once ____________ (with a 25% grace period).

A. per hour

B. every 2 hours

C. every 4 hours

D. every 6 hours

Answer: A

Excerpts from OPDP-1:

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0046 Page 24 of 71

3.4.3 Control Board Monitoring (continued)

1. If the OATC must become engaged in an evolution that will distract him/her from this primary responsibility of monitoring plant parameters, then responsibility for OATC shall be temporarily transferred to another Unit Operator.

2. If the OATC must leave the Reactor Controls Area in response to an annunciator alarm, or to initiate an action for an emergency affecting the safety of operations, he/she shall make an announcement that they are leaving the area, such that the NUSO is aware of their status.

   a. They shall remain within the confines of the Control Room Surveillance Area in these instances.

3. When temporary relief is necessary, the UO being relieved briefs his/her relief on the following:

   a. General plant status.

   b. Abnormal or unusual conditions.

   c. Any evolutions in progress.

   d. Any actions anticipated during the relief period.

   e. And where he/she may be reached in the plant while absent.

   f. If the person relieving is the CRO on the same unit, the brief need only discuss significant changes or activities.

   g. Upon return, the brief should consist of changes.

   h. The NUSO shall be notified whenever the person holding the OATC position changes.

4. The OATC should not perform any other duties that distract from monitoring the plant.

5. The OATC may perform peer checks for activities inside the Reactor Controls Area.

6. Activities such as answering the phone and assisting maintenance personnel shall be minimized, and conducted by other Licensed personnel on the applicable unit.

C. Licensed operator walkdowns in the main control room will be conducted as follows:

1. A walk down of the Reactor Control Area panels is to be performed approximately once per hour, to ensure that indications are within established bands.

2. The walk down of the panels in the Reactor Controls Area shall be conducted by any licensed operator assigned to the unit.

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0046 Page 25 of 71

3.4.3 Control Board Monitoring (continued)

3. Twice a shift, a walk down of the MCR panels outside the Reactor Controls Area will either be conducted by the assigned Control Room Operator, or the OATC. If it is performed by the OATC, he / she will be temporarily relieved by another licensed individual prior to leaving the Reactor Controls Area.

Excerpt from 0-GOI-300-1:

BFN Unit 0	ATTACHMENT 16 UNIT 1 OPERATOR AT THE CONTROLS DUTY STATION CHECKLIST	0-GOI-300-1/ATT-16 Rev. 0016 Page 4 of 17

Date ________

1.0 HOURLY CONTROL ROOM PANEL CHECKS

[1] Perform panel walkdown within the first 15 minutes of every hour and initial upon completion. Next to the time slot is a section number that should be performed in addition to the hourly board walkdown.

Excerpts from 1-SR-2:

BFN Unit 1	Instrument Checks and Observations	1-SR-2 Rev. 0038 Page 23 of 181

Attachment 2
(Page 1 of 102)

Surveillance Procedure Data Package - Modes 1, 2, & 3

TABLE 1.1 CORE THERMAL POWER AND CORE POWER DISTRIBUTION **DAY SHIFT** WEEK: ______________ to ______________

APPLICABILITY: Mode 1 when ≥ 23% RTP
RECORD the readings as soon as possible after the generator breaker has been closed.

Criteria Source: 3.2.1.1; 3.2.2.1; 3.2.3.1; DEFINITIONS SECTION 1.1 - FSAR 3.7.7

LOCATION: ICS Computer (Case Summary - CSUM)

DAY	TIME Note 2	Core Thermal Power (MWt)	Percent Power (% RTP)	LIMIT (AC)	MFLCPR Note 3	MAPRAT Note 3		MFDLRX Note 3	LIMIT (AC)	Review Initials Unit Operator	Review Initials Unit Supvr
Friday	0800										
	1000										
	1200										
	1400										
	1600										
	1800										

BFN Unit 1	Instrument Checks and Observations	1-SR-2 Rev. 0038 Page 25 of 181

Attachment 2
(Page 3 of 102)

Surveillance Procedure Data Package - Modes 1, 2, & 3

DAY SHIFT WEEK: ______________ to ______________

(1) Compliance with the Licensed Power Limit (LPL) (3952 Mwt) is demonstrated by the following process:

A. No actions are allowed that would intentionally raise core thermal power above 3952 Mwt for any period of time. Small, short-term fluctuations in power that are not under the direct control of the unit operator are not considered intentional.

B. Closely monitor the thermal power during steady-state power operation with the goal of maintaining the two-hour average at or below 3952 Mwt. If the core thermal power average for a 2-hour period is found to exceed 3952 Mwt, Operations take timely action to ensure that thermal power is less than or equal to 3952 MWt. (This is implemented by taking action when any running average less than or equal to the 2 hour average exceeds 3952 Mwt.)

C. The core thermal power for an 8 hour period (8 hr average) is not to exceed 3952 Mwt.

D. If an evolution is expected to cause a transient increase in reactor power that could exceed 3952 Mwt, action should be taken to lower core power prior to performing the evolution.

E. IF power is > 3957, REDUCE power.

F. IF power is 3952 to 3957 MWt after allowing time for recent perturbations to settle, REDUCE power and EVALUATE the trend.

G. IF any running 30 min Avg, 1 hr average, or 2 hr average run is > 3952 MWt, REDUCE power.

(2) Core Thermal Power is normally recorded every 2 hours when required. However, these readings may be marked N/A during TIP trace runs, control rod pattern adjustments, or anytime Core Monitoring System is blocked and/or < 23% power. The Reactor Engineer is responsible for monitoring Core Thermal Limits. Monitoring of Core Thermal Power and other Core Thermal Limits is recommended following completion of planned rise in power and following any unexpected power change. If core monitoring software becomes unavailable, the Shift Manager and Reactor Engineer shall determine the appropriate frequency for monitoring Core Thermal Power but should not exceed 24 hours, using backup core monitoring computer, and taking into consideration current core conditions and margin to thermal limits. Power changes should not normally be made without the core monitoring software being available.

BFN Unit 1	Instrument Checks and Observations	1-SR-2 Rev. 0038 Page 27 of 181

Attachment 2
(Page 5 of 102)

Surveillance Procedure Data Package - Modes 1, 2, & 3

TABLE 1.2 DRYWELL UNIDENTIFIED LEAKAGE **DAY SHIFT** WEEK: ____________ to ____________

APPLICABILITY:	Modes 1, 2 & 3 Readings are required at all times.												
Surveillance Requirements:	3.4.4.1						LOCATION: Panel 1-9-4, 1-FR-77-6						
	Col. A.1	Col. B.1	Col. C.1	Col. D.1	Col. E.1	Col. F.1	Col. G.1	Col. H.1	Col. I.1		Review Init		
Preferred reading times are 0800, 1200 and 1600	Current Point 3 (1-FQ-77-6) Reading (gals) Notes 1, 2	Previous Days 1-FQ-77-6 Reading from Col. A.1 (gals) Note 2	Gallons Pumped Col. A.1 - Col. B.1 Note 2	Current Time Note 2	Previous Days Time from Col. D.1 Note 2	Elapsed Time Col. D.1 - Col. E.1 (min) Note 2	Current Leakrate Col. C.1 ÷ Col. F.1 (gpm) Note 2, 5	Previous Days Leakrate from Col. G.1 (gpm) Note 2	Change in Leakrate Col. G.1 - Col. H.1 (gpm) Note 2, 3, 5	LIMITS (AC)	UO	Unit Supvr Note 4	

Examination Outline Cross-reference:	Level	RO	SRO
295003 (APE 3) Partial or Complete Loss of A.C. Power / 6	Tier #	1	------
AA1.03 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER:	K/A #	295003AA1.03	
• Systems necessary to assure safe plant shutdown	Importance Rating	4.4*	------

Proposed Question: **# 12**

Unit 1 is operating at 100% Rated Thermal Power (RTP) with the following conditions:

- Loss of Offsite Power occurs
- ‘A’ Emergency Diesel Generator (EDG) mechanically fails

Subsequently:

- Loss of Coolant (LOCA) occurs
- Reactor Water Level is (-) 130 inches and lowering
- Drywell Pressure is 10 psig and rising
- Reactor Pressure is 600 psig and lowering

Give the conditions above, which **ONE** of the following completes the statements below?

RHR Pump 1C ___**(1)**___ automatically start.

Core Spray Pump 1C ___**(2)**___ automatically start.

A. (1) will
(2) will

B. (1) will
(2) will NOT

C. (1) will NOT
(2) will

D. (1) will NOT
(2) will NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). The second part is incorrect but plausible in that the RHR and Core Spray System interlocks are very complex and often confused at Browns Ferry; Core Spray Pump 1C has power from its associated EDG and it is reasonable to assume that a candidate could mistake the interlocks.

B **CORRECT**: *(See attached)* RHR Pumps do not have an interlock with their companion pump's power supply that would prevent an automatic start. 1C RHR Pump has power from its associated EDG and will start after the requisite time delay. For second part, in accordance with 1-OI-75, Core Spray System, a Core Spray Pump will not start when its companion pump does not have power available. Because 'A' EDG cannot be started during the Loss of Offsite Power, 4KV Shutdown Board 'A' will be de-energized. Therefore, 1C Core Spray Pump will not start because 1A Core Spray Pump does not have power.

C INCORRECT**:** The first part is incorrect but plausible in that the RHR and Core Spray System interlocks are very complex at Browns Ferry, and it is reasonable to assume that a candidate could mistake the interlocks. RHR Pumps do not have an interlock with the companion pump's power supply. The second part is incorrect but plausible (*See A*).

D INCORRECT: The first part is incorrect (*See C*). The second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the effect of a Loss of Offsite Power with a LOCA on the Core Spray and RHR Pump start sequences. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (4) Assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems and/or safety functions.

Technical Reference(s): 1-OI-75, Rev.37 (Attach if not previously provided)
OPL171.045, Rev.22

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.045 Obj. 2h (As available)

Question Source:			
	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.045-02 006 #1528	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1528. OPL171.045-02 005

Given the following conditions:

- Unit 2 was at 100% power
- A small break LOCA has occurred due to a faulted weld in the bottom head drain line
- A reactor scram has occurred due to high DW pressure
- At 0930, the 'A' 4 KV SD Board lost power
- 'A' DG started but failed to automatically tie to the bus
- At time 1000, RPV pressure reached 450 psig
- At time 1015, RPV pressure dropped to 300 psig
- No other operator actions have been performed

Which ONE of the following completes the statements below?

At time 1005, Core Spray Pumps ___(1)___ running.

At 1015, operating Core Spray Pumps are___(2)___.

A.✓ (1) 2B and 2D ONLY are
(2) injecting

B. (1) 2B, 2C, and 2D are
(2) injecting

C. (1) 2B, 2C, and 2D are
(2) NOT injecting

D. (1) 2B and 2D ONLY are
(2) NOT injecting

Excerpt from 1-OI-75:

BFN Unit 1	Core Spray System	1-OI-75 Rev. 0037 Page 12 of 141

3.4 Initiations

A. The CS System will auto initiate from the following signals:

1. RPV water level at or below -122 inches

2. DW pressure at or above 2.45 psig and RPV pressure at or below 450 psig

B. Manually stopping a Core Spray pump after auto initiation will disable automatic restart of that pump until the initiation signal is clear and has been reset. The affected Core Spray pump may still be started manually.

3.5 Isolations

A. PSC PUMP SUCTION INBD and OUTBD ISOL VALVE, 1-FCV-75-57 and 1-FCV-75-58, will close on Group II Isolation, tripping PSC Head Tank Pumps 1A and 1B.

B. The Core Spray test valve receives an auto closure signal on any CS auto initiation.

C. The Core Spray minimum flow valves receive a closure signal when flow is approximately 2600 gpm rising and receives an open signal when flow lowers to approximately 2200 gpm.

3.6 Trips

A. Electrical

3.7 Interlocks

A. The CS inboard and outboard injection valves have in-line valve interlocks to prevent both valves from being opened with RPV pressure at or above 450 psig. Both receive auto open signals when there is a CS initiation signal and RPV pressure is below 450 psig. The inboard valve may be throttled immediately after initiation.

3.8 Power Supplies

A. Core Spray breaker closure with the breaker racked to the test position will result in a auto start of the EECW pumps if the NVA or DGVA relay is allowed to time-out prior to opening of the breaker.

B. If one pump in a Core Spray System has its 4kV Shutdown Board de-energized, NEITHER pump in that loop will auto start and may **NOT** be considered operable per Tech Spec 3.5.1/3.5.2.

Excerpt from OPL171.045:

Obj. ILT 2.h
Obj. LOR 1.h
Obj. LOR 1.g

i. If a shutdown board is deenergized, neither CS pump in that system will automatically start. Systems are designed for only two-pump operation in automatic.

Example: Assume Shutdown Board A has no power at time of CS initiation. CS pumps A and C will not receive an automatic start signal. The C pump could be manually started or, if it was running prior to the initiation, it would remain running. If power is restored to Shutdown Board A, both pumps would automatically start.

Must have either K29 or K30 contact closed

AND

K31 or K32 contact closed.

Note: DGVA is deenergize to function.

ii. If a CS pump is stopped with an initiation signal present, it will not automatically start again until the CS initiation logic is reset. This condition is indicated by an amber light on Panel 9-3 (vertical section). This is accomplished by the stop handswitch energizing K21 which blocks auto start signals until the CS initiation signal is reset dropping out K25. Note that K21 is also deenergized by a loss and restoration of SD boards. (Loss of offsite power with initiation signal present) such that all pumps will restart when the DGs repower the boards.

iii. Unit 1 & 2 Only - An accident signal on one Unit will prevent the nonpreferred pumps on the opposite Unit from starting.

k. CS pumps will trip due to electrical fault protection. This condition is indicated by:

i. White and green light above the pump control switch.

ii. 2) CS Sys pump trip alarm

Obj. ILT 5.f
Obj. LOR 4.f

l. If the CS inboard injection valve (FCV 75-25) is throttled with a CS initiation signal present, an amber light above the control switch will be lit and only manual valve control will be available until the CS initiation, K13, is reset.

Examination Outline Cross-reference:	Level	RO	SRO
295006 (APE 6) Scram / 1	Tier #	1	------
G2.4.1 (10CFR 55.41.10)	Group #	1	------
Knowledge of EOP entry conditions and immediate action steps.	K/A #	295006G2.4.1	
	Importance Rating	4.6	------

Proposed Question: **# 13**

Which **ONE** of the following completes the statement below in accordance with NOTE #1 from 3-EOI-1, RPV CONTROL?

Following a Reactor SCRAM, the Unit 3 Reactor will remain subcritical **WITHOUT** boron under all conditions when _________.

A. All Control Rods are at position '02'

B. Reactor Power is on range 7 of the IRMs and lowering

C. All Control Rods are inserted to or beyond position '00' except two that are fully withdrawn

D. Any 19 Control Rods are at notch '02' with all other Control Rods fully inserted

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that this was the case for Unit 1 ONLY in the past.

B INCORRECT: Incorrect but plausible in that when used in the EOIs, the Reactor is subcritical when Reactor Power on Range 7 of the IRMs. However, this does not ensure that the Reactor will remain subcritical under all conditions without boron.

C INCORRECT: Incorrect but plausible in that this would be true if all Control Rods except one are inserted to or beyond position '00'. In accordance with EOIPM, 0-V-M for ARC-1 from EOI-1A, ATWS RPV Control Bases related to NOTE #1, positive confirmation that the Reactor will remain subcritical under all conditions is best obtained by determining that no Control Rod is withdrawn beyond the Maximum Subcritical Banked Withdrawal Position.

D **CORRECT**: (*See attached*) In accordance with NOTE #1 from EOI-1, RPV CONTROL and EOI-1A, ATWS RPV CONTROL, following a Reactor SCRAM, a Reactor will remain subcritical without boron under all conditions when any 19 Control Rods are at notch '02' with all other Control Rods fully inserted. Following the EOI-1 entry conditions requiring a Reactor SCRAM, this determines the immediate action steps that will be performed either in EOI-1 and/or EOI-1A.

RO Level Justification: Tests the candidate's knowledge of Emergency Operating Instruction (EOI) entry conditions and immediate action steps to mitigate a SCRAM and Anticipated Transient Without a SCRAM (ATWS). This question is rated as Memory due to the requirement to strictly recall facts related to EOIs.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 3-EOI-1, Rev. 13 (Attach if not previously provided)
3-EOI-1A, Rev. 2
EOIPM 0-V-M, Rev. 0

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.202 Obj. 2 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1804 #5	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

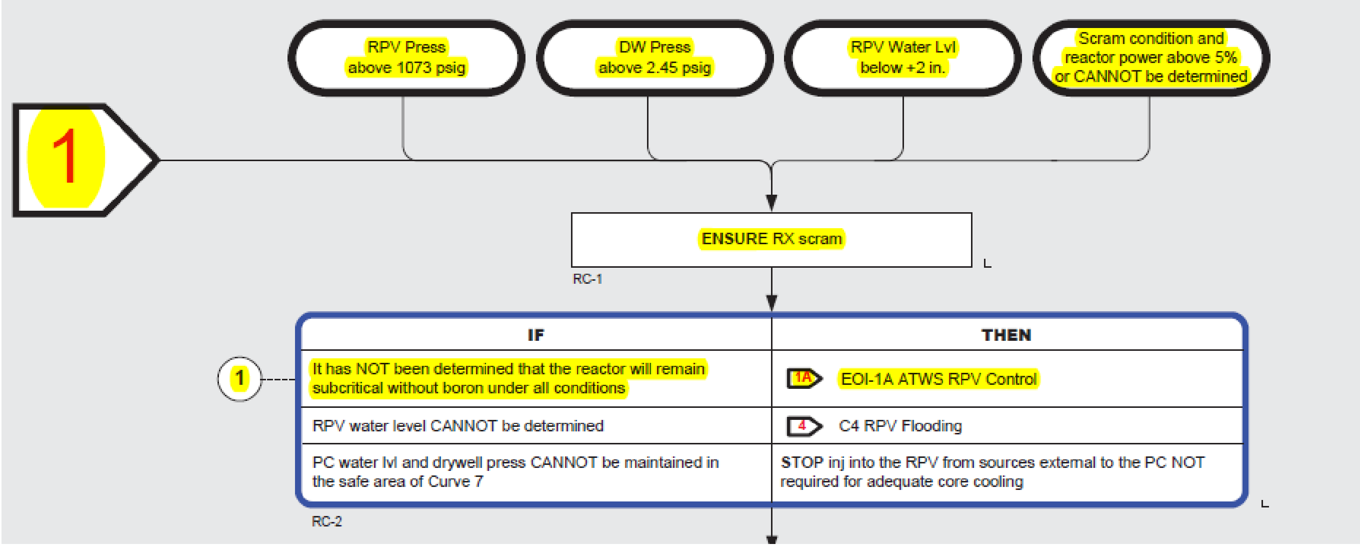
Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Excerpts from 3-EOI-1: Illustrates EOI-1 and EOI-1A Entry Conditions as it relates to NOTE #1

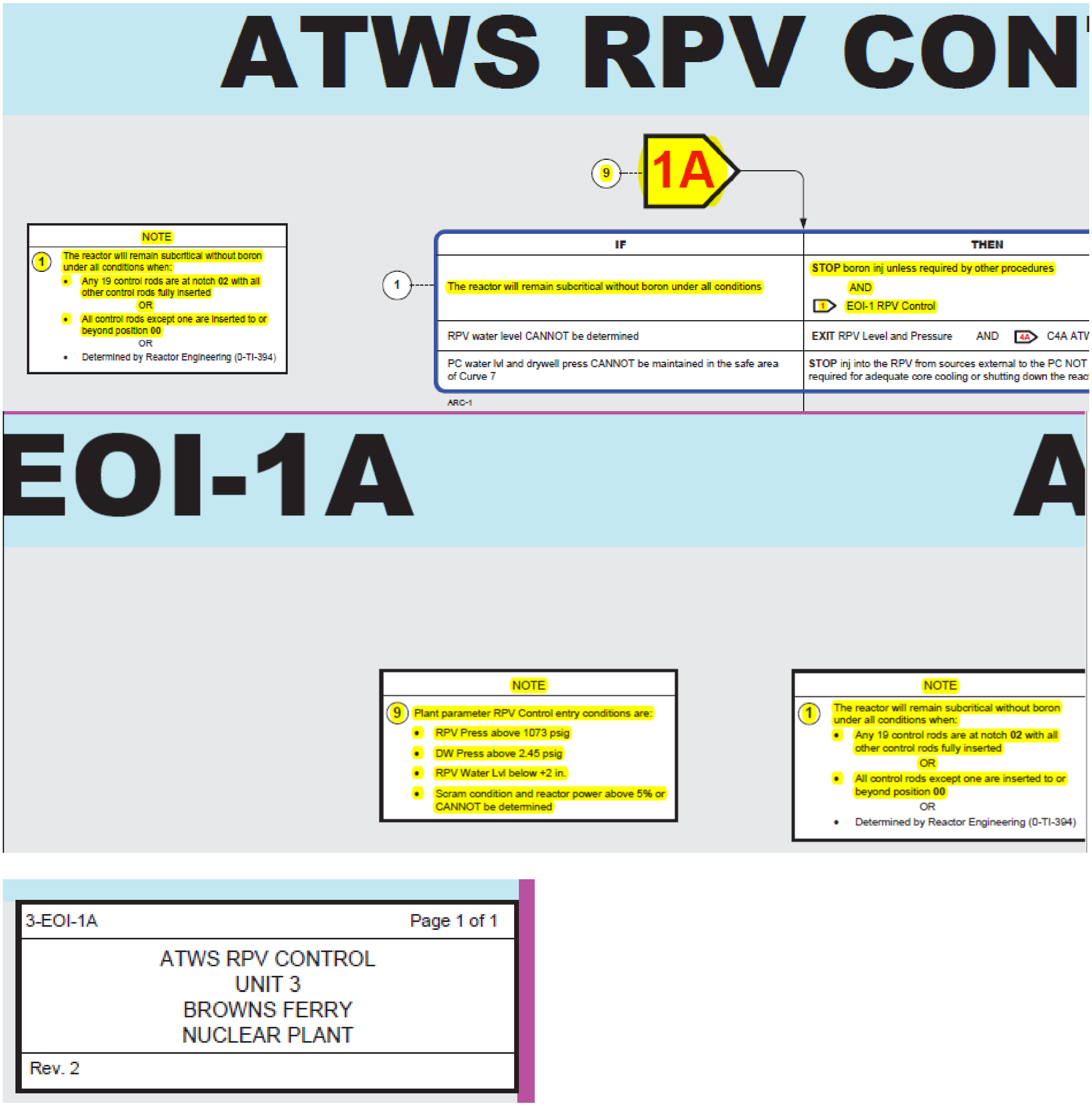
RPV CONTROL



NOTE	
1	The reactor will remain subcritical without boron under all conditions when: • Any 19 control rods are at notch 02 with all other control rods fully inserted OR • All control rods except one are inserted to or beyond position 00 OR • Determined by Reactor Engineering (0-TI-394)
2	TSC staff may recommend an alternate curve for Station Blackout per 0-AOI-57-1A

3-EOI-1	Page 1 of 1
RPV CONTROL UNIT 3 BROWNS FERRY NUCLEAR PLANT	
Rev. 13	

Excerpts from 3-EOI-1A: Illustrates EOI-1 and EOI-1A Entry Conditions as it relates to NOTE #1



Excerpt from 1-EOI-1 (previous revision): Supports Distractor 'A'

1-EOI-1	PAGE 1 OF 1
RPV CONTROL	
UNIT 1 BROWNS FERRY NUCLEAR PLANT	
REV: 0	

NOTES	
1	THE REACTOR WILL REMAIN SUBCRITICAL WITHOUT BORON UNDER ALL CONDITIONS WHEN: • ALL CONTROL RODS ARE INSERTED TO OR BEYOND POSITION **02** OR • ALL CONTROL RODS EXCEPT ONE ARE INSERTED TO OR BEYOND POSITION **00** OR • DETERMINED BY REACTOR ENGINEERING
2	TSC STAFF MAY RECOMMEND AN ALTERNATE CURVE FOR STATION BLACKOUT PER 0-AOI-57-1A

Examination Outline Cross-reference:	Level	RO	SRO
295025 (EPE 2) High Reactor Pressure / 3	Tier #	1	------
EK2.08 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:	K/A #	295025EK2.08	
• Reactor/turbine pressure regulating system: Plant-Specific	Importance Rating	3.7	------

Proposed Question: **# 14**

Unit 2 is operating at 100% RTP when an inadvertent closure of **ALL** MSIVs occurs.

Which **ONE** of the following completes the statement below?

Following the closure of the MSIVs, the digital Electro-Hydraulic Control (EHC) System will transfer to ___**(1)**___ Pressure Control at ___**(2)**___.

A. (1) Reactor
(2) 700 psig

B. (1) Reactor
(2) 955 psig

C. (1) Header
(2) 700 psig

D. (1) Header
(2) 955 psig

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that Reactor Pressure Control is the normal control mode. The second part is correct (*See C*).

B INCORRECT**:** The first part is incorrect but plausible (*See A*). The second part is incorrect but plausible in that this is the normal Control Pressure.

C **CORRECT**: (*See Attached*) In accordance with 2-OI-47, Turbine-Generator System, Attachment 1, while in Reactor Pressure Control, if header pressure drops below 700 psig, the EHC Controlling Pressure logic will automatically transfer Header Pressure Control. When the MSIVs are closed, Main Steam Header Pressure drops below 700 psig within seconds. For second part, EHC will automatically initiate the pressure control method swap to Header Pressure at 700 psig.

D INCORRECT: First part is correct but plausible *(See C).* Second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's knowledge of the operation of the EHC System as it relates to a closure of the MSIVs. When the MSIVs are closed, Reactor Pressure will rise and be controlled by the MSRVs. This question is rated as Memory due to the requirement to strictly recall facts.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (3) The progression of an event.

Technical Reference(s): 2-OI-47, Rev.186 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.228.1.b (As available)

Question Source:

Bank #	BFN 1205 #21
Modified Bank #	
New	

(Note changes or attach parent)

Question History: Last NRC Exam 2012

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Copy of Bank Question:

QUESTION 21

Unit 2 has experienced an inadvertent MSIV closure.

- RCIC is controlling Reactor Level from (+)2 to (+)51 inches
- HPCI is controlling Reactor Pressure from 800 to 1000 psig
- MSIVs remain closed

Which ONE of the following completes the statement below?

The digital EHC system is in __(1)__ pressure control mode with the pressure setpoint currently set at __(2)__ psig.

A. (1) Header
(2) 700

B. (1) Header
(2) 955

C. (1) Reactor
(2) 700

D. (1) Reactor
(2) 955

Correct: A

Excerpt from 2-OI-47:

BFN Unit 2	Turbine-Generator System	2-OI-47 Rev. 0186 Page 238 of 280

Attachment 1
(Page 6 of 22)

EHC Control System
Panel 2-9-7
Controls and Indications

1.2 Components (REFER TO Figures 1 and 2) (continued)

- HEADER PRESS A (B) BYPASS pushbuttons (2-HS-47-16A, 16B)

  These pushbuttons allow the operator to manually bypass the header pressure inputs from the two pressure transmitters, 2-PT-001-0016A and 2-PT-001-0016B. Each pushbutton will illuminate when manually selected or when the associated header pressure input has been automatically bypassed.

  Any header pressure signal which becomes faulted will be automatically bypassed and its respective bypass pushbutton will illuminate. Once the pressure input is corrected, the respective bypass pushbutton is required to be manually depressed to unbypass the pressure input.

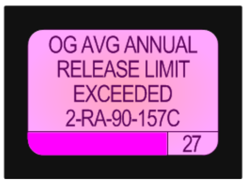
  To remain in Header Pressure Control, the EHC Controlling Pressure logic will allow only one header pressure input to be manually or automatically bypassed. If a second header pressure input is manually or automatically bypassed, the EHC Controlling Pressure logic will automatically transfer to Reactor Pressure Control.

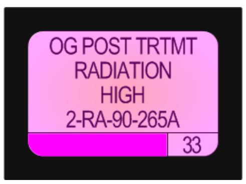
  While in reactor pressure control, if header drops below 700 psig, the EHC Controlling Pressure logic will automatically transfer to header pressure control. If desired, the operator can manually transfer pressure control back to reactor pressure control. When header pressure rises above 875 psig, the pressure control logic will re-enable the header pressure auto transfer logic.

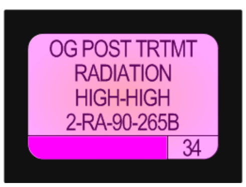
Examination Outline Cross-reference:	Level	RO	SRO
295017 (APE 17) High Off-Site Release Rate / 9	Tier #	1	------
AA2.03 (10CFR 55.41.10)	Group #	2	------
Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE:	K/A #	295017AA2.03	
• Radiation levels: Plant-Specific	Importance Rating	3.1	------

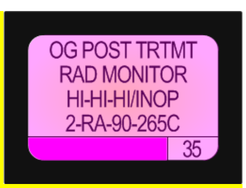
Proposed Question: **# 15**

In accordance with Unit 2 ARPs, which **ONE** of the following, when alarming, requires that Operators ensure 2-FCV-66-28, OFFGAS SYSTEM ISOLATION VALVE, is CLOSED?

A. 

B. 

C. 

D. 

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that this alarm is often confused due to being similar to the Window 35 alarm. A probable cause of this alarm is possible resin trap failure (RWCU or Condensate demins) or fuel damage, and may require Reactor Power reduction

B INCORRECT: Incorrect but plausible in that a probable cause of this alarm is possible resin trap failure (RWCU or Condensate demins) or fuel damage, and may require Reactor Power reduction. The automatic action that occurs on receipt of this alarm is the Offgas System 2-FCV-66-113B, ADSORBER BYPASS VALVE, closes and 2-FCV-66-113A, ADSORBER INLET VALVE, opens.

C INCORRECT: Incorrect but plausible in that a probable cause of this alarm is possible resin trap failure (RWCU or Condensate demineralizers) or fuel damage. The receipt of this alarm requires possible action in accordance with Technical Specifications and may also require a reduction in Reactor Power.

D **CORRECT**: *(See attached)* A probable cause of this alarm is possible resin trap failure (RWCU or Condensate demineralizers) or fuel damage, and may require Reactor Power reduction. 2-FCV-66-28, OFFGAS SYSTEM ISOLATION VALVE, automatically closes after a five (5) second time delay upon receipt of 2-9-4C, Window 35 annunciator. Subsequent Operator actions include ensuring 2-FCV-66-28 is closed.

RO Level Justification: Tests the candidate's ability to operate and monitor the Process Radiation Monitor System and isolation valve operation as it pertains to Offsite Release Rates given four different alarms. This question is rated as C/A due to the integrated aspects of the question to predict an outcome when given the different alarms. This requires mentally using specific knowledge with parameter and system monitor and controls and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 2-ARP-9-4C, Rev. 35 (Attach if not previously provided)
OPL171.030, Rev. 20

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.030, Obj. 6, 10 (As available)

Question Source:

Bank #	BFN 1909 #29	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam 2019

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

Examination Outline Cross-reference:

295017 (APE 17) High Off-Site Release Rate / 9

AA1.07 (10CFR 55.41.7)

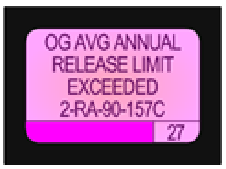
Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE:

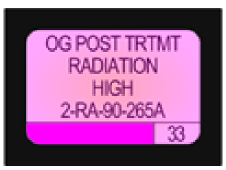
- Process radiation monitoring system

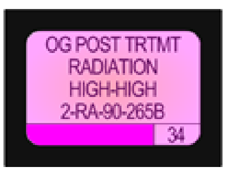
Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295017AA1.07	
Importance Rating	3.4	------

Proposed Question: **# 29**

In accordance with the Unit 2 Alarm Response Procedures, which **ONE** of the following, when alarming, requires that Operators ensure 2-FCV-66-28, OFFGAS SYSTEM ISOLATION VALVE, is CLOSED?

A. 

B. 

C. 

D. 

Proposed Answer: **D**

Excerpts from 2-ARP-9-4C: (all of the provided alarms)

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0035 Page 44 of 44

OG POST TRTMT RAD MONITOR HI-HI-HI/INOP 2-RA-90-265C	35

(Page 1 of 1)

Sensor/Trip Point:

2-RM-90-265A	6.2×10^5 cps
2-RM-90-266A	6.2×10^5 cps

Sensor Location: 2-RE-90-265 Panel 2-25-94 Off-Gas Building
2-RE-90-266 Elevation 538.5

Probable Cause:
A. Resin trap failure (RWCU or Condensate demins).
B. Fuel damage.

Automatic Action: OFFGAS SYSTEM ISOLATION VALVE 2-FCV-66-28 closes after a 5 second time delay

Operator Action:
A. CHECK alarm condition on the following
- OFFGAS RADIATION recorder, 2-RR-90-266 on Panel 2-9-2
- OG POST-TREATMENT CHAN A RAD MON RTMR radiation monitor, 2-RM-90-266A on Panel 2-9-10.
- OG POST-TREATMENT CHAN B RAD MON RTMR radiation monitor, 2-RM-90-265A on Panel 2-9-10.

B. ENSURE OFF-GAS SYSTEM ISOLATION VALVE, 2-FCV-66-28 has the Mechanical Restraint DISENGAGED and 2-FCV-66-28 is CLOSED.
C. REFER TO 2-AOI-66-2.

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0035 Page 34 of 44

OG AVG ANNUAL RELEASE LIMIT EXCEEDED 2-RA-90-157C	27

(Page 1 of 2)

Sensor/Trip Point:

2-RE-90-157 133 R/hr (Alarm from recorder) (1.33×10^5 mR/hr)

Sensor Location: Elevation 565'
Turbine Building
Column T-7 B-LINE
Recorder is on Panel 2-9-2.

Probable Cause:
A. Abnormal flow in the off gas system.
B. Resin trap failure (RWCU or Condensate Demins).
C. Fuel damage.

Automatic Action: None

Operator Action:
A. To determine if the Off Gas Annual Release Rate Limit is exceeded, **PERFORM** the following:
1. **CHECK** alarm condition on the following:
   - OFFGAS RADIATION recorder, 2-RR-90-266, Panel 2-9-2.
   - OG PRETREATMENT RAD MON RTMR monitor, 2-RM-90-157, Panel 2-9-10.

NOTE
High off-gas flow can sweep settled particulates into flow stream causing momentary rise in monitor parameters. Low off-gas flow can result in improper dilution causing rise in monitor parameters.

2. **CHECK** off-gas flow and monitor sample flow normal.
3. **NOTIFY** Radiation Protection.
4. **REQUEST** Chemistry perform radiochemical analysis to determine source.
5. With OPS MGT and Shift Manager's permission, **PLACE** charcoal beds in parallel with another unit. **REFER TO** 2-OI-66.
6. **IF** fuel damage is suspected, **THEN**
   REFER TO 2-SR-3.4.6.1 for dose equivalent iodine - 131 determination.

Continued on Next Page

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0035 Page 35 of 44

OG AVG ANNUAL RELEASE LIMIT EXCEEDED 2-RA-90-157C, Window 27
(Page 2 of 2)

Operator Action: (Continued)

7. **REFER TO** 0-SI-4.8.B.1.a.1 and 2-SR-3.4.6.1-a for ODCM compliance and to determine if power level reduction is required.
8. **IF** directed by Shift Manager or Unit Supervisor/SRO, **THEN** **REDUCE** reactor power to maintain off-gas radiation within ODCM limits.
9. **REFER TO** EPIP-1.

References:	2-45E620-4	2-47E610-90-1	GE 2-729E814 series
	2-47E610-55-14	FSAR Sections 1.6.4.4.6, 7.12.2.2, and 13.6.2	Technical Specifications 3.4.6.1-a.
	Technical Requirements Manual Sections 3.3.5.1, 3.3.9.1, & 3.7.2.1		

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0035 Page 41 of 44

OG POST TRTMT RADIATION HIGH 2-RA-90-265A	33

(Page 1 of 1)

Sensor/Trip Point:

2-RM-90-265A	6.2×10^4 cps
2-RM-90-266A	6.2×10^4 cps

Sensor Location: 2-RE-90-265 Panel 2-25-94 Off-Gas Building,
2-RE-90-266 Elevation 538.5

Probable Cause:
A. Off-Gas flow change.
B. Adsorber lineup change.
C. Resin Trap Failure (RWCU or Condensate demins).
D. Fuel damage.

Automatic Action: Close signal to ADSORBER BYPASS VALVE 2-FCV-66-113B. Open signal to ADSORBER INLET VALVE 2-FCV-66-113A.

Operator Action:
A. **CHECK** alarm condition and **MONITOR** activity on the following:
- OFFGAS RADIATION recorder, 2-RR-90-266 on Panel 2-9-2.
- OG POST-TREATMENT CHAN A RAD MON RTMR radiation monitor, 2-RM-90-266A on Panel 2-9-10.
- OG POST-TREATMENT CHAN B RAD MON RTMR radiation monitor, 2-RM-90-265A on Panel 2-9-10.

B. **ENSURE** Charcoal Adsorbers in service.
C. **NOTIFY** Unit 1 and 3 operators of conditions and that verification of proper operation of Unit 1 and 3 Off-Gas system is required.
D. **CHECK** STACK GAS/CONT RM RADIATION recorder, 0-RR-90-147 on Panel 1-9-2.
E. **NOTIFY** Radiation Protection.
F. **REQUEST** Chemistry perform radiochemical analysis to determine source.
G. **REFER TO** 0-SI-4.8.b.1.a.1 and 0-SR-3.4.6.1-a for ODCM compliance and to determine if power level reduction is required.
H. **IF** directed by Shift Manager or Unit Supervisor/SRO, **THEN** **REDUCE** reactor power to maintain off-gas radiation within ODCM limits.

References:
2-45E620-4 2-47E610-90-2 GE 2-729E814-6
FSAR Sections 1.6.4.4.6, 7.12.2.2, 7.12.2.3, 7.12.3.3, 9.5.4, and 13.6.2
Technical Specifications Section 3.4.6.1-a.
Technical Requirements Manual Sections 3.3.5.1, 3.3.9.1, & 3.7.2.1

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0035 Page 42 of 44

OG POST TRTMT RADIATION HIGH-HIGH 2-RA-90-265B	34

(Page 1 of 2)

Sensor/Trip Point:

2-RM-90-265A	3.1×10^5 cps
2-RM-90-266A	3.1×10^5 cps

Sensor Location: 2-RE-90-265 Panel 2-25-94 Off-Gas Building
2-RE-90-266 Elevation 538.5

Probable Cause:
A. Off-Gas flow change.
B. Adsorber lineup change.
C. Resin trap failure (RWCU or Condensate demins).
D. Fuel damage.

Automatic Action: None

Operator Action:
A. **CHECK MONITOR** high activity on the following:
- OFFGAS RADIATION recorder, 2-RR-90-266 on Panel 2-9-2.
- OG POST-TREATMENT CHAN A RAD MON RTMR radiation monitor, 2-RM-90-266A on Panel 2-9-10.
- OG POST-TREATMENT CHAN B RAD MON RTMR radiation monitor, 2-RM-90-265A on Panel 2-9-10.

B. **ENSURE** Charcoal Adsorbers in service.
C. **NOTIFY** Unit 1 and 3 operators of conditions and that verification of proper operation of Unit 1 and 3 Off-Gas system is required.
D. **CHECK** STACK GAS/CONT RM RADIATION RECORDER, 0-RR-90-147 on Panel 1-9-2.
E. **NOTIFY** Radiation Protection.
F. **REQUEST** Chemistry perform radiochemical analysis to determine source.
G. **REFER TO** 0-SI-4.8.b.1.a.1 and 0-SR-3.4.6.1-a for Technical Specification compliance and to determine if power level reduction is required.
H. **IF** directed by Shift Manager or Unit Supervisor/SRO, **THEN** REDUCE reactor power to maintain off-gas radiation within ODCM limits.

Excerpts from 2-AOI-66-2:

BFN Unit 2	Offgas Post-Treatment Radiation HI-HI-HI	2-AOI-66-2 Rev. 0022 Page 4 of 8

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a High-High-High radiation condition in the Offgas System.

2.0 SYMPTOMS

A. Annunciators in alarm will include, but are NOT limited to, the following:

1. OG POST TRTMT RADIATION HIGH (2-XA-55-4C, Window 33).
2. OG POST TRTMT RADIATION HIGH-HIGH (2-XA-55-4C, Window 34).
3. OG POST TRTMT RAD MONITOR HI-HI-HI/INOP (2-XA-55-4C, Window 35)
4. OG PRETREATMENT RADIATION HIGH (2-XA-55-3A, Window 5).
5. STACK GAS RADIATION HIGH (2-XA-55-3A, Window 13).
6. STACK GAS RADIATION HIGH-HIGH (2-XA-55-3A, Window 6).
7. OG AVG ANNUAL RELEASE LIMIT EXCEEDED (2-XA-55-4C, Window 27).
8. OFFGAS ISOLATION VALVE CLOSED (2-XA-55-7A, Window 4).

B. Increased activity on OFFGAS RADIATION recorder, 2-RR-90-266, Panel 2-9-2.

C. Increased activity on STACK GAS RADIATION recorder, 0-RR-90-147, located on Panel 1-9-2.

BFN Unit 2	Offgas Post-Treatment Radiation HI-HI-HI	2-AOI-66-2 Rev. 0022 Page 5 of 8

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** scram has not occurred, **THEN**

PERFORM the following:

[1.1] **IF** core flow is above 60%, **THEN**

REDUCE core flow to between 50-60%. ☐

[1.2] **MANUALLY SCRAM** the Reactor. (Reference 2-AOI-100-1). ☐

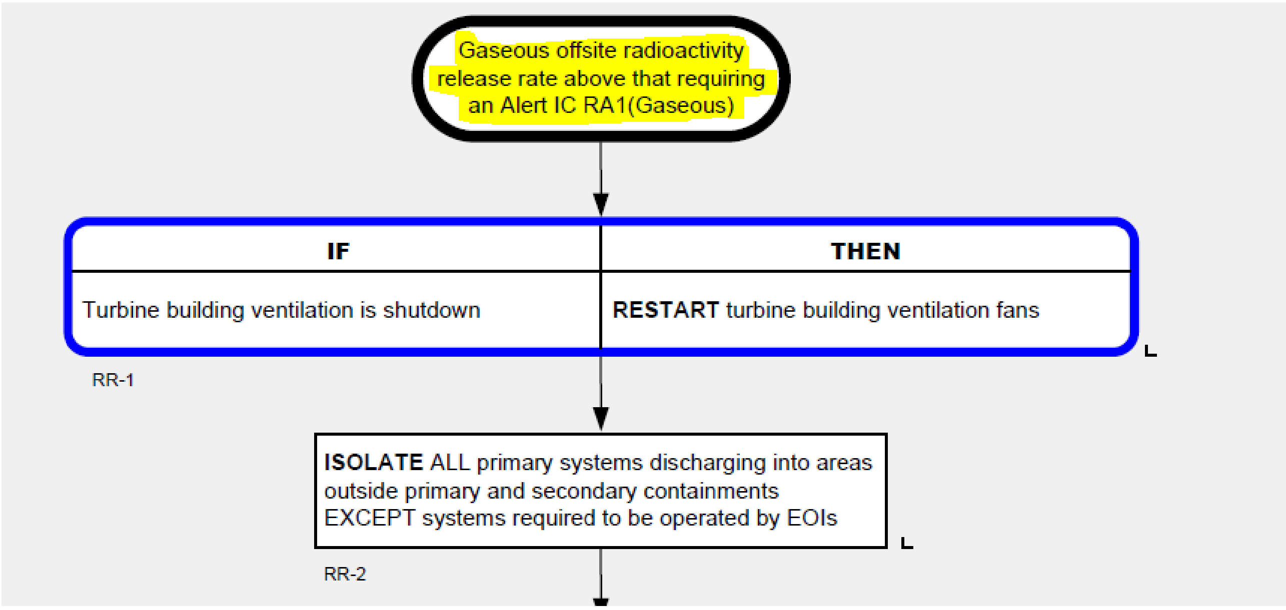
4.2 Subsequent Actions

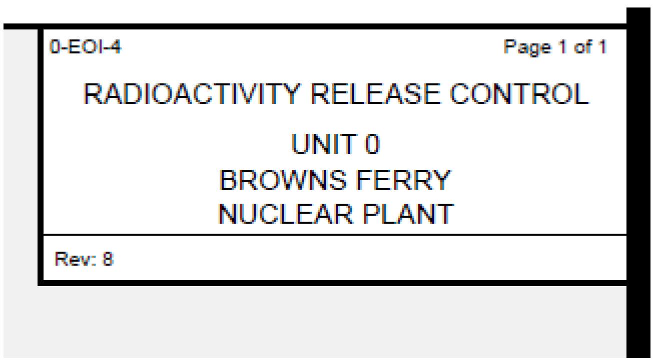
[1] **IF** OFFGAS SYSTEM ISOLATION VALVE, 2-FCV-066-0028 has been mechanically restrained open due to plant conditions **THEN**

DISENGAGE 2-FCV-066-0028 mechanical restraint by rotating the restraining handwheel fully in the counterclockwise direction, locally at the Stack. (Otherwise N/A) ☐

[2] **VERIFY CLOSED** OFFGAS SYSTEM ISOLATION VALVE, 2-FCV-66-28 on Panel 2-9-53 or locally. ☐

Excerpt from 0-EOI-4:




0-EOI-4 Page 1 of 1

RADIOACTIVITY RELEASE CONTROL

UNIT 0
BROWNS FERRY
NUCLEAR PLANT

Rev: 8

Examination Outline Cross-reference:	Level	RO	SRO
295024 (EPE 1) High Drywell Pressure / 5	Tier #	1	------
EK3.06 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: • Reactor SCRAM	K/A #	295024EK3.06	
	Importance Rating	4.0*	------

Proposed Question: **# 16**

Unit 2 is operating at 100% RTP.

Which **ONE** of the following completes the statements below?

In accordance with OPDP-1, Conduct of Operations, when a High Drywell Pressure condition exists and continues to rise, the Operators are required to **FIRST** manually ___**(1)**___.

In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, when Drywell Pressure reaches 2.2 psig, the reason a manual Reactor SCRAM is inserted is that ___**(2)**___.

A. (1) trip the Main Turbine, **THEN** insert a Reactor SCRAM
(2) a pre-determined trigger value has been reached

B. (1) trip the Main Turbine, **THEN** insert a Reactor SCRAM
(2) an automatic Reactor SCRAM should have occurred

C. (1) insert a Reactor SCRAM, **THEN** trip the Main Turbine
(2) a pre-determined trigger value has been reached

D. (1) insert a Reactor SCRAM, **THEN** trip the Main Turbine
(2) an automatic Reactor SCRAM should have occurred

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate confuses the order of operations during a condition requiring a SCRAM. Second part is correct (*See C).*

B INCORRECT: First part is incorrect but plausible (*See A).* Second part is incorrect but plausible if candidate confuses the Trigger Value of 2.2 psig Drywell Pressure with the SCRAM setpoint of 2.45 psig. Another Trigger Value of 2.0 psig Drywell Pressure requires a Recirc Core flow runback to be inserted as listed in BFN-ODM-4.20, Strategies for Successful Transient Mitigation.

C **CORRECT**: *(See attached)* In accordance with OPDP-1, Licensed Operators shall take no manual action that will cause an automatic SCRAM. Given the conditions in the question, the Operators must FIRST insert a manual Reactor SCRAM, then trip the Main Turbine, otherwise an automatic SCRAM would occur. For second part, in accordance with BFN-ODM-4.20, when Drywell Pressure reaches 2.2 psig, the reason a manual Reactor SCRAM is inserted is a pre-determined Trigger Value (2.2 psig) has been reached. This conservative action prevents an automatic SCRAM from occurring when Drywell Pressure reaches 2.45 psig. Additionally, it is a simulator training phase expectation for BFN Initial License candidates to use Standard Trigger Values such that a Recirc Core flow runback is inserted at 2.0 psig Drywell Pressure in preparation for inserting a manual Reacto SCRAM at 2.2 psig.

D INCORRECT: First part is correct (S*ee C).* Second part is incorrect but plausible (S*ee B).*

RO Level Justification: Tests the candidate's knowledge of the reasons for a Reactor SCRAM being performed as it relates to High Drywell Pressure. This question is rated as Memory due to the requirement to strictly recall facts.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): BFN-ODM-4.20, Rev. 6 (Attach if not previously provided)
OPDP-1, Rev. 46

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.276 Obj. 7 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Excerpts from OPDP-1:

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0046 Page 32 of 71

3.5.1 Reactivity Management (continued)

5. Ensure that pre-job briefs for work activities address potential reactivity effects. Personnel involved in reactivity manipulations or working on reactivity control equipment shall be properly trained, understanding their roles and responsibilities, and shall be briefed on management expectations.

6. Determine the experience level of personnel involved in reactivity manipulations and ensure appropriate oversight is provided for those with little or no reactor maneuvering experience.

7. Control Room panel manipulations shall be performed only by licensed Operations personnel, or by trainees under the direction and in the presence of a licensed UO or SRO as part of the individual's participation in NRC approved license training program to qualify for an Operator license. [10CFR50.54 (i), 10CFR55.13].

8. For Sequoyah and Watts Barr - for startup dilutions used to establish conditions to conduct a startup (coming off of refueling boron concentrations), direct oversight is not required as long as the following requirements are met:

   a. Reactor will remain subcritical at all times based upon rod position and RCS temperature.

   b. Reactivity Pre-job brief is conducted to include termination criteria for the dilution.

   c. OAC is to remain engaged in the dilution with no other duties besides normal watchstanding duties.

   d. Periodic oversight is required by the SRO.

D. Unit Operators are charged to:

1. Monitor reactor parameters to ensure the unit is operating within prescribed bands and monitor prescribed parameters and instrumentation to verify plant response is as expected during reactivity manipulations. If the unit is determined to be operating above its licensed core thermal power limit take prompt (typically no more than 10 minutes from time of determination) action to reduce power below the core thermal power limit.

2. Take conservative action, including action for manual scram/reactor trip when abnormal reactor conditions are encountered. The operator shall not rely solely on the Reactor Protection System to protect the reactor during reactivity events.

3. Monitor nuclear instrumentation during refueling activities that could affect the reactivity of the core so that abnormal reactivity events can be mitigated.

4. Directly oversee trainees manipulating reactivity related controls, as if the Unit Operator were performing the manipulation personally. The trainees shall be enrolled in an approved licensed training program.

5. Know and monitor the effects of the reactivity change.

3.5.1 Reactivity Management (continued)

6. Understand and compare reactivity management plan and actual plant performance during core maneuvers.

7. Stop and question unexpected situations involving reactivity, criticality, power level, or core anomalies. Meet anomalous indication with conservative action.

8. Perform no actions except those related to the reactivity manipulation, monitoring for expected changes. This is a dedicated action for the UO performing the reactivity change. Peer checks are expected to be performed for all reactivity management evolutions. Reactivity controls are placed in a safe and stable condition before attending to any emergent conditions.

9. For Browns Ferry - The first control rod movements or flow adjustments shall require place keeping techniques. Subsequent BWR control rod movements or recirculation flow adjustments, during the sequence of the evolution, will be at the discretion of the NUSO. The NUSO should use judgement as to when the place keeping rule may be waived, considering the repetitive number of times the rod movements or flow adjustments will be made. Any changes to personnel or positions, or pause in the evolution, will require resequencing using place keeping rules and practices.

3.5.2 Inserting a Manual Scram or Manual Reactor Trip

A. Licensed Operators shall take no manual action that will cause an automatic scram.

   1. For example, if tripping the turbine will cause a reactor trip, and the turbine requires tripping due to high vibrations, operators shall first trip the reactor, then trip the turbine.

B. Operators shall without hesitation insert a manual scram/manual reactor trip whenever any of the following conditions occurs:

   1. Safety of the reactor is in jeopardy.

   2. Operating parameters exceed any of the reactor protection set-points and an automatic shutdown does not occur.

   3. Core thermal hydraulic instability is observed and mitigating actions are ineffective (BWR).

   4. As directed by plant procedures.

   5. When a pre-determined trigger value is reached.

3.5.3 Manual Control of Automatic Systems

A. If an automatic controller, or an automatic action, is confirmed to have malfunctioned, take prompt actions to place that controller in manual or to accomplish the desired function. Examples are as follows:

Excerpt from BFN-ODM-4.20:

BFN Operations Directive Manual	Strategies for Successful Transient Mitigation	BFN-ODM-4.20 Rev. 0006 Page 11 of 25

4.3.3 Stabilizing Plant Parameters

Operator action to stabilize a critical plant parameter, without NUSO direction, is expected provided that no contradicting order or direction has been given. For example, operator action to stabilize reactor pressure below the high pressure scram setpoint (1073 psig) by opening the MSRVs following an inadvertent closure of the MSIVs is appropriate. Similarly, operating the HPCI flow controller to minimize HPCI injection rate and limit the rate of rise of reactor water level is also appropriate and expected.

4.3.4 Trigger Values

Senior Reactor Operators, specifically the NUSO, are expected to establish control bands and trigger values for critical plant parameters and establish contingency actions based upon the rate of change of critical plant parameters.

Example from the NUSO and OATC: *"Attention for an update."* The NUSO confirms hands are raised and continues with: *"Drywell pressure is 2.0 psig and rising due to a leak in the drywell. We will insert a manual scram if drywell pressure reaches 2.2 psig. End of update."* Then the NUSO provides an individual order to the OATC to insert a manual scram if drywell pressure reaches 2.2 psig. Later, if drywell pressure rose to 2.2 psig, the OATC would update the crew and initiate a manual scram.

Excerpt from BFN ILT Class Standardized Expectations: Used in ILT Simulator Training Phase to ensure multiple crews are trained by consistent standards

ILT Class Standardized Expectations

Airborne Release Offsite	Liquid Release Offsite
☐ Minor releases within federally approved limits[1]	☐ Minor releases within federally approved limits[1]
☐Releases above federally approved limits[1]	☐Releases above federally approved limits[1]
☐ Release information not known	☐ Release information not known

- **STANDARD TRIGGER VALUES:**

Core Flow Runback (when time allows):
2.0 psig Drywell Pressure (rising)
Low Condenser Vacuum alarm on Panel 9-7

When a procedure instructs the performer to perform a core flow runback - the performer will depress the core flow runback push-button and wait until core flow is complete. Then insert SCRAM.

If in the judgment of the US a SCRAM needs to be inserted before completion of lowering core flow or completion of runback, then the SCRAM should be inserted.

Rx SCRAM:
2.2 psig Drywell Pressure (rising)
10" Reactor Water Level (lowering)
1" Margin to trip Condenser Vacuum

Remember that these are standards to allow us to be consistent, Yet as the US, if conditions are degrading at a more rapid rate than usual, you are allowed to direct these actions as you deem which would be more conservative.

Examination Outline Cross-reference:	Level	RO	SRO
295005 (APE 5) Main Turbine Generator Trip / 3	Tier #	1	------
AK3.05 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: • Extraction steam/moisture separator isolations	K/A #	295005AK3.05	
	Importance Rating	2.5	------

Proposed Question: **# 17**

Unit 2 is operating at 21% RTP, when the Main Turbine trips.

Which **ONE** of the following completes the statements below?

Following a Main Turbine trip, the Extraction Non-Return Valves close in order to prevent Main Turbine ___**(1)**___. As a result of this transient, the Reactor___**(2)**___ SCRAM.

A. (1) overspeed
(2) will

B. (1) overspeed
(2) will NOT

C. (1) overheating
(2) will

D. (1) overheating
(2) will NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). The second part is incorrect but plausible in that when the Main Turbine trips while online, a Reactor SCRAM is probable based on Main Turbine loading. However, the given Reactor Power level of 21% is within the Turbine Bypass Valve capacity to sufficiently control Reactor Pressure, which will prevent the Reactor SCRAM under the given conditions.

B **CORRECT:** (*See attached*) In accordance with 2-AOI-47-1, Unplanned Turbine Trip Below 26% Reactor Power (Without Reactor Scram) Automatic Actions, the Extraction Non-Return Valves close when the Main Turbine is tripped. This protects the Main Turbine from an overspeed condition due to the isolation of Extraction Steam. For second part, the Turbine Bypass Valves have the capacity to sufficiently control Reactor Pressure to pass the steam flow for 21% Reactor Power, which will prevent a SCRAM at the given Reactor Power Level.

C INCORRECT: First part is incorrect but plausible in that at low steam loads/flows, turbine blade heating occurs which may possibly cause turbine blade expansion and damage; however the Extraction Non-Return valve is closed to prevent an overspeed condition. The second part is incorrect but plausible (*See A*).

D INCORRECT**:** First part is incorrect but plausible *(See C).* Second part is correct (*See B).*

RO Level Justification: Tests the candidate's knowledge of the reason Extraction Steam isolates on a Main Turbine Trip and the effect on Reactor Operation. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 2-AOI-47-1, Rev. 17 (Attach if not previously provided)
2-OI-47, Rev. 186
OPL171.010, Rev.15

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.095 Obj. 2, 7a (As available)
OPL171.010 Obj. 8

Question Source:	Bank #		
	Modified Bank #	ILT 1909 #16	(Note changes or attach parent)
Question History:	New		
	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

Examination Outline Cross-reference:	Level	RO	SRO
295005 (APE 5) Main Turbine Generator Trip / 3	Tier #	1	------
AK3.03 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP:	K/A #	295005AK3.03	
• Feedwater temperature decrease	Importance Rating	2.8	------

Proposed Question: **# 16**

Unit 2 is operating at 25% RTP with a Reactor Shutdown in progress.

Which **ONE** of the following completes the statements below?

Given that the Main Turbine trips, Extraction Non-Return Valves automatically close in order to prevent Main Turbine ___**(1)**___.

Feedwater Temperature will ___**(2)**___ as a result of the Main Turbine trip.

A. **(1)** overspeed
(2) lower

B. **(1)** overspeed
(2) remain the same

C. **(1)** overheating
(2) lower

D. **(1)** overheating
(2) remain the same

Excerpt from 2-OI-47:

BFN Unit 2	Turbine-Generator System	2-OI-47 Rev. 0185 Page 133 of 279

6.9 Control Valve Tightness Test

NOTES

1) This test is performed from Panel 2-9-7 unless specifically stated otherwise.
2) Turbine speed must lower to less than 900 RPM for this test to be considered satisfactorily completed. However valves are designed to bring turbine to a standstill.
3) If this test is unsatisfactorily completed, turbine should be shut down until repairs are completed.
4) This test should be performed once per operating cycle.

CAUTION

DO NOT perform this test unless reactor power is less than 21.3% to ensure the capacity of the Bypass Valves is sufficient to control reactor pressure.

Excerpt from 2-AOI-47-1:

BFN Unit 2	Unplanned Turbine Trip Below 26% Reactor Power (Without Reactor Scram)	2-AOI-47-1 Rev. 0017 Page 5 of 11

3.0 AUTOMATIC ACTIONS

A. Turbine stop valves close.

B. Turbine control valves close.

C. Turbine combined intermediate valves close.

D. Extraction non-return valves close.

E. Turbine bypass valves open to maintain reactor pressure.

F. The generator air circuit breaker opens.

G. Exciter field breaker opens.

H. Generator load set runs back to minimum.

I. Generator voltage regulator transfers to manual.

J. Recorder 2-XR-47-16 transfers from control valve position to turbine speed.

Excerpt from OPL171.010:

OPL171.010 MAIN TURBINE REV 15

Lesson Plan Content

Outline of Instruction	Instructor Notes
4 CIV Valve quarterly testing: During quarterly testing the combined intermediate valves are individually tested. a. The valves test button is pushed and held in on Panel 9-7 in the control room.	NLO-14, ILT-19 Monitor Turbine indications for unexpected conditions during test
b. The intercept valve slowly closes to 5% open, and then a fast acting solenoid actuates to quickly close the valve.	(IV)
c. When the intercept valve is fully closed, then the intermediate stop valve slowly closes in the same manner as the intercept valve.	(ISV)
d. When the intermediate stop valve is fully closed, the operator releases the pushbutton. e. The intermediate stop valve fully opens and then the intercept valve fully opens.	
E. Extraction Non-return Valves 1. Purpose	
To protect the turbine from an over speed condition, this might occur when the turbine is tripped. A subsequent lowering of pressure in the turbine and heaters, due to vacuum in the condenser will cause hot water from the heater to flash to steam. The reverse steam flow back through the extraction steam piping to the Main Turbine could cause blade damage.	Figure-6 ILT-8 LOR-3 NLO-25

Excerpt from OPL171.010:

OPL171.010 MAIN TURBINE REV 15

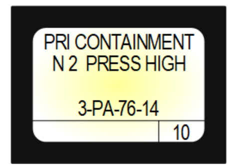
Lesson Plan Content

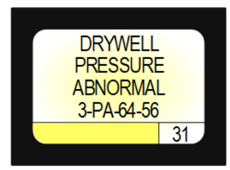
Outline of Instruction	Instructor Notes
(3) Six-flow: Number of exhaust flow paths to the main condenser	
e. Non-reheat: Steam is not reheated before returning to the LP turbines.	
3. Steam Flow Path	ILT-2,NLOR-2 NLO-2, ILT-23, LOR-11
a. Four main steam lines (24 inches)	
b. 8-inch lines to Bypass Valves from mixing header (9 valves ~21% design capacity) 6-inch lines tapping off of the mixing header supply going to:	
(1) Off-gas pre-heaters	
(2) Reactor feedwater pump turbine high-pressure steam supply	
(3) Seal steam regulators	
(4) Steam jet air ejector regulators	
c. Through four main Turbine Stop Valves (TSV)	

Examination Outline Cross-reference:	Level	RO	SRO
295016 (APE 16) Control Room Abandonment / 7	Tier #	1	------
AA2.05 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT:	K/A #	295016AA2.05	
• Drywell pressure	Importance Rating	3.8	------

Proposed Question: **# 18**

Due to a fire on Unit 3, the Unit Operator is performing the Immediate Actions of 3-AOI-100-2, Control Room Abandonment and has reached the step to start the Unit 3 EDGs, when the following occurs:





- PRIMARY CONTAINMENT N2 PRESSURE HIGH (3-9-3B, Window 10) alarms
- DRYWELL PRESSURE ABNORMAL (3-9-5B, Window 31) alarms

Given the conditions above, which **ONE** of the following completes the statements below?

AT THE MOMENT the above alarms are received, the Unit 3 EDGs ___**(1)**___.

Reactor Water Level will be controlled at the Backup Control Panel using ___**(2)**___.

A. (1) must be manually started
(2) High Pressure Coolant Injection (HPCI)

B. (1) must be manually started
(2) Reactor Core Isolation Cooling (RCIC)

C. (1) will receive an automatic start signal
(2) High Pressure Coolant Injection (HPCI)

D. (1) will receive an automatic start signal
(2) Reactor Core Isolation Cooling (RCIC)

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See* B). The second part is incorrect but plausible in that Control Room abandonment is an infrequently performed evolution, and the controls on the Backup Control Panel are often mixed up. HPCI automatically starts on Drywell Pressure of 2.45 psig and a low Reactor Water Level of (-) 45 inches, so this system will quite possibly be running when the operators reach the Backup Control Panel. Additionally, HPCI and RCIC are often mistaken for each other due to the system similarity and there are actions to disable HPCI in the subsequent actions of 3-AOI-100-2, Control Room Abandonment.

B **CORRECT**: (*See attached*) First part is correct in that the EDGs will automatically start when Drywell Pressure reaches 2.45 psig. There are several alarms for Drywell Pressure, but the listed alarms have set points below 2.45 psig; therefore the EDGs will have to be manually started in accordance with 3-AOI-100-2. The second part is correct in that in accordance with 3-AOI-100-2, RCIC is used for Reactor Water Level Control at the Backup Control Panel.

C INCORRECT: First part is incorrect but plausible in that given the Drywell Pressure Alarms it is reasonable to assume that the EDGs may have started on the Drywell Pressure signal of 2.45 psig instead of manually starting them in accordance with 3-AOI-100-2. The second part is incorrect but plausible (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is correct (*See B*).

RO Level Justification: Tests the candidate’s knowledge of the effect of Drywell Pressure on the Control Room Abandonment procedure. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site’s procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 3-AOI-100-2, Rev.26 (Attach if not previously provided)
3-ARP-9-3B, Rev.23
3-ARP-9-5B, Rev.32

Proposed references to be provided to applicants during examination: **PRIMARY CONTAINMENT N2 PRESSURE HIGH (3-9-3B, Window 10) DRYWELL PRESSURE ABNORMAL (3-9-5B, Window 31)**

Learning Objective: OPL171.016 Obj. 11 (As available)
OPL171.208 Obj. 8

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Excerpt from 3-AOI-100-2:

BFN Unit 3	Control Room Abandonment	3-AOI-100-2 Rev. 0026 Page 6 of 92

4.0 OPERATOR ACTIONS

4.1 Immediate Action

NOTES

1) The immediate action to "DEPRESS REACTOR SCRAM A and B pushbuttons" is required to be completed prior to evacuating the control room.

2) Steps should be performed in order, however, Steps 4.1[7], 4.1[10], 4.1[11], and 4.1[12] may be performed at anytime while performing the immediate actions.

[1] **IF** core flow is above 60%, **THEN:** (Otherwise N/A)

LOWER core flow to between 50-60%. ☐

[2] **DEPRESS** REACTOR SCRAM A and B pushbuttons. ☐

[3] **PLACE** REACTOR MODE SWITCH in SHUTDOWN. ☐

NOTE

If rods fail to insert or scram solenoids fail to deenergize in Steps 4.1[4] and 4.1[5], then Step 4.2[1] will pull RPS Scram Solenoid Fuses.

[4] **CHECK** ALL control rods fully inserted. ☐

[5] **CHECK** all eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights extinguished. ☐

[6] **TRIP** Reactor Recirc Pumps. ☐

[7] **ISOLATE** RWCU. ☐

[8] **ENSURE** Main Turbine tripped. ☐

[9] **TRIP** Reactor Feed Pumps as necessary to prevent tripping on high water level. ☐

[10] **START** Emergency Diesel Generators. ☐

[11] **ENSURE** each EECW header has at least one pump in service. ☐

Excerpt from 3-9-ARP-3B:

BFN Unit 3	Panel 9-3 3-XA-55-3B	3-ARP-9-3B Rev. 0023 Page 13 of 38

PRI CONTAINMENT N2 PRESS HIGH 3-PA-76-14 10	Sensor/Trip Point:	
	PT-76-14	1.5 psig (alarm from recorder).

(Page 1 of 1)

Sensor Location:	Panel 25-6 Rx Bldg, El 593, R-19 Q-LINE
Probable Cause:	A. Drywell Cooler(s) failure. B. Steam or water leak inside Drywell. C. Loss of RBCCW to Drywell Coolers. D. Low pressure front moving through area.
Automatic Action:	None
Operator Action:	A. **CHECK** containment pressure using multiple indications. ☐ B. **CHECK** containment temperature. ☐ C. **REFER TO** 3-OI-64 Venting the Drywell with Standby Gas Treatment Fan. ☐
References:	3-45N620-3 47W600-57 3-47E610-76-1 GE 730E933-1

Excerpt from 3-9-ARP-5B:

BFN Unit 3	Panel 9-5 3-XA-55-5B	3-ARP-9-5B Rev. 0032 Page 37 of 44

DRYWELL PRESSURE ABNORMAL 3-PA-64-56 31	Sensor/Trip Point:	
	3-PS-64-56E	1.65 psig rising
	3-PS-64-56F	0.1 psig lowering

(Page 1 of 1)

Sensor Location: Panel 25-5B
Elevation 593

Probable Cause:
A. Drywell DP air compressor failure.
B. Loss of RBCCW.
C. Breach of Primary Containment.
   1. Drywell vent valves open or leaking.
   2. Drywell vacuum breaker open or leaking.

D. LOCA.
E. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **CHECK** alarm using multiple indications.

B. **IF** RBCCW has been lost, **THEN** **REFER TO** 3-AOI-70-1.

C. **REFER TO** 3-AOI-64-1.

References:

3-45E620-6	3-47E610-64-1	3-730E915-17
3-AOI-70-1	3-AOI-64-1	

Excerpt from 0-OI-82:

BFN Unit 0	Standby Diesel Generator System	0-OI-82 Rev. 0169 Page 14 of 219

3.0 PRECAUTIONS AND LIMITATIONS (continued)

P. Personnel working in the D/G rooms should remain aware that the possibility exists of CO_2 discharge into the room. Upon CO_2 initiation, an alarm will sound. Personnel then have 20 seconds to evacuate the area before CO_2 is dispensed. For detection purposes, a wintergreen odor is injected into CO_2 discharge.

Q. Environmental calculations assume DG battery ambient temperatures are within 40°F to 110°F.

R. When the D/G is the only feed to the shutdown board and in single unit operations, starting an RHR Pump with other 4kV motor loads running on the associated board may result in D/G overload.

S. After operation of 4160V breakers, the charging spring is required to be verified to have recharged by verifying locally the breaker closing spring target indicates charged and the amber breaker spring charged light is on to ensure future breaker operation.

T. Diesel Generators will automatically start, as follows:

1. Degraded voltage or undervoltage on 4-kV Shutdown Board A, B, C, or D will start its associated Diesel Generator.
2. A Pre-Accident Signal (Reactor Vessel Low Low Low water level OR High Drywell pressure) on Unit 1, Unit 2 or Unit 3 will start all eight Diesel Generators.

U. Under normal conditions, any of the following will auto trip the Diesel Generator output breaker:

1. Differential overcurrent
2. Timed overcurrent
3. Reverse power
4. Loss of field
5. Overspeed
6. Common Accident Signal (Low Low Low Reactor water level OR Low Reactor pressure in conjunction with High Drywell pressure on Unit 1, 2 or Unit 3.)

Excerpts from 3-AOI-100-2:

BFN Unit 3	Control Room Abandonment	3-AOI-100-2 Rev. 0026 Page 12 of 92

4.2 Unit 3 Subsequent Actions (continued)

[9] **INITIATE** RCIC as follows:

[9.1] At Panel 3-25-32, **CHECK OPEN** 3-FCV-71-9 (Red Light above switch) RCIC TURB TRIP/THROT VALVE RESET, 3-HS-71-9D. ☐

[9.2] At 250V DC RMOV Bd 3B, Compt. 5D, **PLACE** 3-HS-071-0034C, RCIC PUMP MIN FLOW VALVE EMER HAND SWITCH, in OPEN. (Unit 3 Turbine Building AUO) ☐

[9.3] At 250V DC RMOV Bd 3C, Compt. 4B, **PLACE** 3-HS-071-0008C, RCIC TURB STM SUPPLY VALVE EMER HAND SWITCH, in OPEN. (Unit 3 Reactor Building AUO) ☐

NOTE
RCIC Turbine should start and flow should stabilize at 620 gpm.

[9.4] At Panel 3-25-32, **CHECK** turbine speed 2100 rpm or above using RCIC TURBINE SPEED, 3-SI-71-42B. ☐

[9.5] At 250V DC RMOV Bd 3B, Compt. 5D, **PLACE** RCIC PUMP MIN FLOW VALVE EMER HAND SWITCH, 3-HS-071-0034C, in CLOSE. (Unit 3 Turbine Building AUO) ☐

[9.6] At Panel 3-25-32, **ADJUST** flowrate as necessary using RCIC SYSTEM FLOW/CONTROL, 3-FIC-71-36B . ☐

BFN Unit 3	Control Room Abandonment	3-AOI-100-2 Rev. 0026 Page 13 of 92

4.2 Unit 3 Subsequent Actions (continued)

NOTE
The following step prevents HPCI operation and automatic opening of HPCI MAIN PUMP MINIMUM FLOW VALVE, 3-FCV-73-30.

[10] At 250V DC RMOV Bd 3A, **PERFORM** the following:

[10.1] Compt. 3D, **ENSURE CLOSED** HPCI STM SPLY VLV TO TURB FCV-73-16 (MO 23-14). ☐

[10.2] Compt. 3D, **PLACE** HPCI TURBINE STEAM SPLY VLV TRANS, 3-XS-073-0016, in EMERG. ☐

[10.3] **IF** desired to verify HPCI MAIN PUMP MINIMUM FLOW VLV, 3-FCV-073-0030, closed prior to opening breaker, **THEN** (Otherwise N/A)

DIRECT operator to verify locally. ☐

[10.4] Compt. 8D, **PLACE** 3-BKR-073-0030 HPCI MAIN PUMP MIN FLOW VLV FCV-73-30, in OFF. ☐

[11] **ESTABLISH** plant cooldown as follows:

Examination Outline Cross-reference:	Level	RO	SRO
295018 (APE 18) Partial or Complete Loss of CCW / 8	Tier #	1	------
AK1.01 (10CFR 55.41.10)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:	K/A #	295018AK1.01	
• Effects on component/system operations	Importance Rating	3.5	------

Proposed Question: **# 19**

All three Units are operating at 100% RTP when the following conditions occur:

- Emergency Equipment Cooling Water (EECW) Pump A3 is operating and subsequently trips
- EECW Pump C3 fails to start
- **NO** Operator actions have been taken

Given the conditions above, which **ONE** of the following completes the statements below in accordance with 0-OI-67, Emergency Equipment Cooling Water System?

0-FCV-67-53, EECW SYSTEM BACKUP WATER SUPPLY VALVE to the Control Air Compressors will automatically open when Raw Cooling Water (RCW) Header Pressure lowers below the setpoint of ___**(1)**___.

EECW Pumps A3 and C3 supply the ___**(2)**___ header of the EECW System.

A. (1) 15 psig
(2) North

B. (1) 15 psig
(2) South

C. (1) 30 psig
(2) North

D. (1) 30 psig
(2) South

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT**:** The first part is incorrect but plausible in that EECW Pumps A3, B3, C3, D3 will auto-start when Low Raw Cooling Water (RCW) header pressure setpoint lowers below 15 psig for RBCCW heat exchangers. The second part is correct (*See C*).

B INCORRECT: The first part is incorrect but plausible (*See A*). The second part is incorrect but plausible in that the EECW System normally provides cooling for numerous loads through a North (A3/C3 pumps with A1/C1 serving as backup pumps) header and a South header (B3/D3 pumps with B1/D1 serving as backup pumps). Additionally, specific EECW System backup cooling valves only open if EECW header pressure is adequate. Given this, candidates could easily confuse specific EECW pumps/valves, pressures, normal and/or backup supported loads.

C **CORRECT:** *(See attached)* In accordance with 0-OI-67, Emergency Equipment, RHRSW Pumps A3, B3, C3, D3 are assigned as EECW Pumps. They will auto-start when Raw Cooling Water (RCW) header pressure lowers below the setpoint of 30 psig for Control Air Compressors. For second part, EECW System FCV-67-53, EECW NORTH HEADER SUPPLY VALVE TO AIR COMP is the backup water supply valve to Control Air Compressors which will auto OPEN (if EECW pressure is greater than or equal to 106 psig) when RCW Header Pressure lowers to each respective setpoint only. Since the EECW Pumps A3/C3 supply the North header (not available) and only one of the B3/D3 South header supply pumps are normally running, the North header will NOT have adequate pressure for 0-FCV-67-53 to automatically OPEN to provide EECW backup water supply to the Control Air Compressors.

D INCORRECT: The first part is correct (*See C*). The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the operational impact from a partial or complete loss of Raw Cooling Water and Emergency Equipment Cooling Water System component/operations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 0-OI-67, Rev. 121 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OPL171.051, Obj. 7a, 7b (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.046-07 003 #1596	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1596. OPL171.046-07 003

Given the following conditions:

- All three units are operating at 100% power
- The operating A3 EECW Pump trips
- The C3 EECW pump fails to start manually

Which ONE of the following describes the impact of the A3 and C3 EECW Pump failures? Assume NO additional operator actions.

The backup cooling water supply to the ______ is NOT available.

A. control air compressors ONLY

B. Unit 1 and 2 RBCCW heat exchangers ONLY

C.✓ control air compressors and the Unit 3 RBCCW heat exchanger ONLY

D. control air compressors and all three units' RBCCW heat exchangers

C CORRECT

Excerpts from 0-OI-67:

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0121 Page 9 of 105

3.0 PRECAUTIONS AND LIMITATIONS

A. RHRSW Pumps A3, B3, C3, and D3 are assigned to the EECW System and are referred to in this procedure as RHRSW pumps.

B. RHRSW Pumps A1, B1, C1, and D1 may supply either the RHRSW or EECW Systems. 0-OI-23 should be referred to when using A1, B1, C1, or D1 RHRSW pumps for RHRSW operation.

C. The EECW System is aligned as follows:

1. At least one RHRSW pump, assigned to the EECW System, should be running on each header to maintain the header charged at all times. If no pumps are running on a header and header pressure lowers to ≤ 0 psig, the header shall be declared inoperable and appropriate actions taken, as required by Technical Specifications.

2. Two additional RHRSW pumps, one on each of the north and south headers, are normally lined up to start automatically for EECW system operation, if NOT already running.

3. Only one RHRSW pump in a given RHRSW pump room may be counted toward meeting Technical Specification 3.7.2 requirements for EECW pump operability.

D. If a Number 1 RHRSW Pump is needed to meet minimum Technical Specification operable EECW pump requirements, that pump may be aligned to EECW. To meet EECW requirements, Number 1 RHRSW pumps must be aligned to EECW, the pump started, and should remain running. Number 1 RHRSW pumps do NOT have the same auto start signals as the associated Number 3 RHRSW Pump. When a Number 1 RHRSW Pump is aligned for EECW, its RHRSW function required by the Safe Shutdown Program, NFPA 805 is inoperable. NFPA 805 FPR or FPRM requirement after implementation shall be addressed. **REFER TO** Sections 8.1 through 8.4.

E. RHRSW Pumps A3, B3, C3, and D3 as well as A1, B1, C1, and D1, when lined up for EECW operation, will auto-start when either:

1. Any unit Common Accident Signal Relay is energized. (High Drywell Pressure in conjunction with low reactor pressure, or Low-Low-Low Reactor Water Level.)

2. Low Raw Cooling Water header pressure at control air compressor (less than 30 psig).

3. Low Raw Cooling Water pressure at RBCCW heat exchanger (less than 15 psig).

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0121 Page 11 of 105

3.0 PRECAUTIONS AND LIMITATIONS (continued)

J. DCN 70834-03: removes limit switch 0-ZS-67-49 and spare associated cable between 4KV SD BD B and IPS.

1. The start logic for RHRSW pump C1 has been modified by replacing the limit switch function associated with 0-SHV-067-0049 with a mode switch located on 4KV SD BD B Compt. 10. Placing RHRSW PUMP C1 EECW MODE SWITCH, 0-HS-067-0049 to the "EECW" position will align the C1 RHRSW pump auto start logic circuit.

2. When the RHRSW PUMP C1 EECW MODE SWITCH, 0-HS-067-0049 is in the "EECW" position and C2 RHRSW pump is supplying the RHR HX, the HX outlet valve will not automatically close if the C2 RHRSW trips which will cause a low flow and low pressure condition.

K. DCN 70834-04: removes limit switch 0-ZS-67-48 and spare associated cable between 4KV SD BD 3ED and IPS.

1. The start logic for RHRSW pump D1 has been modified by replacing the limit switch function associated with 0-SHV-067-0048 with a mode switch located on 4KV SD BD 3ED Compt. 6. Placing RHRSW PUMP D1 EECW MODE SWITCH, 0-HS-067-0048 to the "EECW" position will align the D1 RHRSW pump auto start logic circuit.

2. When the RHRSW PUMP D1 EECW MODE SWITCH, 0-HS-067-0048 is in the "EECW" position and D2 RHRSW pump is supplying the RHR HX, the HX outlet valve will not automatically close if the D2 RHRSW trips which will cause a low flow and low pressure condition.

L. [NRC/C] The unavailability of an entire EECW header (north or south) should be limited until the modifications described in Attachment S, Table S-2 of the to TVA letter to NRC, License Amendment Request to Adopt NFPA 805, Performance-Based Standard for Light Water Reactor Electric Generating Plants (2001 Edition) (Tech Spec Change TS-480), dated March 27, 2013. These are the interim actions applied during the transition phase from Appendix R Fire Protection to NFPA 805.[NRC 114569379]

M. EECW System backup water supply valve (FCV-67-53) to the control air compressors will auto open at 30 psig lowering RCW pressure, if EECW pressure is ≥ 106 psig. The valve will auto close on EECW pressure dropping to < 106 psig.

Supports Distractors A(1), B(1):

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0121 Page 12 of 105

3.0 PRECAUTIONS AND LIMITATIONS (continued)

N. EECW System backup water supply valves (FCV-67-50 [North header] and 51 [South header]) to the RBCCW heat exchangers will open at 15 psig lowering RCW pressure if EECW pressure is equal to or greater than the setpoint. These valves will close on EECW pressure dropping below the setpoint. Once closed, the closure seals in until manually reset in accordance with Section 8.7. The north header supply to Unit 1 RBCCW, the north header supply to Unit 2 RBCCW and the South header supply to Unit 3 RBCCW are normally isolated with a manual valve; therefore no flow will occur when either 1-FCV-67-50, 2-FCV-67-50 or 3-FCV-67-51 opens. The EECW pressure setpoints for these valves are listed below in psig:

	Unit 1	Unit 2	Unit 3
FCV-67-50	90	91	92
FCV-67-51	107	109	113

O. The EECW discharge strainer automatically starts its cleaning cycle on pump discharge flow, and the flush valve opens automatically.

P. The shutdown boards will load shed their respective EECW loads when a shutdown board undervoltage condition exists or a LOCA signal in conjunction with a loss of offsite power is received.

RHRSW (EECW) PUMP	SHUTDOWN BOARD
A1	A
A3	3EA
B1	3EC
B3	C
C1	B
C3	3EB
D1	3ED
D3	D

Q. Because the EECW system is common to all three units, the Unit Operators should contact each other whenever changes to the system are made.

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0121 Page 22 of 105

5.2 **Startup of the North EECW Header (continued)**

[7] **START** A3 or C3 RHRSW pump using one of the following:

- RHRSW PUMP A3(C3) EECW NORTH HDR, 0-HS-23-85A/1(91A/1) on Unit 1
- RHRSW PUMP A3(C3) EECW NORTH HDR, 0-HS-23-85A/2(91A/2) on Unit 2
- RHRSW PUMP A3(C3) EECW NORTH HDR, 0-HS-23-85A/3(91A/3) on Unit 3

CAUTION

Minimum RHRSW/EECW pump flow is 1700 gpm for A3 and C3 pumps. These pumps should NOT be operated below minimum flow requirements.

[8] **CHECK** pump flow greater than 1700 gpm.

[9] IF NOT, **THEN**

THROTTLE OPEN valve utilized in Step 5.2[5.2] or 5.2[6.2] until flow is greater than 1700 gpm. (Otherwise N/A)

[10] WHEN approximately 2 minutes has elapsed, THEN

START the second RHRSW pump using one of the following:

- RHRSW PUMP A3(C3) EECW NORTH HDR, 0-HS-23-85A/1(91A/1) on Unit 1
- RHRSW PUMP A3(C3) EECW NORTH HDR, 0-HS-23-85A/2(91A/2) on Unit 2
- RHRSW PUMP A3(C3) EECW NORTH HDR, 0-HS-23-85A/3(91A/3) on Unit 3

CAUTION

Minimum RHRSW/EECW pump flow is 1700 gpm for A3 and C3 pumps. These pumps should NOT be operated below minimum flow requirements.

[11] **CHECK** pump flow greater than 1700 gpm.

Supports Distractors B(2), D(2):

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0121 Page 27 of 105

5.3 **Startup of the South EECW Header (continued)**

[7] **START** B3 or D3 RHRSW pump using one of the following:

- RHRSW PUMP B3(D3) EECW SOUTH HDR, 0-HS-23-88A/1(94A/1) on Unit 1
- RHRSW PUMP B3(D3) EECW SOUTH HDR, 0-HS-23-88A/2(94A/2) on Unit 2
- RHRSW PUMP B3(D3) EECW SOUTH HDR, 0-HS-23-88A/3(94A/3) on Unit 3

CAUTION

Minimum RHRSW/EECW pump flow is 1700 gpm for B3 and D3 pumps.
These pumps should NOT be operated below minimum flow requirements.

[8] **CHECK** pump flow greater than 1700 gpm.

[9] **IF** NOT, **THEN**

THROTTLE OPEN valve utilized in Step 5.3[5.2] or 5.3[6.2] until flow is greater than 1700 gpm. (Otherwise N/A)

[10] **WHEN** approximately 2 minutes has elapsed, **THEN**

START the second RHRSW pump using one of the following:

- RHRSW PUMP B3(D3) EECW SOUTH HDR, 0-HS-23-88A/1(94A/1) on Unit 1
- RHRSW PUMP B3(D3) EECW SOUTH HDR, 0-HS-23-88A/2(94A/2) on Unit 2
- RHRSW PUMP B3(D3) EECW SOUTH HDR, 0-HS-23-88A/3(94A/3) on Unit 3

CAUTION

Minimum RHRSW/EECW pump flow is 1700 gpm for B3 and D3 pumps.
These pumps should NOT be operated below minimum flow requirements.

[11] **CHECK** pump flow greater than 1700 gpm.

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0121 Page 56 of 105

8.7 EECW to RCW Crossties for Control Air & RBCCW

NOTES

1) When any unit's HS-67-53A (EECW N HDR SUPPLY VLV TO AIR COMP) 1(2)(3)-LPNL-925-0032 is in the "OPEN" position and EECW header pressure greater than setpoint, the 0-FCV-67-53 will OPEN.

2) When all unit's HS-67-53A (EECW N HDR SUPPLY VLV TO AIR COMP) 1(2)(3)-LPNL-925-0032 are in "AUTO" and EECW header pressure greater than setpoint (106 psig) and RCW pressure lowers below setpoint (30 psig), then 0-FCV-67-53 will OPEN.

3) When 1(2) (3)-HS-67-50A (51A) (EECW NORTH (SOUTH) HDR SPLY TO RBCCW HTXS) 1(2) (3)-LPNL-925-0032, is in the "OPEN" position and EECW header pressure greater than setpoint, then 1(2) (3)-FCV-67-50 (51) will OPEN.

4) When 1(2)(3)-HS-67-50A(51A) (EECW NORTH(SOUTH) HDR SPLY TO RBCCW HTXS) 1(2)(3)-LPNL-925-0032 is in "AUTO" and EECW header pressure greater than setpoint and RCW pressure lowers below setpoint, then 1(2)(3)-FCV-67-50(51) will OPEN. The North header supply to Unit 1 RBCCW, the North header supply to Unit 2 RBCCW and the South header supply to Unit 3 RBCCW are normally isolated with a manual valve; therefore no flow will occur when either 1-FCV-67-50, 2-FCV-67-50 or 3-FCV-67-51 opens.

	Unit 1	Unit 2	Unit 3
FCV-67-50	90	91	92
FCV-67-51	107	109	113

[1] **IF** EECW header drops to below setpoint and subsequently is restored above the setpoint, **THEN**

OPEN 0-FCV-67-53 and 1(2)(3)-FCV-67-50 and 1(2)(3)-FCV-67-51 by performing the following:

[1.1] **RESET** 0-FCV-67-53 using 1(2) (3)-HS-67-53B, 1(2)(3)-LPNL-925-0032, EECW NORTH HDR SPLY VLV TO AIR COMP RESET CKT and **CHECK** BLUE Light extinguished (any unit's RESET switch will reset all three Panel 25-32 lights).

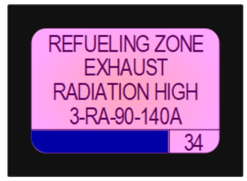
[1.2] **RESET** 1(2) (3)-FCV-67-50 (51) using 1(2) (3)-HS-67-50B (51B) EECW NORTH (SOUTH) HDR SPLY TO RBCCW HTX RESET, 1(2)(3)-LPNL-925-0032 and **CHECK** BLUE Light extinguished.

Examination Outline Cross-reference:	Level	RO	SRO
295023 (APE 23) Refueling Accidents / 8	Tier #	1	------
AK2.03 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between REFUELING ACCIDENTS and the following: • Radiation monitoring equipment	K/A #	295023AK2.03	
	Importance Rating	3.4	------

Proposed Question: **# 20**

Unit 3 is in MODE 5 when a refueling accident results in the following:

- REFUELING ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 34) alarms



Given the conditions above, which **ONE** of the following completes the statement below that describes the Unit 3 Ventilation System response?

___**(1)**___ Zone Ventilation System(s) isolates and ___**(2)**___ Zone dampers align to the Standby Gas Treatment System.

A. (1) **ONLY** the Refueling
(2) **ONLY** the Refueling

B. (1) **ONLY** the Refueling
(2) **BOTH** the Refueling **AND** Reactor

C. (1) The Refueling **AND** Reactor
(2) **ONLY** the Refueling

D. (1) The Refueling **AND** Reactor
(2) **BOTH** the Refueling **AND** Reactor

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See Attached*) In accordance with the given 3-ARP-9-3A, Window 34, automatic actions that occur as a result of this alarm are that Control Room and Refuel Zone ventilation isolates and Standby Gas Treatment (SGT) initiates. For second part, this alarm results in a Group 6 isolation and in accordance with 3-AOI-64-2D, Group 6 Ventilation System Isolation, Refuel Zone Ventilation will isolate and align to the SGT.

B INCORRECT: The first part is correct (*See* A). The second part is incorrect but plausible in that the Refueling Zone isolates, but SGT does not align to the Reactor Zone for a Refuel Radiation Monitor alarm only. SGT aligning to the Reactor Zone is a normal system response for Reactor Zone high radiation, but not when only Refuel Floor radiation is high.

C INCORRECT: The first part is incorrect but plausible in that that a Refuel Zone isolation will occur, but a Reactor Zone isolation will not. A Reactor Zone isolation is a normal system response for Reactor Zone high radiation, but both zones do not isolate when only Refuel Floor radiation is high. The second part is correct (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of how the Reactor and Refuel Ventilation Systems respond to a high radiation level on the Refuel Floor. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (4) Assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems and/or safety functions.

Technical Reference(s): 3-ARP-9-3A, Rev. 57 (Attach if not previously provided)

3-AOI-64-2D, Rev.19

Proposed references to be provided to applicants during examination: **REFUELING ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 34)**

Learning Objective: OPL171.067 Obj. 3 (As available)
OPL171.018 Obj. 4.

Question Source:	Bank #	1205 #10	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2012	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Copy of Bank Question:

QUESTION 10

Unit 3 is in Mode 5 when a refueling accident results in the following annunciator:

- REFUELING ZONE EXHAUST RADIATION HIGH (3-9-3A, window 34)

Which ONE of the following completes the statement below that describes the expected plant response?

A. ONLY the Refueling Zone Ventilation System isolates and ONLY the Refueling Zone dampers re-align to SGTS.

B. ONLY the Refueling Zone Ventilation System isolates and the Reactor AND Refueling Zones dampers re-align to SGTS.

C. The Refueling AND Reactor Zone Ventilation Systems isolate and ONLY the Refueling Zone dampers re-align to SGTS.

D. The Refueling AND Reactor Zone Ventilation Systems isolate and the Reactor AND Refueling Zones dampers re-align to SGTS.

Correct Answer: A

Excerpt from 3-ARP-9-3A:

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0057 Page 56 of 59

REFUELING ZONE EXHAUST RADIATION HIGH 3-RA-90-140A

34

(Page 1 of 2)

Sensor/Trip Point:

3-RE-90-140A	72 MR/HR	
3-RE-90-140B	72 MR/HR	
3-RE-90-141A	72 MR/HR	Required setting of ≤ 100 MR/HR.
3-RE-90-141B	72 MR/HR	

Sensor Location: Rx Bldg, El 664' (Refuel Floor), R-17 P-LINE

Probable Cause:
A. Radiation levels have risen above alarm setpoint.
B. Refueling accident.
C. Dry Cask loading/unloading activities in progress.
D. Loss of power to NUMAC drawer.

Automatic Action:
A. Control Room and Refuel Zone ventilation isolates.
B. SGTS initiates.
C. Control Room Emergency Pressurization units start.

Operator Action:
A. **CHECK** alarm condition on the following:
   1. REACTOR & REFUEL ZONE EXHAUST RADIATION recorder, 3-RR-90-144 points 3 and 4 on Panel 3-9-2.
   2. RX & REFUEL ZONE EXH CH A RAD MON RTMR, 3-RM-90-140/142 on Panel 3-9-10.
   3. RX & REFUEL ZONE EXH CH B RAD MON RTMR, 3-RM-90-141/143 on Panel 3-9-10.

B. **IF** Dry Cask loading/unloading activities are in progress, **THEN NOTIFY** the Cask Supervisor to place the MPC in a safe condition per one of the following methods:
   1. MSI-0-079-DCS400.1
   2. MSI-0-079-DCS400.1FW
   3. As directed by Radiation Protection.

C. **NOTIFY** Shift Manager, Unit 1 and Unit 2.

Excerpts from 3-AOI-64-2D:

BFN Unit 3	Group 6 Ventilation System Isolation	3-AOI-64-2D Rev. 0019 Page 4 of 17

1.0 PURPOSE

This procedure provides symptoms, automatic actions and operator actions for a Group 6 Ventilation System Isolation.

2.0 SYMPTOMS

NOTES

1) PCIS Group 6 Isolation is initiated by any one of the following signals:
   - Reactor vessel water level (LEVEL 3)
   - Drywell pressure at 2.45 psig
   - Reactor zone exhaust radiation at 72 mr/hr
   - Refuel zone exhaust radiation at 72 mr/hr
2) High Refuel Zone exhaust radiation causes only the automatic actions listed in Section 3.1.
3) Refuel Zone isolation due to Group 6 isolation initiated on Unit 1 or Unit 2.

A. Any one or more of the following annunciators in ALARM:

1. REACTOR ZONE EXHAUST RADIATION HIGH (3-XA-55-3A, Window 21)
2. REFUELING ZONE EXHAUST RADIATION MONITOR DOWNSCALE (3-XA-55-3A, Window 28)
3. REFUELING ZONE EXHAUST RADIATION HIGH (3-XA-55-3A, Window 34)
4. RX ZONE EXH RADIATION MONITOR DNSC (3-XA-55-3A, Window 35)
5. RX BLDG VENTILATION ABNORMAL (3-XA-55-3D, Window 3)
6. RX VESSEL WTR LEVEL LOW HALF SCRAM (3-XA-55-4A, Window 2)
7. DRYWELL PRESSURE HIGH HALF SCRAM (3-XA-55-4A, Window 8)
8. REACTOR ZONE DIFFERENTIAL PRESSURE LOW (3-XA-55-3D, Window 32)

BFN Unit 3	Group 6 Ventilation System Isolation	3-AOI-64-2D Rev. 0019 Page 6 of 17

3.0 AUTOMATIC ACTIONS

3.1 Refueling Zone Isolation

A. The following equipment TRIP and ISOLATE:

1. Refuel Zone Supply/Exhaust Fans/Dampers:

    a. 3-FCO-064-0003A, REFUEL ZONE EXH FAN 3A DMPR

    b. 3-FCO-064-0003B, REFUEL ZONE SPLY FAN 3A DMPR

    c. 3-FCO-064-0004A, REFUEL ZONE EXH FAN 3B DMPR

    d. 3-FCO-064-0004B, REFUEL ZONE SPLY FAN 3B DMPR

    e. 3-FCO-064-0005, REFUEL ZONE SPLY OUTBD ISOL DMPR

    f. 3-FCO-064-0006, REFUEL ZONE SPLY-INBD ISOL DMPR

    g. 3-FCO-064-0009, REFUEL ZONE EXH OUTBD ISOL DMPR

    h. 3-FCO-064-0010, REFUEL ZONE EXH INBD ISOL DMPR

2. Drywell DP Compressor

3. Primary Containment H_2/O_2 Analyzer

4. 3-RM-90-256, Drywell Radiation Monitor (3-MON-90-50, AIR PARTICULATE MONITOR CONSOLE)

B. The following valves CLOSE:

1. 3-FCV-076-0017, PRI CTMT N_2 MAKEUP OUTBD ISOLATION VLV

2. 3-FCV-076-0018, DRYWELL N_2 MAKEUP INBD ISOLATION VLV

3. 3-FCV-076-0019, SUPPR CHBR ATM SPLY INBD ISOLATION VLV

4. 3-FCV-076-0024, PRI CTMT N_2 PURGE OUTBD ISOLATION VLV

5. 3-FCV-064-0017, DW/SUPPR CHBR AIR PURGE ISOL VLV

6. 3-FCV-064-0018, DRYWELL ATM SUPPLY INBD ISOLATION VLV

7. 3-FCV-064-0030, DRYWELL VENT OUTBD ISOLATION VLV

8. 3-FCV-064-0031, DRYWELL INBD ISOLATION VLV

9. 3-FCV-064-0032, SUPPR CHBR VENT INBD ISOL VLV

BFN Unit 3	Group 6 Ventilation System Isolation	3-AOI-64-2D Rev. 0019 Page 7 of 17

3.1 Refueling Zone Isolation (continued)

10. 3-FCV-064-0033, SUPPR CHBR VENT OUTBD ISOL VLV
11. 3-FCV-064-0034, SUPPR CHBR INBD ISOLATION VLV
12. 3-FCV-084-0020, PRI CTMT VENT TO SGT ISOL VALVE
13. 3-FSV-043-0050, RHR SAMPLE INLET INBD ISOL VALVE
14. 3-FSV-043-0056, RHR SAMPLE INLET OUTBD ISOL VALVE
15. 3-FSV-043-0040, PASS SAMPLE RETURN INBD ISOL VALVE
16. 3-FSV-043-0042, PASS SAMPLE RETURN OUTBD ISOL VALVE
17. 3-FCV-064-0019, SUPPR CHAMBER ATM SUPPLY INBD ISOL VLV
18. 3-FCV-064-0029, DRYWELL EXHAUST INBD ISOL VLV
19. 3-FCV-064-0140, DRYWELL DP CPRSR DISCH VLV
20. 3-FCV-064-0139, DRYWELL DP CPRSR SUCT VLV

C. Standby Gas Treatment System starts

D. 1-FCO-64-44, REFUEL ZONE EXH TO SGT CROSSTIE DMPR, OPENS

E. 3-FCO-64-44, REFUEL ZONE EXH TO SGT CROSSTIE DMPR, OPENS

F. 1-FCO-64-45, REFUEL ZONE EXH TO SGT CROSSTIE DMPR, OPENS

G. CREV Units start

BFN Unit 3	Group 6 Ventilation System Isolation	3-AOI-64-2D Rev. 0019 Page 8 of 17

3.2 Reactor Zone Isolation

A. Refuel Zone Isolation Actions occur as listed in Section 3.1.

B. Reactor Zone Supply and Exhaust fans trip and isolate:

1. 3-FCO-064-0011A, REACTOR ZONE EXH FAN 3A AND DMPR
2. 3-FCO-064-0011B, REACTOR ZONE SPLY FAN 3A DMPR
3. 3-FCO-064-0012A, REACTOR ZONE EXH FAN 3B DMPR
4. 3-FCO-064-0012B, REACTOR ZONE SUP FAN 3B DMPR
5. 3-FCO-064-0013, REACTOR ZONE SPLY OUTBD ISOL DMPR
6. 3-FCO-064-0014, REACTOR ZONE SPLY INBD ISOL DMPR
7. 3-FCO-064-0042, REACTOR ZONE EXH INBD ISOL DMPR
8. 3-FCO-064-0043, REACTOR ZONE EXH OUTBD ISOL DMPR

C. 3-FCO-064-0040, RX ZONE EXH TO SGTS, OPENS.

D. 3-FCO-064-0041, RX ZONE EXH TO SGTS, OPENS.

Examination Outline Cross-reference:	Level	RO	SRO
295026 (EPE 3) Suppression Pool High Water Temperature / 5	Tier #	1	------
EA2.02 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: • Suppression pool level	K/A #	295026EA2.02	
	Importance Rating	3.8	------

Proposed Question: **# 21**

Unit 2 suffered a small break LOCA with the following conditions:

- 2-EOI-2 Primary Containment Control, Suppression Pool Temperature leg is being executed
- Suppression Pool Temperature is 190 °F

Given the conditions above, which **ONE** of the following completes the statement below?

In accordance with EOI Curve 3, Heat Capacity Temp Limit (HCTL), action is **REQUIRED** if Reactor Pressure reaches ___**(1)**___ when Suppression Pool Level is ___**(2)**___.

[REFERENCE PROVIDED]

A. (1) 500 psig
(2) 17.5 feet

B. (1) 500 psig
(2) 16.5 feet

C. (1) 700 psig
(2) 17.5 feet

D. (1) 700 psig
(2) 16.5 feet

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible with Reactor Pressure at 500 psig and the given Suppression Pool Temperature, the intersecting point is located using Suppression Pool Level provided in second part. Second part is incorrect but plausible using Suppression Pool Level of 17.5 feet and the intersecting point with 500 psig from first part, Suppression Pool Temperature would have to be at or above 192 °F to exceed Heat Capacity Temperature Limit (HCTL). With the given Suppression Pool Temperature at 190 °F, HCTL is not exceeded, therefore action is not required.

B INCORRECT: First part is incorrect but plausible with Reactor Pressure at 500 psig and the given Suppression Pool Temperature, the intersecting point is located using Suppression Pool Level provided in second part. Second part is incorrect but plausible using Suppression Pool Level of 16.5 feet and the intersecting point from 500 psig in first part, Suppression Pool Temperature would have to be closer to 200 °F to exceed HCTL. With the given Suppression Pool Temperature at 190 °F, HCTL is not exceeded, therefore action is not required.

C **CORRECT:** *(See attached)* In accordance with EOI Curve 3, HCTL, action is required if **above** the curve for existing RPV Pressure listed at the bottom of the Curve (not provided on candidate's Curve 3). Using the given Suppression Pool Temperature line of 190 °F, follow the 700 psig Reactor Pressure curve to find the intersecting points. Second part, then follow the given 190 °F Suppression Pool Temperature line to where it intersects Suppression Pool Level of 17.5 feet. The resulting intersecting point is **above** the existing Reactor Pressure indicating that action is REQUIRED since HCTL has been exceeded.

D INCORRECT: First part is incorrect but plausible with Reactor Pressure at 700 psig and the given Suppression Pool Temperature, the intersecting point is located using Suppression Pool Level provided in second part. Second part is incorrect but plausible using Suppression Pool Level of 16.5 feet and the intersecting point from 700 psig in first part, Suppression Pool Temperature would have to be at or above 192 °F to exceed HCTL. With the given Suppression Pool Temperature at 190 °F, HCTL is not exceeded, therefore action is not required.

RO Level Justification: Tests the candidate's ability to determine and interpret Secondary Containment parameters as it relates to the EOI Curve 3, Heat Capacity Temperature Limit. Specifically, using Suppression Pool Level and Reactor Pressure to find the intersecting point above or below a given Suppression Pool Temperature, to determine if action is required in accordance with the EOIs. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must look at the conditions and put several pieces of mental data together to come up with a solution.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (4) Assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems and/or safety functions.

Technical Reference(s): 2-EOI-2, Rev. 16 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **2-EOI-2, Curve 3 – Heat Capacity Temperature Limit**

Learning Objective: OPL171.203, Obj. 12 As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1703 #19	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2017	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	X
	55.43	

Copy of Bank Question:

QUESTION 19 Rev 4

A LOCA occurred on Unit 2 and the crew is implementing 2-EOI-2, Primary Containment Control, Suppression Pool Temperature leg.

Suppression Pool Temperature is 190° F.

Which one of the following completes the statement below?

Action is required if Reactor Pressure is __ (1) __ psig and Suppression Pool Level is __ (2) __ feet.

[REFERENCE PROVIDED]

A. (1) 700
(2) 16

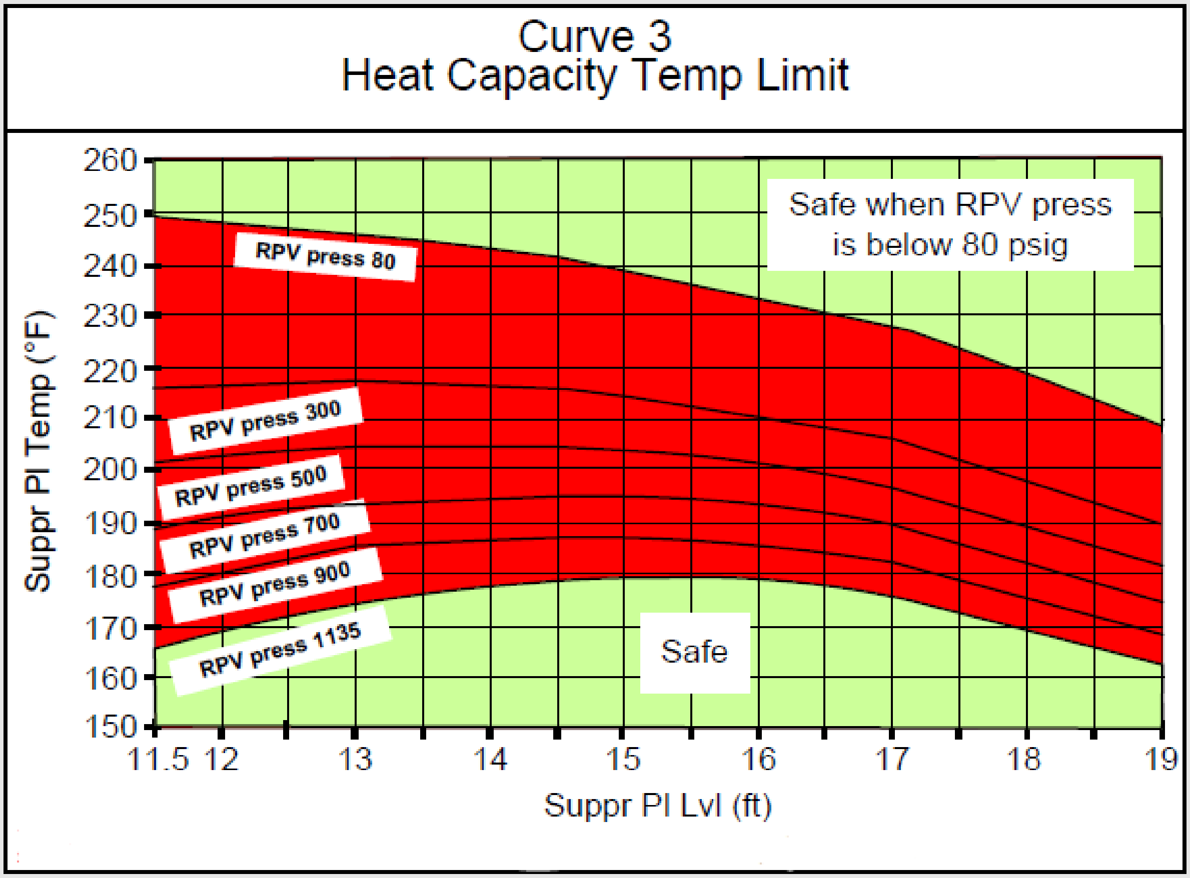
B. (1) 700
(2) 17

C. (1) 900
(2) 15

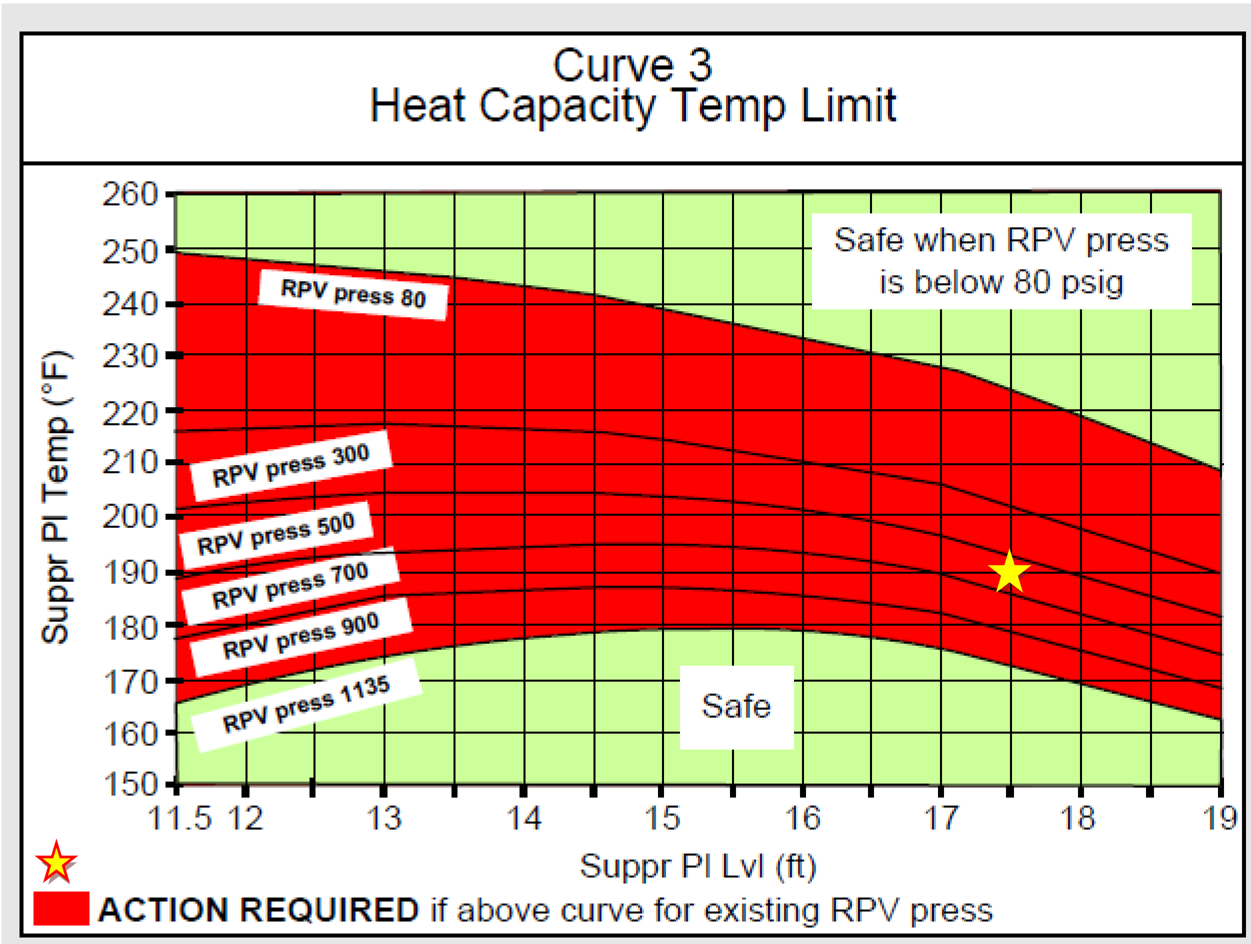
D. (1) 500
(2) 18

Answer: C

REFERENCE PROVIDED to candidate: 2-EOI-2, Curve 3, HCTL without ACTION REQUIRED statement



Excerpt from 2-EOI-2, Curve 3, HCTL: illustrating the intersecting point of the given Suppression Pool Temperature and Level as it relates to Reactor Pressure, indicates ACTION REQUIRED since above curve



Examination Outline Cross-reference:	Level	RO	SRO
215003 (SF7 IRM) Intermediate-Range Monitor	Tier #	2	------
K2.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of electrical power supplies to the following:	K/A #	215003K2.01	
• IRM channels/detectors	Importance Rating	2.5*	------

Proposed Question: **# 22**

Regarding the power supplies to IRMs, which **ONE** of the following completes the statement below?

IRM Channels/Detectors __________.

A. C and G are powered from **A** channel ± 24VDC System; D and H are powered from **B** channel ± 24VDC System

B. D and H are powered from **A** channel ± 24VDC System; C and G are powered from **B** channel ± 24VDC System

C. C and G are powered from Division **I**, 250 VDC System; D and H are powered from Division **II**, 250 VDC System

D. D and H are powered from Division **I**, 250 VDC System; C and G are powered from Division **II**, 250 VDC System

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 0-OI-57D, DC Electrical System, IRM recorders A, C, E, G are powered from **A** channel ± 24VDC System and B, D, F, H are powered from **B** channel ± 24VDC System.

B INCORRECT: Incorrect but plausible in that both SRM and IRM power supply arrangements are often confused by candidates especially as it relates to the complex DC System.

C INCORRECT: Incorrect but plausible in that the DC System power supplies are complex and often confused by candidates. Division I, II 250VDC relates to ECCS power supplies.

D INCORRECT: Incorrect but plausible in that the DC System power supplies are complex and often confused by candidates. Division I, II 250VDC relates to ECCS power supplies.

RO Level Justification: Tests the candidate's knowledge of electrical power supplies as it relates to the IRM Channels/Detectors from the DC System. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): 0-OI-57D, Rev. 174 (Attach if not previously provided)
OPL171.037, Rev. 16
OPL171.020, Rev. 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.037 Obj. 4 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1703 #38	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2017	

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTION 38

What are the power supplies to the SRM Channels / detectors?

A. 'A' & 'B' are powered from the 'A' channel $\pm$ 24VDC System and 'C' & 'D' are powered from the 'B' channel $\pm$ 24VDC System.

B. 'A' & 'C' are powered from the 'A' channel $\pm$ 24VDC System and 'B' & 'D' are powered from the 'B' channel $\pm$ 24VDC System.

C. 'A' & 'B' are powered from Division I, 250 VDC System and 'C' & 'D' are powered from Division II, 250 VDC System.

D. 'A' & 'C' are powered from Division I, 250 VDC System and 'B' & 'D' are powered from Division II, 250 VDC System.

Answer: B

Excerpt from 0-OI-57D:

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0174 Page 124 of 336

5.11 Placing Unit 3 ± 24V DC Neutron Monitoring Battery A(B) in Service to Battery Board 3 (continued)

[9] **ENSURE** the following equipment is in normal operation in accordance with 3-OI-90 Radiation Monitoring System, 3-OI-92 Source Range Monitoring, and 3-OI-92A Intermediate Range Monitoring:

[9.1] ± 24V Neutron Monitoring Channel A.

A. Panel 9-5

- IRM recorders A, C, E and G
- SRM Channel A and C indicators, recorders and period meters

B. Panel 9-10

- 3-RM-90-266A, OG POST-TREATMENT CH A RAD MON RTMR

[9.2] ± 24V Neutron Monitoring Channel B.

A. Panel 9-5

- IRM recorders B, D, F and H
- SRM Channel B and D indicators, recorders and period meters

Excerpt from OPL171.020 Lesson Plan:

OPL171.020 , Intermediate Range Monitor System, Rev# 12

d) Each trip unit has two inputs:
   (1) a reference input
   (2) a signal input
e) The trip unit trips when an input signal exceeds a preset reference signal.
f) Trip units produce two types of outputs
   (1) One output signal provides a seal-in indication which must be manually reset by the operator.
   (2) The other output signal resets automatically when the condition clears.
g) Signals are provided as inputs to the following:
   (1) Panel 9-5 indicating lights.
   (2) Panel 9-5 annunciators.
   (3) Panel 9-12 IRM drawer indicating lights.
   (4) Reactor Manual Control System (RMCS) for associated control rod blocks.
   (5) Reactor Protection System (RPS) for associated reactor scram signals.

14. Calibration Circuits
   a) Provide a built-in method of calibrating the indications and trips generated by the IRM.
   b) A signal generator initiates signals of either 40 or 125.
   c) These signals are input into the amplifier attenuator when the mode switch (S1) is in the 40 or 125 position.

Image 4 - IRM Channel 'A' S1

C. Power Supplies
1. The IRM power supplies receives unregulated $\pm$ 24 VDC power from the Neutron Monitoring battery and convert it to regulated voltages and proper magnitude for use by the IRM detectors and logic circuits.
   a) A loss of 24 VDC will give an INOP trip signal.
   b) Additionally, a loss of 24 VDC would result in a loss of IRM indication
2. Neutron monitoring battery chargers are fed from its unit's 250V Battery Board, Panel 8, which in turn is fed from I&C 'A' and 'B' regulating transformers.
3. Detector Drive Units are powered from I&C 'A'
   a) A loss of this power supply would result in an inability to insert or withdraw IRMs.

ILT Objective 3

Excerpt from OPL171.037 Lesson Plan:

OPL171.037, DC Systems, Rev: 16

C. Components 1. Chargers a) The chargers are solid-state (full-wave silicon rectifier type). One is connected in parallel with each 24V section of the center-tapped 48V batteries. They are capable of recharging a fully discharged battery while supplying the connected loads. Power supply to the chargers is from the 120V AC I&C buses.	Obj 5a
2. Batteries a) The batteries are 24-cell lead-acid types. Their 8-hour discharge rating is 75 amp-hours. They are capable of supplying their connected loads for 3 hours without recharging, to a minimum cell potential of 1.75V	Obj 4a
D. Distribution 1. The ± 24V DC Power System supplies power to various neutron-monitoring instrumentation during all modes of plant operation. Distribution (both channels) is from Panel 12 of each Unit's 250V Battery Board.	Obj 4c

Supports Distractors C and D:

6. The 250V RMOV Boards are supplied from Unit Battery Boards as follows:

250 RMOV BD	NORMAL SOURCE	ALTERNATE SOURCE
1A	BB-1	BB-2
1B	BB-3	BB-1
1C	BB-2	BB-1
2A	BB-2	BB-3
2B	BB-3	BB-1
2C	BB-1	BB-2
3A	BB-3	BB-2
3B	BB-1	BB-3
3C	BB-2	BB-3

*All transfers for these boards are manual only.

Obj 1d

0-OI-57d P&L

7. Loss of 250VDC from each unit's 'A' or 'B' 250V RMOV Board results in loss of that divisions ECCS ATU Inverter and the redundant 24VDC converter power supply. This results in a complete loss of voltage to the divisions ECCS ATU logic panels (panel 9-81 or 9-82) and loss of voltage to the other components/systems served by the inverter.

Obj 8

AOI-57-11

Supports Distractors C and D:

OPL171.037, DC Systems, Rev: 16

250V RMOV POWER SUPPLIES

U1			U2			U3		
250V RMOV BOARD	NORMAL	ALTERNATE	250V RMOV BOARD	NORMAL	ALTERNATE	250V RMOV BOARD	NORMAL	ALTERNATE
1A	BB1	BB2	2A	BB2	BB3	3A	BB3	BB2
1B	BB3	BB1	2B	BB3	BB1	3B	BB1	BB3
1C	BB2	BB1	2C	BB1	BB2	3C	BB2	BB3

0-45E701-1/0-45E702-1/0-45E703-1/0-OI-57-D

TYPICAL LOADS ON 250V RMOV BOARDS (U1 AND U3 SIMILAR)

250V RMOV 2A	250V RMOV BD 2B	250V RMOV BD 2C
2-FCV-73-34 HPCI DISCH VALVE 2-FCV-73-3 HPCI STEAM LINE ISOL 2-FCV-73-16 HPCI STM SUP VLV 2-FCV-73-40 HPCI PMP CST SUCT VLV 2-FCV-73-26 HPCI SC SUCT VLV 2-FCV-73-27 HPCI SC SUCT VLV 2-FCV-73-30 HPCI MIN FLOW VLV 2-FCV-73-44 HPCI PMP DISCH VLV 2-FCV-73-35 HPCI BYP TO CST VLV HPCI TURB AUX OIL PMP HPCI GLAND SEAL COND BLOWER HPCI GLAND SEAL COND CNDS PMP HPCI SYS LOGIC DIV II PNL 9-39 ECCS DIV II ATU INV CS SYS LOGIC DIV II-2 PNL 9-33 RHR SYS LOGIC II-2 PNL 9-33 RCIC SYS AND ADS LOGIC DIV II-2 PNL 9-33 VARIOUS MSRV POWER 4KV RPT BD 2-II NORM CONTROL PWR BU SCRAM VALVES PNL 9-15 MSIV OUTBD ISLN VLV DIV II PNL 9-43 MSIV OUTBD ISLN VLVS DIV II-2 PNL 25-32 2-FCV-74-47 RHR SDC SUCTION OUTBD 2-FCV-64-222 HARDENED SC OUTBD ISOL VLV RECIRC VFD 2A PNL 25-23	2-FCV-73-36 HPCI SHUTOFF TO CST VLV 2-FCV-71-3 RCIC STM LINE ISOL VLV 2-FCV-71-34 RCIC MIN FLOW VLV RCIC SYS LOGIC DIV I PNL 25-31 ECCS DIV I ATU INV CS SYS LOGIC DIV I PNL 9-32 RHR SYS LOGIC DIV I PNL 9-32 HPCI SYS LOGIC DIV I PNL 9-32 ADS SYS LOGIC BUS A DIV I PNL 9-30 ADS DIV I BUS B PNL 9-30 VARIOUS MSRV POWER MSIV INBD ISLN VLVS DIV I PNL 9-43 MSIV INBD ISLN VLVS DIV I PNL 25-32 2-FCV-69-2 RWCU ISOL VLV U2 HCVS INST TRANSFER SW 2-FCV-64-221 HARDENED SC OUTBD ISOL VLV FCV-74-108 RHR FLUSH DISCH VLV 2-FCV-1-56 MSL WARMING ISOLATION VLV RECIRC VFD 2B PNL 25-24	2-FCV-71-F RCIC TURB TRIP/THROTTLE VLV 2-FCV-71-37 RCIC PUMP DISCH VLV 2-FCV-71-39 RCIC PUMP INJ VLV 2-FCV-71-8 RCIC TURB STM SUP VLV 2-FCV-71-19 RCIC CST SUCT VLV 2-FCV-71-18 RCIC SC SUCT VLV 2-FCV-71-17 RCIC SC SUCT VLV 2-FCV-71-38 RCIC TEST BYP VLV RCIC GLAND SEAL VAC TK COND PMP 2FCV-71-25 RCIC LUBO COOLING WTR VLV RCIC TURB TRIP SOLENOID CKT VARIOUS MSRV POWER

2-45E712-1/-2/-3

Fig- 5 250V RMOV Board Loads

Examination Outline Cross-reference:	Level	RO	SRO
295031 (EPE 8) Reactor Low Water Level / 2	Tier #	1	------
EA2.01 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL:	K/A #	295031EA2.01	
• Reactor water level	Importance Rating	4.6*	------

Proposed Question: **# 23**

A Unit 2 LOCA has occurred and resulted in the following conditions:

- 2-PI-3-207A, REACTOR PRESSURE indicates 500 psig
- 2-LI-3-52 and 2-LI-3-62, RX WTR LEVEL ACCIDENT RANGE indicates (-) 220 inches

Given the conditions above, which **ONE** of the following completes the statements below?

Actual Reactor Water Level is ___**(1)**___ Top of Active Fuel (TAF).

To help ensure accurate Reactor Water Level can be reported, 2-LI-3-52 and 2-LI-3-62 require the use of correction curves due to being calibrated at ___**(2)**___.

[REFERENCE PROVIDED]

A. (1) above
(2) 0 psig / 212 °F

B. (1) above
(2) Normal Operating Pressure / Normal Operating Temperature

C. (1) below
(2) 0 psig / 212 °F

D. (1) below
(2) Normal Operating Pressure / Normal Operating Temperature

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible if the candidate correlates the determined Reactor Water Level from the correction curves to be above actual Top of Active Fuel or (-) 162 inches. Any of the lines on the chart could be the Top of Active Fuel, as the lines are not labeled. The lines have the respective Reactor Water Level that they correspond to but it requires the candidate to know what each numerical Reactor Water Level is since the chart's legend is not given to the candidate. The second part is correct (*See C*).

B INCORRECT: The first part is incorrect (*See A*). The second part is incorrect but plausible in that other (Narrow Range) Level Instruments are calibrated for Normal Operating Pressure (1035 psig) / Normal Operating Temperatures (550 °F) as indicated in 2-SR-2, Instrument Checks and Observations (*attached*), NOT Accident Conditions and are Temperature Compensated by a pressure signal.

C **CORRECT**: To ensure Adequate Core Cooling via Core Submergence in accordance with RPV Control (2-EOI-1) Table L-3, PIP 95-64 for 2-LI-3-52 / 2-LI-3-62 and BFN-ODM-4.20, Reactor Water Level must be above TAF (-) 162 inches. For the given Accident Conditions, the Operator is required to use the Correction Curves from the Operator aid on Panel 9-3. Candidate must determine that the given indicated Reactor Water Level at (-) 220 inches and the provided Reactor Pressure of 500 psig results in Reactor Water Level being below TAF (-) 162 inches just above the actual (-) 180 inches, therefore the Reactor Core is not fully submerged at this time. For second part, these two Post Accident Range Instruments (LI-3-52 / LI-3-62) are calibrated to indicate from (-) 268 inches to (+) 32 inches, a Reactor Water Level range. These Post Accident Monitoring (PAM) conditions assume the Reactor is depressurized to 0 psig (212 °F) in Accident Conditions and are NOT temperature compensated. Per Safety Analysis on Water Level instruments, the conclusion is that the Accident range instruments adequately indicate Water Level provided they are corrected for off-calibration conditions for RPV Pressure utilizing the operator aid (PIP 95-64) on Panel 9-3 for Reactor Water Level Correction.

D INCORRECT: The first part is correct (*See C*). The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability, utilizing Accident Condition parameters, to determine actual Reactor Water Level by correctly interpreting the required Reactor Water Correction Curves. This will help to ensure the status of Adequate Core Cooling via Core Submergence. This question is rated as C/A due to the requirement to assemble and integrate parameters using a graph to predict an outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (4) Assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems and/or safety functions.

Technical Reference(s): (Attach if not previously provided)
EOIPM 0-V-C, Rev.3
PIP 95-64, Rev. 3
BFN-ODM-4.20, Rev. 6
OPL171.003, Rev. 26
2-SR-2, Rev. 85

Proposed references to be provided to applicants during examination: **2-LI-3-52 & 62 CORRECTION CURVES, PIP 95-64 (without legend)**

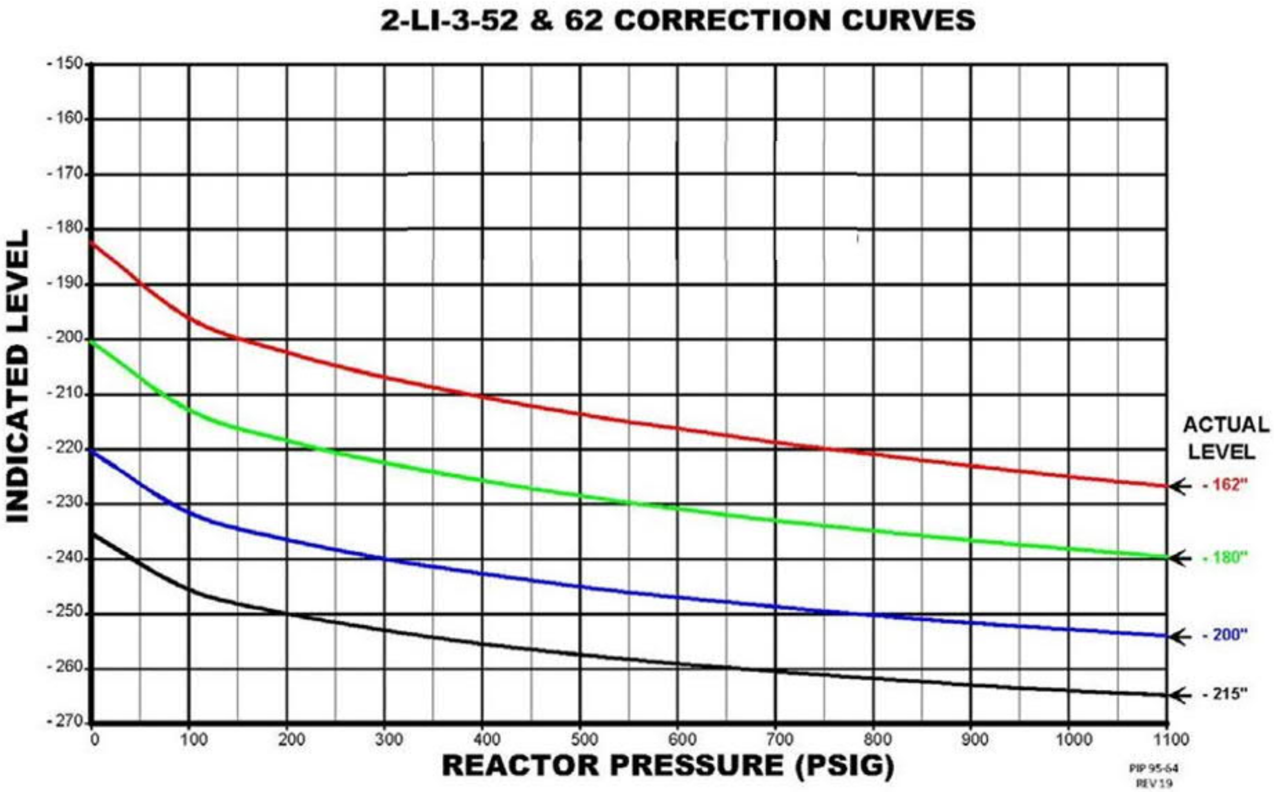
Learning Objective: OPL171.003 Obj. 15 (As available)
OPL171.202 Obj. 7

Question Source:	Bank #		
	Modified Bank #	BFN 1804 #16	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

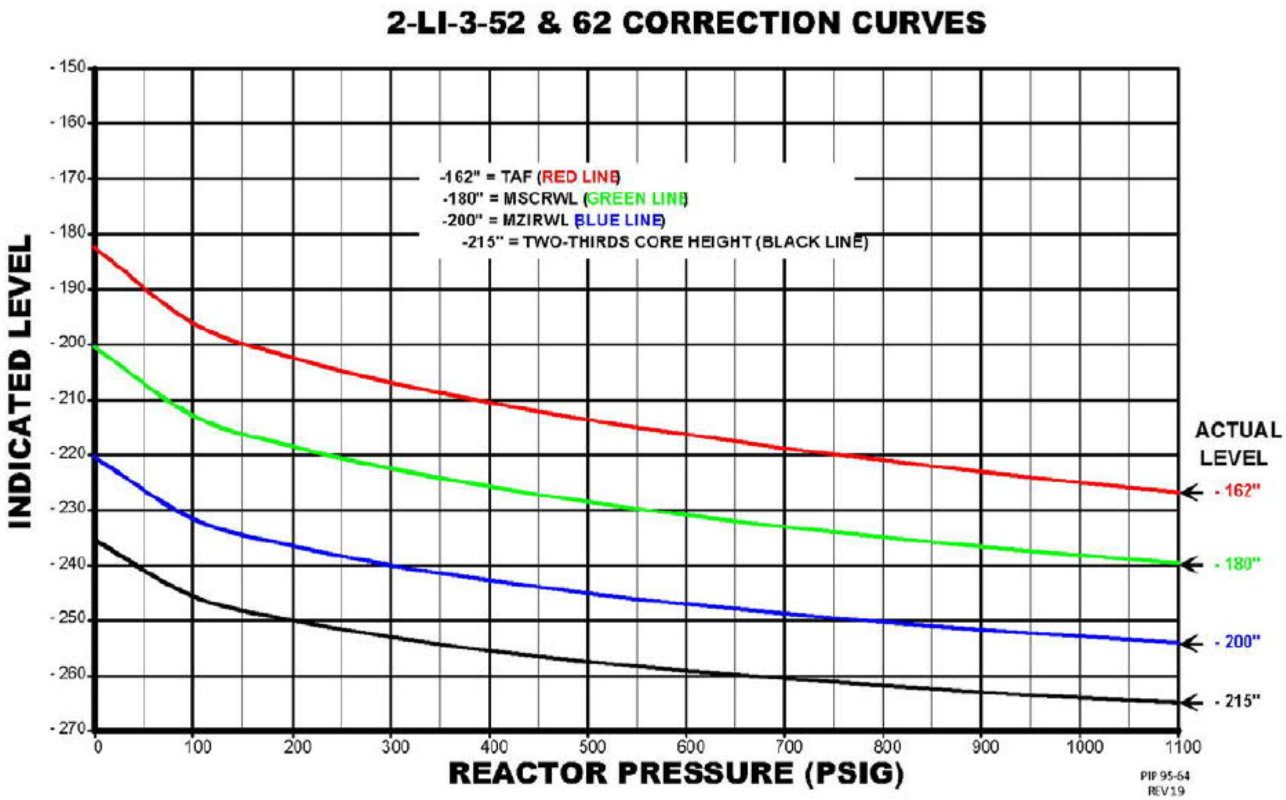
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

REFERENCE PROVIDED to candidate: PIP 95-64 without legend



PIP 95-64 with legend:



Copy of Bank Question:

ILT 1804 Written Exam

16. A Unit 2 Loss of Coolant Accident (LOCA) has occurred and resulted in the following conditions:

- 2-PI-3-207A, REACTOR PRESSURE indicates 400 psig
- 2-LI-3-52 and 2-LI-3-62, RX WTR LEVEL ACCIDENT RANGE indicates (-) 190 inches

Given the current conditions, which ONE of the following completes the statements below?

The Top of Active Fuel (TAF) ___**(1)**___ submerged at this time.

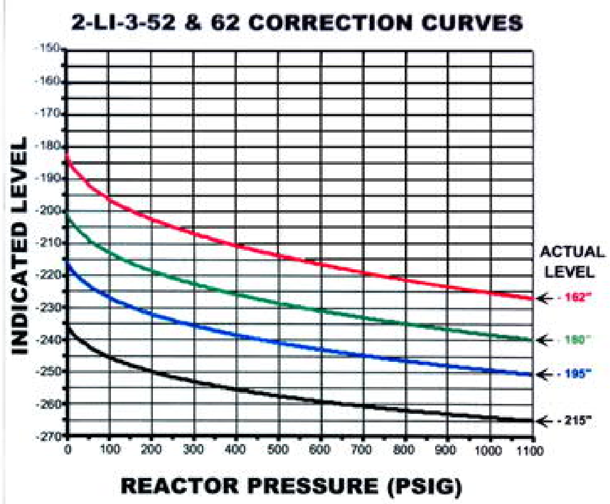
To help ensure accurate Reactor Water Level can be reported, 2-LI-3-52 and 2-LI-3-62 require the use of correction curves due to being calibrated at ___**(2)**___ .

A. **(1)** is
(2) 0 psig / 212 °F

B. **(1)** is
(2) Normal Operating Pressure / Normal Operating Temperature

C. **(1)** is NOT
(2) 0 psig / 212 °F

D. **(1)** is NOT
(2) Normal Operating Pressure / Normal Operating Temperature



Excerpts from EOIPM 0-V-C:

BFN Unit 0	EOI-1, RPV Control Bases	EOIPM Section 0-V-C Rev. 0003 Page 23 of 141

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/L flowpath

The RPV water level control flowpath establishes and maintains adequate core cooling through core submergence. A list of the preferred injection systems for use in doing so is provided; along with a list of alternate injection subsystems which may be used to augment RPV water level control (refer to the definition of alternate injection subsystems in EOIPM Section 0-I-C).

BFN Unit 0	EOI-1, RPV Control Bases	EOIPM Section 0-V-C Rev. 0003 Page 75 of 141

1.0 EOI-1, RPV CONTROL BASES (continued)

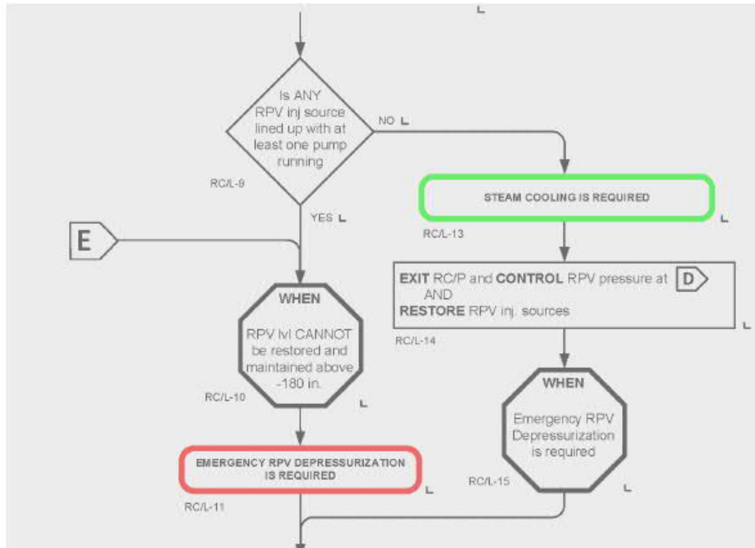
DISCUSSION: RC/L-12

Adequate core cooling with injection is ensured following emergency RPV depressurization as long as one of three conditions exists (Table L-3):

- The core is completely submerged in which case override RC/L-4 requires return to Step RC/L-3.
- RPV water level can be restored and maintained above **A.71** (Minimum Steam Cooling RPV Water Level). The core is then cooled by a combination of submergence and steam cooling even with no core spray flow.
- Design core spray flow requirements are satisfied and RPV water level can be restored and maintained at or above **A.49** (Two-Thirds Core Height Water Level).

1.0 EOI-1, RPV CONTROL BASES (continued)

RC/L-12 (cont'd)



RESTORE and **MAINTAIN** RPV water lvl above -180 in. using one or more Preferred Injection Systems (Table L-1)

- OK to use ANY of the Alternate Injection Subsystems (Table L-2)

IF	THEN
RPV water lvl cannot be restored and maintained above -180 in. AND Spray Cooling cannot be restored and maintained (Table L-3)	**MAXIMIZE** injection with ALL available sources
Maximizing injection with ALL available sources cannot restore and maintain Spray Cooling or RPV water lvl above -180 in. (Table L-3)	**WHEN** the TSC team assumes command and control **THEN** **EXIT** all EOIs and **ENTER** all SAMGs

RC/L-12

Table L-3
Adequate Core Cooling

Adequate core cooling exists if one of the following is met:

• Core submergence:	RPV water lvl above -162 in.
• Steam Cooling with injection:	RPV water lvl above -180 in.
• Spray cooling:	Either CS subsystem operating with at least 6,250 gpm to the RPV AND RPV water lvl above -215 in.
• Steam cooling without injection:	RPV water lv above -200 in.

BFN Unit 0	EOI-1, RPV Control Bases	EOIPM Section 0-V-C Rev. 0003 Page 79 of 141

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/L-13, RC/L-14, RC/L-15

If the decreasing RPV water level trend has not been reversed before RPV water level drops to the top of the active fuel and no source of injection into the RPV is available (NO path from RC/L-12), the only mechanism able to provide adequate core cooling is steam cooling.

Flow path D, Steam Cooling is an RPV pressure control action and is thus entered by way of the third override in RC/P-4. While in flow path D, Step RC/L-14 continues attempts to restore RPV injection capability. Steam cooling is not initiated until RPV water level decreases to the top of the active fuel because:

- Adequate core cooling exists so long as RPV water level remains above the top of the active fuel.

Excerpt from BFN-ODM-4.20:

BFN Operations Directive Manual	Strategies for Successful Transient Mitigation	BFN-ODM-4.20 Rev. 0006 Page 17 of 25

4.8.2 RPV Control (EOI-1)

A. Level Leg of flowchart

Direct an initial reactor water level band of +2 to +51" or as directed by the EOI's. This gives a manageable band with level control still in the indicating range and in automatic level control.

Reporting RWL at TAF: For an Operator to report that Reactor Water Level has reached Top of Active Fuel (-162 inches), the Emergency Range instrument reading -155 inches is not adequate. The LI-3-52 & 62 Correction Curves PIP and LI-52 or LI-62 level indications must be referenced before declaring that Reactor Water Level is at -162 inches.

Use of the Emergency High Pressure Makeup Pump (EHPM) while in EOI-1 is allowed in order to restore Reactor Water Level to +2" to +51" (see Override in RC/L-3)

The achievement of saturation conditions in the Drywell is not sufficient in and of itself to call the levels instruments unreliable. WHEN indications of reference leg flashing in the instruments are observed, THEN the water level should be considered unreliable for that instrument. Indications of flashing include erratic indications, large oscillations and large mismatches between level indicators. IF all indications are unreliable, THEN level is "unknown". Another Licensed operator or the STA should be used to validate these conditions.

Excerpt from 2-SR-2:

BFN Unit 2	Instrument Checks and Observations	2-SR-2 Rev. 0085 Page 151 of 156

Attachment 4
(Page 1 of 1)

Reactor Water Level Indication Correction

The Reactor Water Level Instrumentation tables in Attachment 2 are arranged such that only instruments in the same compensation group are compared. However, determination of corrected level indication may be required during operation at off-normal conditions or if desired to compare instruments from different compensation groups.

Corrected level indication may be used for satisfying MAX DEV criteria provided the following are observed:

- The parameter correction is appropriately applied to all instruments being compared.
- Both the indicated and corrected level indications are recorded and annotated in Attachment 2 along with the bases for the corrected level indication.

Corrected level indication can be determined from the following table which provides commonly needed corrections or from Technical Instruction 2-TI-149. The following table presents Reactor Water Level as: indicated, corrected for Reactor Vessel Temperature 100°F, and corrected for Reactor Vessel Temperature 212°F for various water level instruments. Enter the indicated Rx water level and find the correct instrument column and use the closest Rx vessel temperature. (Matching corrected levels between instruments and subtracting the associated indicated levels will yield an approximate deviation value in inches between those instruments. i.e., If the Narrow Range Compensated Instrument is reading 38", the corrected level would be 32". Also, using a corrected value of 32" in the Narrow Range Uncompensated column shows the instrument should be reading 47". Therefore, a deviation of approximately 9" would be expected between the Narrow Range Compensated instrument and the Narrow Range Uncompensated instrument.)

INDICATED LEVEL	CORRECTED LEVEL											
Indicated Reactor Water Level	Narrow Range Compensated 2-LIS-3-53(60) (206) (253) Level		Narrow Range Uncompensated 2-LIS-3-184, 185,203(A-D), 208(A-D) Level		Wide Range 2-LI-3-58A(B), 2-LIS-3-56A(D) Level		Post Accident 2-LR-3-62 2-LI/LIS-3-52 2-LI/LIS-3-62A Level		Floodup 2-LI-3-55 Level		Wide Range 2-LI-3-46A(B) Level
	100°	212°	100°	212°	100°	212°	100°	212°	100°	212°	
50	40.5	42.5	34.5	36					48.5	50	
48	39	41	33	35					46.5	48	
46	37.5	39.5	31.5	33					44.5	46	
44	36	38	30	32					42.5	44	
42	35	36.5	28.5	30.5					40.5	42	
40	33.5	35	27	29	Note 1	Note 2	No Calculated Correction Value		38.5	40	No Calculated Correction Value
38	32	34	26	27.5					36.5	38	
36	30.5	32.5	24.5	26					34.5	35.5	
34	29	31	23	24.5					32.5	33.5	
32	28	29.5	21.5	23					30.5	31.5	
30	26.5	28	20	21.5					28.5	29.5	
28	25	26.5	19	20					26.5	27.5	

(1) Indicates > 60" if actual Water Level is > 5".
(2) Indicates > 60" if actual Water Level is > 11.5".

Excerpts from OPL171.003 Lesson Plan:

OPL171.003 REACTOR VESSEL PROCESS INSTRUMENTATION REV 26

c. Definitions — Obj. ILT-4, NLO/NLOR-1
   1) Reactor vessel zero:
      a) Reactor pressure vessel bottom head invert (the inside of the bottom head)
      b) Provides reference for all in core components and vessel nozzle taps.
   2) Instrument zero:
      a) 528 inches above vessel zero

d. Four ranges of level indication — Obj. ILT-5
   1) Normal Control Range (Narrow Range) — Obj. NLO/NLOR-2
      a) 0 to +60 inch range covering the normal operating range (analog) with +60" up to +70" digital and 0" down to – 10" digital readings. — Obj. ILT-8
      b) Referenced to instrument zero — Obj. NLO-4/ NLOR-3
      c) Four of these instruments are used by Feedwater Level Control System (FWLCS). — Obj. ILT-11.i/ LOR-7.i, ILT-13. LOR-8,NLO-6
         (1) The level signal utilized by the FWLCS is NOT directed through the Analog Trip System.
         (2) Temperature compensated by a pressure signal
         (3) Most accurate level indication available to the operator
         (4) Calibrated for normal operating pressure and temperature
      d) These indicators and a recorder point (average of the four) are located on Panel 9-5

         NOTE: An air bubble or leak in the reference leg can cause inaccurate readings in a nonconservative direction resulting in a mismatch between level indicators. This problem is particularly prevalent after extended outages when starting up from cold shutdown conditions and at low reactor pressures
      e) Four other narrow range instruments are located in the control room, two above the FWLCS level indicators on panel 9-5 (3-208A & D), one above HPCI (3-208B) and one above RCIC (3-208C) on panel 9-3.
      f) Calibration Conditions:
         (1) Reactor Pressure ~ 1035 psig
         (2) Reactor Temperature ~ 550 F
         (3) DW Temperature 135 F
         (4) Reactor Bldg Temp 80 F
         (5) No Recirc flow impact

4) Post-accident Flood Range

a) -268" to +32" range covering active core area and overlapping the lower portion of the Normal Control Range.

Obj.ILT-15./ LOR-9/ NLO-7

b) Referenced to instrument zero

c) Calibration Conditions:
(1) Accident conditions
(2) Reactor pressure at 0 psig (212°F)
(3) DW temperature 212°F
(4) RB Temperature 212°F
(5) Recirculation pumps tripped.

d) Variable leg tap is from diffuser of jet pumps 1 and 6 (or 11 and 16).

e) Per Safety Analysis on water level instruments the conclusion is that the accident range instruments adequately indicate water level-- provided they are corrected for off-calibration conditions of RPV pressure utilizing the operator aid on Panel 9-3 for level correction.

Obj.ILT-15./LOR-9 /NLO-7

Obj. ILT-11.d/ LOR-7.d

f) An interlock associated with this range will prevent using the RHR System for containment de-pressurization when it is needed to flood the core region.

e. Level instrumentation and piping layout

1. The major portions of the reference legs for the normal control range (A and B) and the Emergency Systems Range (A and B) are outside the drywell, thereby greatly reducing the drywell temperature effects on indicated water level. The reference leg for level transmitter LT-3-55 is still in the drywell.

2. Reference leg piping penetrates the containment on the south side through the penetration for old head spray piping. On the north side the reference leg piping penetrates the primary containment via the drywell purge line.

3. Each reactor instrument line (inside the drywell) connects to a condensing chamber, which is just an enlarged volume in the piping.

4. Piping run is uphill from the vessel to the condensing chamber, and then it is sloped downward to the transmitters, making the condensing chamber the highest point of the instrument line.

OPL171.003 REACTOR VESSEL PROCESS INSTRUMENTATION REV 26

h. Density effects on reactor vessel level ranges — Obj. ILT-9./ LOR-5

1) All vessel level indicators are calibrated at the reactor conditions for which they are to be used.

a) Narrow and Wide ranges are calibrated hot. Rated Operating pressure/ temperature

b) Flood-up and Post-accident ranges are calibrated cold.

2) Unless some density compensation is used, any level instrument actual measurement span will change with vessel pressure and temperature. This is due to changes in reactor water density with little or no change in reference column density.

The variable legs passing through the RWCU pump room may be at elevated temperatures introducing additional level mismatches between the A and B side instruments.

3) Plant transients can cause errors in the sensed level primarily because of the vessel pressure and drywell temperature variations.

a) Reference leg flashing is still possible during rapid de-pressurization transients; however, the effect is minimal since most of the reference legs (vertical runs) are located outside the drywell and are now much cooler than before (during reactor power operations). Additionally "RVLIS" minimize this effect. — Obj. ILT-10/ LOR-6

Reference leg flashing results in erroneous high level indications. Level setpoints are shifted in a non-conservative direction.

Transient flashing effects can cause indicated level to oscillate or be erratic. As the reference leg refills, the indicated level approaches a more accurate water level indication. The RVLIS mod decreases the time necessary for this refill to occur

i. Normal Control Range (Narrow Range) and Emergency Systems Range (Wide Range) Level Discrepancies — Obj. ILT-9/ LOR-5; Obj. ILT-15/LOR-9; Obj. NLO-7

1) Narrow Range level instrumentation is calibrated to be most accurate at rated temperature and pressure (particularly the instruments for FWLCS, since they are temperature compensated). At cold conditions the non-FWLCS instruments read high (not temperature compensated).

2) Wide Range instruments are also calibrated for rated temperature and pressure

OPL171.003 REACTOR VESSEL PROCESS INSTRUMENTATION REV 26

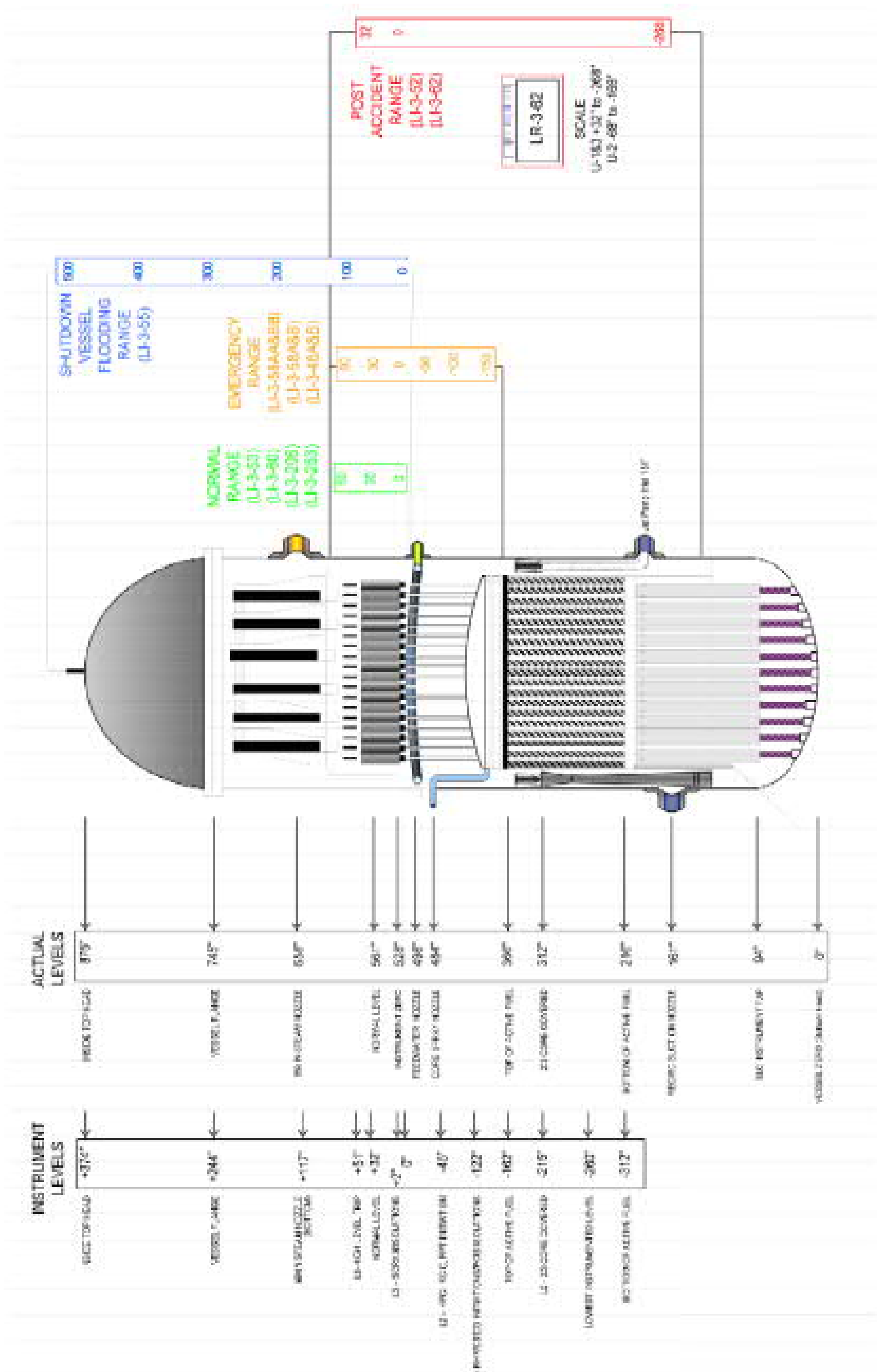


Figure - 3 Vessel Level Instrument Ranges

Examination Outline Cross-reference:

218000 (SF3 ADS) Automatic Depressurization

A2.02 (10CFR 55.41.5)

Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

- Large break LOCA

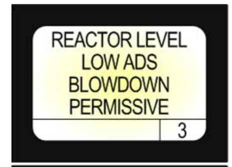
Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	218000A2.02	
Importance Rating	3.5	------

Proposed Question: **# 24**

Unit 3 is operating at 100% RTP, when a LOCA occurs with the following conditions:

At 09:02:00

- REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 3) alarms



- DRYWELL PRESSURE APPROACHING SCRAM (3-9-3B, Window 30) alarms

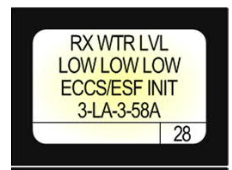
- The Operator inserts a manual Reactor SCRAM

At 09:03:00

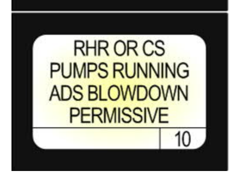
- ADS BLOWDOWN HIGH DRYWELL PRESSURE SEAL-IN (3-9-3C, Window 33) alarms

At 09:04:00

- RX WTR LOW-LOW-LOW ECCS/ESF INITIATE (3-9-3C, Window 28) alarms



- RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 10) alarms



NO further Operator actions have been taken

Which **ONE** of the following identifies the **EARLIEST** time that the Automatic Depressurization System (ADS) will automatically initiate?

A. 09:05:35

B. 09:07:25

C. 09:08:25

D. 09:10:00

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* ADS will initiate after 95 seconds with a Reactor Water Level of (-) 122 inches and a Drywell Pressure of 2.45 psig as long as sufficient ECCS pumps are running. At time 09:04:00 the conditions are met for the 95-second timers to start so at 09:05:35 ADS will initiate.

B INCORRECT: At 09:07:25 is incorrect but plausible in that since it is 265 seconds after Drywell Pressure reaches 2.45 psig. The 265 second timer is the High Drywell Pressure bypass timer. This is a misapplication of the 265 second timer which is intended for situations when High Drywell Pressure does not exist but Low-Low-Low Reactor Water Level does exist.

C INCORRECT: At 09:08:25 is incorrect but plausible in that since one of the ADS timers is 265 seconds and 09:08:25 is 265 seconds after the required conditions are met for the 95-second timer to start.

D INCORRECT: Incorrect but plausible in that 09:10:00 is plausible if the candidate thinks both timers must time out. 09:10:00 is 360 seconds (265-second timer plus 95-second timer) after conditions exist to start the 95-Second timer.

RO Level Justification: Tests the candidate’s ability to correctly recognize when the conditions are met for automatic initiation of ADS based on the respective logic. This question is rated as C/A due to the requirement to assemble, sort, and integrate the permissive logic given time and plant conditions to predict an outcome. The candidate must tie several alarms to setpoints and interpret their meaning to achieve the integrated outcome.

Technical Reference(s): (Attach if not previously provided)
3-OI-1, Rev. 47
3-ARP-9-3B, Rev. 23
3-ARP-9-3C, Rev. 32
OPDP-1, Rev. 46
2-730E929-2, Rev. 24

Proposed references to be provided to applicants during examination: **REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 3), DRYWELL PRESSURE APPROACHING SCRAM (3-9-3B, Window 30), ADS BLOWDOWN HIGH DRYWELL PRESSURE SEAL-IN (3-9-3C, Window 33), RX WTR LOW-LOW-LOW ECCS/ESF INITIATE (3-9-3C, Window 28), RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 10)**

Learning Objective: OPL171.043 Obj. 6 (As available)

Question Source:

Bank #	BFN 1804 #40	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam 2018

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Copy of Bank Question:

Unit 3 is operating at 100% RTP, when a LOCA occurs.

At time 09:02:00

- REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 3) alarms
- DRYWELL PRESSURE APPROACHING SCRAM (3-9-3B, Window 30) alarms
- The Operator inserts a manual Reactor SCRAM

At time 09:03:00

- ADS BLOWDOWN HIGH DRYWELL PRESSURE SEAL-IN (3-9-3C, Window 33) alarms

At time 09:04:00

- RX WTR LOW-LOW-LOW ECCS/ESF INITIATE (3-9-3C, Window 28) alarms
- RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 10) alarms

NO FURTHER OPERATOR ACTIONS HAVE BEEN TAKEN

Which ONE of the following identifies the **EARLIEST** time that the Automatic Depressurization System (ADS) will automatically initiate?

A. 09:05:35

B. 09:07:25

C. 09:08:25

D. 09:10:00

Proposed Answer: **A**

Excerpt from 3-OI-1:

BFN Unit 3	Main Steam System	3-OI-1 Rev. 0047 Page 14 of 71

3.4 Main Steam Relief Valve (MSRV / ADS) (continued)

F. ADS will initiate when ALL of the following conditions are met:

1. A confirmatory low reactor water level signal (+2.0 inches), REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE, 3-9-3C Window 3,

2. Two coincident signals for each of the following parameters:

   a. high drywell pressure (+2.45 psig) in conjunction with low-low-low reactor water level (-122 inches), ADS BLOWDOWN HIGH DRYWELL PRESS SEAL-IN, 3-9-3C Window 33 and RX WTR LVL LOW LOW LOW ECCS/ESF INIT 3-LA-3-58A, 3-XA-55-9-3C Window 28

   OR

   b. ow-low-low reactor water level (-122 inches), RX WTR LVL LOW LOW LOW ECCS/ESF INIT 3-LA-3-58A, 3-XA-55-9-3C Window 28, for 265 seconds (High drywell pressure bypass)

   c. When the above logic is satisfied, the 95 second timer starts (ADS BLOWDOWN TIMERS INITIATED, 3-XA-55-9-3C, Window 11).

3. ADS 95 second timer timed out.

4. One RHR pump OR two Core Spray pumps (A or B and C or D) running, RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE, 3-XA-55-9--3C Window 10.

Excerpts from 3-ARP-9-3C: (all annunciators given)

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0032 Page 6 of 43

REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE 3

(Page 1 of 1)

Sensor/Trip Point:

LIS-3-184	RPV level
LIS-3-185	≤ +2.0 inches

Sensor Location:

LIS-3-184	LIS-3-185
Panel 9-81	Panel 9-82
Aux Inst. Rm	Aux Inst. Rm

Probable Cause:

A. SI/SR is in progress.
B. Low reactor water level (Level 3).
C. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** Reactor water level by multiple indications.
B. **DISPATCH** personnel to Aux Instrument Rm El 593, to check relays energized:
   1. Panel 9-30, Relay 2E-K29
   2. Panel 9-33, Relay 2E-K24

C. **REFER TO** Tech Spec Section 3.3.5.1 and 3.5.1.
D. **EVALUATE** equipment associated with this alarm that is out of service to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0032 Page 14 of 43

RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE 10

(Page 1 of 2)

Sensor/Trip Point:

Core Spray Pump	Instrument	Panel	Relay	Setpoint
A	PS-75-7	25-1	14A-K27A	≥ 185 psig
B	PS-75-35	25-60	14A-K27B	≥ 185 psig
C	PS-75-16	25-1	14A-K28A	≥ 185 psig
D	PS-75-44	25-60	14A-K28B	≥ 185 psig

RHR Pump	Instrument	Panel	Setpoint
A	PS-74-8A & 8B	25-59	≥ 100 psig
B	PS-74-31A & 31B	25-62	≥ 100 psig
C	PS-74-19A & 19B	25-59	≥ 100 psig
D	PS-74-42A & 42B	25-62	≥ 100 psig

Panel 25-1	Rx Bldg, El 519'	R-16 N-LINE
Panel 25-59	Rx Bldg, El 519'	R-15 T-LINE
Panel 25-60	Rx Bldg, El 519'	R-20 N-LINE
Panel 25-62	Rx Bldg, El 519'	R-21 T-LINE

Probable Cause:

A. One RHR pump running with ≥ 100 psig discharge pressure or one CS pump running with ≥ 185 psig discharge pressure.

NOTE

Logic for ADS requires only one RHR pump or two CS pumps, (3A or 3B) and (3C or 3D) to initiate either Logic Bus if all other initiation requirements are met. Either Logic Bus initiation will cause the ADS System to actuate after all relays have timed out.

RHR Pumps Switches	Relays
PS-74-8A and PS-74-19A	Relay 10A-K102A
PS-74-8B and PS-74-19B	Relay 10A-K103A
PS-74-31A and PS-74-42A	Relay 10A-K102B
PS-74-31B and PS-74-42B	Relay 10A-K103B

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0032 Page 35 of 43

RX WTR LVL LOW LOW LOW ECCS/ESF INIT 3-LA-3-58A	28

(Page 1 of 1)

Sensor/Trip Point:

LIS-3-58A, B, C and D ≤ -122 inches (RPV low-low-low level)(Level 1)

Sensor Location: Panel 9-81, 82 Aux Instrument Room

Probable Cause:

A. Reactor Water Low Level
B. SI/SR in progress.

Automatic Action: (One out-of-two taken twice logic)

A. The following receive auto start signals:
- Core Spray System
- RHR System LPCI Mode
- Diesel Generators
- RHRSW (EECW) pump

B. ADS blowdown logic input.

Operator Action:

A. **CHECK** RPV water level using multiple indications.
B. **REFER TO** the EOIs.
C. **EVALUATE** equipment associated with this alarm that is out of service to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0032 Page 41 of 43

ADS BLOWDOWN HIGH DRYWELL PRESS SEAL-IN 33	Sensor/Trip Point:	
	PIS-64-57A	Drywell Press ≥ 2.45 psig
	PIS-64-57B	
	PIS-64-57C	
	PIS-64-57D	

(Page 1 of 1)

Sensor Location:	PIS-64-57 A and C Panel 9-82 Aux Inst Rm El 593'	PIS-64-57 B and D Panel 9-81 Aux Inst Rm El 593'

Probable Cause:

A. Drywell press ≥ 2.45 psig.
B. SI/SR in progress.
C. Sensor malfunction.

Automatic Action: ADS high Drywell press signal seals in.

Operator Action:

A. CHECK Drywell pressure by multiple indications.
B. IF alarm is valid, THEN
   ENTER 3-EOI-1 Flowchart and 3-EOI-2 Flowchart.
C. IF alarm is NOT valid, THEN
   RESET high Drywell pressure seal-ins, 3-XS-1-158 and -160, on Panel 3-9-3.
D. if alarm will NOT reset, THEN
   DISPATCH personnel to Aux Instrument Room to investigate the following:
   1. Panel 3-9-30 Relays 2E-K2 or 2E-K3 energized.

Excerpt from 3-ARP-9-3B: (annunciator given)

BFN Unit 3	Panel 9-3 3-XA-55-3B	3-ARP-9-3B Rev. 0023 Page 33 of 38

DRYWELL PRESS APPROACHING SCRAM 3-PA-64-58	
	30

(Page 1 of 1)

Sensor/Trip Point:

PIS-64-58E
PIS-64-58F 1.96 psig
PIS-64-58G
PIS-64-58H

Sensor Location:	PIS-64-58E, 58G Aux Inst Rm Panel 9-81	PIS-64-58F, 58H Aux Inst Rm Panel 9-82

Probable Cause:
A. Drywell pressure rising.
B. Drywell cooler(s) failure.
C. Steam or water leak inside Drywell.
D. Loss of RBCCW to Drywell coolers.

Automatic Action: Containment Spray Pressure Permissive is met by PIS-64-58E-H at 1.96 psig.

Operator Action:
A. CHECK containment pressure and temperature using multiple indications. ☐
B. REFER TO 3-AOI-64-1. ☐

Excerpt from 3-ARP-9-3C: (NOT given) illustrates the timer logic inputs for ADS

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0032 Page 16 of 43

ADS BLOWDOWN TIMERS INITIATED

11

(Page 1 of 2)

Sensor/Trip Point:

Auxiliary relays (2E-K5 and 2E-K16) energize when all the following conditions are met:

A. High Drywell Pressure, PS-64-57A(B), ≥ 2.45 psig.

B. RPV low-low-low level, LS-3-58A(C), ≤ -122 inches (Seals in and times out in 265 seconds to bypass High Drywell Pressure 2.45 psig).

C. RPV low level, LIS-3-184(185), ≤ +2.0 inches.

Sensor Location:	Panel 9-81A and 9-82 Aux Inst Rm El 593	Panel 9-30 and 9-33 Aux Inst Rm El 593

Probable Cause:

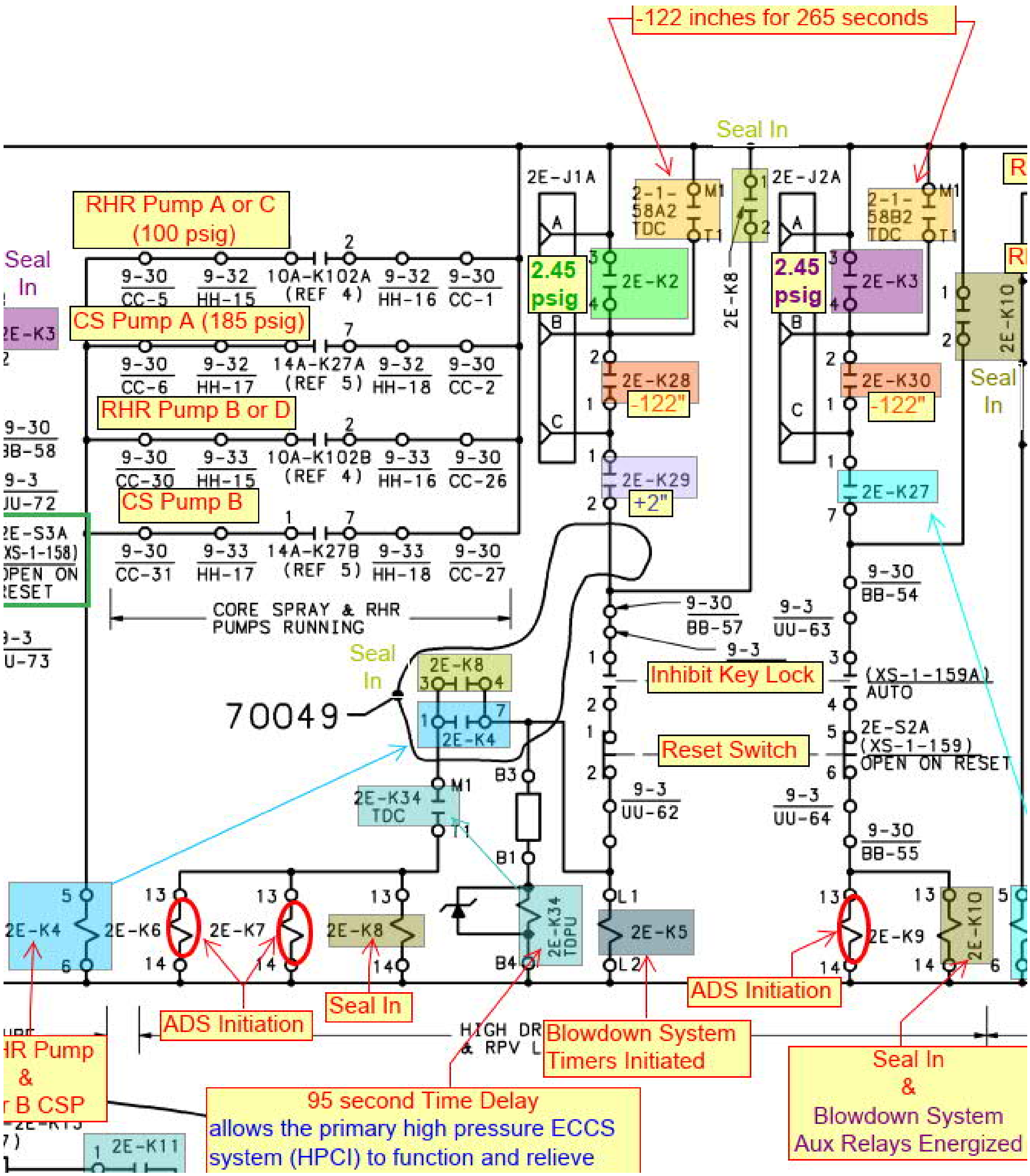
A. Possible LOCA
B. SI/SR in progress
C. Sensor malfunction

Automatic Action:

A. Aux Relays 2E-K9 (Bus A) and 2E-K20 (Bus B) energize immediately upon the initiating signals.
B. ADS timers relays 2E-K34 (Bus A) and/or 2E-K35 (Bus B) also energizes immediately upon the initiating signals.
C. After 95 seconds from energization of ADS timers relays 2E-K34 (Bus A) and/or 2E-K35 (Bus B), relays 2E-K6 (Bus A) and/or 2E-K17 (Bus B) will energize.
D. When either Logic Bus A (2E-K9 and 2E-K6) or Logic Bus B (2E-K20 and 2E-K17) are energized, the ADS is initiated.

Continued on Next Page

Excerpts from 2-730E929-2: Illustrating ADS automatic initiation logic



Examination Outline Cross-reference:

295002 (APE 2) Loss of Main Condenser Vacuum / 3

AK1.04 (10CFR 55.41.10)

Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM:

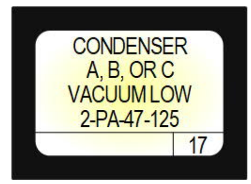
- Increased offgas flow

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295002AK1.04	
Importance Rating	3.0	------

Proposed Question: **# 25**

Unit 2 is at 85% Reactor Power when the following conditions occur:

- 2-XR-66-20, OG FLOW TO SIX HOUR HOLDUP VOLUME, is 100 scfm and rising
- CONDENSER A, B, OR C VACUUM LOW (2-9-7B, Window 17) alarms



Given the conditions above, which **ONE** of the following completes the statements below?

In accordance with the associated ARP, the probable cause for the conditions above is ___**(1)**___.

In accordance with 2-AOI-47-3, Loss of Condenser Vacuum, the Unit Operator (UO) will be directed to ___**(2)**___ in an attempt to maintain Condenser Vacuum.

Note: Steam Jet Air Ejector (SJAE)

A. (1) SJAE stalling
(2) reduce Reactor Power

B. (1) SJAE stalling
(2) start a Mechanical Vacuum Pump

C. (1) Main Condenser air in-leakage
(2) reduce Reactor Power

D. (1) Main Condenser air in-leakage
(2) start a Mechanical Vacuum Pump

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that in accordance with 2-OI-66, Off-Gas System, Precaution and Limitation 3.0.U, a SJAE stall will cause low Main Condenser Vacuum. However, Off-Gas flow would lower in that case. The second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). The second part is incorrect but plausible in that in accordance with Precaution and Limitation 3.0.K of 2-OI-66, Mechanical Vacuum Pumps are not to be used when Reactor Power is above 5%.

C **CORRECT**: *(See attached)* In accordance with 2-AOI-47-3, Loss of Main Condenser Vacuum, rising Off-Gas Flow would indicate Condenser in-leakage as long as the Off-Gas System is functioning properly. No failure of the Off-Gas System is mentioned in the stem of the question. For second part, Reactor Power is required to be reduced in order to maintain condenser vacuum.

D INCORRECT: The first part is correct (*See C*). The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the causes and operational implications of high Off-Gas flow. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 2-OI-66, Rev. 117 (Attach if not previously provided)

2-AOI-47-3, Rev. 22

Proposed references to be provided to applicants during examination: **CONDENSER A, B, OR C VACUUM LOW (2-9-7B, Window 17)**

Learning Objective: OPL171.202 Obj. 12_ (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.010-12 001 #389	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

389. OPL171.010-12 001

Given that Main Condenser Vacuum is degrading, which **ONE** of the following completes the statements?

A RISING Off-Gas flow on OG FLOW TO 6-HOUR HOLDUP VOLUME, 3-FR-66-20, would be indicative of __(1)__.

To ensure a Manual Scram is inserted prior to automatic action occurring, a Control Room Panel 3-9-6 "Hotwell Pressure" reading of __(2)__ inches Hg Vacuum would be a valid Trigger Value in accordance with OPDP-1, "Conduct of Operations."

A. (1) air in-leakage to the Main Condenser.
(2) 22.5

B. (1) First Stage Steam Jet Air Ejector stalling.
(2) 22.5

C✓ (1) air in-leakage to the Main Condenser.
(2) 25

D. (1) First Stage Steam Jet Air Ejector stalling.
(2) 25

Excerpt from 2-OI-66:

BFN Unit 2	Off-Gas System	2-OI-66 Rev. 0117 Page 12 of 159

3.0 PRECAUTIONS AND LIMITATIONS (continued)

4. [II/C] When notified by Rad Con of confirmed airborne radioactivity in the SBGT building, maximum blocking flow may be obtained by removing Unit 1 dilution fan from service and placing 2A, 2B, 3A, and 3B dilution fans in parallel service. [BFPER 980030]

5. Anytime all stack Dilution Fans are removed from service, a train of SGT is required to be placed in service. A Stack Dilution Fan or Standby Gas Treatment is required to be in operation when any potentially radioactive gas is being discharged out the stack. This will dilute potential hydrogen and prevent backflow into the Standby Gas Treatment System.

G. Following startup, while still at low power, recombiner performance and hydrogen concentration should be closely monitored.

H. If one of the Hydrogen Analyzers is inoperable and the operable Hydrogen Analyzer enters CALIBRATE mode, the unit would not have an operable analyzer while in the CALIBRATE mode.

Placing the operable analyzer in BYPASS prevents the operable analyzer from entering CALIBRATE mode.

I. When hydrogen concentration is suspected of being greater than 4%, Do not take any action that will change off-gas valve positions until after the unit is shutdown, with the following exception: SJAE's may be started following an isolation and alternated, if required, with greater that 4% hydrogen. SJAE's have non-sparking valve seats, and hydrogen flammability lower limit is not a concern in a saturated steam environment. [BFPER 03-016548]

J. Do not operate the mechanical vacuum pumps to purge the main condenser if hydrogen concentration is suspected of being present.

K. Do not operate the mechanical vacuum pumps when reactor power is greater than 5% unless being electrically rotated for Preventive Maintenance.

Excerpts from 2-AOI-47-3:

BFN Unit 2	Loss of Condenser Vacuum	2-AOI-47-3 Rev. 0022 Page 5 of 12

NOTES

1) Rising Off-Gas flow would indicate condenser inleakage if the Off-Gas System is functioning properly. Low Off-Gas flow in conjunction with low condenser vacuum could be indicative of an Off-Gas problem.

2) During operations with valid CONDENSER A, B, OR C VACUUM LOW 2-PA-47-125 alarm, and condensate temperature of 136°F or greater at the inlet of the SJAE (ICS point 2-28), reduced SJAE First Stage performance (stalling) may occur. This condition will cause reduced Off Gas flow and a loss of vacuum/turbine trip.[BFPER 02-016091-000]

BFN Unit 2	Loss of Condenser Vacuum	2-AOI-47-3 Rev. 0022 Page 6 of 12

Date ________

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **IF** ANY EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s). ☐

CAUTION

[NRC/C] Operations outside of the allowable regions shown on the Recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. **REFER TO** 2-GOI-100-12A for required actions and monitoring to be performed during a power reduction. [NCO 940245001]

[2] **MONITOR** Condenser Vacuum (Turb Exhaust) Margin To Trip using 2-XR-002-0026 Channel 7. ☐

[3] **IF** Condenser Vacuum (Turb Exhaust) Margin To Trip as indicated on 2-XR-002-0026, approaches 0 inches Hg, with Reactor power less than 26%, **THEN**

TRIP the main turbine. ☐

[4] **IF** condenser vacuum is lost, **THEN**

OPEN the HOTWELL SAMPLE TO FL DR, 2-DRV-043-1019 (557'@ T-10 C-Line) and CON DEMIN SAMPLE TO FL DR, 2-DRV-043-1020 (557'@ T-6 G-Line), to establish flow through the sample lines. ☐

[5] **REDUCE** reactor power in an attempt to maintain condenser vacuum. ☐

Examination Outline Cross-reference:	Level	RO	SRO
261000 (SF9 SGTS) Standby Gas Treatment	Tier #	2	------
A4.01 (10CFR 55.41.7)	Group #	1	------
Ability to manually operate and/or monitor in the control room: • Off-site release levels: Plant-Specific	K/A #	261000A4.01	
	Importance Rating	3.2*	------

Proposed Question: **# 26**

In order to minimize off-site radioactive release levels, which **ONE** of the following completes the statements below?

Standby Gas Treatment System carbon bed filters ____**(1)**____ designed to remove iodine.

If required, **ALL** three trains of Standby Gas Treatment ____**(2)**____ be manually started from the Unit 3 Control Room.

A. (1) are
(2) can

B. (1) are
(2) can NOT

C. (1) are NOT
(2) can

D. (1) are NOT
(2) can NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with 0-OI-65, Standby Gas Treatment System, in the event that train charcoal filter temperature rises to 150 ºF due to Iodine **adsorption** following a LOCA, decay heat removal mode of operation should be initiated when the train is no longer in service. The carbon (charcoal) bed filters are designed to remove iodine through **adsorption**. For second part, in accordance with 0-OI-65, all three trains of SGT System can be started from the Unit 3 Control Room. Only Unit 3 has push buttons to start all three SGT trains from Panel 9-25.

B INCORRECT**:** First part is correct *(See A)*. Second part is incorrect but plausible in that Unit 1 has control switches to operate two SGT trains ONLY, Unit 2 has control switches to operate one SGT train ONLY. This is often confused among candidates.

C INCORRECT**:** First part is incorrect but plausible in that multiple components/filters exist within the SGT System. The candidate could confuse the HEPA filters that exist both before and after the carbon bed filter as the Iodine removing component. Second part is correct *(See A)*.

D INCORRECT**:** First part is incorrect but plausible *(See C)*. Second part is incorrect but plausible *(See B)*.

RO Level Justification: Tests the candidate's ability to manually operate the Standby Gas Treatment System in the Control Room as it relates to Off-site release levels. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): 0-OI-65, Rev. 55 (Attach if not previously provided)

OPL171.018, Rev. 11

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.018 Obj. 10e (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1102 #45	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2011	

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

Examination Outline Cross-reference:

261000 SGTS

K4.05 (10CFR 55.41.7)

Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following:

- Fission product gas removal

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	261000K4.05	
Importance Rating	2.6	------

Proposed Question: **# 45**

Which ONE of the following completes the statement?

Standby Gas Treatment System __**(1)**__ are designed to remove a **MAXIMUM** of __**(2)**__ of elemental iodine.

A. **(1)** HEPA Filters
(2) 70%

B. **(1)** Carbon Beds
(2) 70%

C. **(1)** HEPA Filters
(2) 99.9%

D. **(1)** Carbon Beds
(2) 99.9%

Proposed Answer: **D**

Excerpts from 0-OI-65:

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0055 Page 8 of 42

3.0 PRECAUTIONS AND LIMITATIONS

A. Upon a secondary containment isolation, the SGT System is designed to maintain a negative 1/4-inch of H_20 vacuum in Secondary Containment with an inleakage flow of 12,000 cfm.

B. [NRC/C] All three trains will remain in operation during an accident to satisfy single failure criteria and to minimize the potential release of radioactivity from the Reactor Building into the Control Building air supply intake ducts. [NRC NCO 88 0193 004]

C. [NER/C] Steps should be taken to minimize dust loading and to prevent paint vapors, petroleum fumes, welding smoke, and other airborne contaminants from reaching the HEPA filters and charcoal adsorbers. Normal ventilation should be in operation for a minimum of two (2) hours after painting, fire, smoke, or chemical release has terminated prior to operating SGT System. [CAQR SQP890064]

D. If the SGT System is run within 16 hours of the completion of painting in the areas specified in MAI-5.3 or MAI-5.7, Control of Volatile Organic Compounds section, a determination is to be made using those procedures as to whether additional actions are required to verify SGT System operability. Exceeding MAI-5.7 limits requires performing 0-SR-3.6.4.3.2(A)(B)(C) to verify SGT can perform its intended function.

E. When all SGT Trains are secured and any evolution has the potential to discharge radioactive effluents through the main stack, one Unit 2 and one Unit 3 Stack Dilution Fan should remain in operation. This requirement provides clean air flow through the dilution cross-tie to SGT ducts. This prevents the potential back flow of radioactive effluents through the SGT duct work.

F. The alignment of SBGT trains to perform the PURGING function cannot be used when the average reactor coolant temperature is above 212°F since a postulated LOCA could impact the ability for the SBGT trains to perform their safety function. If the primary containment purge system is inoperable and the average reactor coolant temperature is less than or equal to 212°F, the standby gas treatment system venting path will provide the required filtration. The standby gas treatment system is **NOT** the normal means for PURGING operations since the vent path from containment is a much more restrictive flowpath (slower) than the purge system.

G. In the event that the train charcoal filter temperature rises to 150°F due to iodine adsorption following a LOCA, decay heat removal mode of operation should be initiated when the train is no longer in service.

H. An open decay heat removal damper in a particular train renders that train inoperable for Secondary Containment purposes.

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0055 Page 9 of 42

3.0 PRECAUTIONS AND LIMITATIONS (continued)

I. If an initiation signal is received from another unit while purging through the SGT System, purging operations should be stopped.

J. The SGT Building may be locked to limit access. Entry can be attained by contacting the Control Room.

K. [NRC/C] This operating instruction is used for three units. Valves, electrical boards, switches, and instruments will have a specific unit prefix if they apply only to that unit. If they exist on all units, there will be no prefix number. Valves common to all units have a "0" prefix. [NRC/C RPT 82-13]

L. Since the SGT System is common to all three units, the Unit Operator on each unit should stay fully aware of the system status and condition at all times. The Unit Operators should communicate to each other any change in system status.

M. When a SGT Train is operated, the Unit Operator should always record in the Narrative Log the start and stop time and the filter bank differential pressure at initiation and prior to shutdown. Also, the elapsed time for each train is to be recorded in 1-SR-2 .

N. Any release of potentially radioactive effluent through SGT System will normally be monitored by the performance of 0-SI-4.8.B.1.a.1.

O. Iodine desorption can be expected to begin at charcoal filter exit air temperatures of approximately 270°F.

P. In situations where decay heat cooling is required for all three trains of SGT System together, combinations of Sections 8.5, 8.7 and 8.9 can be performed to accommodate this situation. The easiest combination would be to perform Section 8.7.1, followed by Section 8.5.2 and Section 8.9.

Excerpts from OPL171.018 Lesson Plan:

OPL171.018, Standby Gas Treatment (SGT) System, Rev.11

6. HEPA filter

   Removes 99.9% of 0.3 micron particles

7. Charcoal Bed (Adsorber Type)

   a) Designed to remove at least 95% of iodine in the form of methyl iodine (CH3I) and 99.9% of elemental iodine upon entering conditions of 70% relative humidity at 190ºF.

   b) Made up of individual rectangular canisters of activated charcoal.

   c) Vertical air flow

8. HEPA afterfilter

   a) Identical to first HEPA filter

   b) Captures any parts of the charcoal bed that may break loose.

   c) Prevents passing charcoal fines through fan and out plant stack.

9. Fan

OPL171.018, Standby Gas Treatment (SGT) System, Rev.11

SGT is designed to maintain a -.25 inches H2O Vacuum pressure inside the secondary containment. Since, during isolation conditions, SGT is the only system exhausting from the secondary containment, failure of SGT would result in containment pressure equalizing with atmospheric pressure. A ground level release of potentially radioactive materials to the environment would occur.	ILT Objective 10c LOR/NLOR Objective 7c
10. Secondary Containment Radiation/Contamination Levels During isolation conditions, SGT is exhausting air from secondary containment. Failure of SGT would allow radioactivity to buildup which could increase secondary containment radiation/contamination levels.	ILT Objective 10d LOR/NLOR Objective 7d
11. Off-Site Release Rates SGT filters out fission products prior to discharging to the stack. Failure of the charcoal filters to perform properly would result in more fission products discharged. This would increase release rates.	ILT Objective 10e LOR/NLOR Objective 7e
12. 120V AC I&C I&C A provides power for SGT A indication and alarms. I&C B provides power for SGT B and C indication and alarms. Failure of these power supplies would result in a loss of SGT indication and alarm functions.	

OPL171.018, Standby Gas Treatment (SGT) System, Rev.11

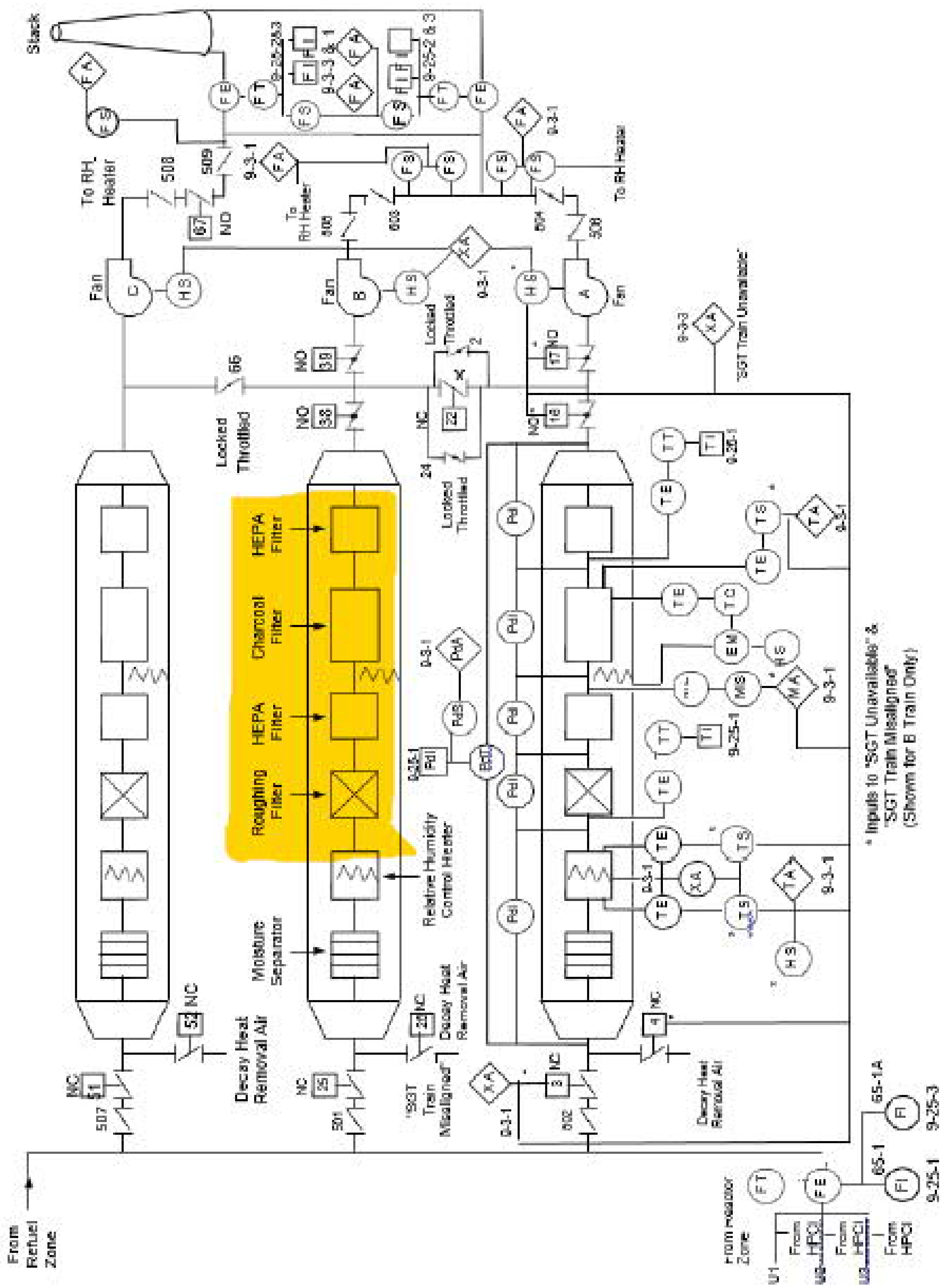


Figure 1: Standby Gas Train Flow Path and Instrumentation

Examination Outline Cross-reference:

295020 (APE 20) Inadvertent Containment Isolation / 5 & 7

AA1.02 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to INADVERTENT CONTAINMENT ISOLATION:

- Drywell ventilation/cooling system

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295020AA1.02	
Importance Rating	3.2	------

Proposed Question: **# 27**

Unit 1 is operating at 100% RTP. The Instrument Mechanics (IMs) are setting up to perform maintenance in the Auxiliary Instrument Room, when a PCIS Group 6 Isolation occurs.

Given the conditions above, which **ONE** of the following completes the statements below?

The Drywell Blowers ___**(1)**___ running.

The Drywell ___**(2)**___ be vented via the **NORMAL** vent path in accordance with 1-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage Into Drywell.

A. (1) are
(2) can

B. (1) are
(2) can NOT

C. (1) are NOT
(2) can

D. (1) are NOT
(2) can NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). The second part is incorrect but plausible in that when a Group 6 PCIS Isolation occurs, the normal vent path outlined in 1-AOI-64-1 cannot be used without the isolation being bypassed. Bypassing the isolation is not allowed when using 1-AOI-64-1, however Group 6 isolation signals can be bypassed using 1-EOI-Appendix-8E, 12, or 13, if necessary, in accordance with the EOIs.

B **CORRECT**: *(See attached)* Drywell Blowers are not tripped or isolated during a Group 6 PCIS Isolation. For second part, the normal Drywell vent path is not available using 1-AOI-64-1, as bypassing the isolation signals is not allowed in the AOI.

C INCORRECT: The first part is incorrect but plausible in that a PCIS Group 6 isolation isolates ventilation systems in the event of a leak in the Drywell or a radiation issue in Secondary Containment. Drywell Blowers are part of the ventilation system, so it is reasonable that they would trip with the ventilation isolation. The isolation affects a large number of ventilation system components. The second part is incorrect but plausible (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the effect of a Primary Containment Isolation signal on Drywell Cooling. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): OPL171.017, Rev. 21 (Attach if not previously provided)
1-AOI-64-1, Rev. 2
1-EOI-1A, Rev. 2
1-AOI-64-2D, Rev. 20
1-EOI-Appendix-8E, Rev. 0

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.071, Obj.4 (As available)

Question Source: Bank #		
Modified Bank #		(Note changes or attach parent)
New	X	
Question History: Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Excerpt from OPL171.017 Lesson Plan:

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 21

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
b) A brief description of available Isolation bypasses is: (1) Group 1 (a) RPV low low-low level (-122" Level 1), is bypassed by the installation of jumpers per EOI Appendix 8A. All Isolations bypassed by jumper installation per EOI Appendix 11H.	ILT- 3c LOR- 3c 2-730E927-8,9 DCN72701 replaces jumpers installed per EOI APPX 8A with four Keylocks on 9-4. U1/U2 completed, U3 spring 2020
(2) Group 2 (a) The RPV low level (+2" or Level 3) and Drywell High Pressure (2.45 psig) Isolation signals to the PSC Head Tank Pump Isolation valves (FCV-75-57, 58) are bypassed by installing jumpers per EOI Appendix 7G. This is done to allow the PSC Head Tank Pumps to be used as an alternate injection system.	ILT- 3c LOR- 3c
(3) Group 4 (a) The HPCI Low Reactor Pressure Isolation can be bypassed by de-terminating a wire per EOI Appendix 16C. High temperature Isolation can be bypassed by booting contacts per EOI Appendix 16K.	ILT- 3c LOR- 3c
(4) Group 5 (a) The RCIC Low Reactor Pressure Isolation can be bypassed by de-terminating a wire per EOI Appendix 16A. High temperature Isolation can be bypassed by booting contacts per EOI Appendix 16K.	ILT- 3c LOR- 3c
(5) Group 6 (a) The Group 6 RPV Low Level (+2" or Level 3)) and Drywell High Pressure (2.45 psig) Isolations can be bypassed by installing jumpers per EOI Appendix 8E.	ILT- 3c LOR- 3c 730E927RF sheet 16, 17,18
(b) The DW/SUPPR CHBR TRAIN A(B) VENT Keylock Switches (HS-84-35/36) and the TRAIN A(B) VENT TO SGT SYSTEM Keylock Switches (HS-84-20/19) Panels 9-54/55 are used to bypass all Group 6 Isolations as necessary to allow containment venting through SGT (or through large vent path) per EOI Appendix 12 and 13. Cannot vent with loss of RPS A.	ILT- 3c LOR- 3c Normally for refueling outages

Excerpt from 1-EOI-Appendix-8E:

BFN UNIT 1	BYPASSING GROUP 6 LOW RPV LEVEL AND HIGH DRYWELL PRESSURE ISOLATION INTERLOCKS	1-EOI APPENDIX-8E Rev. 0 Page 1 of 2

LOCATION:	Unit 1 Auxiliary Instrument Room	
ATTACHMENTS:	1. Tools and Equipment	(✓)

1. **REFER** to Attachment 1 and **OBTAIN** two banana jack jumpers from the EOI Equipment Storage Box. ______

2. **BYPASS** Group 6 Low RPV Level and High Drywell Pressure Isolation Interlocks as follows:

   a. **LOCATE** terminal strip BB in 1-PNLA-009-0015, Bay 3, Rear. ______

   b. **JUMPER** BB-22 to BB-23, 1-PNLA-009-0015. ______

   c. **LOCATE** terminal strip DD in 1-PNLA-009-0015, Bay 1, Rear. ______

   d. **JUMPER** DD-22 to DD-23, 1-PNLA-009-0015. ______

3. **NOTIFY** Unit Operator that Group 6 RPV Low Level and High Drywell Pressure Isolation Interlocks are bypassed. ______

Excerpt from 1-AOI-64-1:

BFN Unit 1	Drywell Pressure and/or Temperature High, or Excessive Leakage Into Drywell	1-AOI-64-1 Rev. 0002 Page 7 of 12

4.2.2 Drywell Pressure is High

[1] **CHECK** Drywell pressure using multiple indications. ☐

[2] **ALIGN** and **START** additional Drywell coolers and fans as necessary. **REFER TO** 1-OI-64. ☐

WARNING

Stack release rates exceeding 1.4 X 10^7 µci/sec, or a SI-4.8.B.1.a.1 release fraction above one will result in ODCM release limits being exceeded.

[3] **VENT** Drywell as follows: ☐

[3.1] **CLOSE** SUPPR CHBR INBD ISOLATION VLV, 1-HS-64-34 (Panel 1-9-3). ☐

[3.2] **VERIFY OPEN**, DRYWELL INBD ISOLATION VLV, 1-HS-64-31 (Panel 1-9-3). ☐

[3.3] **VERIFY** TRAIN A VENT TO SGTS, 1-FIC-84-20 is in AUTO and SET at 100 scfm (Panel 1-9-55). ☐

[3.4] **VERIFY** Running, required Standby Gas Treatment Fan(s) SGTS TRAIN A(B,C) OPERATING, (XI-65-18(40,69)B/1), (Panel 1-9-20). ☐

[3.5] **IF** required, **THEN**

REQUEST Unit 1 Operator to START Standby Gas Treatment Fan(s) SGTS TRAIN A(B) FAN, 0-HS-65-18(40)A/1.(Panel 1-9-25) (Otherwise **N/A**) ☐

NOTE

If 1-FCV-84-20 closes after placing 1-HS-64-35 to open, the valve's closure signal must be reset and 1-HS-64-35 must be returned to the OPEN position in order for 1-FCV-84-20 to RE-OPEN.

[3.6] **IF** required, **THEN**

RECORD venting data in 1-SI-4.7.A.2.a (Otherwise **N/A**) ☐

Excerpts from 1-AOI-64-2D:

BFN Unit 1	Group 6 Ventilation System Isolation	1-AOI-64-2D Rev. 0020 Page 4 of 17

1.0 PURPOSE

This procedure provides symptoms, automatic actions, and operator actions for a Group 6 Ventilation System Isolation.

2.0 SYMPTOMS

NOTES

1) PCIS Group 6 Isolation is initiated by any one of the following signals:
   - Reactor vessel water level at +2.0"
   - Drywell pressure at 2.45 psig
   - Reactor zone exhaust radiation at 72 mr/hr
   - Refuel zone exhaust radiation at 72 mr/hr

2) High Refuel Zone exhaust radiation causes only the automatic actions listed in Section 3.1.

3) Refuel Zone isolation due to Group 6 isolation initiated on Unit 2 or Unit 3.

A. Any one or more of the following annunciators in ALARM:

1. REACTOR ZONE EXHAUST RADIATION HIGH 1-RA-90-142A (1-XA-55-3A, Window 21)
2. REFUELING ZONE EXHAUST RADIATION MONITOR DOWNSCALE 1-RA-90-140B (1-XA-55-3A, Window 28)
3. REFUELING ZONE EXHAUST RADIATION HIGH 1-RA-90-140A (1-XA-55-3A, Window 34)
4. RX ZONE EXH RADIATION MONITOR DNSC 1-RA-90-142B (1-XA-55-3A, Window 35)
5. REAC BLDG VENTILATION ABNORMAL (1-XA-55-3D, Window 3)
6. REAC VESSEL LOW LEVEL HALF SCRAM at +2 (1-XA-55-4A, Window 2)
7. REACTOR ZONE DIFFERENTIAL PRESSURE LOW (1-XA-55-3D, Window 32)
8. DRYWELL HIGH PRESSURE HALF SCRAM (1-XA-55-4A, Window 8)
9. ANA-76-89 DRYWELL/SUPP CHAMBER H_2O_2 ANALYZER FAILURE (1-XA-55-7C, Window 22)

BFN Unit 1	Group 6 Ventilation System Isolation	1-AOI-64-2D Rev. 0020 Page 7 of 17

3.1 Refueling Zone Isolation (continued)

B. The following valves CLOSE:

1. 1-FCV-76-17, PRI CTMT N_2 MAKEUP OUTBD ISOL VALVE
2. 1-FCV-76-18, DRYWELL N_2 MAKEUP INBD ISOL VALVE
3. 1-FCV-76-19, SUPPRESSION CHAMBER N_2 INBD ISOL VALVE
4. 1-FCV-76-24, PRI CONTAINMENT N_2 PURGE OUTBD ISOL VALVE
5. 1-FCV-64-17, DW/SUPPR CHBR AIR PURGE ISOL VLV
6. 1-FCV-64-18, DRYWELL ATM SUPPLY INBD ISOLATION VLV
7. 1-FCV-64-19, SUPPR CHBR ATM SPLY INBD ISOLATION VLV
8. 1-FCV-64-29, DRYWELL VENT INBD ISOL VALVE
9. 1-FCV-64-30, DRYWELL VENT OUTBD ISOLATION VLV
10. 1-FCV-64-31, DRYWELL INBD ISOLATION VLV
11. 1-FCV-64-32, SUPPR CHBR VENT INBD ISOL VALVE
12. 1-FCV-64-33, SUPPR CHBR VENT OUTBD ISOLATION VLV
13. 1-FCV-64-34, SUPPR CHBR INBD ISOLATION VLV
14. 1-FCV-84-19, TRAIN B VENT TO SGTS
15. 1-FCV-84-20, TRAIN A VENT TO SGTS
16. 1-FCV-64-140, DRYWELL DP COMP DISCH VLV
17. 1-FCV-064-0139, DRYWELL DP COMP SUCTION VLV

C. Standby Gas Treatment System starts

D. 1-FCO-64-44, RFF SGT SUCT DMPR OPR, OPENS

E. 3-FCO-64-44, RFF SGT SUCT DMPR OPR, OPENS

F. 1-FCO-64-45, RFF SGT SUCT DMPR OPR, OPENS

G. CREV Units start

Examination Outline Cross-reference:	Level	RO	SRO
295032 (EPE 9) High Secondary Containment Area Temperature / 5	Tier #	1	------
EK3.01 (10CFR 55.41.5)	Group #	2	------
Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: • Emergency/normal depressurization	K/A #	295032EK3.01	
	Importance Rating	3.5	------

Proposed Question: **# 28**

Which **ONE** of the following completes the statement below in accordance with EOI-3, Secondary Containment Control Program Manual?

Emergency Depressurization is required when any Secondary Containment Temperature exceeds its MAXIMUM SAFE value in ____**(1)**____ area(s) to place the Primary System in its lowest energy state by rejecting heat to the ____**(2)**____.

A. (1) **ONLY** one
(2) Main Condenser

B. (1) **ONLY** one
(2) Suppression Pool

C. (1) two **OR** more
(2) Main Condenser

D. (1) two **OR** more
(2) Suppression Pool

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate confuses the Max Safe criteria to Emergency Depressurize. Second part is incorrect but plausible if the candidate confuses the reasons requiring Emergency Depressurization using the SRVs versus Rapidly Depressurizing using the Main Turbine Bypass Valves which reject heat to the Main Condenser.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is correct *(See D).*

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT:** *(See attached)* In accordance with EOI-3, Secondary Containment Control Bases, if Secondary Containment parameters continue to increase and exceed their Max Safe values in two or more areas, the Reactor Pressure Vessel (RPV) must be Emergency Depressurized. For the second part, RPV Emergency Depressurization places the Primary System in its lowest energy state and rejects heat to the Suppression Pool in preferences to outside the containment. This is performed using the Main Steam Relief Valves.

RO Level Justification: Tests candidate's knowledge of the reasons to Emergency Despressurize as it relates to High Secondary Containment Area Temperature. This question is rated as Memory due to the fact that it requires the strict recall of facts.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): EOIPM 0-V-E, Rev. 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.204 Obj. 1 (As available)

Question Source:

Bank #		
Modified Bank #	Quad Cities #26	(Note changes or attach parent)
New		

Question History: Last NRC Exam 2009

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

26 **ID: QDC.ILT.15518** **Points: 1.00**

What is the purpose of performing a QGA 500-1 "RPV Blowdown" when directed by QGA 300 "Secondary Containment Control"?

QGA 500-1 "RPV Blowdown" is performed in order to…

A. (1) facilitate RPV level restoration.
(2) place the primary system in its lowest energy state.
(3) reduce the flow from the break into the Reactor Building.

B. (1) place the primary system in its lowest energy state.
(2) reduce the flow from the break into the Reactor Building.
(3) reject heat to the Torus in preference to the Reactor Building.

C. (1) facilitate RPV level restoration.
(2) place the primary system in its lowest energy state.
(3) reject heat to the Torus in preference to the Reactor Building.

D. (1) facilitate RPV level restoration.
(2) reduce the flow from the break into the Reactor Building.
(3) reject heat to the Torus in preference to the Reactor Building.

Answer: B

Excerpt from EOIPM 0-V-E:

BFN Unit 0	EOI-3 Secondary Containment Control Bases	EOIPM Section 0-V-E Rev. 0003 Page 35 of 47

1.0 EOI-3, SECONDARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: SC-9, SC-10

If secondary containment parameters continue to increase and exceed their Max Safe values in two or more areas, the RPV must be depressurized. RPV depressurization places the primary system in its lowest possible energy state, rejects heat to the suppression pool in preference to outside the containment, and reduces the driving head and flow of primary systems that are unisolated and discharging into the secondary containment.

The criteria of “2 or more areas” specified in this step identifies the rise in the secondary containment parameter value as a wide-spread problem which may pose a direct and immediate threat to plant equipment and to personnel both on and off the site.

One parameter (e.g., radiation) above its Max Safe value in one area and a different parameter (e.g., temperature or water level) above its Max Safe value in the same or another area is not a condition which requires emergency RPV depressurization. A combination of parameters exceeding Max Safe values in one area does not necessarily indicate that control of a given parameter cannot be maintained or that previous actions have not been effective in confining the trouble to one area. Expanding the application to encompass multiple parameters might lead to depressurization of the RPV when such action is not, in needed.

Examination Outline Cross-reference:	Level	RO	SRO
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5	Tier #	1	------
EK2.01 (10CFR 55.41.7)	Group #	2	------
Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL and the following:	K/A #	295036EK2.01	
• Secondary containment equipment and floor drain system	Importance Rating	3.1	------

Proposed Question: **# 29**

Which **ONE** of the following completes the statements below in accordance with EOI-3, Secondary Containment Control?

A Reactor Building Floor Drain Sump Water Level of 67 inches ____**(1)**____ require EOI-3 entry.

Given this condition, an alarm will **FIRST** be indicated in the ____**(2)**____ Control Room.

A. (1) does
(2) Main

B. (1) does
(2) Radwaste

C. (1) does NOT
(2) Main

D. (1) does NOT
(2) Radwaste

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). The second part is incorrect but plausible if the candidate confuses the location of the FIRST alarm indication. The Main Control Room only receives alarms once there is greater than or equal to 2 inches of water on the **FLOOR** in respective Reactor Building locations indicating Secondary Containment **Area** Water Levels rising beyond sump capability.

B **CORRECT**: *(See attached)* In accordance with EOI-3, Secondary Containment Control, a Floor Drain Sump Water Level above 66 inches meets the entry condition. For second part, in accordance with 0-ARP-25-17A, the Reactor Building **Floor** Drain Sump 'A' or 'B' Abnormal level alarms at **66 inches** from the bottom of the sump. Alarm Panel 25-17A is ONLY located in the Radwaste Control Room, therefore the ARP directs the RW Operator to immediately notify the affected Unit of a sealed in alarm requiring an EOI-3 entry condition. If Secondary Containment Water Level is rising beyond sump capability, from the Main Control Room, the SUPPRESSION CHAMBER ROOM FLOOD LEVEL HIGH (1/2/3-9-4C, Window 3) and HPCI ROOM FLOOD LEVEL HIGH (1/2/3-9-4C, Window 10) alarms once there is greater than or equal to 2 inches of water on the floor indicating Secondary Containment **Area** Water Levels.

C INCORRECT: The first part is incorrect but plausible in that EOI-3 has nine different entry conditions with two of nine associated with Secondary Containment Water Level. Any Secondary Containment **Area** Water Level above 2 inches is the other EOI-3 entry condition. The second part is incorrect but plausible (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the interrelations between both Secondary Containment High Sump and Area Water Levels as it relates to EOI-3, Secondary Containment Control. This question is rated as Memory due to the fact that it requires the strict recall of facts.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 2-EOI-3, Rev. 17 (Attach if not previously provided)
2-ARP-9-4C, Rev. 35

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.204 Obj. 2 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1909 #34	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

ILT 1909 Written Exam

34. Which **ONE** of the following completes the statements below?

Entry into EOI-3, Secondary Containment Control, is required when **ANY** Secondary Containment Area Water Level is above ___**(1)**___.

In accordance with EOI-3, ___**(2)**___ is required when a Primary System is discharging into Secondary Containment and Secondary Containment Water Level exceeds Max Safe in two or more areas.

A. **(1)** 2 inches
(2) a normal Reactor Shutdown

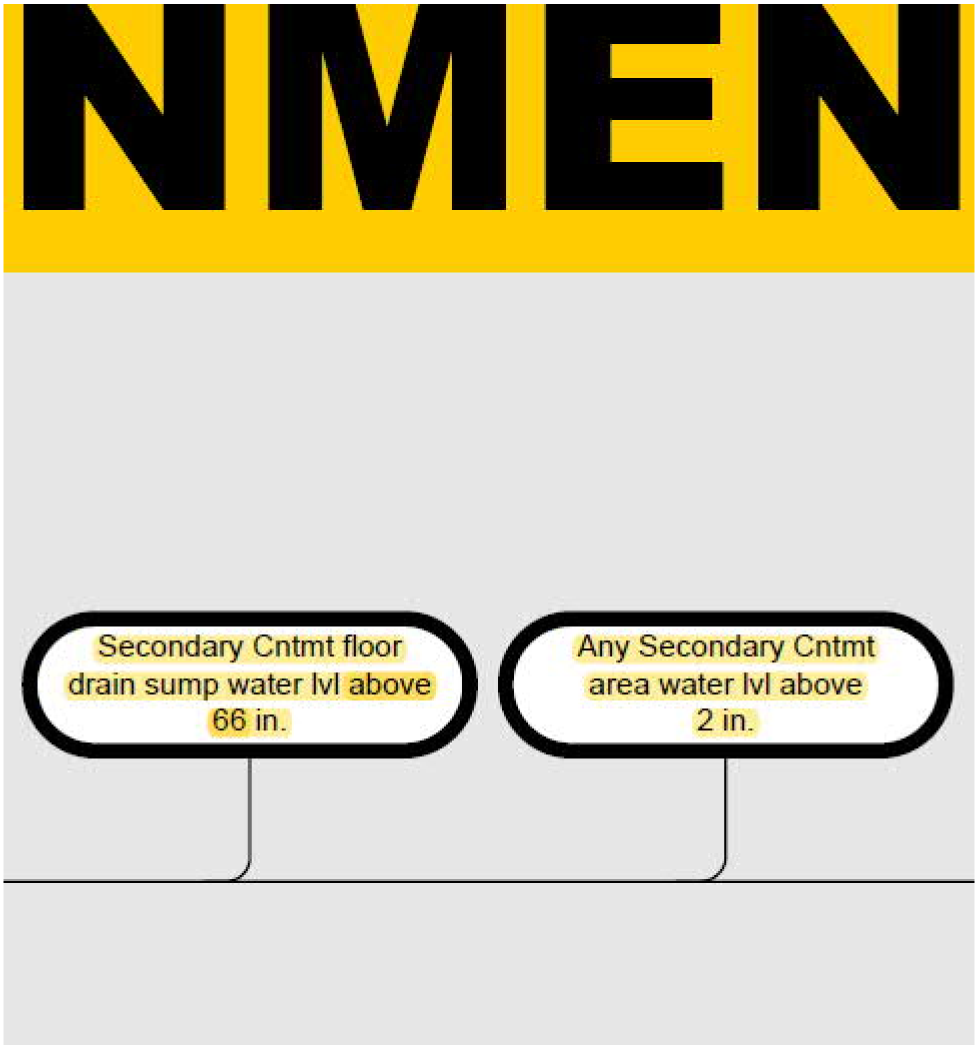
B. **(1)** 2 inches
(2) Emergency Depressurization

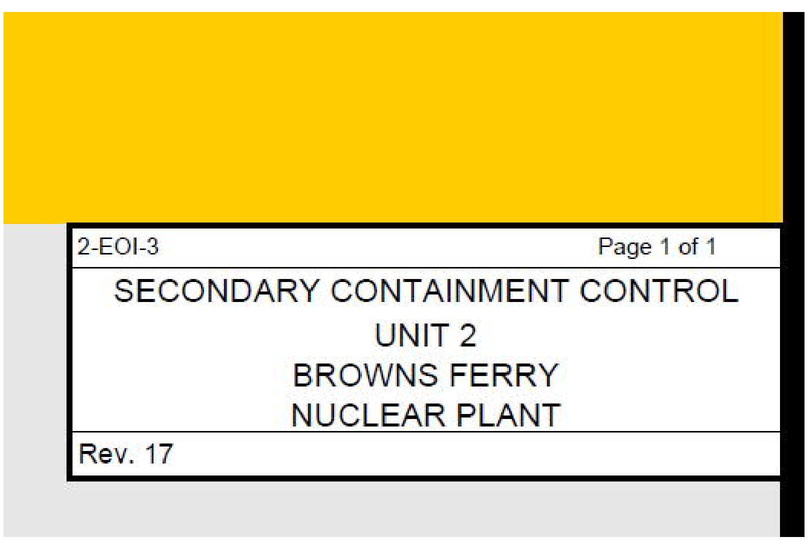
C. **(1)** 66 inches
(2) a normal Reactor Shutdown

D. **(1)** 66 inches
(2) Emergency Depressurization

Correct Answer: **B**

Excerpt from 2-EOI-3:




2-EOI-3 Page 1 of 1

SECONDARY CONTAINMENT CONTROL

UNIT 2

BROWNS FERRY

NUCLEAR PLANT

Rev. 17

Excerpts from 2-ARP-9-4C: Supports Distractors B(2), D(2)

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0035 Page 7 of 44

SUPPR CHMBR RM FLOOD LEVEL HIGH 2-LA-77-25F 3

(Page 1 of 1)

Sensor/Trip Point:

2-LS-77-25F ≥2 inches of water on the floor

Sensor Location: Sensor is located near the floor of the Suppression Chamber room, Column R-11 N-LINE

Probable Cause: Greater than two inches of water on the floor.

Automatic Action: None

Operator Action:

A. **DISPATCH** personnel to visually check the suppression chamber room.

B. IF alarm is valid, **THEN**
PERFORM the following:

- **CHECK** the floor drain sump pumps running.
- **CHECK** the floor drains for proper drainage.
- **IF** possible, **THEN**
  DETERMINE the source of the leak and the leak rate.
- **ENTER** 2-EOI-3 FLOWCHART.

NOTE
The floor drain and equipment drain sump pumps may need to the locked out to prevent Radwaste flooding.

- **NOTIFY** Radwaste Operator to monitor drain collector tank and waste collector tank levels.
- **NOTIFY** Radiation Protection.

References: 0-47E610-77-1 47W600-8 45N620-4

FSAR Sections 13.6.2 and F.7.15

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0035 Page 17 of 44

HPCI ROOM FLOOD LEVEL HIGH 2-LA-77-25E 10	Sensor/Trip Point:	
(Page 1 of 1)	2-LS-77-25E	≥ 2 inches of water on the floor

Sensor Location: Sensor is located near the floor of the HPCI room.

Probable Cause:
A. Greater than two inches of water on the floor.
B. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **DISPATCH** personnel to visually check the HPCI room.
B. **IF** alarm is valid, **THEN** **PERFORM** the following:
- **CHECK** the floor drain sump pumps running.
- **CHECK** the floor drains for proper drainage.
- **IF** possible, **THEN** **DETERMINE** the source of the leak and the leak rate.
- **ENTER** 2-EOI-3 FLOWCHART.

NOTE
The floor drain and equipment drain sump pumps may need to the locked out to prevent Radwaste flooding.

- **NOTIFY** Radwaste Operator to monitor drain collector tank and waste collector tank levels.
- **NOTIFY** Radiation Protection.

References: 0-47E610-77-1 45N620-4
FSAR Sections 13.6.2 and F.7.15

Examination Outline Cross-reference:	Level	RO	SRO
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode	Tier #	2	------
A3.01 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including:	K/A #	203000A3.01	
• Valve operation	Importance Rating	3.8*	------

Proposed Question: **# 30**

Unit 1 was operating at 100% RTP when a LOCA occurred resulting in the following conditions:

- Reactor Pressure is 400 psig
- Reactor Water Level is (-) 140 inches and slowly lowering

Given the conditions above, which **ONE** of the following completes the statement below?

1-FCV-74-52, RHR SYSTEM I LPCI OUTBOARD INJECTION VALVE is ___**(1)**___, and
1-FCV-74-7, RHR SYSTEM I MINIMUM FLOW VALVE is ___**(2)**___.

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See Attached)* Reactor Water Level is below the Common Accident Signal of (-) 122 inches, which causes RHR pumps to start for injection. Reactor Pressure is below the 450 psig RHR Injection Valve interlock; therefore RHR Pumps are running and the RHR System Injection Valves are open. For second part, Reactor Pressure is above the shutoff head value of 320 psig for RHR Pumps. Therefore, RHR System Flow is below 5800 gpm, so the Minimum Flow Valve will also be open.

B INCORRECT: The first part is correct (*See* A). The second part is incorrect but plausible in that since the RHR Pumps are running and the Injection Valves are open, the candidate may assume that RHR is injecting to the Reactor, which would cause the Minimum Flow Valve to close at proper flow rate.

C INCORRECT**:** The first part is incorrect but plausible in that in accordance with 1-EOI-1, RPV Control, the shutoff head for RHR Pumps is 320 psig. However, the candidate may confuse the shutoff head with the Injection Valve Interlock, and since Reactor Pressure is above the shutoff head pressure believe the Injection Valve has not opened yet. The second part is correct (*See A*).

D INCORRECT: Incorrect but plausible (*See C*). The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of how the RHR System Valves operate with an accident signal present and Reactor Pressure above the shutoff head for the RHR System. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

Technical Reference(s): 1-OI-74, Rev.108 (Attach if not previously provided)
1-EOI-1, Rev.6
1-ARP-9-3C, Rev.29

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044 Obj. 4d, 4e (As available)

Question Source:

Bank #	ILT Exam Bank OPL171.044-18 001, #1498	
Modified Bank #		(Note changes or attach parent)
New		
Last NRC Exam		

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

1498. OPL171.044-18 001

Unit 1 was operating at 100% Reactor Power when a LOCA occurred resulting in the following:

- Reactor Pressure is 405 psig
- Reactor Level is (-) 140 inches

Which **ONE** of the following completes the statement?

The RHR SYS I LPCI OUTBD INJECT VALVE, 1-FCV-74-52, is **__(1)__** **AND** the RHR SYS I MIN FLOW VALVE, 1-FCV-74-7, is **__(2)__**.

A. **(1)** OPEN
(2) CLOSED

B. **(1)** CLOSED
(2) CLOSED

C✓ **(1)** OPEN
(2) OPEN

D. **(1)** CLOSED
(2) OPEN

Excerpt from 1-ARP-9-3C:

BFN Unit 1	Panel 1-9-3 1-XA-55-3C	1-ARP-9-3C Rev. 0029 Page 34 of 41

RX WTR LVL
LOW LOW LOW
ECCS/ESF INIT
1-LA-3-58A
28

(Page 1 of 1)

Sensor/Trip Point:

1-LIS-003-0058A	≤ -122 inches
1-LIS-003-0058B	RPV Low-Low-Low level (Level 1)
1-LIS-003-0058C	
1-LIS-003-0058D	

Sensor Location: 1-LIS-003-0058A&B 1-PNLA-009-0081 | 1-LIS-003-0058C&D 1-PNLA-009-0082

Probable Cause:
A. Reactor Water Low Level.
B. SI/SR in progress.

Automatic Action: (One out of two taken twice logic.)

A. The following receive Auto Start Signals:
- Core Spray System
- RHR (LPCI mode) System
- Diesel Generators
- RHRSW (EECW) Pump

B. ADS Blowdown Logic Input

Operator Action:
A. **CHECK** RPV Level using multiple indications.
B. **IF** alarm is valid, **THEN** **REFER TO** EOIs.
C. **IF** alarm is **NOT** valid, **THEN** **RETURN** auto start systems to Standby Readiness per respective OI's.
D. **EVALUATE** equipment associated with this alarm that is out of service to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References:

1-45E620-2-1	1-47E610-3-1	1-47W600-58
GE 730E930-3	GE 0-730E930 -4, and -9	Tech Spec 3.3.5.1

Excerpt from 1-EOI-1:

Table L-1 Preferred Injection Systems		
SOURCES	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available 2 3 6	5C, 20M	1200 psig
HPCI with CST suction if available 2 6 7	5D, 20N	1200 psig
CNDS	6A	480 psig
CS 2	6D, 6E	330 psig
LPCI 2	6B, 6C	320 psig

Excerpts from 1-OI-74:

3.2 RHR Pumps

A. To minimize system vibration, RHR Pump operation should be minimized below 7,000 gpm or above 10,000 gpm, or for more than 3 minutes at minimum flow.

B. [NRC/C] The RHR Pumps are considered to be operable without the seal cooler under the following conditions:

1. Always operable in the LPCI and Containment Cooling Mode.
2. During Shutdown Cooling, operable up to a suction temperature of 215°F.
3. Operable for an emergency with suction temperature between 215°F and 400°F. Operation in this condition for more than two days will require an inspection of the seal surfaces. [NRC LER 296/83047 R1]

C. To prevent Recirculation Pump operation at shutoff head, RHR pumps must not be operated in parallel with Recirculation Pumps.

3.3 LPCI

A. LPCI will initiate on any of the following signals:

1. Reactor Vessel low-low-low water level (-122 inches)(Level 1).
2. High Drywell Pressure (2.45 psig) with low Reactor Vessel Pressure (450 psig).

B. Manually stopping an RHR Pump after LPCI Initiation disables automatic restart of that pump until the initiation signal is reset. The affected RHR Pump can still be started manually.

C. Upon an automatic LPCI Initiation with normal power available, RHR Pump 1A will start immediately, THEN 1B, 1C, 1D sequentially start at 7 second intervals. Otherwise, all RHR Pumps start immediately once Diesel power is available (and normal power unavailable).

D. As soon as practicable after an RHR Pump(s) auto start, the corresponding Control Room hand switch should be placed in normal-after-start position to ensure the hand switch disagreement light(s) and pump tripped annunciator(s) function as designed.

3.6 Interlocks

A. The RHR System is equipped with pump and valve interlocks to assure the following:

1. All RHR Pump flow is directed to the LPCI Injection path during ECCS initiation.

2. Protection of low pressure piping from high reactor pressures.

3. A pump suction path is fully open prior to pump start.

4. Suction Path Interlocks:

   a. An RHR Pump will not start or will trip, if running, unless its corresponding torus suction valve is open or the SDC suction valve and the SDC suction supply valves, 1-FCV-74-47 and 48, are open.

   b. The Torus Suction valves cannot be opened unless the corresponding pumps SDC suction valve is fully closed.

   c. The SDC suction valves cannot be opened unless the corresponding pumps Torus Suction valve is fully closed.

5. RHR Minimum Flow Valve Interlocks:

   a. The RHR minimum flow valves auto close if both pumps in the corresponding loop are off and either pump's SDC suction valve is open.

   b. The minimum flow valves open and close on a low flow of 5800 gpm. The tolerance of the flow switch may allow the setpoint of the min flow valve to be from 4500 gpm to 7000 gpm. Operation outside of this expanded range should be investigated. The analytical limit as listed in design criteria BFN-50-7074 is 11000 gpm for min flow valve closure. [PER-238791-01, 06]

   c. Placing the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), in INHIBIT, will simulate a high flow and the minimum flow valve will remain closed regardless of flow.

BFN Unit 1	Residual Heat Removal System	1-OI-74 Rev. 0108 Page 20 of 434

3.6 Interlocks (continued)

d. Opening RHR SYSTEM I(II)MIN FLOW VALVE, 1-HS-74-7A(30A), from 1-PNL-9-3, with the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), in INHIBIT, will cause the minimum flow valves to travel full open and full close unless the RHR SYSTEM I(II) MIN FLOW VALVE, 1-HS-74-7A(30A) is placed in closed position to break the OPEN seal in contacts.

e. [II/C] Local operation of the RHR minimum flow valves will bypass the intended function of the Minimum Flow Inhibit switch and can cause inadvertent drainage of the Reactor vessel to the Suppression Pool. [BFPER941099]

f. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), with the respective RHR Loop in standby readiness, can cause inadvertent damage to that loop RHR Pump(s) should RHR Pump(s) auto start. [BFA-890790003P]

g. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), with the respective RHR loop in Shutdown Cooling, can cause inadvertent drainage of the Reactor Vessel to the Suppression Pool. [BFA-890790003P]

6. The RHR Outboard LPCI Injection valves, 1-FCV-74-52(66), have throttling capability. They receive an auto open signal in the presence of a LPCI initiation signal when Reactor Pressure is ≤450 psig and are interlocked open under these conditions until the appropriate LPCI SYS I(II) OUTBD INJ VLV BYPASS SEL keylock switch, 1-HS-74-155A(155B) is placed in the BYPASS position. Additionally these valves are interlocked to prevent opening when reactor pressure is greater than 450 psig if its in-line companion valve 1-FCV-74-53(67) is not fully closed.

Examination Outline Cross-reference:	Level	RO	SRO
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection **K5.06** (10CFR 55.41.5)	Tier #	2	------
Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM:	Group #	1	------
	K/A #	206000K5.06	
• Turbine speed measurement: BWR-2,3,4	Importance Rating	2.6*	------

Proposed Question: **# 31**

Which **ONE** of the following completes the statements below?

In accordance with 3-OI-73, High Pressure Coolant Injection System (HPCI), operating the HPCI Turbine below a **MINIMUM** of ___**(1)**___ should be minimized to prevent system damage.

If HPCI trips on an overspeed condition with a valid initiation signal, the overspeed trip ___**(2)**___.

A. (1) 2100 rpm
(2) will automatically reset

B. (1) 2100 rpm
(2) must be manually reset

C. (1) 2400 rpm
(2) will automatically reset

D. (1) 2400 rpm
(2) must be manually reset

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that the minimum allowable speed for RCIC is 2100 RPM. HPCI and RCIC are often confused because of system similarity. The second part is correct (*See C*).

B INCORRECT: The first part is incorrect but plausible *(See A).* The second part is incorrect in that while HPCI will automatically reset the overspeed trip (*See C*), RCIC requires operator action to manually reset its overspeed trip. HPCI and RCIC are often confused because of system similarity.

C **CORRECT**: *(See attached)* In accordance with 3-OI-73, High Pressure Coolant Injection System, HPCI Turbine operation below 2400 RPM should be minimized to ensure proper oil flow, reduce vibration, and prevent possible water hammer. Additionally, the EOI Appendices for HPCI mention this limitation in the notes. For second part, in accordance with 3-OI-73, if HPCI were to overspeed 3-FCV-73-18, HPCI TURBINE STOP VALVE, will close by spring pressure, the piston of the hydraulic trip resets. The device automatically resets when HPCI's speed is between 2500-3000 RPM and HPCI will inject to the Reactor again.

D INCORRECT: The first part is correct (*See C)*. The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability and knowledge pertaining to the operational speed limitations for the HPCI Turbine to prevent system damage and how the HPCI System responds to an overspeed condition. This question is rated as memory due to strictly recalling facts concerning the limitations on HPCI Turbine Speed and how the overspeed device resets following an overspeed condition.

Technical Reference(s): 3-OI-73, Rev.63 (Attach if not previously provided)
3-OI-71, Rev.63

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.042 Obj. 8,10 (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.042-10 008 #1330	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT

for ILT Exam Bank 08 22 2018

1330. OPL171.042-10 008

Which one of the following completes the statements below in accordance with 1-EOI APPENDIX-5D INJECTION SYSTEM LINEUP HPCI?

Operating HPCI Turbine at ___(1)___ rpm or less may result in unstable system operation and equipment damage.

Operating HPCI Turbine with suction temperatures above a **maximum** of ___(2)___ °F may result in equipment damage.

A. (1) 2100
(2) 140

B. (1) 2100
(2) 160

C✓ (1) 2400
(2) 140

D. (1) 2400
(2) 160

Excerpts from 3-OI-73:

BFN Unit 3	High Pressure Coolant Injection System	3-OI-73 Rev. 0063 Page 10 of 99

3.0 PRECAUTIONS AND LIMITATIONS (continued)

E. When any of the following signals are received, HPCI SUPPR POOL OUTBD SUCT VLV, 3-FCV-073-0027, and HPCI SUPPR POOL INBD SUCT VALVE, 3-FCV-073-0026 automatically open, unless a HPCI isolation signal is present.

1. Suppression Pool Level High at +5.25 in.

2. HPCI Pump Suction Condensate Header Level Low at approximately 7000 gallons (El. 552'6" on 3-LS-073-0056A and 0056B)

F. When HPCI SUPPR POOL OUTBD SUCT VLV 3-FCV-073-0027 and HPCI SUPPR POOL INBD SUCT VLV 3-FCV-073-0026 are fully open, HPCI CST SUCTION VALVE 3-FCV-073-0040 automatically closes.

G. When either HPCI SUPPR POOL OUTBD SUCT VLV, 3-FCV-073-0027, or HPCI SUPPR POOL INBD SUCT VLV, 3-FCV-073-0026, is FULL OPEN, the HPCI/RCIC CST TEST VLV, 3-FCV-073-0036, and HPCI PUMP CST TEST VLV, 3-FCV-073-0035, close.

H. When the HPCI TURBINE STEAM SUPPLY VALVE, 3-FCV-073-0016, is opened, the following valves close:

1. HPCI HOTWELL PUMP INBD ISOL VLV, 3-FCV-073-0017A

2. HPCI HOTWELL PUMP OUTBD ISOL VLV, 3-FCV-073-0017B

3. HPCI STEAM LINE INBD DRAIN VLV, 3-FCV-073-0006A

4. HPCI STEAM LINE OUTBD DRAIN VLV, 3-FCV-073-0006B

I. The HPCI PUMP MIN FLOW VALVE, 3-FCV-073-0030, automatically opens when system flow is at or below 900 gpm (lowering) if a system initiation signal is present, and automatically closes when system flow is at or above 1255 gpm (rising) regardless of presence of initiation signal.

J. HPCI PUMP MIN FLOW VALVE, 3-FCV-073-0030, opens on receipt of an initiation signal, even with HPCI Auxiliary Oil pump in PULL-TO-LOCK position, resulting in slowly draining CST to Suppression Chamber.

K. When a HPCI System isolation signal is reset, the steam line isolation valves do not automatically open, and are required to be opened via handswitch operation, even if a system initiation signal is present.

L. HPCI turbine operation below 2,400 rpm should be minimized to ensure adequate oil pressure from the turbine driven oil pump, to reduce system vibration, and prevent possible water hammer in the exhaust line.

BFN Unit 3	High Pressure Coolant Injection System	3-OI-73 Rev. 0063 Page 9 of 99

3.0 PRECAUTIONS AND LIMITATIONS

A. The HPCI turbine automatically trips on any of the following:

1. RPV water level high at +51 inches
2. Low pump suction pressure at 19.3" HG Vacuum (4.7 sec time delay)
3. Turbine high exhaust pressure at 140 psig
4. Any isolation signal
5. Remote Manual HPCI TURBINE TRIP pushbutton, 3-HS-73-18A

B. HPCI turbine overspeed at 122% (~5000 rpm) of rated speed (~4100 rpm) results in a hydraulic trip. The hydraulic trip occurs when operating oil is ported from the HPCI TURBINE STOP VALVE, 3-FCV-073-0018, causing the stop valve to close under spring force. Once the stop valve is closed, the piston of the hydraulic trip resets. With the HPCI turbine under load, the field-adjusted reset should occur between 2500 and 3000 rpm, and the startup sequence should commence. Since the overspeed trip condition does not result in any automatic trip signals in the HPCI control circuit, the HPCI PUMP MIN FLOW VALVE, 3-FCV-73-0030 does not close as a direct result of the turbine overspeed.

Excerpt from 3-OI-71:

BFN Unit 3	Reactor Core Isolation Cooling System	3-OI-71 Rev. 0063 Page 10 of 84

3.1 General Precautions (continued)

E. RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34, opens on receipt of an initiation signal even with RCIC turbine manually tripped resulting in slowly draining CST to Suppression Chamber.

F. [NER/C] If RCIC Turbine is tripped by mechanical overspeed trip device, RCIC TRIP/THROT VALVE, 3-FCV-71-9, is required to be manually reset at the turbine. [INPO SOER 82-008] If there is NO positive indication in Control Room that 3-FCV-71-9 is reset, personnel should verify position and document in rounds sheet.

G. RCIC Turbine operation below 2100 rpm may result in unstable system operation and equipment damage.

H. RCIC TURB EXHAUST VACUUM RELIEF SOV, 3-FCV-71-59, is normally de-energized in the open position and is required to be re-energized and closed to minimize leakage from primary containment following a LOCA when RCIC is shutdown and NO longer required.

I. Technical Specification 3.5.3 requires RCIC System operability be determined within 12 hours after RPV pressure is above 150 psig, or prior to startup using auxiliary steam.

J. RCIC Turbine oil drain and sample valves should **NOT** be operated without permission from the Unit Supervisor.

K. Injection of Suppression Pool water into the RPV should be avoided whenever possible to prevent degradation of primary system water quality.

L. [NER/C] Failure to manually trip the RCIC Turbine if speed exceeds 5700 rpm may result in equipment failure. [IE notice 90-045] Operation of the RCIC turbine can be stopped using the RCIC TURBINE TRIP pushbutton, 3-HS-71-9A. Section 8.4 is used to restore the turbine to operation if required.

M. When operable, RCIC SYSTEM FLOW/CONTROL controller, 3-FIC-71-36A, should be in AUTO in order to provide more stable system operation.

N. When RCIC SYSTEM FLOW/CONTROL controller, 3-FIC-71-36A, is operated in MANUAL, turbine speed should be raised as rapidly as possible to prevent turbine exhaust check valve chatter.

O. Whenever the RCIC STEAM LINE INBD or OUTBD ISOLATION VLVs, 3-FCV-71-2 or 3, are closed, MN STM LINE DRAIN INBD and OUTBD ISOLATION VLVs, 3-FCV-1-55 and 56, should be open to drain the RCIC Steam Line.

Examination Outline Cross-reference:

209001 (SF2, SF4 LPCS) Low-Pressure Core Spray

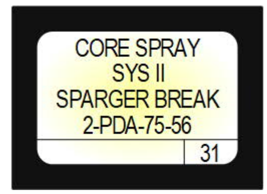
G2.4.31 (10CFR 55.41.10)

Knowledge of annunciator alarms, indications, or response procedures.

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	209001G2.4.31	
Importance Rating	4.2	------

Proposed Question: **# 32**

Which **ONE** of the following completes the statements below in accordance with the associated ARP?



- CORE SPRAY SYSTEM II SPARGER BREAK (2-9-3F, Window 31)

Core Spray Sparger break detection is used to detect a possible Core Spray System piping break inside the Reactor Vessel, **(1)** to the Core Shroud.

Due to break detection instrumentation design, the given annunciator **(2)** be sealed in during normal cold shutdown conditions on a Unit.

A. (1) external
(2) would

B. (1) external
(2) would NOT

C. (1) internal
(2) would

D. (1) internal
(2) would NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with CORE SPRAY SYS II SPARGER BREAK (2-9-3F, Window 31), indicates possible Core Spray pipe break inside the Reactor Vessel, external to the core shroud. For second part, the provided alarm would be in alarm during normal cold shutdown conditions on a Unit at 2 psig differential and lowering. This is due to the low-side pressure (above-Core plate pressure plus the pressure due to the height of water in the Reactor Vessel) being greater than high side pressure (Core exit pressure plus pressure due to height of water in the sensing leg) creating a negative differential pressure.

B INCORRECT: First part is correct *(See A)*. The second part is incorrect but plausible in that the provided alarm would NOT be expected during normally steady state power operation. The candidate could confuse the differential pressure sensing points related to the given alarm.

C INCORRECT: First part is incorrect but plausible in that the Core Spray pipe break detection system monitors multiple pressure locations within the Reactor Vessel which could confuse the candidate. This involves differential pressures related to jet pump driving force, across core plate, steam separators and steam dryer. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the annuciator alarms, indications and procedures related to Core Spray Pipe Break Detection System. This question is rated as C/A due to the requirement to assemble, sort, and integrate at least two different parts of the question to predict an outcome. This requires mentally using specific knowledge of the five different sensing points of the Core Spray Pipe Break Detection System to determine the correct outcome as it relates to plant conditions.

Technical Reference(s): 2-ARP-9-3F, Rev. 40 (Attach if not previously provided)
OPL171.045, Rev. 22

Proposed references to be provided to applicants during examination: **CORE SPRAY SYSTEM II SPARGER BREAK (2-9-3F, Window 31)**

Learning Objective: OPL171.045 Obj. 2d (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.045-02 004 #1466	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1466. OPL171.045-02 004

Which ONE of the following completes the statements below?

Core Spray Sparger break detection shares an RPV penetration with __(1)__.

Core Spray Sparger break detection is used to detect a possible Core Spray break inside the vessel, __(2)__ to the core shroud

A.✓ (1) SLC
(2) external

B. (1) SLC
(2) internal

C. (1) a RPV level instrument DP cell
(2) external

D. (1) a RPV level instrument DP cell
(2) internal

Excerpt from 2-ARP-9-3F:

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0040 Page 35 of 41

CORE SPRAY SYS II SPARGER BREAK 2-PDA-75-56 (31)	Sensor/Trip Point: 2-PDIS-075-0056	Decreasing 2 psig DP (15 second time delay)

(Page 1 of 1)

Sensor Location: Panel 25-27
El 565', R-10 S-LINE

Probable Cause:
A. Core spray piping break between reactor vessel wall and reactor shroud if alarm comes in during steady-state power operation.
B. Low core flow or unit in cold condition.
C. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **DISPATCH** personnel to Panel 2-25-27 to compare 2-PDIS-75-56 with 2-PDIS-75-28. The normal reading should be approximately 3.5 psid at high power operation.
B. **IF** instrument verification is necessary, **THEN** **REQUEST** that IMs check instrument operation.
C. **IF** indications confirm a broken core spray header, **THEN**, The associated Core Spray System is inoperable **REFER TO** Tech Spec 3.5.1 and 3.5.2, TRM 3.3.3.3.
D. **IF** there are NO indications of a core spray header break, **THEN** **REFER TO** Tech Spec Table 3.3.5.1.

Excerpts from OPL171.045 Lesson Plan:

OPL171.045, Core Spray System, Rev. 19

3. Significant Alarms

Item	Setpoints	Function	
"Core Spray Sys I (II) Sparger Break	2 psid decreasing (15 sec. T.D)	Alarm only. Indicates possible Core Spray pipe break inside vessel, external to core shroud	Normally in alarm in cold shutdown Obj. ILT 2.d(OF-5), 5.b,h Obj. LOR 1.d(OF-5), 4.b,h Obj. NLOR 5(OF-5) Obj. NLO 7(OF-5)

OPL171.045, Core Spray System, Rev. 19

3) The Unit 2, division 2, preferred pumps receive a trip from the Unit 1 CS or RHR initiation logic (14-K11B or 10-K73B) when a Unit 1 LOCA is sensed provided a Unit 2 div 2 CS initiation doesn't exist.	Note the Unit 2 Div 2 CS initiation completely blocks the Unit 1 signal and initiation response is normal and uninterrupted.
4) For Unit 1, division 1 works exactly like division 2 on Unit 2. Unit 1 division 2 is the Unit 1 non-preferred division and works the same as Unit 2 division 1.	
o. The Diesel Generator Output breaker logic includes Unit Priority Retrip Logic. The purpose of Unit Priority Re-trip logic is to allow the preferred pumps on the Unit with the first accident signal to continue to run but readies the second accident Unit for its preferred pump starts. This is done by selectively tripping the appropriate DG output breakers to facilitate a 4kv load shed.	CASA & CASB send an initial trip signal to all 8 DG breakers to shed all 4kv loads on the SD boards being powered by DGs. Eg. Bds are on DGs because a LOSP has first occurred.
p. The priority retrip will retrip the DG breakers for the 4kv boards which will have pumps starting on them. Ie. For a Unit 3 initiated CAS signal, a Unit 1 or 2 LOCA will start all 8 of that Units ECCS pumps on boards A, B, C, & D. Those 4 DG output breakers trip in that case. For a Unit 1 or 2 initiated CAS signal, the subsequent initiation from Unit 2/1 will cause diesel breaker retrip for the preferred division for that Unit only (A/B for Unit 1, or C/D for Unit 2).	
6. Leak Detection	
Core Spray piping penetrates the drywell, reactor vessel and shroud. If a pipe break occurred between vessel wall and the shroud, Core Spray function would be lost. Pipe break detection system monitors the integrity of the Core Spray piping and alarms in Control Room.	Obj. ILT 2.d(OF-5) Obj. LOR 1.d(OF-5) Obj. NLOR 5(OF-5) Obj. NLO 7(OF-5)
a. Pressure 1 (P1) is greater than P5 due to the jet pump driving force.	See TP-3 encircled numbers as referenced in each of five steps at left
b. P1 is greater than P2 due to the pressure drop across the core plate.	

OPL171.045, Core Spray System, Rev. 19

c. P2 is greater than P3 due to the pressure drop across the core. (This ΔP is small.)

d. P3 is greater than P4 by 7 psi due to the pressure drop across the steam separators.

e. P4 is greater than P5 by 7" of water due to the pressure drop across the steam dryer.

f. The low side of the detector senses above-core plate pressure (P2) plus the pressure due to the height of water in the vessel. Under normal conditions the high side of the detector senses core exit pressure (P3) plus pressure due to the height of water in the sensing leg. With the plant operating at rated conditions the detector reads +3.5 psid. P3 is slightly less than P2 due to the ΔP across the core. Therefore, the pressure differential detected is mainly due to the height of cold water (135°F) in the high leg of piping. If the Core Spray piping breaks between the reactor vessel and the shroud, piping is now sensing P5 instead of P3, and the high-side pressure at the detector would decrease by 7 psig. Sensed low-side pressure will remain the same. This would cause the ΔP to decrease, causing an alarm to sound at 2 psid decreasing (following a 15-sec time delay). During cold shutdown conditions this alarm will normally be in. This is due to low-side pressure being greater than high side pressure (negative ΔP).

Actual ΔP instruments are located on Elev. 565 of Reactor Building, only alarms in control room. Instrument readings satisfy TRM requirements.

Based on Tech. Support calculations, alarm will clear at 300-330 psig, depending on drywell temp. Loop 2 typically reads higher than Loop 1. (Ref. R)

D. Relationships With Other Systems

1. Combines with other ECCS to provide adequate core cooling over the entire break spectrum. — Obj. ILT 4, Obj. LOR 3, Obj. NLOR 7

2. Torus provides the normal source of water. — Obj. NLO 9

3. The CST provides an alternate source of water.

4. Keep Fill System maintains Core Spray piping full of water to prevent water hammer.

5. Emergency Equipment Cooling Water furnishes cooling water to the Core Spray room coolers. — NLOR 3

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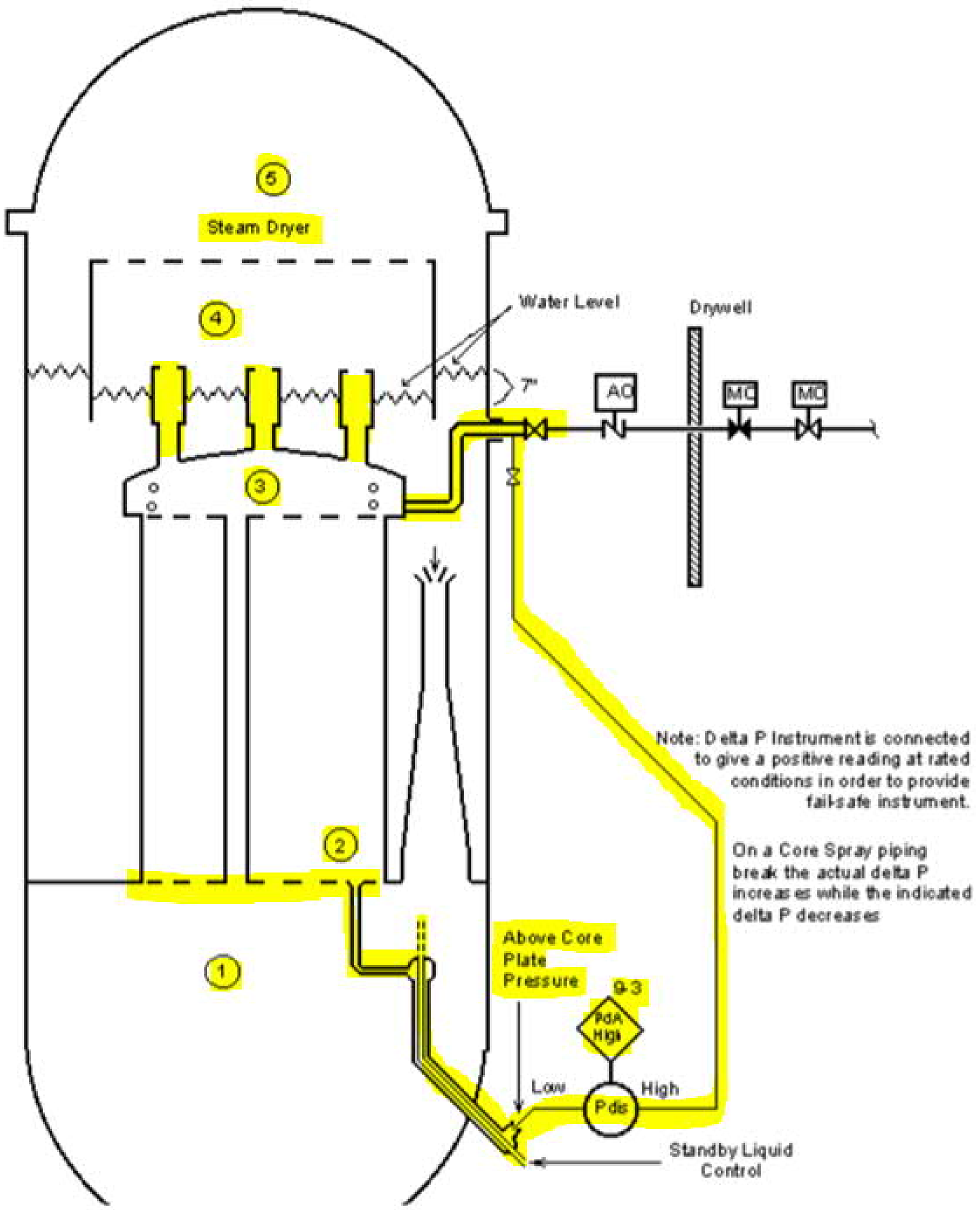


TP-3: CORE SPRAY PIPE BREAK DETECTION INSTRUMENTATION

Examination Outline Cross-reference:	Level	RO	SRO
211000 (SF1 SLCS) Standby Liquid Control	Tier #	2	------
K1.03 (10CFR 55.41.8)	Group #	1	------
Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following:	K/A #	211000K1.03	
• Plant air systems: Plant-Specific	Importance Rating	2.5	------

Proposed Question: **# 33**

Unit 1 has suffered a loss of the Service Air System.

Given the condition above, which **ONE** of the following completes the statement below?

The loss of Service Air will **_________** on the SLC System Storage Tank.

A. have no effect

B. cause a loss of level indication **ONLY**

C. cause a loss of mixing capability **ONLY**

D. cause a loss of level indication **AND** a loss of mixing capability

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that the Control Air system also provides air to the SLC Storage Tank for level indication. It is plausible that Control Air provides both air mixing and level indication, and that a loss of Service Air will not affect the SLC System.

B INCORRECT**:** Incorrect but plausible in that Control Air provides for level indication of the SLC Storage Tank.

C **CORRECT**: (*See attached*) In accordance with 1-OI-63, Service Air provides for air mixing of the SLC Storage Tank.

D INCORRECT: Incorrect but plausible in that Servie Air does provide for air mixing in the SLC Storage Tank, but level indictation is provided for by Control Air.

RO Level Justification: Tests the candidate's knowledge of the SLC System's connections with plant air systems and the effect of a loss of Service Air on the SLC System. This question is rated as Memory due to the requirement to strictly recall facts related to how plant air systems are connected to the SLC System.

Technical Reference(s): 1-OI-63, Rev.6 (Attach if not previously provided)

OPL171.039, Rev. 21

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.039, Obj. 8 (As available)

Question Source:	Bank #	Quad Cities #46	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2011	

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Copy of Bank Question:

EXAMINATION ANSWER KEY

Quad Cities 2011 ILT NRC Exam (RO Portion)

46 **ID: QDC.ILT.16498** **Points: 1.00**

What effect, if any, does a loss of Service Air have on the Standby Liquid Control (SBLC) System?

A. No effect.

B. Loss of storage tank level indication only.

C. Loss of storage tank mixing capability only.

D. Loss of storage tank level indication and mixing capability.

Answer: C

Excerpt from 1-OI-63:

BFN Unit 1	Standby Liquid Control System	1-OI-63 Rev. 0006 Page 22 of 32

8.3 Preparing SLC Storage Tank For Boron Sampling

NOTES

1) Preparation of the SLC Storage Tank should be done when any one of the following conditions exist:
   - 1-SR-3.1.7.3 is to be performed.
   - A level change in the tank has occurred.
   - As directed by the Unit Supervisor.
2) All operations are performed locally unless otherwise noted.

CAUTION

When the air valve is opened in the following steps and air is admitted into the SLC Storage Tank, SLC is considered INOPERABLE due to the possibility of air entrapment into the positive displacement pumps. The system is considered inoperable only while air is being supplied to air mix the boron. (Tech Spec 3.1.7-B [BFPER 99-004803-000])

[1] **ENSURE** Service Air System is available to supply air mixing to the SLC Storage Tank. ☐

[2] **NOTIFY** the Unit Supervisor that air mix in the SLC tank is about to take place and SLC System will be considered inoperable while air is supplied to the tank for mixing. (Reference Tech Spec 3.1.7). ☐

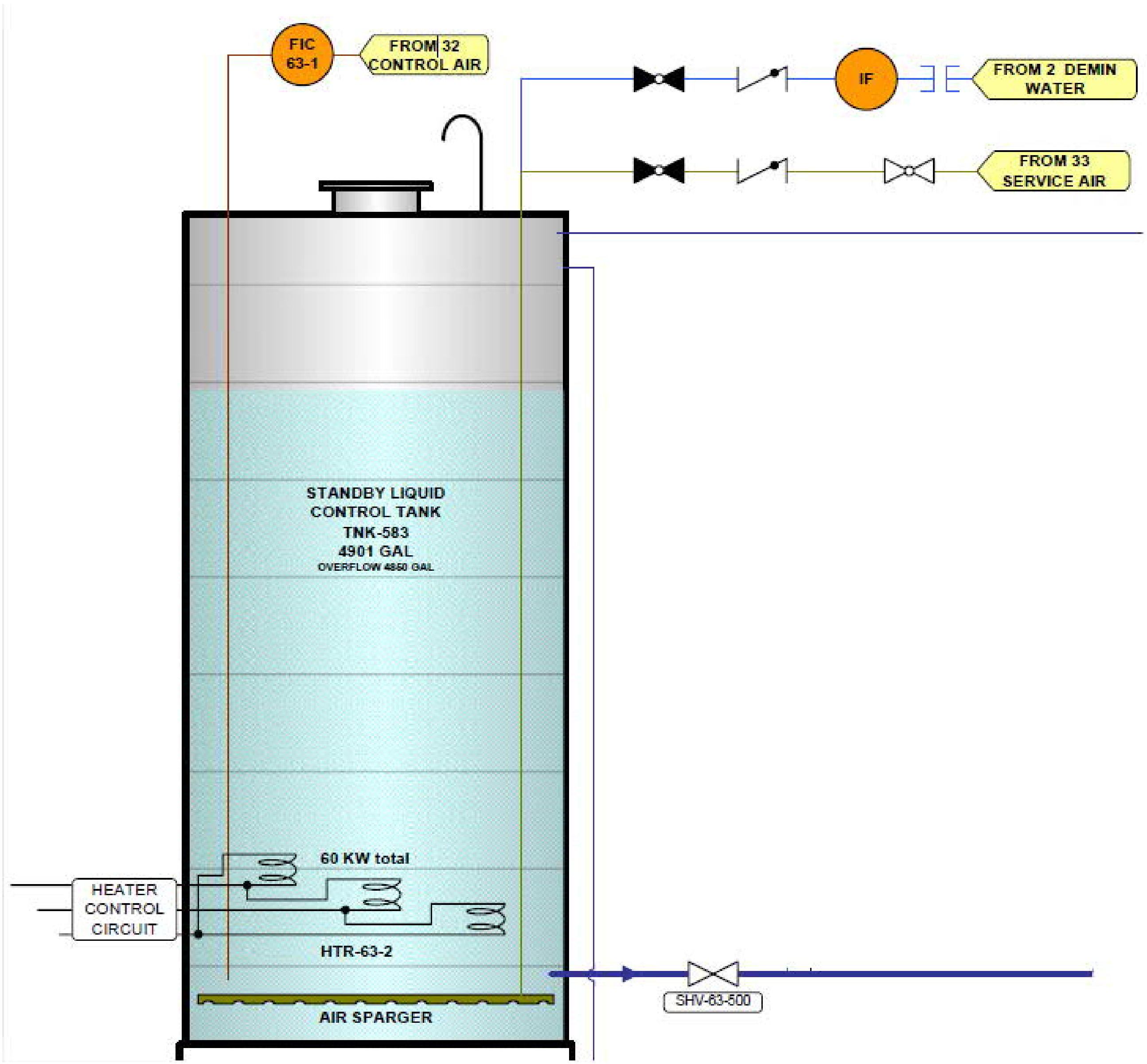
CAUTION

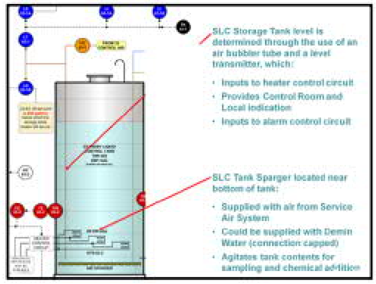
Adequate mixing time (20 minutes) is required to be strictly enforced to ensure representative sampling. Excessive mixing times should be avoided (i.e. approximately 1 hour).

[3] **UNLOCK** and **OPEN** SLC STORAGE TANK SERVICE AIR, 1-SHV-063-0536. ☐

[4] **ENSURE OPEN** SA SPLY TO SLC (R2P), 1-SHV-033-0756. ☐

Excerpts from OPL171.039 Lesson Plan:





38

OBJ:
NLO 3.a, 5.e, 8.a
LO 3.a, 5.f, 10.a

The SLC Storage Tank liquid level is determined through the use of an air bubbler tube (or purge system) and a level transmitter. An open tube is immersed in the storage tank liquid such that the open end is off the bottom of the tank. Air from the Control Air System, manually adjusted at SLC STORAGE TANK LEVEL CONTROLLER (FIC-63-1) to control between 1 and 1.5 SCFH, is forced into the top of the tube until bubbles constantly stream from the bottom of the tube. In the event tank level increases, the increased level will cause the hydrostatic head (backpressure) within the bubbler to increase. On the other hand, if tank level decreases, the hydrostatic head within the bubbler will decrease. (Objective 5e)

Changes in the pressure of the hydrostatic head of the liquid due to increasing or decreasing tank level will be sensed by SLC STORAGE TANK LEVEL TRANSMITTER (LT-63-1), which is powered by 120V AC I&C B. This transmitter transmits an electrical output signal proportional to storage tank level to a 0-100% range SLC STORAGE TANK LEVEL indicator on Control Room Panel 9-5 (LI-63-1A). Local SLC STORAGE TANK LEVEL indication is also available at Panel 25-19 (LI-63-1B).

Annunciator SLC TANK LEVEL ABNORMAL, (9-5B Window 21) is provided to warn operators of an improper storage tank level. The high level alarm activates at an indicated tank level of 95.4% increasing, which corresponds to an actual level of approximately 4,626 gallons. The low level alarm activates at an indicated level of 85.6% decreasing, which corresponds to an actual level of approximately 4,150 gallons. (Objective 5g)

Included inside, near the bottom of the storage tank, is a sparger that can be supplied with air via the Service Air System or demineralized water via the Demineralized Water System. Service Air can be supplied to the tank sparger through a normally locked closed SLC STORAGE TANK SERVICE AIR SHUTOFF VALVE (SHV-63-536) in order to agitate the tank contents for mixing during chemical addition and routine sampling operations. Demineralized water could be supplied through the tank sparger, via a normally closed SLC STORAGE TANK DEMIN WATER SHUTOFF VALVE (SHV-63-534); but, currently that connection is capped. All chemical additions to the SLC Storage Tank are batch-conducted through a temporary SLC Addition Stand on Reactor Building EL 621 ft. Demineralized water only additions are conducted through a temporary hose, attached to a Demineralized Water System connection, directly into the SLC Storage Tank.

Examination Outline Cross-reference:

212000 (SF7 RPS) Reactor Protection

K3.05 (10CFR 55.41.7)

Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following:

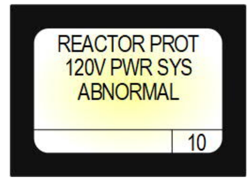
- RPS logic channels

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	212000K3.05	
Importance Rating	3.7	------

Proposed Question: **# 34**

Unit 2 is operating at 100% RTP with the following conditions:

- 2A Reactor Protection System (RPS) Motor Generator (MG) trips
- REACTOR PROTECTION 120V POWER SYSTEM ABNORMAL (2-9-5B, Window 10) alarms



Given the conditions above, which **ONE** of the following completes the statements below?

___**(1)**___ RPS logic channels are impacted. As a result, a ___**(2)**___ occurs.

A. (1) A1 **AND** A2
(2) Full SCRAM

B. (1) A1 **AND** A2
(2) Half-SCRAM **ONLY**

C. (1) A1, A2 **AND** A3
(2) Full SCRAM

D. (1) A1, A2 **AND** A3
(2) Half-SCRAM **ONLY**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate confuses the two separately powered RPS trip systems A and/or B as having only 2 channels per trip system. Second part is incorrect but plausible in that given the complexity of RPS, the candidate could confuse the impact from the combination of 2A RPS loss with the condition causing the given alarm as resulting in a Full SCRAM.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is correct *(See D).*

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT**: *(See attached)* RPS logic consists of two separately powered (A or B RPS MGs) trip systems each having three channels. Two channels are utilized to produce automatic SCRAM signals (trip channels A1/A2 and/or B1/B2 respectively). The third is used to produce manual SCRAM signals (A3 and B3). Given the trip of the 2A RPS MG with REACTOR PROT 120V PWR SYS ABNORMAL alarm, indicates a loss of power to one of the RPS channnels. The loss of 2A RPS MG will impact A1/A2 (auto SCRAM) and A3 (manual SCRAM) RPS logic channels. For second part, in accordance with the given alarm response procedure, the Operator is referred to 2-AOI-99-1 which states RPS trip logic A(B) half-SCRAM occurs.

RO Level Justification: Tests the candidate's knowledge of the effect that a loss or malfunction of the Reactor Protection System has on its respective logic channels. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-99, Rev. 88 (Attach if not previously provided)
2-AOI-99-1, Rev. 30
2-ARP-9-5B, Rev. 31
OPL171.028, Rev. 21

Proposed references to be provided to applicants during examination: **REACTOR PROT 120V PWR SYS ABNORMAL (2-9-5B, Window 10)**

Learning Objective: OPL171.028 Obj. 18a (As available)

Question Source:	Bank #		(Note changes or attach parent)
	Modified Bank #		
	New	X	
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Excerpts from OPL171.028 Lesson Plan: Discusses the RPS power supply, trip system/channel and logic arrangement

Outline of Instruction	Instructor Notes and Methods
b) This action limits uncontrolled release of radioactive material by terminating excessive temperature and pressure rise.	
B. Component Description	Objective 2,10 IL-1
1. The Reactor Protection System includes the motor-generator power supplies with associated control and indicating equipment, sensors, relays, bypass circuitry, and switches that supply a signal to the Control Rod Drive (CRD) system to cause rapid insertion of control rods (SCRAM) to shut down the reactor. It also includes outputs to the process computer system and annunciators. The system includes two motor-generator power supplies with an alternate supply via a transformer and the sensors, relays, bypass circuitry and switches that operate valves to cause rapid insertion of all control rods.	
2. Normal power to RPS Buses A and B on each unit is supplied by two motor-generator (MG) sets / one per bus which powers two independent trip systems.	
a) Motor: (1) 480VAC (2) 3 phase (3) Powered from 480V RMOV Boards A and B on the respective unit.	Objective 3a, 11a OF-5
b) Generator: (1) 120VAC output (2) Single phase (3) 60 Hz (4) Motor-generator flywheel maintains voltage and frequency within 5 percent of rated values for at least 1.0 second following total loss of power to the drive motor. This allows for loss of power due to switching operations.	
3. Alternate Power to RPS is supplied from 480V RMOV Board B through a transformer.	
a) Unit 1 alternate power is supplied through a transformer from 480V RMOV Board 1B	Objective 3b, OF-5
b) Unit 2 alternate power is supplied through a transformer shared with the Unit Preferred System (UPS) from 480V RMOV Board 2B. Unit Preferred transformer can only supply UPS or one of two RPS busses at any one time. This is due to the load restrictions of the transformer.	Unit Difference Objective 14
c) Unit 3 alternate power is supplied through a transformer from 480V RMOV Board 3B.	
d) Interlocked so that both RPS buses cannot be	Objective 4c

OPL171.028 , Reactor Protection System, Rev# 21

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
b) RPS B supply breakers/loads:	
(1) Breaker 953: RPS B Logic, Sensors, and HCU Solenoids	
(2) Breaker 954: HVPS for PRNM/LPRM/RBM	
(3) Breaker 955: Rx Bldg/Refuel Floor Vent Rad Monitor 90-141/143, MSL Rad Mon B and D	
(4) Breaker 956: PCIS Div II Logic, Sensors, and MSIV Outboard AC Solenoids	
C. Reactor Protection Logic	
1. Logic is designed so that failure of a single component will neither cause a SCRAM nor prevent a SCRAM. To achieve this, all sensors, actuating relays and most of the switches are arranged in redundant logics.	Objective 4a,b Objective 12a,b Objective 13b
2. Logic consists of two separately powered trip systems each having three channels:	IL-2
a) Two channels are utilized to produce automatic SCRAM signals (trip channels A1 and A2).	2-730E915RF-11 2-730E915RF-12 2-730E915-13
b) The third is used to produce manual SCRAM signals (trip channel A3).	
c) The channels for trip system B are designated B1, B2 and B3.	
3. Both of the automatic channels in each trip system monitor critical reactor parameters.	
a) At least four channels for each monitored parameter are required for the trip system logic.	Objective 4a Objective 12d
b) If either of the two channels sense a parameter which exceeds a setpoint, then this would place the associated trip system (A or B) into a tripped condition.	IL-3 2-730E915RF-11 2-730E915RF-12
c) To produce a SCRAM, both trip systems must be tripped. This is called a "one-out-of-two-taken twice" arrangement.	Objective 5a0
4. Each trip system logic may also be manually tripped.	Objective 12f IL-4
a) Each trip system contains manual SCRAM switches on Panel 9-5 which cause a trip in the respective trip system when actuated.	
b) The reactor mode switch has contacts in both the A3 and B3 channels. Placing the reactor mode switch in SHUTDOWN will result in a trip of both trip systems.	2-730E915RF-11 2-730E915RF-12
c) A trip in both channels A3 and B3 initiates a reactor SCRAM.	
5. During normal operation all sensor and trip contacts essential to safety are closed.	

OPL171.028 , Reactor Protection System, Rev# 21

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
a) Channels, logics, and actuators are energized.	
b) When a SCRAM signal is received, the logic relays de-energize to cause a SCRAM.	Objective 12h
c) Loss of power to one RPS bus will result in a half-SCRAM. Loss of power to both RPS buses will result in a full SCRAM.	2-730E915-13
D. Reactor SCRAM Signals and Arrangement	Objective 4a,b;12a,b,c Refer to OI-99 for setpoints
1. Four Channel test switches, one per channel allows for testing each channel's trip function.	
a) Test switches located on Panel 9-15 and 9-17 in Aux. Inst. Room.	Objective 6, 17, 18
b) Key-locked, two positions - NORMAL and TRIP	2-730E915RF-11
c) Placing the switch to TRIP de-energizes that channel's relays producing a half-SCRAM.	2-730E915RF-12
2. Turbine Stop Valves, 10 percent closure anticipates the pressure and neutron flux rise caused by the rapid closure of the Turbine Stop Valves.	
a) Each of the four Turbine Stop Valves is equipped with two limit switches. One limit switch is assigned to RPS A and one to RPS B.	IL-5 Objective 13a, 13c
b) These switches will provide a valve-closed signal to the RPS trip logic.	2-730E915-9, 10
c) The position switch contacts are arranged so that any two Stop Valves can be closed causing no more than a half-SCRAM.	
d) Closure (< 90% full open) of any combination of three Stop Valves will cause a full SCRAM in all cases.	2-730E915RF-11 2-730E915RF-12
e) From the logic it can be determined that: (1) Closing one valve does not cause a half-SCRAM. (2) Closing 1 and 4, or 2 and 3, at the same time does not yield a half-SCRAM. (3) Closing any other combination of two valves will cause a half-SCRAM. (4) Any combination of three or more valves closed will cause a reactor SCRAM.	
3. Generator Load Reject SCRAM (Control Valve fast closure) (<850 psig ETS pressure)	
a) Anticipates the rapid rise in pressure and neutron flux resulting from fast closure of the Turbine Control Valves due to a load rejection.	
b) Definition of load reject: Greater than 40 percent mismatch between generator stator amps and turbine cross-under pressure. Generally speaking, a load reject is a sudden loss of electrical grid load-like a loss of all offsite feeders.	Objective 13a Mismatch between steam demand and electrical load on hand.

Excerpt from 2-ARP-9-5B:

BFN Unit 2	Panel 9-5 2-XA-55-5B	2-ARP-9-5B Rev. 0031 Page 13 of 43

REACTOR PROT
120V PWR SYS
ABNORMAL

10

(Page 1 of 1)

Sensor/Trip Point:

Bus A	42T-A circuit breaker
Bus B	42T-B circuit breaker
Bus A	2A1 circuit protector
Bus A	2A2 circuit protector
Bus B	2B1 circuit protector
Bus B	2B2 circuit protector
Unit Preferred TUP-2	2C1 circuit protector
Unit Preferred TUP-2	2C2 circuit protector

Sensor Location: Battery Board 2, Panel 9A, Bus A and B
A11 circuit protectors Battery Board Room 2.

Probable Cause:

A. Circuit breaker open.
B. RPS MG Set A or B abnormal.
C. One RPS channel on the alternate power supply.
D. Circuit protector tripped on over voltage, under voltage or under frequency. (Placing the Unit Preferred XFMR2 (TUP2) in service to an RPS bus without a burn-in period to stabilize transformer output voltage, can result in elevated output voltage.)
E. Sensor malfunction.
F. Cleared fuse for control power to any circuit protector.

Automatic Action: De-energizing one RPS channel actuates every scram-related alarm for the affected channel and actuates PCIS inboard or outboard logic.

Operator Action:

A. **REFER TO** 2-AOI-99-1. ☐
B. **REFER TO** Tech Spec Sect 3.3.8.2. ☐

References: 2-45E620-6 2-45E641-4 2-AOI-99-1
Technical Specifications

Excerpts from 2-AOI-99-1:

BFN Unit 2	Loss of Power to One RPS Bus	2-AOI-99-1 Rev. 0030 Page 3 of 18

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic action and operator action for the loss of power to one of the two RPS buses.

2.0 SYMPTOMS

2.1 Annunciators in alarm:

A. REACTOR PROT 120V PWR SYS ABNORMAL (2-XA-55-5B, Window 10).

1. Alternate power supply breaker (42T-A or 42T-B) CLOSED, or
2. Any one Circuit Protector tripped (2A1, 2A2, 2B1, 2B2, 2C1, 2C2).

B. All Reactor Scram and Primary Containment Isolation alarms associated with de-energized RPS bus.

C. RX BLDG VENTILATION ABNORMAL (2-XA-55-3D, Window 3).

1. Trip of Reactor and Refuel Zone supply/exhaust fans.

2.2 Control Room Indication:

A. Loss of Quadruple Low Voltage Power Supplies (QLVPS) as indicated by the extinguished green lights on the front of the QLVPS.

B. The Affected APRM/LPRM will have a "FAULT" indicated in Inverse Video on the top bar of the display panel. This indication will clear just after RPS is restored.

BFN Unit 2	Loss of Power to One RPS Bus	2-AOI-99-1 Rev. 0030 Page 4 of 18

3.0 AUTOMATIC ACTIONS

NOTE

An overview of automatic actions for RPS Bus A(B) is provided here. A detailed list of actions is provided in 2-OI-99, Illustration 1, which lists actions that occur when RPS buses are de-energized on a transfer of power supply.

A. RPS trip logic A(B) half-scram occurs.

B. PCIS Group 1 half-trip logic de-energizes.

C. PCIS Group 2 isolation, RHR Shutdown Cooling Mode:

1. Bus A inboard.
2. Bus B outboard.

D. PCIS Group 3 isolation, RWCU:

1. Bus A inboard and outboard.
2. Bus B outboard.

E. PCIS Group 6 isolation, Primary Containment Vent and Purge and Reactor Building Ventilation:

1. Bus A or B inboard and outboard.

F. Group 8 isolation, TIP.

G. Control Room Emergency Ventilation System start.

H. Standby Gas Treatment System starts.

Excerpts from 2-OI-99: Supports Distractors A(1), B(1):

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0088 Page 25 of 107

6.1 Reset of One RPS Trip Logic Channel (continued)

[6] **CHECK** the following conditions:

A. All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED.

B. The following four lights ILLUMINATED:

- SYSTEM A BACKUP SCRAM VALVE, 2-IL-99-5A/AB.
- SYSTEM B BACKUP SCRAM VALVE, 2-IL-99-5A/CD.

C. Scram Discharge Volume vent and drain valves indicate OPEN.

D. Points SOE033 (Channel A3 manual scram) and SOE035 (Channels A1 & A2 Auto Scram) on ICS computer or on the First Out Printer reads "NOTTRIP" for RPS "A".

E. Points SOE034 (Channel B3 manual scram) and SOE036 (Channels B1 & B2 Auto Scram) on ICS computer or on the First Out Printer reads "NOTTRIP" for RPS "B".

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0088 Page 61 of 107

8.5 Restoration to Normal Following RPS Bus Power Loss or Transfer (continued)

[3] **CHECK** the following conditions:

A. All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED.

B. The following four lights ILLUMINATED:

- SYSTEM A BACKUP SCRAM VALVE, 2-IL-99-5A/AB.
- SYSTEM B BACKUP SCRAM VALVE, 2-IL-99-5A/CD.

C. Scram Discharge Volume vent and drain valves indicate OPEN.

D. Points SOE033 and SOE035 on ICS computer or on the First Out Printer reads "NOTTRIP" for RPS "A".

E. Points SOE034 and SOE036 on ICS computer or on the First Out Printer reads "NOTTRIP" for RPS "B".

[4] At Panel 2-9-4, **RESET** PCIS trip logic as follows:

Examination Outline Cross-reference:	Level	RO	SRO
215003 (SF7 IRM) Intermediate-Range Monitor	Tier #	2	------
A3.03 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including:	K/A #	215003A3.03	
• RPS status	Importance Rating	3.7	------

Proposed Question: **# 35**

Unit 2 is in MODE 2 with the following conditions:

- All IRMs are on Range 4
- During Control Rod withdrawal the following IRM indications are noted:

IRM 'E' – 108/125

IRM 'F' – 118/125

IRM 'H' – 117/125

Given the conditions above, which **ONE** of the following describes the response of RPS and/or the Reactor Manual Control System (RMCS) to these conditions?

A. Full SCRAM

B. Rod Block **ONLY**

C. RPS A Rod Block and Half SCRAM

D. RPS B Rod Block and Half SCRAM

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that the IRMs are divided up into the two RPS Channels, each with four IRMs (RPS Channel A – IRMs A/C/E/G, RPS Channel B – IRMs B/D/F/H), and the IRMs in each channel are often mixed up. IRMs F and H are in RPS Channel B and are above their SCRAM setpoints. IRM E is in RPS Channel A, but is only above the IRM High alarm setpoint.

B INCORRECT: Incorrect but plausible in that a Rod Block is generated, but IRMs F and H are above the SCRAM setpoint and will cause a half Reactor SCRAM.

C INCORRECT**:** Incorrect but plausible in that while IRMs F and H are above ths SCRAM setpoint, the candidate may confuse which IRMs are assigned to which RPS Channels.

D **CORRECT**: *(See attached)* In accordance with 2-OI-92A, Intermediate Range Monitors, a Rod Block is generated because the listed IRMs are above 90, and a Reactor SCRAM is generated because IRMs F and H are above 116.4.

RO Level Justification: Tests the candidate's knowledge of the effect the IRMs on RPS and the automatic actions that occur within RPS as a result of the IRM levels. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-92A, Rev.29 (Attach if not previously provided)
OPL171.020, Rev.12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.020 Obj. 6, 8 (As available)

Question Source:

Bank #	BFN 1703 #37	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam 2017

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 X
55.43

Comments:

Copy of Bank Question:

QUESTION 37 Rev 1

Unit 2 is in MODE 2.

- All IRMs are on range 4
- During Control Rod withdrawal the following IRM indications are noted:

  IRM 'E' – 108/125
  IRM 'F' – 118/125
  IRM 'H' – 117/125

What is the response of the Reactor Protection System (RPS) and/or Reactor Manual Control System (RMCS) to these plant conditions?

A. Rod Block **ONLY**

B. Rod Block and Half Scram RPS A

C. Rod Block and Half Scram RPS B

D. Full Scram

Answer: C

Excerpts from 2-OI-92A:

BFN Unit 2	Intermediate Range Monitors	2-OI-92A Rev. 0029 Page 14 of 20

NOTE

1) All IRM Rod Block Trips and IRM Scram Trips are automatically bypassed when the reactor mode switch is in the RUN position.
2) More than one IRM detector may be withdrawn at a time if needed. During a Reactor Startup one IRM Power Indication per channel should used to monitor power during IRM withdraw.
3) The DRIVE IN Circuitry may need to be reset by pressing the DRIVE IN 2-HS-92-7C/S2 if the IRM Detectors fail to withdraw.
4) All operations are performed on Panel 2-9-5 unless specifically stated otherwise.
5) IRMs are separated into two sections for RPS "A" and RPS "B". This section refers to withdrawing IRM detector "A" but each IRM can be substituted using the appropriate detector designator in place of the "A" reference.
   - RPS A- IRM A, C, E, G
   - RPS B- IRM B, D, F, H

BFN Unit 2	Intermediate Range Monitors	2-OI-92A Rev. 0029 Page 20 of 20

Illustration 1
(Page 1 of 1)

IRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
IRM High	> 90 ON 125 SCALE	Rod block unless REACTOR MODE SWITCH in RUN
IRM Inop	A. Module unplugged B. Mode switch not in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless REACTOR MODE SWITCH in RUN Reactor Scram unless REACTOR MODE SWITCH in RUN
IRM Downscale	< 7.5 on 125 SCALE	Rod block unless IRMs on range 1 unless REACTOR MODE SWITCH in RUN
IRM Detector Wrong Position	detector not full in	Rod block unless detector full-in, or REACTOR MODE SWITCH in RUN
IRM High-High	> 116.4 ON 125 SCALE	Reactor Scram unless REACTOR MODE SWITCH in RUN

Excerpt from OPL171.020 Lesson Plan:

OPL171.020 , Intermediate Range Monitor System, Rev# 12

5. Detector Assignments
   a) Four IRM channels (one detector per channel) are assigned to each Reactor Protection (RPS) channel for a total of 8 channels
   b) The arrangement of IRM channels allows one IRM channel in each RPS channel to be bypassed without compromising intermediate range neutron monitoring.

IRM Channel	RPS Channel
A	A1
C	A2
E	A1
G	A2
B	B1
D	B2
F	B1
H	B2

Examination Outline Cross-reference:	Level	RO	SRO
295030 (EPE 7) Low Suppression Pool Water Level / 5	Tier #	1	------
G2.2.42 (10CFR 55.41.10)	Group #	1	------
Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	K/A #	295030G2.2.42	
	Importance Rating	3.9	------

Proposed Question: **# 36**

Unit 2 is operating at 100% RTP under **NORMAL** conditions.

Which **ONE** of the following completes the statement below?

Given the conditions above, Suppression Pool Water Level shall be greater than or equal to _________ in accordance with Technical Specification (Tech Spec) 3.6.2.2, Suppression Pool Water Level.

A. (-) 5.50 inches

B. (-) 6.00 inches

C. (-) 6.25 inches

D. (-) 7.25 inches

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that (-) 5.50 inches is the Suppression Chamber Water Level Abnormal alarm setpoint.

B INCORRECT: Incorrect but plausible in that (-) 6.00 inches Suppression Pool Water Level is related to the low monitor and control band in accordance with SP/L-1 in 2-EOI-2, Primary Containment Control.

C **CORRECT**: (*See attached*) In accordance with Technical Specification 3.6.2.2, Suppression Pool Water Level shall be greater than or equal to (-) 6.25 inches **WITH** Primary Containment differential pressure control established. Given the conditions above, the RO candidate must cognitively analyze both Reactor Power and what NORMAL conditions means in order to determine if Primary Containment differential pressure control is or is not required. This requires the candidate to know that Primary Containment differential pressure control between the Drywell and Suppression Chamber will be established within 24 hours after thermal power is greater than 15% RTP following a startup in accordance with 2-OI-64, Primary Containment System. Once established, the Primary Containment differential pressure impacts the lowest acceptable Suppression Pool Water Level.

D INCORRECT: Incorrect but plausible in that (-) 7.25 inches Suppression Pool Water Level is the Technical Specification 3.6.2.2 acceptable level **WITHOUT** Primary Containment differential pressure control established.

RO Level Justification: Tests the candidate's ability to recognize system parameters as it relates to low Suppression Pool Water Level that are entry-level conditions of Technical Specifications. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): Unit 2 Tech Spec 3.6.2.2, Amend. 253 (Attach if not previously provided)

2-OI-64, Rev. 6

2-ARP-9-3B, Rev. 37

2-EOI-2, Rev. 16

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.016 Obj. 11 (As available)

Question Source:		
	Bank #	BFN 1703 #1
	Modified Bank #		(Note changes or attach parent)
	New	
Question History:	Last NRC Exam	2017

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

QUESTION 1 Rev 1

Which one of the following completes the statement below?

The EOI-2, Primary Containment Control, **Entry Condition** set point for Low Suppression Pool Level is _______ inches.

A. (-) 5.50

B. (-) 6.00

C. (-) 6.25

D. (-) 7.25

Answer: C

Excerpt from Unit 2 Tech Spec 3.6.2.2:

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be ≥ -6.25 inches with and -7.25 inches without differential pressure control and ≤ -1.0 inches.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool water level not within limits.	A.1 Restore suppression pool water level to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

Excerpts from 2-OI-64:

BFN Unit 2	Primary Containment System	2-OI-64 Rev. 0127 Page 36 of 151

6.8 Normal Operations

[1] **COMPLETE** applicable section of 2-SI-4.7.A.2.a, each time nitrogen is added or venting is performed. ☐

[2] **MAINTAIN** the following parameters:

- Nitrogen makeup less than 60 SCFM. ☐
- Drywell temperature less than or equal to 135°F. ☐
- Drywell pressure less than or equal to 1.5 psig. ☐
- Drywell to Suppression Chamber DP between 1.15 and 1.30 psid. ☐
- Drywell oxygen content less than 4 percent. ☐
- Drywell hydrogen content less than 4 percent. ☐
- Suppression Chamber oxygen content less than 4 percent. ☐
- Suppression Chamber hydrogen content less than 4 percent. ☐
- Suppression Pool level between -2 inches and -5.5 inches. ☐
- Suppression Pool water temperature below 95°F during normal power operation. ☐

BFN Unit 2	Primary Containment System	2-OI-64 Rev. 0127 Page 10 of 151

3.0 PRECAUTIONS AND LIMITATIONS

A. TOE 0-97-064-0823 evaluated the impact of inerting or purging the Suppression Chamber and the Drywell concurrently (Both FCV 64-19 and FCV 64-18 open at the same time). This evaluation concluded there is a potential to overpressurize primary containment in the event of a large break LOCA with both FCV 64-18 and FCV 64-19 open at the same time with the Reactor NOT in Cold Shutdown (Mode 4). This situation could create a large bypass flow path between the Drywell and the Suppression Chamber. Therefore, The Suppression Chamber and the Drywell shall NOT be inerted or purged at the same time unless the Reactor is in Cold Shutdown (Mode 4).

B. Drywell/Suppression Chamber purging operations may begin 24 hours prior to a scheduled shutdown (24 hours prior to reducing thermal power to <15% RTP prior to the next scheduled shutdown).

C. Drywell/Suppression Chamber nitrogen inerting must be completed (oxygen concentration less than 4% by volume) within 24 hours of going into RUN (24 hours after thermal power is >15% RTP following startup).

D. No Containment entry is permitted without special breathing equipment unless a natural air atmosphere has been established (oxygen greater than or equal to 19.5%), as verified by Chemistry obtaining a grab sample IAW CI-403.

E. Differential pressure control between the Drywell and Suppression Chamber will be established within 24 hours after thermal power is >15% RTP following startup and will be maintained between 1.15 and 1.30 psid to provide a margin to the Tech Spec limit.

F. The Drywell/Suppression Chamber differential pressure may be reduced to less than 1.10 psid 24 hours prior to reducing thermal power to <15% RTP prior to the next scheduled reactor shutdown or for a maximum of 4 hours during required operability testing of HPCI, RCIC, or the Drywell Pressure Suppression Chamber Vacuum Breakers.

G. Suppression Chamber water level will normally be maintained between minus 2-inches and minus 5-1/2-inches to provide adequate margin to Tech Spec limits.

H. The minimum Suppression Chamber water temperature is 50°F to assure adequate margin to analyzed material structural limits.

I. If both the primary and secondary indications on any Safety Relief Valve (tailpipe temperature and acoustic monitor) are inoperable, Suppression Chamber water temperature will be checked and recorded in the Narrative log at least once per shift to observe any unexplained temperature rises which might indicate an open Relief Valve.

Excerpt from 2-ARP-9-3B:

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0037 Page 19 of 39

SUPPR CHAMBER WATER LEVEL ABNORMAL
2-LA-64-54A
15
(Page 1 of 1)

Sensor/Trip Point:

LT-64-54 ≤ -5.5" H_2O

≥ -1.75" H_2O

Sensor Location: RX Bldg, El 519'
NW corner room just inside door

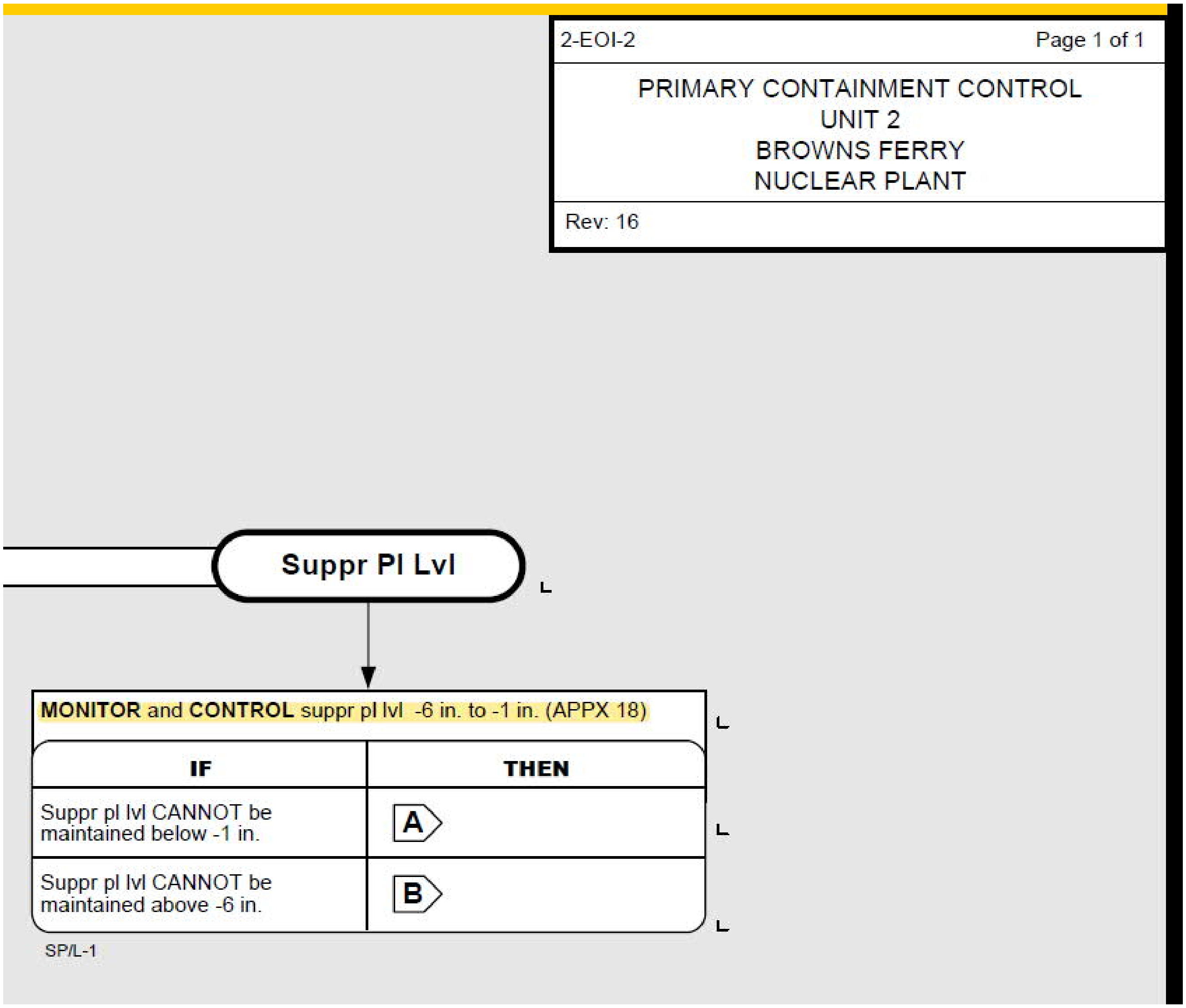
Probable Cause:
A. Suppression Chamber water level abnormal.
B. Placing Suppression Pool Cooling in service
C. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **CHECK** Suppression Pool level using multiple indications. ☐
B. **IF** level is low, **THEN**
DISPATCH personnel to check for leaks. ☐
C. **IF** level is high, **THEN**
CHECK for RCIC, HPCI, Core Spray, or RHR draining to Suppression Pool, and **CHECK** 2-TR-64-161 and -162. ☐
D. **REFER TO** 2-OI-74, Section 8.0. ☐
E. **REFER TO** Tech Spec 3.6.2.2. ☐
F. **IF** level is above -1" or below -6.25" **AND NOT** in Mode 4 or Mode 5 **THEN** (otherwise N/A)
ENTER 2-EOI-2 Flowchart. ☐
G. **IF** level is above -1" or below -6.25" **AND** in Mode 4 or Mode 5 **THEN** (otherwise N/A)
1. **EVALUATE** plant conditions to **DETERMINE** if 2-EOI-2 entry is appropriate. ☐
2. **RECORD** actions in NOMS log. ☐

References: 2-45E620-3 2-47E610-64-1 GE 730E943-1
Technical Specifications 3.6.2.2

Excerpt from 2-EOI-2:

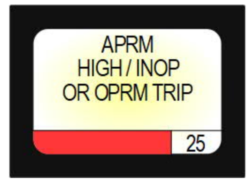


Examination Outline Cross-reference:	Level	RO	SRO
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor	Tier #	2	------
K6.04 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the following will have on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM:	K/A #	215005K6.04	
• Trip units	Importance Rating	3.1	------

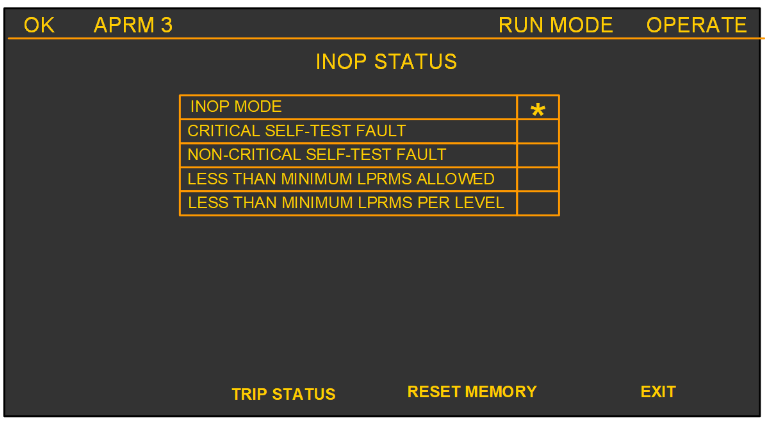
Proposed Question: **# 37**

Unit 1 is operating at 100% RTP with the following conditions:

- APRM HIGH/INOP OR OPRM TRIP (1-9-5A, Window 25) alarms



The Operator responds to find APRM 3 as indicated below:



In accordance with 1-OI-92B, Average Power Range Monitoring, which **ONE** of the following completes the statement below?

Given the conditions above, a trip output is sent to ___**(1)**___of the Voter Logic Modules and APRM 3 ___**(2)**_____ be bypassed.

A. (1) **ALL**
(2) can

B. (1) **ALL**
(2) can NOT

C. (1) **ONLY** 1
(2) can

D. (1) **ONLY** 1
(2) can NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 1-OI-92B, Average Power Range Monitors, each 2/4 Logic Module (Voter) receives trip status from each APRM instrument and APRM channel bypass information for each channel (APRM 1 thru 4 respectively). If one of the APRM channels is bypassed, the logic automatically reverts to 2/3 logic. For second part, only one APRM can be bypassed at a time.

B INCORRECT: The first part is correct (*See A*). The second part is incorrect but plausible in that the Power Range Nuclear Monitoring System contains Local Power Range Monitors (LPRMs), APRMs and Oscillation Power Range Monitors (OPRMs). Candidates often confuse the trip signals, setpoints, actions and capabilities with the system.

C INCORRECT**:** The first part is incorrect but plausible in that the candidate could confuse voter logic as it relates specifically to SRMs, IRMs, LPRMs or APRMs/OPRMs to generate a Rod Block and/or SCRAM signal. The second part is correct (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is incorrect but plausible (*See B*).

RO Justification: Tests the candidate's knowledge of the effect that a loss or malfunction has on the Average Power Range Monitoring System (APRM). This question is rated as C/A due to the requirement to assemble, sort, and integrate at least two different parts of the question to predict an outcome. It requires determining the appropriate APRM response as it relates to the complexity of all of the Reactor Power monitoring systems.

Technical Reference(s): 1-OI-92B, Rev. 14 (Attach if not previously provided)
1-ARP-9-5A, Rev. 28
1-OI-92, Rev. 10
1-OI-92A, Rev. 10

Proposed references to be provided to applicants during examination: **APRM HIGH/INOP OR OPRM TRIP (1-9-5A, Window 25), APRM 3 drawer screen indication for INOP STATUS**

Learning Objective: OPL171.148, Obj. 13d (As available)

Question Source:		
Bank #		
Modified Bank #		(Note changes or attach parent)
New	**X**	

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content:
55.41 **X**
55.43

Excerpt from 1-ARP-9-5A:

BFN Unit 1	Panel 9-5 1-XA-55-5A	1-ARP-9-5A Rev. 0028 Page 32 of 48

APRM HIGH/INOP OR OPRM TRIP

25

(Page 1 of 2)

Sensor/Trip Point:

APRM:

A. HIGH.
   1. (0.61W + 67.4%) with Rx Mode Sw. in RUN.
   2. 12% with Rx Mode Sw. NOT in RUN.
   3. 119% in any mode.
   4. (0.55(W-ΔW)+64.5%) Single loop operation with Mode Switch in RUN.

B. INOP.
   1. APRM Chassis Mode switch NOT in OPERATE.
   2. Loss of Input Power.
   3. Watchdog Timer timed out.
   4. Self Test Detected Critical Fault

OPRM TRIP:

C. Any one of four algorithms, period, growth, amplitude or CDA exceeds its trip value setpoint for an operable OPRM cell.

Sensor Location: Control Room Panel 1-9-14.

Probable Cause:

A. Flux level at or above setpoint.
B. Testing in progress.
C. Malfunction of sensor.
D. Control rod drop accident.
E. Thermal-Hydraulic Instability detected

Automatic Action:

A. Input signal to all four Voters, no output to RPS.
B. Full Reactor scram if two sensors actuate.

Operator Action:

A. CHECK alarm by multiple indications.
B. With SRO permission, BYPASS initiating channel to reset the alarm. REFER TO 1-OI-92B.

Continued on Next Page

Excerpts from 1-OI-92B:

BFN Unit 1	Average Power Range Monitoring	1-OI-92B Rev. 0014 Page 20 of 28

Attachment 1
(Page 1 of 6)

APRM/OPRM Trip Outputs and PRNMS Overview

TRIP SIGNAL	SETPOINT	ACTION
APRM Downscale	≥5%	1. Rod Block if REACTOR MODE SWITCH in RUN.
APRM Inop	1. APRM Chassis Mode not in OPERATE (keylock to INOP). 2. Loss of Input Power to APRM. 3. Self Test detected Critical Fault in the APRM instrument. 4. Firmware Watchdog timer has timed out	1. One Channel detected, no alarm or RPS output signal. 2. Two Channels detected, RPS output signal to all four Voters (Full Reactor Scram).
APRM Inop Condition	1. < 20 LPRMs in OPERATE, or < 3 per level.	1. <20 LPRMs total or <3 per level results in a Rod Block and a trouble alarm on the display panel. This does not yield an automatic APRM trip, but does, however, make the associated APRM INOP.
APRM High	1. DLO ≤ (0.61W + 62.0%) SLO ≤ (0.55(W-dw) + 58.5%) [W = Total Recirc Drive Flow in % rated]. 2. Neutron Flux Clamp Rod Block ≤ 113% 3. ≤ 8% APRM Flux.	1. Rod Block if REACTOR MODE SWITCH in RUN. 2. Rod Block if REACTOR MODE SWITCH in RUN. 3. Rod Block in all REACTOR MODE SWITCH positions except RUN.
APRM High High	1. DLO ≤ (0.61W + 67.4%) SLO ≤(0.55(W-dw) + 64.5%) [W = Total Recirc Drive Flow in % rated]. 2. ≤ 119% APRM FLUX. 3. ≤ 12% APRM Flux.	1. Scram. 2. Scram. 3. Scram in all REACTOR MODE SWITCH positions except RUN.
Recirc Flow Compare Recirc Flow Upscale	1. ≤ 5% mismatch between APRM Channels. 2. 107% Flow monitor upscale.	1. Flow compare inverse video alarm. 2. Rod Block.

TRIP SIGNAL	SETPOINT	ACTION
OPRM Inop	Less than 8 responsive OPRM cells and the Rx Mode Switch is in RUN	Annunciation Only
OPRM ALARM	Any one of two algorithms, period or CDA exceeds its pre-trip alarm setpoint for an operable OPRM cell.	Rod Block
OPRM Trip	Any one of four algorithms, period, growth, amplitude or CDA exceeds its trip value setpoint for an operable OPRM cell.	1. One Channel detected, no RPS output signal. 2. Two Channels detected, RPS output signal to all four Voters (Full Reactor Scram).
All OPRM setpoints are bypassed when the Reactor Mode Switch is not in RUN or the Reactor is not operating in the Power/Flow region where instabilities can occur (≥ 23% Power & <75% Recirc Drive Flow).		

BFN Unit 1	Average Power Range Monitoring	1-OI-92B Rev. 0014 Page 24 of 28

Attachment 1
(Page 5 of 6)

APRM/OPRM Trip Outputs and PRNMS Overview

3.0 2/4 LOGIC PROCESSING

A. Each 2/4 Logic Module (Voter) receives trip status from each APRM/OPRM instrument and APRM/OPRM channel bypass information for each channel. Based on this information, each 2/4 Logic Module determines whether to send trip signals to the RPS or not. Trip signals are sent if any two of the same type of non-bypassed APRM/OPRMs are providing trip signals to the 2/4 Logic Module. A separate, non-safety section of the 2/4 Logic Module serves as an interface between the associated APRM channel and other plant equipment outside Panel 1-9-14, such as the annunciator system, indicators on Panel 1-9-5, and the Reactor Manual Control System.

B. The 2/4 logic does not latch the input trip conditions. This means that no reset is required and no output trip signal occurs if one APRM/OPRM instrument generates a trip input to the logic and then clears before another APRM/OPRM instrument generates a trip. A trip output occurs only if two or more of the same type of inputs indicate a trip condition.

C. The 2/4 logic produces a trip state at two redundant trip outputs(X and Y) if any two of the same type of non-bypassed APRM/OPRM channels indicate a trip state. If one of the APRM/OPRM channels is bypassed, the logic automatically reverts to 2/3 logic. The 2/4 logic monitors for an active input from the channels. A non-bypassed input channel is processed as if in a trip state, if the input is not active.

BFN Unit 1	Average Power Range Monitoring	1-OI-92B Rev. 0014 Page 7 of 28

2.2.4 Miscellaneous Documents

SIL-111 Revision 1, Neutron Monitor and Flow Bypass Switch Modification

3.0 PRECAUTIONS AND LIMITATIONS

A. Each operable APRM channel requires a minimum of 20 LPRM inputs and at least 3 LPRM inputs per level (REFER to Tech Spec Section 3.3.1.1, Reactor Protection System (RPS) Instrumentation, and associated basis).

B. Only one APRM/OPRM can be bypassed at a time. The APRM BYPASS selector switch, 1-HS-92-7B/S3 on Panel 1-9-5, bypasses both the APRM AND OPRM for the channel selected.

C. In order to prevent an inadvertent rod withdrawal block or Reactor scram while operating the APRM BYPASS selector switch;1-HS-92-7B/S3:

Ensure that the previously bypassed channel returns to normal status by observing the blue BYPASSED light extinguished on all four of the Voters at Panel 1-9-14 for the applicable APRM channel.

After bypassing a channel, the applicable blue BYPASSED lights on all four of the Voters are required to be illuminated prior to testing, operating, or working on that channel.

Excerpts from 1-OI-92 and 1-OI-92A: Supports Distractors B(2), D(2):

BFN Unit 1	Source Range Monitors	1-OI-92 Rev. 0010 Page 20 of 20

Attachment 1
(Page 1 of 1)
SRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
SRM High	6.8 X 10^4 counts per second	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Inop	A. Module unplugged B. Mode switch not in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Downscale	5 counts per second	Rod block unless IRMs on range 3 (or higher) or REACTOR MODE SWITCH in RUN
SRM Detector Wrong Position	145 counts per second	Rod block unless detector full-in, IRMs on range 3 (or higher), or REACTOR MODE SWITCH in RUN
SRM High-High	2 x 10^5 counts per second	Scram if shorting links removed

BFN Unit 1	Intermediate Range Monitors	1-OI-92A Rev. 0010 Page 20 of 20

Illustration 1
(Page 1 of 1)
IRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
IRM High	>104.6 on 125 SCALE	Rod block unless REACTOR MODE SWITCH in RUN
IRM Inop	A. Module unplugged B. Mode switch not in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless REACTOR MODE SWITCH in RUN Reactor Scram unless REACTOR MODE SWITCH in RUN
IRM Downscale	<7.5 on 125 SCALE	Rod block unless IRMs on range 1 or REACTOR MODE SWITCH in RUN
IRM Detector Wrong Position	detector not full in	Rod block unless detector full-in, or REACTOR MODE SWITCH in RUN
IRM High-High	>116.4 on 125 SCALE	Reactor Scram unless REACTOR MODE SWITCH in RUN

Examination Outline Cross-reference:

295038 (EPE 15) High Offsite Radioactivity Release Rate / 9

EA1.05 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE:

- Post accident sample system (PASS): Plant-Specific

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295038EA1.05	
Importance Rating	3.0*	------

Proposed Question: **# 38**

Given the following Unit 1 conditions after a LOCA:

- Fuel damage has occurred
- An offsite release is in progress
- DRYWELL RADIATION HIGH (1-ARP-7C, Window 15) alarms

Which **ONE** of the following completes the statements below?

In accordance with the Alarm Response Procedures (ARPs), Drywell Radiation Level will be verified on Panel ___**(1)**___.

In accordance with the associated ARP, SLC is injected to ensure Suppression Pool pH remains above 7.0 in order to keep ___**(2)**___ from becoming airborne.

A. (1) 1-9-10
   (2) iodine

B. (1) 1-9-10
   (2) cesium

C. (1) 1-9-54
   (2) iodine

D. (1) 1-9-54
   (2) cesium

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that there are numerous Radiation Monitors on Panel 1-9-10 that are used for verification of radiation alarms. The second part is correct (*See C*).

B INCORRECT: The first part is incorrect but plausible (*See A*). The second part is incorrect but plausible in that, cesium is a fission product, and cesium-iodide is deposited into the Suppression Pool, which eventually causes pH to become acidic. The acidic pH results in a release of iodine.

C **CORRECT:** *(See attached)* In accordance with the given 1-ARP-9-7C, Window 15, DRYWELL RADIATION HIGH, operators check the alarm on 1-RR-90-272A on Panel 1-9-54 and the alarm on 1-RR-90-273 on Panel 1-9-55. For second part, in accordance with BFN FSAR Section 14.6.3.5, maintaining Suppression Pool pH above 7.0 ensures that the particulate iodine deposited in the Suppression Pool during a LOCA does not re-evolve and become airborne as iodine.

D INCORRECT: The first part is correct (*See C*). The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of where to monitor Post Accident Radiation Monitors as it relates to high radiation in the Drywell and the reason for an Alarm Response Procedure action responding to the radiation alarm. This question is rated as C/A due to the requirement to assemble, sort, and integrate two distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (3) The progression of an event.

Technical Reference(s): 1-ARP-9-7C, Rev.29 (Attach if not previously provided)

UFSAR Section 3.8 and 14.6 Amend. 27

Proposed references to be provided to applicants during examination: **DRYWELL RADIATION HIGH (1-ARP-7C, Window 15)**

Learning Objective: OPL171.033 Obj. 6 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1804 #58	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

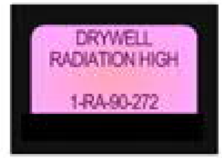
10 CFR Part 55 Content: 55.41 **X**

55.43

Copy of Bank Question:

Proposed Question: **# 58**

Given the following Unit 1 plant conditions after a Loss of Coolant Accident (LOCA) has occurred:



- EOI-1A, ATWS RPV Control and EOI-2, Primary Containment Control entered
- Operator at the Controls (OATC) is inserting the Control Rods
- Standby Liquid Control (SLC) is injecting
- Annunciator DRYWELL RADIATION HIGH (1-ARP-7C, Window 15) is in alarm

Subsequently, the Unit Supervisor exits EOI-1A and enters EOI-1, RPV Control.

Which ONE of the following completes the statement below?

Based on the current conditions, SLC is designed to inject to ________.

A. provide a high pressure source of injection to help recover RPV Water Level

B. minimize iodine release by maintaining Suppression Pool pH < 7.0

C. ensure sufficient negative reactivity added to Reactor to remain shutdown due to changes in core geometry

D. minimize iodine release by maintaining Suppression Pool pH ≥ 7.0

Excerpts from 1-ARP-9-7C:

BFN Unit 1	Panel 9-7 1-XA-55-7C	1-ARP-9-7C Rev. 0029 Page 20 of 41

DRYWELL RADIATION HIGH

1-RA-90-272

15

(Page 1 of 2)

Sensor/Trip Point:

1-RE-90-272A	100 R/HR
1-RE-90-273A	160 R/HR
1-RE-90-272B 1-RE-90-273B	Alarm setpoint disabled by raising adjustment to full scale value.

Sensor Location:
1-RM-90-272A, Panel 1-9-54
1-RM-90-273A, Panel 1-9-55
1-RM-90-272B, Panel 1-9-54
1-RM-90-273B, Panel 1-9-55

Probable Cause:
A. Noise spikes.
B. Sensor malfunction.
C. High radiation (post accident monitor).

Automatic Action: None

Operator Action:
A. **CHECK** alarm on 1-RR-90-272A on Panel 1-9-54 and 1-RR-90-273A on Panel 1-9-55.
B. **CHECK** 1-RR-90-256 for rising indication.
C. **ATTEMPT** to isolate equipment to stop source.
D. (NRC/C)**IF** the alarm is determined to be valid, **THEN**, **PERFORM** the following :
- **OPEN** UPSTREAM MSL DRAIN TO CONDENSER 1-FCV-001-0058.
- **OPEN** DOWNSTREAM MSL DRAIN TO CONDENSER 1-FCV-001-0059.
- **ENSURE** 1-PCV-001-0147 is Closed by taking STEAM SEAL REGULATOR, 1-HS-1-147 to **CLOSE**. (Panel 1-9-7)

Continued on Next Page

BFN Unit 1	Panel 9-7 1-XA-55-7C	1-ARP-9-7C Rev. 0029 Page 21 of 41

DRYWELL RADIATION HIGH 1-RA-90-272, Window 15
(Page 2 of 2)

Operator
Action: (Continued)

E. **IF** ALL the following conditions exist:
- Alarm is determined to be valid.
- The reactor will remain subcritical without boron injection under all conditions
- Leakage of primary coolant into primary containment is indicated

THEN within 2 hours of alarm, **INJECT** SLC for alternate source term control by placing SLC PUMP 1A/1B, 1-HS-63-6A in the START A OR START B position.

F. **REFER TO** EPIPs.

G. **IF** started at Operator Action Step E, **THEN WHEN** SLC tank reaches "0", **STOP** the running SLC Pump.

H. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References: 1-45E620-9-1, 2 0-47E610-90-2
Technical Specifications 3.3.3.1

Excerpt from BFN UFSAR Section 3.8:

3.8 STANDBY LIQUID CONTROL SYSTEM

3.8.1 Safety Objective

The safety objective of the Standby Liquid Control System is to provide a backup method, which is independent of the control rods, to make the reactor subcritical over its full range of operating conditions and provide sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage (see Section 14.6.3.5). Making the reactor subcritical is essential to permit the nuclear system to cool to the point where corrective actions can be carried out. Maintaining the suppression pool pH at or above 7.0 following a LOCA involving fuel damage supports the LOCA radiological dose analyses that do not consider the re-evolution of iodine to the containment atmosphere. The Standby Liquid Control System is classified as a special safety system.

3.8.2 Safety Design Basis

1. Backup capability for reactivity control shall be provided, independent of normal reactivity control provisions in the nuclear reactor, to shut down the reactor if the normal control is impaired so that cold shutdown (MODE 4) cannot be obtained with control rods alone.

2. The backup system shall have the capacity for controlling the reactivity difference between the steady-state rated operating condition of the reactor and the cold shutdown condition (MODE 4), including shutdown margin, to assure complete shutdown from the most reactive condition at any time in the core life.

3. The time required for actuation and effectiveness of the backup reactivity control shall be consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions (MODE 4). A scram of the reactor or operational control of fast reactivity transients is not specified to be accomplished by this system.

4. Means shall be provided by which the functional performance capability of the system components can be verified periodically under conditions approaching actual use requirements. Demineralized water, rather than the actual neutron absorber solution, is injected into the reactor to test the operation of all components of the redundant control system.

5. The neutron absorber shall be dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage, dilution, or imperfect mixing.

Excerpts from BFN UFSAR Section 14.6.3.3 and 14.6.3.5:

b. The primary to secondary containment leak rate was taken as two percent volume per day (232 cfh).

c. The four main steam lines are assumed to leak a total of 150 scfh which is the Technical Specification limit.

d. The containment vent system flow path operates for a period of 24 hours at a flow rate of 139 cfm at 10 days, 20 days, and 29 days post-accident. This flow is filtered via the SGTS filters.

e. The hardened containment vent isolation valves leak a total of 10 scfh to the independent release points above the Unit 1, 2, and 3 Reactor Buildings. Release associated with leakage from the hardened containment vent isolation valves is assumed to begin at 11 hours.

f. Twenty gpm ECCS leakage into secondary containment in accordance with NUREG-0800, Section 15.6.5, Appendix B.

g. No credit is taken for spray removal in the containment.

h. Natural removal rates for particulates in the drywell are based on the correlations of NUREG-CR-6604. For elemental iodine, the natural removal coefficients for removal of plateout are based on the expressions of SRP 6.5.2.

i. For the purpose of suppression pool pH control, the accident is assumed to be a recirculation line break.

Additionally, an analysis evaluated the suppression pool pH in the event of a DBA LOCA involving fuel damage. The objective of the analysis was to demonstrate that the suppression pool pH remains at or above 7.0; thus, ensuring that the particulate iodine (Cesium Iodide - CsI) deposited into the suppression pool during this event does not re-evolve and become airborne as elemental iodine.

The calculation methodology was based on the approach outlined in NUREG-1465 and NUREG/CR-5950. Specifically, credit was taken for sodium pentaborate solution addition to the suppression pool water as a result of SLCS operation.

The initial effects on suppression pool pH come from rapid fission product transport and formation of cesium compound, which would result in increasing the suppression pool pH. As radiolytic production of nitric acid and hydrochloric acid proceeds and these acids are transported to the suppression pool over the first days of the event, the suppression pool water would become more acidic. The buffering

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d. The core inventory release fractions, timing, and chemical form are those specified in Regulatory Guide 1.183. Table 14.6-7 gives the bounding core inventory of each isotope .

14.6.3.5 Fission Product Release From Primary Containment

Fission products are released from the primary containment to the secondary containment via primary containment penetration leakage at the Technical Specification leakage limit. Primary containment atmosphere is released via main steam isolation valve leakage to the high and low pressure turbines and the condenser. Primary containment atmosphere is released directly to the Standby Gas Treatment System during operation of the Containment Atmospheric Dilution (CAD) System. Primary containment atmosphere is released above the Units 1 and 2 Reactor Buildings via leakage of the Unit 1 and 2 hardened containment venting system isolation valves. Primary containment atmosphere is released to the top of the stack via leakage of the Unit 3 hardened wetwell vent isolation valves. The Emergency Core Cooling Systems (ECCS) leak into the secondary containment. The following assumptions were used in calculating the amounts of fission products released from the primary containment:

a. The primary containment minimum free volume (drywell and wetwell) is 278,400 ft^3. The drywell volume is 159,000 ft^3 and the torus gas space volume is 119,400 ft^3. The drywell torus gas space volumes are treated as separate volumes until after the activity release to the containment is complete and then these volumes are assumed to be well mixed. The activity release is entirely to the drywell.

b. The primary to secondary containment leak rate was taken as two percent volume per day (232 cfh).

c. The four main steam lines are assumed to leak a total of 150 scfh which is the Technical Specification limit.

d. The containment vent system flow path operates for a period of 24 hours at a flow rate of 139 cfm at 10 days, 20 days, and 29 days post-accident. This flow is filtered via the SGTS filters.

e. The Unit 3 hardened wetwell vent isolation valves leak a total of 10 scfh to the top of the offgas stack. The Unit 1 and 2 hardened containment vent isolation valves leak a total of 10 scfh to the independent release points above the Unit 1 and 2 Reactor Buildings. Release associated with leakage from the Unit 1 and 2 hardened containment vent isolation valves is assumed to begin at 11 hours.

f. Twenty gpm ECCS leakage into secondary containment in accordance with NUREG-0800, Section 15.6.5, Appendix B.

g. No credit is taken for spray removal in the containment.

h. Natural removal rates for particulates in the drywell are based on the correlations of NUREG-CR-6604. For elemental iodine, the natural removal coefficients for removal of plateout are based on the expressions of SRP 6.5.2.

i. For the purpose of suppression pool pH control, the accident is assumed to be a recirculation line break.

Additionally, an analysis evaluated the suppression pool pH in the event of a DBA LOCA involving fuel damage. The objective of the analysis was to demonstrate that the suppression pool pH remains at or above 7.0; thus, ensuring that the particulate iodine (Cesium Iodide - CsI) deposited into the suppression pool during this event does not re-evolve and become airborne as elemental iodine.

The calculation methodology was based on the approach outlined in NUREG-1465 and NUREG/CR-5950. Specifically, credit was taken for sodium pentaborate solution addition to the suppression pool water as a result of SLCS operation.

The initial effects on suppression pool pH come from rapid fission product transport and formation of cesium compound, which would result in increasing the suppression pool pH. As radiolytic production of nitric acid and hydrochloric acid proceeds and these acids are transported to the suppression pool over the first days of the event, the suppression pool water would become more acidic. The buffering effect of SLCS injection within several hours is sufficient to offset the effects of these acids that are transported to the pool. Sufficient sodium pentaborate solution is available to maintain the suppression pool pH at or above 7.0 for 30 days post-accident.

14.6.3.6 Fission Product Release to Environs

Secondary Containment Releases

The fission product activity in the secondary containment at any time (t) is a function of the leakage rate from the primary containment, the volumetric discharge rate from the secondary containment and radioactive decay. During normal power operation, the secondary containment ventilation rate is 75 air changes per day; however, the normal ventilation system is turned off and the Standby Gas Treatment System (SGTS) is initiated as a result of low reactor water level, high drywell pressure, or high radiation in the Reactor Building. Any fission product removal effects in the secondary containment such as plateout are neglected. The fission product activity released to the environs is dependent upon the fission product inventory airborne in the secondary containment, the volumetric flow from the secondary containment, and the efficiency of the various components of the SGTS.

Examination Outline Cross-reference:	Level	RO	SRO
295009 (APE 9) Low Reactor Water Level / 2	Tier #	1	------
AK2.04 (10CFR 55.41.7)	Group #	2	------
Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: • Reactor water cleanup	K/A #	295009AK2.04	
	Importance Rating	2.6	------

Proposed Question: **# 39**

Unit 1 is operating at 100% RTP when a LOCA occurs in the Drywell, with the following conditions:

- A manual Reactor SCRAM was inserted
- Primary Containment Isolation System (PCIS) Groups 2, 3, 6, and 8 have successfully isolated

Given the conditions above, which **ONE** of the following completes the statements below?

The Reactor Water Cleanup (RWCU) System isolated on **(1)** .

As a result of the RWCU Group Isolation signal, 1-FCV-69-12, RWCU RETURN ISOLATION VALVE **(2)** automatically close.

A. (1) 2.45 psig Drywell Pressure
(2) will

B. (1) 2.45 psig Drywell Pressure
(2) will NOT

C. (1) (+) 2 inches Reactor Water Level
(2) will

D. (1) (+) 2 inches Reactor Water Level
(2) will NOT

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that the 2.45 psig Drywell Pressure signal causes isolations in other PCIS Groups (2, 6, and 8). The reason for the Drywell Pressure isolation is to isolate Primary Containment in the event of a leak in the Drywell, and the numerous PCIS groups are often confused. The second part is correct (*See* C).

B INCORRECT**:** The first part is incorrect but plausible (*See A*). The second part is incorrect but plausible in that 1-FCV-69-12, RWCU RETURN ISOLATION VALVE is NOT a Primary Containment Isolation Valve (PCIV), so it is reasonable to believe that 1-FCV-69-12 would not automatically isolate.

C **CORRECT**: (*See attached*) RWCU will isolate as a result of the (+) 2 inches Reactor Water Level signal (PCIS Group 3 – given in the stem). This purpose of the Group 3 Isolation Signal is to isolate a Reactor Coolant System leak from the RWCU System. For second part, 1-FCV-69-12 will isolate when a Group 3 Isolation Signal is received, although it is NOT a PCIV.

D INCORRECT: The first part is correct (*See* C). The second part is incorrect but plausible (*See* B).

RO Level Justification: Tests the candidate's knowledge of the operation of the RWCU System given a Low Reactor Water Level isolation signal. This question is rated as C/A due to the requirement to assemble, sort, and integrate at least two different parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 1-AOI-64-2A, Rev.2 (Attach if not previously provided)
1-OI-69, Rev.79
OPL171.017, Rev. 21

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.013 Obj. 4d (As available)
OPL171.017, Obj. 2b

Question Source:	Bank #		
	Modified Bank #	BFN 1501 #22	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2015	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

Q 22

Unit 1 is at 100% Reactor power when a failure of the Reactor Feed Water Control System results in an automatic Reactor scram on low RPV water level.

Which ONE of the following completes the statements below?

The reason the RWCU pumps automatically trip is __ (1) __.

The RWCU Return Isolation valve, 1-FCV-69-12, will be __ (2) __.

A. (1) system flow less than 56 gpm
(2) closed

B. (1) system flow less than 56 gpm
(2) open

C. (1) RWCU INBD (OUTBD) SUCT Isolation valves, 1-FCV-69-1(2) NOT full open
(2) closed

D. (1) RWCU INBD (OUTBD) SUCT Isolation valves, 1-FCV-69-1(2) NOT full open
(2) open

Answer: C

Excerpt from 1-AOI-64-2A:

BFN Unit 1	Group 3 Reactor Water Cleanup Isolation	1-AOI-64-2a Rev. 0002 Page 4 of 7

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 3 Reactor Water Cleanup Isolation.

2.0 SYMPTOMS

NOTE

Reactor Water Cleanup System Isolation is initiated by any one of the following signals:

- Reactor Vessel Water Level Low (PCIS Group 3 isolation)
- RWCU Isolation Logic for Area Temperatures (PCIS Group 3 isolation)
- SLC Injection Initiation
- RWCU Non-Regenerative HX Discharge Temperature High

A. Any of the following annunciators in alarm:

1. RWCU ISOL LOGIC CHANNEL A(B) TEMP HIGH (1-XA-55-5B, Window 32 or 33)
2. RX VESSEL WTR LEVEL LOW HALF SCRAM (1-XA-55-4A, Window 2)
3. RWCU LEAK DETECTION TEMP HIGH (1-XA-55-3D, Window 17)
4. RWCU NON-REGENERATIVE HX DISCH TEMP HIGH (1-XA-55-4B, Window 17)
5. SLC INJECTION FLOW TO REACTOR (1-XA-55-5B, Window 14)

3.0 AUTOMATIC ACTIONS

A. 1-FCV-069-0001 closes as indicated at RWCU INBD SUCT ISOLATION VALVE, 1-HS-69-1.

B. 1-FCV-069-0002 closes as indicated at RWCU OUTBD SUCT ISOLATION VALVE, 1-HS-69-2A.

C. 1-FCV-069-0012 closes as indicated at RWCU RETURN ISOLATION VALVE, 1-HS-69-12A.

D. RWCU PUMPS 1A & 1B trip as indicated at 1-HS-69-4A-A & 1-HS-69-4B-A.

Excerpt from 1-OI-69:

BFN Unit 1	Reactor Water Cleanup System	1-OI-69 Rev. 0079 Page 16 of 148

3.7 RWCU Pump Trip Signals

A. The following signals will cause an automatic trip of an RWCU Pump:

1. Low flow 56 gpm (30 second time delay if the control switch is in NORMAL after start)
2. 1B Inboard Bearing high temperature 180°F (30 sec. time delay).
   1A pump Cooling water high temperature 140°F (30 sec. time delay).
3. RWCU INBD SUCT ISOLATION VLV, 1-FCV-069-0001 not full open.
4. RWCU OUTBD SUCT ISOL VLV, 1-FCV-069-0002 not full open.
5. RWCU SYS RETURN ISOL VLV, 1-FCV-069-0012 fully closed.
6. 480V Shutdown Board Undervoltage (5 second TD) or Overcurrent.

3.8 RWCU Isolation Signals

B. The following signals will cause an automatic isolation of the RWCU system:

1. Reactor water level low (Level 3).
2. Non-regenerative heat exchanger outlet high temperature 140°F.
3. RWCU Pump Room 1A area high temperature 148°F.
4. RWCU Pump Room 1B area high temperature 148°F.
5. Main Steam Tunnel/RWCU piping high temperature 197°F.
6. RWCU pipe trench area high temperature 131°F.
7. RWCU Heat Exchanger Room pipe chase area high temperature 166°F.
8. RWCU Heat Exchanger Room high temperature 139°F.
9. Standby Liquid Control system initiation.

Excerpt from OPL171.017 Lesson Plan:

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 21

Lesson Plan Content

Outline of Instruction	**Instructor Notes and Methods**
b) Group 2	
(1) This group includes Residual Heat Removal (RHR), drywell floor/equipment drain sump, and PSC Head Tank Pump suction valves.	ILT- 2g LOR- 2g 730E927-7,8
(2) The signals which will initiate a Group 2 Isolation are: (2-730E927-13)	730E927-13
(a) RPV low level (+2"or Level 3) (b) Drywell High Pressure (+2.45 psig) (c) Reactor High pressure (100 psig) (SDC) only.	NLO / NLOR- 2
c) Group 3	
(1) This group includes only the inboard and outboard Reactor Water Cleanup (RWCU) supply Isolation valves. FCV-69-12 also closes but is not PCIS Valve.	ILT- 2b LOR- 2b 2-730E927-7,8 2-730E927-13
(2) The signals which will initiate a Group 3 Isolation are as follows: **NOTE: NRHX HI Temp (TIS 69-11) Isolates the system, but is not PCIS (no alarm off of TIS-69-11)** (a) RPV Low Level (+2" or Level 3)	NLO / NLOR- 2
(b) RWCU Area High temperature (U1/U3 131-197ºF) (U2, 131-185ºF)	Unit Difference
(c) SLC Pump Hand Switch	

Examination Outline Cross-reference:	Level	RO	SRO
262001 (SF6 AC) AC Electrical Distribution	Tier #	2	------
K4.04 (10CFR 55.41.7)	Group #	1	------
Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following:	K/A #	262001K4.04	
• Protective relaying	Importance Rating	2.8	------

Proposed Question: **# 40**

Unit 3 is operating at 100% RTP, when a fault occurs on the Unit 3 Main Transformer.

Given the conditions above, which **ONE** of the following correctly describes the transfer scheme for 4KV Unit Boards 3A and 3B?

A. 4KV Unit Boards 3A **AND** 3B will remain de-energized until the alternate feeder breakers are **MANUALLY** closed.

B. 4KV Unit Boards 3A **AND** 3B will **FAST** transfer to alternate **AND** must be **MANUALLY** transferred to normal after power is restored.

C. 4KV Unit Boards 3A **AND** 3B will **FAST** transfer to alternate **AND** will **AUTOMATICALLY** transfer back to normal after power is restored.

D. 4KV Unit Boards 3A **AND** 3B will **SLOW** transfer to alternate **AND** must be **MANUALLY** transferred to normal after power is restored.

Proposed Answer**: B**

Explanation (Optional):

A INCORRECT**:** Incorrect but plausible in that the Browns Ferry 4160 V Electrical System and the transfer schemes for circuit breakers in this system are very complex and often confused by candidates. The transfer from alternate to normal is a manual transfer, so it is plausible that the candidate could believe that the transfer to alternate is a manual transfer.

B **CORRECT**: (*See attached)* In accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System, a high-speed transfer from normal to alternate is initiated by main transformer protective relays. 4KV Unit Boards 3A and 3B will automatically high-speed (fast) transfer from the normal to the alternate source on the fault. The transfer back to the USST (normal source) is manual only.

C INCORRECT: Incorrect but plausible in that while Unit Boards 3A and 3B will fast transfer to the alternate power supply when a fault occurs on the Main Transformer, the boards will not automatically transfer back to normal when power is restored to the normal supply.

D INCORRECT: Incorrect but plausible in that upon a loss of normal voltage, Unit Boards 3A and 3B will slow transfer to alternate, and must be manually transferred to normal when normal power is restored.

RO Level Justification: Tests the candidate's knowledge of the protective relaying in effect during a Main Transformer fault and the response of the Unit 3 Unit Board supply breakers. The Browns Ferry transfer schemes are very complex, and therefore this question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-OI-57A, Rev.166 (Attach if not previously provided)

OPL171.036, Rev.22

Proposed references to be provided to applicants during examination: NONE

OPL171.036 Obj. 8d (As available)

Question Source:	Bank #		
	Modified Bank #	ILT Exam Bank OPL171.036-08 007 #992	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

992. OPL171.036-08 007

A fault occurs out on the Athens 161Kv line, which is automatically cleared by breaker operation. The system line voltage is subsequently restored by automatic reclosure.

Which ONE of the following describes the response of the plant's electrical distribution system? (ASSUME NO OPERATOR ACTION)

A. Start Busses 1A **AND** 2A will **SLOW** transfer to alternate on undervoltage and **AUTOMATICALLY** return to the normal source 40 cycles after power is restored.

B✓ Start Busses 1A **AND** 2A **FAST** transfer to alternate and **AUTOMATICALLY** return to the normal source 40 cycles after power is restored.

C. Start Busses 1A **AND** 2A **FAST** transfer to alternate and must be **MANUALLY** transfered back to normal.

D. Start Bus 1A de-energizes until anternate feeder breaker is **MANUALLY** closed.

Excerpt from 0-OI-57A:

BFN Unit 0	Switchyard and 4160V AC Electrical System	0-OI-57A Rev. 0166 Page 190 of 210

Attachment 1
(Page 3 of 7)

Auxiliary Power Supplies and Bus Transfer Schemes

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE	REMARKS
7	4kV Recirculation Pump Boards: (Unit 1,2,3)			
	A. Recirc VFD set A	Unit SS TR A (BKRs 1122,1222, 1322)	Start Bus 2A (BKRs 1436,1438, 1442)	Automatic high speed transfer from the normal to the alternate source is initiated by main generator unit trip relays. Automatic delayed transfer from the normal to the alternate source is initiated by high-speed voltage relay. (Breakers are listed in Unit 1, 2, 3 order.)
	B. Recirc VFD set B	Unit SS TR A (BKRs 1124,1224, 1324)	Start Bus 2B (BKRs 1534,1536, 1538)	

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE	ALTERNATE 2	REMARKS
8	4kV Unit Boards (Unit 1)				Automatic high-speed transfer from the normal to the alternate source is initiated by main transformer protective relays. Automatic delayed transfer from the normal to the alternate source has been defeated by DCN W14030A On 1A, 1B, 2A and 2B Unit Bds. Unit Bds 1C, 2C, 3A, 3B, & 3C will still auto delay transfer from normal to alternate source. To permit use of the condensers as heat sinks for the possible case of normal power outage at the plant, the plant design includes a mode of operation for one running condenser circulating water pump. The controls provide for backfeeding from the shutdown boards to the 4160V unit boards A and B on each unit. For Unit 1 and 2, each shutdown bus and 4160V unit boards A and B provide for trip and lockout of unit transformer and start bus sources to the selected 4160V unit boards, before closure of the selected 4160V shutdown bus breaker which feeds the diesel generator power from the shutdown bus to the 4160V unit board. For Unit 3 only, the backfeed to the Unit boards A and B is directly from the shutdown boards.
	A. 4kV Unit Bd. 1A	Unit SS TR 1B (BKR 1112)	Start Bus 1A (BKR 1424)	Backfeed from shutdown buses	
	B. 4kV Unit Bd. 1B	Unit SS TR 1B (BKR 1114)	Start Bus 1B (BKR 1524)	Backfeed from shutdown buses	
	C. 4kV Unit Bd. 1C	Unit SS TR 1A (BKR 1116)	Start Bus 1B (BKR 1532)		
	4kV Unit Boards (Unit 2)				
	D. 4kV Unit Bd. 2A	Unit SS TR 2B (BKR 1212)	Start Bus 1A (BKR 1428)	Backfeed from shutdown buses	
	E. 4kV Unit Bd. 2B	Unit SS TR 2B (BKR 1214)	Start Bus 1B (BKR 1526)	Backfeed from shutdown buses	
	F. 4kV Unit Bd. 2C	Unit SS TR 2A (BKR 1216)	Start Bus 1A (BKR 1426)		
	4kV Unit Boards (Unit 3)				
	G. 4kV Unit Board 3A	Unit SS TR 3B (BKR 1312)	Start Bus 1A (BKR 1432)	Backfeed from shutdown bds	
	H. 4kV Unit Bd. 3B	Unit SS TR 3B (BKR 1314)	Start Bus 1B (BKR 1528)	Backfeed from shutdown bds	
	I. 4kV Unit Bd. 3C	Unit SS TR 3A (BKR 1316)	Start Bus 1A (BKR 1434)		

Excerpts from OPL171.036 Lesson Plan:

OPL171.036 , AC Power Distribution, Rev# 22

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
2. There are nine 4kV Unit Boards - three per unit. They are located in the turbine building on Elev. 604 (A and C Boards) and Elev. 586 (B Boards). The USSTs are the normal supply and start buses are the alternate supplies.	
a) USST A is the normal supply to 4kV Unit Board C and USST B is the normal power supply to 4kV Unit Boards A and B. (All Units)	ILT/NLO/NLOR Obj. 6.d LOR Obj.1.d
b) 4kV Start Bus 1A is the alternate power supply to 4kV Unit Boards 1A, 2A, 2C, 3A, and 3C.	
c) 4kV Start Bus 1B is the alternate power supply to 4kV Unit Boards 1B, 1C, 2B, and 3B.	
3. U1 and U2 4kV Unit Boards A and B supply power to 4kV Shutdown Buses 1 and 2 thereby providing off-site power to the Standby AC Power System. 3A and 3B 4kV Unit Boards supply power directly to the U3 4kV Shutdown Boards.	ILT/NLO/NLOR Obj. 6.a, 6.c, 7 LOR Obj.1.a, 1.c
4. <u>Control Room Indications</u> Indication of the 4kV Unit Boards' voltages and amperages are available on Panel 9-8. In addition, each boards pump motor amps is also available (except CRD pumps).	
5. <u>Transfer Schemes</u>	
a) General Operation The 4kV Unit Boards are normally fed from the Unit Station Service Transformers with an alternate feed from the 4kV Start Buses.	45E763
Transfer to the Start Buses may be manual or automatic but transfer back to the USST is manual only. All manual transfers and transformer trip-actuated transfers are fast transfers. Undervoltage relay-actuated transfer is delayed until bus voltage has decreased to 30% normal. A voltage relay prevents automatic transfer to a dead bus. The breakers are electrically interlocked to prevent paralleling the Unit and Common transformers.	ILT/NLO/NLOR Obj. 8.d LOR Obj. 2.d, OF-5 The 30% UV delayed transfer has been removed from 1A/1B, 2A/2B Unit Boards.

OPL171.036 , AC Power Distribution, Rev# 22

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
b) Automatic fast transfer of Unit Boards occurs on main transformer protective relaying or USST relaying.	0-OI-57A Attachment 1
To automatically fast transfer from normal to alternate: (1) normal feed breaker tripped (2) 43 selector switch in AUTO	
(3) Alternate feed line-side voltage available 27SU(x) (4) Alternate feeder breaker closes, provided no lock-outs are present.	27SUA - A UB 27SUB - B UB 27SUC - C UB
c) To automatically transfer from normal to alternate on undervoltage: (1) 43 transfer switch in AUTO (2) Alternate voltage available (3) Undervoltage on bus (bus voltage < 30%) and the normal feeder breaker trips. (4) Alternate feeder breaker closes after normal feeder breaker trips (as sensed by 52B finger) provided no lockouts are present.	
d) Manual transfer Unit Board supplies (1) 43 switch in MAN (2) Alternate feeder breaker control switch in CLOSE (3) Open normal feeder breaker and the alternate feeder breaker closes when 52b finger contact from normal feeder breaker closes.	

Examination Outline Cross-reference:	Level	RO	SRO
201006 (SF7 RWMS) Rod Worth Minimizer	Tier #	2	------
G2.4.6 (10CFR 55.41.10)	Group #	2	------
• Knowledge of EOP mitigation strategies.	K/A #	201006G2.4.6	
	Importance Rating	3.7	------

Proposed Question: **# 41**

Which **ONE** of the following completes the statements below in accordance with 1-EOI Appendix-1D, Insert Control Rods Using Reactor Manual Control System?

Rod Worth Minimizer (RWM) ____**(1)**____ required to be bypassed.

IF the Reactor SCRAM or ARI **CANNOT** be reset, **THEN** 1-SHV-085-586, CHARGING WATER ISOLATION VALVE is required to be ____**(2)**____ to manually drive Control Rods.

A. (1) is
(2) CLOSED

B. (1) is
(2) OPEN

C. (1) is NOT
(2) CLOSED

D. (1) is NOT
(2) OPEN

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with 1-EOI Appendix-1D, Rod Worth Minimizer is required to be manually bypassed to allow Control Rods to be inserted. For second part, if the Reactor SCRAM or ARI **CANNOT** be reset, then dispatch personnel to close 1-SHV-085-586, CHARGING WATER ISOLATION VALVE.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible if the candidate does not recall that 1-SHV-085-586 must be closed to assist in inserting Control Rods if the Reactor SCRAM or ARI **CANNOT** be reset.

C INCORRECT: First part is incorrect but plausible if the candidate does not recall that RWM is required to be manually bypassed to allow Control Rods to be inserted. Second part is correct *(See A)*.

D INCORRECT: First part is incorrect but plausible *(See C)*. Second part is incorrect but plausible *(See B)*.

RO Level Justification: Tests the candidate's knowledge of the Emergency Operating Instructions (EOIs) mitigation strategies as it relates to Rod Worth Minimizer. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): 1-EOI APPENDIX-1D, Rev. 1 (Attach if not previously provided)
1-EOI-1A, Rev. 2
OPL171.024, Rev. 16

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.202 Obj. 20 (As available)
OPL171.024, Obj. 6

Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Excerpt from 1-EOI Appendix-1D:

BFN UNIT 1	INSERT CONTROL RODS USING REACTOR MANUAL CONTROL SYSTEM	1-EOI APPENDIX-1D Rev. 1 Page 1 of 2

LOCATION: Unit 1 Control Room, Panel 1-9-5

ATTACHMENTS: 1. Core Position Map (√)

NOTE

This EOI Appendix may be executed concurrently with EOI Appendix 1A or 1B at SRO's discretion when time and manpower permit.

1. **VERIFY** at least one CRD pump in service. ______

NOTE

Closing 1-SHV-085-0586, CHARGING WTR ISOL, valve may reduce the effectiveness of EOI Appendix 1A or 1B.

2. IF .... Reactor Scram or ARI CANNOT be reset,

   THEN........... **DISPATCH** personnel to close 1-SHV-085-0586, CHARGING WTR ISOL (RB NE, El 565 ft). ______

3. **VERIFY** REACTOR MODE SWITCH in SHUTDOWN. ______

4. **BYPASS** Rod Worth Minimizer. ______

5. **REFER TO** Attachment 2 and **INSERT** control rods in the area of highest power as follows:

   a. **SELECT** control rod. ______

   b. **PLACE** CRD NOTCH OVERRIDE switch in EMERG ROD IN position UNTIL control rod is NOT moving inward. ______

   c. **REPEAT** Steps 5.a and 5.b for each control rod to be inserted. ______

6. WHEN.......... NO further control rod movement is possible or desired,

   THEN........... **DISPATCH** personnel to **VERIFY** open 1-SHV-085-0586, CHARGING WTR ISOL (RB NE, El 565 ft). ______

Excerpt from 1-EOI-1A:

Control Rod Insertion

RESET ARI
DEFEAT ARI logic trips if necessary (APPX 2)

ARC/Q-12

INSERT control rods using ANY Alternate Control Rod Insertion Methods (Table Q-2)

ARC/Q-13

Table Q-2
Alternate Control Rod Insertion Methods

CONDITIONS	METHODS	APPX
Scram valves failed to open	**DEENERGIZE** scram solenoids	1A
	VENT scram air header	1B
Scram valves opened but SDV is full	1. **RESET** scram **DEFEAT** RPS logic if necessary 2. **DRAIN** SDV 3. **RECHARGE** accumulators 4. **INITIATE** scram	1F
Manual control rod insertion methods	**DRIVE** control rods **BYPASS** RWM and **RAISE** CRD drive water differential pressure if necessary	1D
	RAISE CRD cooling water header dp	1G
	SCRAM individual control rods	1C
	VENT control rod over piston volumes	1E

1-EOI-1A	Page 1 of 1
ATWS RPV CONTROL UNIT 1 BROWNS FERRY NUCLEAR PLANT	
Rev: 2	

Excerpt from OPL171.024 Lesson Plan:

OPL 171.024 Rod Worth Minimizer (RWM) Rev.16

This list may be modified and saved by the Reactor Engineer or System Manager using these off-line functions.

c) This listing of Emerg. Insert Rods do not match the Emerg Shove Sheet Instructions. This RWM Display should not be used.

III. Operational Summary **

A. RWM Startup - Review OI-85 for placing RWM in service. RWM is only placed in service after officially entering Mode 2 – We cannot place mode switch in STARTUP simply to perform RWM SR – We must be in Mode 2. RWM is placed in service immediately after entering Mode 2 before any control rods are withdrawn.

B. RWM is typically not used during plant shutdowns because the plant is usually manually scrammed at approx. 30-40% power.

C. Review whose approval is required to perform a startup without RWM in service.

D. During EOI ATWS scenarios when manual control rod insertion is required, RWM is manually bypassed to allow any control rod to be inserted. Otherwise, RWM insert blocks will prevent control rod insertion using the EMERGENCY-IN Switch (HS-85-47).

E. Program Aborts

a) The RWM program can be aborted for a variety of reasons listed later.

Examination Outline Cross-reference:

G2.4.21 (10CFR 55.41.7)

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.4.21	
Importance Rating	4.0	------

Proposed Question: **# 42**

In accordance with 0-OI-48, Integrated Computer System (ICS), which **ONE** of the following completes the statement below?

The Safety Parameter Display System (SPDS) component of ICS ___**(1)**___ qualified as independent, decision-making instrumentation for operating the plant and will display parameters in ___**(2)**___ when the input data is **BAD**.

A. (1) is
(2) yellow

B. (1) is
(2) blue

C. (1) is NOT
(2) yellow

D. (1) is NOT
(2) blue

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate confuses the aspect of qualified vs. not qualified instrumentation pertaining to decision making as related to operating the plant. Second part is incorrect but plausible (*See C*).

B INCORRECT: First part incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT: First part is correct (*See D*). Second part is incorrect but plausible in that a multitude of SPDS display colors exist for specific Quality Codes associated with the status of components and/or parameters. Yellow indicates when a plant component and/or parameter point's current value has exceeded the defined High or Low Operating Limit with the assigned Quality Code of HALM or LALM.

D **CORRECT**: (*See attached*) In accordance with 0-OI-48, the Safety Parameter Display System (SPDS) component of the Integrated Computer Syste (ICS) is NOT qualified instrumentation, and therefore cannot be used as the sole guide in operating the plant. For second part, the Operator can determine the status of components and/or parameters having BAD display input by using the SPDS Summary Menu to see the impacted description and related Quality Codes. This will be indicated using the display color of blue.

RO Level Justification: Tests the candidate's knowledge of the parameters and logic used to assess the status of data input points from a multitude of systems indicating Emergency and/or Abnormal conditions exist. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to emergency plant conditions.

Technical Reference(s): 0-OI-48, Rev. 50 (Attach if not previously provided)
OPL171.099, Rev. 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.099, Obj. 7d (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1909 #74	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

ILT 1909 Written Exam

74. Unit 1 is operating at 100% RTP.

Which **ONE** of the following completes the statement below?

When assessing the EOI Exclusion Plot Status Boxes on the Safety Parameter Display System (SPDS) while using Integrated Computer System (ICS), ___**(1)**___ is expected to be colored RED.

In accordance with 0-OI-48, Integrated Computer System, the SPDS component of ICS ___**(2)**___ qualified as independent decision making instrumentation for operating the plant.

Note: Curve 5 – Drywell Spray Initiation Limit
Curve 6 – Pressure Suppression Pressure

A. **(1)** Curve 5
(2) is

B. **(1)** Curve 5
(2) is NOT

C. **(1)** Curve 6
(2) is

D. **(1)** Curve 6
(2) is NOT

Correct Answer: **B**

Excerpts from 0-OI-48:

BFN Unit 0	Integrated Computer System	0-OI-48 Rev. 0050 Page 7 of 50

3.0 PRECAUTIONS AND LIMITATIONS

A. The Safety Parameter Display System (SPDS) component of the Integrated Computer System (ICS) is **NOT** qualified instrumentation, and therefore cannot be used as the sole guide in operating the plant. It is a highly reliable operator's aid. Installed, qualified plant instruments are to be checked to back up any information shown by the SPDS before any actual plant manipulations are performed.

BFN Unit 0	Integrated Computer System	0-OI-48 Rev. 0050 Page 44 of 50

Attachment 1
(Page 1 of 2)

Quality Codes Descriptions

DISPLAY COLOR	CODE	DESCRIPTION
BLUE	UNK	Unknown; point NOT yet processed. If a point is deleted from processing when SAIPMS Is first activated, "UNK" quality code will be assigned. This quality code will also be displayed for calculated or derived points that have NOT yet cycled through their first processing period.
BLUE	DEL	Point has been deleted from processing. If a point was active when the SAIPMS software was activated, and was subsequently disabled from processing, the quality code "DEL" is assigned and no further EU conversion is attempted.
BLUE	INVL	DAS multiplexer error. See following discussion on Alarm Sensor Messages.
BLUE	RDER	Sensor read error. See following discussion on Alarm Sensor Messages.
BLUE	OTC	Open thermocouple. See following discussion on Alarm Sensor Messages.
BLUE	BAD	Input counts exceed sensor range. See following discussion on Alarm Sensor Messages.
BLUE	HRL	Point exceeds high reasonable limits. This condition is tested after EU conversion and if the value exceeds the defined High

BFN Unit 0	Integrated Computer System	0-OI-48 Rev. 0050 Page 43 of 50

8.5 **Determining Disabled ICS Alarms**

[1] **IF** it is desired to determine which ICS alarms are currently deleted from processing, **THEN**

SELECT Display Dele From Proc on the ICS SUMMARY MENU.

[2] **IF** it is desired to determine which ICS alarms are currently inhibited from display, **THEN**

SELECT Display Inhibited on the ICS SUMMARY MENU.

[3] **IF** it is desired to determine which ICS alarms are currently substituted, **THEN**

SELECT Display Substituted on the ICS SUMMARY MENU.

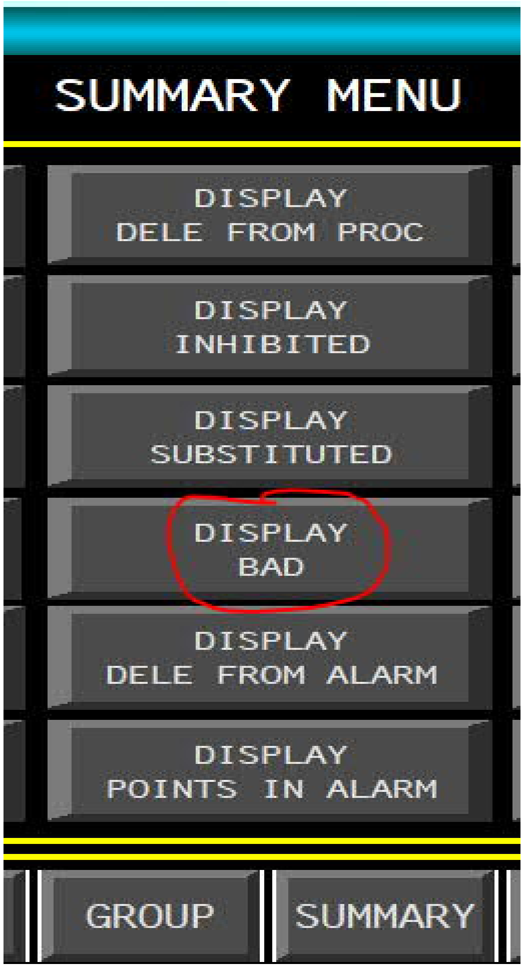
[4] **IF** it is desired to determine which ICS alarms are currently bad, **THEN**

SELECT Display Bad on the ICS SUMMARY MENU .

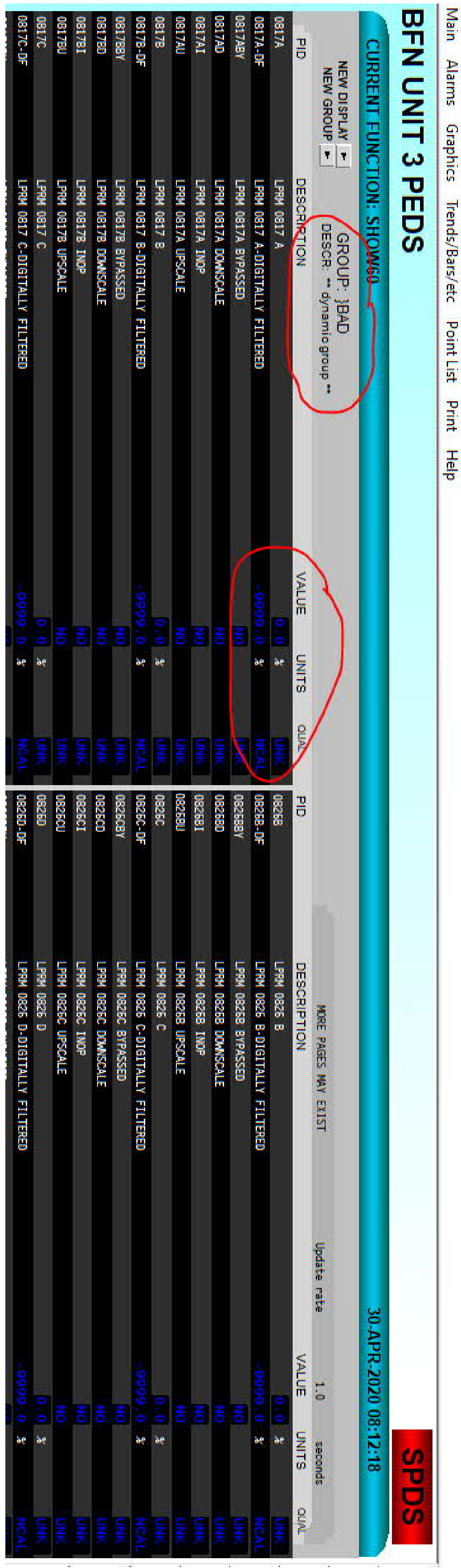
[5] **IF** it is desired to determine which ICS alarms are currently deleted from alarm, **THEN**

SELECT Display Dele From Alarm on the ICS SUMMARY MENU.

Excerpts from Safety Parameter Display System (SPDS) as referenced on previous page:



Illustrated BLUE for BAD data:



Supports Distractors A(2), C(2):

BFN Unit 0	Integrated Computer System	0-OI-48 Rev. 0050 Page 45 of 50

Attachment 1
(Page 2 of 2)

Quality Codes Descriptions

DISPLAY COLOR	CODE	DESCRIPTION
RED	HIHI	Point above high alarm limit. This condition is met when a point's current value has exceeded the defined High Alarm limit, and is assigned a quality code of "HIHI".
RED	LOLO	Point above low alarm limit. This condition is met when a point's current value has exceeded the defined Low Alarm limit, and is assigned a quality code of "LOLO".
RED	ALM	State/Change-of State alarm. Any logical-value point may be alarm monitored against either a defined logical state (that is, "TRUE", or "FALSE"), or a defined change-of-state condition (that is, "TRUE" to "FALSE", "FALSE" to "TRUE", or either state change). A quality code if "ALM" is assigned if the point meets any of the above conditions.
YELLOW	HALM	Point above high warning limit. This condition is met when a point's current value has exceeded the defined High Operating Limit, and is assigned a quality code of "HALM".
YELLOW	LALM	Point above low warning limit. This condition is met when a point's current value has exceeded the defined Low Operating Limit, and is assigned a quality code of "LALM".

Excerpt from OPL171.099 Lesson Plan:

OPL171.099, Integrated Computers System, Rev. 12

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
D. Safety Parameter Display System (SPDS)	
1. Purpose	Obj.4
a. The SPDS generates the information necessary for rapid detection of abnormal or emergency operating conditions.	Instructor: detail why this came about (TMI accident, for accident conditions monitoring)
b. The SPDS monitors the plants response to corrective actions.	
2. Component Description	

QA Record. Non-RP - Retain in ECM (Lifetime Retention)
RP LPs - Retain in ECM (Life of Nuclear Insurance Policy, plus 10 years)

Page 11 of 32

OPL171.099, Integrated Computers System, Rev. 12

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
a. The SPDS is part of the Integrated Computer System (ICS)	
b. The input points were chosen to support the EOI entry conditions.	

Examination Outline Cross-reference:

202002 (SF1 RSCTL) Recirculation Flow Control

K6.02 (10CFR 55.41.7)

Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM:

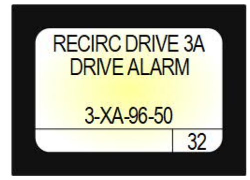
- D.C. power

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	202002K6.02	
Importance Rating	2.6	------

Proposed Question: **# 43**

Unit 3 is operating at 100% RTP with the following conditions:

- RECIRC DRIVE 3A DRIVE ALARM (3-9-4A, Window 32) alarms



Given the conditions above, which **ONE** of the following completes the statements below in accordance with 3-OI-68, Reactor Recirculation System?

If the alarm is due to a power cell experiencing a Direct Current (DC) bus undervoltage, the Variable Frequency Drive's (VFD) affected cell ___**(1)**___ bypass.

Once the respective cell is bypassed, the affected Recirc Pump will experience a speed drop of approximately ___**(2)**___.

A. (1) will automatically
(2) 200 rpm

B. (1) will automatically
(2) 345 rpm

C. (1) requires a manual
(2) 200 rpm

D. (1) requires a manual
(2) 345 rpm

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with the given alarm, a probable cause is a low voltage power supply problem. 3-OI-68, Reactor Recirculation System, states the Recirc Drive is equipped with a cell auto bypass feature which will automatically bypass a power cell should a failure be sensed such as DC bus overvoltage/undervoltage. For second part, if a cell bypasses while a Recirc Pump is running, a drop of approximately 200 rpm will occur. The drive will remain at the speed at which it stopped. Operator action is required to raise speed following a cell bypass.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that 345 rpm is the Recirc Drive speed associated with the Drive Start and Drive Shutdown demand as well as the Drive Minimum speed.

C INCORRECT: First part is incorrect but plausible in that once a cell automatically bypasses while a Recirc Pump is running and a drop of 200 rpm occurs, the Drive will remain at that speed. The Operator is required to manually raise speed following an automatic cell bypass. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the effect that a loss or malfunction of DC power will have on the Reactor Recirculation Flow Control System as it relates to the Variable Frequency Drives (VFD). This question is rated as memory due to strictly recalling facts related to the Reactor Recirculation System.

Technical Reference(s): 3-OI-68, Rev. 99 (Attach if not previously provided)
3-ARP-9-4A, Rev. 48
OPL171.007A, Rev. 11

Proposed references to be provided to applicants during examination: **RECIRC DRIVE 3A DRIVE ALARM (3-ARP-9-4A, Window 32)**

Learning Objective: OPL171. 007A, Obj. 3a_ (As available)

Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	**X**
Question History:	Last NRC Exam	

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Excerpt from 3-ARP-9-4A:

BFN Unit 3	Panel 9-4 3-X-55-4A	3-ARP-9-4A Rev. 0048 Page 43 of 47

RECIRC DRIVE 3A
DRIVE ALARM
3-XA-96-50
32

(Page 1 of 1)

Sensor/Trip Point:
Active Alarm (DRIVE ALARM)
Various alarms (Re-Flash on additional cell bypasses)
REFER TO ICS Group Display "GD @VFDADA" for a list of the alarms

Sensor Location: Elevation 639', Rx Building, Recirc Drive 3A

Probable Cause:
A. Drive problem
B. Coolant system problem
C. Low voltage power supply problem
D. Network communication problem

Automatic Action: None

Operator Action:
A. **REFER TO** ICS Group Display "GD @VFDADA"
B. **DETERMINE** cause of alarm.
C. **IF** the VFD has shutdown, **THEN** **NOTIFY** engineering to determine if a random reboot has occurred by reviewing the fault logs.
D. **IF** a random reboot is confirmed by engineering, **THEN** **START** the VFD per 3-OI-68.(Reference CR 1251203)
E. **IF** a problem with the cooling water system is indicated, **THEN** **ENSURE** proper operation of cooling water system.
F. **IF** the problem is conductivity in the cooling water system, **THEN** **ENSURE** demineralizer is in service.

Excerpts from 3-OI-68:

BFN Unit 3	Reactor Recirculation System	3-OI-68 Rev. 0099 Page 20 of 210

3.6 VFD's

A. Keying a Radio while the Recirc Drive control cabinet door is open has caused Recirc Drive trips.

B. The output of the Recirc Drive is monitored by 2 sets of protective relays, Motor Monitoring Relays (MMR) and Digital Frequency Relays (DFR). These relays are arranged in an energize to trip "2 out of 3" logic trip system. Either set (MMR or DFR) will trip the drive. If the relay loses power it will be unable to generate a trip to the logic system thus placing it in a "2 out of 2" logic trip system. If more than one MMR relay loses power, the trip from the MMR trip system will not generate a trip, the same is true for the DFR relays.

C. The Recirc Drive is equipped with a cell auto bypass feature. This feature will automatically bypass a power cell should a failure be sensed. The system will only bypass one cell in each phase and the bypass will only be the one phase in which the failure is sensed.

D. The signal which will cause a cell to automatically bypass are as follows:

1. Input fuse "CLEARED"
2. Over temperature Cell Heat Sink.
3. Communication fault.
4. Control power fault.
5. IGBT out of saturation.
6. DC bus overvoltage/undervoltage
7. DC bus capacitor fault
8. Cell control board hardware fault

E. The Recirc Drive are sized to allow one cell in each phase to be bypassed and still provide full output to the Recirc Pumps. With more than one cell in a phase bypassed the output from the drive may be reduced below the full output requirement of the Recirc Pumps.

BFN Unit 3	Reactor Recirculation System	3-OI-68 Rev. 0099 Page 21 of 210

3.6 VFD's (continued)

G. The following is a list of "DRIVE ALARM'S" and setpoints where applicable. The appropriate System Engineer should be contacted upon receipt of any of the following alarms that do not clear within 5 minutes.

1. Coolant conductivity ≥ 3µS for 12 seconds.
2. Coolant conductivity ≥ 5µS for 12 seconds.
3. Loss of conductivity signal

BFN Unit 3	Reactor Recirculation System	3-OI-68 Rev. 0099 Page 22 of 210

3.6 VFD's (continued)

23. Loss of coolant tank level signal.
24. Heat exchanger sump high level.
25. Emergency stop push button depressed.
26. Failure of a 24 vdc source.
27. Failure of a 15 vdc source.
28. Failure of a 12 vdc source.
29. Failure of a 5 vdc source.
30. Input/Output ground fault.
31. Failure of 120vac power to the Recirc Drive internal cooling fans.
32. Power cell is bypassed.
33. Speed demand in local control.
34. Power cell is bypassed. (Alarm Re-Flashes on additional cell bypass events)

BFN Unit 3	Reactor Recirculation System	3-OI-68 Rev. 0099 Page 23 of 210

3.6 VFD's (continued)

L. If a cell bypasses while a recirc pump is running, a drop of ≈200 rpm will occur. The drive will remain at the speed at which it stopped. Operator action is required to raise speed following a cell bypass.

M. During transfer of 4KV Recirc Boards 3A(3B) from Normal to Alternate and Alternate to Normal the expected response of the VFD's will be to lower speed approximately 100 rpm's and subsequently return to the previous demand at a rate of 1 rpm/sec.

N. If the standby coolant pump starts it will not shutdown until the condition which caused the pump start is corrected and the FAULT RESET push button is depressed to reset the alarms and faults.

O. After removing 4KV input power from a Recirc VFD, the cooling water system for that VFD should remain in operation for a minimum of two hours. Maintaining the VFD Cooling Water System in service for at least two hours after removing 4KV input power to the Recirc Drive precludes damaging the cooling system hoses due to overheating. This cooldown time is contingent on plant conditions. There may be times when the Unit SRO would want the VFD Cooling System shutdown sooner, such as during the winter when Reactor Building ambient temperature is low, or when cooling system conditions, such as leaks, warrant shutting down the cooling water system in a timely manner.

Supports Distractors B(2), D(2):

BFN Unit 3	Reactor Recirculation System	3-OI-68 Rev. 0099 Page 196 of 210

Attachment 7
(Page 1 of 1)

Recirc Drive Speed Control and Acceleration Rates

Recirc Drive Speed Demand	Acceleration Rate
Drive Start	Accelerate to 345 rpm at 100 rpm per second
Raise Slow	Accelerate 1 rpm at 1 rpm per second
Raise Medium	Accelerate 5 rpm at 1 rpm per second
Lower Slow	Decelerate 1 rpm at 1 rpm per second
Lower Medium	Decelerate 5 rpm at 1 rpm per second
Lower Fast	Decelerate 50 rpm at 15 rpm per second
75% Runback	Lowers speed to 1130 rpm at 15 rpm per second and limits speed to 1130 rpm if rpm is less than 1130 rpm.
28% Limiter	Lowers speed to 480 rpm at 15 rpm per second and limits speed to 480 rpm if rpm is less than 480 rpm. If the 28% limiter resets while the Recirc Drive is decelerating the rate of change will lower to 1 rpm/sec. When the 28% limiter resets selecting a RAISE and LOWER SPEED demand push button will stop the Recirc Drive from decelerating
Drive Shutdown	Decelerate to 345 rpm at 25 rpm per second, once 345 rpm is reached drive will shutdown.
Drive Minimum Speed	345 rpm
Drive Maximum Speed	Adjustable up to ~1682 rpm.
High Power Runback	Lowers speed to a steam flow of 14.8 Mlbm/hr or 700-750 rpm at 15 rpm per second. When Speed is within ~1% of desired setpoint the rate of change changes to 1 rpm per second.
Mid Power Runback	Lowers speed to a steam flow of 10.9 Mlbm/hr or 700-750 rpm at 15 rpm per second. When Speed is within ~1% of desired setpoint the rate of change changes to 1 rpm per second.
Core Flow Runback	Lowers speed to a core flow of 60 Mlbm/hr or 700-750 rpm at 15 rpm per second. When Speed is within ~1% of desired setpoint the rate of

Excerpt from OPL171.007A Lesson Plan:

OPL171.007A, Variable Frequency Drives (VFDs), Rev 11

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
I. General Introduction	(Notes optional at instructor discretion.)
Notes and Methods	
A. Introduce self and/or guest(s)	
1. Take Attendance	
2. Handout trainee feedback forms	
3. Introduce Topic / Goal	
4. Learning Objectives	
5. Description of how class will be conducted	
6. Evaluation Method	
7. What's In It For Me?	
II. Presentation	
A. System Description Overview	
1. Each of the Variable Frequency Drives (VFDs) consists of an Input Transformer, Power Cells, Control System, and a Cooling Water System. All of these major components are located inside or near cabinets on the 639' elevation of each Unit's Reactor Building.	Refer to PowerPoint presentation
2. The Input Transformer consists of one (1) primary and fifteen (15) secondaries. Each secondary feeds a separate power cell.	
3. The Power Cells receive the AC power from the transformer and convert it to DC. The DC is then inverted and controlled to provide an alternating output (AC).	
4. The VFD Control System controls the output of the power	

Examination Outline Cross-reference:

259002 (SF2 RWLCS) Reactor Water Level Control

K3.06 (10CFR 55.41.7)

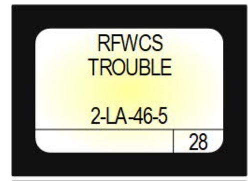
Knowledge of the effect that a loss or malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following:

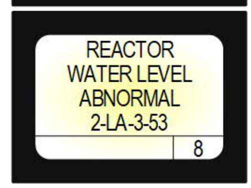
- Main turbine

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	259002K3.06	
Importance Rating	2.8	------

Proposed Question: **# 44**

Unit 2 is operating at 25% RTP with the following conditions:





- REACTOR FEEDWATER CONTROL SYSTEM TROUBLE (2-9-6C, Window 28) alarms
- REACTOR WATER LEVEL ABNORMAL (2-9-5A, Window 8) alarms
- Reactor Feedwater Pump (RFPT) Speeds are rising

Which **ONE** of the following completes the statements below?

Assuming **NO** Operator action is taken, the Main Turbine will automatically trip when Reactor Water Level reaches ___**(1)**___.

The Reactor Water Level Instruments that cause the Main Turbine Trip are ___**(2)**___.

Note: 3-203A-D, REACTOR WATER LEVEL NORMAL RANGE
3-208A-D, REACTOR WATER LEVEL NARROW RANGE

A. (1) 51 inches
(2) 3-203A-D

B. (1) 51 inches
(2) 3-208A-D

C. (1) 55 inches
(2) 3-203A-D

D. (1) 55 inches
(2) 3-208A-D

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that HPCI and RCIC trip when Reactor Water Level reaches 51 inches. The 51-inch trip for HPCI/RCIC and 55-inch trip for Main Turbine/Reactor Feedwater Pumps are often confused by ILT Candidates. The second part is incorrect but plausible in that there is a large number of Reactor Water Level Instruments that perform various functions to trip or initiate different systems and protective features. Level Instruments 3-203A-D are responsible for the low level (+2 inch) Reactor SCRAM and PCIS isolations (Group 2, 3, 6, and 8).

B INCORRECT: The first part is incorrect but plausible (*See* A). The second part is correct (*See* D).

C INCORRECT: The first part is correct (*See* D). The second part is incorrect but plausible (*See* A).

D **CORRECT:** (*See attached*) In accordance with 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low, RFPTs and Main Turbine trip at Level 8 (+ 55 inches) to protect the turbines from damage. For second part, in accordance with 2-OI-3, Reactor Feedwater System, Level Instruments 3-208A-D are responsible for tripping the Main Turbine, Reactor Feedwater Pumps (RFPTs), HPCI, and RCIC on high Reactor Water Level.

RO Level Justification: Tests the candidate's knowledge of the effect of high Reactor Water Level on the Main Turbine, and which instruments are used to cause a Main Turbine trip on high Reactor Water Level. This question is rated as Memory due to the fact that it requires the strict recall of facts concerning the high Reactor Water Level trip and the instrument numbers.

Technical Reference(s): (Attach if not previously provided)

2-AOI-3-1, Rev.23

2-OI-47, Rev.186

PIP-95-71

2-OI-71, Rev.74

2-OI-73, Rev.101

Proposed references to be provided to applicants during examination: **REACTOR FEEDWATER CONTROL SYSTEM TROUBLE (2-9-6C, Window 28), REACTOR WATER LEVEL ABNORMAL (2-9-5A, Window 8)**

Learning Objective: OPL171.010, Obj. 23 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1909 #65	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Copy of Bank Question:

ILT 1909 Written Exam

65. Unit 1 is operating at 24% RTP following a Startup with the following conditions:

- Reactor Feedwater Pump (RFPT) 1A is in service
- 1-LI-3-208D, RX WATER LEVEL NORMAL RANGE, failed **DOWNSCALE**

Subsequently, the Unit Operator observes 1-LI-3-208A, RX WATER LEVEL NORMAL RANGE drifting **DOWNSCALE**.

Given the conditions above, which **ONE** of the following completes the statements below?

If actual Reactor Water Level **RISES** to (+) 55 inches, the Main Turbine __**(1)**__ trip.

In accordance with OPDP-1, Conduct of Operations, a manual Reactor SCRAM __**(2)**__ required.

A. **(1)** will
(2) is

B. **(1)** will
(2) is NOT

C. **(1)** will NOT
(2) is

D. **(1)** will NOT
(2) is NOT

CORRECT: D

Excerpt from 2-AOI-3:

BFN Unit 2	Loss of Reactor Feedwater or Reactor Water Level High/Low	2-AOI-3-1 Rev. 0023 Page 6 of 16

2.0 SYMPTOMS (continued)

I. Possible mismatch between Feedwater Line Flows (FI -3-78A, FI-3-78B) during HPCI or RCIC inadvertent injections.

J. Possible Reactor power rise due to improved moderation from rising reactor water level and/or Feedwater flow.

3.0 AUTOMATIC ACTIONS

A. The following trip at greater than or equal to +55" (normal range level instruments 2-3-208A, 208B, 208C and 208D):

1. Main Turbine (this will cause a Reactor scram above 26% power).
2. RFPT.

B. The following trip at greater than or equal to +51" (normal range level instruments 2-3-208A, 208B, 208C, and 208D):

1. RCIC Turbine (if steam supply valve is open)
2. HPCI Turbine

Excerpt from 2-OI-47:

BFN Unit 2	Turbine-Generator System	2-OI-47 Rev. 0186 Page 16 of 280

3.2 Tech Specs

A. The COLR Thermal Limit analysis allows for a Turbine Bypass Valve and/or Recirc Pump Trip to be out of service. Therefore, EOC-RPT logic can remain disabled should a Turbine Bypass Valve become inoperative. Unit 2 TRM COLR should be referred to for appropriate Thermal limits and off-rated corrections when either Turbine Bypass out-of-service conditions exist or when Recirculation Pump Trip is out-of-service.

B. Placing LEVEL 8 TRIP BYPASS handswitch, 2-HS-047-0087/8 on Panel 2-9-31 (Aux Instrument Room), in BYPASS position, can inop Turbine High Water Level Trip Instrumentation. Unit SRO approval is required prior to placing this switch in BYPASS position. Refer to Tech Spec 3.3.2.2I

3.3 Turbine Trips

3.3.1 Automatic Trips

A. High Reactor Water Level Trip Logic:

1. High Reactor Water Level Trip logic for the Main Turbine at +55 inches is taken from Narrow Range level instruments 2-LI-3-208A, 2-LI-3-208B, 2-LI-3-208C, and 2-LI-3-208D. Logic is arranged in two channels; Channel A is fed from 2-LI-3-208A and 2-LI-3-208C. Channel B is fed from 2-LI-3-208B and 2-LI-3-208D. A trip of the Main Turbine and the RFPTs will occur if both instruments in Channel A, or Channel B sense reactor water level at ≥ +55 inches.

2. [TSAR/C] Operation of the Turbine-Generator with inoperable high level trip instrumentation may result in equipment damage or a personnel hazard. [Item D-91]

B. Condenser Vacuum Trip:

1. An ICS screen(CONDVAC) will display the current vacuum and the alarm and trip setpoint.

2. A Turbine trip on low main condenser vacuum occurs when one condenser section is at the trip setpoint AND another section is at the alarm setpoint.

Excerpt from PIP-95-71:

CONDENSING POT A (3-821)
A NORMAL CONTROL RANGE INSTRUMENTS (0 TO +60)

LOCAL PANEL	INSTRUMENT NUMBER	MASTER TRIP	SLAVE TRIP	PANEL NUMBER	FUNCTION
25-5C	LT-3-203A			9-83	RX SCRAM, PCIS 2,3,6,8
25-5C	LT-3-208A			9-81	MT,RFPT, RCIC TRIP
PNL-0426	LT-3-206			9-18	FW LEVEL CONTROL
PNL-0426	PT-3-207				FW LEVEL CONTROL
25-5C	PT-3-204A			9-81	ATWS/ARI-RPT INIT

Excerpt from 1-OI-71:

BFN Unit 2	Reactor Core Isolation Cooling	2-OI-71 Rev. 0074 Page 9 of 78

3.0 PRECAUTIONS AND LIMITATIONS

3.1 GENERAL PRECAUTIONS

A. Turbine controls provide for automatic shutdown of the RCIC turbine upon receiving any of the following signals (REFER TO Section 8.4 for auto actions):

1. High RPV water level (+51 in.); 579 in. above vessel zero. The RCIC TURBINE STEAM SUPPLY VLV, 2-FCV-71-8, and RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, will close at +51 in. and will RE-OPEN when RCIC re-initiates at -45 in. RPV water level.
2. Turbine overspeed (Mechanical, 121.0% of rated speed).
3. Pump low suction pressure (10 inches Hg vacuum).
4. Turbine high exhaust pressure (50 psig).
5. Any isolation signal.
6. Remote manual trip (RCIC TURBINE TRIP push-button, 2-HS-71-9A, depressed).

Excerpt from 1-OI-73:

BFN Unit 2	High Pressure Coolant Injection System	2-OI-73 Rev. 0101 Page 13 of 97

3.6 Trips

A. HPCI turbine automatically trips on the following:

1. RPV water level high at +51 inches.
2. Low pump suction pressure at 15" Hg Vacuum (4.7 sec time delay).
3. Turbine high exhaust pressure at 140 psig.
4. Any isolation signal.
5. Remote Manual HPCI TURBINE TRIP pushbutton, 2-HS-73-18A.

Examination Outline Cross-reference:	Level	RO	SRO
215001 (SF7 TIP) Traversing In-Core Probe	Tier #	2	------
K1.10 (10CFR 55.41.7)	Group #	2	------
Knowledge of the physical connections and/or cause effect relationships between TRAVERSING IN-CORE PROBE and the following:	K/A #	215001K1.10	
• Area radiation monitoring system: (Not-BWR1)	Importance Rating	2.6	------

Proposed Question: **# 45**

With respect to the MAXIMUM SAFE radiation value for Traversing Incore Probe (TIP), which **ONE** of the following completes the statements below?

If the MAXIMUM SAFE value for 2-RI-90-22A, TIP ROOM EL 565 RX BLDG is reached during TIP operation with the probes outside their shields, then automatic withdrawal to the in-shield position **___(1)___** occur.

A Group 8 Isolation **___(2)___** directly caused by the TIP Room MAXIMUM SAFE Value.

A. (1) will
(2) is

B. (1) will
(2) is NOT

C. (1) will NOT
(2) is

D. (1) will NOT
(2) is NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that the candidate could confuse the MAXIMUM SAFE radiation value for the TIP Room Radiation monitor of 100,000 mr/hr with other PCIS Groups which will automatically isolate at the Secondary Containment radiation value of 72 mr/hr. The second part is incorrect but plausible in that there are several ancillary systems that cause isolations of PCIS Groups, and these radiation montiors could cause isolations like any of the others. 2-AOI-64-2E, TIP Isolation, lists REACTOR BUILDING AREA RADIATION HIGH (2-ARP-9-3A, Window 22) as a symptom for the isolation. That alarm is caused by several Radiation Monitors, with 2-RI-90-22A being one of the monitors that brings in the Reactor Building High Radiation Alarm.

B INCORRECT: The first part is incorrect but plausible (*See A*). The second part is correct (*See D*).

C INCORRECT: The first part is correct (*See D*). The second part is incorrect but plausible (*See* A).

D **CORRECT**: *(See Attached)* In accordance with 2-EOI-3, the MAXIMUM SAFE value for the TIP Room Radiation monitor is 100,000 mrem/hr as indicated on 2-RI-90-22A. However, if TIP operation is in progress with the probes outside their shields when the MAXIMUM SAFE value is reached, then automatic withdrawal to the in-shield position will NOT occur. For second part, PCIS Group 8 Isolation is caused by (+) 2 inches Reactor Water Level and 2.45 psig in the Drywell, not the TIP Room Radiation level as indicated from 2-RI-90-22A. However, there are PCIS Group Isolations that will occur based on radiation levels.

RO Level Justification: Tests the candidate's knowledge of the effect of the Transversing In-Core Probe (TIP) Area Radiation Monitor's reading on EOI Entry and how that reading compares to other Radiation Monitors. This question is rated as memory due to strictly recalling facts related to the TIP Area Radiation Monitor, the EOI Entry Condition related to the Maximum Safe Reading, and how it compares to the other Secondary Containment Radiation Monitors.

Technical Reference(s): 2-EOI-3, Rev.17 (Attach if not previously provided)
2-AOI-64-2E, Rev.17
2-ARP-9-3A, Rev.55
2-AOI-64-2D, Rev. 34

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL141.034, Obj. 1 (As available)
OPL171.204, Obj. 2

Question Source: Bank #
(Note changes or attach parent)
New X

Question History: Last NRC Exam

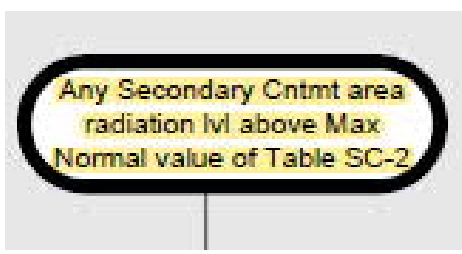
Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Excerpt from 2-EOI-3:

Table SC-2
Secondary Cntmt Area Radiation

Area	Applicable Radiation Indicators	Max Normal Value mR/hr	Max Safe Value mR/hr	Potential Isolation Sources
RHR sys I pumps	90-25A	Alarmed	1000	FCV-74-47, 48
RHR sys II pumps	90-28A	Alarmed	1000	FCV-74-47, 48
HPCI room	90-24A	Alarmed	1000	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	90-26A	Alarmed	1000	FCV-71-2, 3, 39
CS sys II pumps	90-27A	Alarmed	1000	None
Top of torus General area	90-29A	Alarmed	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB el 565 W	90-20A	Alarmed	1000	FCV-69-1, 2, 12 SDV vents & drains
RB el 565 E	90-21A	Alarmed	1000	SDV vents & drains
RB el 565 NE	90-23A	Alarmed	1000	None
TIP room	90-22A	Alarmed	100,000	TIP ball valve
RB el 593	90-13A, 14A	Alarmed	1000	FCV-74-47, 48
RB el 621	90-9A	Alarmed	1000	FCV-43-13, 14
Recirc MG sets	90-4A	Alarmed	1000	None
Refuel floor	90-1A, 2A 3A	Alarmed	1000	None



Excerpt from 2-AOI-64-2E:

BFN Unit 2	Traversing Incore Probe Isolation	2-AOI-64-2E Rev. 0017 Page 4 of 7

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 8, Traversing Incore Probe (TIP) Isolation and detection of a reactor coolant leak in a TIP guide tube.

2.0 SYMPTOMS

NOTES

1) PCIS Group 8 Isolation is initiated by any of the following signals:

- Reactor Vessel Water Level Low.
- Drywell High Pressure.

A. Any one or more of the following annunciators in alarm:

1. RX VESSEL WTR LEVEL LOW HALF SCRAM (2-XA-55-4A, Window 2). Group 8 Isolation only.
2. DRYWELL PRESSURE HIGH HALF SCRAM (2-XA-55-4A, Window 8). Group 8 Isolation only.
3. AIR PARTICULATE MONITOR RADIATION HIGH 2-RA-90-50A (2-XA-55-3A, Window 2). TIP guide tube leak only.
4. RX BLDG AREA RADIATION HIGH 2-RA-90-1D (2-XA-55-3A, Window 22). TIP guide tube leak only.

3.0 AUTOMATIC ACTIONS

[1] **IF** a Group 8 isolation occurred, **THEN** the following are automatic actions:

- **IF** TIP probes are outside their shields, **THEN** TIP withdrawal initiated to IN-SHIELD position.
- TIP Ball Valves receive a close signal, or close after TIP probes are withdrawn to their IN-SHIELD position.
- TIP Purge Valves closes (no indications provided).

Excerpt from 2-ARP-9-3A:

BFN Unit 2	Panel 9-3 2-XA-55-3A	2-ARP-9-3A Rev. 0055 Page 38 of 60

RX BLDG AREA RADIATION HIGH 2-RA-90-1D 22

(Page 1 of 3)

Sensor/Trip Point:

RI-90-4A	RI-90-24A	For setpoints
RI-90-9A	RI-90-25A	**REFER TO**
RI-90-13A	RI-90-26A	2-SIMI-90B.
RI-90-14A	RI-90-27A	
RI-90-20A	RI-90-28A	
RI-90-21A	RI-90-30A	
RI-90-R22A	RI-90-29A	
RI-90-23A		

Sensor Location:

RE-90-4	MG set area	Rx Bldg El 639' R-10 S-LINE
RE-90-9	Clean-up System	Rx Bldg El 621' R-9 T-LINE
RE-90-13	North Clean-up Sys	Rx Bldg El 593' R-9 P-LINE
RE-90-14	South Clean-up Sys	Rx Bldg El 593' R-9 S-LINE
RE-90-20	CRD-HCU West	Rx Bldg El 565' R-9 R-LINE
RE-90-21	CRD-HCU East	Rx Bldg El 565' R-13 R-LINE
RE-90-22	TIP Room	Rx Bldg El 565' R-12 P-LINE
RE-90-23	TIP Drive	Rx Bldg El 565' R-12 P-LINE
RE-90-24	HPCI Room*	Rx Bldg El 519' R-14 U-LINE
RE-90-25	RHR West	Rx Bldg El 519' R-8 U-LINE
RE-90-26	Core Spray-RCIC	Rx Bldg El 519' R-9 N-LINE
RE-90-27	Core Spray	Rx Bldg El 519' R-14 N-LINE
RE-90-28	RHR East	Rx Bldg El 519' R-14 U-LINE
RE-90-30	Fuel Storage Pool	Rx Bldg El 664' R-12 P-LINE
RE-90-29	Suppression Pool	Rx Bldg El 519' R-14 U-LINE

Probable Cause:

A. Radiation levels have risen above alarm setpoint.

B. Dry Cask Storage activities in progress (activities could affect rad levels sensed by 2-RE-90-30)

NOTE
Due to the location of the Rad Monitor in relation to the Test line in the HPCI Quad, the HPCI Room Rad Alarm may be received when the HPCI Flow test is in progress.

C. HPCI Flow Rate Surveillance in Progress.

Automatic Action: None

Continued on Next Page

Excerpt from 2-AOI-64-2D: Supports Distractors A(1), B(1)

BFN Unit 2	Group 6 Ventilation System Isolation	2-AOI-64-2D Rev. 0034 Page 4 of 15

1.0 PURPOSE

This procedure provides symptoms, automatic actions, and operator actions for a Group 6 Ventilation System Isolation.

2.0 SYMPTOMS

NOTES

1) PCIS Group 6 Isolation is initiated by any one of the following signals:
   - Reactor vessel water level at +2.0"
   - Drywell pressure at 2.45 psig
   - Reactor zone exhaust radiation at 72 mr/hr
   - Refuel zone exhaust radiation at 72 mr/hr
2) High Refuel Zone exhaust radiation causes only the automatic actions listed in Section 3.1.
3) Refuel Zone isolation due to Group 6 isolation initiated on Unit 1 or Unit 3.

A. Any one or more of the following annunciators in ALARM:

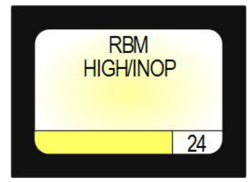
1. REACTOR ZONE EXHAUST RADIATION HIGH (2-XA-55-3A, Window 21)
2. REFUELING ZONE EXHAUST RADIATION MONITOR DOWNSCALE (2-XA-55-3A, Window 28)
3. REFUELING ZONE EXHAUST RADIATION HIGH (2-XA-55-3A, Window 34)
4. RX ZONE EXH RADIATION MONITOR DNSC (2-XA-55-3A, Window 35)
5. RX BLDG VENTILATION ABNORMAL (2-XA-55-3D, Window 3)
6. RX VESSEL WTR LEVEL LOW HALF SCRAM (2-XA-55-4A, Window 2)
7. DRYWELL PRESSURE HIGH HALF SCRAM (2-XA-55-4A, Window 8)
8. REACTOR ZONE DIFFERENTIAL PRESSURE LOW (2-XA-55-3D, Window 32)

Examination Outline Cross-reference:	Level	RO	SRO
215002 (SF7 RBMS) Rod Block Monitor	Tier #	2	------
A4.02 (10CFR 55.41.7)	Group #	2	------
Ability to manually operate and/or monitor in the control room: • RBM back panel switches, meters and indicating lights: BWR-3,4,5	K/A #	215002A4.02	
	Importance Rating	2.9	------

Proposed Question: **# 46**

Unit 1 is operating at 100% RTP with the following conditions:

- ROD BLOCK MONITOR HIGH/INOP (1-9-5A, Window 24) alarms



In accordance with 1-OI-92C, Rod Block Monitor, which **ONE** of the following completes the statement below?

The Rod Block Monitor (RBM) **HIGH** setpoint is ___**(1)**___.

If the Panel 1-9-14 RBM Chassis Mode Switch is **NOT** in OPERATE, a Control Rod Block ___**(2)**___ occur.

A. (1) 109.2%
(2) will NOT

B. (1) 109.2%
(2) will

C. (1) 119%
(2) will NOT

D. (1) 119%
(2) will

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible if the candidate confuses the seven different RBM Trip Outputs that cause a Control Rod Block. Additionally, this is further complicated with countless switches located on Main Control Room boards and panels.

B **CORRECT**: (*See attached*) In accordance with the given alarm response procedure, 1-OI-92C, Rod Block Monitor, is referenced. Both procedures state a RBM Upscale HIGH occurs when Simulated Thermal Power is greater than 80% at 109.2%. For second part, RBM INOP occurs when the local RBM chassis (located on Panel 1-9-14 in MCR) Mode Switch is NOT in OPERATE, causing a Control Rod Block.

C INCORRECT: First part is incorrect but plausible in that the RBM Upscale LOW occurs at a higher percentage and the Upscale HIGH occurs at a lower percentage. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

RO Level Justification: Tests the candidate's ability to manually operate switches and monitor indications in the Main Control Room associated with Rod Block Monitor. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome as it relates to numerous Rod Block Monitor Trip Outputs.

Technical Reference(s): OPL171.148, Rev. 15 (Attach if not previously provided)
1-OI-92C, Rev. 10
1-ARP-9-5A, Rev. 28

Proposed references to be provided to applicants during examination: **ROD BLOCK MONITOR HIGH/INOP (1-9-5A, Window 24)**

Learning Objective: OPL171.148 Obj. 26a, b (As available)

Question Source:

Bank #		
Modified Bank #		(Note changes or attach parent)
New	**X**	

Question History:

Last NRC Exam	

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Excerpts from 1-ARP-9-5A:

BFN Unit 1	Panel 9-5 1-XA-55-5A	1-ARP-9-5A Rev. 0028 Page 30 of 48

RBM
HIGH/INOP
24

(Page 1 of 2)

Sensor/Trip Point:

Relay K1
A10K1 in RBM
Interface module

Relay K3
A10K5 in RBM
Interface module

A. RBM HIGH
1. LOW SETTINGS 25% to 60% STP Alarms at 119.0%.
2. INTERMEDIATE SETTING >60% to 80% STP Alarms at 114.0%.
3. HIGH SETTING >80% STP Alarms at 109.2%.

B. RBM INOP
1. Mode switch NOT in operate.
2. RBM fails to null.
3. Less than 50% of assigned LPRM inputs
4. Loss of Input Power (Module unplugged).
5. More than one rod selected.
6. Critical self test fault detected
7. Circuit Board not in circuit.
8. Self Test Detected Critical Fault.

Sensor Location: Panel 1-9-14, MCR.

Probable Cause:
A. One or more sensor greater than or equal to setpoint.
B. SI (or SR) in progress.
C. Malfunction of sensor.

Automatic Action: Rod withdrawal block of presently selected control rod.

BFN Unit 1	Panel 9-5 1-XA-55-5A	1-ARP-9-5A Rev. 0028 Page 31 of 48

RBM HIGH/INOP, Window 24
(Page 2 of 2)

Operator Action: (Continued)

B. **IF NOT** moving control rods but a rod is selected, **THEN CHECK** Rod Out Permit light is NOT illuminated to ensure selected rod withdrawal is inhibited. (Receiving a rod block when not moving a rod, may be an indication of a failure of the RBM or an indication of a Reactor Power reduction with a rod selected.)
C. **NOTIFY** Reactor Engineer if additional assistance is required.
D. **REFER TO** 1-OI-92C, RBM Failure.
E. **REFER TO** Tech. Spec. Table 3.3.2.1-1, TRM Tables 3.3.2.1-1 and 3.3.4-1.

References: 1-AOI-92-1   1-45E620-6-1   1-107E5784-4, 20

Excerpts from 1-OI-92C:

BFN Unit 1	Rod Block Monitor	1-OI-92C Rev. 0010 Page 9 of 17

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Y. Rod Block Monitor uses STP to select one of three predefined setpoints to give Rod Blocks and alarms. A CONTROL ROD BLOCK is generated when the average of the selected LPRM detector signals reaches or exceeds the setpoint. {percents are in Rated Core Thermal Power} The Setpoints and alarms are as follows:

1. At less than 25% RBM is bypassed.
2. From 25% to 60% the alarm setpoint is 119.0%.
3. From > 60% to 80% the alarm setpoint is 114%.
4. At greater than 80% the alarm setpoint is 109.2%.

The Operators Display Assembly will also display the alarm setpoints in symbol form. Small up arrows (^) will appear on the bargraph; 1 up arrow is the LOW setpoint, 2 up arrows is the INTERMEDIATE setpoint, 3 up arrows is the HIGH setpoint.

BFN Unit 1	Rod Block Monitor	1-OI-92C Rev. 0010 Page 15 of 17

Illustration 1
(Page 1 of 1)

RBM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
RBM Downscale	≤ 92%	Rod Block
RBM Inop	1. Local RBM Chassis Mode Switch NOT in OPERATE. 2. LOSS of Input Power (Module unplugged). 3. RBM fails to null. 4. Less than 50% of LPRM inputs operable for rod selected. 5. Null sequence in progress. 6. Self-Test Detected Critical Fault. 7. More than one rod selected.	Rod Block
RBM Upscale 1. Low 2. Intermediate 3. High 4. Recirc Flow Upscale	 1. 25% to 60% STP alarms at 119.0%. 2. >60% to 80% STP alarms at 114.0%. 3. >80% STP alarms at 109.2%. 4. 107%	Rod Block
1. Recirc Flow Compare	1. 5% mismatch between APRMs	Flow Compare Inverse Video Alarm.

Examination Outline Cross-reference:	Level	RO	SRO
223001 (SF5 PCS) PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES	Tier #	2	------
K3.01 (10CFR 55.41.7)	Group #	2	------
Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES will have on the following:	K/A #	223001K3.01	
• Secondary Containment	Importance Rating	3.6	------

Proposed Question: **# 47**

Unit 2 was operating at 100% RTP when a LOCA occurred.

Subsequently,

- All Drywell Sprays have failed
- 2-FCV-64-222, HARDENED CONTAINMENT VENT OUTBOARD VALVE, failed in the CLOSED position
- The preferred vent path is **NOT** available.

Which **ONE** of the following completes the statement below?

In accordance with 2-EOI Appendix-13, Emergency Venting Primary Containment, the containment vent path will be from the ___**(1)**___ and Reactor Building ventilation ductwork failure ___**(2)**___ expected.

A. (1) Drywell
(2) is

B. (1) Drywell
(2) is NOT

C. (1) Suppression Chamber
(2) is

D. (1) Suppression Chamber
(2) is NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** In accordance with 2-EOI-Appendix-13, Section 1.0 [3.0] if the Suppression Chamber vent path is unavailable (preferred vent path), then Primary Containment will be vented via the Drywell. With the failure of 2-FCV-64-222, HARDENED CONTAINMENT VENT OUTBOARD VALVE, the Suppression Chamber vent path will not be available. For second part, the Caution in 2-EOI-Appendix-13 states that the actions to vent the Drywell will fail ductwork inside the Reactor Building and may fail Secondary Containment integrity. This is not applicable if venting from the Suppression Chamber.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that this procedure is infrequently performed, and the candidate may not correlate venting the Drywell with failing Reactor Building ductwork.

C INCORRECT: First part is incorrect in that the Suppression Chamber vent path is through 2-FCV-64-222. With this failure 2-EOI-Appendix-13 states to vent the Drywell. The second part is correct (*See A*).

D INCORRECT: The first part is incorrect but plausible *(See C).* The second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's knowledge of the effect of Primary Containment emergency venting has on Secondary Containment. This question is rated as C/A due to the requirement to assemble, sort, and integrate at multiple parts of the question to predict an outcome.

Technical Reference(s): 2-EOI-Appendix-13 Rev. 9 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203 Obj. 9 (As available)

Question Source:			
	Bank #		
	Modified Bank #	BFN 1804 #72	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

ILT 1804 Written Exam

72. Which one of the following completes the statement below in accordance with 2-EOI Appendix-13, Emergency Venting Primary Containment?

It ___(1)_____ permitted to exceed the Off-Gas Release Rate Limits when Emergency Venting of Primary Containment is in progress.

The **PREFERRED** Primary Containment vent path is from the ___(2)_____.

A. **(1)** is
(2) Drywell

B. **(1)** is NOT
(2) Drywell

C. **(1)** is
(2) Suppression Chamber

D. **(1)** is NOT
(2) Suppression Chamber

Excerpts from 2-EOI-Appendix 13:

BFN Unit 2	Emergency Venting Primary Containment	2-EOI Appendix-13 Rev. 0010 Page 3 of 16

1.0 INSTRUCTIONS

LOCATION:	Unit 2 Control Room
ATTACHMENTS	1. Tools and Equipment 2. Vent System Overview 3. Hardened Vent Flow Path 4. HCVS Battery Alignment 5. HCVS Nitrogen Bottle Alignment 6. HCVS Operation from the Remote Operating Station

[1] **NOTIFY** SHIFT MANAGER/SED of the following:

- Emergency Venting of Primary Containment is in progress. ☐
- Off-Gas Release Rate Limits will be exceeded. ☐

NOTES

1) HARDENED CONTAINMENT VENT VALVES 2-FCV-64-221 and 222 may be operated locally with handwheels (U2 RB el. 580, top of clean room, southwest corner).

2) If an alternate DC power source is needed for the HCVS valve solenoids, Att. 4 HCVS Battery Alignment may be performed.

3) If an alternate air supply is needed for the HCVS valves, Att. 5 HCVS Nitrogen Bottle Alignment may be performed.

4) If required, HCVS valves may be operated from the U3 DG building using Att. 6, HCVS Operation from the Remote Operating Station.

[2] **VENT** the Suppression Chamber as follows (Panel 2-9-3):

[2.1] **IF** EITHER of the following exists:

- Suppression Pool water level CANNOT be determined to be below 26 ft., **OR**
- Suppression Chamber CANNOT be vented, **THEN**

CONTINUE in this procedure at Step 1.0[3]. ☐

BFN Unit 2	Emergency Venting Primary Containment	2-EOI Appendix-13 Rev. 0010 Page 4 of 16

1.0 INSTRUCTIONS (continued)

[2.2] **PLACE** keylock switch 2-HS-64-222B, HARDENED CONTAINMENT VENT OUTBD PERMISSIVE, in PERM. ☐

[2.3] **CHECK** blue indicating light above 2-HS-64-222B, HARDENED CONTAINMENT VENT OUTBD PERMISSIVE, illuminated. ☐

[2.4] **OPEN** 2-FCV-64-222, HARDENED CONTAINMENT VENT OUTBD ISOL VLV. ☐

[2.5] **PLACE** keylock switch 2-HS-64-221B, HARDENED CONTAINMENT VENT INBD PERMISSIVE, in PERM. ☐

[2.6] **CHECK** blue indicating light above 2-HS-64-221B, HARDENED CONTAINMENT VENT INBD PERMISSIVE, illuminated. ☐

[2.7] **OPEN** 2-FCV-64-221, HARDENED CONTAINMENT VENT INBD ISOL VLV. ☐

[2.8] **CHECK** Drywell and Suppression Chamber Pressure lowering. ☐

[2.9] **MAINTAIN** Primary Containment Pressure below 55 psig using 2-FCV-64-222, HARDENED CONTAINMENT VENT OUTBD ISOL VLV, as directed by SRO. ☐

[3] **IF** Suppression Chamber vent path is NOT available, **THEN**

VENT the Drywell as follows:

[3.1] **NOTIFY** SHIFT MANAGER/SED that Secondary Containment integrity failure is possible. ☐

[3.2] **NOTIFY** RADCON that Reactor Building is being evacuated due to imminent failure of Primary Containment vent ducts. ☐

[3.3] **EVACUATE** ALL Reactor Buildings using P.A. System. ☐

[3.4] **START** ALL available SGTS trains. ☐

[3.5] **ENSURE CLOSED** 2-FCV-64-36, DW/SUPPR CHBR VENT TO SGT (Panel 2-9-3). ☐

BFN Unit 2	Emergency Venting Primary Containment	2-EOI Appendix-13 Rev. 0010 Page 5 of 16

1.0 INSTRUCTIONS (continued)

[3.6] **ENSURE OPEN** the following dampers (Panel 2-9-25):

- 2-FCO-64-40, REACTOR ZONE EXH TO SGTS ☐
- 2-FCO-64-41, REACTOR ZONE EXH TO SGTS ☐

[3.7] **ENSURE CLOSED** 2-FCV-64-29, DRYWELL VENT INBD ISOL VALVE (Panel 2-9-3 or Panel 2-9-54). ☐

[3.8] **DISPATCH** personnel to Unit 2 Auxiliary Instrument Room to perform the following:

[3.8.1] **REFER TO** Attachment 1 and **OBTAIN** one 12-in. Banana Jack Jumper from EOI Equipment Storage Box. ☐

[3.8.2] **LOCATE** terminal strip DD in 2-PNLA-009-0043, Front. ☐

[3.8.3] **JUMPER** DD-76 to DD-77 (2-PNLA-009-0043). ☐

[3.8.4] **NOTIFY** Unit Operator that jumper for 2-FCV-64-30, DRYWELL VENT OUTBD ISOLATION VLV, is in place. ☐

[3.9] **ENSURE OPEN** 2-FCV-64-30, DRYWELL VENT OUTBD ISOLATION VLV (Panel 2-9-3). ☐

CAUTIONS

1) The following step will fail ductwork inside Secondary Containment and may fail Secondary Containment Integrity.

2) Off-Gas Release Rate Limits will be exceeded.

[3.10] **PLACE** keylock switch 2-HS-84-36, SUPPR CHBR/DW VENT ISOL BYP SELECT, to DRYWELL (Panel 2-9-54). ☐

[3.11] **ENSURE OPEN** 2-FCV-64-29, DRYWELL VENT INBD ISOL VALVE (Panel 2-9-54). ☐

[3.12] **CHECK** Drywell and Suppression Chamber pressure lowering. ☐

Examination Outline Cross-reference:	Level	RO	SRO
234000 (SF8 FH) Fuel-Handling Equipment	Tier #	2	------
K4.02 (10CFR 55.41.7)	Group #	2	------
Knowledge of FUEL HANDLING EQUIPMENT design feature(s) and/or interlocks which provide for the following:	K/A #	234000K4.02	
• Prevention of control rod movement during core alterations	Importance Rating	3.3	------

Proposed Question: **# 48**

The following conditions exist on Unit 3:

- 3-HS-99-5A-S1, REACTOR MODE SWITCH is in REFUEL
- **ALL** Control Rods are inserted
- The Refueling Bridge Operator grappled a fuel bundle and raised the grapple
- The fuel bundle was then moved towards the Reactor Core

In accordance with 0-GOI-100-3A, Refueling Operations (In-Vessel Operations), which **ONE** of the following completes the statement below?

Given the conditions above, as the Refueling Bridge moves towards the Reactor Core, it ____**(1)**____ the Core **AND** a Control Rod Block ____**(2)**____ occur.

A. (1) continues over
(2) will

B. (1) continues over
(2) will NOT

C. (1) stops before it reaches
(2) will

D. (1) stops before it reaches
(2) will NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 0-GOI-100-3A, Refueling Operations (In-Vessel Operations) the Refueling Bridge will continue over the Core given none of interlocks have been met to stop it. For second part, a Control Rod Block will occur due to meeting the interlock criteria: any platform hoist loaded, Refueling platform near or over the Core with the MODE SWITCH in REFUEL.

B INCORRECT: The first part is correct (*See A*). The second part is incorrect but plausible in that the Refueling Rod Block and Platform interlocks are easily confused. A Control Rod Block will NOT occur if grapple was unloaded.

C INCORRECT: First part is incorrect but plausible in that the Refueling Bridge will stop before it reaches the Core if one Control Rod was withdrawn and a second Control Rod was selected. The second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of Fuel Handling equipment design features and interlocks as it relates to Control Rod Blocks and Refueling Bridge Platform during Refueling. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-GOI-100-3A, Rev. 86 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.053 Obj. 5 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.053-04 001, #1679	
Modified Bank #		(Note changes or attach parent)
New		
Last NRC Exam		

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

1679. OPL171.053-04 001

On Unit 3, the Mode Switch is in REFUEL **AND ALL** control rods are inserted. The Refueling Bridge operator grappled a fuel bundle, raised the grapple, **AND** commenced moving the bundle towards the core.

Which ONE of the following describes what will result as the Refueling Bridge moves towards the core?

The Refueling Bridge _____.

A✓ continues over the core **AND** initiates a control rod block

B. continues over the core **AND** causes **NO** protective actions

C. stops before it reaches the core **AND** initiates a control rod block

D. stops before it reaches the core **AND** causes **NO** protective actions

Excerpts from 0-GOI-100-3A:

BFN Unit 0	Refueling Operations (In-Vessel Operations)	0-GOI-100-3A Rev. 0086 Page 18 of 218

3.3 Refuel Bridge Operation (continued)

F. A Rod Block will occur if any of the following conditions are met:

1. Any platform hoist loaded or main grapple **NOT** full up with the platform near or over the core with the Mode Switch in REFUEL.

2. Service platform dummy plug not installed.

3. One rod withdrawn and a second rod selected with the Mode Switch in REFUEL.

4. Platform near or over the core with the Mode Switch in STARTUP.

5. One rod withdrawn with the Mode Switch in REFUEL. (As long as all rods are full in, a rod may be selected and withdrawn as long as rod withdrawal signal is present. When the initial withdrawal signal ceases this block will enforce.)

BFN Unit 0	Refueling Operations (In-Vessel Operations)	0-GOI-100-3A Rev. 0086 Page 17 of 218

3.3 Refuel Bridge Operation

A. For safety and cleanliness reasons, fuel handling areas and equipment travel paths shall remain free of unnecessary equipment and control or power cables that cross travel paths shall be elevated or diverted.

B. Refueling equipment operators shall **NOT** rely solely on limit switches and stops for termination of refueling equipment travel. These switches and devices shall serve as backup protection. The primary means for stopping travel is operator action.

C. When operating the refuel bridge in any speed other than JOG, ensure that the grapple or devices being transported have adequate clearance above items stored in the SFSP and Reactor Cavity.

D. Bridge travel toward the core will be stopped if any of the following conditions are met (except when interlocks are jumpered out by instruction in this procedure):

1. Any platform hoist loaded or main grapple **NOT** full up and all rods **NOT** full in with the platform near or over the core.

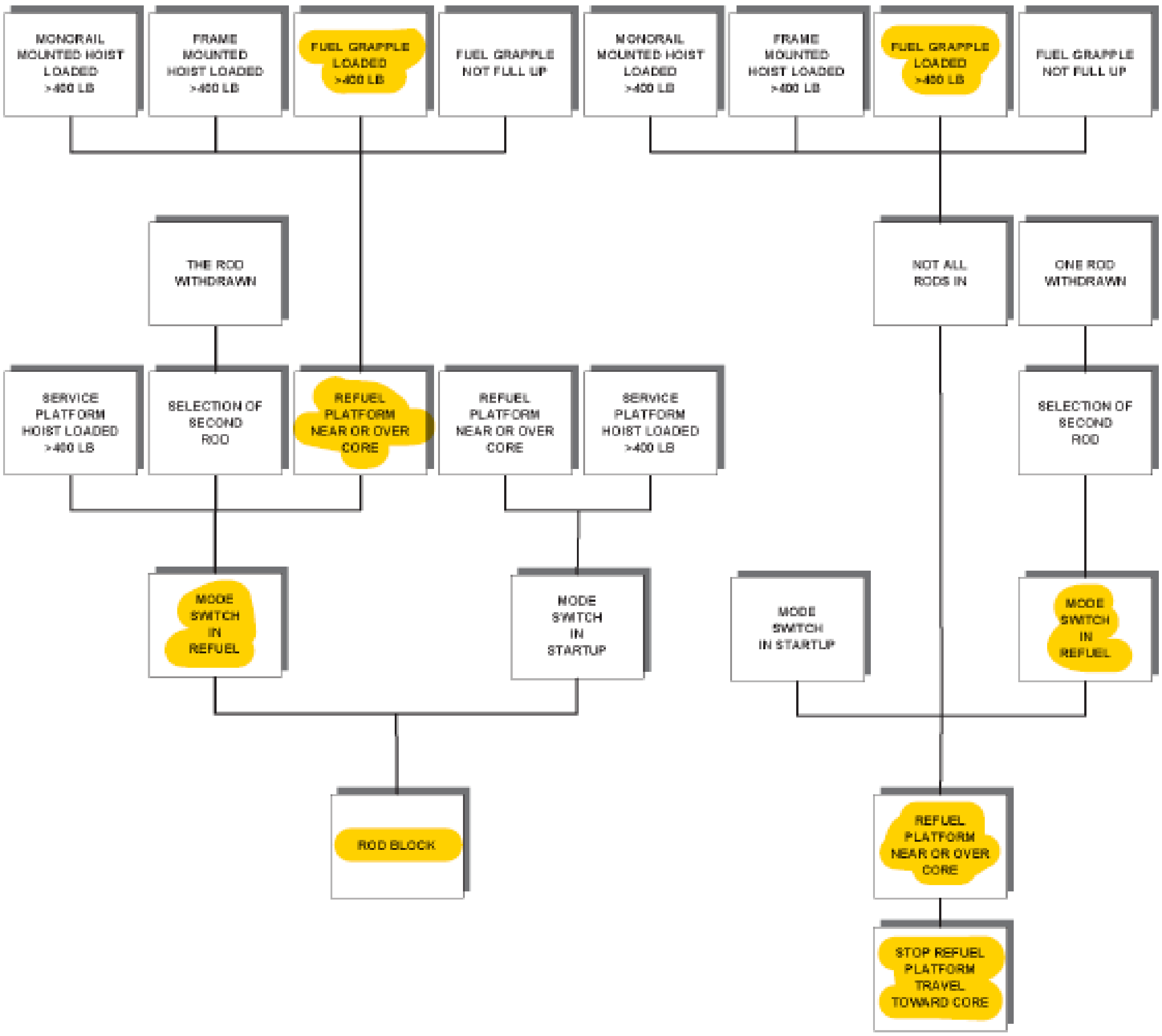
2. Platform near or over the core with the Mode Switch in other than REFUEL.

3. One rod withdrawn and when withdrawn rod is initially deselected with the Mode Switch in REFUEL. (As long as the rod that is withdrawn is never deselected bridge travel may continue and not be blocked by this interlock.)

BFN Unit 0	Refueling Operations (In-Vessel Operations)	0-GOI-100-3A Rev. 0086 Page 211 of 218

Attachment 30
(Page 2 of 2)

Rod, Bridge, and Hoist Blocks, Block Diagram



Examination Outline Cross-reference:	Level	RO	SRO
263000 (SF6 DC) DC Electrical Distribution	Tier #	2	------
K4.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following:	K/A #	263000K4.01	
• Manual/ automatic transfers of control: Plant-Specific	Importance Rating	3.1	------

Proposed Question: **# 49**

In accordance with 0-OI-57D, DC Electrical System, which **ONE** of the following completes the statements below concerning the 2B 250V Battery Charger?

The 2B 250V Battery Charger ___**(1)**___ be aligned to supply Battery Board 5.

There is a/an ___**(2)**___ interlock that prevents the 2B 250V Battery Charger from supplying more than one Battery Board at a time.

A. (1) can
(2) electrical

B. (1) can
(2) mechanical

C. (1) can NOT
(2) electrical

D. (1) can NOT
(2) mechanical

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). The second part is incorrect but plausible in that there are many breaker interlocks that are electrical, for example both normal and alternate feeder breakers to a 250V Reactor MOV Board are electrically interlocked to prevent simultaneous closure of DC sources.

B **CORRECT:** (*See attached*) In accordance with 0-OI-57D, DC Electrical System, the 2B 250V Battery Charger can be aligned to Battery Board 5. For second part, 0-OI-57D states that it is required to position the mechanical interlock to align the 2B Battery Charger output to Battery Board 4, 5, or 6.

C INCORRECT: The first part is incorrect but plausible in that 7 different 250V battery chargers exist in the DC System. While 6 of the 7 different 250V battery chargers can NOT be aligned to supply all the respective Battery Boards, 2B 250V Battery Charger (known as the Spare 250V Battery Charger) can be aligned to service Battery Boards 1, 2, 3, 4, 5, 6. The second part is incorrect but plausible (*See B*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is incorrect but plausible (*See A*).

RO Level Justification: Tests the candidate's knowledge of the DC Electrical System design features and/or interlocks as it relates to manual transfers of control. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): 0-OI-57D, Rev. 175 (Attach if not previously provided)

OPL171.037, Rev. 14

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.037 Obj. 5_ (As available)

Question Source:

Bank #	
Modified Bank #	
New	X

(Note changes or attach parent)

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 X

55.43

Comments:

Excerpts from 0-OI-57D:

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0175 Page 217 of 336

8.1 Placing the 250V BATTERY CHARGER 2B, 0-CHGA-248-0002B in Service to Battery Board 1(2,3,4,5,6) (continued)

[3] **ENSURE** the appropriate emergency tie to DC board is on:

- 250V BATTERY CHARGER 2B TIE TO DC BD 1, 0-BKR-280-0001/609 on Battery Board 1.
- 250V BATTERY CHARGER 2B EMER TIE TO DC BD 2, 0-BKR-280-0002/607 on Battery Board 2.
- EMERGENCY SUPPLY FROM 250V DC BATTERY CHARGER 2B, 0-BKR-280-0003/609 on Battery Board 3.
- 250V BAT CHGR 2B EMER TIE TO DC BD 4, 0-BKR-280-0004/202 on Battery Board 4.
- BAT BD NO 5 BKR 202 OUTPUT TRANSFER, 0-BKR-280-0005/202 on Battery Board 5.
- BAT BD NO 6 BKR 202 OUTPUT TRANSFER, 0-BKR-280-0006/202 on Battery Board 6.

[4] **IF** Battery Board 4, 5 or 6 is to be connected to 250V BATTERY CHARGER 2B, 0-CHGA-248-0002B, **THEN** (Otherwise N/A)

PERFORM the following in Battery Board Room 4:

[4.1] **ALIGN** mechanical interlock, BATTERY CHARGER OUTPUT TRANSFER SWITCH 2BA, 0-XSW-248-0002BA to the appropriate disconnect switch.

[4.2] **PLACE** the appropriate disconnect switch to ON:

- ALT FDR BAT BD 5 BKR 202
- ALT FDR BAT BD 4 BKR 202
- ALT FDR BAT BD 6 BKR 202

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0175 Page 218 of 336

8.1 Placing the 250V BATTERY CHARGER 2B, 0-CHGA-248-0002B in Service to Battery Board 1(2,3,4,5,6) (continued)

NOTE

It is required to position the mechanical interlock to align Battery Charger 2B Output Transfer Switch to 0-XSW-248-0002B CONNECT TO 0-XSW-248-0002BA, for Battery Board 4 or 5 or 6.

CAUTION

The transfer switch mechanical interlock on 250V Battery Charger 2B output Transfer Switch prevents supplying more than one of the four possible Battery Boards with 250V Battery Charger 2B.

[5] **ALIGN** Mechanical Interlock BATTERY CHARGER 2B OUTPUT TRANSFER SWITCH 2B, 0-XSW-248-0002B, to the appropriate disconnect switch, **THEN**

PLACE the appropriate disconnect switch to ON:

- Battery Board 1: TO BATTERY BD 1 BKR 609
- Battery Board 2: TO BATTERY BD 2 BKR 607
- Battery Board 3: TO BATTERY BD 3 BKR 609
- Battery Board 4: CONNECT TO 0-XSW-0002BA. (N/A if Battery Board 4 is not being connected)
- Battery Board 5: CONNECT TO 0-XSW-0002BA. (N/A if Battery Board 5 is not being connected)
- Battery Board 6: CONNECT TO 0-XSW-0002BA. (N/A if Battery Board 6 is not being connected)

[6] **PERFORM** the following at 250V BATTERY CHARGER 2B, 0-CHGA-248-0002B:

- **ENSURE** the NORMAL SUPPLY INPUT FROM 480V SD BD 2B/6D, 0-BKR-248-0002BA is OFF.
- **ENSURE** the ALTERNATE SUPPLY INPUT FROM 480V CMN BD 1/3A, 0-BKR-248-0002BB is OFF.
- **ENSURE** POWER ON, 0-HS-248-0002B is OFF.

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0175 Page 241 of 336

8.9 Transfer of Power Supplies to 250V Reactor MOV Boards

8.9.1 Transfer of Power Supplies to 250V Reactor MOV Boards from Normal Supply to Alternate Supply

[1] **REVIEW** all Precautions and Limitations in Section 3.0.

NOTES

1) Tripping of the normal or alternate feeder breakers to 250V Reactor MOV Boards on overcurrent results in lockout of both breakers.

2) Both normal and alternate feeder breakers to a 250V Reactor MOV Board are electrically interlocked to prevent simultaneous closure (paralleling) of DC sources.

3) The normal and alternate feeder breakers are located on the 250V Reactor MOV Board which they supply.

4) Trip Test push-buttons are used only for testing racked out normal and alternate feeder breakers.

5) Transfer requires two operators due to the distance between the normal and alternate feeder breakers.

6) Prior to transferring any 250VDC RMOV Board to the alternate supply, Precaution and Limitation 3.0A must be complied with.

7) Transfer of 250V RMOV BD 3A will cause annunciation of the following alarms:
- 3-XA-55-3C, Window 1, RCIC RELAY LOGIC POWER FAILURE
- 3-XA-55-3C, Window 32, ADS BLOWNDOWN POWER FAILURE
- 3-XA-55-3E, Window 23, 480V RX MOV BD D BACKUP SW IN EMER POSN
- 3-XA-55-3F, Window 28, HPCI GLAND SEAL CONDENSER HOTWELL LEVEL LOW
- 3-XA-55-5B, Window 34, PNL 9-47 FUSE FAILURE
- 3-XA-55-4B, Window 22, 4160V RPT BD 3-II CONTROL ABNORMAL

CAUTIONS

1) When any unit 250VDC RMOV Board A or B is transferred to alternate power supply, it is possible that a transfer of EHC from Reactor Pressure Control to Header Pressure Control to occur due to a loss of power to 2 of the 4 reactor press instruments. [PER 109297]

2) Failure to unload and shutdown the affected downstream ECCS ATU Inverter may cause the plant to enter unplanned LCO's. (0-OI-57C)

Excerpt from OPL171.037 Lesson Plan:

250V Battery Charger	Normal Source	Alternate Source	
1	480V SD Bd 1A Comp 6D	480V Common Bd 1 Comp 3A	Obj 5 OF-2 0-45E710-1 0-45E710-7
2A	480V SD Bd 2A Comp 6D	480V Common Bd 1 Comp 3A	
2B	480V SD Bd 2B Comp 6D	480V Common Bd 1 Comp 3A	
3	480V SD Bd 3A Comp 6D	480V Common Bd 1 Comp 3A	
4	480V SD Bd 3B Comp 6D	480V Common Bd 1 Comp 3A	
5	480V Com Bd 1 Comp 5C	(no alternate)	
6	480V Com Bd 3 Comp 3D	(no alternate)	

2B spare charger DC output can be directed to any of four feeders. Three DC outputs can be connected to battery board 1, 2, or 3. The fourth output is connected to another output transfer switch (located in battery board room 4) which charges batteries 4, 5, or 6 plant batteries. A mechanical interlock permits closing only one output feeder at a time. (A slide bar is utilized in battery board room 2 and a Kirk key interlock is used in battery board room 4) — 0-OI-57D

D. Distribution System — Obj 5, OF-2, 3-45E779-5, 51, 0-45E704

1. The 250V Unit and Plant/Station DC subsystems distribute power via the breakers on their Battery Boards. These Boards have been identified as one of the most significant systems to mitigate the events of significance in the Browns Ferry Probabilistic Risk Assessment (PRA).

2. Manual alignment of DC supplies from battery boards 1, 2 and 3 are considered to be important operator actions required to mitigate the events of significance in the Browns Ferry PRA

3. The Unit subsystems provide power for DC motor-operated valves, DC pump motors, and ECCS control and logic circuits. They supply Main Control Room 9-9 Panels, and provide control power for all 480V Shutdown Boards and Cooling Tower switch gear. Caution should be exercised when transferring or opening loads from the battery boards due to effects on other units. — Obj 1D

45E712 Series

Examination Outline Cross-reference:

241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating

A1.23 (10CFR 55.41.5)

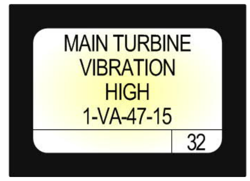
Ability to predict and/or monitor changes in parameters associated with operating the REACTOR/TURBINE PRESSURE REGULATING SYSTEM controls including:

- Main turbine vibration

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	241000A1.23	
Importance Rating	2.8	------

Proposed Question: **# 50**

Unit 1 is operating at 100% RTP with the following conditions:



- MAIN TURBINE VIBRATION HIGH (1-9-7B, Window 32) alarms
- The Balance of Plant Operator (BOP) observes vibration on Journal Bearing # 4 reading **8** mils and Journal Bearing # 5 at **7.5** mils
- **BOTH** Journal Bearing vibration levels are rising at a rate of **2** mils per minute

Which **ONE** of the following completes the statement below?

Given the conditions above, the **EARLIEST** time the Main Turbine **MUST** be tripped is __________ in accordance with the associated ARP.

A. immediately

B. in 2 minutes

C. in 4 minutes

D. in 15 minutes

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that IMMEDIATELY may be selected if the candidate takes the given vibration of 8 mils, which activates the given alarm, as the required action in accordance with the given 1-9-7B, Window 32.

B **CORRECT**: (*See attached*) The calculation for 2 minutes with the given vibration rate of change yields 12 mils. In accordance with the given ARP, Table 1, the Main Turbine Trip is required IMMEDIATELY after any Journal Bearing vibration exceeds 12 mils. NOTE: the second journal bearing vibe of 7.5 mils is needed to confirm that the vibrations are real. The automatic turbine trip on vibration (if not bypassed) looks at adjacent bearings to detect the proper combination to trip the turbine. This trip is normally bypassed so the Operator is taking the responsibility for the automatic function. See Attachment 1 (page 44) of ARP 9-7B.

C INCORRECT: Incorrect but plausible in that the calculation for 4 minutes with the given vibration rate of change yields 16 mils. In accordance with the given ARP, Table 1, the Main Turbine Trip is immediately required between 800 and 1400 RPM with 14 mils.

D INCORRECT: Incorrect but plausible in that the candidate can easily confuse the given ARP, Table 1, Main Turbine Trip time required after any Journal Bearing vibration exceeds 10 mils for 15 minutes with the given vibration levels and rate.

RO Level Justification: Tests the candidate's ability to predict the impacts of Main Turbine Journal Bearing vibration and use procedures to mitigate the consequences. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome. To successfully answer, the candidate must understand the mode of operation of the plant relative to the vibration chart to make two different determinations.

Technical Reference(s): 1-OI-47 Rev. 60 (Attach if not previously provided)
1-ARP-9-7B, Rev. 25

Proposed references to be provided to applicants during examination: **MAIN TURBINE VIBRATION HIGH (1-9-7B, Window 32)**

Learning Objective: OPL171.147 Obj. 2 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1804 NRC #62	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

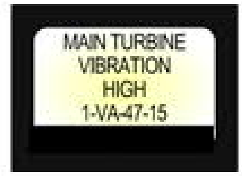
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

62. Unit 1 is operating at 100% RTP when the following occurs:



- MAIN TURBINE VIBRATION HIGH alarms (1-9-7B, Window 32)
- The BOP observes vibration on Journal Bearing # 4 reading **8** mils and Journal Bearing # 5 at **7.5** mils
- **BOTH** Journal Bearing vibration levels are rising at a rate of **1** mil per minute

Which ONE of the following completes the statement below in accordance with 1-ARP-9-7B, Window 32?

Given these plant conditions, the **EARLIEST** time the Main Turbine **MUST** be tripped is ________.

A. immediately

B. in 2 minutes

C. in 4 minutes

D. in 15 minutes

Correct Answer: C

Pedigree Information:

Excerpt from 1-OI-47:

BFN Unit 1	Turbine-Generator System	1-OI-47 Rev. 0060 Page 69 of 281

5.4 Turbine Roll (continued)

NOTES

1) Main Turbine vibration readings can be monitored on TURBINE-GENERATOR VIBRATION recorder (1-XR-47-15), ICS, EHC WORKSTATIONS, or locally using hand held vibration monitoring devices deemed appropriate by Components Engineering. During Turbine startup, it may be preferred to use the hand held vibration monitoring devices for reasons specified in Precaution 3.7B.2. If it is desired to use local vibration detectors, field personnel shall be in direct communications with the Control Room.

2) Nuclear Mutual Limited Company is notified when vibration levels exceed 6 mils. The next action level in the procedure is 10 mils and 12 mils. Actions between 6 mils and the next action level (10 or 12 mils) is a plant management decision made on fundamental characteristics of the rise in vibration levels and economic practicality.

CAUTION

BWROG Scram Frequency Reduction recommends to mitigate the potential of a turbine rotor rub event and the consequential highvibration when operating < 30% Reactor Thermal Power (RTP), the following parameters should be maintained in these bands:

1) Main Condenser vacuum <28"Hg.
2) Steam Packing Exhauster vacuum 10"-12" H2O.
3) Steam Seal header pressure 2-3 psig.

TABLE 1: NORMAL VIBRATION LIMITS

SPEED (RPM)	TRIP AFTER ANY JOURNAL VIBRATION EXCEEDS ____MILS FOR	____MINUTES	TRIP IMMEDIATELY IF JOURNAL BEARING VIBRATION EXCEEDS	NORMAL VIBRATION LEVEL FOR CONTINUED OPERATION
LESS THAN 800			8 MILS	
800 - 1400	10	2	14 MILS	7 MILS
1400 - RUNNING SPEED	10	15	12 MILS	≤ 5 MILS

[47] **IF** any of the vibration limits requiring a trip are met in Table 1, **THEN**:

DEPRESS Turbine TRIP pushbutton, 1-HS-47-67D. (Otherwise N/A)

Excerpts from 1-ARP-9-7B:

BFN Unit 1	Panel 9-7 1-XA-55-7B	1-ARP-9-7B Rev. 0025 Page 42 of 47

MAIN TURBINE VIBRATION HIGH 1-VA-47-15 32	Sensor/Trip Point: 1-XM-47-87E/7 CH DO/22 Inputs from 1-VE-47-15 A through M — 8 mils

(Page 1 of 3)

Sensor Location: Main Turbine
El 617'
Turbine Bldg

Probable Cause:

A. Startup/rolling up to speed.
   1. Passing through critical speeds.
   2. Improper rolling rate for metal temperature.
   3. Bearing oil temperature abnormal.
   4. Improper steam seal header pressure.
   5. ImproperTurbine startup drain valve alignment.
   6. Turbine imbalance.
   7. Bearing failure.
   8. Excessive turbine operation at FSNL (full speed no load)

B. Operating at speed.
   1. Improper load or unload rate.
   2. Bearing oil temperature abnormal.
   3. Bearing failure.
   4. Turbine imbalance.
   5. Generator MVAR loading.

C. Sensor malfunction.

D. High Generator Hydrogen Cooler outlet differential temperature (greater than or equal to 10 deg C)

E. High Exciter Cooler outlet differential temperature (greater than or equal to 10 deg F)

Automatic Action: A turbine trip can occur when high vibration is sensed by combinations of bearings (assuming the trip is not bypassed). Reference Attachment 1 to this alarm response for these combinations.

Continued on Next Page

BFN Unit 1	Panel 9-7 1-XA-55-7B	1-ARP-9-7B Rev. 0025 Page 43 of 47

MAIN TURBINE VIBRATION HIGH 1-VA-47-15, Window 32
(Page 2 of 3)

Operator Action:

A. CHECK the following:
- On EHC WORKSTATION, Turbine Vibration screen.
- On ICS, MAIN TURBINE BEARINGS (TURBBRG) screen.
- TURBINE GENERATOR VIBRATION recorder, 1-XR-47-15 (Panel 1-9-7).
- Computer points 47-15A thru 47-15M.

CAUTION
If Main Turbine trips Unit SRO will decide if Condenser Vacuum should be broken to lower Main Turbine speed at a higher rate.

B. **IF** alarm is valid, **THEN** **PERFORM** the following:
1. **DETERMINE** cause by checking PROBABLE CAUSE section above.
2. **REDUCE** load and **OBSERVE** vibration.
3. **IF** any of the vibration limits requiring a trip are met in Table 1, **THEN** **DEPRESS** Turbine TRIP pushbutton, 1-HS-47-67D:

TABLE 1: NORMAL VIBRATION LIMITS				
SPEED (RPM)	**TRIP AFTER** ANY JOURNAL VIBRATION EXCEEDS		**TRIP IMMEDIATELY IF** JOURNAL BEARING VIBRATION EXCEEDS	NORMAL VIBRATION LEVEL FOR CONTINUED OPERATION
	___MILS FOR	___MINUTES		
LESS THAN 800			8 MILS	
800 - 1400	10	2	14 MILS	7 MILS
1400 - RUNNING SPEED	10	15	12 MILS	≤ 5 MILS

Continued on Next Page

BFN Unit 1	Panel 9-7 1-XA-55-7B	1-ARP-9-7B Rev. 0025 Page 44 of 47

MAIN TURBINE VIBRATION HIGH 1-VA-47-15, Window 32
(Page 3 of 3)

Operator
Action: (Continued)

4. IF Main Turbine was tripped due to high vibration AND a significant loss of Hydrogen pressure is observed, THEN DUMP generator hydrogen with Unit SRO concurrence. REFER TO Dumping Hydrogen section of 1-OI-35

References: 1-45E620-10-2 1-47E610-47-3 and 4 1-45E602-28,44

Attachment 1

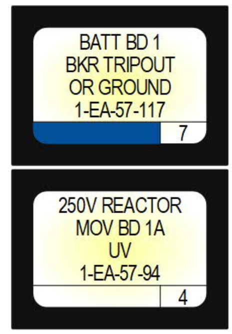
Turbine high vibration trip logic has been modified such that trips will be generated only when certain combinations of bearings experience high and high high vibration. The confirmatory high setpoint is 8 mils for number 1 through 12 bearings. The high high setpoint for all bearings is 10 mils. Any one of the following combinations will cause a turbine trip (assuming the trip is NOT bypassed):

- #1 or #3 Conf high and #2 high high
- #3 or #5 Conf high and #4 high high
- #5 or #7 Conf high and #6 high high
- #7 or #9 Conf high and #8 high high
- #9 or #11 Conf high and #10 high high
- #11 Conf high and #12 high high
- #2 Conf high and #1 high high
- #2 or #4 Conf high and #3 high high
- #4 or #6 Conf high and #5 high high
- #6 or #8 Conf high and #7 high high
- #8 or #10 Conf high and #9 high high
- #10 or #12 Conf high and #11 high high

Examination Outline Cross-reference:	Level	RO	SRO
295004 (APE 4) Partial or Complete Loss of D.C. Power / 6	Tier #	1	------
AK3.02 (10CFR 55.41.5)	Group #	1	------
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: • Ground isolation/fault determination	K/A #	295004AK3.02	
	Importance Rating	2.9	------

Proposed Question: **# 51**

Unit 1 is operating at 100% RTP with the following conditions:



- BATTERY BOARD 1 BREAKER TRIPOUT OR GROUND (1-9-8C, Window 7) alarms
- 250V REACTOR MOV BOARD 1A UNDERVOLTAGE (1-9-8C, Window 4) alarms
- Battery Board 1, 250V RMOV Board 1A Normal Feeder Breaker tripped
- **NO** Operator actions have been taken

Which **ONE** of the following completes the statement below in accordance with 1-AOI-57-11, Loss of Power to an ECCS ATU Panel / ECCS Inverter?

Given the conditions above, 250V DC RMOV Board 1A provides ___**(1)**___ to the respective ECCS Analog Trip Unit (ATU) **AND** ___**(2)**___ ECCS ATU is de-energized.

A. (1) **ONLY** one power source
(2) Division I, Panel 1-9-81

B. (1) **ONLY** one power source
(2) Division II, Panel 1-9-82

C. (1) **BOTH** the normal **AND** redundant power sources
(2) Division I, Panel 1-9-81

D. (1) **BOTH** the normal **AND** redundant power sources
(2) Division II, Panel 1-9-82

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the BFN DC Systems consist of 250V DC, 125 V DC, 48 V DC, +/- 24V DC, countless batteries/chargers/panels and Unit differences which could easily confuse candidates on power supply arrangements. Second part is incorrect but plausible in that the ECCS ATU Division power supplies/division are often confused given opposite board convention.

B INCORRECT: First part is incorrect but plausible *(See A)*. Second part is correct *(See D)*.

C INCORRECT: First part is correct *(See D)*. Second part is incorrect but plausible *(See A).*

D CORRECT: (*See attached*) In accordance with the given 1-ARP-9-8C, Window 4 and Window 7, a ground fault has occurred on 250V DC Battery Board 1 resulting in a loss of 250V DC RMOV Board 1A. This in turn results in the loss of both the normal and alternate power supply to specifics ECCS ATUs. For second part, given the loss of 250V DC RMOV Board 1A, both the normal and alternate power supply to ECCS ATU Division II, Panel 1-9-82 is lost.

RO Level Justification: Tests the candidate's knowledge of a Loss of DC Power as it relates to an electrical ground and/or fault determination. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (2) Diagnosis that leads to selection of procedures that should be used to respond to the evolution.

Technical Reference(s): 1-AOI-57-11, Rev. 6 (Attach if not previously provided)
1-ARP-9-8C, Rev. 17
OPL171.037, Rev. 16

Proposed references to be provided to applicants during examination: **BATTERY BOARD 1 BREAKER TRIPOUT OR GROUND (1-9-8C, Window 7), 250V REACTOR MOV BOARD 1A UNDERVOLTAGE (1-9-8C, Window 4)**

Learning Objective: OPL171.037 Obj. 5a (As available)

Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	**X**
Question History:	Last NRC Exam	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Excerpts from 1-ARP-9-8C:

BFN Unit 1	Panel 1-9-8 1-XA-55-8C	1-ARP-9-8C Rev. 0017 Page 7 of 48

250V REACTOR MOV BD 1A UV 1-EA-57-94 (4)	Sensor/Trip Point:	
	Relay 72N-BA	Normal supply overcurrent.
	Relay 72E-BA	Alternate supply overcurrent.
	Relay 27EX	Normal supply undervoltage.
	Relay 27B	MOV board undervoltage (7 sec TDDO)

(Page 1 of 2)

Sensor Location: 250V RX MOV Board 1A, EL 621', R-1 Q-Line, A Shutdown Board Room

Probable Cause:

A. Loss of normal supply (250V Battery Board 1, Pnl 2, Breaker 202).
B. Overcurrent on normal or alternate supply to the board.
C. Fuse failure.
D. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** alarm by:
- Loss of HPCI indicating lights on Panel 1-9-3
- Loss of backup scram valve lights on Panel 1-9-5

B. **DISPATCH** Personnel to 250V MOV Beard to check for abnormal conditions: undervoltage, breaker tripped, etc.

NOTE

[II/C] If the mechanism resets (hear click and feel resistance when pushing in), this indicates that an overcurrent condition tripped the breaker.

C. **IF** Normal or Alternate feeder breaker tripped, **THEN**

MANUALLY **DEPRESS** mechanical trip/reset mechanism on the breaker face to reset Bell Alarm lockout device. [NER/C II-B-92-069]

D. **ENSURE** Breaker 202 closed at Battery Board Room 1, Panel 2, EL 593'.

Continued on Next Page

BFN Unit 1	Panel 1-9-8 1-XA-55-8C	1-ARP-9-8C Rev. 0017 Page 12 of 48

BATT BD 1 BKR TRIPOUT OR GROUND 1-EA-57-117 (7)	Sensor/Trip Point: Breaker tripout. Ground on Battery Board 1

(Page 1 of 2)

Sensor Location: Battery Board Room 1, El 593'

Probable Cause:

A. Breaker overload or fault (thermal and magnetic trip) on Panels 1-14
B. A ground fault exists on 250V DC Battery Board 1.
C. Fuse failure to instrumentation circuit, Panel 1.
D. Blown light bulb in ground detector indicator photo cell.
E. A ground fault exists on 48V DC,(Battery Board 1 PNL 10R)

Automatic Action: Potential loss of any of the following equipment fed from Battery Board 1.

- 250V DC RMOV Board 1A, 2C and 3B
- Feedwater inverter
- 250V DC Cabinet 1, Panel 1-9-9

Operator Action:

A. **CHECK** Battery Board 1 volts (EI-57-37) and amps (EI-57-38) on Panel 1-9-8 to determine if load is on the battery or the charger.

B. **CHECK** Battery Board 1 for abnormal conditions:
- Loss of volt and amp indication.
- Breaker position.
- Ground indication, 250V DC and 48V DC.
- Fuse failure.
- Check ground detector photo cell light bulb (requires electrical maintenance).

C. **CHECK** condition of equipment fed from the tripped breaker and attempt reclosure as directed by Unit SRO.

Continued on Next Page

Excerpts from 1-AOI-57-11:

BFN Unit 1	Loss of Power to An ECCS ATU Panel / ECCS Inverter	1-AOI-57-11 Rev. 0006 Page 3 of 33

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions, operator actions, technical specification requirements, and reportability requirements resulting from a loss of power to ECCS ATU Panel 1-9-81 or 1-9-82 or loss of an ECCS inverter.

NOTES

1) Each Inverter provides 120V AC power to selected divisional instruments, controls and logic and one of two redundant power supplies to its divisional ATU cabinet. The power supplies to ECCS ATU Panel 1-9-81 (Div. I), ECCS ATU Panel 1-9-82 (Div. II) and the ECCS inverters are as follows:

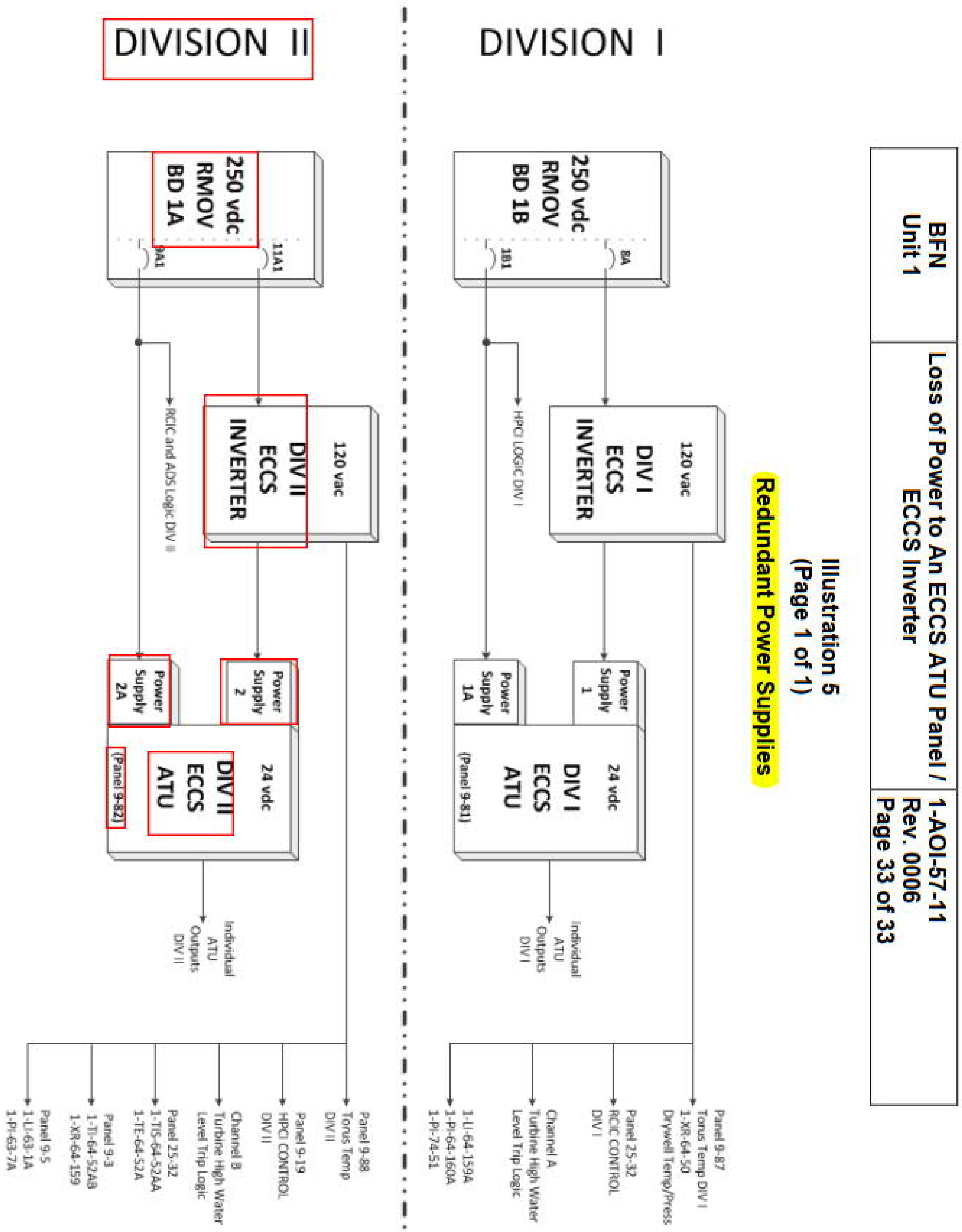
Panel 1-9-81

• Division I ECCS inverter Refined Signal	250V RMOV Board 1B, compartment 8A
• Division I 250/24vdc Black Box Conversion	250V RMOV Board 1B, compartment 1B1, in case inverter fails

Panel 1-9-82

• Division II ECCS inverter Refined signal	250V RMOV Board 1A, comp 11A1
• Division II 250/24vdc Black Box Conversion	250V RMOV Board 1A, comp 9A1

- Power will be lost to an ECCS ATU Panel due to the loss of the respective 250V RMOV board listed above
- Opening/loss of both of the breakers listed above
- Loss of ECCS ATU Panel internal fuses
- Simultaneous loss of both redundant 24vdc power supplies in each ECCS ATU panel.


BFN
Unit 1
Loss of Power to An ECCS ATU Panel / ECCS Inverter
1-AOI-57-11
Rev. 0006
Page 33 of 33
Illustration 5
(Page 1 of 1)
Redundant Power Supplies
DIVISION I
250 vdc RMOV BD 1B
8A
1B1
120 vac DIV I ECCS INVERTER
HPCI LOGIC DIV I
Power Supply 1
Power Supply 1A
24 vdc DIV I ECCS ATU
(Panel 9-81)
Individual ATU Outputs DIV I
Panel 9-87 Torus Temp DIV I 1-XR-64-50 Drywell Temp/Press
Panel 25-32 RCIC CONTROL DIV I
Channel A Turbine High Water Level Trip Logic
1-LI-64-159A 1-PI-64-160A 1-PI-74-51
DIVISION II
250 vdc RMOV BD 1A
11A1
9A1
120 vac DIV II ECCS INVERTER
RCIC and ADS Logic DIV II
Power Supply 2
Power Supply 2A
24 vdc DIV II ECCS ATU
(Panel 9-82)
Individual ATU Outputs DIV II
Panel 9-88 Torus Temp DIV II
Panel 9-19 HPCI CONTROL DIV II
Channel B Turbine High Water Level Trip Logic
Panel 25-32 1-TIS-64-52AA 1-TE-64-52A
Panel 9-3 1-TI-64-52AB 1-XR-64-159
Panel 9-5 1-LI-63-1A 1-PI-63-7A

Excerpt from OPL171.037 Lesson Plan:

OPL171.037, DC Systems, Rev: 16

D. Distribution System

1. The 250V Unit and Station DC subsystems distribute power via the breakers on associated Battery Boards. — Fig 2,3,4

2. These Boards are identified among the most significant systems to mitigate the events of significance in the Browns Ferry Probabilistic Risk Assessment (PRA). — 3-45E779-5, 51 0-45E704
Manual alignment of DC supplies from battery boards 1, 2 and 3 are considered to be important operator actions required to mitigate the events of significance in the Browns Ferry PRA

3. The Unit subsystems provide power ECCS control and logic circuits, DC motor-operated valves, and DC pump motors. They supply Main Control Room 9-9 cabinet 1, and provide control power for all 480V Shutdown Boards and Cooling Tower switch gear. — Obj 1d

4. Unit batteries supply alternate control power for A-D and 3EB 4kV Shutdown Boards. 3EA, 3EC, and 3ED 4kV Shutdown Boards receive both normal and alternate control power from the 250V Unit DC Systems. — Obj 1d, Obj 5b, 45E712 Series

5. Exercise caution when transferring or opening loads from the battery boards due to potential effect on other units.

6. The 250V RMOV Boards are supplied from Unit Battery Boards as follows: — Obj 1d, 0-OI-57d P&L

250 RMOV BD	NORMAL SOURCE	ALTERNATE SOURCE
1A	BB-1	BB-2
1B	BB-3	BB-1
1C	BB-2	BB-1
2A	BB-2	BB-3
2B	BB-3	BB-1
2C	BB-1	BB-2
3A	BB-3	BB-2
3B	BB-1	BB-3
3C	BB-2	BB-3

*All transfers for these boards are manual only.

7. Loss of 250VDC from each unit's 'A' or 'B' 250V RMOV Board results in loss of that divisions ECCS ATU Inverter and the redundant 24VDC converter power supply. This results in a complete loss of voltage to the divisions ECCS ATU logic panels (panel 9-81 or 9-82) and loss of voltage to the other components/systems served by the inverter. — Obj 8, AOI-57-11

Examination Outline Cross-reference:	Level	RO	SRO
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8	Tier #	1	------
AA1.02 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:	K/A #	295019AA1.02	
• Instrument air system valves: Plant-Specific	Importance Rating	3.3	------

Proposed Question: **# 52**

Unit 2 is operating at 100% RTP when a Control Air leak develops, resulting in header pressure lowering to 25 psig.

Given the condition above, which **ONE** of the following completes the statements below?

The ___**(1)**___ Main Steam Isolation Valves (MSIVs) are CLOSED.

In accordance with 2-AOI-32-2, Loss of Control Air, ___**(2)**___ is aligned to re-open the MSIVs.

A. (1) outboard
(2) Drywell Control Air

B. (1) outboard
(2) Containment Atmosphere Dilution (CAD)

C. (1) inboard
(2) Drywell Control Air

D. (1) inboard
(2) Containment Atmosphere Dilution (CAD)

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) The first part is correct in that in accordance with 2-AOI-32-2, Loss of Control Air, the outboard MSIVs will close if Control Air Pressure is less than 45 psig. The second part is correct in that in accordance with 2-AOI-32-2, Loss of Control Air, Drywell Control Air is aligned to supply the outboard MSIVs using Attachment 2.

B INCORRECT: The first part is correct (*See A*). The second part is incorrect but plausible in that in accordance with 2-EOI-1, RPV Control, Operators are directed to cross-tie CAD to Drywell Control Air to the MSIVs in accordance with 2-AOI-Appendix-8G, Crosstie CAD to Drywell Control Air if necessary. 2-AOI-32-2 does not direct cross-tieing CAD to re-open MSIVs.

C INCORRECT: The first part is incorrect but plausible in that the inboard MSIVs are supplied air from Drywell Control Air, and the outboard MSIVs are supplied air from Control Air. Because there are two different air sources for the MSIVs, it is plausible that Drywell Control Air supplies the outboard MSIVs. The second part is correct (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability to diagnose the effect of a Loss of Control Air on the Inboard and Outboard MSIVs, and the system that is used in acordance with the AOIs to provide the necessary air for operation given a loss of Control Air. This question is rated as C/A due to the requirement to assemble, sort, and integrate two distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 2-AOI-32-2, Rev.37 (Attach if not previously provided)
2-EOI-1, Rev.18
OPL171.054, Rev. 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.54, Obj. 6, 8 (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.054-08 008 #1721	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1721. OPL171.054-08 008

Given the following conditions for Unit 2:

- Reactor has scrammed due to a loss of control air header pressure.
- Suppression pool temperature is 115° F
- Control Air header pressure is 25 psig
- RPV water level was deliberately lowered to (-) 55 inches

Which ONE of the following completes the statements below?

MSIV status is __(1)__.

__(2)__ contains the steps to re-align Drywell Control Air to the MSIVs.

NOTE: 2-AOI-32-2, Loss of Control Air
2-EOI Appendix-8B, Reopening MSIVs/Bypass Valve Operation

A. (1) all MSIVs CLOSED

(2) 2-AOI-32-2

B. (1) all MSIVs CLOSED

(2) 2-EOI Appendix-8B

C✓ (1) inboard MSIVs OPEN, outboard MSIVs CLOSED

(2) 2-AOI-32-2

D. (1) inboard MSIVs OPEN, outboard MSIVs CLOSED

(2) 2-EOI Appendix-8B

Excerpts from 2-AOI-32-2:

BFN Unit 2	Loss of Control Air	2-AOI-32-2 Rev. 0037 Page 8 of 24

4.2 **Subsequent Actions (continued)**

[7.2.1] **ESTABLISH** lube oil temperature between 80°F and 100°F using the following TCV BYPASS VALVE(s):

- For A RFP use 2-24-626A or 3-24-627A
- For B RFP use 2-24-626B or 3-24-627B
- For C RFP use 2-24-626C or 3-24-627C

[7.3] **CLOSE** EHC fluid cooler TCV isolation valves 2-24-592 or 2-24-593, **THEN**

ESTABLISH fluid temperature between 85°F and 125°F on TI-47-59 using TCV BYPASS VALVE 2-24-590 or 2-24-591.

[8] **ENSURE** drywell control air system is being supplied by either the Nitrogen System or CAD system.

NOTES

1) DRYWELL control air can be valved into control air lines for outboard MSIV's.

2) Control air must be restored in order to close main steam line drain valves 2-FCV-1-58 and 2-FCV-1-185. These valves must be closed in order to develop pressure downstream of outboard MSIVs

[9] **IF** Unit SRO determines Outboard MSIV's need to be opened in order to establish the main condenser as a heat sink,
AND
Control Air is available

THEN

PERFORM Attachment 2. (Otherwise N/A)

BFN Unit 2	Loss of Control Air	2-AOI-32-2 Rev. 0037 Page 18 of 24

Attachment 1
(Page 1 of 6)

Expected System Responses

1.0 MAIN STEAM

A. If the loss of control air is instantaneous, when control air pressure drops to < 45 psig, the MSIV accumulator air will be routed to close the OutBd MSIVs.

BFN Unit 2	Loss of Control Air	2-AOI-32-2 Rev. 0037 Page 24 of 24

Attachment 2
(Page 1 of 1)

Align Drywell Control Air to Open Outboard MSIV's

The following steps will align the Drywell Control Air System to Outboard MSIVS:

- **CLOSE** 2-32-1878 located at Elev 565 Rx Bldg outside steam vault (this will prevent supplying Drywell Control Air to components outside the Drywell other than the outboard MSIV's).
- **OPEN** 2-32-334 located above the TIP room (aligns Drywell Control Air to the outboard MSIV's).

REOPEN MSIV's.

Excerpt from 2-EOI-1:

STABILIZE RPV press below 1073 psig using the main turbine bypass vlvs (APPX 8B)

- OK to use ANY Alternate RPV Pressure Control Systems (Table P-1)
- Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary

IF	THEN
DW Control Air is or becomes unavailable	PLACE the control switch for each MSRV in CLOSE/AUTO AND PLACE MSRV auto actuation logic inhibit XS-1-202 to INHIBIT

RC/P-4

Excerpt from OPL171.054 Lesson Plan:

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
air system, any leak or break inside the drywell will directly introduce oxygen into the inerted drywell.	
b. Any leakage into the drywell from the drywell control system will also contribute to a higher containment pressure and subsequent additional venting requirements.	
c. A calculation to evaluate the rupture of the DCA headers inside the drywell during an accident determined that injecting all the nitrogen from the 6000 gallon Containment Inerting system liquid nitrogen storage tank would take approximately 12 days and would only raise the drywell and suppression pool pressure to approximately 33 psig.	
d. When CAD is aligned to drywell control air only nitrogen will be introduced to the system.	
(1) If a break of the drywell control air system occurs while CAD is aligned the only net effect is a higher containment pressure and a depletion of the nitrogen supply.	
(2) The CAD to drywell crosstie provides long term MSRV accumulator gas supply in order to satisfy Station Blackout (SBO) analysis and provide an alternate supply for NFPA 805 Fire Protection requirements. It can also be used during short periods as a backup to drywell control air without compromising the SBO or NFPA analysis as long as nitrogen pressure and level are maintained above minimum values for functionality.	
e. Drywell control air can be cross-tied to supply the outboard MSIVs on all three units.	
f. Drywell Control Air supplies normal air supply to the inboard MSIVs, MSRVs, and other pneumatically operated equipment inside the drywell.	Obj ILT 8 Obj LOR 7
g. Loss of DWCA without a backup source available will force all pneumatically controlled components within the drywell to go to their fail position. For example, see items 2 and 6 below.	
2. Main Steam System	
a. MSIVs and MSRVs-The air accumulators for the MSIVs contain enough air for one closing actuation. When control air pressure drops to < 45 psig, the MSIV accumulator air will be routed to the MSIVs.	

Examination Outline Cross-reference:

295021 (APE 21) Loss of Shutdown Cooling / 4

AK1.01 (10CFR 55.41.9)

Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING:

- Decay heat

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295021AK1.01	
Importance Rating	3.6	------

Proposed Question: **# 53**

Unit 2 is in MODE 4 when the RHR Pump in service for Shutdown Cooling trips.

Subsequently, Shutdown Cooling flow is being restored to remove decay heat.

Which **ONE** of the following completes the statement below?

Given the conditions above, the **MINIMUM** required RHR Shutdown Cooling pump flowrate is __________ in accordance with 2-AOI-74-1, Loss of Shutdown Cooling.

A. 1500 gpm

B. 3500 gpm

C. 6250 gpm

D. 7000 gpm

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that 1350 gpm is the minimum RHRSW pump flowrate used to support RHR Shutdown Cooling.

B INCORRECT: Incorrect but plausible in that 1700 – 4500 gpm is the acceptable RHRSW flowrate RHR Shutdown Cooling.

C INCORRECT: Incorrect but plausible in that 6,000 – 6,500 gpm is the acceptable RHR Shutdown Cooling flowrate for MODE 5 with 1 or more fuel bundles removed from the Reactor Core.

D **CORRECT:** (*See attached*) In accordance with 2-AOI-74-1, Loss of Shutdown Cooling, RHR flow should be reestablished to remove decay heat using the restarted pump with the acceptable RHR Shutdown Cooling flowrate between 7,000 – 10,000 gpm with the Reactor in MODE 4. RHR Loop flowrate requirements to support Shutdown Cooling lowers with one or more fuel bundles removed from the Reactor Core.

RO Level Justification: Tests the candidate's knowledge of the operational flowrate required to adequately remove decay heat using the Shutdown Cooling mode of the Residual Heat Removal System. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 2-OI-74, Rev. 183 (Attach if not previously provided)

2-AOI-74-1, Rev. 40

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044 Obj. 11d (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.074-02 047 #2039
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	X
55.43	

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

2039. OPL171.074-02 047

The following conditions exist on Unit 2:

- The Reactor is shutdown in MODE 4.
- 2-AOI-74-1, Loss of Shutdown Cooling, has been entered due to a trip of the ONLY running RHR pump.

Subsequently,

- The tripped RHR pump is restarted.

Which ONE of the following completes the statement below?

In accordance with 2-AOI-74-1, Loss of Shutdown Cooling, which one of the following identifies an acceptable RHR Shutdown Cooling flow value?

A. 1500 gpm

B. 3500 gpm

C. 6250 gpm

D✓ 8000 gpm

Excerpts from 2-AOI-74-1:

BFN Unit 2	Loss of Shutdown Cooling	2-AOI-74-1 Rev. 0040 Page 4 of 30

1.0 PURPOSE

This instruction provides the symptoms and operator actions for a Loss of Shutdown Cooling.

2.0 SYMPTOMS

A. RHR Pump Trip while in Shutdown Cooling Mode:

1. RHR SYS I PUMP A(C) TRIPPED, various conditions [2-XA-55-3D, Window 13(14)]

2. RHR SYS II PUMP B(D) TRIPPED, various conditions [2-XA-55-3E, Window 13(14)]

B. Low RHR Shutdown Cooling Flow while in Shutdown Cooling Mode:

1. RHR SD CLG FLOW LOW at 3700 GPM (2-XA-55-3D, Window 11)

C. Automatic isolation (PCIS Group II or 100 psig) or manual isolation of RHR System while in Shutdown Cooling Mode:

1. RHR SYS I/II DISCH OR SD CLG HDR PRESS HIGH at 100 psi (2-XA-55-3E, Window 32).

2. 2-FCV-74-47, RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV CLOSED.

3. 2-FCV-74-48, RHR SHUTDOWN COOLING SUCT INBD ISOL VLV CLOSED.

4. 2-FCV-74-53, RHR SYS I LPCI INBD INJECT VALVE CLOSED.

5. 2-FCV-74-67, RHR SYS II LPCI INBD INJECT VALVE CLOSED.

D. Loss of RHRSW while in Shutdown Cooling Mode:

1. RHRSW HDR PRESS LOW at 50 psi lowering (2-XA-55-3E, Window 31)

E. High RHR cooling water temperature while in Shutdown Cooling Mode:

1. RHR/FPC HX OUTLET TEMP HIGH at 125°F (2-XA-55-3E, Window 18)

   a. Indication of pressure on RPV or rising Reactor coolant temperature while in a Cold Shutdown condition (Mode 4 or Mode 5).

   b. Indication of RPV water level below Level 3 or Drywell pressure above 2.45 psig.

BFN Unit 2	Loss of Shutdown Cooling	2-AOI-74-1 Rev. 0040 Page 13 of 30

4.2 Subsequent Actions (continued)

[14] **IF** the Group 2 PCIS Isolation has been reset, **THEN** (Otherwise **N/A**)

RETURN the affected loop of RHR to Shutdown Cooling as follows. ☐

[14.1] **CLOSE** RHR SYS I(II) LPCI OUTBD INJECT VALVE, 2-FCV-74-52(66). ☐

[14.2] **OPEN** RHR SYS I(II) LPCI INBD INJECT VALVE, 2-FCV-74-53(67) ☐

[14.3] **VERIFY** RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 2-HS-74-148(149) in **INHIBIT** ☐

[14.4] **VERIFY CLOSED** RHR SYSTEM I(II) MIN FLOW VALVE, 2-FCV-74-7(30). ☐

[14.5] **VERIFY CLOSED** RHR PUMP 2A(2B) and 2C(2D) SUPPR POOL SUCT VLVs, 2-FCV-74-1(24) and 2-FCV-74-12(35). ☐

NOTE

EQV 70933 requires 2-FCV-074-0002, 0013, 0025 and 0036 requires that due to limitations on the valves actuator, the valves shall not be stroked OPEN if the differential pressure across the disc exceeds 82 psid.

[14.6] **VERIFY OPEN** RHR PUMP 2A(2B) and 2C(2D) SD COOLING SUCT VLVs, 2-FCV-74-2(25) and 2-FCV-74-13(36). ☐

[14.7] **OPEN** RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48 ☐

[14.8] **IF** the tripped pump has been determined to be in operating condition and with Unit Supervisor permission, **THEN:**

RESTART tripped RHR pump(s) RHR PUMP 2A(2C)(2B)(2D) using 2-HS-74-5A(16A)(28A)(39A) ☐

Also supports Distractor 'C':

4.2 Subsequent Actions (continued)

[14.9] **THROTTLE** RHR SYS I(II) LPCI OUTBD INJECTION VALVE, 2-FCV-74-52(66), to establish and maintain RHR flow as indicated by 2-FI-74-50(64), RHR SYS I(II) FLOW, as follows: ☐

RHR Pumps in Operation	1	2
Loop Flow	7,000 to 10,000	14,000 to 20,000
Loop Flow (1 or more fuel bundles removed from core)	6,000 to 6,500	N/A

[14.10] **WHEN** time permits after RHR pump is started, **THEN**

VERIFY RHR Pump Breaker charging spring recharged by observing amber breaker spring charged light is on and closing spring target indicates charged. ☐

[14.11] **VERIFY** inservice RHRSW pump for the appropriate header. **REFER TO** 0-OI-23. ☐

[14.12] **SLOWLY THROTTLE** RHR HX 2A(2C)(2B)(2D) RHRSW OUTLET VALVE, 2-FCV-23-34(40)(46)(52), to obtain desired cooldown rate. ☐

[15] **PERFORM** the following as required:

[15.1] **RAISE** RWCU flow rate to maximum AND **MAXIMIZE** RWCU blowdown as required to maintain reactor coolant temperatures less than 200°F on all indications. **REFER TO** 2-OI-69. ☐

[15.2] **RAISE** CRD flow. **REFER TO** CRD Pump Operation at Elevated Flow section. **REFER TO** 2-OI-85 ☐

Excerpt from 2-OI-74: Supports Distractors ‘A’ and ‘B’:

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0183 Page 142 of 521

8.8.1 Initiation / Operation of RHR Loop I in Shutdown Cooling (continued)

CAUTIONS

1) When little decay heat is present, RHR Heat Exchanger RHRSW Outlet Valves should be throttled very slowly to prevent excessive cooldown rates.

2) **DO NOT EXCEED** 4500 gpm RHRSW flow through any RHR Heat Exchanger or cooldown rate of greater than 90°F/hr.

3) During RHRSW low flows, such as shutdown cooling split flows, the initial flow rate from any RHRSW heat exchanger is required to be greater than or equal to 600 gpm. This flow rate ensures operation of the off-line radiation monitor. Off-line monitors receive their start signal from a TDPU relay which is energized by the RHRSW heat exchanger's discharge flow rate greater than 600 gpm. Upon reaching this flow rate, the flow may be lowered or split as desired to establish a cooldown rate or maintain consistent shutdown temperatures.

4) When operating RHRSW through the heat exchangers, damage can occur to the RHRSW discharge valves for the RHR Heat Exchanger if operating at low flows and high differential pressures for long periods. In order to lower the differential pressure the valves experience, flow through the in service heat exchanger(s) should be established such that the total header flow is greater than or equal to 4000 gpm. When operating RHRSW in split mode with other units, this is calculated by adding the individual flows from each of the in service RHR heat exchangers. BFPER 00-003901-000

5) It may be necessary to establish RHRSW flow through another unit's heat exchanger or through EECW to prevent operating the RHRSW pump at less than 1700 gpm. (Minimum flow for B1 RHRSW pump is 1350 gpm) REFER TO 0-OI-23.

[30.3] **SLOWLY THROTTLE** 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV, to obtain desired cooldown rate of less than or equal to 90°F/hr.

[30.4] **IF** RWCU is **NOT** in service, **THEN**:

PLACE RWCU in service. **REFER TO** 2-OI-69. (May be marked N/A when in Mode 5 and Fuel Pool Cooling gates removed.)

[30.5] At Panel 2-9-21, 2-TR-74-80, Point 3,

CHECK RHR HX A/C COM DISCH temperature less than 140°F.

Examination Outline Cross-reference:	Level	RO	SRO
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5	Tier #	1	------
EK3.05 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE:	K/A #	295028EK3.05	
• Reactor SCRAM	Importance Rating	3.6	------

Proposed Question: **# 54**

Which **ONE** of the following completes the statement below?

When Drywell Temperature reaches ___**(1)**___ entry into EOI-2, Primary Containment Control is required.

In accordance with EOI-2, the Reactor is required to be SCRAMMED in preparation for spraying the Drywell before Drywell Temperature rises to ___**(2)**___.

A. (1) 150 ºF
(2) 200 ºF

B. (1) 150 ºF
(2) 280 ºF

C. (1) 160 ºF
(2) 200 ºF

D. (1) 160 ºF
(2) 280 ºF

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 150 ºF is the Technical Specification Limit for Drywell Temperature. The second part is incorrect but plausible in that in previous EOI-2, Primary Containment Control revisions, before Drywell Temperature reached 200 ºF, a Reactor SCRAM was required to before continuing.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT**: *(See Attached)* The entry condition for EOI-1 is 160 ºF. For second part, in accordance with 2-EOI-2, DW/T-4, before Drywell Temperature reaches 280 ºF, EOI-1 is entered. In accordance with EOIPM 0-V-D, EOI-1 is entered to ensure the Reactor is SCRAMMED before Drywell Sprays are initiated.

RO Level Justification: Tests the candidate's knowledge of the reasons for a Reactor SCRAM as it applies to High Drywell Temperature. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (3) The progression of an event.

Technical Reference(s): 2-EOI-2, Rev. 16 (Attach if not previously provided)
EOIPM 0-V-D, Rev. 2
Tech Spec 3.6.1.4, Amend.253

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203 Obj. 4 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1501 #14	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2015	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTION 14 Rev 3

Which ONE of the following completes the statements below?

In accordance with EOI-2, Primary Containment Control, step DW/T-4, before Drywell Temperature rises to __ (1) __ entry into EOI-1, RPV Control, is required.

In accordance with EOIPM 0-V-D, Primary Containment Control Bases, the reason for entering EOI-1 at step DW/T-4 is to scram before __ (1) __.

A. (1) 160 °F
(2) Drywell sprays are initiated

B. (1) 160 °F
(2) the Suppression Chamber design temperature limit is exceeded

C. (1) 200 °F
(2) Drywell sprays are initiated

D. (1) 200 °F
(2) the Suppression Chamber design temperature limit is exceeded

Answer: C

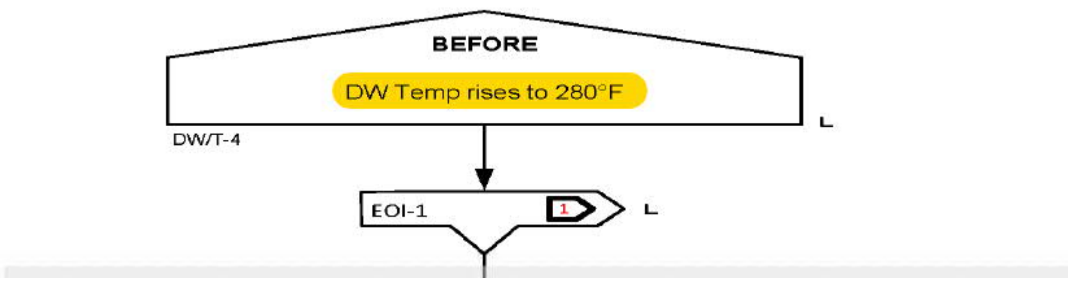
Excerpt from 2-EOI-2:

Excerpt from EOIPM 0-V-D:

BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V-D Rev. 0002 Page 18 of 119

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DW/T-4



BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V-D Rev. 0002 Page 19 of 119

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: DW/T-4

If drywell temperature cannot be controlled by operation of all available drywell cooling, the RPV Control flowchart is entered and drywell spray is required before applicable component qualification and structural design temperature limits are reached. Entering the RPV Control flowchart ensures that, if possible, the reactor is scrammed before drywell sprays are initiated and in anticipation of possible RPV depressurization in this flowpath. This helps ensure that actions are taken to limit the drywell temperature increase prior to substantially exceeding the temperature limits of non-EQ equipment.

Entry into the RPV Control flowchart must be explicitly stated because conditions requiring entry into the Primary Containment Control flowchart do not necessarily require entry into the RPV Control flowchart. Therefore, a scram may not have yet been initiated. Directing that the RPV Control flowchart be entered, rather than explicitly stating here "Initiate a reactor scram," coordinates actions currently being executed if the RPV Control flowchart has already been entered. (Note that the RPV Control flowchart requires initiating a reactor scram only if one has not previously been initiated.) In addition, entry to the RPV Control flowchart must be made because it is through that flowchart that the transfer to Flowchart C2, Emergency RPV Depressurization, is affected.

Entry to the RPV Control flowchart before reaching the drywell design temperature provides the following benefits:

- The recirculation pumps are not qualified for continuous operation in a spray environment and are therefore tripped before drywell sprays are initiated. The reactor is scrammed before the recirculation pumps are tripped to reduce the magnitude of the transient and avoid high power-to-flow conditions.
- While the increase in core void fraction following emergency RPV depressurization in subsequent steps would temporarily shut down the reactor, a potential for subsequent core damage exists and sudden insurges of cold water could result in power spikes as RPV pressure decreases below the shutoff head of low pressure injection systems. Emergency depressurization with the reactor at power should therefore be avoided.
- The override in the RC/P flowpath of the RPV Control flowchart permits rapid depressurization through the main turbine bypass valves in anticipation of emergency RPV depressurization.

Excerpt from Tech Spec 3.6.1.4:

3.6 CONTAINMENT SYSTEMS

3.6.1.4 Drywell Air Temperature

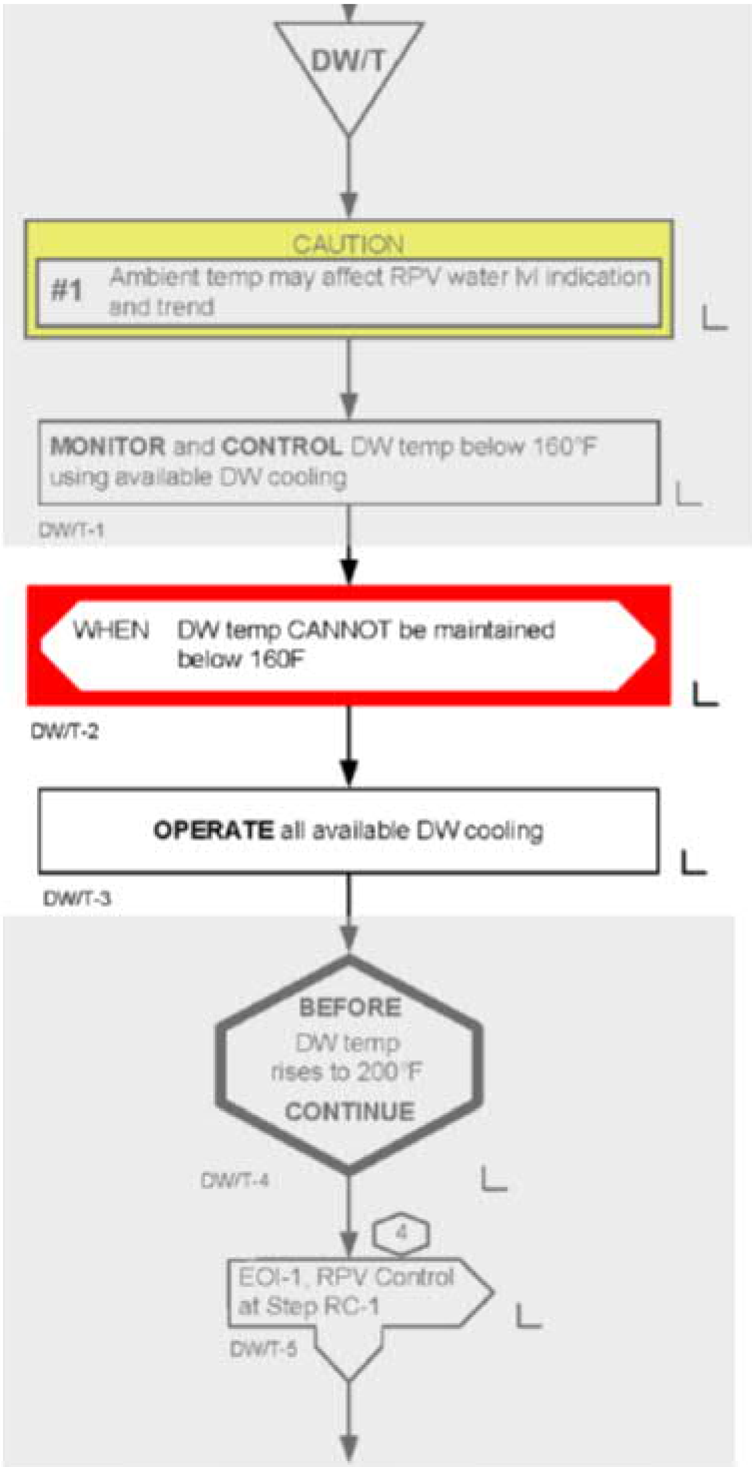
LCO 3.6.1.4 Drywell average air temperature shall be ≤ 150°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell average air temperature not within limit.	A.1 Restore drywell average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours

Excerpt from older EOI-2 revision: Supports Distractor A(2), C(2)



Examination Outline Cross-reference:	Level	RO	SRO
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9	Tier #	1	------
G2.4.34 (10CFR 55.41.10)	Group #	2	------
Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	K/A #	295034G2.4.34	
	Importance Rating	4.2	------

Proposed Question: **# 55**

In accordance with the EOI Appendices, which **ONE** of the following completes the statements below with respect to restoring Secondary Containment **VENTILATION**?

High Radiation Isolation signals ___**(1)**___ be bypassed.

Upon entering EOI-3, Secondary Containment Control, the respective ventilation isolations are bypassed to restore ___**(2)**___.

A. (1) can NOT
(2) Reactor Zone Ventilation **ONLY**

B. (1) can NOT
(2) Reactor Zone Ventilation **AND** Refuel Zone Ventilation

C. (1) can
(2) Reactor Zone Ventilation **ONLY**

D. (1) can
(2) Reactor Zone Ventilation **AND** Refuel Zone Ventilation

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct *(See B).* The second part is incorrect but plausible if the candidate assumes that ventilation isolations are bypassed to restore Reactor Zone Ventilation only as related to each specific Unit given that the Refuel Zone is one common area for all three Units.

B **CORRECT**: (*See attached*) EOI Appendices are written to provide a means for bypassing various isolation signals to allow the restoration of some systems. The low Reactor Water Level of (+) 2 inches and high Drywell Pressure of 2.45 psig isolation signals can be bypassed in accordance with EOI-Appendix-8E, Bypassing Group 6 Low RPV Level and High Drywell Pressure Isolation Interlocks. However, the High Radiation isolation signals from the Reactor and Refuel Zone Radiation Monitors cannot be bypassed, but it is reasonable to assume that any signal could be bypassed in accordance with the EOI Appendices. For second part, in accordance with EOI-3, if Reactor AND Refuel Zone ventilation is isolated and their respective exhaust radiation levels are less than 72 mr/hr, then perform EOI-Appendix-8F, Restoring Refuel Zone and Reactor Zone Ventilation Fans Following Group 6 Isolation. If necessary, defeat/bypass isolation interlocks by performing EOI-Appendix-8E, at Panel 9-15 in the Aux Instrument Room using jumpers.

C INCORRECT: The first part is incorrect but plausible in that in accordance with EOI-Appendix-8E, some signals can be bypassed for a Group 6 PCIS Isolation; however the High Radiation Isolation signals from Reactor and Refuel Radiation Monitors cannot be bypassed. It is reasonable to assume that any signal could be bypassed in accordance with the EOI Appendices. The second part is incorrect but plausible *(See A).*

D INCORRECT**:** The first part is incorrect but plausible (*See C*). The second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of RO tasks outside the Control Room to bypass a high radiation signal that causes a PCIS Group 6 Isolation in accordance with the Emergency Operating instructions. This question is rated as Memory due to the fact that it requires the strict recall of facts.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 2-EOI-Appendix-8E, Rev. 3 (Attach if not previously provided)
2-EOI-Appendix-8F, Rev. 6
2-730E927-7, Rev. 24
OPL171.017, Rev. 21

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.017, Obj. 3c (As available)

Question Source:	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
	Last NRC Exam		

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 **X**

Comments:

Excerpt from 2-EOI-Appendix-8E:

BFN Unit 2	Bypassing Group 6 Low RPV Level and High Drywell Pressure Isolation Interlocks	2-EOI Appendix-8E Rev. 0003 Page 3 of 4

1.0 INSTRUCTIONS

Location:	Unit 2 Auxiliary Instrument Room
Attachments:	1. Tools and Equipment

[1] **REFER** to Attachment 1 and **OBTAIN** two banana jack jumpers from the EOI Equipment Storage Box. ☐

[2] **BYPASS** Group 6 Low RPV Level and High Drywell Pressure Isolation Interlocks as follows:

[2.1] **LOCATE** terminal strip BB in Panel 9-15, Bay 3, Rear ☐

[2.2] **JUMPER** BB-22 to BB-23, Panel 9-15 ☐

[2.3] **LOCATE** terminal strip DD in Panel 9-15, Bay 1, Rear ☐

[2.4] **JUMPER** DD-22 to DD-23, Panel 9-15. ☐

[3] **NOTIFY** Unit Operator that Group 6 RPV Low Level and High Drywell Pressure Isolation Interlocks are bypassed. ☐

END OF TEXT

Excerpt from 2-EOI Appendix-8F:

BFN Unit 2	Restoring Refuel Zone and Reactor Zone Ventilation Fans Following Group 6 Isolation	2-EOI Appendix-8F Rev. 0006 Page 3 of 6

1.0 INSTRUCTIONS

LOCATION:	Unit 2 Control Room
ATTACHMENTS	None

[1] **VERIFY** PCIS Reset. ☐

[2] **PLACE** Refuel Zone Ventilation in service as follows (Panel 2-9-25):

[2.1] **VERIFY** 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, is in OFF. ☐

NOTE

When Refuel Zone supply and exhaust fans start, Refuel Zone supply and exhaust dampers open automatically.

[2.2] **PLACE** 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, to SLOW A (SLOW B). ☐

[2.3] **CHECK** two SPLY/EXH A (B) green lights above 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, extinguish and two SPLY/EXH A (B) red lights illuminate. ☐

[2.4] **VERIFY OPEN** the following dampers:

- 2-FCO-64-5, REFUEL ZONE SPLY OUTBD ISOL DMPR ☐
- 2-FCO-64-6, REFUEL ZONE SPLY INBD ISOL DMPR ☐
- 2-FCO-64-9, REFUEL ZONE EXH OUTBD ISOL DMPR ☐
- 2-FCO-64-10, REFUEL ZONE EXH INBD ISOL DMPR. ☐

BFN Unit 2	Restoring Refuel Zone and Reactor Zone Ventilation Fans Following Group 6 Isolation	2-EOI Appendix-8F Rev. 0006 Page 4 of 6

1.0 INSTRUCTIONS (continued)

[3] **PLACE** Reactor Zone Ventilation in service as follows (Panel 2-9-25):

[3.1] **VERIFY** 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS, is in OFF. ☐

NOTE
When Reactor Zone supply and exhaust fans start, Reactor Zone supply and exhaust dampers open automatically.

[3.2] **PLACE** 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS, in SLOW A (SLOW B). ☐

[3.3] **CHECK** two SPLY/EXH A (B) green lights above 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS, extinguish and two SPLY/EXH A (B) red lights illuminate. ☐

[3.4] **VERIFY OPEN** the following dampers:

- 2-FCO-64-13, REACTOR ZONE SPLY OUTBD ISOL DMPR ☐
- 2-FCO-64-14, REACTOR ZONE SPLY INBD ISOL DMPR ☐
- 2-FCO-64-42, REACTOR ZONE EXH INBD ISOL DMPR ☐
- 2-FCO-64-43, REACTOR ZONE EXH OUTBD ISOL DMPR. ☐

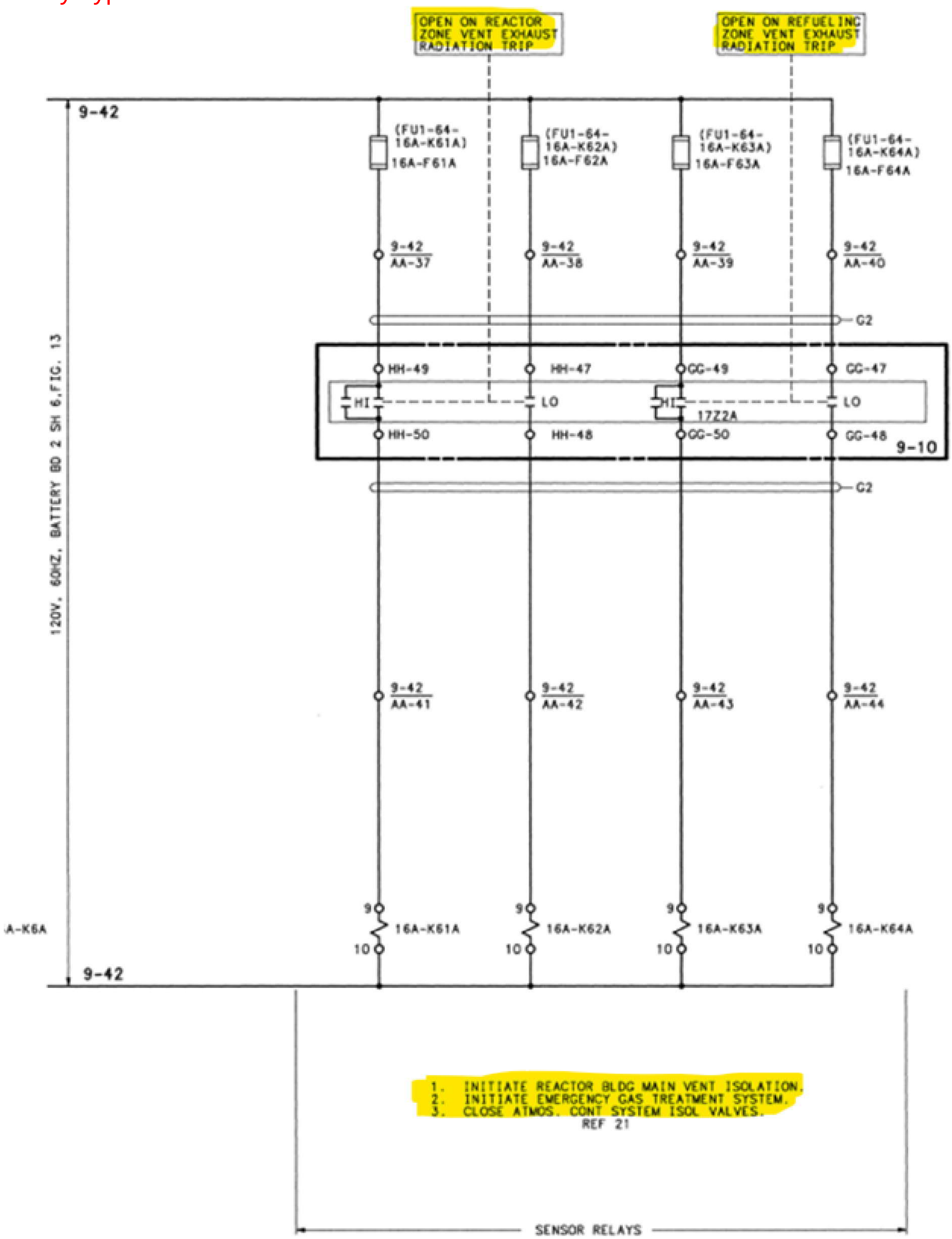
Excerpt from OPL171.017 Lesson Plan:

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 21

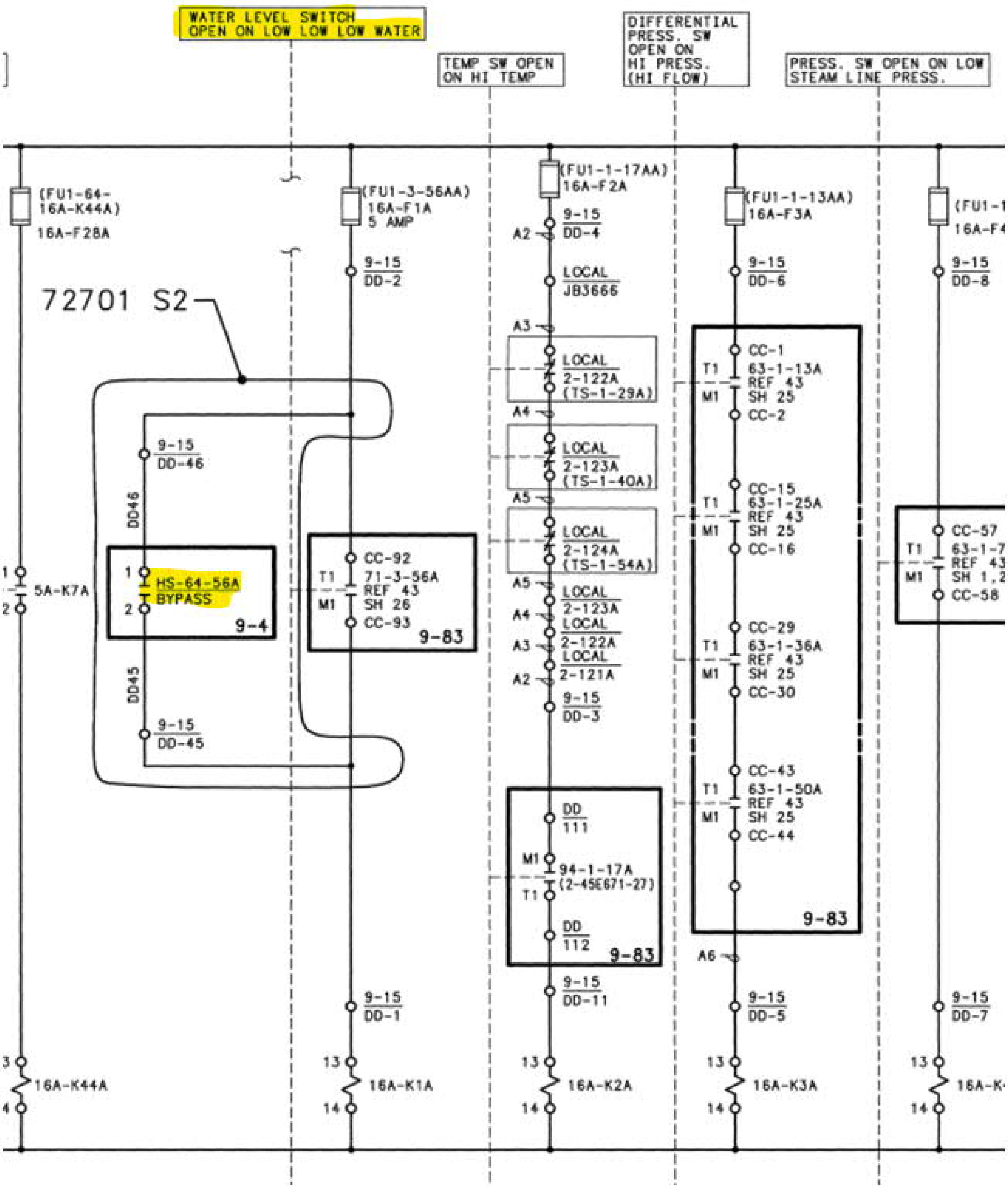
Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
b) A brief description of available Isolation bypasses is: (1) Group 1 (a) RPV low low-low level (-122" Level 1), is bypassed by the installation of jumpers per EOI Appendix 8A. All Isolations bypassed by jumper installation per EOI Appendix 11H.	ILT- 3c LOR- 3c 2-730E927-8,9 DCN72701 replaces jumpers installed per EOI APPX 8A with four Keylocks on 9-4. U1/U2 completed, U3 spring 2020
(2) Group 2 (a) The RPV low level (+2" or Level 3) and Drywell High Pressure (2.45 psig) Isolation signals to the PSC Head Tank Pump Isolation valves (FCV-75-57, 58) are bypassed by installing jumpers per EOI Appendix 7G. This is done to allow the PSC Head Tank Pumps to be used as an alternate injection system.	ILT- 3c LOR- 3c
(3) Group 4 (a) The HPCI Low Reactor Pressure Isolation can be bypassed by de-terminating a wire per EOI Appendix 16C. High temperature Isolation can be bypassed by booting contacts per EOI Appendix 16K.	ILT- 3c LOR- 3c
(4) Group 5 (a) The RCIC Low Reactor Pressure Isolation can be bypassed by de-terminating a wire per EOI Appendix 16A. High temperature Isolation can be bypassed by booting contacts per EOI Appendix 16K.	ILT- 3c LOR- 3c
(5) Group 6 (a) The Group 6 RPV Low Level (+2" or Level 3)) and Drywell High Pressure (2.45 psig) Isolations can be bypassed by installing jumpers per EOI Appendix 8E.	ILT- 3c LOR- 3c 730E927RF sheet 16, 17,18
(b) The DW/SUPPR CHBR TRAIN A(B) VENT Keylock Switches (HS-84-35/36) and the TRAIN A(B) VENT TO SGT SYSTEM Keylock Switches (HS-84-20/19) Panels 9-54/55 are used to bypass all Group 6 Isolations as necessary to allow containment venting through SGT (or through large vent path) per EOI Appendix 12 and 13. Cannot vent with loss of RPS A.	ILT- 3c LOR- 3c Normally for refueling outages
(c) The Analyzer A(B) SAMPLE ISOLATION BYPASS Keylock switches (HA-76-69/79) on Panels 9-54/55 are used to bypass all Group 6 Isolations as necessary to allow placing the H2/O2 Analyzers in service per EOI-2, Step PC/H-1.	ILT- 3c LOR- 3c

Excerpts from 2-730E927-7: Illustrates Reactor/Refuel Zone Vent Exhaust Radiation Trip signals cannot by bypassed



Supports Distractors C(1), D(1), illustrates bypass capability for Low Reactor Water Level



Examination Outline Cross-reference:	Level	RO	SRO
205000 (SF4 SCS) Shutdown Cooling	Tier #	2	------
A1.03 (10CFR 55.41.5)	Group #	1	------
Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including:	K/A #	205000A1.03	
• Recirculation loop temperatures	Importance Rating	3.3	------

Proposed Question: **# 56**

Unit 1 is being shut down for an outage with the following conditions:

- 1-HS-99-5A-S1, REACTOR MODE SWITCH is in SHUTDOWN
- Reactor Vessel Head Closure Bolts are still fully tensioned
- Residual Heat Removal (RHR) Loop I is in Shutdown Cooling

Subsequently, a complete Loss of Shutdown Cooling occurs resulting in the following:

TIME	Reactor Coolant Temperature (°F)
0800	110 °F
0802	114 °F
0804	118 °F

Given the conditions above, which **ONE** of the following completes the statements below?

The current Heatup Rate ___**(1)**___ within the limit specified in Tech Spec 3.4.9, RCS Pressure and Temperature Limits?

Given that all other associated MODE requirements remain unchanged and based upon the constant trend ___**(2)**___ is the **EARLIEST** time that Unit 1 will enter MODE 3 due to the rising Reactor Coolant Temperature.

A. (1) is
(2) 0852

B. (1) is
(2) 0845

C. (1) is NOT
(2) 0852

D. (1) is NOT
(2) 0845

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that Tech Spec 3.4.9 is applicable at all times. Candidates easily confuse the heatup rate criteria. The target heatup rate of 80 °F/hr is confused to be the same as the Tech Spec limit specified in the surveillance. During startup or shut down, Operations targets the optimum heatup/cooldown rate limit of 20 °F every 15 minutes which equates to 80 °F/hr. This ensures the administrative limit of 90 °F/hr is not exceeded which provides margin to the 100 °F/hr Tech Spec as listed in the SR. The second part is correct *(See C)*.

B INCORRECT: The first part is incorrect but plausible *(See A)*. Second part is incorrect but plausible in that this time is based upon the candidate choosing the wrong temperature of 200 °F for the MODE change versus the actual temperature of 213 °F. In this case, subtracting a starting temperature of 110 °F from 200 °F results in 90 °F delta. 90 °F ÷ 120 °F/hr = .75 hours; which yields .75 x 60 = 45 minutes. Then add 45 minutes to the 0800 starting time for calculating heatup rate. This would incorrectly result in the EARLIEST MODE change time being 0845.

C **CORRECT**: *(See attached)* In accordance with 1-AOI-74-1, Loss of Shutdown Cooling, heatup/cooldown rate Surveillance Requirement (SR) is to be performed. The provided data in the table analytically indicates a heatup rate of 4 °F every two (2) minutes. This translates to a heatup rate of 120 °F/hr which is NOT within the limit of the procedurally specified Tech Spec 3.4.9 Limit in the SR of 100 °F/hr. For second part, subtracting a starting temperature of 110 °F from 213 °F (Where MODE change actually occurs in reality) results in 103 °F delta. 103 °F ÷ 120 °F/hr = .8583 hours; which yields .8583 x 60 = 51.5 minutes added to the 0800 starting time for calculating heatup rate. This would result in the EARLIEST MODE change time being between 0851 and 0852.

D INCORRECT: First part is correct *(See C)*. The second part is incorrect but plausible *(See B)*.

RO Level Justification: Tests the candidate's ability to predict and monitor changes in parameters associated with Shutdown Cooling as it relates to Recirculation Loop Temperatures. This question is rated as C/A due to the requirement to assemble, sort, and integrate at least two different parts of the question to predict an outcome. This requires mentally using specific knowledge and performing calculations to determine the correct outcome. The candidate must calculate correctly and integrate the data with current plant conditions, understand and recall procedure requirements, and apply MODE specifics from Tech Specs.

Technical Reference(s): 1-AOI-74-1, Rev. 8 (Attach if not previously provided)
1-SR-3.4.9.1(1), Rev. 14
1-GOI-100-1A, Rev. 56
1-GOI-100-12A, Rev. 26
Unit 1 Tech Spec 1.0, Amend 234
Unit 1 Tech Spec 3.4.9, Amend 234

Proposed references to be provided to applicants during examination: **Reactor Coolant Temperature and Time Table**

Learning Objective:	OPL171.04 Obj. 7, 11d	(As available)	
Question Source:	Bank #		
	Modified Bank #	BFN 1909 #20	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

Proposed Question: **# 20**

Unit 2 is being shut down for an outage with the following conditions:

- 2-HS-99-5A-S1, REACTOR MODE SWITCH is in SHUTDOWN
- Reactor Vessel Head Closure Bolts are still fully tensioned
- Residual Heat Removal (RHR) Loop I is in Shutdown Cooling

Subsequently, a complete Loss of Shutdown Cooling occurs and results in the following Reactor Coolant Temperature response:

TIME	Reactor Coolant Temperature (°F)
0800	113 °F
0802	116 °F
0804	119 °F

Given the conditions above, which **ONE** of the following completes the statements below?

The current Heatup Rate is ___**(1)**___the limit specified in 2-SR-3.4.9.1(1), Reactor Heatup and Cooldown Rate Monitoring.

Given that all other associated MODE requirements remain unchanged and based upon the constant trend, ___**(2)**___ is the **EARLIEST** time that Unit 2 will enter MODE 3 due to the rising Reactor Coolant Temperature.

A. **(1)** below
(2) 0858

B. **(1)** below
(2) 0906

C. **(1)** above
(2) 0858

D. **(1)** above
(2) 0906

Proposed Answer: **B**

Excerpt from 1-AOI-74-1:

BFN Unit 1	Loss of Shutdown Cooling	1-AOI-74-1 Rev. 0008 Page 10 of 29

4.2 Subsequent Actions (continued)

NOTES

1) With the Reactor in Cold Shutdown Condition (Mode 4 or Mode 5), reactor coolant stratification may be indicated by one of the following:

- Reactor pressure above 0 psig with any reactor coolant temperature indication reading at or below 212°F.
- Differential temperatures of 50°F or greater between either RX VESSEL BOTTOM HEAD (FLANGE DR LINE) 1-TE-56-29 (8) temperatures and RX VESSEL FW NOZZLE N4B END (N4B INBD)(N4D END)(N4D INBD) 1-TE-56-13(14)(15)(16) temperatures from the REACTOR VESSEL METAL TEMPERATURE recorder, 1-TR-56-4.
- With recirculation pumps and shutdown cooling out of service, a Feedwater sparger temperature of 200°F or greater on any RX VESSEL FW NOZZLE N4B END (N4B INBD)(N4D END)(N4D INBD) 1-TE-56-13(14)(15)(16) temperatures from the REACTOR VESSEL METAL TEMPERATURE recorder, 1-TR-56-4.

2) [NER/C] For purposes of thermal stratification monitoring, the bottom head drain line is more representative as long as there is flow in the line. [GE SIL 251 and 430]

[7] **PLOT** heatup/cooldown rate as necessary in accordance with 1-SR-3.4.9.1(1). ☐

[8] **REQUEST** the SRO to estimate the following times at least once per shift until a method of decay heat removal is restored:

[8.1] **DETERMINE** the time since shutdown. ☐

[8.2] **DETERMINE** the current RPV heat-up rate from 1-SR-3.4.9.1(1) or **USE** Illustration 1 if reactor coolant stratification is suspected. ☐

[8.2.1] **IF** additional information is required to determine the heat-up rates, **THEN**

NOTIFY Reactor Engineer. ☐

Excerpt from 1-SR-3.4.9.1(1):

BFN Unit 1	Reactor Heatup and Cooldown Rate Monitoring	1-SR-3.4.9.1(1) Rev. 0014 Page 4 of 20

1.0 INTRODUCTION

1.1 Purpose

This Surveillance Procedure is performed to ensure the operating limits on the Reactor Vessel Pressure and Temperature, during Reactor Coolant System (RCS) Heatup or Cooldown Operations with the Reactor critical, are satisfied within the limits specified in Figure 3.4.9-1, Curves #1 and #2. 1-SR-3.4.9.1(2) will be used to ensure these requirements during In-Service Hydrostatic or Leak Testing. The average rate of Reactor Coolant Temperature change is required to **NOT** exceed 100 degrees F per hour whenever averaged over a one-hour period and RCS pressure and temperature is required to be within the limits specified in Figure 3.4.9-1, Curve #3.

1.2 Scope

This procedure fully implements the requirements of Technical Specification (TS) Surveillance Requirement (SR) 3.4.9.1.b and 3.4.9.2 for Heatup and Cooldown, and Figure 3.4.9-1, (Curves #2 and #3). TS requirements for RCS In-Service Hydrostatic or Leak Testing are covered by 1-SR-3.4.9.1(2).

1.3 Frequency

This procedure is performed 15 minutes prior to control rod withdrawal for the purpose of achieving criticality and every 30 minutes, during RCS Heatup and Cooldown Operations, in Modes 2 or 3. This SR is **NOT** required, during Heatup Operations, in Modes 4 or 5. Data recording may conclude whenever:

During Heatup, Reactor Pressure is >900 psig and any two of the five instruments recorded change by <10°°F over a 30 minute time period.

During Cooldown, Reactor Coolant Temperature is <190°F and any two of the five instruments recorded change by <10°F over a 30 minute time period.

During Heatup or Cooldown hold periods, on occasions where any two of the five instruments recorded change by <10°F over a 30 minute time period.

1.4 Applicability

LCO - At ALL Times

Excerpt from Unit 1 Tech Spec 3.4.9:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE------------ Required Action A.2 shall be completed if this Condition is entered. ---------------------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.9.1	-------------------------NOTES----------------------- 1. Only required to be performed during RCS heatup and cooldown operations or RCS inservice leak and hydrostatic testing when the vessel pressure is > 312 psig. 2. The limits of Figure 3.4.9-2 may be applied during nonnuclear heatup and ambient loss cooldown associated with inservice leak and hydrostatic testing provided that the heatup and cooldown rates are ≤ 15°F/hour. 3. The limits of Figures 3.4.9-1 and 3.4.9-2 do not apply when the tension from the reactor head flange bolting studs is removed. --- Verify: a. RCS pressure and RCS temperature are within the limits specified by Curves No. 1 and No. 2 of Figures 3.4.9-1 and 3.4.9-2; and b. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period.	30 minutes
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.9-1, Curve No. 3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

Excerpt from 1-GOI-100-12A:

BFN Unit 1	Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations	1-GOI-100-12A Rev. 0026 Page 12 of 98

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General:

A. Unit Supervisor's permission is required to reject water to main condenser from Reactor Water Cleanup (RWCU) System without RWCU filter in service.

B. Maximum cooldown rate is 90°F per hour.

C. One method of Reactor decay heat removal should not be stopped prior to establishing another method.

D. [II/F] Prior to initiating any event that adds or has potential to add heat energy to the Suppression Chamber; the Unit Supervisor shall evaluate the necessity of placing Suppression Pool Cooling in service. This is due to the potential of developing thermal stagnation during sustained heat additions. [II-B-91-129]

E. Prior to opening any MSIV or MSIV drain isolation valve following Mode 4 or 5, operators are to verify secondary containment integrity will be maintained. Both an inboard and outboard MSIV on the same main steam line OR an inboard MSIV and both main steam line drain isolation valves are NOT to be opened at the same time unless Unit 1 Outage Shift Manager or Secondary Containment System engineer confirm no work in progress which could violate Secondary Containment or that a Secondary Containment Breach Permit is in place to support the activity.

Work is not to commence on an outboard MSIV unless 1-OI-64 has been addressed for the requirement to use extended secondary containment.

F. Operators are to identify and use Multiple and Diverse Indication of Reactor Water Level (RWL) during significant changes in RWL inventory. [INPO IER: 3-12-19]

Excerpt from 1-GOI-100-1A:

BFN Unit 1	Unit Startup	1-GOI-100-1A Rev. 0056 Page 19 of 207

3.2.3 Coolant and Metal Temperatures

A. Lowering reactor head flange and/or head temperature below the temperature of fully tensioning reactor head bolts may result in bolt relaxation and potential leakage when reactor vessel is pressurized during startup.

B. The following limitations apply to reactor heatup and/or cooldown:

1. When Reactor coolant temperature is less than 215°F, a maximum heatup rate limit of 40°F/hr will reduce the Oxygen and Hydrogen Peroxide content of the coolant, and mitigate the base metal from the reactor flange closure region from exceeding a 100°F/hr maximum Heatup rate limit. In addition, it is recommended that Reactor Criticality from a cold shutdown condition should occur at temperatures greater than 180°F. [PER 558516]

2. During Reactor heatup with reactor coolant temperature greater than or equal to 215°F, and during Reactor Cooldown, the optimum rate of temperature change is 20°F every 15 minutes. This ensures the administrative limit of 90°F/Hr is NOT exceed.

   Do NOT Attempt to "makeup" for time intervals which fall short of 20°F. If the 20°F is exceeded in any 15 minute period, subtract the amount of heatup/cooldown rate over 20°F from the 20°F for the next 15 minute period. These guidelines assist in achieving a target heatup/cooldown rate of 80°F/Hr and ensure the administrative limit of 90°F/Hr is NOT exceeded.

3. Past experience has shown that during startup from cold conditions RPV flange metal temperature heatup rates of greater than 100°F could occur when the MSIVs are opened later in the startup. To prevent this from occurring, the Reactor should NOT be taken critical until moderator temperature is at least 180°F. This will provide sufficient time for the flange

Excerpt from Unit 1 Tech Spec 1.0:

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[(a)] or Startup/Hot Standby	NA
3	Hot Shutdown[(a)]	Shutdown	> 212
4	Cold Shutdown[(a)]	Shutdown	≤ 212
5	Refueling[(b)]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Examination Outline Cross-reference:	Level	RO	SRO
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling	Tier #	2	------
K2.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of electrical power supplies to the following: • RCIC initiation signals (logic)	K/A #	217000K2.02	
	Importance Rating	2.8*	------

Proposed Question: **# 57**

Unit 2 has experienced a loss of 250 VDC RMOV BOARD 2B.

Which **ONE** of the following statements below describes the effect on the RCIC system?

A. **ONLY** the RCIC Flow Controller would fail downscale.

B. RCIC will **NOT** automatically **INITIATE** upon a valid signal.

C. RCIC will **NOT** automatically **ISOLATE** upon a valid signal.

D. RCIC will initiate upon a valid signal and **ONLY** trip on High Reactor Water Level.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** Incorrect but plausible in that the RCIC Flow Controller is powered from the Div I ECCS ATU Inverter, which has lost power, but it is not the ONLY effect of the loss of the 2B 250VDC RMOV Board.

B **CORRECT:** (*See attached*) In accordance with 2-ARP-9-3C, Window 1, RCIC isolation, initiation, and trip logic is lost when Div I of RCIC Logic loses its power supply.

C INCORRECT**:** Incorrect but plausible in that while Logic Bus ‘A’ of RCIC Logic is lost (which causes a loss of isolation logic for Div I), however the isolation logic has a redundant channel (Logic Bus ‘B’), powered from 2A 250VDC RMOV Board (which powers Div II of RCIC Logic). Logic Bus ‘B’ will also cause RCIC to automatically isolate.

D INCORRECT: Incorrect but plausible in that in accordance with 2-AOI-100-2, Control Room Abandonment, when 2-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE transfer switch has been placed in EMERGENCY, RCIC will not trip on High Reactor Water Level.

RO Level Justification: Tests the candidate’s knowledge of the effect that a loss of electrical power to RCIC logic has on the RCIC System. This question is rated as Memory due to the requirement to strictly recall facts related to the power supplies to RCIC logic and the effect of a loss of power on trips, initiations, and isolations.

Technical Reference(s): 2-ARP-9-3C, Rev.28 (Attach if not previously provided)
2-AOI-57-11, Rev.16
2-AOI-100-2, Rev.60
2-45E626-1, Rev.17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.040, Obj. 7 (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.040-07 002 #1148	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

10 CFR Part 55 Content:

55.41	**X**
55.43	

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 06 27 2013

1148. OPL171.040-07 002

Unit 2 experienced a loss of 250 VDC RMOV BD 2B

Which ONE of the following statements describes the operation of the RCIC system?

A. RCIC will NOT automatically isolate.

B✔ RCIC will NOT automatically initiate.

C. The RCIC Flow Controller remains functional i.e. is not affected.

D. ONLY the manual isolation is functional.

Excerpt from 2-ARP-9-3C:

BFN Unit 2	Panel 9-3 2-XA-55-3C	2-ARP-9-3C Rev. 0028 Page 4 of 42

RCIC LOGIC POWER FAILURE	1

(Page 1 of 1)

Sensor/Trip Point:

Logic Bus A: Relay 13A-K1, 13A-K-24, or 13A-K40 deenergized

Logic Bus B: Relay 13A-K34 deenergized.

Sensor Location:	Logic Bus A Panel 25-31	Logic Bus B Panel 9-33
	Rx Bldg, El 621', R-13 Q-Line Aux Instr Rm, El 593'	

Probable Cause:
A. Cleared Fuse(s)
B. Loss of 250V DC power supply to panels.

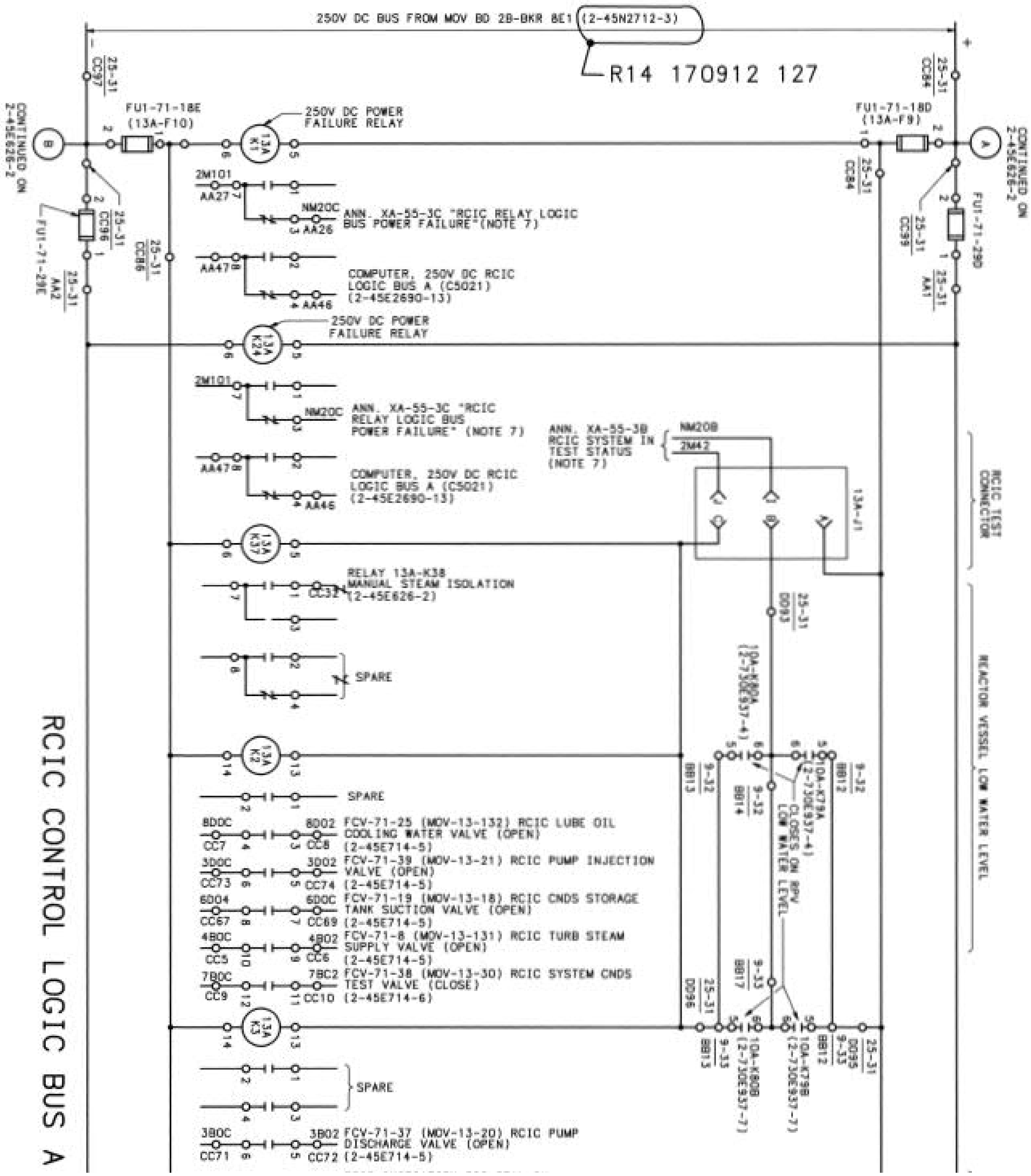
Automatic Action: None

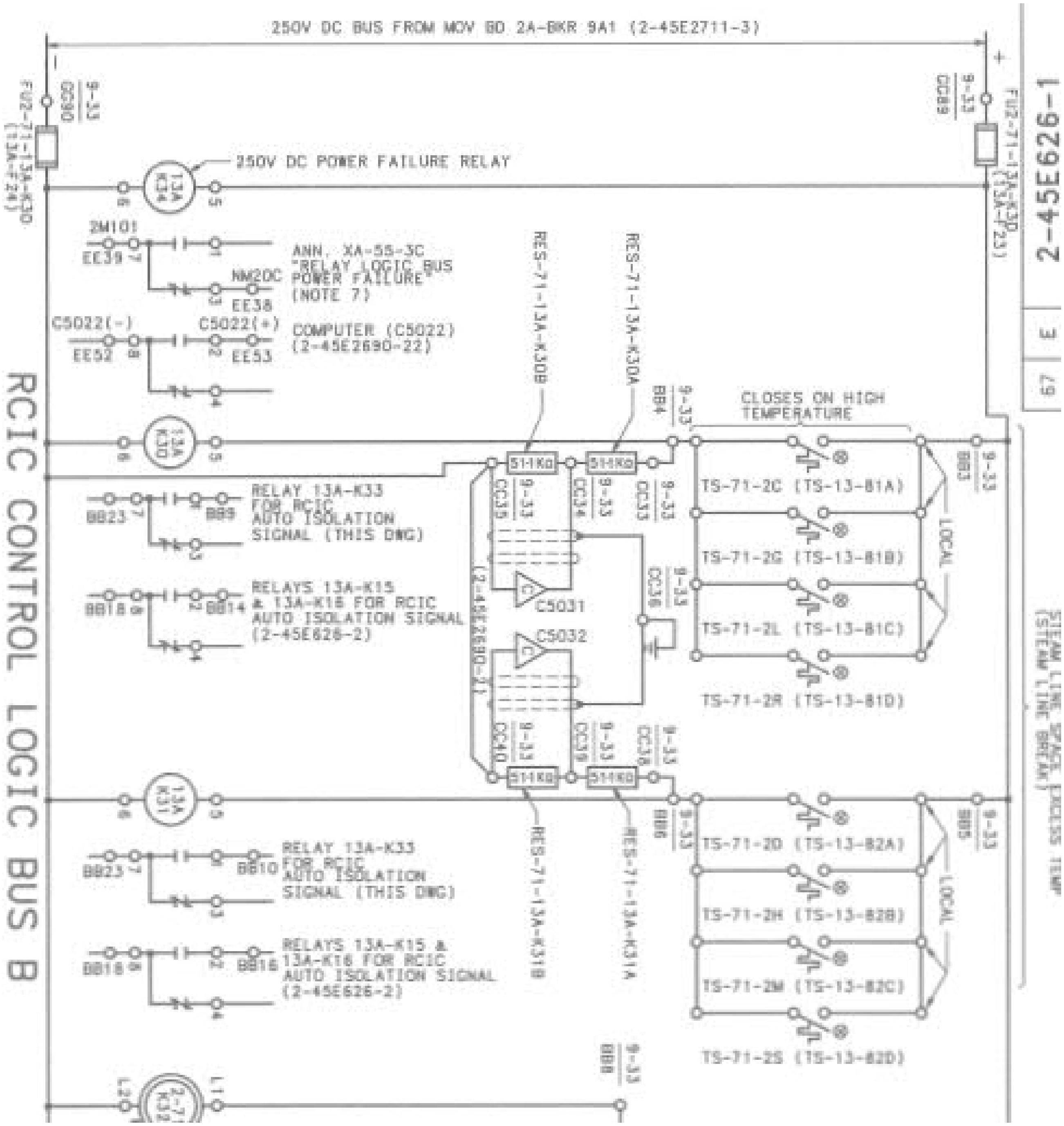
Operator Action:

A. **DETERMINE** which logic bus (A or B) has failed and **DISPATCH** personnel(s) to investigate the following:
   1. Logic Bus A
      a. Power supply 250V DC RMOV Bd 2B, Compt 8E1. Loss of EGM Control Box and items listed in 2, 3, and 4.
      b. Fuses 2-FU1-071-0018D (13A-F9) and 2-FU1-071-0018E (13A-F10) (10 amp) -Panel 2-25-31, fuse block CC. Loss of RCIC initiation, isolation, and trip logics.
      c. Fuses 2-FU1-071-0029D and 2-FU1-071-0029E (3 amp)- Panel 2-25-31, fuse block AA. Loss of isolation on Rupture Disc high pressure.
      d. Fuses 2-FU1-071-0013AA (13A-F28) and 2-FU1-071-13AB (13A-F29) (10 amp) - Panel 2-25-31, fuse block. Loss of trip function on Turbine Exhaust Pressure High and Pump Suction Low Pressure.
   2. Logic Bus B
      a. Power supply - 250V DC RMOV Bd 2A, Compt 9A1.
      b. Fuses 2-FU2-71-13A-K30 (13A-F23) and 2-FU2-71-13A-K30 (13A-F24) (10 amp) - Panel 2-9-33, fuse block GG. Loss of Logic Bus B isolation logic. RCIC continues to function.

B. **REFER TO** Tech Specs Sections 3.3.5.2, 3.3.6.1, and 3.5.3.

Excerpts from 2-45E626-1:



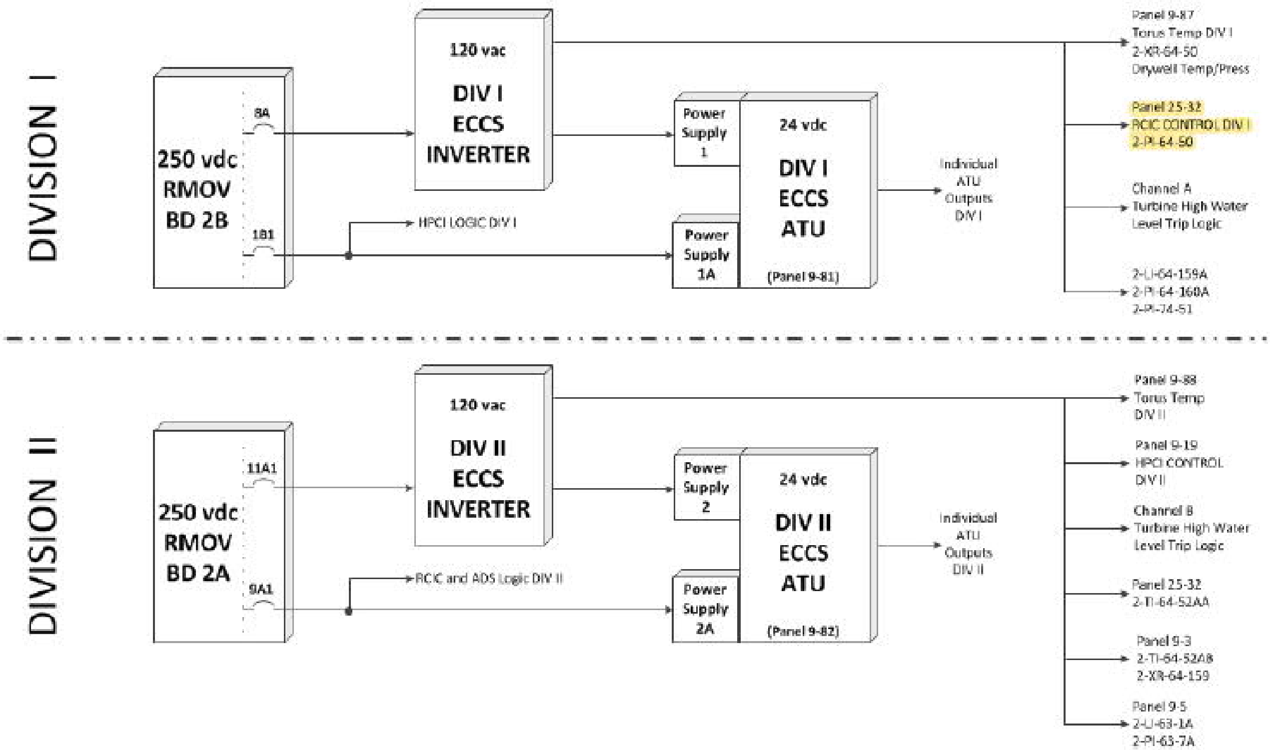

250V DC BUS FROM MOV BD 2A-BKR 9A1 (2-45E2711-3)
250V DC POWER FAILURE RELAY
ANN. XA-55-3C "RELAY LOGIC BUS POWER FAILURE" (NOTE 7)
COMPUTER (C5022) (2-45E2690-22)
RES-71-13A-K30B
RES-71-13A-K30A
CLOSES ON HIGH TEMPERATURE
RELAY 13A-K33 FOR RCIC AUTO ISOLATION SIGNAL (THIS DWG)
RELAYS 13A-K15 & 13A-K16 FOR RCIC AUTO ISOLATION SIGNAL (2-45E626-2)
TS-71-2C (TS-13-81A)
TS-71-2G (TS-13-81B)
TS-71-2L (TS-13-81C)
TS-71-2R (TS-13-81D)
C5031
C5032
RELAY 13A-K33 FOR RCIC AUTO ISOLATION SIGNAL (THIS DWG)
RELAYS 13A-K15 & 13A-K16 FOR RCIC AUTO ISOLATION SIGNAL (2-45E626-2)
RES-71-13A-K31B
RES-71-13A-K31A
TS-71-2D (TS-13-82A)
TS-71-2H (TS-13-82B)
TS-71-2M (TS-13-82C)
TS-71-2S (TS-13-82D)
LOCAL
RCIC CONTROL LOGIC BUS B
STEAM LINE SPACE EXCESS TEMP (STEAM LINE BREAK)
2-45E626-1

Excerpt from 2-AOI-57-11:

BFN Unit 2	Loss of Power to an ECCS ATU Panel/ECCS Inverter	2-AOI-57-11 Rev. 0016 Page 31 of 31

Attachment 5
(Page 1 of 1)

Redundant Power Supplies



Excerpt from 2-AOI-100-2:

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0060 Page 11 of 95

4.2 Unit 2 Subsequent Actions (continued)

NOTES

1) Attachment 1 provides normal backup control stations and available communications.
2) Attachment 10 provides PAX extensions and locations.

[7] **ESTABLISH** communication with the following personnel and **DIRECT** attachments be completed as follows:

- U-2 Unit Operator complete Attachment 2, Part A. ☐
- U-2 Rx Bldg AUO complete Attachment 3, Part A. ☐
- U-2 Turb Bldg AUO complete Attachment 4, Part A. ☐

[8] Upon completion of attachments, **RE-ESTABLISH** communication using the best available means and continue procedure. ☐

CAUTION

1) RCIC TURBINE STEAM SUPPLY VALVE, 2-FCV-71-8, transfer switch has been placed in EMERGENCY and will **NOT** trip on Reactor Water Level High (+51 inches). Failure to maintain level below this value may result in equipment damage.
2) RCIC will still trip on low suction pressure, high turbine exhaust pressure, mechanical overspeed, and trip push button on pnl 25-32.

Examination Outline Cross-reference:	Level	RO	SRO
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff	Tier #	2	------
G2.1.32 (10CFR 55.43.2)	Group #	1	------
• Ability to explain and apply system limits and precautions.	K/A #	223002G2.1.32	
	Importance Rating	3.8	------

Proposed Question: **# 58**

In accordance with 1-OI-1, Main Steam System, which **ONE** of the following completes the statement below?

Main Steam Tunnel Temperature should **NOT** be allowed to exceed a **MAXIMUM** of

___(1)___, **___(2)___**.

A. (1) 170 °F
   (2) to prevent MSIV isolation

B. (1) 170 °F
   (2) due to environmental qualification requirements

C. (1) 189 °F
   (2) to prevent MSIV isolation

D. (1) 189 °F
   (2) due to environmental qualification requirements

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 170 °F is the temperature at which MAIN STEAM LINE LEAK DETECTION TEMP HIGH (1-9-3D, Window 24) alarms from the sensor 1-TE-1-60A, Main Steam Tunnel. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that BFN FSAR 4.6, Main Steam Isolation Valves discusses the environmental qualification requirements from 10CFR50.49 as it relates to MSIVs.

C **CORRECT:** *(See attached)* In accordance with 1-OI-1, Main Steam System, Precaution and Limitations 3.2.2. MSIV Isolation, Main Steam Tunnel Temperature should not be allowed to exceed 189 °F. For second part, this is to prevent MSIV isolation which is a PCIS Group 1 Isolation.

D INCORRECT: First part is correct *(See C).* Second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's ability to explain and apply Main Steam System Precautions and Limitations as it applies to Primary Containment Isolations. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): 1-OI-1, Rev. 16 (Attach if not previously provided)

1-ARP-9-3D, Rev. 30

BFN FSAR 4.6, Rev. 28

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.009_Obj. 14c_ (As available)

Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Excerpts from 1-OI-1:

3.2 Main Steam Isolation Valves (MSIV)

3.2.1 MSIV Closure

A. The MSIVs should be fast closed when the reactor is shutdown and no steam flow, unless required to be slow closed by surveillance, test instruction, or an abnormal condition. [BFNPER 164499]

B. When an MSIV is closed at power, the potential exists for an isolation of the Hydrogen Water Chemistry System to occur. This is due to the possibility of a hydrogen bubble becoming entrained in the main steam line drains and subsequently being released when the main steam line drains reposition in response to a MSIV closure. This scenario can result in a small Off Gas System hydrogen spike of sufficient strength to cause an automatic isolation of the Hydrogen Water Chemistry System.

C. Closure of all MSIVs could cause turbine shaft damage if main condenser vacuum is maintained and seal steam supply is not established from the auxiliary boiler.

3.2.2 MSIV Isolation

A. Main steam tunnel temperature should not be allowed to exceed 189°F to prevent MSIV isolation.

B. Whenever reactor pressure is reduced to 852 psig and the reactor mode switch is in RUN position, the MSIVs will close.

3.2.2 MSIV Isolation (continued)

F. The PCIS group one trip parameters do not exceed trip setpoints.

1. Reactor water level above -122 in.
2. MSL flow less than 135%.
3. MSL tunnel temperature less than 189°F.
4. MSL pressure greater than 852 psig if in Mode 1.

Excerpts from 1-ARP-9-3D: also supports Distractors A(1), B(1)

BFN Unit 1	Panel 9-3 XA-55-3D	1-ARP-9-3D Rev. 0030 Page 30 of 44

MAIN STEAM LINE LEAK DETECTION TEMP HIGH 1-TA-1-60

24

(Page 1 of 2)

Sensor/Trip Point:

1-TE-1-60A (1-TIS-1-60A)	170°F
1-TE-1-60B through D (1-TR-69-29)	160°F

Sensor Location:

1-TE-1-60A	Main Steam Tunnel
1-TE-1-60B	Main Steam Line Steam Vault Area Elev 565
1-TE-1-60C	Main Steam Line Bypass Vlv Area Elev 586
1-TE-1-60D	Main Steam Line Control Vlv Area Elev 617

Probable Cause:

A. Main Steam, RWCU, Feedwater, RCIC, or HPCI discharge (only with HPCI in service and elevated Suppression Pool water Temp.) line break.
B. TB or RB Coolers out of service
C. Sensor malfunction
D. Steam Vault Exhaust Booster Fan out of service.

Automatic Action: Impending MSIV Isolation at 189°F area temp.

BFN Unit 1	Panel 9-3 XA-55-3D	1-ARP-9-3D Rev. 0030 Page 31 of 44

MAIN STEAM LINE LEAK DETECTION TEMP HIGH 1-TA-1-60, Window 24
(Page 2 of 2)

Operator Action: (Continued)

C. **IF** RCIC is **NOT** in service and 1-FI-71-1A(B), RCIC STEAM FLOW indicates flow, **THEN**
ISOLATE RCIC and check temperatures lowering. ☐
D. **CHECK** for elevated RAD levels on the following instruments:
- 1-RM-90-20, CRD-HCU West AREA EL. 565 RX BLDG. ☐
- 1-RM-90-29, SUPPR POOL AREA EL. 519 RX BLDG. ☐

E. **IF** HPCI is injecting with elevated Suppression Pool temperature, **THEN**
CONSIDER securing HPCI to determine if HPCI is the source of the leak. ☐
F. **IF** Rx Building Main Steam Tunnel temperature is above 170°F on 1-TIS-1-60A on Panel 1-9-3, **THEN**
PERFORM the following:
1. **ENTER** 1-EOI-3 Flowchart. ☐
2. **ENSURE** Rx Zone fans, 1-HS-64-11A at Panel 1-9-25, in Fast Speed. ☐
3. **ENSURE** Steam Vault Exhaust Booster Fan in service. ☐
4. **REFER TO** 1-OI-30B. ☐

Excerpt from BFN FSAR 4.6: Supports Distractors A(2), C(2)

is indicated, each valve may be checked individually by opening the other valve in the same steam line with all other MSIVs closed, evacuating and closing the steam chest, and checking for pressure rise.

Redundancy is provided by two MSIVs in each steam line so that either can perform the isolation function, and either can be tested for leakage after closing the other. The inside valve, the outside valve and their control systems are physically separated. Considering the redundancy, the mechanical strength, the closing forces, and the leakage tests discussed above, the main steam isolation valves satisfy safety design bases "c", "d", and "e" to limit the release of reactor coolant or radioactive materials, within the margins evaluated in Section 14.0, "Plant Safety Analysis."

The MSIVs and their installation are designed as seismic Class I equipment for inclusion of seismic loadings, as delineated in Appendix C.

The design of the MSIVs for seismic loadings is discussed in paragraph 4.6.3 above. These loads are small compared with the pressure and operating loads the valve components are designed to withstand. The cantilevered support of the air cylinder, hydraulic cylinder, springs, and controls is the key area. The increase in loading at the joints between the support shafts and the valve bonnet caused by the specified earthquake loading is negligible. Therefore, the seismic loading requirement of design basis "f" is met.

Electrical equipment, associated with the MSIVs, that operates in an accident environment is limited to the wiring, solenoid valves, and position switches on the MSIVs. The design and purchase specifications for the wiring, solenoid valves, and position switches for accident environmental conditions are contained in the BFN 10 CFR 50.49 program. Under the accident conditions, ambient pressure and temperature increase to approximately 50 psig and 337°F; each valve is required to close within a 2 minute exposure to these conditions. The valve closing is completed during this two minute time frame.

Operation of the valves in the normal operating conditions and postulated accident environments is ensured by the requirements of the purchase specifications, review and approval of equipment design and vendor drawings, extensive control of materials, fabrication procedures, fabrication tests, nondestructive examinations, Excerpt from 10CF50.49: Supports Distractors A(2), C(2)

§ 50.49 Environmental qualification of electric equipment important to safety for nuclear power plants.

(a) Each holder of or an applicant for an operating license issued under this part, or a combined license or manufacturing license issued under part 52 of this chapter, other than a nuclear power plant for which the certifications required under § 50.82(a)(1) or § 52.110(a)(1) of this chapter have been submitted, shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section. For a manufacturing license, only electric equipment defined in paragraph (b) which is within the scope of the manufactured reactor must be included in the program.

(b) Electric equipment important to safety covered by this section is:

(1) Safety-related electric equipment.[3]

(i) This equipment is that relied upon to remain functional during and following design basis events to ensure—

(e) The electric equipment qualification program must include and be based on the following:

(1) *Temperature and pressure*. The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.

(2) *Humidity*. Humidity during design basis accidents must be considered.

Examination Outline Cross-reference:	Level	RO	SRO
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)	Tier #	2	------
A1.02 (10CFR 55.41.5)	Group #	1	------
Ability to predict and/or monitor changes in parameters associated with operating the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) controls including:	K/A #	262002A1.02	
• Motor generator outputs	Importance Rating	2.5	------

Proposed Question: **# 59**

All three Units are operating at 100% RTP with the following conditions:

- 240V Lighting Board 2A is tagged out of service for scheduled work
- Electrical fault causes 240V Lighting Board 3B to de-energize

Given the conditions above, which **ONE** of the following completes the statement below?

The Plant Preferred Motor Generator will start ___**(1)**___ and energize ___**(2)**___.

A. (1) immediately
(2) **ONLY** Battery Board 2 Panel 14

B. (1) immediately
(2) Panel 9-9 Cabinet 4 on **ALL** three Units

C. (1) after a 6-second time delay
(2) **ONLY** Battery Board 2 Panel 14

D. (1) after a 6-second time delay
(2) Panel 9-9 Cabinet 4 on **ALL** three Units

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the BFN 120V AC Distribution System is complex and often confused especially as it relates to all 3 Unit's associated breakers, panels, boards, power supplies and transfer schemes. In accordance with 0-OI-57C, 208V/120V AC Electrical System, if Unit Preferred MMG set trips, the appropriate Unit's Breaker 1002 should be closed immediately to supply bus power. Second part is incorrect but plausible in that 0-AOI-57-3, Loss of Plant Preferred, Attachment 1 illustrates that Battery Board 2 Panel 14 is normally (alternately) powered from 240V Lighting Board 2A(3B), however given the conditions, the Plant Preferred Auto Transfer Switch has realigned to the Plant Preferred MG. Battery Board 2 Panel 14 is now left de-energized without manual Operator action.

B INCORRECT: First part is incorrect but plausible *(See A)*. Second part is correct *(See D)*

C INCORRECT: First part is correct *(See D)*. Second part is incorrect but plausible *(See A).*

D **CORRECT**: (*See attached*) In accordance with 0-AOI-57-3, the Plant Preferred Motor Generator (MG) starts when Plant Preferred bus voltage drops to 95% for 6 seconds. Second part, the Plant Preferred Auto Transfer Switch transfers from the Lighting Board to the Plant Preferred MG when Plant Preferred bus voltage drops to 95% for 6 seconds AND MG set output is at 90% rated voltage and frequency. This will energize Battery Board 2 Panel 13 thereby energizing Panel 9-9 Cabinet 4 on ALL 3 Units.

RO Level Justification: Tests the candidate's ability to predict and monitor changes in 120V AC Electrical System parameters as it relates to motor generator outputs. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and it's meaning to predict the correct outcome related to the complex 120V AC Electrical System.

Technical Reference(s): 0-OI-57C, Rev. 132 (Attach if not previously provided)
0-AOI-57-3, Rev. 57
OPL171.102, Rev. 10

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.102 Obj. 3b_ (As available)

Question Source:	Bank #	BFN 1510 #46	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2015	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTION 46 Rev 1

All three Units are operating at 100% power.

- 240V Lighting Board 2A is tagged out of service for scheduled work.

An electrical fault causes 240 V Lighting Board 3B to deenergize.

Which one of the following completes the statements below?

The Plant Preferred MG will start __ (1) __ and energize __ (2) __.

A. (1) immediately
(2) Panel 9-9 cabinet 4 on all 3 units

B. (1) immediately
(2) Battery Board 2 Panel 14

C. (1) after a 6 second time delay
(2) Panel 9-9 cabinet 4 on all 3 units

D. (1) after a 6 second time delay
(2) Battery Board 2 Panel 14

Answer: C

Excerpts from 0-AOI-57-3:

BFN Unit 0	Loss of Plant Preferred	0-AOI-57-3 Rev. 0057 Page 6 of 30

3.0 AUTOMATIC ACTIONS

NOTES
1) Battery Board 2 Panel 13 is normally supplied power from the Nonpreferred Transfer Switch (240V Lighting Board 2A or 3B). Upon loss of transfer switch power, the Plant Preferred MG should start and load to energize the Plant Preferred system.
2) The Plant Preferred MG starts when Plant Preferred bus voltage drops to 95% for 6 seconds. The Plant Preferred Auto Transfer Switch transfers from the lighting board to the Plant Preferred MG when Plant Preferred bus voltage drops to 95% for 6 seconds and MG set output is at 90% rated voltage and frequency. When the lighting board voltage reaches 97% rated for one minute (adjustable up to 30 minutes) the Plant Preferred Auto Transfer Switch transfers back to the lighting board and MG set is automatically disconnected.

A. RSW Storage Tank Isolation Valve, 0-FCV-25-32, will close.

B. CO_2 Master Solenoids 1 and 2, 0-FCV-39-11 and 4 open.

C. D/G rooms A, B, C, and D; 0-FCV-39-7, 8, 9, and 10 close.

D. Electrical Board rooms A and B, 0-FCV-39-5 and 6 close.

E. If plant Preferred is the only power supply to Panel 1 (2)(3)-9-53, then 1(2)(3)-FCV-66-28,Off Gas Isolation Valve, closes.(The normal power supply for Panel1(2)(3)-9-53 is Panel 1(2)(3)-9-9 cabinet 5, breaker 522. Plant Preferred serves as an alternate power supply for Panel 1(2)(3)-9-53).

F. Auxiliary Boiler A will trip, if running.

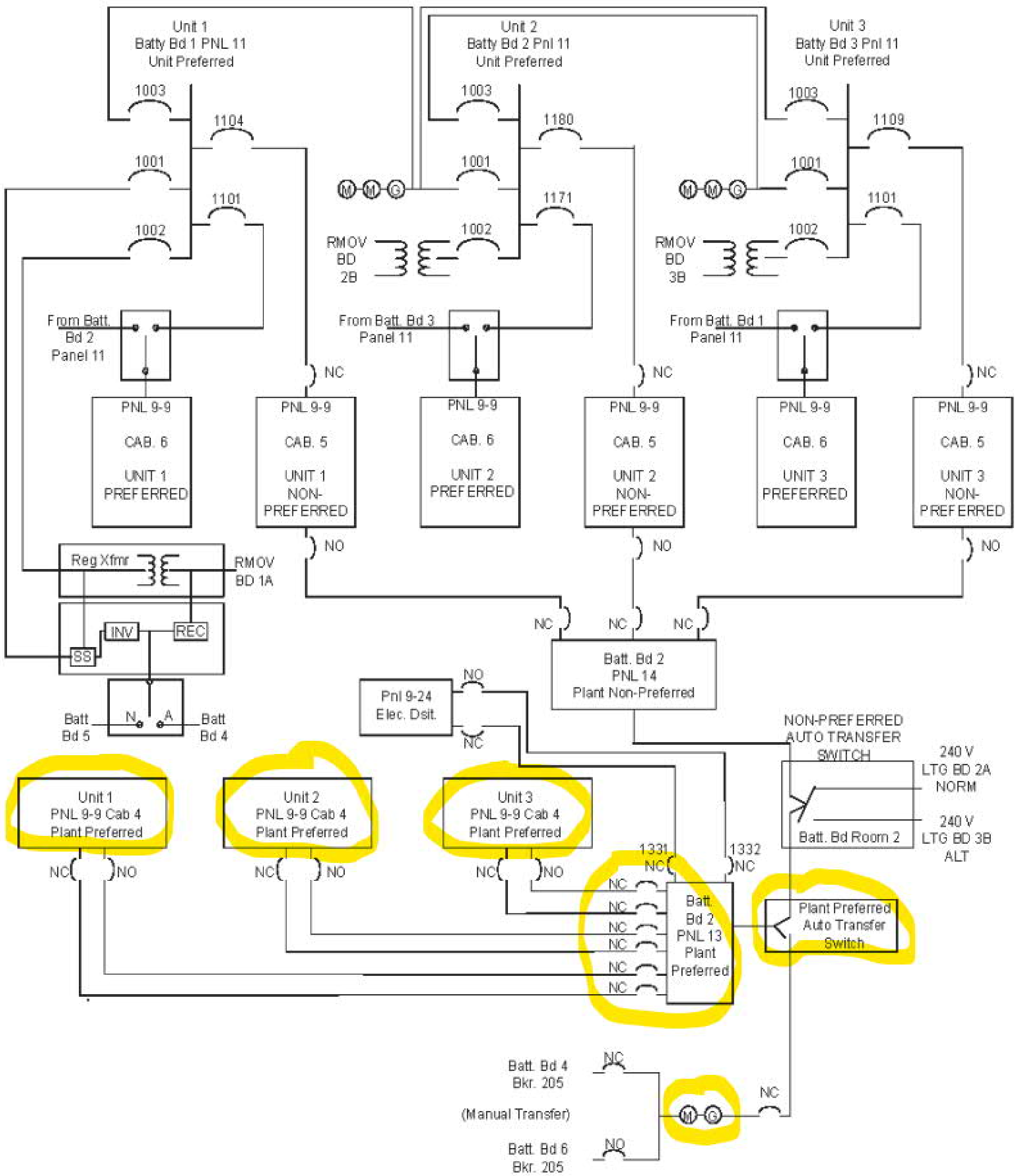
G. Auxiliary Boiler Steam Dump Line Control Valve, 0-FCV-012-0078, will close if being controlled by Auxiliary Boiler A.

Illustrates the normal and alternate power supplies for Plant Preferred loads via the Auto Transfer Switch

BFN Unit 0	Loss of Plant Preferred	0-AOI-57-3 Rev. 0057 Page 16 of 30

Attachment 1
(Page 1 of 1)

Vital 120V AC Distribution



Excerpt from OPL171.102 Lesson Plan:

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
(d) If the Main Turbine is not at rated speed, power to EHC could be lost requiring MSRV for pressure control. (e) Refer to AOI-57-4 for other effects.	until another RWM rod scan is called for. Note: Unit Non-Preferred is also lost. This requires manual xfr. Panel 9-53 auto xfrs to 9-9, cab 4
3. Plant Preferred power system	
a) Purpose and design	IL-6
(1) Plant preferred supplies power to loads which can withstand short power interruptions, but may be needed for plant shutdown after a loss of AC power.	NLO/NLOR Obj. 3.b.
(2) The system is normally powered from the Non-Preferred system, with a DC powered motor-generator set as a backup. The motor-generator (MG) set auto starts on low system voltage.	NLO/NLOR Obj. 3.a LOR Obj. 1 95% for 6 seconds
(3) When MG set output voltage is near normal 90% voltage and frequency the auto transfer switch will shift to allow the MG set to supply system loads. When power to the Non-Preferred system is restored, the transfer switch will shift back to the Non-Preferred system after a minute time delay. The MG must be manually shutdown. Normally, the MG set is powered from battery board 4. It can be manually transferred to battery board 6.	ILT Obj. 3.b NLO/NLOR Obj. 3.a, 3.c, 3.d 97% for 1minute
b) Distribution	
(1) Battery Board 2, panel 13 contains the distribution breakers for the Plant Preferred system.	
(2) From Battery Board 2 power is sent to each unit's panel 9-9, cabinet 4 and to unit 1 panel 9-24. Each of these panels has a normal and an alternate supply breaker from Battery Board 2. The normal and alternate supply breakers are manually transferred.	ILT Obj. 3.a NLO/NLOR Obj. 3.a NLO Obj. 3.a
c) The alarm "Panel 9-9 PFD or Non-PFD BKR Tripout" could indicate a trip of a load or supply breaker on panel 9-9, cabinet 4. This would indicate a loss of Plant Preferred to that unit. Other indications of a loss of Plant Preferred would be: (1) RSW Head Tank Isolation Valve FCV 25-32	ILT Obj. 3.b, 3.c. LOR Obj. 4 NLO Obj. 3.f

Excerpt from 0-OI-57C: supports Distractors A(1), B(1)

8.1 Placing the Unit Preferred Transformer in Service with the Normal MMG Set Supplying Power to the Bus (continued)

CAUTION

If the Unit Preferred MMG set supplying power to the bus Trips, the appropriate Unit 1(2,3) Breaker 1002 should be closed immediately to supply power to the bus.

[5] **DEPRESS** and **HOLD** for at least 10 seconds UNIT 2(3) MMG AC MOTOR STOP/DC TRANSFER, 2(3)-HS-252-02AF and **OBSERVE** the following:

A. AC Motor stops, UNIT 2(3) MMG AC MOTOR OFF, 2(3)-IL-252-02AJ illuminated.

B. Amps remain the same as indicated on UNIT 2(3) MMG GENERATOR amps, 2(3)-II-252-02C/1AMR.

NOTE

Unit Preferred Transformer Volts - Hertz are read as Incoming.

[6] **PLACE** UNIT 2(3) PFD SYSTEM TRANSFORMER SOURCE SYNC SS-5(8), 2(3)-HS-252-02/SS-5)(03/SS-8) to ON.

[7] **MATCH** Running to Incoming voltage using UNIT 2(3) MMG VOLTAGE ADJUST, 2(3)-HS-252-02CD.

[8] **ADJUST** MMG speed using UNIT 2(3) MMG FREQUENCY **ADJUST**, 2(3)-HS-252-02CF to obtain a synchroscope slowly rotating in the FAST (clockwise) direction as indicated on SYNCHROSCOPE, 0-SCP-252-000A/B.

Examination Outline Cross-reference:

300000 (SF8 IA) Instrument Air

A2.01 (10CFR 55.41.5)

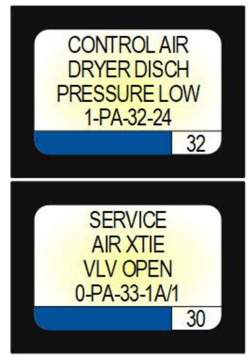
Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

- Air dryer and filter malfunctions

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	300000A2.01	
Importance Rating	2.9	------

Proposed Question: **# 60**

Unit 1 is operating at 100% RTP when a Control Air leak develops, resulting in the following conditions:



- CONTROL AIR DRYER DISCHARGE PRESSURE LOW (1-9-20, Window 32) alarms
- SERVICE AIR CROSSTIE VALVE OPEN (1-9-20, Window 30) alarms
- Control Air Pressure is currently 69 psig and slowly lowering

Given the conditions above, which **ONE** of the following completes the statement below?

0-FCV-33-1, SERVICE AIR CROSSTIE VALVE opened at ___**(1)**___ Control Air Pressure and a manual Reactor SCRAM ___**(2)**___ **CURRENTLY** required in accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors.

A. (1) 70 psig
(2) is

B. (1) 70 psig
(2) is NOT

C. (1) 85 psig
(2) is

D. (1) 85 psig
(2) is NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 70 psig Control Air Pressure is when the provided Window 32 alarms. The second part is incorrect but plausible if the candidate confuses the Control Air Pressure and alarms associated with requiring a manual Reactor SCRAM. If SCRAM PILOT AIR HEADER PRESS LOW (1-9-5B, Window 28) was in alarm at 66 psig (not provided), a manual Reactor SCRAM would be required.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is correct *(See D).*

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT:** *(See attached)* In accordance with the given Alarm Response Procedure for 1-9-20B, Window 32, which alarms at 70 psig Control Air Pressure and lowering, it lists a Control Air dryer malfunction as a probable cause. It also states 1-9-20B, Window 30 (also given) is in alarm and references 0-AOI-32-1, Loss of Control and Service Air Compressors. 0-AOI-32-1 states that 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE opens at Control Air Pressure less than or equal to 85 psig. For second part, in accordance with 0-AOI-32-1, if Control Air Pressure continues to lower below 55 psig, then manually SCRAM the Reactor. However, SCRAM PILOT AIR HEADER PRESS LOW (1-9-5B, Window 28) is NOT in alarm at 66 psig (not provided). Therefore, a manual Reactor SCRAM is NOT currently required.

RO Level Justification: Tests the candidate's ability to predict the impact of component malfunctions as it relates to the Control Air System and the use of Alarm Response Procedures and Abnormal Operating Instructions to mitigate the conditions. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-OI-32, Rev. 141 (Attach if not previously provided)
1-ARP-9-20B, Rev. 38
1-ARP-9-5B, Rev. 22
0-AOI-32-1, Rev. 56

Proposed references to be provided to applicants during examination: **CONTROL AIR DRYER DISCH PRESSURE LOW (1-9-20B, Window 32), SERVICE AIR XTIE VLV OPEN (1-9-20B, Window 30)**

Learning Objective: OPL171.054 Obj. 8 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1703 #14	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2017	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Copy of Bank Question:

QUESTION 14 Rev 4

All three units are operating at 100% power when the “G” Air compressor trips.

AIR COMPRESSOR ABNORMAL, (1-9-20B window 29) has alarmed.

Conditions are as follows:

- ‘A’ and ‘B’ compressors are running
- ‘C’ and ‘D’ compressors failed to start.
- 1-PI-32-20, Control Air header pressure, is lowering

Which one of the following completes both statements in accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors?

0-FCV-33-1, Service Air crosstie to Control Air valve, automatically opens when control air header pressure first lowers to __ (1) __ psig.

Reactor SCRAM on Unit 1 is required if Control and Service Air Compressors cannot maintain Control Air Header pressure above __ (2) __ psig.

A. (1) 70
   (2) 55

B. (1) 70
   (2) 66

C. (1) 85
   (2) 55

D. (1) 85
   (2) 66

Answer: C

Excerpts from 1-ARP-9-20B:

BFN Unit 1	Panel 9-20 1-XA-55-20B	1-ARP-9-20B Rev. 0038 Page 36 of 39

CONTROL AIR DRYER DISCH PRESSURE LOW 1-PA-32-24	
	32

(Page 1 of 1)

Sensor/Trip Point:

Relay 20B	1-PS-032-0024	70 psig lowering

Sensor Location:
Relay 20B
Panel 1-9-20
Main Control Rm.

1-PS-032-0024
Panel 1-LPNL-925-0204
Elevation 565
Col. T-2 H-LINE

Probable Cause:
A. Leak(s).
B. Valve malfunction or incorrect alignment.
C. Control air dryer malfunction.

Automatic Action: None

Operator Action:
A. **CHECK** for indications of control air pressure lowering.
- On Panel 1-9-20, CONTROL AIR PRESSURE, 1-PI-32-20. ☐
- ICS Display (AIR COMPRESSOR G). ☐
- Annunciator SERVICE AIR XTIE VLV OPEN, (1-XA-55-20B, Window 30), in alarm. ☐
- Annunciator AIR COMPRESSOR ABNORMAL, (1-XA-55-20B, Window 29), in alarm. ☐
- Annunciator SCRAM PILOT AIR PRESSURE LOW, (1-XA-55-5B, Window 28), in alarm. ☐

B. **DISPATCH** personnel to investigate. (**COORDINATE** with other units). ☐
C. **NOTIFY** Unit Supervisor, U2 and U3. ☐
D. **REFER TO** 0-AOI-32-1 and 1-AOI-32-2. ☐

References: 1-45E620-12-2 0-45E769-5 1-47E610-32-1

BFN Unit 1	Panel 9-20 1-XA-55-20B	1-ARP-9-20B Rev. 0038 Page 34 of 39

SERVICE AIR XTIE VLV OPEN 0-PA-33-1A/1	
	30

(Page 1 of 1)

Sensor/Trip Point:

Relay 20C 0-PS-33-1 85 psig

Sensor Location: Relay 20C Panel 9-20 0-PS-33-1 Elevation 565 Col. T-2 N-LINE

Probable Cause:

A. Air compressor malfunction.
B. Excessive air usage.
C. System leakage.
D. Sensor malfunction.

Automatic Action: FCV-33-1 opens to supply Control Air from Service Air.

Operator Action:

A. **CHECK OPEN** FCV-33-1, using 0-HS-33-1A/1, on Panel 1-9-20, or **OPEN** CONTROL AIR BYPASS, 0-33-501 and **MONITOR** CONTROL AIR PRESSURE, 1-PI-32-20. (**COORDINATE** with Unit 3.) ☐
B. **DISPATCH** personnel to Control Air Compressor G Control Panel 0-LPNL-925-0692 and Control Panel 0-LPNL-925-0118 to investigate local alarms and indications. ☐
C. **NOTIFY** Unit Supervisor. ☐
D. **IF** unable to maintain air pressure, **THEN PERFORM** the following:
   - **CHECK** running or **START** all available Control and Service Air Compressors locally. (**COORDINATE** with Unit 3.) ☐
   - **ENSURE** Control Air Compressors are loading as required. (G Compressor can be monitored on ICS). ☐

E. **IF** Control Air Compressors are **NOT** loading as required, **THEN HAVE** personnel perform Alternate Method for Manually Loading Control Air Compressors. **REFER TO** 0-OI-32. ☐
F. **INITIATE** a search to find and isolate air leaks. (**COORDINATE** with other units). ☐
G. **REFER TO** 1-AOI-32-2. (**COORDINATE** with other units). ☐

References: 0-45E769-5 1-45E620-12-2 0-45E781-3

Excerpts from 0-AOI-32-1:

BFN Unit 0	Loss of Control and Service Air Compressors	0-AOI-32-1 Rev. 0056 Page 4 of 35

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for loss of Control and Service Air Compressors.

2.0 SYMPTOMS

- AIR COMPRESSOR ABNORMAL, (1-XA-55-20B, Window 29) is IN ALARM.
- CONTROL AIR COMP G BKR DE-ENERGIZED, (1/2-XA-55-23B, Window 38), will alarm.
- CONT AIR COMP G MTR AMPS, 0-EI-32-2901, on Panel 1-9-20, indicates approximately zero amps.
- Control Air Compressor G breaker trips.
- Air Compressor G ICS Display shows Compressor G in unloaded or shutdown condition.
- Air Compressor G ICS Display shows lowering Control Air System pressure.
- SERVICE AIR XTIE VLV OPEN (FCV-33-1 OPEN), (1(3)-XA-55-20B, Window 30) is IN ALARM.
- CONTROL AIR PRESS LOW (2(3)-XA-55-20B, Window 32) is IN ALARM.
- CONTROL AIR DRYER DISCH PRESSURE LOW (1-XA-55-20B, Window 32) is IN ALARM.
- SCRAM PILOT AIR HEADER PRESS LOW (1(2)(3)-XA-55-5B, Window 28) is IN ALARM.
- Outboard MSIV's close or start to close.
- Air Compressor E ICS Display shows Compressor E in unloaded or shutdown condition.
- Air Compressor F ICS Display shows Compressor F in unloaded or shutdown condition.
- Main Steam Line Drain valves 1(2)(3)-FCV-1-58, 185, 168, 169, 170, and 171 fail open.

BFN Unit 0	Loss of Control and Service Air Compressors	0-AOI-32-1 Rev. 0056 Page 5 of 35

3.0 AUTOMATIC ACTIONS

- Service Air crosstie to Control Air valve, 0-FCV-33-1, opens at control air header pressure less than or equal to 85 psig.
- Control Air Compressors A,B,C,D will start as their on-line pressure setpoints are reached.
- The Emergency Control Bay Air Compressor will start at Control Air Header pressure less than or equal to 73 psig.
- Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901, will close when Control Air Header pressure reaches equal to or less than to 65 psig at the valve.
- Unit 1 to Unit 2 Control Air Crosstie, 1-PCV-032-3901, will close when Control Air Header pressure reaches equal to or less than to 65 psig at the valve.

BFN Unit 0	Loss of Control and Service Air Compressors	0-AOI-32-1 Rev. 0056 Page 6 of 35

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

NOTE
If necessary, place keeping marks may be made directly in the Control Room copy of this instruction. Management Services should be contacted for a replacement copy when time permits.

4.2 Subsequent Actions

[1] **CHECK** automatic actions. ☐

[2] **PERFORM** automatic actions that failed to occur. (Otherwise N/A) ☐

[3] IF ANY EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s) (otherwise N/A). ☐

[4] **IF** CONTROL AIR PRESSURE is continuing to lower as indicated by 1-PI-32-20 on Panel 1-9-20 or 2(3)-PI-32-88 on Panel 2(3)-9-20, **AND CONTROL AIR PRESSURE** lowers below 55 psig, **THEN** (Otherwise N/A)

MANUALLY SCRAM the reactor. Refer to 1(2)(3)-AOI-100-1 and 1(2)(3)-AOI-32-2. ☐

Excerpt from 1-ARP-9-5B: Supports Distractors A(2), C(2)

BFN Unit 1	Panel 9-5 1-XA-55-5B	1-ARP-9-5B Rev. 0022 Page 31 of 42

SCRAM PILOT
AIR HEADER
PRESS LOW
1-PA-85-38B
28

(Page 1 of 2)

Sensor/Trip Point:

1-PS-085-0038 56.0 psig

Sensor Location: 1-LPNL-925-0018
Elev. 565
Rx. Bldg.
Column R5, N line

Probable Cause:
A. SI (or SR) in progress.
B. Failure of scram pilot header pressure regulators 1-PCV-085-0066 or 1-PCV-085-0067.
C. Control air system failure.
D. Sensor malfunction.

Automatic Action: None

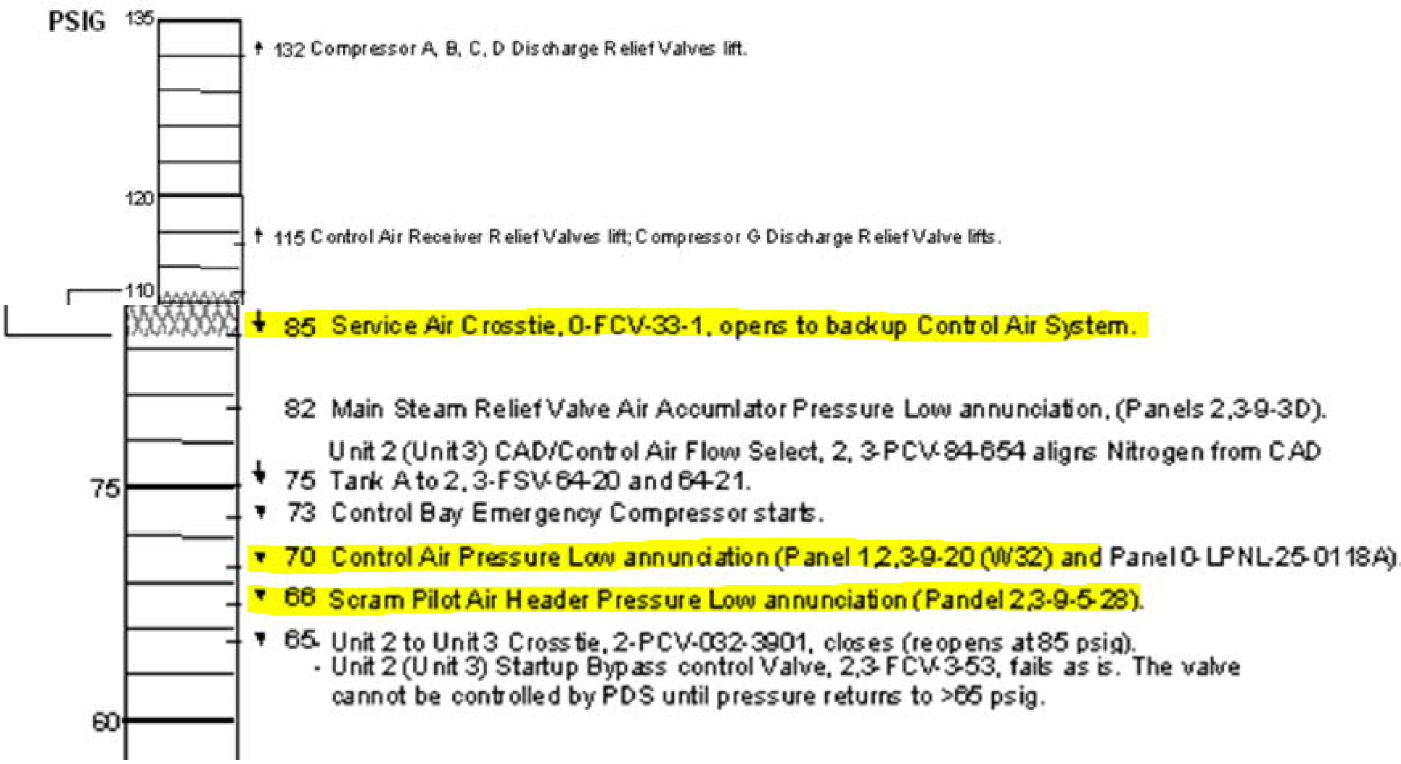
Operator Action:
A. **CHECK** 1-PI-32-20 on Panel 1-9-20 for control air pressure. ☐
B. **IF** low, **THEN**
REFER TO 0-AOI-32-1 ☐
C. On Panel 1-9-20, **CHECK OPEN** 1-FCV-32-91. ☐
D. **DISPATCH** personnel to check local pressure indicator, CRD SCRAM VALVE PILOT AIR HDR PRESS, 1-PI-085-0038 on 1-LPNL-925-0018, elevation 565', Rx building. ☐
E. Behind 1-LPNL-925-0018, RX. BLDG. EL. 565', **CHECK** CRD CA FILTER INLET, 1-PI-085-0066A (-0067A) and CRD CA FILTER OUTLET, 1-PI-085-0066B (-0067B). ☐
F. **IF** DP across CRD CA FILTER to 1-PCV-085-0067 is high, **THEN** **PERFORM** the following:
1. **CHECK OPEN** 1-SHV-085-0244, HDR X-TIE TO 1-FSV-085-0035A&B. ☐
2. **CLOSE** the following valves:
   - 1-SHV-085-0243, HDR ISOL TO 1-FSV-085-0035A&B. ☐
   - 1-SHV-085-0262, HEADER SHUTOFF VLV. ☐
3. **BLOW DOWN** filter by opening then releasing petcock on filter. ☐
4. **OPEN** the following valves:
   - 1-SHV-085-0243, HDR ISOL TO 1-FSV-085-0035A&B. ☐
   - 1-SHV-085-0262, HEADER SHUTOFF VLV. ☐

Excerpt from 0-OI-32: Illustrates related Control Air System Pressure spectrum

BFN Unit 0	**Control Air System**	**0-OI-32 Rev. 0141 Page 73 of 117**

Attachment 1
(Page 1 of 1)

Control Air System Pressure Spectrum



Examination Outline Cross-reference:	Level	RO	SRO
400000 (SF8 CCS) Component Cooling Water	Tier #	2	------
A3.01 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operations of the CCWS including: • Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS	K/A #	400000A3.01	
	Importance Rating	3.0	------

Proposed Question: **# 61**

In accordance with 2-OI-27, Condenser Circulating Water (CCW) System, which **ONE** of the following completes the statements below?

CCW Pump Discharge Valves ___**(1)**___ OPEN when its respective CCW Pump is started.

If a CCW Pump is stopped or tripped when other CCW Pumps remain running, the associated CCW Pump Discharge Valves ___**(2)**___ CLOSE.

A. (1) automatically
(2) automatically

B. (1) automatically
(2) must be manually taken to

C. (1) must be manually taken to
(2) automatically

D. (1) must be manually taken to
(2) must be manually taken to

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with 2-OI-27, Condenser Circulating Water System, CCW Pump discharge valves automatically OPEN when its CCW Pump is started. For second part, CCW Pump discharge valves automatically CLOSE if the associated CCW Pump is stopped or tripped unless the CCW Pump was the last one running.

B INCORRECT: The first part is correct (*See A*). The second part is incorrect but plausible in that the majority of component cooling water system discharge valves (EECW, RBCCW, RCW…) must be manually taken to CLOSE if the respective pump is stopped or tripped. The CCW System has a number of unique automatic opening and/or closure signals associated with the normal operation of its pumps and discharge valves.

C INCORRECT: The first part is incorrect but plausible in that the majority of component cooling water system discharge valves (EECW, RBCCW, RCW…) must be manually taken to OPEN when its respective pump is started. The CCW System has a number of unique automatic opening and/or closure signals associated with the normal operation of its pumps and discharge valves. The second part is correct (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability to monitor automatic operations of the Condenser Circulating Water (CCW) System as it relates to a number of unique automatic opening and/or closure signals associated with the normal operation of its pumps and discharge valves. This question is rated as Memory due to the requirement to strictly recall specific facts related to the CCW System automatic operation.

Technical Reference(s): 2-OI-27, Rev. 94 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.050 Obj. 7 (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.050-11 001 #1609	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1609. OPL171.050-11 001

The following Condenser Circulating Water (CCW) configuration currently exists:

2A CCW pump OFF; discharge valve OPEN
2B CCW pump OFF; discharge valve CLOSED
2C CCW pump OFF; discharge valve CLOSED

Based on this configuration, which ONE of the following is correct?
(assume ONLY the pump control switches were manipulated)

A. 2A CCW pump was the FIRST pump stopped;
Cannot be restarted.

B✓ 2A CCW pump was the LAST pump stopped;
Cannot be restarted unless its discharge valve is closed fully.

C. 2A CCW pump was the FIRST pump stopped;
Cannot be restarted unless its discharge valve is at least 95% closed.

D. 2A CCW pump was the LAST pump stopped;
Can be restarted with its discharge valve open as long as 2B and 2C CCW pump discharge valves are closed.

Excerpt from 2-OI-27:

BFN Unit 2	Condenser Circulating Water System	2-OI-27 Rev. 0094 Page 8 of 124

3.0 PRECAUTIONS AND LIMITATIONS

A. A Condenser Circulating Water Pump will NOT start unless its discharge valve is closed and the remaining CCW pumps are either running or have their discharge valves closed.

B. Condenser Circulating Water Pumps will automatically trip if any of the following conditions occur:

1. 4160V undervoltage.
2. CCW Pump Discharge Valve motor controller closing coil is energized and neither of the two remaining CCW Pumps is running.
3. CCW Pump Discharge Valve motor controller closing coil is energized and the valve is ≥ 95% closed.
4. CCW Pump motor overcurrent.
5. CCW Pump motor high differential phase current (83 or 50 device).

C. Condenser Circulating Water Pump Discharge Valves:

1. Automatically open when its CCW Pump is started.
2. Automatically close if the associated CCW Pump is stopped or tripped unless the associated CCW Pump was the last one running.

D. Radwaste Discharge Valve, 2-FCV-77-61, automatically closes or is prevented

Examination Outline Cross-reference:	Level	RO	SRO
201002 (SF1 RMCS) Reactor Manual Control	Tier #	2	------
A4.05 (10CFR 55.41.7)	Group #	2	------
Ability to manually operate and/or monitor in the control room: • Rod select matrix	K/A #	201002A4.05	
	Importance Rating	3.1	------

Proposed Question: **# 62**

A Reactor Startup is in progress on Unit 2, with the following conditions:

- Control Rod 30-31 is selected
- The Operator at the Controls (OATC) has withdrawn Control Rod 30-31 from position 08 to position 12
- No further Control Rods will be withdrawn

In accordance with 2-OI-85, Control Rod Drive System, which **ONE** of the following completes the statements below?

A white light on ____**(1)**____ indicates that Control Rod 30-31 is selected.

To de-select Control Rod 30-31, the OATC will ____**(2)**____.

A. (1) the Control Rod Select Matrix **ONLY**
(2) depress 2-XS-85-40, Control Rod 30-31 Select Switch

B. (1) the Control Rod Select Matrix **ONLY**
(2) cycle 2-HS-85-46, CRD Power Switch OFF and then ON

C. (1) the Control Rod Select Matrix **AND** the Full Core Display
(2) depress 2-XS-85-40, Control Rod 30-31 Select Switch

D. (1) the Control Rod Select Matrix **AND** the Full Core Display
(2) cycle 2-HS-85-46, CRD Power Switch OFF and then ON

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that there is a myriad of indicating lights for different functions on the Full Core Display, and it is plausible that the selection of a Control Rod would not be indicated anywhere but the Control Rod Select Matrix. When the OATC selects a Control Rod, they verify the selection on the Rod Select Matrix. The second part is incorrect but plausible in that in accordance with 2-OI-85, Control Rod Drive System, to select a Control Rod the OATC will depress the desired Control Rod Select Switch. It is reasonable to assume that depressing the Control Rod Select Switch for Control Rod 30-31 will de-select the Control Rod.

B INCORRECT: The first part is incorrect but plausible (*See A*). The second part is correct (*See D*).

C INCORRECT: The first part is correct (*See D*). The second part is incorrect but plausible (*See A*).

D **CORRECT**: (*See Attached)* In accordance with 2-OI-85, once a Control Rod is selected the OATC will check that the lights on the Full Core Display and the Rod Select Matrix are illuminated. For second part, in accordance with 2-OI-85, when Control Rod movement is no longer desired and de-selecting Control Rods is desired, the Operator will place 2-HS-85-46, CRD POWER SWITCH in off and back on.

RO Level Justification: Tests the candidate's knowledge of the operation of the Rod Select Matrix during Control Rod withdrawals. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-85, Rev.146 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.005 Obj. 33 (As available)

Question Source:	Bank #	Nine Mile #36	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2017	

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

Proposed Question: #36

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

- Troubleshooting is in progress on the Rod Select Matrix.
- Control rod 26-11 has been selected by depressing its Rod Select pushbutton.
- It is now desired to de-select control rod 26-11.

Given the following separate operator actions:

(1) Depress and release the control rod 18-19 Rod Select pushbutton.

(2) Cycle the CONTROL ROD POWER switch to OFF and then back to ON.

Which of these operator actions, if any, will result in control rod 26-11 being de-selected in accordance with N1-OP-5, Control Rod Drive?

A. (1) only

B. (2) only

C. Either (1) or (2)

D. Neither (1) Nor (2)

Excerpts from 2-OI-85:

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0146 Page 64 of 255

6.6.3 Control Rod Notch Withdrawal

[1] **SELECT** the desired control rod by depressing the appropriate CRD ROD SELECT pushbutton, 2-XS-85-40.

[2] **ENSURE** CRD DRIVE WTR HDR DP is between 250 -270 psid on 2-PDI-85-17A by throttling CRD DRIVE WATER PRESS CONTROL VLV, 2-HS-85-23A, as necessary.

[3] **IF** selected control rod has been identified as a "fast notching" control rod by engineering, **THEN**

THROTTLE OPEN 2-PCV-085-0023 CRD, CRD DRIVE WATER PRESS CONTROL VLV, to as low as 180 psid using 2-HS-85-23A. (Otherwise N/A)

[4] **OBSERVE** the following for selected control rod:

- CRD ROD SELECT pushbutton is brightly ILLUMINATED.
- White light on the Full Core Display ILLUMINATED
- Rod Out Permit light ILLUMINATED.

[5] **ENSURE** ROD WORTH MINIMIZER operable and LATCHED in to correct ROD GROUP when Rod Worth Minimizer is enforcing.

[6] **PLACE** CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH and **RELEASE**.

[7] **OBSERVE** control rod settles into desired position **AND** ROD SETTLE light extinguishes.

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0146 Page 66 of 255

6.6.4 Continuous Rod Withdrawal

NOTES

1) Continuous control rod withdrawal may be used when a control rod is to be withdrawn greater than three notches.

2) When in areas of high notch worth, single notch withdrawal should be used instead of continuous rod withdrawal. Information concerning high notch worth is identified by Reactor Engineering in Control Rod Coupling Integrity Check, 2-SR-3.1.3.5A.

3) When continuously withdrawing a control rod to a position other than position 48, the CRD Notch Override Switch is held in the Override position and then the CRD Control Switch is held in the Rod Out Notch position.

- Both switches should be released when the control rod reaches two notches prior to its intended position.

  (Example: If a control rod is to be withdrawn from position 00 to position 12, the CRD Notch Override Switch and the CRD Control Switch would be used to move the control rod until reaching position 08, then both switches would be released.)

- If the rod settles in a notch prior to the intended position, the CRD Control Switch should be used to withdraw the rod to the intended position.

  (using the above example; If the control rod settles at a notch prior to the intended position of 12, the CRD Control Switch would be used to withdraw the control rod to position 12.)

[1] **SELECT** the desired control rod by depressing the appropriate CRD ROD SELECT pushbutton, 2-XS-85-40.

[2] **OBSERVE** the following for selected control rod:

- CRD ROD SELECT pushbutton is brightly ILLUMINATED.
- White light on the Full Core Display ILLUMINATED
- Rod Out Permit light ILLUMINATED.

[3] **ENSURE** ROD WORTH MINIMIZER operable and LATCHED in to correct ROD GROUP when Rod Worth Minimizer is enforcing.

[4] **CHECK** Control Rod is being withdrawn to a position greater than three notches.

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0146 Page 70 of 255

6.6.5 Return to Normal after Completion of Control Rod Withdrawal

[1] **WHEN** control rod movement is no longer desired **AND** deselecting control rods is desired, **THEN**:

[1.1] **PLACE** CRD POWER, 2-HS-85-46, in OFF.

[1.2] **PLACE** CRD POWER, 2-HS-85-46, in ON.

Examination Outline Cross-reference:

290003 (SF9 CRV) Control Room Ventilation

A3.01 (10CFR 55.41.7)

Ability to monitor automatic operations of the CONTROL ROOM HVAC including:

- Initiation / reconfiguration

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	290003A3.01	
Importance Rating	3.3	------

Proposed Question: **# 63**

An event occurred, resulting in 0-RM-90-259A, CONTROL ROOM VENTILATION RADIATION MONITOR, failing **UPSCALE** and 0-RM-90-259B reading 120 counts per minute.

Which **ONE** of the following predicts how the Main Control Room ventilation systems will respond?

Note: Control Room Emergency Ventilation System (CREV)

A. **NO** Control Room Ventilation isolation; **NEITHER** CREV fan auto starts.

B. **ALL** three Units' Control Room Ventilation Systems isolate and **BOTH** CREV fans auto start.

C. **ALL** three Units' Control Room Ventilation Systems isolate and **ONLY** the selected CREV fan auto starts.

D. **ONLY** Unit 1 and 2 Control Room Ventilation Systems isolate and **ONLY** the selected CREV fan auto starts.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that the candidate may believe that both Radiation Monitors must exceed the isolation setpoint in order to cause an isolation and a CREV start.

B INCORRECT: Incorrect but plausible in that when one of the Radiation Monitors exceeds the initation setpoint (221 cpm), all three Units' ventilation systems will isolate; however, only the selected CREV Fan will automatically start on an initiation signal.

C **CORRECT**: *(See attached)* In accordance with 1-OI-90, Radiation Monitoring System, when one of the Control Room Ventilation Radiation Monitors exceeds the initation setpoint (221 cpm), all three Units' ventilation systems will isolate and the selected CREV unit would auto start after a time delay.

D INCORRECT: Incorrect but plausible in that Radiation Monitor 0-RM-90-259A is physically located in the Unit 1 Mechanical Equipment room and 0-RM-90-259B is located in the Unit 3 Mechanical Equipment Room. Therefore, it is logical to assume that 0-RM-90-259A would control the isolation of Units 1 and 2 Control Room Ventilation System and 0-RM-90-259B would control the isolation of the Unit 3 Control Room Ventilation System. The selected CREV would auto start after a time delay.

RO Level Justification: Tests the candidate's knowledge of the isolation of the Control Room Ventilation Systems and the auto start of CREV given a failed high Control Room Radiation Monitor. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-OI-90, Rev. 71 (Attach if not previously provided)
0-OI-31, Rev. 162
OPL171.067, Rev. 22

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.067, Obj. 3 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.045-02 007 #1469
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Copy of Bank Question:

QUESTION 72 Rev 2

Control Room Ventilation Radiation Monitor 0-RM-90-259A fails upscale and 0-RM-90-259B reads 120 counts per minute.

Which one of the following predicts how the Control Room Ventilation System will respond?

A. **NO** ventilation isolation; **NEITHER** CREV fan auto starts.

B. **ONLY** Unit 1 and 2 Control Room Ventilation Systems isolate and **ONLY** the selected CREV fan auto starts.

C. **ALL** 3 Units' Control Room Ventilation Systems isolate and **ONLY** the selected CREV fan auto starts.

D. **ALL** 3 Units' Control Room Ventilation Systems isolate and **BOTH** CREV fans auto start.

Answer: C

Excerpt from 1-OI-90:

BFN Unit 1	Radiation Monitoring System	1-OI-90 Rev. 0071 Page 40 of 45

Illustration 1
(Page 3 of 3)

Radiation Monitoring System Operational Summary

SUBSYSTEM	OPERATION
Ventilation Exhaust Continuous Air Monitors	The following Continuous Air Monitors sample for total gas activity: • 1-RM-90-250; Monitors Reactor/Refueling Zone and Turbine Building exhaust ducts. • 1-RM-90-249 and 251; Monitors Turbine Building Exhaust Roof Ventilators. • 0-RM-90-252; Monitors Radwaste Building Exhaust Ducts.
Drywell Continuous Air Monitor 1-RM-90-256	Continuous Air Monitor, samples for total gas activity. A Primary Containment Isolation signal (high drywell pressure or low reactor water level) will auto close monitor isolation valves, 1-FSV-090-0254A and B, -0255, and -0257A and B. These valves can be reopened when the isolation signal is reset and the reset pushbuttons are depressed.
Control Room Isolation Radiation Monitors 0-RM-90-259A & B	One radiation detector monitors the fresh air supply duct to the Units 1 and 2 Control Room. Unit 3 Control Room Ventilation is monitored by a separate detector. High radiation at either detector initiates Emergency Control Room isolation and pressurization on all units. Panel 25-230

Excerpt from 0-OI-31:

BFN Unit 0	Control Bay and Off-Gas Treatment Building Air Conditioning System	0-OI-31 Rev. 0162 Page 88 of 243

5.29 Manual Initiation of Control Room Emergency Ventilation (CREV) System

NOTES

1) The CREV System automatically initiates from:

- PCIS Group 6

  Reactor vessel water level at "LEVEL 3"

  Drywell pressure at 2.45 psig

  Reactor zone exhaust radiation at 72 mr/hr

  Refuel zone exhaust radiation at 72 mr/hr

- Control Room High Radiation

  221 CPM above background on U1 & 2 (3) Control Room

  Radiation-Gas Radiation Recorder, 0-RR-90-259A(B)

2) The CREV System is normally in Standby Readiness.

3) Performance of this instruction requires the use of two (2) Briggs and Stratton keys.

4) Dampers 0-FCO-31-150B, D, E, F, and G close automatically on auto initiation or a start from the control room using CREV TRAIN A (B) INIT/CB ISOL 0-HS-31-150A (0-HS-31-150B) on Panel 2-9-22. The dampers will NOT operate in response to the local fan control switches (0-HS-31-7214B or 0-HS-31-7213B) or to the control room fan control switches (0-HS-31-7214A or 0-HS-31-7213A).

Excerpts from OPL171.067 Lesson Plan:

pushbuttons. A local direct acting thermostat controls temperature.

b. Communications Room ACUs provide cooling solely to the Communications Room. These are two 100% capacity A/C Units. One normally in service and one in standby, using a local manual selector switch and pushbuttons. A local direct acting thermostat controls temp.

7. Control Room Emergency Ventilation (CREV) is designed to supply and process the outdoor air needed for pressurization during isolated conditions. There are 2 CREV units rated at 3000 cfm each. A CREV unit consists of Motor-driven fan, (power supply is from 480V RMOV Bd 1A for CREV Fan A; RMOV Bd 3B for CREV Fan B), HEPA filter (common), charcoal filter assemblies located in the CREVS Equipment Room, charcoal heater, and inlet isolation damper and a backflow check outlet damper. They are designed to maintain a positive pressurization to 1/8" w.g. minimum to the control room.

Tech. Spec. 3.7.3
ILT/LOR 2j,e
ILT 7,8 NLO 6 ,7
(Old CREV Units abandoned in place as Aux Pressurization Systems) Figure -4
2-47E2865-4

a. A CREV may be started manually from control room Panel 2-9-22 if local control switch is in AUTO position via a 3 position, spring-return to center switch. (STOP-AUTO-START). Actuates only the CREVS unit & associated damper, not the isolation dampers.

Red indicating lights on panel 3-9-21 to provide indication of
CREV Fan A and/or B running on Unit 3.
Annunciators are on panel 9-6 for all units.

b. There is also a 2 position maintained contact, one per train, AUTO-INITIATE/TEST switch which is used to perform system level actions for that train (primarily testing). It provides the same response as auto start.

c. Local start at local control station in Relay Room is done using a 2 position maintained contact, one per train, AUTO-TEST switch. Isolation dampers do not operate automatically if started from local panel.

d. Automatic start signals are:

i. High radiation of 221 cpm above background + 2 Min TD (270 cpm Tech Specs) in air inlet ducts to Control Room from (Radiation monitor RM 90-259A Units 1 & 2, Radiation monitor RM 90-259B Unit 3). Either monitor starts selected CREV unit.

ILT/LOR 2e,g
ILT 13, NLO 14

T. S. 3.3.7.1

ii. Reactor zone ventilation systems radiation high $\geq$72 MR/hr

iii. Refuel zone ventilation systems radiation high $\geq$72 MR/hr

iv. Low reactor water level at +2 inches above instrument zero

v. High primary containment pressure $\geq$2.45 psig

e. On receipt of a start signal, normal outside air paths (see below) to elevation 3C are isolated. The selected CREV unit starts once the inlet damper is full open. This supplies

ILT 18, NLO 17
The inlet damper is normally closed & fails

OPL171.067 , Plant Heating, Ventilation, and Air Conditioning (HVAC) Systems, Rev# 22

e. On receipt of a start signal, normal outside air paths (see below) to elevation 3C are isolated. The selected CREV unit starts once the inlet damper is full open. This supplies pressurizing air to the Unit 1, 2 and 3 Control Rooms. One CREV unit can supply all three control rooms, so the STBY CREV unit will not normally start. Once started, the CREV

ILT 18, NLO 17
The inlet damper is normally closed & fails closed. Damper opening takes ~70 seconds. While in the

Examination Outline Cross-reference:	Level	RO	SRO
290002 (SF4 RVI) Reactor Vessel Internals	Tier #	2	------
K5.07 (10CFR 55.41.5)	Group #	2	------
Knowledge of the operational implications of the following concepts as they apply to REACTOR VESSEL INTERNALS:	K/A #	290002K5.07	
• Safety limits	Importance Rating	3.9	------

Proposed Question: **# 64**

In accordance with Reactor Pressure Safety Limits, which **ONE** of the following completes the statements below?

The Reactor Steam Dome Pressure Safety Limit is ___**(1)**___.

If violated, the REQUIRED ACTION is to restore Reactor Steam Dome Pressure Safety Limit compliance and insert **ALL** insertable Control Rods within ___**(2)**___.

A. (1) 1325 psig
(2) 2 hours

B. (1) 1325 psig
(2) 10 hours

C. (1) 1375 psig
(2) 2 hours

D. (1) 1375 psig
(2) 10 hours

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with the Tech Spec Safety Limits 2.0, the Reactor Coolant System Pressure Safety Limit, states that the Reactor Steam Dome Pressure shall be ≤1325 psig. For second part, in accordance with Tech Specs, when any Safety Limit has been violated, within 2 hours, restore compliance and Insert all insertable Control Rods.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible if the candidate confuses a Safety Limit violation as a Loss of Safety Function (LOSF) as stated in OPDP-8, Operability Determination Process and Limiting Conditions for Operation Tracking. This would lead them to believe to insert all insertable Control Rods to be in MODE 2 within 10 hours in accordance with LCO 3.0.3.

C INCORRECT**:** First part is incorrect but plausible in that the Safety Limit at the lowest elevation of the Reactor Coolant System is 1375 PSIG, but the question is asking for the Reactor Steam Dome Safety Limit. The second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible *(See C*). The second part is incorrect but plausible (*See B*).

RO Level Justification: Tests candidate's knowledge of the Safety Limit for Reactor Pressure and the actions for exceeding a Technical Specification Reactor Pressure Limit. This question is rated as Memory due to strictly recalling facts related to Technical Specification Reactor Pressure limits.

Technical Reference(s): Tech Spec 2.0, Amend. 299 (Attach if not previously provided)
Tech Spec 3.0, Amend. 266
OPDP-8, Rev. 26

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.087, Obj. 14 (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.009-14 007, #388	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

388. OPL171.009-14 007

Which ONE of the following completes the statements below?

The Limiting Condition for Operation (LCO) statement for Unit 1 Reactor Steam Dome Pressure, LCO 3.4.10, has been violated; the required action is to restore Reactor Steam Dome pressure to within the limit in __(1)__ .

The Reactor Coolant System Pressure Safety Limit is __(2)__.

A✓ (1) 15 minutes
(2) 1325 psig

B. (1) 1 hour
(2) 1375 psig

C. (1) 15 minutes
(2) 1375 psig

D. (1) 1 hour
(2) 1325 psig

Excerpt from Tech Spec 2.0:

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 585 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 585 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.06 for two recirculation loop operation or ≥ 1.08 for single loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Excerpt from Tech Spec 2.0 Bases:

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure - High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1965 Edition, including Addenda through the summer of 1965 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1, 1967 Edition (Ref. 6), and the additional requirements of GE design and procurement specifications (Ref. 7) which were implemented in lieu of the outdated B31 Nuclear Code Cases - N2, N7, N9, and N10, for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig for suction piping and 1326 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

Excerpt from Tech Spec 3.0:

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.
LCO 3.0.3	When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 2 within 10 hours; b. MODE 3 within 13 hours; and c. MODE 4 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, and 3.

(continued)

Excerpt from OPDP-8: Supports Distractors B(2), D(2)

NPG Standard Department Procedure	Operability Determination Process and Limiting Conditions for Operation Tracking	OPDP-8 Rev. 0026 Page 56 of 115

Attachment 6
(Page 6 of 31)

3.3.1 Perform "LOSF Evaluation". (continued)

J. If an LOSF is identified, ensure that the most appropriate action is taken considering the current MODE and plant conditions:

1. The appropriate conditions and required actions of the LCO in which the LOSF exist are entered, or

2. If no appropriate LCO condition and required actions exist for the LOSF, then LCO 3.0.3 shall be evaluated for entry and other appropriate limitations and remedial or compensatory actions are to be taken commensurate with the LOSF, or

3. If no LCO exists for the LOSF and the plant is in a MODE where LCO 3.0.3 is not applicable, then appropriate limitations and remedial or compensatory actions are to be taken commensurate with the LOSF.

4. Refer to NPG-SPP-01.12, TVA Nuclear Event Response Procedure, to determine additional follow-up actions.

5. Ensure Condition Report is initiated.

Examination Outline Cross-reference:	Level	RO	SRO
259001 (SF2 FWS) Feedwater	Tier #	2	------
A2.06 (10CFR 55.41.5)	Group #	2	------
Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	259001A2.06	
• Loss of A.C. electrical power	Importance Rating	3.2	------

Proposed Question: **# 65**

With respect to the Unit 3 Reactor Feedwater Control System (RFWCS), which **ONE** of the following completes the statements below.

The normal power supply to the RFWCS Panel Display Station (PDS) on Panel 3-9-5 is **___(1)___**. If the normal power supply is lost to the PDS, RFWCS **___(2)___**.

A. (1) Unit Preferred
(2) **CONTINUES** control at the last known Reactor Water Level setpoint

B. (1) Unit Preferred
(2) **MUST** be transferred to manual

C. (1) Plant Preferred
(2) **CONTINUES** control at the last known Reactor Water Level setpoint

D. (1) Plant Preferred
(2) **MUST** be transferred to manual

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 3-AOI-57-4, Loss of Unit Preferred, RFWCS components are impacted from Panel 3-9-9 Cabinet 6 Unit Preferred loss of power. RFWCS Panel Display Stations (PDS) on Panel 3-9-5 are disabled during a Loss of Unit Preferred. PDS controls are inoperative and displays become blank. For second part, even though the RFWCS PDS are disabled, the system continues controlling Reactor Water Level on the last known setpoint.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that the RFWCS Panel Display Stations are in fact disabled, the candidate could believe the system will lose automatic control thereby requiring transferring to manual control. Manual governor control is only applicable if the Operator needed to control RFPT speed, but not applicable to PDS controls.

C INCORRECT: First part is incorrect but plausible if the candidate confuses the given power loss impact as from Panel 9-9 Cabinet 4 Plant Preferred, instead of Panel 3-9-9 Cabinet 6 Unit Preferred. A Plant Preferred loss would impact some Feedwater components, just not specific to the Feedwater Control aspect. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests candidate's ability to predict the impact of a loss A.C. electrical power on the Reactor Feedwater Control System. Specifically tests the required Abnormal Operating Instruction response as it relates to BFN's complex 120VAC Electrical Distribution System. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-OI-3, Rev. 109 (Attach if not previously provided)
3-AOI-57-4, Rev. 38
0-AOI-57-3, Rev. 57

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.012 Obj. 2b (As available)

Question Source:

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

Question History:

Last NRC Exam	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Excerpts from 3-AOI-57-4:

BFN Unit 3	Loss of Unit Preferred	3-AOI-57-4 Rev. 0038 Page 4 of 31

2.0 SYMPTOMS (continued)

H. RFW Control System Panel Display Stations on Panel 3-9-5 disabled. PDS Controls are inoperative and displays become blank. The RFW Control System continues to control system parameters according to water level setpoint.

I. The following RFW Control System annunciators in alarm on Panel 3-9-6:

1. RFPT GOVERNOR POWER FAILURE OR GOVERNOR ABNORMAL, 3-XA-46-8, 9 & 10 (3-XA-55-6C window 12)

2. RFWCS TROUBLE, 3-LA-46-5A (3-XA-55-6C Window 28)

J. Loss of power to Cabinet 6 will cause a loss of flow signal to CNDS FLOW CONTROL SHORT CYCLE, 3-FIC-2-29 when in auto. This will cause 3-FCV-2-29A/B to open resulting in rising condensate flow. This can adversely affect Condensate & Feedwater system NPSH and Reactor water level.

K. The following EHC Control System annunciators in alarm on Panel 3-9-6:

1. EHC POWER ABNORMAL (3-XA-55-7B Window 5)

2. EHC/TSI SYSTEM TROUBLE (3-XA-55-7B Window 6)

L. EHC Control System PLU 1 (power load unbalance) can bypass with a sustained loss of power to Panel 9-9 Cabinet 5. An uninterruptible power supply will keep the PLU energized for approximately 15 minutes after normal power is lost.

M. EHC Control System HMI on Panel 3-9-31 may become blank if power is lost to Panel 9-9 Cabinet 6. An uninterruptible power supply will keep this component energized for approximately 15 minutes after normal power is lost. RECIRC FLOW SYSTEM TROUBLE ALARM (3-XA-55-4A, Window 23).

N. Loss of power to CRD select modules.

O. PNL 9-21 SYS LEAK DETECTION POWER FAILURE (3-XA-55-3D, Window 31) on loss of power to Panel 3-9-21 Steam Leak Detection Panel.

BFN Unit 3	Loss of Unit Preferred	3-AOI-57-4 Rev. 0038 Page 6 of 31

3.0 AUTOMATIC ACTIONS

A. Panel 3-9-9 Cabinet 6 Unit Preferred automatically transfers to the alternate power source.

B. RFWCS Panel Display Stations are disabled and the system continues controlling on the last known signal.

C. 3-FCV-85-11(A/B) closes.

D. 3-FCV-1-58, UPSTREAM MSL DRAIN TO CONDENSER, auto opens. (REFER TO NOTE ABOVE)

E. Loss of one out of two auctioneered power sources for EHC Control System (ICS is the other power source).

F. Loss of power to Turbine Supervisory Instrumentation Panel 3-9-46.

G. Loss of one out of two auctioneered power sources for RFPT 3C Woodward Governor and Final Driver (ICS is the other power source).

H. Loss of one out of two auctioneered power sources for Recirc Flow Control System (ICS is the other power source).

BFN Unit 3	Loss of Unit Preferred	3-AOI-57-4 Rev. 0038 Page 28 of 31

Attachment 2
(Page 1 of 4)

Panel 3-9-9 Cabinet 6 Unit Preferred Loads

A. BKR 601, PANEL 3-9-5 FEEDWATER CONTROL

1. RFW Turbine 3A Speed Control, 3-SIC-46-8
2. RFW Turbine 3B Speed Control, 3-SIC-46-9
3. RFW Turbine 3C Speed Control, 3-SIC-46-10
4. RFW START-UP LEVEL CONT, 3-LIC-3-53

BFN Unit 3	Loss of Unit Preferred	3-AOI-57-4 Rev. 0038 Page 29 of 31

Attachment 2
(Page 2 of 4)

Panel 3-9-9 Cabinet 6 Unit Preferred Loads

E. BKR 605, PANEL 3-9-29 FW, STEAM AND CNDS

1. HOTWELL PRESS INDICATION (3-XR-2-26)
2. HOTWELL TEMP INDICATION (3-XR-2-26)
3. CNDS FLOW SJAE A AND B (3-FI-2-42)
4. CNDS PRESS AFTER DEMIN (3-PI-2-46)
5. CST FLOW FROM HOTWELL INDICATION (3-XR-2-26)
6. CST FLOW TO HOTWELL INDICATION (3-XR-2-26)
7. FEEDWATER HEATER DRAIN CLR A-5 FLOW TO CNDR (3-FI-6-16)
8. FEEDWATER HEATER DRAIN CLR B-5 FLOW TO CNDR (3-FI-6-34)
9. FEEDWATER HEATER DRAIN CLR C-5 FLOW TO CNDR (3-FI-6-52)
10. CNDS PUMPS DISCH HEADER FLOW INDICATION (3-XR-2-26)
11. RCW HEADER PRESSURE INDICATOR (3-PI-24-18)
12. CONTROL AIR PRESSURE (3-PI-032-0088)
13. SERVICE AIR PRESSURE INDICATOR (0-PI-33-3A/3)
14. HP FIRE PROTECT. HDR PRESSURE INDICATOR (0-PI-26-44A/3)

F. BKR 606, (PANEL 3-9-5)

REACTOR WATER LEVEL CONTROL, 3-LIC-46-5

Excerpt from 3-OI-3:

BFN Unit 3	Reactor Feedwater System	3-OI-3 Rev. 0109 Page 295 of 330

Attachment 2
(Page 4 of 4)

RFPT Speed Control Panel Display Station

3.0 DESCRIPTION

RFPT Speed Control switch provides Unit Operator direct control of RFPT speed between 600 and 5800 rpm. Manual governor control is established when switch is depressed in MANUAL GOVERNOR position and amber indicating light is illuminated. With switch in this position, a speed demand signal (4-20 ma) is sent to the Woodward Governor. Taking switch to Raise or Lower adjusts RFPT speed at a rate of 50 rpm per second. When switch is pulled up, amber light extinguishes and RFPT speed control is transferred to associated RFPT Speed Control PDS located below Speed Control Switch.

Just like the Speed Control switch, Speed Control PDS offers Unit Operator RFPT governor control from 600 to 5800 rpm. With Speed Control switch pulled up, PDS placed in MANUAL (amber light illuminated) Column 3 (CO) selected on PDS, the Ramp RAISE/Ramp LOWER pushbuttons can be used to adjust RFPT speed at a linear rate.

When PDS is placed in AUTO (blue light illuminated), then control of RFPT is transferred to Reactor Water Level Control PDS (Master Controller).

4.0 FAILURE MECHANISMS

RFPT Trip	Speed Control trips to MANUAL GOVERNOR. The amber indicating light illuminates. However, Unit Operator must depress Speed Control switch.
Loss of Unit Preferred	All indications are lost on PDS. If PDS was controlling RFPT, then RFW Control System will continue to control with last known demand values. PDS is rendered inoperative. Unit Operator can still control RFPT with Speed Control switch in MANUAL GOVERNOR.

Excerpt from 0-AOI-57-3: Supports Distractors C(1), D(1)

BFN Unit 0	Loss of Plant Preferred	0-AOI-57-3 Rev. 0057 Page 28 of 30

Attachment 3
(Page 9 of 10)

Panel 9-9 Cabinet 4 Plant Preferred Loads

13.0 PANEL 9-59

A. TURBINE-GENERATOR TEMPERATURE RECORDER, TR-47-164

B. PANEL 25-277 INSTRUMENTATION POWER SUPPLY; GENERATOR CONDITION MONITORING, XANIS-35-85

C. GENERATOR CONDITION MONITOR, ZANR-35-85

14.0 RAW COOLING WATER INSTRUMENTATION E/P TRANSDUCERS POWER SUPPLY

A. ALL UNITS

1. RBCCW HEAT EXCHANGER A TEMPERATURE MODIFIER, TM-24-80
2. RBCCW HEAT EXCHANGER B TEMPERATURE MODIFIER, TM-24-85
3. FEEDWATER LINE A(B) TEMPERATURE TRANSMITTER, TT-3-48C(50C)
4. REACTOR FEED PUMP A OIL TEMPERATURE MODIFIER, TM-24-56
5. REACTOR FEED PUMP B OIL TEMPERATURE MODIFIER, TM-24-54
6. REACTOR FEED PUMP C OIL TEMPERATURE MODIFIER, TM-24-52

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.29 (10CFR 55.41.10)	Tier #	3	------
Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	Group #	------	------
	K/A #	G2.1.29	
	Importance Rating	4.1	------

Proposed Question: **# 66**

In accordance with BFN-ODM-4.5, Operator Aids and Operator Information Systems, which **ONE** of the following completes the statement below regarding system status?

During a Main Control Room panel walk down, the Operator determines that a system is aligned correctly when the **NORMALLY RUNNING** pumps have a ___**(1)**___ red lens cover and the **NORMALLY CLOSED** valves have a ___**(2)**___ red lens cover.

A. (1) clear
(2) diffused

B. (1) clear
(2) clear

C. (1) diffused
(2) diffused

D. (1) diffused
(2) clear

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that for pumps, the unlit green light shall be a clear green lens cover. Second part is incorrect but plausible in that for valves, if NORMALLY CLOSED, it shall have a diffused green lens cover.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT: First part is correct (*See D)*. Second part is incorrect but plausible (*See A*).

D **CORRECT**: (*See attached*) In accordance with BFN-ODM-4.5, Operator Aids and Operator Information Systems, to help the Operators as they walk the boards down, the following lens convention has been developed to aid the process: for pumps, if the pump is NORMALLY RUNNING it shall have a lit diffused red lens cover and the unlit green light shall be a clear green lens cover. For second part, for valves, if NORMALLY CLOSED it shall have lit diffused green lens cover and the red light shall be a clear unlit red lens cover.

RO Level Justification: Tests the candidate's knowledge of how to conduct system lineups/status checks on pumps and valves in the Main Control Room. This question is rated as Memory due to strictly recalling facts.

Technical Reference(s): BFN-ODM-4.5, Rev. 4.5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.065 Obj. 11 (As available)

Question Source:

	Bank #	BFN 1510 #68	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2015	

Question Cognitive Level:

Memory or Fundamental Knowledge	X
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	X
55.43	

Comments:

Copy of Bank Question:

QUESTION 68 Rev 1

In accordance with ODM-4.5, Operator Aids and Operator Information System, how does the Unit Operator determine during the panel walk down, that a system is aligned correctly?

The normally running pumps shall have a lit ___ (1) ___ red lens cover.

The normally opened valves shall have an extinguished ___ (2) ___ green lens cover.

A. (1) clear
   (2) clear

B. (1) diffused
   (2) diffused

C. (1) diffused
   (2) clear

D. (1) clear
   (2) diffused

Answer: C

Excerpts from BFN-ODM-4.5:

BFN Operations Directive Manual	Operator Aids and Operator Information Systems	BFN-ODM-4.5 Rev. 0006 Page 5 of 7

E. Light Lens Cover Convention

1. OPDP-1 currently requires a walk down of the control panels to monitor plant status. During the walk downs Operators are to check switch positions, instrumentation readings, and status lights.

BFN Operations Directive Manual	Operator Aids and Operator Information Systems	BFN-ODM-4.5 Rev. 0006 Page 6 of 7

1.1 ODM-4.5 DIRECTIVE (continued)

2. To help the operator as they walk the board down, the following lens convention has been developed to aid the process:

For Pumps:

- If the pump is normally running, it shall have a lit diffused red lens cover. The unlit green light for this pump shall be a clear green lens cover.
- If the pump is normally off it shall have a lit diffused green lens cover. The unlit red light for this pump shall be a clear red lens cover.

For Valves:

- If the valve is normally open it shall have a lit diffused red lens cover. The green light for this valve shall be a clear unlit green lens cover.
- If the valve is normally closed it shall have a lit diffused green lens cover. The red light for this valve shall be a clear unlit red lens cover.

For status lights:

- If the status light is normally lit it shall have a diffused lens cover of the appropriate color.
- If the status light is normally unlit it shall have a clear lens cover of the appropriate color.

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.35 (10CFR 55.41.10)	Tier #	3	------
Ability to determine Technical Specification Mode of Operation.	Group #	------	------
	K/A #	G2.2.35	
	Importance Rating	3.6	------

Proposed Question: **# 67**

Which **ONE** of the following sets of conditions satisfies the Tech Spec definition of MODE 2?

A. A Reactor SCRAM has just occurred. The MODE SWITCH is in SHUTDOWN. Moderator Temperature is 480º F. The MSIVs are closed.

B. The Reactor is shutdown. The MODE SWITCH is in START&HOT STBY. Moderator Temperature is 180º F. All Control Rods are fully inserted.

C. Preparations are in progress for a Reactor Startup. The MODE SWITCH is in SHUTDOWN. Moderator Temperature is 135º F. All Control Rods are fully inserted.

D. The Reactor is shutdown. The MODE SWITCH is in REFUEL. Moderator Temperature is 140º F. All Reactor Vessel Head Closure Bolts are **NOT** fully tensioned.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that the given conditions place the plant in MODE 3. There are 5 different Modes listed in Tech Specs with various MODE SWITCH positions and Moderator Temperature considerations, and the candidates could easily confuse the different MODES.

B **CORRECT**: *(See attached)* In accordance with Tech Spec 1.0, Table 1.1-1, when The MODE SWITCH is in START&HOT STBY, regardless of Moderator Temperature or Control Rod position, the plant is in MODE 2.

C INCORRECT: Incorrect but plausible in that the given conditions place the plant in MODE 4.

D INCORRECT: Incorrect but plausible (*See A*).

RO Level Justification: Tests the candidate's ability to determine the plant Mode in accordance with Technical Specifications. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): Unit 1 Tech. Specs 1.0, Amend. 234 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.087 Obj. 8 (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.087-09 003 #2237	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

2237. OPL171.087-09 003

Which ONE of the following sets of plant conditions satisfies the Technical Specification definition of MODE 4?

A. A reactor scram has just occurred. Moderator temperature is 480 F. The MSIVs are closed. The MODE SWITCH is in "**Shutdown**".

B. The reactor is shutdown. Moderator temperature is 135 F. The MODE SWITCH is in "**Start/Hot Stby"**. All control rods are fully inserted.

C. The reactor is shutdown. Moderator temperature is 140 F. All reactor vessel head closure bolts are **NOT** fully tensioned. The MODE SWITCH is in "**Shutdown**".

D✓ Preparations are in progress for a reactor startup. All control rods are fully inserted with the MODE SWITCH in "**Shutdown**". Moderator temperature is 180 F.

Correct Answer: D

Excerpt from Tech Spec 1.0:

Definitions
1.1

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[a] or Startup/Hot Standby	NA
3	Hot Shutdown[a]	Shutdown	> 212
4	Cold Shutdown[a]	Shutdown	≤ 212
5	Refueling[b]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Examination Outline Cross-reference:

G2.2.41 (10CFR 55.41.10)

Ability to obtain and interpret station electrical and mechanical drawings.

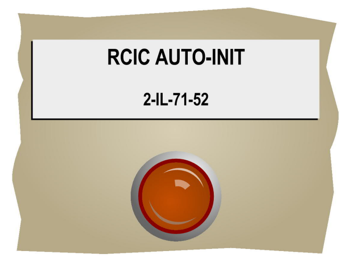
	RO	SRO
Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.2.41	
Importance Rating	3.5	------

Proposed Question: **# 68**

Given the following drawing excerpt of Unit 2 RCIC Initiation Logic, which **ONE** of the following completes the statements below in accordance with 2-45E626-1, Wiring Diagram, RCIC System Schematic Diagram?

The four primary contacts in the Reactor Vessel Low Water Level portion of the circuit are actuated directly by ___**(1)**___ System Relays.

On a RCIC automatic initiation signal, if relay 13A-K**3** fails to energize, the RCIC AUTO-INITIATE light, shown here, ___**(2)**___.



[SEE THE ATTACHED RCIC DRAWING, 2-45E626-1]

A. (1) RHR
(2) will illuminate

B. (1) RHR
(2) will NOT be illuminated

C. (1) RCIC
(2) will illuminate

D. (1) RCIC
(2) will NOT be illuminated

Proposed Answer: **B**

Explanation (Optional):

A First part is correct *(See B)*. The second part is incorrect but plausible in that the relay 13A-K3 is given as 'fails to energize'. If the candidate misreads the print relay and contact convention of energize to actuate for DC power circuits on a RCIC automatic initiation signal, which requires relay 13A-K3 to energize, this would lead then to believe the relay 13A-K4 may still energize. Thereby closing the contact 13A-K4 in the logic string to illuminate 2-IL-71-52, RCIC AUTO-INITIATE light.

B **CORRECT**: (*See attached*) The relays associated with the four primary contacts in the Reactor Vessel Low Water Level portion of the circuit are designated as "**10A**-KX," which translates in GE terms to RHR System Relays, components, and switches. RCIC Relay designators are **13A**-KX. For second part, if relay 13A-K**3** fails to **energize** upon a RCIC automatic initiation signal, then relay 13A-K**4** will NOT be energized to seal in. In turn, the contact 13A-K**4** will NOT close in the logic string for 2-IL-71-52, RCIC AUTO-INITIATE light, as shown. Therefore, the RCIC AUTO-INITIATE light will NOT be illuminated.

C INCORRECT: First part is incorrect but plausible in that normally, in system logic, the relays that initiate, trip, or isolate a system are in fact designated as related to that system, by GE designators. It is an extremely small percentage of the overall number of system relays at BFN that cross over boundaries. The second part is incorrect but plausible *(See A).*

D INCORRECT: First part is incorrect but plausible *(See C)*. The second part is correct *(See B)*.

RO Level Justification: Tests the candidate's ability to obtain and interpret Browns Ferry's electrical drawings as it relates to the RCIC System. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): (Attach if not previously provided)
- 2-45E626-1, Rev. 17
- 2-730E937 Sht 1, Rev. 26
- 2-730E937 Sht 4, Rev. 21
- OPL171.040, Rev. 31

Proposed references to be provided to applicants during examination: **Excerpt from RCIC Drawing, 2-45E626-1, Panel 2-9-3 with 2-IL-71-52, RCIC Auto-Initiation light**

Learning Objective: OPL171.040 Obj. 4, 5 (As available)
OPL171.093 Obj. 4

Question Source:	Bank #		
	Modified Bank #	BFN 1909 #68	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis **X**

10 CFR Part 55 Content:
- 55.41 **X**
- 55.43

Copy of Bank Question:

68. Given the following drawing excerpt of Unit 2 RCIC Initiation Logic, which **ONE** of the following completes the statements below in accordance with 2-45E626-1, Wiring Diagram, RCIC System Schematic Diagram?

The four primary contacts in the Reactor Vessel Low Water Level portion of the circuit are actuated directly by ___**(1)**___ System Relays.

The **LAST** RCIC Relay that energizes to cause the RCIC AUTO-INITIATE light, shown here, to illuminate on an initiation signal ___**(2)**___ a seal-in relay.

[SEE THE ATTACHED RCIC DRAWING, 2-45E626-1]

A. **(1)** RHR
(2) is

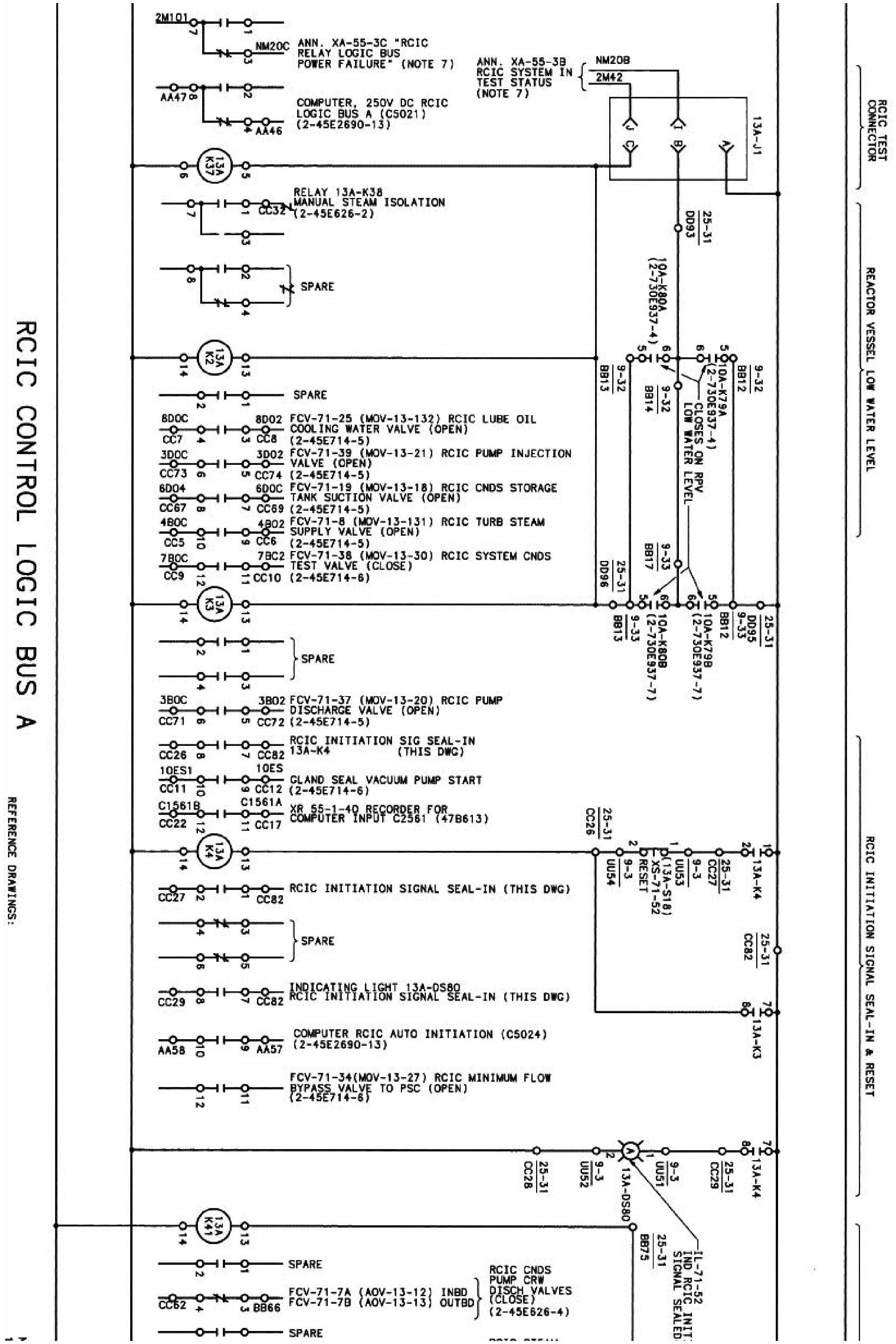
B. **(1)** RHR
(2) is NOT

C. **(1)** RCIC
(2) is

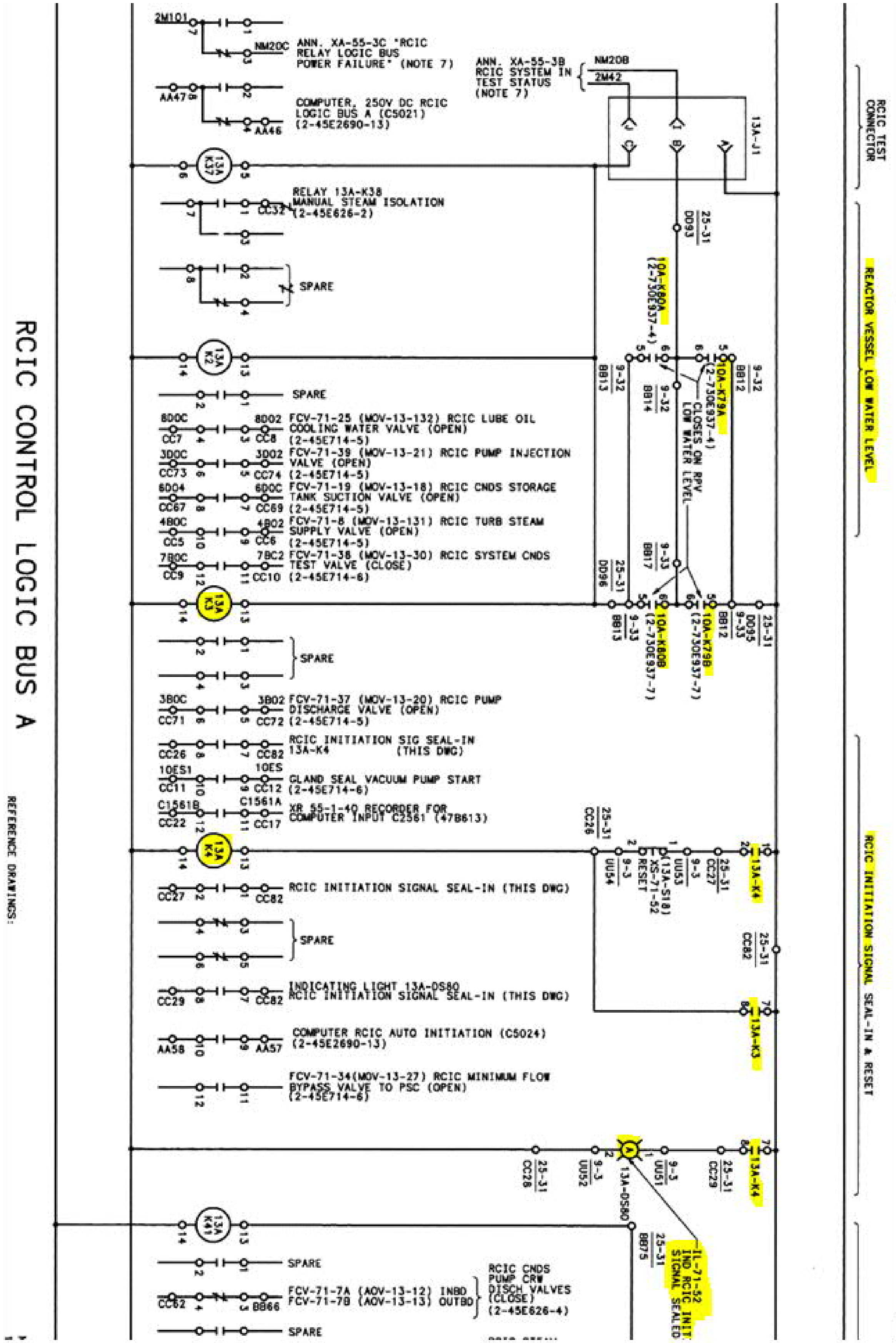
D. **(1)** RCIC
(2) is NOT

Correct Answer: **A**

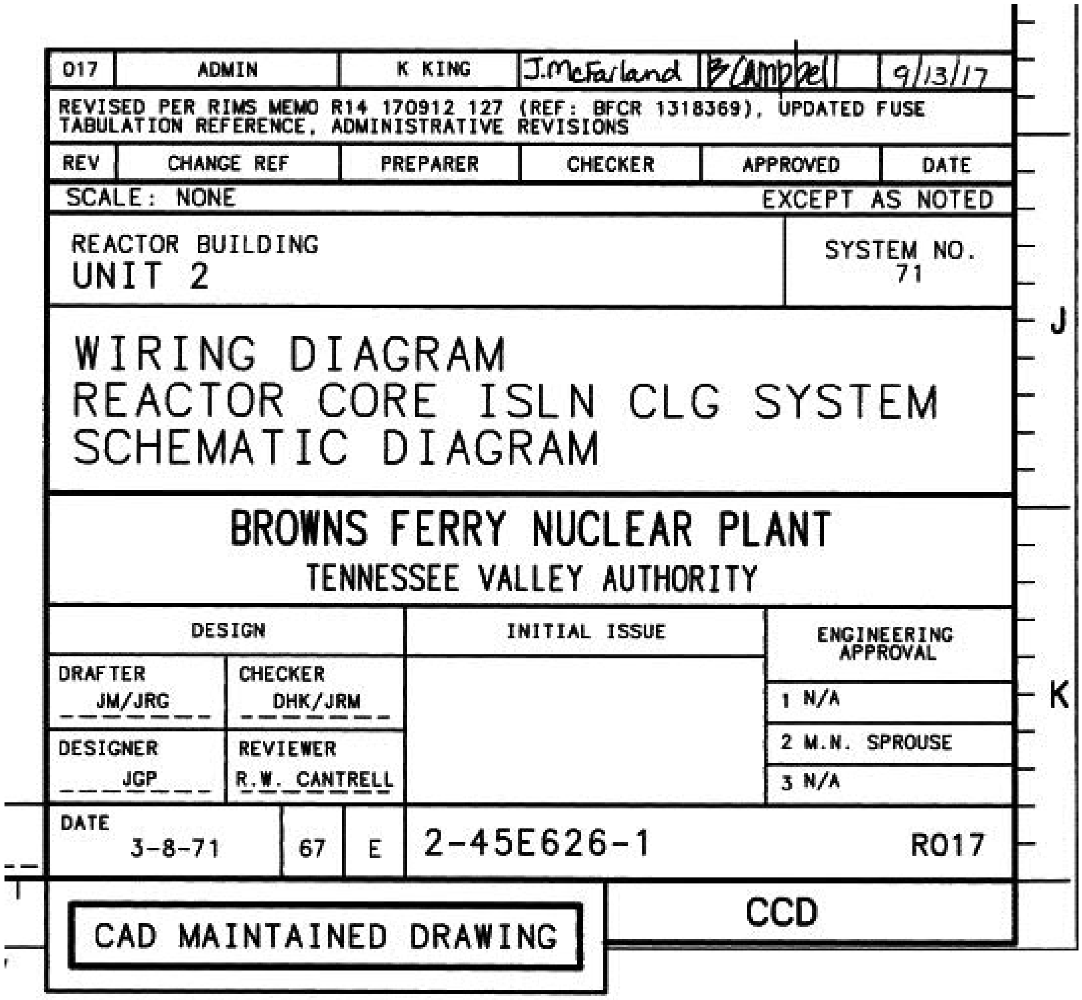
[Attachment provided to candidate: excerpt from 2-45E626-1]



Excerpt from 2-45E626-1: highlighting related relays and contacts:



Excerpt from 2-45E626-1 (to show title and rev for included drawing excerpt):



REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE
017	ADMIN	K KING	J.McFarland	B Campbell	9/13/17

REVISED PER RIMS MEMO R14 170912 127 (REF: BFCR 1318369), UPDATED FUSE TABULATION REFERENCE, ADMINISTRATIVE REVISIONS

SCALE: NONE EXCEPT AS NOTED

REACTOR BUILDING
UNIT 2

SYSTEM NO. 71

WIRING DIAGRAM
REACTOR CORE ISLN CLG SYSTEM
SCHEMATIC DIAGRAM

BROWNS FERRY NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY

DESIGN: DRAFTER JM/JRG; CHECKER DHK/JRM; DESIGNER JGP; REVIEWER R.W. CANTRELL

INITIAL ISSUE

ENGINEERING APPROVAL
1 N/A
2 M.N. SPROUSE
3 N/A

DATE 3-8-71 | 67 | E | 2-45E626-1 | R017

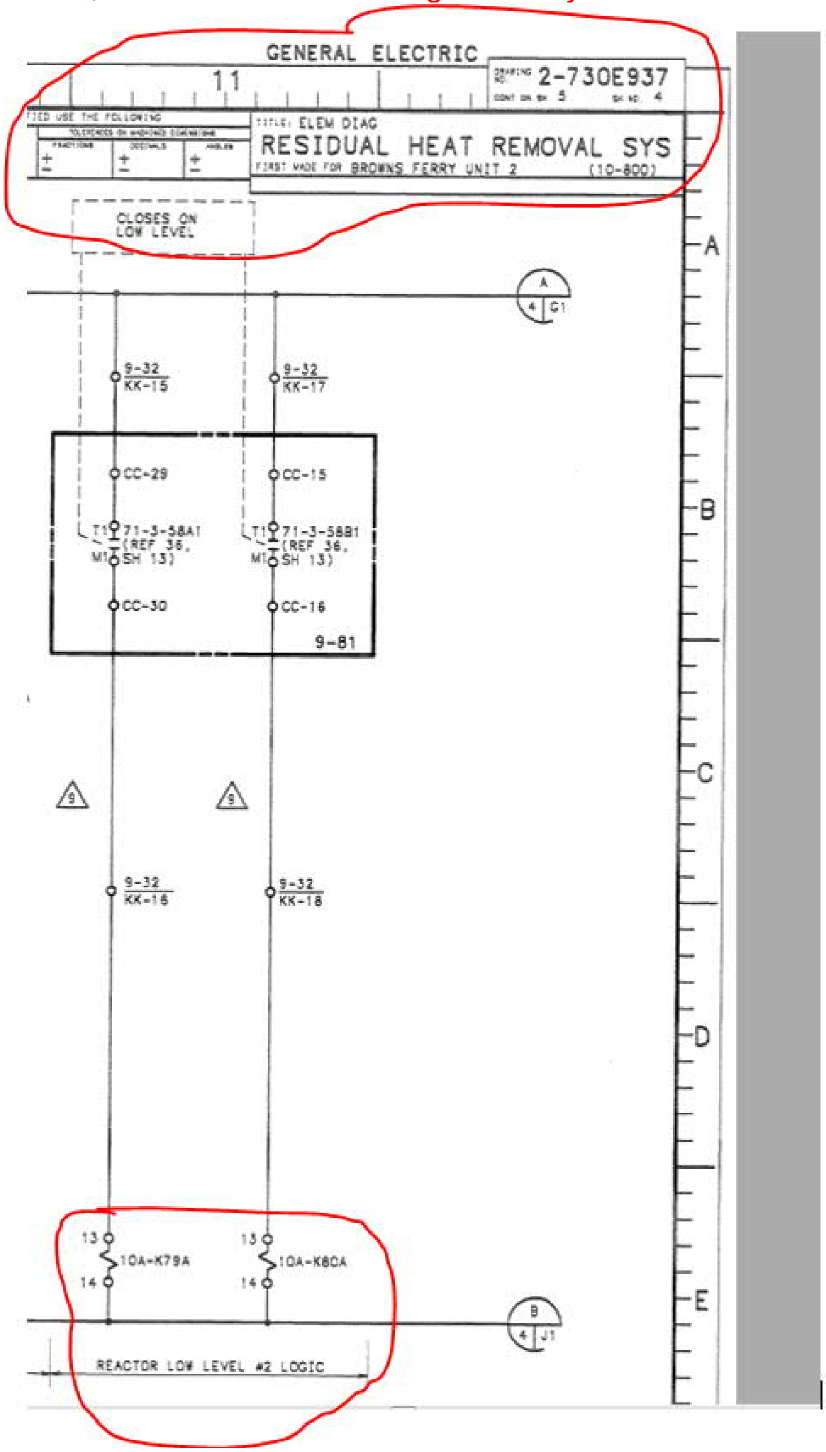
CAD MAINTAINED DRAWING

CCD

J

K

Excerpt from 2-730E937, Sheet 4: Illustrates origin of relay 10A-K79A and 80A from RHR



021	ADMIN	M.L.POOLE	WCHODGES	DPWalker 6-26-2009
REVISED PER RIMS MEMO R14 090624 102 (REF: BFPER 161548)				
REV	CHANGE REF	PREPARER	CHECKER	APPROVED / DATE

REVISIONS BY TVA

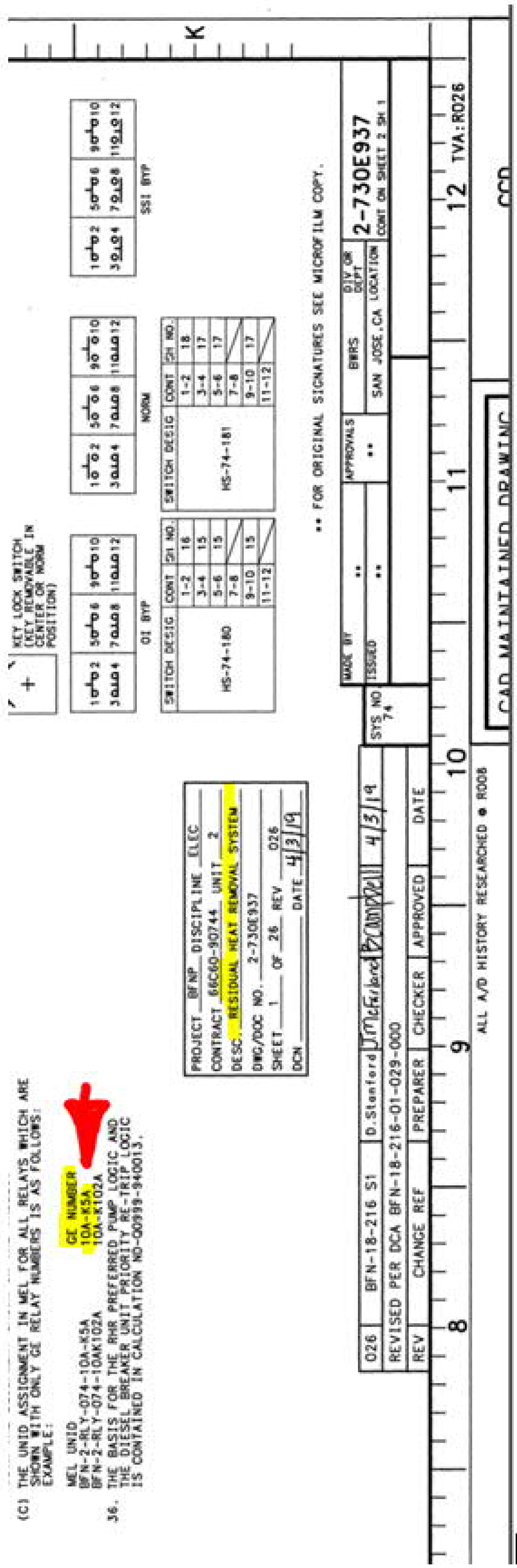
MADE BY P. TOOLE 3-25-70; ISSUED W.T. MERTEN 7-70; APPROVALS W.A.R. 6-5-70; BWRS SAN JOSE; DIV OR DEPT LOCATION; 2-730E937 CONT ON SHEET 5 SHEET 4

CAD MAINTAINED DRAWING — CCD

PROJECT BFNP DISCIPLINE ELEC
CONTRACT 66C60-90744 UNIT 2
DESC. RESIDUAL HEAT REMOVAL SYS
DWG/DOC NO. 2-730E937
SHEET 4 OF 26 REV 021
DCN DATE 6-26-2009
SYS NO. 74, 71

TVA:R021

Excerpt from 2-730E937, Sheet 1: Illustrates GE number designation for RHR relays (10A)



Excerpt from OPL171.040 Lesson Plan:

OPL171.040, Reactor Isolation Cooling (RCIC), Rev.# 31

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
D. Operational Summary	
1. Injection Mode	
a. Automatic operation with flow manually set into a flow controller and referenced to actual flow. The difference between the signals is processed through an electric hydraulic signal network to position the governor valve.	
b. Suction during the injection mode is either from the CST or torus while discharge is to the Feedwater System.	
2. Test Mode	
a. Manual start accomplished by opening the steam valve with the flow controller in automatic, manually set to the desired flow using the auto thumbwheel.	OE BFPER 99-011768 Operating Characteristic
b. Suction during the test mode is from the CST while discharge is to the CST (CST-to CST)	
c. Flow controller response in the test mode is much slower, i.e., takes longer to achieve the flow setpoint.	
(1) Δp across the test mode discharge piping path to the CST is much higher than the Δp across the injection piping path to the vessel.	
(2) When throttling the test Vlv, the discharge pressure change is very quick; however, the time to achieve the corresponding flow setpoint will be very long because the controller must affect turbine rpm to a much greater extent than if RPV press changed.	
d. GE SIL 623 identified that a single failure of the governor valve during surveillance test (CST-to-CST) could result in pump and discharge piping pressures exceeding ASME limits. This is only possible in the surveillance mode (CST-to-CST). Consequently GE recommends that the test valve be throttled to raise discharge pressure AFTER the turbine is already rolling instead of before the turbine is started.	GE SIL 623
e. TOE 3-00-071-11480 On 11/3/00 RCIC surveillance test, RCIC was operated for 4 minutes without minimum flow at 4800 rpm. This occurred because the discharge pressure indication in the control room had failed. It was subsequently determined that the actual discharge piping pressure approached 1506 psig. This is above the design piping pressure of 1500 psig.	BFPER 00-011480
3. System Automatic Initiation Signal	Obj. ILT.4.a. Obj. LOR .2.a. TP-9 NLOR - 2 NLO - 4
a. One-of-two twice on -45" vessel level (Low Level 2)	
b. Level Relays originate from the RHR logic	

Examination Outline Cross-reference:

G2.2.12 (10CFR 55.41.10)
Knowledge of surveillance procedures.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.2.12	
Importance Rating	3.7	------

Proposed Question: **# 69**

Which **ONE** of the following meets the requirements to be considered an "Infrequently Performed Test or Evolution" (IPTE) per NPG-SPP-10.6, Infrequently Performed Test or Evolution?

A. 1-SR-3.5.1.7(COMP), HPCI Comprehensive Pump Test

B. 2-SR-3.5.1.6(RHR I), Quarterly RHR System Rated Flow Test Loop I

C. 0-SR-3.8.1.9(A), Diesel Generator ‘A’ Emergency Unit 1 Load Acceptance Test

D. 0-GOI-300-4, Switchyard Manual, Switching Order to remove the West Point 500KV line

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that it involves a comprehensive surveillance on an Emergency Core Cooling System (ECCS), which is on the screening list for Infrequently Performed Test or Evolutions (IPTEs) (*See attached*). In this case, it is a quarterly surveillance, performed across three units and is not designated as an IPTE.

B INCORRECT**:** Incorrect but plausible in that it involves a comprehensive surveillance on an Emergency Core Cooling System (ECCS), which is on the screening list for IPTEs (*See attached*). In this case, it is a quarterly surveillance, performed across two subsystems, on three units; and is not designated as an IPTE.

C **CORRECT**: (*See attached*) In accordance with NPG-SPP-10.6, Infrequently Performed Tests or Evolutions, one of the distinguishing criteria for determining if a test / evolution is an IPTE is that if that test or evolution involves special, infrequently performed surveillance testing that involves complicated sequencing or placing the plant in an unusual configuration. The EDG Load Acceptance Test (LAT), by definition, is an IPTE. The Attachment 1 Table, used for identifying IPTEs supports this activity as an IPTE; along with excluding the three others listed in the question above. The EDG LAT not only challenges an emergency electrical power source, it has tentacles into several other major important systems such as Core Spray, RHR, RHRSW / EECW, and 480V Load Shed Logic. Additionally, this complex test is only conducted once per Refueling Cycle (every 24 months). Finally, it is designated in its own Surveillance Procedure directly as an IPTE on the cover page.

D INCORRECT: Incorrect but plausible in that switching is a very important, high profile, and safety significant activity. This would be considered “Routine” switching in accordance with 0-GOI-300-4, Switchyard Manual (*See attached*).

RO Level Justification: Tests the candidate's knowledge of the process for conducting special or infrequent tests at BFN. This question is rated as Memory due to strictly recalling specific criteria in determining if a test or evolution is an Infrequently Performed Test or Evolution (IPTE).

Technical Reference(s): NPG-SPP-10.6, Rev.1 (Attach if not previously provided)
0-SR-3.8.1.9(A), Rev. 12
1-SR-3.5.1.7(COMP), Rev. 35
2-SR-3.5.1.6(RHR I), Rev. 47
0-GOI-300-4, Rev. 112

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.078, Obj. 10 (As available)

Question Source:	Bank #	BFN 1909 #69	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

ILT 1909 Written Exam

69. Which **ONE** of the following meets the requirements to be considered an "Infrequently Performed Test or Evolution" (IPTE) per NPG-SPP-10.6, Infrequently Performed Test or Evolution?

A. 1-SR-3.5.1.7(COMP), HPCI Comprehensive Pump Test

B. 2-SR-3.5.1.6(RHR I), Quarterly RHR System Rated Flow Test Loop I

C. 0-SR-3.8.1.9(A), Diesel Generator ‘A’ Emergency Unit 1 Load Acceptance Test

D. 0-GOI-300-4, Switchyard Manual, Switching Order to remove the West Point 500KV line

Correct Answer: C

Excerpts from NPG-SPP-10.6:

NPG Standard Programs and Processes	Infrequently Performed Test or Evolutions	NPG-SPP-10.6 Rev. 0001 Page 8 of 19

3.2.4 Preparation of Infrequently Performed Tests or Evolutions (continued)

2. Validation of the procedure by walk-throughs and trials on a plant simulator when feasible.

3. Establishment of clearly delineated responsibilities for the plant staff during the test.

D. Responsible supervisors shall consider the temporary assignment of additional personnel under the direction of the shift manager to augment the shift personnel (for example assignment of an engineer or coordinator for the test or evolution, assignment of an additional senior reactor operator during control rod manipulations, or assignment of additional data takers when data is not readily available to the assigned shift at their normal shift location). The duties, authority, and responsibilities of extra personnel should be included on the organization chart and made clear in the test briefings. [C.3].

4.0 RECORDS

Records shall be processed in accordance with NPG-SPP-31.2, Records Management.

Test data packages shall include the following:

- Testing instructions or procedures and applicable attachments/appendices, when completed.

Forms or documents developed by other procedures/instructions (for example TVA 41083 (NPG-SPP-22.205-1 NPG Technical Pre-Job Briefing Checklist)) shall be processed per the requirements of that procedure.

4.1 QA Records

Test data packages affecting quality related systems or equipment as designated by the site Q-List.

4.2 Non-QA Records

Non-quality related test data packages

5.0 DEFINITIONS

Infrequently Performed Tests or Evolutions (IPTE) - Infrequently performed tests or evolutions that have the potential to significantly degrade the plant's margin of safety that warrant additional management oversight and control. The following criteria shall be used to identify these types of tests/evolutions:

- Tests or evolutions not specifically covered by existing normal or abnormal operating procedures.

NPG Standard Programs and Processes	Infrequently Performed Test or Evolutions	NPG-SPP-10.6 Rev. 0001 Page 9 of 19

5.0 DEFINITIONS (continued)

- Tests or evolutions that are seldom performed even though covered by existing normal or abnormal procedures (for example, plant startup after a prolonged outage or after any outage that involves significant changes to systems, equipment, or procedures related to the core, reactivity control, or reactor protection).
- Special, infrequently performed surveillance testing that involves complicated sequencing or placing the plant in an unusual configuration.
- Tests or evolutions that require the use of special test procedures in conjunction with existing procedures.

This definition shall be reviewed and updated as part of the normal review cycle, as specified within the review cadence, to ensure that subsequently identified tests or evolutions receive the additional management attention prior to their performance. [C.1]

Senior Manager - A direct report, or manager as designated by, the Plant Manager, General Manager Site Operations, or Site Vice President.

Technical Specification (TS) and/or ISFSI Certificate of Compliance (CoC) Acceptance Criteria - The surveillance requirement (including technical requirements, the offsite dose calculation manual, and fire protection report, if applicable) which the surveillance instruction or surveillance requirement (BFN only) either partially or completely fulfills as defined in the surveillance instruction or surveillance requirement.

Test coordinator -The individual assigned the direct responsibility for coordinating test performance.

Test Stoppage - A suspension of test activities that could affect or impact plant operations. Typically, the shift manager and the test coordinator will discuss the expected test duration and the probability of routine breaks (such as lunches); however, a suspension that has not been planned and/or agreed to between the shift manager and the test coordinator before beginning the test is considered a test stoppage.

6.0 REFERENCES

6.1 Requirements Documents

A. 10CFR50, Appendix B, Criterion XI

B. TVA-NQA-PLN89-A, Nuclear Quality Assurance Plan (NAQP), Sections 9.4, 9.7, 10.2, and 11.2

6.2 Developmental References

A. NPG-SPP-01.2 Administration of Site Technical Procedures

B. NPG-SPP-06.3 Pre-/Post-Maintenance Testing

C. NPG-SPP-06.6 Inspection Program

NPG Standard Programs and Processes	Infrequently Performed Test or Evolutions	NPG-SPP-10.6 Rev. 0001 Page 11 of 19

Attachment 1
(Page 1 of 4)

GUIDELINES FOR IDENTIFYING INFREQUENTLY PERFORMED TESTS OR EVOLUTIONS (IPTE)

1.0 DEPARTMENT APPLICABILITY SCREENING

Procedure Number/Work Order Number:	
Procedure/Evolution Title:	

			YES	NO
1.	Is equipment essential to any of the following safety functions required for the current plant operating conditions affected in any potentially adverse way?			
	a.	**Reactivity and reactor control**: Any activity that may adversely affect a system or component that could change reactor power. This may include maintenance of turbine pressure regulator controls or RCS control valve circuitry. Consider operation of demineralizers, extended operations at low power, and portions of startup and shutdown that may pose a significant risk to safety margins. PWR Only: Consider conditions which could cause an inadvertent dilution.		
	b.	**Reactor Protection System**: Any activity that may adversely affect the reactor protection system or its components. Consider portions of startup or shutdown that challenge operation of the components essential to this function.		
	c.	**Decay Heat Removal**: Any activity that adversely affects decay heat removal systems, including surveillance testing.		
	d.	**ECCS Operability:** Any activity that may prevent actuation of or impair the ability of required systems from performing their ECCS function.		
	e.	**Reactor Coolant System Inventory**: Any work/testing/operations that may have the potential to unintentionally reduce reactor coolant system level. This includes lines or systems that connect below the reactor coolant system level required to be maintained for current plant conditions. Consider activities that could impact the Reactor Coolant Pressure boundary including repair using temporary plug or freeze seal installations. In addition, evolutions to enter into lowered inventory would be included.		
	f.	**Containment Integrity**: Any activity that challenges containment integrity. (BWR Primary and Secondary Containment)		
	g.	**Electric Power Availability**: Any activity that challenges emergency or normal electrical power supply to equipment required for the current operating condition of the plant.		

Excerpt from 0-SR-3.8.1.9(A):

Browns Ferry Nuclear Plant

Unit 0

Surveillance Procedure

0-SR-3.8.1.9(A)

Diesel Generator A Emergency Unit 1 Load Acceptance Test

Revision 0012

Quality Related

Level of Use: Continuous Use

Complex Infrequently Performed Test or Evolution

Excerpt from 2-SR-3.5.1.6(RHR I):

Browns Ferry Nuclear Plant

Unit 2

Surveillance Procedure

2-SR-3.5.1.6(RHR I)

Quarterly RHR System Rated Flow Test Loop I

Revision 0047

Quality Related

Level of Use: Continuous Use

Level of Use or Other Information: Key Number P2337

Excerpt from 1-SR-3.5.1.7:

Browns Ferry Nuclear Plant

Unit 1

Surveillance Procedure

1-SR-3.5.1.7(COMP)

HPCI Comprehensive Pump Test

Revision 0035

Quality Related

Level of Use: Continuous Use

Level of Use or Other Information: Key Number P1352

Effective Date: 03-15-2019

Responsible Organization: OPS, Operations

Prepared By: Michael Teggins

Approved By: Walter Miller

Excerpt from 0-GOI-300-4:

Browns Ferry Nuclear Plant

Unit 0

General Operating Instruction

0-GOI-300-4

Switchyard Manual

Revision 0112

Quality Related

Level of Use: Reference Use

Examination Outline Cross-reference:	Level	RO	SRO
G2.3.13 (10CFR 55.41.12)	Tier #	3	------
Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	Group #	------	------
	K/A #	G2.3.13	
	Importance Rating	3.4	------

Proposed Question: **# 70**

With respect to Area Radiation Monitors (ARMs), which **ONE** of the following completes the statements below?

ARMs are individual detectors that provide indications and alarms in the Main Control Room (MCR) with an amber 'HIGH' light that will **FIRST** illuminate when the MAXIMUM **(1)** radiation value has been reached.

(2) of the ARMs on MCR Panel 9-11 **REQUIRE** EOI entry.

A. (1) SAFE
(2) All

B. (1) SAFE
(2) NOT all

C. (1) NORMAL
(2) All

D. (1) NORMAL
(2) NOT all

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if candidate confuses MAX SAFE with MAX NORMAL as they are related to the high radiation ALARM value and indication pertaining to Area Radiation Monitors (ARMs). Second part is incorrect but plausible in that the majority of the ARMs on Panel 9-11 in MCR do require EOI-3 entry as indicated with a orange EOI flag.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT: First part is correct (*See D*). Second part is incorrect but plausible (*See A*).

D **CORRECT:** (*See attached*) In accordance with 2-ARP-9-3A, Window 22, alarms for specific Reactor Building Area Radiation Monitors (ARMs) are indicated on Control Room Panel 2-9-11. These ARMs will have an amber 'HIGH' light that will FIRST illuminate when reaching their MAX Normal (ALARM) value. For second part, some of the ARMs on Panel 9-11 do **REQUIRE** EOI entry. EOI-3 Table SC-2, Secondary Containment Radiation does list the ARMs requiring EOI-3 entry if above MAX NORMAL. Table SC-2 delineates between the ARMs reaching their MAX NORMAL value versus MAX SAFE. Upon reaching an ARM's MAX SAFE setpoint, Operators will announce 'MAX SAFE' has been reached. This is required to assist the Unit Supervisor in EOI-3 critical decision making.

RO Level Justification: Tests the candidate's knowledge of radiological safety procedures pertaining to Licensed Operator duties specifically for Radiation Monitoring Systems. The fixed Area Radiation Monitors (ARMs), associated annunciators, and the required actions when in alarm as it relates to Emergency Operating Instructions are vital for safe operations of the plant. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): 2-ARP-9-3A, Rev. 55 (Attach if not previously provided)
OPL171.034, Rev. 12
2-EOI-3, Rev. 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.034 Obj. 4 (As available)

Question Source:			
	Bank #		
	Modified Bank #	BFN 1909 #71	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Copy of Bank Question:

ILT 1909 Written Exam

71. Which **ONE** of the following completes the statement below?

The Area Radiation Monitors (ARMs) are individual detectors that provide indications and alarms in the Main Control Room of ___(1)___ radiation levels from selected plant locations and the amber 'HIGH' light will **FIRST** illuminate when the MAX ___(2)___ radiation value has been reached.

A. **(1)** neutron
(2) SAFE

B. **(1)** neutron
(2) NORMAL

C. **(1)** gamma
(2) SAFE

D. **(1)** gamma
(2) NORMAL

Correct Answer: **D**

Excerpts from 2-ARP-9-3A:

BFN Unit 2	Panel 9-3 2-XA-55-3A	2-ARP-9-3A Rev. 0055 Page 38 of 60

RX BLDG AREA RADIATION HIGH

2-RA-90-1D

22

(Page 1 of 3)

Sensor/Trip Point:

RI-90-4A	RI-90-24A	For setpoints
RI-90-9A	RI-90-25A	REFER TO
RI-90-13A	RI-90-26A	2-SIMI-90B.
RI-90-14A	RI-90-27A	
RI-90-20A	RI-90-28A	
RI-90-21A	RI-90-30A	
RI-90-R22A	RI-90-29A	
RI-90-23A		

Sensor Location:			
	RE-90-4	MG set area	Rx Bldg El 639' R-10 S-LINE
	RE-90-9	Clean-up System	Rx Bldg El 621' R-9 T-LINE
	RE-90-13	North Clean-up Sys	Rx Bldg El 593' R-9 P-LINE
	RE-90-14	South Clean-up Sys	Rx Bldg El 593' R-9 S-LINE
	RE-90-20	CRD-HCU West	Rx Bldg El 565' R-9 R-LINE
	RE-90-21	CRD-HCU East	Rx Bldg El 565' R-13 R-LINE
	RE-90-22	TIP Room	Rx Bldg El 565' R-12 P-LINE
	RE-90-23	TIP Drive	Rx Bldg El 565' R-12 P-LINE
	RE-90-24	HPCI Room*	Rx Bldg El 519' R-14 U-LINE
	RE-90-25	RHR West	Rx Bldg El 519' R-8 U-LINE
	RE-90-26	Core Spray-RCIC	Rx Bldg El 519' R-9 N-LINE
	RE-90-27	Core Spray	Rx Bldg El 519' R-14 N-LINE
	RE-90-28	RHR East	Rx Bldg El 519' R-14 U-LINE
	RE-90-30	Fuel Storage Pool	Rx Bldg El 664' R-12 P-LINE
	RE-90-29	Suppression Pool	Rx Bldg El 519' R-14 U-LINE

Probable Cause:

A. Radiation levels have risen above alarm setpoint.

B. Dry Cask Storage activities in progress (activities could affect rad levels sensed by 2-RE-90-30)

NOTE

Due to the location of the Rad Monitor in relation to the Test line in the HPCI Quad, the HPCI Room Rad Alarm may be received when the HPCI Flow test is in progress.

C. HPCI Flow Rate Surveillance in Progress.

Automatic Action: None

Continued on Next Page

BFN Unit 2	Panel 9-3 2-XA-55-3A	2-ARP-9-3A Rev. 0055 Page 39 of 60

RX BLDG AREA RADIATION HIGH 2-RA-90-1D, Window 22
(Page 2 of 3)

Operator Action:

A. DETERMINE area with high radiation level on Panel 2-9-11. (Alarm on Panel 2-9-11 will automatically reset if radiation level lowers below setpoint.)

B. IF Dry Cask storage activities are in progress, THEN NOTIFY CASK Supervisor.

C. IFalarm is from the HPCI Room while Flow testing is performed, NOTIFY personnel at the HPCI Quad to validate conditions.

D. NOTIFY RAD PRO.

BFN Unit 2	Panel 9-3 2-XA-55-3A	2-ARP-9-3A Rev. 0055 Page 40 of 60

RX BLDG AREA RADIATION HIGH 2-RA-90-1D, Window 22
(Page 3 of 3)

Operator Action: (Continued)

J. For all radiation indicators except FUEL STORAGE POOL radiation indicator, 2-RI-90-30, ENTER 2-EOI-3 Flowchart.

K. REFER TO 2-AOI-79-1 or 2-AOI-79-2 if applicable.

L. EVALUATE equipment associated with this alarm to determine compensatory actions required to maintain REP function. REFER TO NPG-SPP-18.3.5.

References: 2-45E620-3 2-45E610-90-1 GE 0-730E356-1
TVA Calc NDQ00902005008/EDC63693

Excerpt from 2-EOI-3: illustrating Radiation entry conditions as it relates to Table SC-2

Any Secondary Cntmt area radiation lvl above Max Normal value of Table SC-2

Table SC-2
Secondary Cntmt Area Radiation

Area	Applicable Radiation Indicators	Max Normal Value mR/hr	Max Safe Value mR/hr	Potential Isolation Sources
RHR sys I pumps	90-25A	Alarmed	1000	FCV-74-47, 48
RHR sys II pumps	90-28A	Alarmed	1000	FCV-74-47, 48
HPCI room	90-24A	Alarmed	1000	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	90-26A	Alarmed	1000	FCV-71-2, 3, 39
CS sys II pumps	90-27A	Alarmed	1000	None
Top of torus General area	90-29A	Alarmed	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB el 565 W	90-20A	Alarmed	1000	FCV-69-1, 2, 12 SDV vents & drains
RB el 565 E	90-21A	Alarmed	1000	SDV vents & drains
RB el 565 NE	90-23A	Alarmed	1000	None
TIP room	90-22A	Alarmed	100,000	TIP ball valve
RB el 593	90-13A, 14A	Alarmed	1000	FCV-74-47, 48
RB el 621	90-9A	Alarmed	1000	FCV-43-13, 14
Recirc MG sets	90-4A	Alarmed	1000	None
Refuel floor	90-1A, 2A 3A	Alarmed	1000	None

2-EOI-3 Page 1 of 1

SECONDARY CONTAINMENT CONTROL
UNIT 2
BROWNS FERRY
NUCLEAR PLANT

Rev. 17

Excerpt from OPL171.034 Lesson Plan:

OPL171.034 , Area Radiation Monitoring System, Rev# 12

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
C. ARM System Instrumentation	
1. Local and remote monitoring points are listed on TP-8 by station number, location description, and instrument scale used in that area.	
2. Each monitoring point consists of a sensor/converter, auxiliary unit & indicator/trip unit.	
Note: Station Number sequence on Pnl 9-11is typically from left to right, top to bottom. EOI-3 instruments are designated with an orange EOI tag.	ILT/LOR –Obj 2 NLO/NLOR-Obj 2
3. ARM alarm setpoints vary depending on detector location and area background radiation, but typically ~ 10 mr/hr is used in low range instrument areas	
4. When the local sensor & converter senses an upscale (high) radiation condition and the trip setpoint has been exceeded the following occurs:	ILT/LOR –Obj 4 NLO/NLOR-Obj 4
a) The local auxiliary unit audibly alarms.	
b) High amber light illuminates on local aux unit	
c) Local Meter indicates radiation level	
d) Remote (control room) indicator trip unit (I/T) Receives input directly from the local sensor &converter which picks up the upscale trip relay and drives the meter indication:	
e) The Upscale trip relay Illuminates the amber (high) light, sends a signal to the local auxiliary unit, and sends a signal to the MCR annunciator. Auto resets when clear.	

Examination Outline Cross-reference:

G2.3.4 (10CFR 55.41.12)

Knowledge of radiation exposure limits under normal or emergency conditions.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.3.4	
Importance Rating	3.2	------

Proposed Question: **# 71**

In accordance with EPIP-15, Emergency Exposure, which **ONE** of the following completes the statement below?

Site Emergency Director (SED) authorization is required when a worker will exceed _________ in order to save a life or avoid extensive exposures to large populations.

A. 2 rem

B. 5 rem

C. 10 rem

D. 25 rem

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that in accordance with NPG-SPP-5.1, Radiological Controls, 2 REM per year is the administrative limit for yearly exposure. In accordance with EPIP-15, while the exposure to workers should be limited to prevent exceeding 2 REM, during an emergency situation exceeding that limit may be justified. The Site Emergency Director (SED) is not required to approve exceeding a dose of 2 REM for a worker.

B INCORRECT: Incorrect but plausible in that in accordance with 10 CFR 20.1201, the dose limit for adults is 5 REM per year. In accordance with EPIP-15, while the exposure to workers should be limited to prevent exceeding the 10 CFR 20.1201 limit of 5 REM, during an emergency situation exceeding that limit may be justified. The SED is not required to approve exceeding a dose of 5 REM for a worker.

C INCORRECT: Incorrect but plausible in that in accordance with EPIP-15, the limit for protecting valuable property is 10 REM. There are no provisions in EPIP-15 for exceeding 10 REM to protect valuable property.

D **CORRECT**: (*See Attached*) In accordance with EPIP-15, the SED to is required to authorize individuals to exceed 25 REM for life saving activities or to avoid extensive exposures to large populations.

RO Level Justification: Tests the candidate's knowledge of emergency radiation exposure limits. This question is rated as Memory due to the requirement to strictly recall facts related to radiation exposure limits.

Technical Reference(s): EPIP-15, Rev.13 (Attach if not previously provided)
NPG-SPP-05.1, Rev.11
10 CFR 20.1021

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.053, Obj. 10 (As available)

Question Source:	Bank #	
	Modified Bank #		(Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Excerpt from NPG-SPP-05.1:

NPG Standard Programs and Processes	Radiological Controls	NPG-SPP-05.1 Rev. 0011 Page 15 of 53

3.2.4 Exposure Control (continued)

TABLE 1

ADMINISTRATIVE DOSE LEVEL PROGRAM

Dose Equivalent (Rem)	Requirement	Authorization to Exceed (signatures)
Up to 0.5 TEDE (or 1.5 LDE, 5.0 SDE and 5.0 SDE,ME)	Statement of current year dose and previous years dose signed by individual	Not applicable
Up to 2.0 TEDE (or 12 LDE, 40 SDE and 40 SDE ME) all sources	NRC FORM-4 or equivalent to document current year and previous years dose equivalent	Not applicable
To exceed 2.0 TEDE all sources	Same as above	RPM/RSO
To exceed 3.0 TEDE all sources	Same as above	RPM/RSO, and Plant Manager[1]
To exceed 4.0 TEDE (or 12 LDE, 40 SDE and 40 SDE ME) all sources	Same as above	RPM/RSO, Plant Manager[1], and Site VP[2]
To exceed 5.0 TEDE[3] all sources	Form-4 information must be verified and a Planned Special Exposure initiated in Accordance with RCTP-114	RPM/RSO, Plant Manager[1], and Site VP[2]

[1] At non-nuclear plant sites, this will be the RSO's immediate supervisor.

[2] At non-nuclear plant sites, this will be the applicable TVA VP.

[3] Authorizations for a planned special exposure will only be considered in an exceptional situation when alternatives that might avoid the dose estimated to result from the planned special exposure are unavailable or impractical.

Excerpt from 10 CFR 20.1021:

Subpart C—Occupational Dose Limits

Source: 56 FR 23396, May 21, 1991, unless otherwise noted.

§ 20.1201 Occupational dose limits for adults.

(a) The licensee shall control the occupational dose to individual adults, except for planned special exposures under § 20.1206, to the following dose limits.

(1) An annual limit, which is the more limiting of—

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin of the whole body, and to the skin of the extremities, which are:

(i) A lens dose equivalent of 15 rems (0.15 Sv), and

(ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

(b) Doses received in excess of the annual limits, including doses received during accidents, emergencies, and planned special exposures, must be subtracted from the limits for planned special exposures that the individual may receive during the current year (see § 20.1206(e)(1)) and during the individual's lifetime (see § 20.1206(e)(2)).

Excerpts from EPIP-15:

BFN Unit 0	EMERGENCY EXPOSURE	EPIP-15 Rev. 0013 Page 6 of 11

3.3 Guidance for All Emergency Dose Limits (continued)

G. Personnel shall not enter any area where dose rates are unknown or not measurable with instruments and dosimetry immediately available.

NOTE

The value below corresponds to the ratio of external (measured) dose rate to estimate TEDE dose, in accordance with default values in TVA's Dose Assessment model. When accident specific nuclide assessment are available, more definitive dose assessments should be performed to adjust the correction factors.

H. Until isotopic assessments of airborne radioactivity are available, an administrative correction factor of 2 should be used to estimate TEDE exposures in airborne activity areas:

Estimated TEDE = Dosimeter Reading × 2

3.4 Dose Limits for Workers During Emergencies

A. Doses to all workers during emergencies should, to the extent practicable be limited to 10 CFR 20.1201 limits. There are, however, some emergency situations for which higher emergency exposures may be justified. Whenever these situations are justified and ALARA considerations have been evaluated the following limits can be administered.

B. RP considers the to-date annual accrued dose to individuals when establishing the maximum dose limits for workers during emergencies. The to-date annual accrued dose would be subtracted from the applicable emergency dose limit to determine the authorized allowable dose for the emergency.

C. Dose Limits for the Protection of Valuable Property

Dose Limit (rem)	Receptor
10	Whole Body (TEDE)
30	Lens of the Eye
100	All Other Organs

BFN Unit 0	EMERGENCY EXPOSURE	EPIP-15 Rev. 0013 Page 7 of 11

3.4 **Dose Limits for Workers During Emergencies (continued)**

D. Dose Limits for Lifesaving Activities and the Protection of Large Populations

Dose Limit (rem)	Receptor
25	Whole Body (TEDE)
75	Lens of the Eye
250	All Other Organs

E. Dose Limits Greater than 25 Rem for Lifesaving Activities or to Avoid Extensive Exposures to Large Populations

1. Situations may occur in which a dose in excess of 25 rem would be required in order to carry out lifesaving operations or to avoid extensive exposures to large populations. It is not possible to prejudge the risk that one person should be allowed to take to save the life of others.

2. Personnel made knowledgeable of the risks involved with radiation exposures through training or briefings utilizing the information contained within Appendix A and selected on a voluntary basis may be allowed to exceed the 25 rem emergency dose exposure limit for the purpose of lifesaving activities or to avoid extensive exposures to large populations.

Examination Outline Cross-reference:	Level	RO	SRO
G2.3.5 (10CFR 55.41.11) Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	Tier #	3	------
	Group #	------	------
	K/A #	G2.3.5	
	Importance Rating	2.9	------

Proposed Question: **# 72**

Determine which **ONE** of the following indications would be observed for an Area Radiation Monitor (ARM) when the detector is **SATURATED** in a radiation field.

A. The indicator remains upscale **BUT** all trips drop out.

B. The indicator drops downscale **AND** the trips drop out except for the downscale trip.

C. The indicator remains upscale **AND** the upscale trip remains in due to the pegging circuit.

D. The indicator drops downscale **BUT** the upscale trip remains in due to the pegging circuit.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that the upscale alarm function remains active, the candidate could assume that the pegging circuit only retains an upscale indication.

B INCORRECT: Incorrect but plausible in that If the candidate doesn't realize the pegging circuit is an actual component that functions. Neither downscale or a downscale trip are expected in a saturated field.

C **CORRECT:** The system is equipped with a protective electronic pegging circuit installed for the adverse GM detector characteristic known as saturation. Once the counts appear to be a solid count to the point it appears that there is no count, the detector would naturally indicate no counts. The installed pegging circuit prevents this false downscale indication by providing the indication as upscale with the associated alarm and indication response. A transistor current amplifier (Holding Circuit), controlled by the GM tube DC output, furnishes a meter pegging current when the radiation intensity exceeds the point of saturation.

D INCORRECT: Incorrect but plausible in that if the candidate assumes that the pegging circuit only retains an upscale indication while a downscale is not expected in a saturated field.

RO Level Justification: Tests the candidate's ability to use radiation monitoring systems, specifically Area Radiation Monitors and interpret their indications. This question is rated as Memory due to the requirement to strictly recall procedural facts.

Technical Reference(s): OPL171.034, Rev. 12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.034 Obj. 4 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.034-03 001 #963	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

963. OPL171.034-03 001

Determine which ONE of the following indications would be observed for an Area Radiation Monitor (ARM) ***when the detector is saturated*** in a radiation field.

A✓ The indicator remains upscale and the upscale trip remains in due to the pegging circuit.

B. The indicator drops downscale and the trips drop out except for the downscale trip.

C. The indicator remains upscale but all trips drop out.

D. The indicator drops downscale but the upscale trip remains in due to the pegging circuit.

Correct Answer: A

Excerpt from OPL171.034 Lesson Plan:

1. Geiger-Mueller Tube (Sensor)

a) The GM tube has an internal electric field which draws ions toward the electrodes. ~600V DC is applied across the anode and cathode from the external power supply — ILT/LOR –Obj 2 NLO/NLOR-Obj 2

b) Gamma radiation entering a GM tube interacts with the atoms of the detector gas, releasing ions to the detector volume.

c) The ions then cause secondary ionization in the contained gas.

d) As radiation intensity rises, the number of ionizations and ion pairs increases. Nominal sensitivity is 3.7×10^{-10} amps/R/hr.

e) The positive and negative ions, created by incident radiation, move toward the anode (+) and cathode (-) of the GM tube.

f) The charge collection on the anode causes a voltage change in the circuit, and a current flow (pulse) results.

g) If the interval between pulse generating events becomes short enough, the GM tube will go into steady DC conduction, "saturation," and stop producing output pulses — ILT/LOR –Obj 3 NLO/NLOR-Obj 3

h) When this happens, the external circuit that is counting and integrating pulses responds as though the events were only one pulse which would cause a decreased meter reading at the point of saturation

i) This possible source of error is overcome by using a high-pass filter in the GM tube circuit which detects this condition and provides a saturation signal which represents the degree of ionization of the GM tube

j) A transistor current amplifier (Holding Circuit), controlled by the GM tube DC output, furnishes a meter pegging current when the radiation intensity exceeds the point of "saturation. — ILT/LOR –Obj 4 NLO/NLOR-Obj 4

k) This assures proper indication. Without pegging circuit, saturation would indicate downscale. With pegging circuit, saturation provides upscale indication.

Examination Outline Cross-reference:

G2.4.27 (10CFR 55.41.10)
Knowledge of “fire in the plant” procedures.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.4.27	
Importance Rating	3.4	------

Proposed Question: **# 73**

A fire has been reported in the Unit 1 Reactor Building and 0-AOI-26-1, Fire Response, has been entered.

Which **ONE** of the following completes the statements below?

In accordance with the AOI, the AUOs will **INITIALLY** report to ___**(1)**___.

If a Standby Gas Train is operated to remove smoke, that train ___**(2)**___.

A. (1) their assigned Control Room
(2) remains OPERABLE

B. (1) their assigned Control Room
(2) will be considered INOPERABLE

C. (1) the Incident Commander for damper realignment
(2) remains OPERABLE

D. (1) the Incident Commander for damper realignment
(2) will be considered INOPERABLE

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). The second part is incorrect but plausible in that in accordance with 0-AOI-26-1, Fire Response and 0-OI-65, Standby Gas Treatment System, if used for smoke removal, renders Standby Gas INOPERABLE. Unit Operators are not normally tasked with Technical Specifications or determining the operability of equipment, however the note in the AOI mentions the operability concern for the Standby Gas trains when using them for smoke removal.

B **CORRECT**: (*See attached*) In accordance with 0-AOI-26-1, the AUOs report to their assigned Control Room and all other AUOs report to the Unit 2 Control Room. The second part is correct in that in accordance with 0-AOI-26-1 and 0-OI-65, using Standby Gas for smoke removal will render the Standby Gas INOPERABLE.

C INCORRECT: The first part is incorrect but plausible in that 0-AOI-26-1 directs that dampers be aligned for smoke removal when directed by the Incident Commander, but AUOs are not directed in the procedure to report to the Incident Commander. The second part is incorrect but plausible *(See A).*

D INCORRECT: The first part is incorrect but plausible *(See C).* Second part is correct *(See B).*

RO Level Justification: Tests the candidate's knowledge of the plant procedures for a fire. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to emergency plant conditions.

Technical Reference(s): 0-OI-65, Rev. 55 (Attach if not previously provided)

0-AOI-26-1, Rev. 21

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.074 Obj. 2 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.074-01 028 #1987
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

1987. OPL171.074-01 028

A fire has been reported in Unit 3 Reactor Building. 0-AOI-26-1, Fire Response, has been entered.

When directed, AUOs will report to ___(1)___.

If a SBGT train is run to remove smoke, the train will be considered ___(2)___.

A. 1. their assigned Control Room(s), all other AUOs will report to Unit 2 MCR
   2. operable

B✓ 1. their assigned Control Room(s), all other AUOs will report to Unit 2 MCR
   2. inoperable

C. 1. the Fire Brigade Leader to align ventilation system dampers/fans as necessary
   2. operable

D. 1. the Fire Brigade Leader to align ventilation system dampers/fans as necessary
   2. inoperable

Excerpts from 0-AOI-26-1:

BFN Unit 0	Fire Response	0-AOI-26-1 Rev. 0021 Page 6 of 77

4.2 Subsequent Actions (continued)

NOTES
1) The Shift Manager will remain in communication with the Incident Commander and reference 0-FSS-001 for applicability based on the severity of the fire.
2) Fire Safe Shutdown procedures contains Tables which assist in depicting the credited plant/unit equipment and instrumentation for that specific Fire Area.
3) The AUOs are assembled in the Control Rooms to ensure FSS Recovery Actions can be completed with maximum efficiency (the exception would be FSS-16-2, Main Control Room Abandonment, which requires the ERs to assemble at the backup control panels for their respective Unit).
4) To ensure that in the event of a severe fire, containment pressure is not vented below that which is needed to maintain RHR pump NPSH, maintain 1(2,3)-FIC-84-19 in normal position of Manual and "0" scfm.

[4] **IF** directed by the Unit Supervisor, **THEN**

PERFORM the following:

[4.1] **NOTIFY** AUOs to report to their assigned Control Room(s), all other AUOs will report to Unit 2 MCR (ERs will report to the Backup Control Room Panels for MCR Abandonment). ☐

[4.2] **REVIEW** applicable FSS for the fire area. ☐

[4.3] **DISTRIBUTE** FSS Attachment/Sections (Recovery Actions for the Fire Area) to assigned Emergency Responders, as directed by the SROs for either the EOIs or the FSSs. ☐

BFN Unit 0	Fire Response	0-AOI-26-1 Rev. 0021 Page 8 of 77

4.2 Subsequent Actions (continued)

[8] **IF** directed by the US, **THEN**

PERFORM the following:

[8.1] **DIRECT** AUOs to standby until notification is received that fire is out or notification to perform assigned FSS Attachment and Sections. ☐

NOTES

1) The ventilation systems can be restarted for smoke removal provided a PCIS Group 6 isolation is **NOT** required and radioactive release limits will **NOT** be exceeded.
2) It may be necessary to wear self-contained breathing apparatus to manually align ventilation dampers.
3) Damper locations are given in 1(2)(3)-OI-30B and 1(2)(3)-OI-30A valve lineup checklists.
4) If SBGT train is run to remove smoke, the train will be technically inoperative thereafter. System Engineering should be contacted to determine tests and SI/SRs necessary to re-establish operability. Since the SGT System is common to all three units, the Unit Operator on each unit should stay fully aware of the system status and condition at all times. The Unit Operators should communicate to each other any change in system status.
5) The Fire Brigade Leader will coordinate changes to ventilation system alignment through the Incident Commander to the Control Room.
6) The Emergency Responders which are not being used for EOI Appendices or FSS Attachments can be utilized to align ventilation system dampers/fans as necessary.

[9] **WHEN** directed by Incident Commander or Shift Manager, **THEN**

- **MANUALLY ALIGN** ventilation system dampers as necessary and ☐
- **PLACE** ventilation systems in service to remove smoke from affected area. ☐

Excerpt from 0-OI-65:

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0055 Page 12 of 42

3.0 PRECAUTIONS AND LIMITATIONS (continued)

HH. Mechanical locking devices are installed (under valve cover plate - two hex head screws) on TRAIN A(B)(C) DECAY HEAT DISCH DAMPERs, 0-DMP-065-0002(0024)(0066), to prevent valve motion. Position is verified by valve position striker arm contact with associated limit switch stop on opposite side of the valve actuator plate.

II. Using SGT System for smoke removal causes the affected train(s) to be inoperative. **REFER TO** Tech Spec 3.6.4.3.

Examination Outline Cross-reference:	Level	RO	SRO
600000 (APE 24) Plant Fire On Site / 8	Tier #	1	------
AK1.01 (10CFR 55.41.10)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to Plant Fire On Site:	K/A #	600000AK1.01	
• Fire Classifications by type	Importance Rating	2.5	------

Proposed Question: **# 74**

An electrical fire is reported in 'A' Diesel Generator room.

Which **ONE** of the following completes the statement below in accordance with the Fire Protection Requirements Manual?

The installed fire suppression system that is effective for this classification of fire and available in this room is __________.

A. Halon agent - Manual initiation **ONLY**

B. Carbon Dioxide agent - Manual initiation **ONLY**

C. Halon agent - Manual **OR** Automatic initiation

D. Carbon Dioxide agent - Manual **OR** Automatic initiation

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that we do have Halon extinguishers at BFN; but, we do not have installed Halon Fire Suppression Systems, either automatic or manual.

B INCORRECT: Plausible in that we do have installed Carbon Dioxide Fire Suppression Systems at BFN in the Diesel Generator Buildings for all three units at BFN; but they can be operated in Manual or Automatic modes.

C INCORRECT: Plausible in that we do have Halon extinguishers at BFN; but, we do not have installed Halon Fire Suppression Systems, either automatic or manual.

D **CORRECT:** *(See attached)* In accordance with the Fire protection Requirements Manual, we do have installed Carbon Dioxide Fire Suppression Systems at BFN in the Diesel Generator Buildings for all three units at BFN, and they can be operated in Manual or Automatic modes.

RO Level Justification: Tests the candidate's knowledge of the operational implications of certain classifications of fires and the viable extinguishing agents that can be used on those certain classification of fires. This question is rated as Memory due to the fact that it requires the strict recall of facts.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): NFPA 805 Fire Protection Requirements Manual, Rev. 12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: N/A (As available)

Question Source:

Bank #	BFN 1804 #19	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam 2018

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 X

55.43

Comments:

Copy of Bank Question:

ES-401 Sample Written Examination Question Worksheet Form ES-401-5

Examination Outline Cross-reference:	Level	RO	SRO
600000 (APE 24) Plant Fire On Site / 8	Tier #	1	------
AK1.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to PLANT FIRE ON SITE:	K/A #	600000AK1.01	
• Fire classifications by type	Importance Rating	2.5	------

Proposed Question: **# 19**

An electrical fire is reported in 'A' Diesel Generator room.

Which ONE of the following completes the statement below in accordance with the Fire Protection Requirements Manual?

The installed fire suppression system that is effective for this classification of fire and available in this room is __________.

A. Halon agent - Manual initiation **ONLY**

B. Carbon Dioxide agent - Manual initiation **ONLY**

C. Halon agent - Manual **OR** Automatic initiation

D. Carbon Dioxide agent - Manual **OR** Automatic initiation

Proposed Answer: D

Excerpt from NFPA 805 Fire Protection Requirements Manual:

Rooms, and for purging the generator-hydrogen systems. The second storage and refrigeration unit (6-ton capacity) is located in the Unit 3 Diesel Generator Building.

There are two master control valves (a valve supplying CO2 to several hazard control valves) located at the storage unit in the Units 1 and 2 Diesel Generator Building. One valve supplies CO2 to hazard control valves in the Turbine, Control, and Service Buildings, and the other valve supplies those in the Units 1 and 2 Diesel Generator Building. The storage unit in the Unit 3 Diesel Generator Building has one master control valve which supplies CO2 to hazard control valves in the Unit 3 Diesel Generator Building. These master control valves are normally closed, and open upon a fire signal from any of the protected hazards to charge the supply headers.

Those portions of the CO2 systems in the Units 1 and 2 Diesel Generator Building and Unit 3 Diesel Generator Building are seismically designed. Those portions of the CO2 systems in the Control Building are also seismically designed to prevent inadvertent release of CO2. Loss of nonseismic portions of the system does not affect the availability of protection in the Diesel Generator Buildings.

The CO2 Fire Protection System is complete with control room annunciation providing flow indication to each specific hazard area. This is accomplished by a pressure switch downstream of each local control (hazard) valve. This allows the control room operator to manually close a failed-open local control (hazard) valve which could be diverting flow from a fire area.

Appropriate discharge time delays are provided to permit personnel egress. CO2 discharge for these protected hazards may also be initiated manually, in the event a fire is observed, before automatic actuation occurs.

A low level indicator on the CO2 storage tanks is connected to an annunciator in the control room. A wintergreen odorizer is injected into the CO2 downstream of the local control (hazard) valve, so that the presence of CO2 is discernible by smell.

The low pressure CO2 systems shall be FUNCTIONAL whenever equipment protected by the CO2 systems is required to be FUNCTIONAL. Doors, dampers, and gas/pressure seals support the functionality of the CO2 systems by providing an adequate seal to prevent loss of extinguishing agent. Automatic closing dampers and doors and automatic stops of fans are tested as part of the CO2 system functional tests. The doors that support functionality of the system are listed in Table T9.3.11.D-2 and passive doors are inspected as part of the daily fire door inspections. Passive, normally closed doors that open into (as opposed to out of) the area protected by CO2 are not required to latch in order to be functional, since the CO2 discharged within the room will pressurize the room and that pressurization will ensure the door stays shut.

Unique system identifiers were developed during the development of the NFPA 805 FPR. Those unique system numbers contain information not contained with the plant equipment database. The FPR contains explanatory information regarding the system identifiers and descriptions. Cross references between the unique system identifiers and the Master Equipment List (MEL) are provided in the FPLCO tables in Section 9.3 below.

requiring a continuous fire watch when corresponding detection systems are NONFUNCTIONAL for Risk Significant barriers versus prescribing a roving fire watch for all LSS barriers).

2.8 Intentionally Left Blank

2.9 Emergency Lighting Units

Emergency lighting is provided to illuminate the areas containing equipment needed for emergency shutdown during a fire as well as the access and egress routes which must be taken to reach the necessary equipment.

Emergency lighting is credited where installed for defense-in-depth in the Fire Protection Program when any of the following criteria are met:

- The existing emergency light illuminates an area where a time critical action (time available less than hour) may be performed.
- The existing emergency light illuminates an ingress/egress path to a location where a time critical action may be performed.

NFPA 805 does not contain specific design requirements, but requires that lighting should be evaluated to ensure sufficient lighting is available to perform the intended action. (NFPA 805 Section B.5.2(e)(3))

Permanent fixed emergency lights located in the Main Control Rooms or at the Backup Control Panels, LPNL-925-0032, and listed in Table T9.3.11.I-1 are credited for risk criteria. Permanent fixed emergency lights listed in Table T9.3.11.I-2 are credited for defense-in-depth. Staged portable hardhat lights are made available as the primary credited means for ensuring sufficient lighting to perform operator actions and ingress/egress to meet the risk criteria in all other locations. 0-45E400-RW-09 defines the required emergency lights listed in Tables T9.3.11.I-1 and T9.3.11.I-2.

Based on staged portable lighting being provided to the operator and emergency lighting being available in the command and control locations, sufficient lighting is available to perform the intended actions. [MDQ0009992014000237 *NFPA 805 Credited Operator Actions*]

All emergency lighting units are considered to be Low Safety Significant (LSS).

2.10 Fire Extinguishers

Fire extinguishers are provided as backup suppression to the fixed fire protection systems and may be used as additional / alternative fire protection equipment when needed.

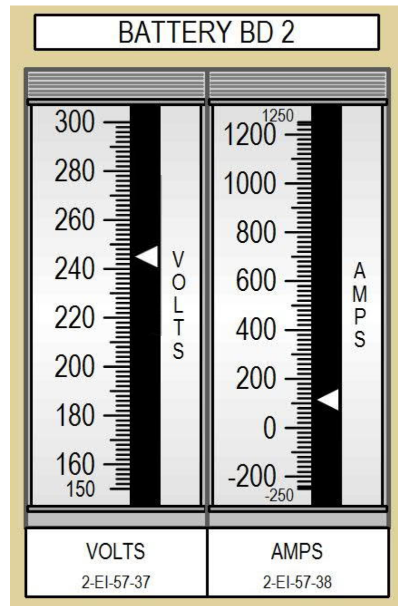
Fire extinguishers were selected to be used on Class A, B, and C fires as defined by NFPA Standard 10, dated 1967. Class ABC dry chemical extinguishers are provided in selected areas of the plant for combating incipient stage fires. In areas occupied by electrical switchboards and control panels, carbon dioxide, dry chemical, or Halon extinguishers are used.

The existing fire extinguisher inspection and replacement program in place at the station is sufficient for monitoring and extinguishers have been excluded from the FPRM.

Examination Outline Cross-reference:	Level	RO	SRO
263000 (SF6 DC) DC Electrical Distribution	Tier #	2	------
A1.01 (10CFR 55.41.5)	Group #	1	------
Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including:	K/A #	263000A1.01	
• Battery charging/discharging rate	Importance Rating	2.5	------

Proposed Question: **# 75**

Unit 2 is operating at 100% RTP with the following conditions:



- BATTERY BOARD 2 Volts (2-EI-57-37) indicates 245V DC
- BATTERY BOARD 2 Amps (2-EI-57-38) indicates (+) 120 amps

Which **ONE** of the following completes the statements below?

Battery #2 is currently ___**(1)**___.

Battery Board 2 Voltage ___**(2)**___ within the **MINIMUM OPERATING** voltage requirement specified in 0-OI-57D, DC Electrical System.

A. (1) discharging
(2) is

B. (1) discharging
(2) is NOT

C. (1) being charged
(2) is

D. (1) being charged
(2) is NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that in accordance with FSAR 8.6, 210V DC is the minimum DESIGN terminal voltage.

B CORRECT: *(See attached)* Given the current is (+) 120 amps above zero indicates that the battery is discharging. For second part, in accordance with 0-OI-57D, DC Electrical System, states Battery Board 2 Volts normal range is greater than 250 Volts as indicated on MCR Panel 9-8.

C INCORRECT: First part is incorrect but plausible in that the candidate could confuse being on the +/- side of zero of given indicated AMPS to conclude the battery is being charged. Second part is incorrect but plausible *(See A).*

D INCORRECT**:** First part is incorrect but plausible *(See C).* Second part is correct *(See B).*

RO Level Justification: Tests the candidate's ability to predict and monitor changes in DC Electrical parameters as it relates to battery charging and discharging rates. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome. The candidate has to assess plant conditions related to the often-confused and complex BFN DC Electrical Distribution System.

Technical Reference(s): 0-OI-57D, Rev. 175 (Attach if not previously provided)
OPL171.037, Rev. 16

Proposed references to be provided to applicants during examination: **Panel 2-9-8, BATTERY BOARD 2 Volts (2-EI-57-37) and Amps (2-EI-57-38)**

Learning Objective: OPL171.037 Obj. 8 (As available)

Question Source:	Bank #	BFN 1703 #49	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2017	

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis **X**

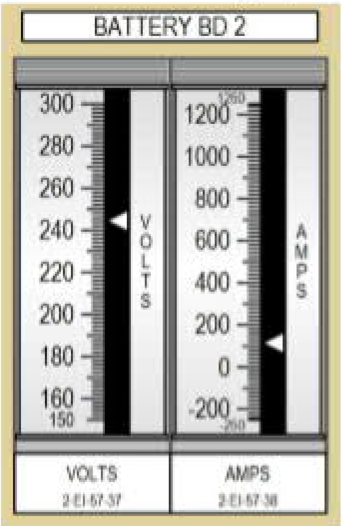
10 CFR Part 55 Content:
55.41 **X**
55.43

Copy of Bank Question:

QUESTION 49 Rev1

The following are indicated on Panel 2-9-8 for Battery Board 2:

BATTERY BD 2 Volts (2-EI-57-37) indicates 245VDC as shown below.
BATTERY BD 2 Amps (2-EI-57-38) indicates (+) 120 amps as shown below.



Which one of the following completes the statements below?

Main Bank Battery #2 is currently __ (1) __.

In accordance with 0-OI-57D, DC Electrical, Battery Board 2 voltage __ (2) __ within 0-OI-57D limits.

A. (1) discharging
(2) is **NOT**

B. (1) discharging
(2) is

C. (1) being charged
(2) is **NOT**

D. (1) being charged
(2) is

Answer: A

Excerpt from 0-OI-57D:

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0175 Page 141 of 336

6.1 Normal Operations (continued)

TABLE 2
250 VOLT DC UNIT BATTERY SYSTEM

LOCATION	PARAMETER	NORMAL RANGE
Battery Board Room No. 2(1,3) 250V Charger 2A (1,3,2B) Control Bay 593'	DC Volts	Greater than 250 Volts
	DC Amperes	Greater than zero less than 300 amps
	Power on Light	Illuminated
	Transformer Overtemp Light	Extinguished
	Overvoltage DC Light	Extinguished
	Undervoltage DC Light	Extinguished
	Undervoltage AC Light	Extinguished
Battery Board Room No. 2(1,3) Panel 1 Control Bay 593'	DC Volts	Greater than 250 Volts
	DC Amperes	Zero amps
	Bus Ground Indication(1)	Zero Volts
Battery Board Room No. 4 250V Charger 4 Turbine Bldg. 586'	DC Volts	Greater than 250 Volts
	DC Amperes	Greater than zero less than 300 amps
	Power on Light	Illuminated
	Transformer Overtemp Light	Extinguished
	Overvoltage DC Light	Extinguished
	Undervoltage DC Light	Extinguished
	Undervoltage AC Light	Extinguished

(1) **IF** a ground of an absolute valve greater than or equal to 30 volts is indicated, **THEN**

REFER TO 0-GOI-300-2.

Excerpt from FSAR 8.6:

charger common with the station system) together with the associated circuitry, switches, indicators, and alarms (Figure 8.6-1a).

The 250-V DC station system consists of three 120-cell lead-acid batteries (one Non-Class 1E battery and battery charger per unit and one Class 1E spare battery charger common with the Unit system) together with the associated circuitry, switches, indicators, and alarms (Figure 8.6-1f).

2. The 250-V DC control power supply system (Shutdown Board Batteries SB-A, SB-B, SB-C, SB-D, and SB-3EB) consists of five 120-cell lead-acid batteries (one battery and battery charger for each shutdown board, and one spare battery charger), together with the associated circuitry, switches, indicators, and alarms (Figure 8.6-1a). The batteries also supply 480-V shutdown boards for Units 1 and 2 and ATWS.

250-V Plant DC System

The battery chargers are of the solid-state, rectifier type, capable of working independently. Each charger is capable of automatically regulating output voltage. Each battery charger has the capacity to furnish floating, equalizing, and fast charge in accordance with the battery manufacturer's recommendations.

Each battery charger provides the 250-V DC supply during normal operations, keeps its associated battery fully charged at all times, and recharges the battery after a discharge. On loss of power to the charger, the battery supplies all required loads. Each battery is equipped with a low-voltage alarm which is actuated before battery voltage falls to 240-V.

Each of the batteries for the 250-V DC system consists of 120 lead-calcium grid type cells.

The unit batteries have a 1-minute rating of 2080 amperes and an 8-hour discharge rating of 2320 ampere-hours.

The station batteries have a 1-minute rating of 2240 amperes and an 8-hour discharge rating of 2320 ampere-hours.

All ratings are to a final terminal voltage of 210-V at a temperature of 77°F.

Excerpt from OPL171.037 Lesson Plan:

OPL171.037, DC Systems, Rev: 16

C. Components 1. Batteries a) The 250 volt batteries are 120-cell lead-calcium type. The Unit Batteries have a manufacturer 1 minute discharge rating of 2080 amps and an 8-hour discharge rating of 2320 amp-hours to a 210V DC minimum (required ECCS components must operate with as low as 200V). b) Two batteries can carry maximum expected load under DBA (Design Basis Accident) conditions without recharging for 30 minutes. c) The Plant/Station Batteries have a manufacturer 1 minute discharge rating of 2240 amps and a 8 hours discharge rating of 2320 amp-hours.	Obj 1a

Examination Outline Cross-reference:	Level	RO	SRO
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	Tier #	------	1
AA2.03 (10CFR 55.43.5 – SRO Only)	Group #	------	1
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:	K/A #	295001AA2.03	
• Actual core flow	Importance Rating	------	3.3

Proposed Question: **# 76**

Unit 2 is at 48% RTP with the following conditions:

- Reactor Recirculation Pump 2A tripped due to a VFD malfunction
- Recirc Pump 2A Discharge Valve is open
- 2-FI-68-46, JET PUMP A FLOW indicates 2 x 10^6 lbm/hr
- 2-FI-68-48, JET PUMP B FLOW indicates 42 x 10^6 lbm/hr

Given the conditions above, which **ONE** of the following completes the statements below?

In accordance with 2-AOI-68-1, Recirc Pump Trip / Core Flow Decrease, an accurate total core flow indication will result from Jet Pump flow in the out of service loop being ___**(1)**___.

The requirements of Tech Spec 3.4.1, Recirculation Loops Operating, are required to be implemented within ___**(2)**___ of entering single loop operations.

A. (1) subtracted instead of added
(2) 12 hours

B. (1) subtracted instead of added
(2) 24 hours

C. (1) added instead of subtracted
(2) 12 hours

D. (1) added instead of subtracted
(2) 24 hours

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). The second part is incorrect but plausible in that a reference is not provided for this question, requiring the candidate to recall the time from memory. Additionally, Tech Spec 3.4.1, Recirculation Loops Operating, Condition B has a completion time of 12 hours.

B **CORRECT**: In accordance with 2-AOI-68-1, Recirc Pump Trip / Core Flow Decrease, Recirc Pump operation with one Recirc Pump out of service and the in-service Jet Pump Flow less than or equal to 41 $x10^6$ lbm/hr can result in inaccurate Core Flow indication. This results from positive Jet Pump Flow in the out of service loop being subtracted instead of added. With flow for the in-service loop being greater than 41 x 10^6 lbm/hr, the out of service loop flow must be subtracted to have an accurate total core flow indication. This subtraction happens automatically following a pump trip/shutdown. The crew must verify the flow in the in-service loop is adequate to have accurate flow indication. For second part, in accordance with 2-AOI-68-1 and Tech Spec 3.4.1, Condition A, the LCO requirements of balanced flow or meeting the conditions for single Recirculation Loop Operation must be met within 24 hours.

C INCORRECT: The first part is incorrect but plausible in that the candidate could confuse core flow indicator/recorder operations as it relates to the out of service loop believing it is added instead of subtracted. The second part is incorrect but plausible (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is correct (*See B*).

SRO Level Justification: Tests the candidate's ability to determine actual Core Flow as it relates to partial or complete loss of forced Core Flow circulation. Additionally, Technical Specification requirement knowledge for Recirc Pump single loop operation is tested within this question. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the technical specifications and their bases. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): Tech Spec 3.4.1, Amend. 258 and 333 (Attach if not previously provided)
2-AOI-68-1, Rev. 37

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.007 Obj. 21, 22. 23 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1205 #76	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2012	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content:	55.41	
	55.43	**X**

Comments:

Copy of Bank Question:

QUESTION 76

Unit 2 is at 60% power after Reactor Recirculation Pump 2A tripped 15 minutes ago due to a VFD malfunction. The following plant conditions are indicated:

- Recirc Pump 2A discharge valve is open
- Loadline 90%
- JET PUMP A FLOW, 2-FI-68-46 2 mlbm/hr
- JET PUMP B FLOW, 2-FI-68-48 40 mlbm/hr

Which ONE of the following completes the statements below?

Total core flow indication on recorder 2-XR-68-50, TOTAL CORE FLOW/CORE PRESS DROP on Panel 2-9-5 __(1)__.

The requirements of Technical Specification 3.4.1 are required to be implemented within __(2)__ of entering single loop operations.

A. (1) is accurate
(2) 12 hours

B. (1) may be inaccurate
(2) 12 hours

C. (1) is accurate
(2) 24 hours

D. (1) may be inaccurate
(2) 24 hours

Correct answer: D

Excerpts from 2-AOI-68-1:

BFN Unit 2	Recirc Pump Trip/Core Flow Decrease	2-AOI-68-1 Rev. 0037 Page 5 of 14

2.0 SYMPTOMS

NOTE

Because a Reactor Recirc Pump seizure provides the same symptoms, the actions described herein cover that condition also. A seizure would most likely NOT be immediately discernible from other pump trips.

CAUTIONS

1) Operation with one recirc pump out of service and the inservice jet pump loop flow less than or equal to 41 x 10^6 lbm/hr (2-FI-68-46 or 2-FI-68-48) can result in inaccurate core flow indication. This results from positive jet pump flow in the out of service loop being subtracted instead of added. If operation in this condition is required, contact Reactor Engineers to perform Attachment 2 of 2-SR-3.4.1(SLO) to determine actual core flow and to substitute that value into the ICS as necessary.

BFN Unit 2	Recirc Pump Trip/Core Flow Decrease	2-AOI-68-1 Rev. 0037 Page 5 of 14

2.0 SYMPTOMS

NOTE

Because a Reactor Recirc Pump seizure provides the same symptoms, the actions described herein cover that condition also. A seizure would most likely NOT be immediately discernible from other pump trips.

CAUTIONS

1) Operation with one recirc pump out of service and the inservice jet pump loop flow less than or equal to 41 x 10^6 lbm/hr (2-FI-68-46 or 2-FI-68-48) can result in inaccurate core flow indication. This results from positive jet pump flow in the out of service loop being subtracted instead of added. If operation in this condition is required, contact Reactor Engineers to perform Attachment 2 of 2-SR-3.4.1(SLO) to determine actual core flow and to substitute that value into the ICS as necessary.

2) Immediately upon the opening of the "DRIVE RUNNING" contacts, the associated jet pump loop flow is subtracted even though the loop flow is still positive. This results in a severe indicated lowering in core flow, then as the tripped loop flow decays toward zero, the core flow indication will rise toward the actual value. The severity of the indicated core flow perturbation will depend upon the cause of the Recirc pump trip and the speed of the Recirc Drive prior to the trip.

3) [NER/C] The Natural circulation line on the Power/Flow map (0-TI-248 or ICS) only shows the approximate, nominal characteristic for operating with both Recirc loops out of service. Therefore, indicated core flow in natural circulation operation may not fall directly on the natural circulation line as depicted on the Power/Flow map. [NRC IN 96-016, GE SIL 516]

4) Per Technical Specifications, the Reactor CAN BE operated indefinitely with one Recirc loop out of service, provided the requirements of T.S. 3.4.1 are implemented within 24 hours of entering single loop operations.

5) [NER] The natural circulation line on the Power/Flow map (0-TI-248 or ICS) is only an approximation. Inaccuracies are evident at Low/No-Flow conditions.

6) Failure to monitor SJAE/OG CNDR CNDS FLOW, 2-FI-2-42, on Panel 2-9-6 for proper flow may only result in SJAE poor performance. The SJAE's will NOT trip on Condensate System low pressure.

7) Changes in Condensate System flow may require adjustment to SPE CNDS BYPASS, 2-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally should be in direct communication with the Control Room.

Excerpt from Unit 2 Tech Spec 3.4.1:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

-----NOTE-----
Single recirculation loop operation is prohibited in the MELLLA+ operating domain.

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation;

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	24 hours
B. Operating in the MELLLA+ operating domain with a single recirculation loop in operation.	B.1	Initiate action to exit the MELLLA+ operating domain.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met. OR No recirculation loops in operation.	C.1	Be in MODE 3.	12 hours

Excerpt from Unit 2 Tech Spec 3.4.1 (previous revision prior to MELLLA +):

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation;

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	24 hours
B. Required Action and associated Completion Time of Condition A not met. OR No recirculation loops in operation.	B.1	Be in MODE 3.	12 hours

Examination Outline Cross-reference:	Level	RO	SRO
295004 (APE 4) Partial or Complete Loss of D.C. Power / 6	Tier #	------	1
AA2.03 **(10CFR 55.43.5 - SRO Only)**	Group #	------	1
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:	K/A #	295004AA2.03	
• Battery voltage	Importance Rating	------	2.9

Proposed Question: **# 77**

Unit 3 is operating at 100% RTP. During the performance of 3-SR-3.8.6.2(DG 3A), Quarterly Check of Diesel Generator 3A Battery, Electrical Maintenance reports the following conditions:

- 3A EDG 125V BATTERY PILOT CELL SPECIFIC GRAVITY is 1.208 (corrected for electrolyte temperature and level)
- The average SPECIFIC GRAVITY of all connected cells is 1.215 (corrected for electrolyte temperature and level)
- All cell electrolyte levels are in between the minimum and maximum level marks
- Float voltage was reported to be 2.10 Volts

Given the conditions above, which **ONE** of the following is required in accordance with Tech Specs?

[REFERENCE PROVIDED]

A. Declare the 3A EDG 125V battery INOPERABLE **IMMEDIATELY**.

B. Verify pilot cell's electrolyte level and float voltage is within the Table 3.8.6-1 Category C limits within 1 hour **AND** once per 7 days thereafter, **AND** Restore battery cell parameters to Category A and B limits of Table 3.8.6-1 within 31 days.

C. Verify battery cell parameters meet Table 3.8.6-1 Category C limits within 24 hours **AND** once per 7 days thereafter, **AND** Restore battery cell parameters to Category A and B limits of Table 3.8.6-1 within 31 days.

D. Verify pilot cell's electrolyte level and float voltage is within the Table 3.8.6-1 Category C limits within 1 hour **AND** Verify battery cell parameters meet Table 3.8.6-1 Category C limits within 24 hours **AND** once per 7 days thereafter, **AND** Restore battery cell parameters to Category A and B limits of Table 3.8.6-1 within 31 days.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible if the candidate confuses the provided battery pilot cell parameters associated with Category A, B, or C from Tech. Spec. 3.8.6, Battery Cell Parameters, Table 3.8.6-1 and misapplies Condition B and respective Required Actions and Completion Time.

B INCORRECT: Incorrect but plausible if the candidate confuses the provided battery pilot cell parameters associated with Category A, B, or C from Tech. Spec. 3.8.6, Table 3.8.6-1and misapplies Condition A and respective Required Actions and Completion Times.

C INCORRECT: Incorrect but plausible if the candidate confuses the provided battery pilot cell parameters associated with Category A, B, or C from Tech. Spec. 3.8.6, Table 3.8.6-1 and misapplies Condition A and respective Required Actions and Completion Times.

D **CORRECT**: (*See attached*) In accordance with Unit 3 Tech Spec 3.8.6, the specific gravity, and electrolyte levels both meet requirements, however Category A and B cell float voltages are NOT within the minimum voltage of ≥ 2.13 V for each Category. Minimum required cell voltage for Category C is > 2.07 V.

SRO Level Justification: Tests the candidate's knowledge of required battery systems, MODES of applicability for DC required systems regarding their ability to apply data to Technical Specification tables and determine the Required Actions and Completion Times. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (2) Diagnosis that leads to selection of procedures that should be used to respond to the evolution.

Technical Reference(s): Unit 3 Tech Spec 3.8.6, Rev. 212 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **Unit 3 Tech Spec 3.8.6 and Table 3.8.6-1 (No Bases)**

Learning Objective: OPL171.037 Obj. 8, 9 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC		

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Excerpts from Unit 3 Tech Spec 3.8.6:

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Unit, Shutdown Board, and DG batteries shall be within the limits of Table 3.8.6-1.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Category A or B limits.	A.1	Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	AND		
	A.2	Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours AND Once per 7 days thereafter
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with average electrolyte temperature of the representative cells not within limits. <u>OR</u> One or more batteries with one or more battery cell parameters not within Category C values.	B.1 Declare associated battery inoperable.	Immediately

Battery Cell Parameters
3.8.6

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameter Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE VALUE FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity (b)(c)(d)	≥ 1.20	≥ 1.195 AND Average of all connected cells > 1.205	Not more than 0.020 below average of all connected cells AND Average of all connected cells ≥ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature.

(c) As an alternative to the specific gravity measurements, a battery charging current of < 1 amp for Unit and Shutdown Board batteries and < 0.5 amp for DG batteries when on float charge is acceptable only during a maximum of 7 days following a battery recharge. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

(d) Alternate values may be used for a limited number of cells provided demonstrated battery capacity at the last discharge test meets the minimum qualifying value.

Examination Outline Cross-reference:

295006 (APE 6) Scram / 1

AA2.01 (10CFR 55.43.5 – SRO Only)

Ability to determine and/or interpret the following as they apply to SCRAM

- Reactor Power

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295006AA2.01	
Importance Rating	------	4.6*

Proposed Question: **# 78**

Unit 2 is operating at 100% RTP when the following conditions occurred:

At 0805:

- Reactor SCRAM
- **NO** Control Rods initially inserted
- Immediate Actions of 2-AOI-100-1, Reactor SCRAM are complete

At 0815:

- MSIVs are CLOSED
- Reactor Pressure is being maintained 800 to 1000 psig with two (2) Main Steam Relief Valves (MSRVs) OPEN and a third being **MANUALLY** cycled

At 0817, the Shift Manager, as the SED, makes an Emergency Plan event declaration.

Given the conditions above, which **ONE** of the following completes the statement below in accordance with EPIP-1, Emergency Classification Procedure?

At 0817, the **HIGHEST** required Emergency Classification Action Level (EAL) to report is a/an **___(1)___** and the State of Alabama is required to be notified **NO** later than **___(2)___**.

Note: Notification of Unusual Event (NOUE)
SED judgement shall **NOT** be used as a basis for classification

[REFERENCE PROVIDED]

A. (1) ALERT (SA5)
(2) 0832

B. (1) ALERT (SA5)
(2) 0835

C. (1) NOUE (SU5)
(2) 0832

D. (1) NOUE (SU5)
(2) 0835

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with EPIP-1, Attachment 3, Emergency Classification Scheme Bases, SA5 (page 124 of 137) given the information in the stem that an ATWS has occurred and subsequent manual actions taken at the Reactor Control Consoles are not successful in shutting down the Reactor. The Immediate Actions of 2-AOI-100-1, Reactor SCRAM, have been completed, which indicates that ARI has been initiated and a manual SCRAM inserted. Given that the MSIVs are CLOSED, Bypass Valves are not available for Reactor Pressure control; however, the fact that 2-3 SRVs must be opened to control Reactor Pressure indicates that Control Rods are still out. Reactor Power is approximately 10-15% since each SRV accounts for 4.5% - 5% RTP. For second part, in accordance with EPIP-2, Alert, the Notification of the State of Alabama is required to be completed as soon as possible, and within 15 minutes of the time of the Emergency Classification. The event was classified at 0817, therefore adding 15 minutes yields a time of no later than 0832.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible in that the candidate may apply the rule that the Site Emergency Director has 15 minutes to classify the event and then 15 minutes to notify the State. Therefore, 30 minutes + 0805 results in a time of 0835.

C INCORRECT: The first part is incorrect but plausible in that with the Immediate Actions of 2-AOI-100-1 being complete, the candidate may believe that all Control Rods are in. The candidate may not correlate that having 2-3 SRVs open to maintain Reactor Pressure with an ATWS condition indicates approximately 10-15% RTP. Therefore, the candidate would select SU5 (page 134 of 137) in accordance with EPIP-1. The second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible *(See C)*. Second part is incorrect but plausible *(See B).*

SRO Level Justification: Test the candidate's ability to determine and interpret Reactor Power as it applies to a Reactor SCRAM and ATWS conditions. SRO only because of the link to 10CFR55.43 (2): Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome especially without a given reference.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents, (2) Diagnosis that leads to selection of procedures that should be used to respond to the evolution and (3) The progression of an event.

Technical Reference(s): EPIP-1, Rev. 59 (Attach if not previously provided)

EPIP-3, Rev. 42

Proposed references to be provided to applicants during examination: **EPIP-1, Attachment 1, HOT INITIATING CONDTIONS- MODES 1-2-3**

Learning Objective: OPL171.075 Obj. 2 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1306 #80	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2013	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	X

Comments:

Copy of Bank Question:

QUESTION 80

Given the following conditions:

- Unit 2 was operating at 100% power
- At time 0805 a scram occurred due to a loss of both RPS Bus A and RPS Bus B
- **NO** control rods initially inserted
- Following the manual scram and ARI, several rods failed to fully insert

At 0815, the following conditions exist:

- Reactor power is UNKNOWN
- Reactor pressure is being maintained 800 to 1000 psig with two (2) SRVs OPEN and a third being manually cycled
- Reactor water level is (-)75 inches and steady, being maintained using HPCI
- Suppression pool temperature is 136°F and rising

At 0817, the Shift Manager, as the Site Emergency Director, made an Emergency Plan event declaration in accordance with EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE

Which ONE of the following completes the statement below?

At 0817, the highest required classification is __(1)___ and the State of Alabama is required to be notified no later than ___(2)____.

[REFERENCE PROVIDED]

A. (1) General Emergency
(2) 0832

B. (1) General Emergency
(2) 0835

C. (1) Site Area Emergency
(2) 0835

D. (1) Site Area Emergency
(2) 0832

Answer: **D**

Excerpt from EPIP-1:

BFN Unit 0	Emergency Classification Procedure Attachment 3 – Bases	EPIP-1 Revision 0059 Page 124 of 137

SA5

ECL: Alert

Initiating Condition: Automatic or manual scram fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.

Operating Mode Applicability: Power Operation, Startup

Emergency Action Levels:

Note:	A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

(1) a. An automatic or manual scram did not shutdown the reactor.

AND

b. Automatic ARI or Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (for example, initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control consoles (for example, locally opening breakers). Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shutdown the reactor is prolonged enough to cause a challenge to the RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS5. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS5 or FS1, an Alert declaration is appropriate for this event.

Supports Distractors C(1), D(1):

BFN Unit 0	Emergency Classification Procedure Attachment 3 – Bases	EPIP-1 Revision 0059 Page 134 of 137

SU5

ECL: Unusual Event

Initiating Condition: Automatic or manual scram fails to shutdown the reactor.

Operating Mode Applicability: Power Operation, Startup

Emergency Action Levels: (1 or 2)

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

(1) a. An automatic scram did not shutdown the reactor.

AND

b. A subsequent automatic ARI or manual action taken at the reactor control consoles is successful in shutting down the reactor.

OR

(2) a. A manual scram did not shutdown the reactor.

AND

b. EITHER of the following:

- A subsequent automatic ARI or manual action taken at the reactor control consoles is successful in shutting down the reactor.
- A subsequent automatic scram is successful in shutting down the reactor.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (for example, initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor. Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

Excerpt from EPIP-3:

BFN Unit 0	ALERT	EPIP-3 Rev. 0042 Page 7 of 29

3.1 State of Alabama Notification

NOTE
Notification of the State of Alabama is required to be completed as soon as possible, within 15 minutes from the time of emergency classification declaration.

[1] **PERFORM** the following:

[1.1] **RECORD** the following information:

- Time of ALERT Event Classification: ____________________

[1.2] **IF** the CECC is NOT activated, **THEN**

CONTINUE in this procedure at step 3.1[2].
Otherwise continue in this step.

RECORD the following information:

- ALERT Classification IC Designator: ____________________
- Site Emergency Director: (Name) ________________________

[1.3] **CONTACT** the CECC Director at 1-423-751-1614 and communicate the information recorded in Step 3.1[1.1] and 3.1[1.2].

[1.4] **CONTINUE** in this procedure at Section 3.2 (skipping completion of Appendix A and Appendix B).

[2] **COMPLETE** Appendix A, “Alert Initial Notification Form.”

[3] **COMPLETE** Appendix B, “State of Alabama and Operations Duty Specialist (ODS) Notification,” utilizing a completed Appendix A.

3.2 Dose Assessment Evaluation

[1] **IF** emergency circumstances warrant dose assessment, **THEN**

CONTACT Radiation Protection at 729-7865 and **REQUEST** the implementation of EPIP-13, “Dose Assessment.”

Examination Outline Cross-reference:

295024 (EPE 1) High Drywell Pressure / 5

G2.2.36 (10CFR 55.43.2 - SRO Only)

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295024G2.2.36	
Importance Rating	------	4.2

Proposed Question: **# 79**

Unit 2 is operating at 100% RTP. During surveillance testing, the IMs report that pressure switch 2-PIS-64-56A, Drywell Pressure Instrument Channel A1, failed an Acceptance Criteria (AC) step to de-energize within its allowable values.

Determine which **ONE** of the following actions / limitations is required in accordance with Tech Spec 3.3.1.1, RPS Instrumentation.

[REFERENCE PROVIDED]

A. No action required.

B. Place channel in trip in 6 hours.

C. Place channel in trip in 12 hours.

D. Must be in MODE 3 in 12 hours.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT**:** Incorrect but plausible if the candidate fails to realize the correct number of CHANNELS (all) in both RPS Trip Systems A and B are required to be OPERABLE as well as knowing the MODES of Applicability. This would lead them to incorrectly believe this condition would require an Information LCO only.

B INCORRECT: Incorrect but plausible in that the 6 hour Completion Time is applicable when both RPS A and RPS B have a Channel that is INOPERABLE and not applicable to this condition since both the given pressure switch (PIS) and the relay are in RPS A logic bus (A1). Often times, candidates confuse functions, required channels and/or trip systems associated with ECCS, PCIS and/or RPS Instrumentation. Table 3.3.1.1-1 requires understanding the current operating condition associated with RPS Instrumentation which all CHANNELS are required to be OPERABLE.

C **CORRECT**: In accordance with Tech Spec 3.3.1.1. RPS Instrumentation, even with the given single CHANNEL failure, RPS capability still exists and RPS B still has both CHANNELS that remain OPERABLE. Table 3.3.1.1-1, FUNCTION 6 for Drywell Pressure – High indicates 4 CHANNELS are required per TRIP system – 2 channels per RPS A and B respectively. In this case, Condition A is applicable with One or more required CHANNELS INOPERABLE, with Required Action to Place CHANNEL in trip within 12 hours.

D INCORRECT: Incorrect but plausible if the candidate confuses functions, required channels and/or trip systems associated with RPS Instrumentation. Misreading Table 3.3.1.1-1 would require entry into Condition G with Required Action to be in MODE 3 within 12 hours.

SRO Level Justification: Tests the candidate's ability to apply Technical Specifications as it relates to LCO status for given specific RPS specific instruments for High Drywell Pressure.
SRO only because of link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (4) Assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems and/or safety functions.

Technical Reference(s): Unit 2 Tech Spec 3.3.1.1, Rev. 258 (Attach if not previously provided)
2-SR-3.3.1.1.13(6A), Rev. 5
2-730E915-9, Rev. 29

Proposed references to be provided to applicants during examination: **Unit 2 Tech Spec 3.3.1.1, Table 3.3.1.1-1 (No Bases)**

Learning Objective: OPL171.028, Obj. 7 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:			

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Excerpt from 2-SR-3.3.1.1.13(6A): Illustrating given instrument High Drywell Pressure

BFN Unit 2	Reactor Protection and Primary Containment Isolation Systems High Drywell Pressure Instrument Channel A1 Calibration	2-SR-3.3.1.1.13(6A) Rev. 0005 Page 12 of 40

6.0 ACCEPTANCE CRITERIA

A. Responses which fail to meet the acceptance criteria stated in this section shall constitute unsatisfactory surveillance results documentation shall be initiated in accordance with NPG-SPP-06.9.1 and require notification of the Unit Supervisor at the time of failure.

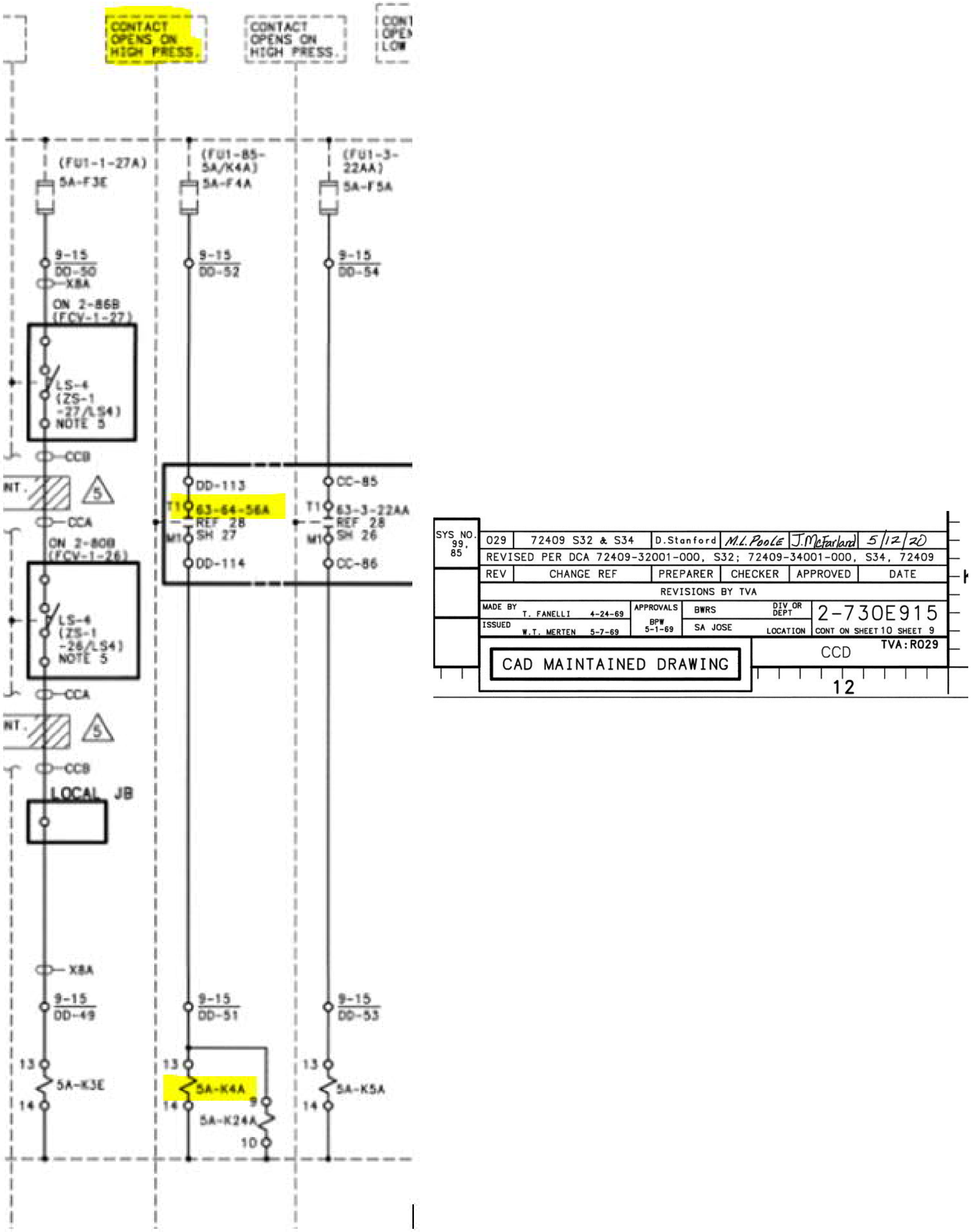
1. Channel trip relay 2-RLY-099-05AK04A de-energizes for a pressure input to 2-PT-064-0056A of 2.50 psig or less increasing and causes an A channel half scram.

2. Relay 2-RLY-064-16AK5A de-energizes when Relay 2-RLY-099-05AK04A de-energizes.

3. A RPS half-scram signal will extinguish SCRAM SOLENOID GROUP A LOGIC RESET, 1 2, 3, and 4 indicating lights(4) on Panel 2-9-5.

B. Steps which determine this criteria are designated by (AC) next to the initial blank.

Excerpt from 2-730E915-9: Illustrating given instrument High Drywell Pressure contact/relay



Excerpt from Unit 2 Tech Spec 3.3.1.1:

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u>	
	A.2 -----NOTE----- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ----- Place associated trip system in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------------NOTE------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ---------------------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	OR		
	B.2	Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours

(continued)

RPS Instrumentation
3.3.1.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
J. Required Action and associated Completion Time of Condition I not met.	J.1	Be in Mode 2.	4 hours

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA(e)
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
(b) Each APRM channel provides inputs to both trip systems.
(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence

Examination Outline Cross-reference:

295026 (EPE 3) Suppression Pool High Water Temperature / 5

G2.4.18 (10CFR 55.43.1 - SRO Only)

Knowledge of the specific bases for EOPs.

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295026G2.4.18	
Importance Rating	------	4.0

Proposed Question: **# 80**

An ATWS has occurred on Unit 3.

Which **ONE** of the following completes the statements below concerning Boron injection?

In accordance with EOI-1A, ATWS RPV Control, before Suppression Pool Temperature rises to **___(1)___**, Boron injection is required.

In accordance with the EOI Program Manual Bases, the reason Boron is injected at this temperature is **___(2)___**.

A. (1) 110 °F
   (2) for alternate source term control

B. (1) 110 °F
   (2) to shut down the Reactor prior to overheating the Suppression Pool

C. (1) 120 °F
   (2) for alternate source term control

D. (1) 120 °F
   (2) to shut down the Reactor prior to overheating the Suppression Pool

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct *(See B)*. The second part is incorrect but plausible in that in accordance with the Alarm Response Procedures, Boron is injected within 2 hours of receiving 3-ARP-9-7C, Window 15, DRYWELL/SUPPR CHAMBER RADIATION HIGH, Boron is injected for alternate source term control.

B **CORRECT**: *(See attached)* In accordance with EOI-1A, ATWS RPV Control, Step ARC/Q-8 states that BEFORE Suppression Pool Temperature rises to 110º F, Boron injection is required. In accordance with EOIPM Section 0-V-M, EOI-1A, ATWS RPV Control Bases states that the reason Boron is injected prior to 110 °F Suppression Pool Temperature is to avoid depressurizing with the Reactor at power prior to reaching the Heat Capacity Temperature Limit. This will shut down the Reactor prior to overheating the Suppression Pool.

C INCORRECT: The first part is incorrect but plausible in that according to Technical Specification 3.6.2.1, Suppression Pool Average Temperature, the Reactor must be depressurized to less than 200 psig if Suppression Pool Temperature is above 120 °F. The second part is incorrect but plausible *(See A)*.

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is correct (*See B*).

SRO Level Justification: Test the candidate's knowledge of the EOI-1A Bases as it relates to High Suppression Pool Water Temperature and Boron Injection. SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as Memory due to the requirement to strictly recall both Tech Spec Bases and EOI facts.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): 3-EOI-1A, Rev. 2 (Attach if not previously provided)
EOIPM O-V-M, Rev. 0
Unit 3 Tech Spec 3.6.1.2, Amend.212

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.202 Obj. 20 (As available)

Question Source:	Bank #	ILT EXAM BANK OPL171.201-10 045 #2556	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

2556. OPL171.201-10 045

An ATWS has occurred on Unit 3.

Which of the following completes both statements?

In accordance with EOI-1A, ARC/Q, before suppression pool temperature rises to __ (1) __ boron injection is required.

In accordance with EOIPM Section 0-V-M, EOI-1A, ATWS RPV Control Bases, the reason boron is injected at this temperature is to __ (2) __.

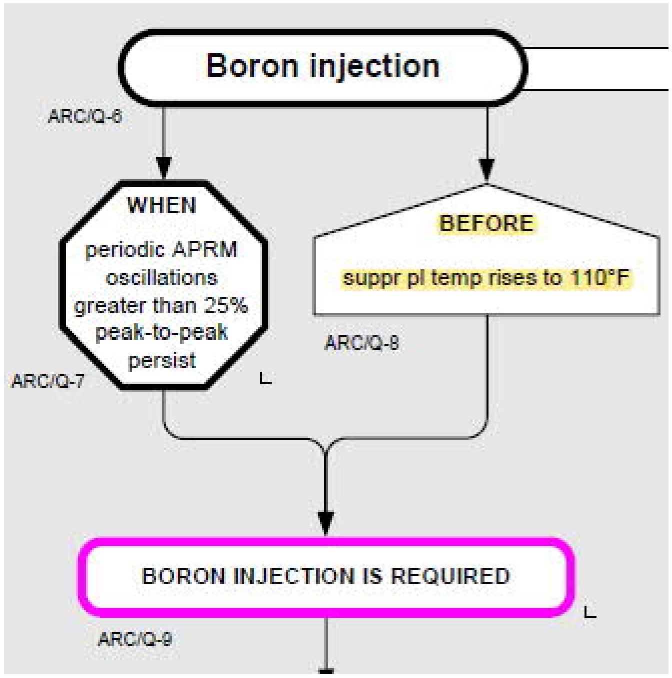
A✓ (1) 110 °F
(2) preclude emergency RPV depressurization

B. (1) 110 °F
(2) prevent Thermal Hydraulic Instabilities (THI)

C. (1) 120 °F
(2) preclude emergency RPV depressurization

D. (1) 120 °F
(2) prevent Thermal Hydraulic Instabilities (THI)

Excerpt from 3-EOI-1A:



Excerpt from EOIPM 0-V-M:

BFN Unit 0	EOI-1A, ATWS RPV Control Bases	EOIPM Section 0-V-M Rev. 0000 Page 155 of 165

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/Q-8

If suppression pool temperature and RPV pressure cannot be maintained in a safe region of the Heat Capacity Temperature Limit (Curve 3), emergency RPV depressurization will be required. To avoid depressurizing the RPV with the reactor at power, it is desirable to shut down the reactor prior to reaching the Heat Capacity Temperature Limit. The Boron Injection Initiation Temperature (BIIT, **A.64**) is defined so as to achieve this goal when practicable.

The Boron Injection Initiation Temperature (BIIT) is the greater of:

- The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.
- The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

Excerpt from Unit 3 Tech Spec 3.6.2.1:

Suppression Pool Average Temperature
3.6.2.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Suppression pool average temperature > 120°F.	E.1 Depressurize the reactor vessel to < 200 psig.	12 hours
	AND	
	E.2 Be in MODE 4.	36 hours

Excerpt from 3-ARP-9-7C: Supports Distractors A(2), C(2)

BFN Unit 3	Panel 9-7 3-XA-55-7C	3-ARP-9-7C Rev. 0038 Page 21 of 41

DRYWELL/SUPPR CHAMBER RADIATION HIGH 3-RA-90-272, Window 15
(Page 2 of 2)

Operator
Action: (Continued)

E. IF ALL the following conditions exist (1, 2, and 3):
   1. Alarm is determined to be valid, AND
   2. The reactor will remain subcritical without boron injection under all conditions, AND
   3. Leakage of primary coolant into primary containment is indicated, THEN

   Within 2 hours of alarm, **INJECT** SLC for alternate source term control by placing SLC PUMP 3A/3B, 3-HS-63-6A in the START A or **START** B position. ☐

F. REFER TO EPIPs. ☐

G. IF started at Operator Action Step E, THEN
   WHEN SLC tank reaches 0", **STOP** the running SLC Pump ☐

H. EVALUATE equipment associated with this alarm to determine compensatory actions required to maintain REP function.
   REFER TO NPG-SPP-18.3.5. ☐

References: 3-45E620-9 0-47E610-90-2 NESSD 3R-090-273A-00
Technical Specifications Section 3.3.3.1 NESSD 3R-090-272A-00

Examination Outline Cross-reference:	Level	RO	SRO
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5	Tier #	------	1
EA2.01 **(10CFR 55.43.5 - SRO Only)**	Group #	------	1
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE:	K/A #	295028EA2.01	
• Drywell temperature	Importance Rating	------	4.1*

Proposed Question: **# 81**

Which **ONE** of the following completes the statements below concerning High Drywell Temperature?

In accordance with Tech Spec Bases 3.6.1.4, Drywell Air Temperature, the highest initial Drywell Average Air Temperature (DW/T) which ensures that peak Drywell Temperature will **NOT** be reached following a Design Basis Accident (DBA) LOCA is ___**(1)**___.

In accordance with EOI-2, Primary Containment Control DW/T leg, Emergency Depressurization is required when DW/T **CANNOT** be restored and maintained below ___**(2)**___.

A. (1) 150 °F
(2) 280 °F

B. (1) 150 °F
(2) 350 °F

C. (1) 160 °F
(2) 280 °F

D. (1) 160 °F
(2) 350 °F

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct *(See B)*. The second part is incorrect but plausible in that EOI-2 states before DW/T rises to 280 °F (DW/T-4), enter EOI-1 and ENSURE Reactor SCRAM.

B **CORRECT**: *(See attached)* In accordance with Tech Spec Bases 3.6.1.4, among the inputs to the Design Basis Analysis is the initial Drywell Air Temperature (DW/T) of 150 °F. This limitation ensures that the Safety Analysis remains valid by maintaining the expected initial conditions and ensures that the peak DW/T does not exceed the maximum calculated temperature of 336 °F. For second part, in accordance with EOI-2, Primary Containment Control, when DW/T cannot be restored and maintained below 350 °F, DW/T-7 and 8 directs an Emergency Depressurization.

C INCORRECT: The first part is incorrect but plausible in that DW/T above 160 °F is an EOI-2 entry condition. It is also listed in the DW/T leg in both steps DW/T-1 and 2. The second part is incorrect but plausible *(See A)*.

D INCORRECT: The first part is incorrect but plausible *(See C)*. The second part is correct *(See B)*.

SRO Level Justification: Test the ability of the candidate to determine and/or interpret High Drywell Temperature as it relates to Technical Specification Bases 3.6.1.4, Drywell Air Temperature and Design Basis Accident. Furthermore, the candidate must distinguish between the EOI-2, Primary Containment Control entry condition and the Emergency Depressurization requirement related to High Drywell Temperature. SRO only because of link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as Memory due to the requirement to strictly recall both Tech Spec Bases and EOI facts.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents.

Technical Reference(s): U2 TS Bases 3.6.1.4, Amend 254 (Attach if not previously provided)
2-EOI-1, Rev. 18
2-EOI-2, Rev. 16
EOIPM 0-V-D, Rev. 2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203, Obj. 4 (As available)

Question Source:	Bank #	BFN 1909 #80	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Copy of Bank Question:

Proposed Question: **# 80**

Which **ONE** of the following completes the statements below concerning High Drywell Temperature (DW/T)?

In accordance with Tech Spec Bases, the highest initial Drywell Average Air Temperature which ensures that the peak Drywell Temperature will **NOT** be reached following a Design Basis Accident (DBA) LOCA is ___**(1)**___.

In accordance with EOI-2, DW/T leg, Emergency Depressurization is required when DW/T cannot be restored and maintained below ___**(2)**___.

A. (1) 160 °F
(2) 280 °F

B. (1) 160 °F
(2) 350 °F

C. (1) 150 °F
(2) 280 °F

D. (1) 150 °F
(2) 350 °F

Proposed Answer: **D**

Excerpt from Tech Spec Bases 3.6.1.4:

B 3.6 CONTAINMENT SYSTEMS

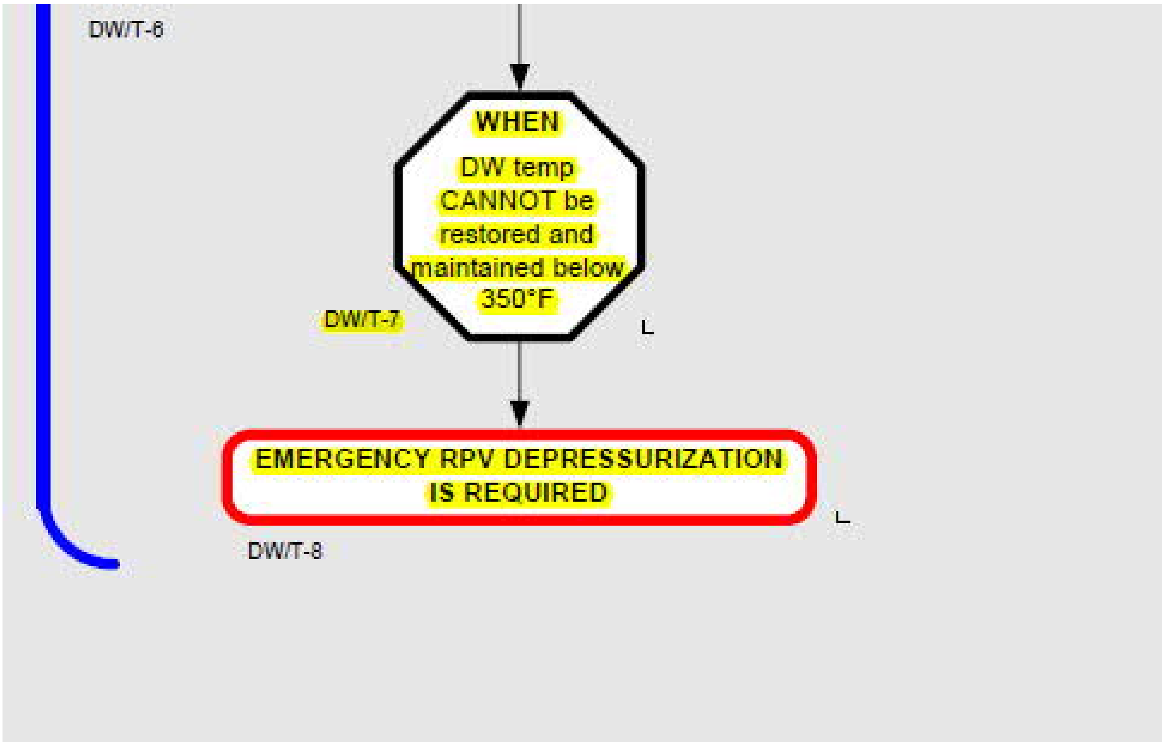
B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND	The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.
APPLICABLE SAFETY ANALYSES	Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak drywell temperature does not exceed the maximum calculated temperature of 336°F (Ref. 2). Exceeding this temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident. Drywell air temperature satisfies Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)

Excerpt from 2-EOI-2, DW/T-7 and 8:



2-EOI-2	Page 1 of 1
PRIMARY CONTAINMENT CONTROL UNIT 2 BROWNS FERRY NUCLEAR PLANT	
Rev: 16	

Excerpt from EOIPM 0-V-D related to DW/T-7 and 8:

BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V-D Rev. 0002 Page 31 of 119

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

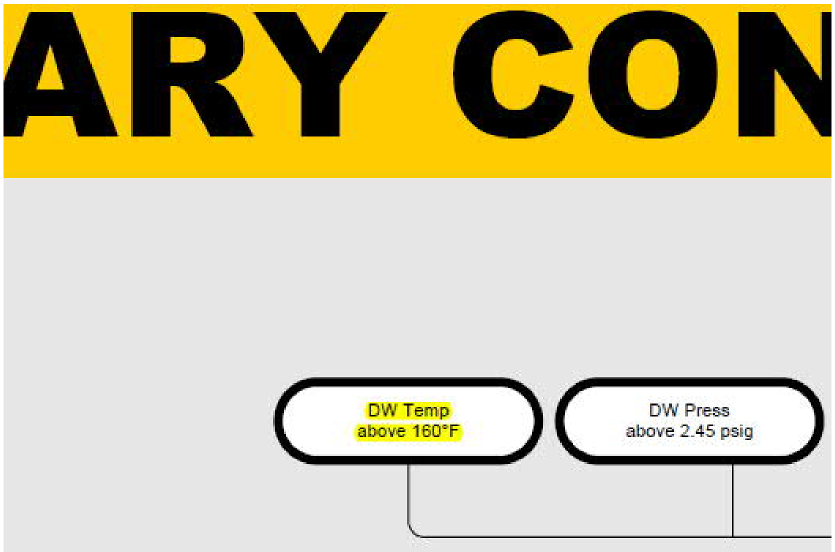
DISCUSSION: DW/T-7, DW/T-8

If drywell temperature cannot be restored and maintained below the ADS qualification temperature, emergency RPV depressurization is performed. This action minimizes any continuing direct energy release to the drywell through a primary system break and ensures that the MSRVs are opened while still operable.

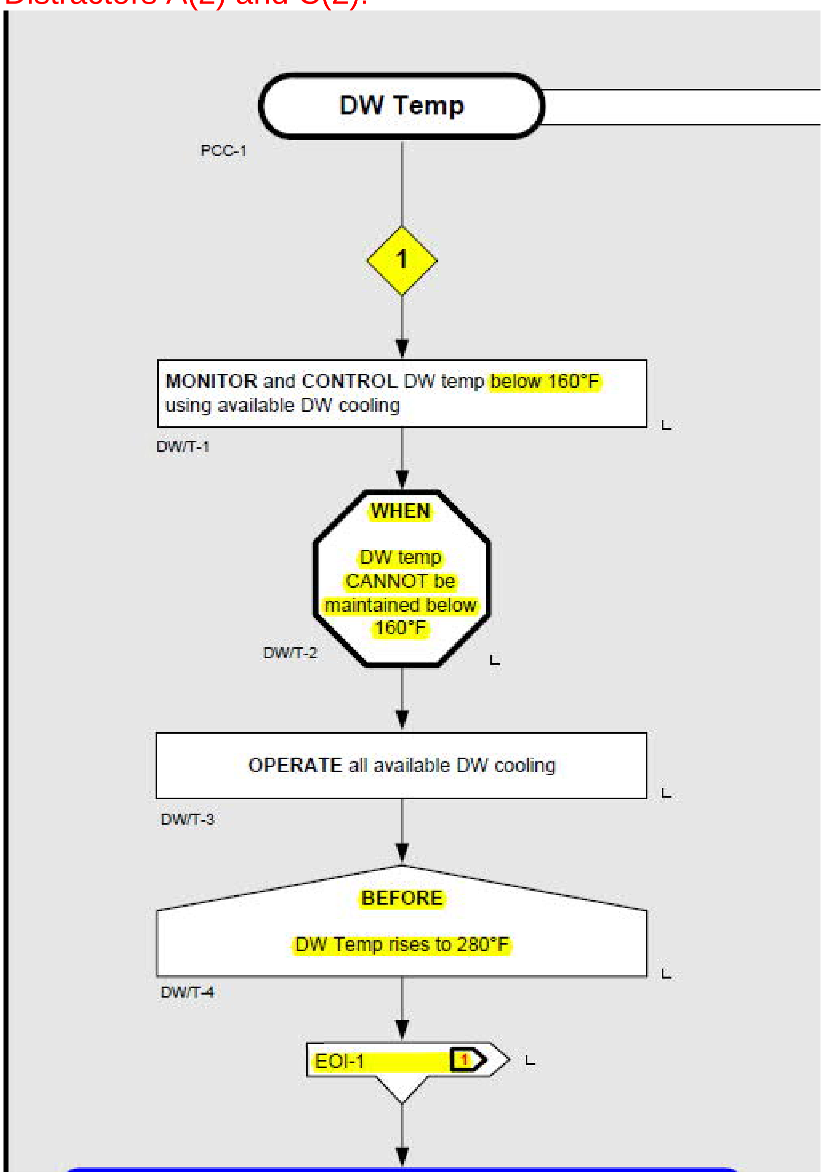
Consistent with the definition of "restore," emergency RPV depressurization is not required until it has been determined that drywell sprays (initiated in Step DW/T-6) are ineffective in reducing drywell temperature. It is not expected that MSRV operability will be immediately challenged when the ADS qualification temperature is reached. If drywell temperature is already above the specified value when Steps DW/T-7 and DW/T-8 are reached, drywell sprays may still be used, if available, in preference to emergency depressurization. If sprays are effective in reducing drywell temperature, emergency depressurization need not be performed. Extended operation above the ADS qualification temperature is not permitted, however.

A determination that drywell temperature cannot be restored and maintained below the ADS qualification temperature may be made when, before, or after temperature actually reaches the specified value.

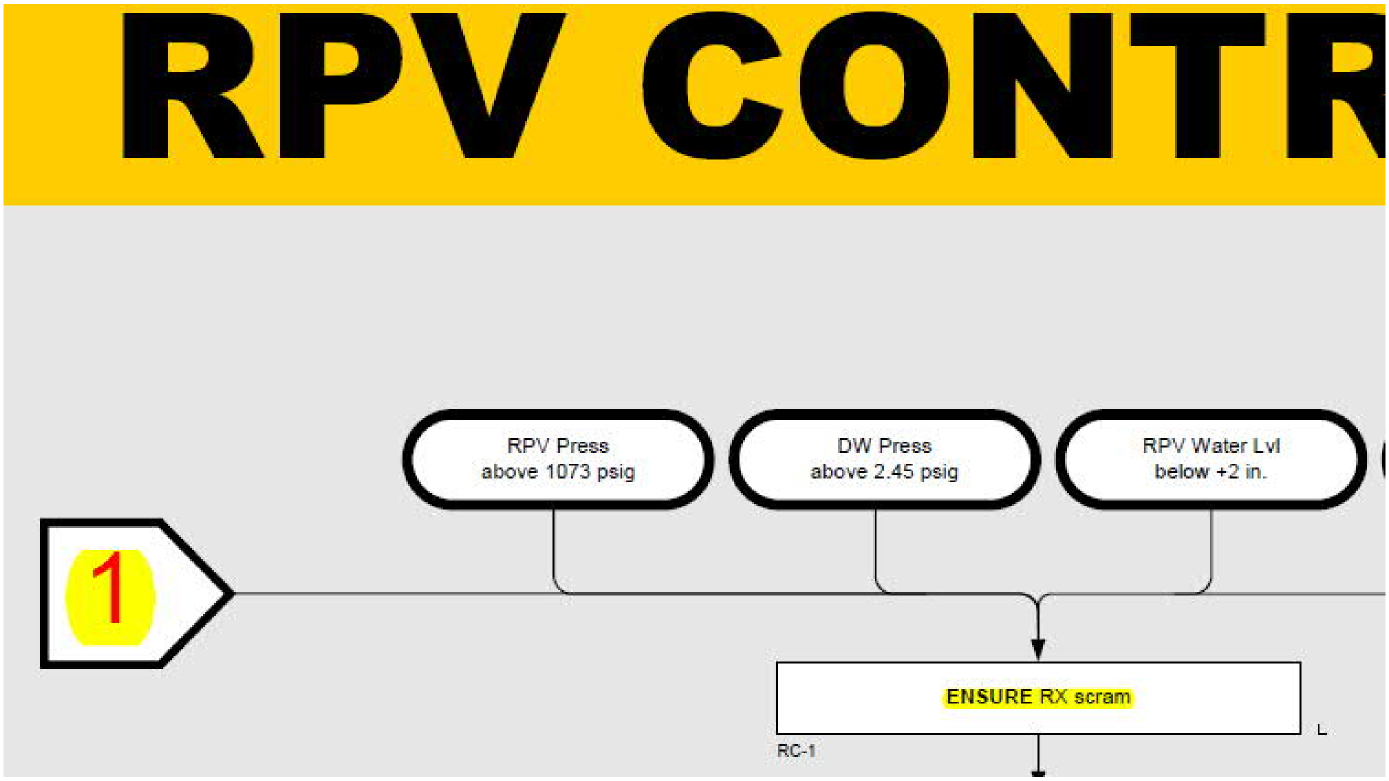
Excerpts from 2-EOI-2: Supports Distractors C(1), D(1)

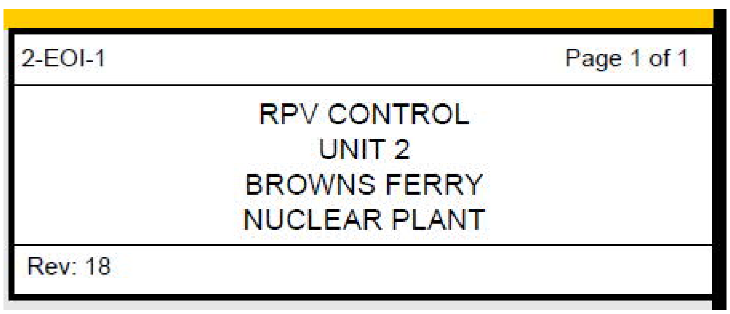


Distractors A(2) and C(2):



Excerpt from 2-EOI-1 showing entry from 2-EOI-2, DW/T-4:





2-EOI-1	Page 1 of 1
RPV CONTROL UNIT 2 BROWNS FERRY NUCLEAR PLANT	
Rev: 18	

Examination Outline Cross-reference:

295037 (EPE 14) SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1

G2.4.9 (10CFR 55.43.5 – SRO Only)

Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295037G2.4.9	
Importance Rating	------	4.2

Proposed Question: **# 82**

A Unit 1 Reactor startup is in progress in accordance with 1-GOI-100-1A, Unit Start Up, when a leak in Primary Containment results in the following conditions:

At 0910:

- Reactor SCRAM has been inserted
- Multiple Control Rods failed to fully insert
- APRMs indicate 6% Reactor Power
- OATC is continuing with ATWS actions

At 0920:

- SLC injection has lowered tank level by 10%
- APRMs indicate 0% Reactor Power
- IRMs are inserted and indicate downscale on Range 2
- ATWS Actions are complete in accordance with 1-AOI-100-1, Reactor SCRAM, OATC Hard Card

Given the conditions above, which **ONE** of the following completes the statements below?

The SRO will direct Reactor **POWER** Control to ___**(1)**___. In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, the **CORRECT** Reactor Water Level band is ___**(2)**___.

Note: 1-EOI-1A, ATWS RPV Control

A. (1) continue with 1-EOI-1A
(2) (+) 2 to (+) 51 inches

B. (1) continue with 1-EOI-1A
(2) (-) 50 to (-) 100 inches

C. (1) transition to 1-AOI-100-1
(2) (+) 2 to (+) 51 inches

D. (1) transition to 1-AOI-100-1
(2) (-) 50 to (-) 100 inches

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B)*. The second part is incorrect but plausible if the candidate only considers the last report of Reactor Power to assume they are to direct the restoration of the normal band for Reactor Water Level to (+) 2 to (+) 51 inches.

B **CORRECT**: In accordance with NOTE 1 from 1-EOI-1, RPV Control, and 1-EOI-1A, ATWS RPV Control, was not met during the SCRAM report from 1-AOI-100-1, Reactor SCRAM. Therefore, boron was injected with Reactor Power above 5% in accordance with the Immediate ATWS Actions of AOI-100-1. The Reactor Power leg of 1-EOI-1A is only to be exited if the Reactor is subcritical AND no boron has been injected. Given, Control Rods are out and SLC was initiated, the ATWS Actions require lowering Reactor Water Level to (-) 50 inches which coincides with 1-EOI-1A. For second part, since ATWS Actions were taken and Reactor Water Level was lowered to (-) 50 inches, the last report at 0920 indicates that none of the conditions in EOI-1A, Table Q-1 exist. In accordance with BFN-ODM-4.20, the NUSO will direct a Reactor Water Level band of (-) 50 to (-) 100 inches.

C INCORRECT: The first part is incorrect but plausible in that with the Immediate Actions (which includes the ATWS Actions for Reactor Power being above 5%) of 1-AOI-100-1 being complete, when at 0920, given APRMs indicate 0% Reactor Power or subcritical with IRMs indicating downscale on Range 2 will allow exiting the Reactor Power control leg of 1-EOI-1A and transition to 1-AOI-100-1. Second part is incorrect but plausible *(See A).*

D INCORRECT: The first part is incorrect (*See C*). Second part is correct (*See B*).

SRO Level Justification: Test the candidate's knowledge of low Reactor Power ATWS conditions as it relates to Immediate Actions and EOI requirements as mitigation strategies. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome especially without a given reference.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (3) The progression of an event.

Technical Reference(s): 1-EOI-1A, Rev. 2 (Attach if not previously provided)
1-AOI-100-1, Rev 26
BFN-ODM- 4.20, Rev 6

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.202 Obj. 13 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	**X**
10 CFR Part 55 Content:	55.41	
	55.43	**X**

Comments:

Excerpts from 1-AOI-100-1:

BFN Unit 1	Reactor Scram	1-AOI-100-1 Rev. 0026 Page 5 of 78

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **DEPRESS** REACTOR SCRAM A and B, 1-HS-99-5A/S3A and 1-HS-99-5A/S3B, on Panel 1-9-5.

[2] **PLACE** REACTOR MODE SWITCH, 1-HS-99-5A/S1, in SHUTDOWN.

[3] **IF** all control rods can NOT be verified fully inserted, **THEN**

INITIATE ARI. (otherwise **MARK** N/A).

[4] **IF** Reactor Power is 5% or BELOW, **THEN:** (otherwise **MARK** N/A)

REPORT the following to the Unit SRO:

- Reactor Scram
- Mode Switch is in Shutdown
- "All rods in" or "rods out "
- Reactor Water Level and trend (recovering or lowering)
- Reactor pressure and trend
- MSIV position (Open or Closed)
- Power level

ATWS Immediate Actions:

BFN Unit 1	Reactor Scram	1-AOI-100-1 Rev. 0026 Page 6 of 78

4.1 Immediate Actions (continued)

NOTES
1) Perform steps 4.1[5.3] and 4.1[5.4] in parallel.
2) Step 4.1[5.8] should be reported IMMEDIATELY when that condition is reached.
3) Step 4.1[5.9] may be performed before step 4.1[5.8] if Reactor Water Level is slowly lowering.

[5] **IF** Reactor Power is ABOVE 5% or unknown, **THEN**

PERFORM the following:
(otherwise **MARK** steps N/A).

[5.1] **REPORT** the following to the Unit SRO:

- REACTOR SCRAM, Mode Switch in SHUTDOWN
- Control Rods out
- Reactor power
- Continuing with ATWS Actions

[5.2] **ENSURE** shutdown both Recirc. Pumps

[5.3] **STOP** and **PREVENT** injection from CONDENSATE and FEEDWATER per APP 4

[5.4] **STOP** and **PREVENT** injection from HPCI, RHR, and CS per APP 4.

[5.5] **INITIATE** SLC and **ENSURE** injection.

[5.6] **INHIBIT** ADS LOGIC.

[5.7] **BYPASS** Group 1 RPV Low-Low-Low Level isolation interlocks per APP 8A.

[5.8] **WHEN** Reactor Water Level reaches -50 inches, **THEN**

REPORT Reactor Water Level AND Reactor Power to Unit SRO.

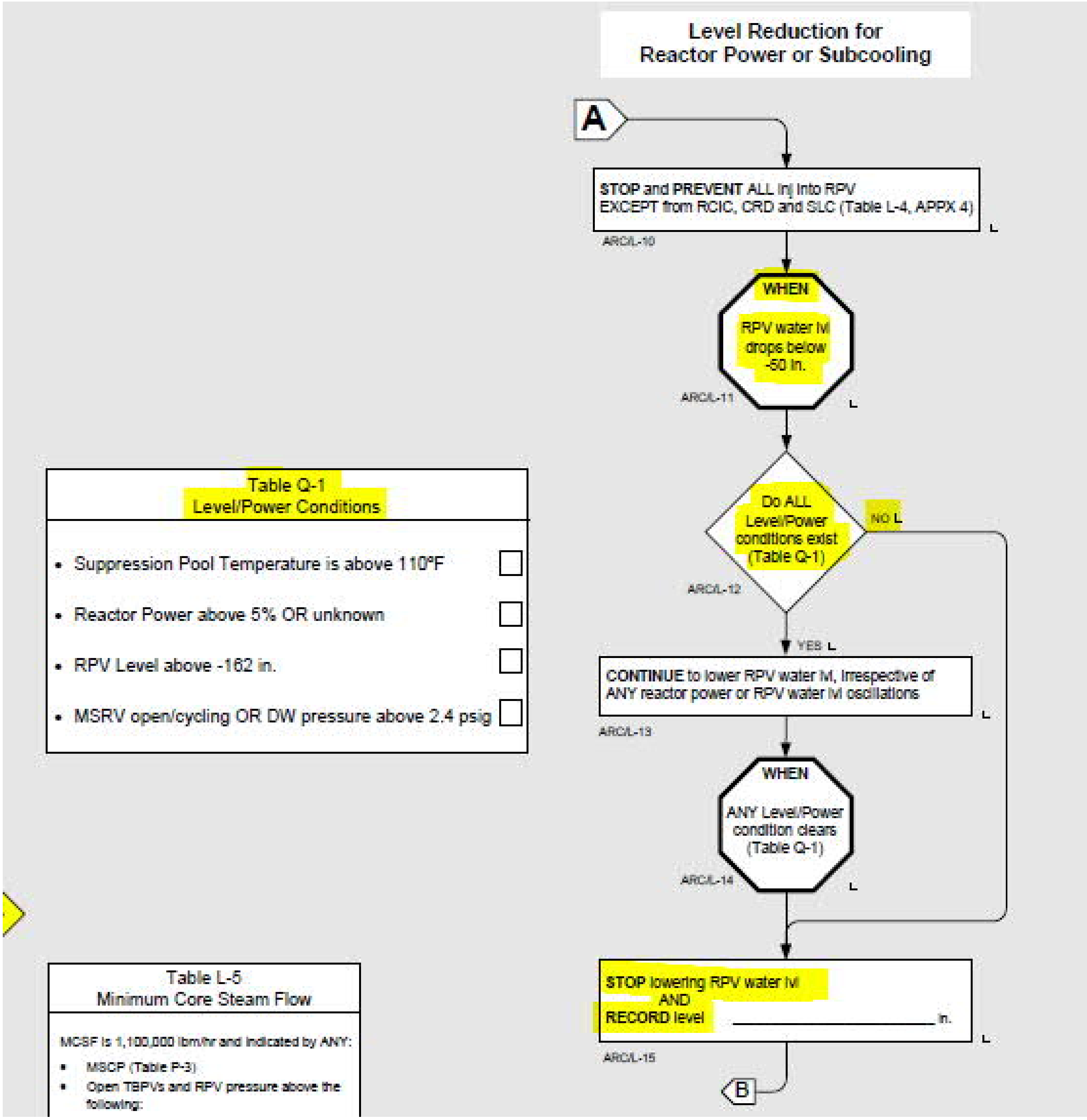
BFN Unit 1	Reactor Scram	1-AOI-100-1 Rev. 0026 Page 7 of 78

4.1 Immediate Actions (continued)

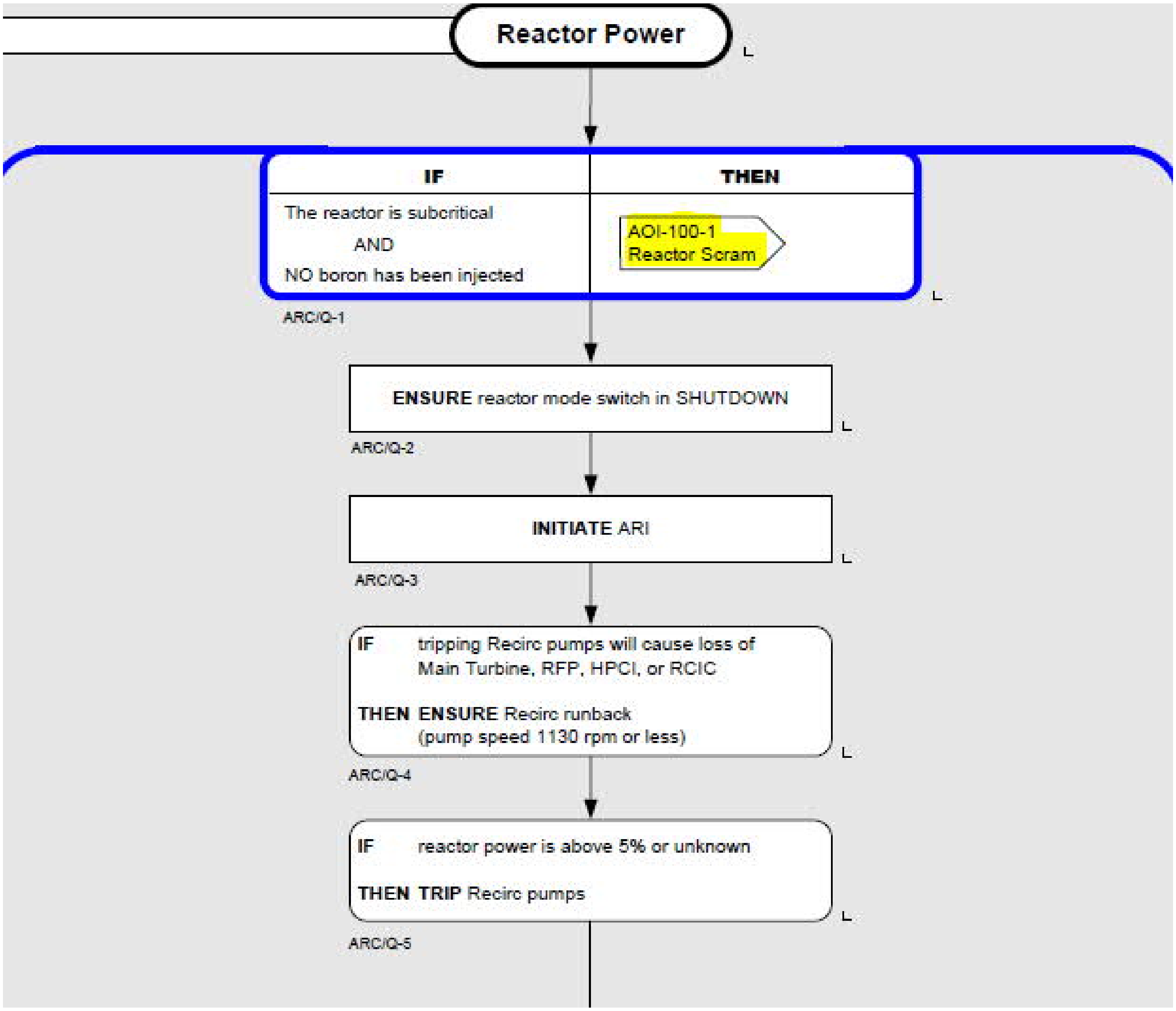
[5.9] **REPORT** "ATWS Actions Complete".

- Reactor pressure and trend.
- MSIV position.
- Power level.
- SLC IS/IS NOT Injecting.

Excerpts from 1-EOI-1A:



Supports Distractors (C1), (D1):



Excerpt from BFN-ODM-4.20:

BFN Operations Directive Manual	Strategies for Successful Transient Mitigation	BFN-ODM-4.20 Rev. 0006 Page 20 of 25

4.8.4 ATWS RPV Control (EOI-1A) (continued)

The strategy is for the NUSO to monitor the OATC and BOP Immediate Operator Actions from AOI-100-1,proceed to the Level leg of EOI-1A, and be cognizant of the conditions in table Q-1. When the OATC reports "Reactor Water Level is -50" and Reactor Power, The NUSO should be prepared to direct a reactor water level band of -50" to -100" if conditions allow. If ALL the conditions of Table Q-1 are met, then the NUSO should direct the UO to continue to lower Reactor water level until one of the conditions of Table Q-1 is cleared. Once the OATC reports "ATWS Actions complete" the NUSO should verify the actions performed in EOI-1A flow chart and prosecute the event as required by the EOI's.

B. When EOI-1A, Step ARC/Q-7 is reached, IF core oscillations are observed, THEN INITIATE SLC.

C. When EOI-1A, Step ARC/Q-8 is reached, IF reactor power is greater than APRM downscale, THEN INITIATE SLC.

Examination Outline Cross-reference:	Level	RO	SRO
295010 (APE 10) High Drywell Pressure / 5	Tier #	------	1
AA2.01 **(10CFR 55.43.5 - SRO Only)**	Group #	------	2
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:	K/A #	295010AA2.01	
• Leak rates	Importance Rating	------	3.8

Proposed Question: **# 83**

Unit 2 is operating at 100% RTP when a leak in the Drywell results in the following conditions:

- Manual Reactor SCRAM
- Drywell Pressure rises to 5.6 psig
- Drywell Floor Drain Integrators indicate a leak rate of 15 gpm

The **HIGHEST** required Emergency Classification Action Level (EAL) to report is a/an ___**(1)**___ and the NRC Notification must **NOT** exceed ___**(2)**___.

Note: SED judgement shall **NOT** be used as a basis for classification

[REFERENCE PROVIDED]

A. (1) NOUE (SU4)
(2) 30 minutes

B. (1) NOUE (SU4)
(2) 60 minutes

C. (1) ALERT (FA1)
(2) 30 minutes

D. (1) ALERT (FA1)
(2) 60 minutes

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that a NOUE does exist since unidentified leakage has been greater than 10 gpm if the candidate misapplies 'for 15 minutes or longer', however it is **not** the highest classification. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible if the candidate applies the rule that the Site Emergency Director has 15 minutes to classify the event and then 15 minutes to notify the State.

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT:** *(See attached)* In accordance with EPIP-1 (under FISSION PRODUCT BARRIERS), with Drywell Pressure at or above 2.45 psig with indications of a leak in the Drywell meets the 1 of 2 criteria for ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS Barrier. This results in the HIGHEST EAL of ALERT (FA1). For second part, in accordance with EPIP-3, the Notification of the NRC is required to be completed as soon as possible, not to exceed 60 minutes from classification declaration.

SRO Level Justification: Tests the candidate's ability to interpret leak rates as it relates to High Drywell Pressure and applying Emergency Classification Action Level requirements. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (2) Diagnosis that leads to selection of procedures that should be used to respond to the evolution.

Technical Reference(s): EPIP-1, Rev. 59 (Attach if not previously provided)
EPIP-3, Rev. 42

Proposed references to be provided to applicants during examination: **EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3**

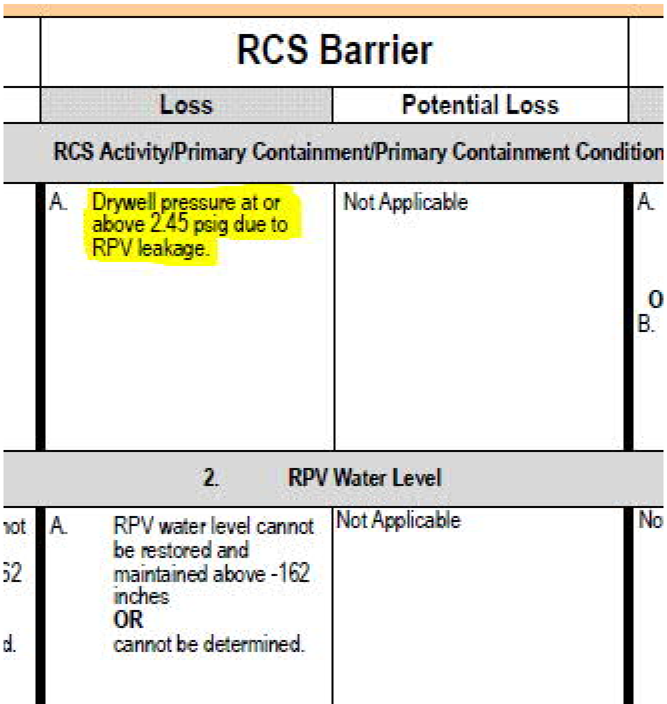
Learning Objective: OPL171.075 Obj. 2 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

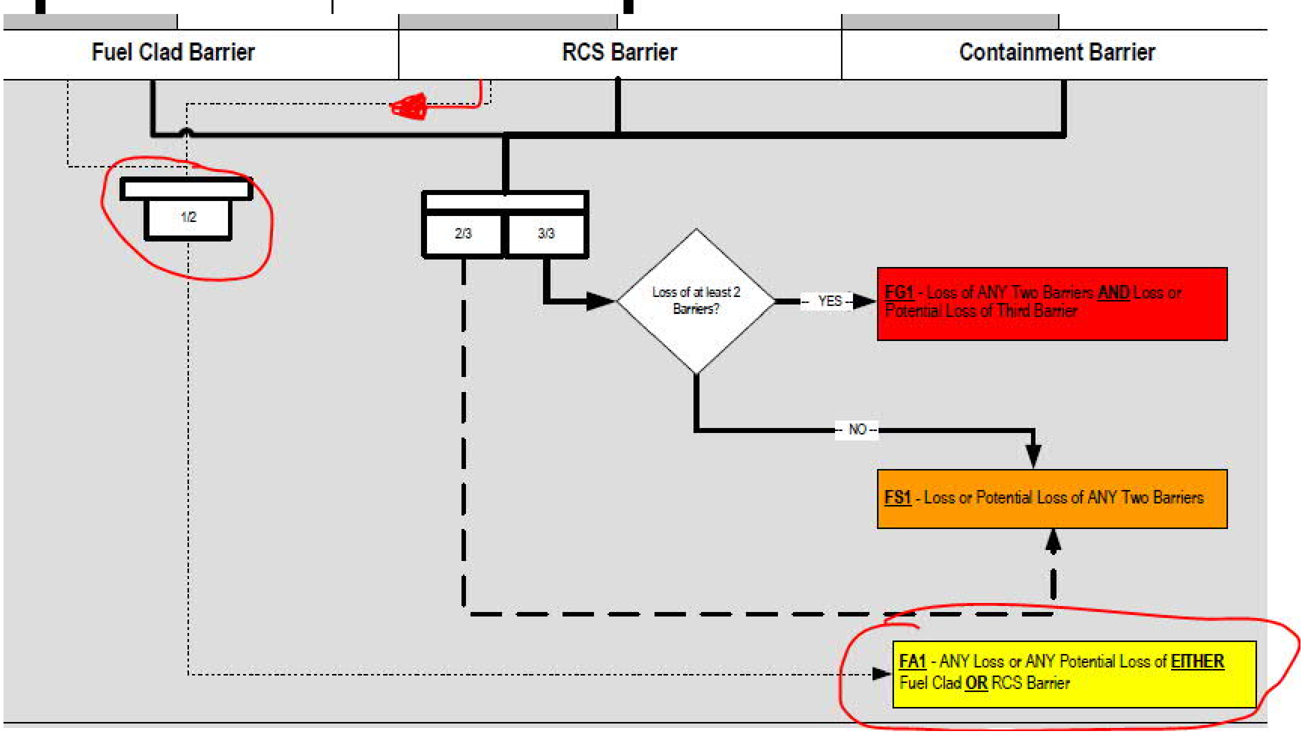
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41
55.43 **X**

Excerpts from EPIP-1:



RCS Barrier	
Loss	**Potential Loss**
RCS Activity/Primary Containment/Primary Containment Condition	
A. Drywell pressure at or above 2.45 psig due to RPV leakage.	Not Applicable
2. RPV Water Level	
A. RPV water level cannot be restored and maintained above -162 inches **OR** cannot be determined.	Not Applicable



SU4 - RCS Leakage for 15 minutes or longer.

The SED should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.

OR

(2) Identified leakage greater than 30 gpm for 15 minutes or longer.

OR

(3) Leakage from the RCS to a location outside containment greater than 30 gpm for 15 minutes or longer.

SU5 - Automatic or manual scram fails to shutdown the reactor. **Applicable in Mode 1 & 2 ONLY**

Excerpts from EPIP-3:

BFN Unit 0	ALERT	EPIP-3 Rev. 0042 Page 4 of 29

1.0 INTRODUCTION

1.1 Purpose

The purpose of this procedure is to provide for the timely notification of appropriate individuals or organizations when the Shift Manager (SM) or the Site Emergency Director (SED) has determined through the use of EPIP-1, "Emergency Classification Procedure," that an event has occurred which is classified as an Alert. Additionally, this procedure provides for periodic evaluation of the current situation by the SM/SED to determine whether the Alert should be terminated, continued or upgraded to a higher emergency classification.

Upon completion of classification utilizing EPIP-1, all initial classification steps are conducted from the body of this instruction.

The steps of this procedure can be completed concurrently. The order or sequencing of the steps is suggested but if the step cannot be completed timely or is delegated, continue through all steps, periodically reviewing all steps not completed. Continue review of the procedure until all steps are completed. Two functions of this procedure are "timed actions." These time critical actions are "Notification of the State" (as soon as possible, within 15 minutes from classification declaration) and "Notification of the NRC" (as soon as possible, not to exceed 60 minutes from classification declaration).

BFN Unit 0	ALERT	EPIP-3 Rev. 0042 Page 9 of 29

3.4 Notification of The Nuclear Regulatory Commission (NRC)

NOTE

Notification of the NRC is required to be completed as soon as possible, not to exceed 60 minutes from classification declaration.

[1] **COMPLETE** Appendix C, "Notification of the Nuclear Regulatory Commission (NRC) (NRC Event Notification Worksheet)."

[2] **COMPLETE** Appendix D, "Notification of Site Personnel."

3.5 Maintaining Communications with the NRC

NOTE

When the TSC is staffed, the open and continuous line of communications with the NRC may be transferred to the NRC Coordinator position.

[1] **IF** requested by the NRC, **THEN**

DIRECT a member of the Operations staff (SRO if available) to maintain an open and continuous line of communications as directed by NRC.

3.6 Monitor/Re-Evaluate the Event

Examination Outline Cross-reference:

295015 (APE 15) Incomplete SCRAM/1

AA2.02 **(10CFR 55.43.5 - SRO Only)**

Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM:

- Control rod position

	RO	SRO
Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295015AA2.02	
Importance Rating	------	4.2*

Proposed Question: **# 84**

Unit 2 is operating at 100% RTP when the following conditions occur:

- Reactor SCRAM
- Multiple Control Rods fail to insert
- Reactor Power is 8%
- **ALL 185** SCRAM inlet/outlet blue lights are illuminated

Given the conditions above, which **ONE** of the following completes the statements below?

In accordance with 2-EOI-1A, ATWS RPV CONTROL, the Recirc Pumps ___**(1)**___ required to be tripped.

To mitigate this condition, the SRO will direct ___**(2)**___.

Note: 2-EOI Appendix-1B, Venting and Repressurizing the SCRAM Pilot Air Header

2-EOI Appendix-1F, Manual SCRAM

A. (1) are
(2) 2-EOI Appendix-1B

B. (1) are
(2) 2-EOI Appendix-1F

C. (1) are NOT
(2) 2-EOI Appendix-1B

D. (1) are NOT
(2) 2-EOI Appendix-1F

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B)*. The second part is incorrect but plausible in that there are numerous EOI Appendices that are available for Control Rod insertion in 2-EOI-1A, and having to recall information from a table on an EOI Flowchart without a reference makes any of them plausible choices. ATWS RPV Control; 2-EOI-Appendix 1B is plausible in that this EOI Appendix is used in the event the SCRAM Valves failed to open, however the candidate may believe that venting the SCRAM Pilot Air Header would release a hydraulic lock situation in the SCRAM Discharge Volume.

B **CORRECT**: (*See attached*) In accordance with 2-EOI-1A, RPV ATWS Control, if Reactor Power is greater than 5% following the SCRAM, the Recirc Pumps are required to be tripped. 2-AOI-100-1 Immediate Actions for ATWS conditions also require the Recirc Pumps to be tripped. For second part, to mitigate this condition, 2-EOI-Appendix-1F would be directed in accordance with 2-EOI-1A, if a SCRAM Discharge Volume is full. This is indicated by the given 'ALL 185 SCRAM inlet/outlet blue lights are illuminated'.

C INCORRECT: First part is incorrect but plausible if the candidate fails to recall the exact Reactor Power level from memory at which EOI-1A directs the Recirc Pumps to be tripped. This would lead a candidate to conclude that Recirc Pumps are not required to be tripped. Second part is incorrect but plausible *(See A).*

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

SRO Level Justification: Tests the candidate's ability to interpret Control Rod Positions following a Hydraulic ATWS and procedurally mitigate the failure of Control Rods to insert. To correctly answer this question, candidate must recognize proper core orientation as it relates to the SCRAM Discharge Volume alignment. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (2) Diagnosis that leads to selection of procedures that should be used to respond to the evolution.

Technical Reference(s): 2-EOI-1A, Rev. 2 (Attach if not previously provided)
2-AOI-100-1, Rev. 115
OPL171.005, Rev. 22

Proposed references to be provided to applicants during examination: NONE

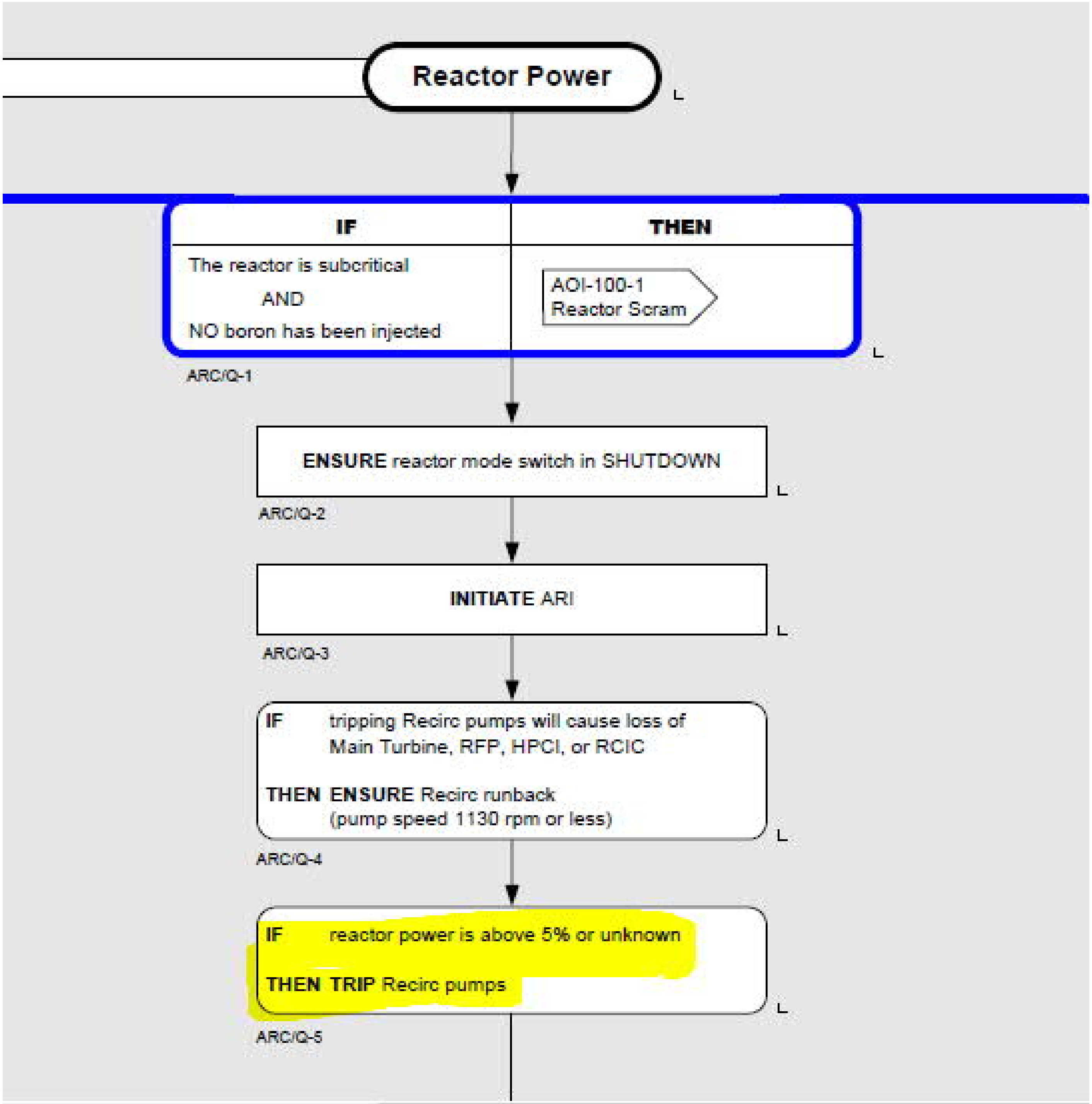
Learning Objective: OPL171.202, Obj. 20 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	**X**
10 CFR Part 55 Content:	55.41	
	55.43	**X**

Comments:

Excerpts from 2-EOI-1A:



Control Rod Insertion

RESET ARI
DEFEAT ARI logic trips if necessary (APPX 2)

ARC/Q-12

INSERT control rods using ANY Alternate Control Rod Insertion Methods (Table Q-2)

ARC/Q-13

Table Q-2
Alternate Control Rod Insertion Methods

CONDITIONS	METHODS	APPX
Scram valves failed to open	**DEENERGIZE** scram solenoids	1A
	VENT scram air header	1B
Scram valves opened but SDV is full	1. **RESET** scram **DEFEAT** RPS logic if necessary 2. **DRAIN** SDV 3. **RECHARGE** accumulators 4. **INITIATE** scram	1F
Manual control rod insertion methods	**DRIVE** control rods **BYPASS** RWM and **RAISE** CRD drive water differential pressure if necessary	1D
	RAISE CRD cooling water header dp	1G
	SCRAM individual control rods	1C
	VENT control rod over piston volumes	1E

Excerpt from 2-AOI-100-1:

BFN Unit 2	Reactor Scram	2-AOI-100-1 Rev. 0115 Page 6 of 77

4.1 Immediate Actions (continued)

NOTES
1) Perform steps 4.1[5.3] and 4.1[5.4] in parallel.
2) Step 4.1[5.8] should be reported IMMEDIATELY when that condition is reached.
3) Step 4.1[5.9] may be performed before step 4.1[5.8] if Reactor Water Level is slowly lowering.

[5] **IF** Reactor Power is ABOVE 5% or unknown, **THEN**

PERFORM the following:
(otherwise **MARK** steps N/A).

[5.1] **REPORT** the following to the US:

- REACTOR SCRAM, Mode Switch in SHUTDOWN
- Control Rods out
- Reactor power
- Continuing with ATWS Actions

[5.2] **ENSURE** shutdown both Recirc. Pumps

[5.3] **STOP** and **PREVENT** injection from CONDENSATE and FEEDWATER per APP 4

[5.4] **STOP** and **PREVENT** injection from HPCI, RHR, and CS per APP 4.

[5.5] **INITIATE** SLC and **ENSURE** injection. ☐

[5.6] **INHIBIT** ADS LOGIC.

[5.7] **BYPASS** Group 1 RPV Low-Low-Low Level isolation interlocks per APP 8A.

[5.8] **WHEN** Reactor Water Level reaches -50 inches, **THEN**

REPORT Reactor Water Level AND Reactor Power to US.

Excerpt from OPL171.005 Lesson Plan:

OPL171.005, Control Rod Drive (CRD) Hydraulics Rev. 22

f. Scram operation	Figure-13
1) Two scram pilot air valves for each HCU are directly connected to the air system so that the inlet and outlet diaphragm type scram valves open in response to scram signals as air is bled from their diaphragms.	
2) The inlet scram valve, when open, permits the scram accumulator to supply the initial energy to rapidly insert the control rod.	Obj. ILT 6.b,Obj 4.b Obj. NLOR6.b Obj. NLO 3.b (OF5)
3) The outlet scram valve, when open, permits water vented from the over piston area of the CRD to exhaust to the scram discharge volume.	
4) Scram pilot air valves	Obj. ILT 6.c Obj. LOR 4.c Obj. NLOR 6.c Obj. NLO 3.c
a) Quantity: Two per HCU (Modification is in progress replacing separate valve bodies with a single valve body with two solenoid valves inside.)	Figure-13
b) Type: 3-way solenoid-operated valves	If either scram p valve deenergiz (open), the rema closed valve kee on the scram inlet/outlet valve
c) Power is supplied by 120VAC Reactor Protection System power, one valve's solenoid powered by RPS Bus A and the other by Bus B.	
d) The solenoid valves are normally energized, routing air to the inlet and outlet scram valve diaphragms to hold the valves closed.	Industry OE – individual rods h scrammed on lo one RPS bus because other solenoid valve d energized becau undetected blow fuse.
e) Note: the single body valves function the same way except the solenoid valves operate a third internal 3 way relay valve to actually supply and vent the scram valve air.	
f) When a full scram occurs, both solenoid valves de-energize to vent air from the inlet and outlet scram valve diaphragms, causing the valves to open	Obj. ILT 6.a(OF! Obj. LOR 4.a Obj. NLOR 6.a Obj. NLO 3.a
5) Scram inlet and outlet valves	Q: What would happen to the ro the outlet scram doesn't open?
a) Both are globe valves with Teflon seats, to minimize leakage.	
b) Normal lineup consists of both air-operated valves held closed by air pressure from the instrument air header.	A: Rod would n scram and exce pressure may bu the index tube.
c) The scram inlet and outlet valves start to open within 0.15 seconds after the pilot air valves lose voltage.	

OPL171.005, Control Rod Drive (CRD) Hydraulics Rev. 22

d) When a scram occurs, both valves open by internal spring pressure on loss of control air pressure, after the scram pilot air valves open to vent.

e) Sequencing is required to prevent buildup of high pressures in the control rod drive, so the outlet valves open slightly ahead of the inlet valves. The outlet valves have stronger opening spring tension.

Obj. ILT 7
Obj. NLO

f) Position indication is provided by means of spring-mounted position switches. When both valves open, position switches causing a blue rod scram light to be illuminated on the Rod Status Display, Panel 9-5.

Examination Outline Cross-reference:

295029 (EPE 6) High Suppression Pool Water Level / 5

G2.2.22 (10CFR 55.43.2 - SRO Only)

Knowledge of limiting conditions for operations and safety limits.

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295029G2.2.22	
Importance Rating	------	4.7

Proposed Question: **# 85**

Unit 2 is in MODE 1 at 100% RTP when a leak into the Suppression Pool has resulted in the following condition:

- At 0200, Suppression Pool Level is (-) 3 inches and rising at 1 inch per hour

Given the condition above, which **ONE** of the following completes the statements below?

The Tech Spec Limit for 3.6.2.2, Suppression Pool Level, will be reached at ___**(1)**___.

The Bases of the Tech Spec Suppression Pool upper level limit is to prevent ___**(2)**___ during a DBA LOCA.

A. (1) 0315
(2) inoperability of the Drywell-Suppression Chamber Vacuum Breakers

B. (1) 0315
(2) excessive clearing loads from MSRV discharges and excessive pool swell loads

C. (1) 0400
(2) inoperability of the Drywell-Suppression Chamber Vacuum Breakers

D. (1) 0400
(2) excessive clearing loads from MSRV discharges and excessive pool swell loads

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the Suppression Chamber Water Level Abnormal alarm will be received at 0315 when greater than or equal to (-) 1.75 inches. Second part is incorrect but plausible in that this is a concern of higher Suppression Chamber Water Level as it relates to Emergency Depressurization from the point of covering the Drywell-Suppression Chamber Vacuum Breakers in EOI-2, Primary Containment Control.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is correct *(See D).*

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT:** *(See attached)* In accordance with Tech Spec 3.6.2.2, Suppression Pool Water Level, with the given condition, the upper limit of less than or equal to (-) 1.0 inches will be reached at 0400. For second part, in accordance with Tech Spec 3.6.2.2 Bases, Suppression Pool Water upper level limit is to prevent excessive clearing loads from SRV discharges and excessive pool swell loads during a DBA LOCA.

SRO Level Justification: Tests the candidate's knowledge of High Suppression Pool Water Level as it relates to the Technical Specification and Bases. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the technical specifications and their bases. This question is rated as Memory due to the requirement to strictly recall procedural facts.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, AOPs, EOPs, and their associated bases documents. (3) The progression of an event.

Technical Reference(s): Unit 2 Tech Spec 3.6.2.2, Amend. 253 (Attach if not previously provided)
Unit 2 Tech Spec Bases 3.6.2.2, Rev. 0
2-ARP-9-3B, Rev. 38
2-EOI-2, Rev. 16
EOIPM, 0-V-D, Rev. 2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.016 Obj. 9 (As available)

Question Source:	Bank #	BFN 1102 #84	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2011	

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Copy of Bank Question:

Proposed Question: **# 84**

A leak into Unit 2 Suppression Pool has resulted in the following indications:

- At 0200 Suppression Pool Level is (-) 3 inches and rising at 1 inch per hour

Which ONE of the following completes the statements below?

The Tech Spec Limit for 3.6.2.2, "Suppression Pool Level," will be reached at __**(1)**__.

The bases of the Tech Spec Suppression Pool upper level limit is to __**(2)**__ during a DBA LOCA.

A. **(1)** 0315
(2) ensure that peak primary containment pressure does not exceed maximum allowable values

B. **(1)** 0315
(2) prevent excessive clearing loads from S/RV discharges and excessive pool swell loads

C. **(1)** 0400
(2) ensure that peak primary containment pressure does not exceed maximum allowable values

D. **(1)** 0400
(2) prevent excessive clearing loads from S/RV discharges and excessive pool swell loads

Proposed Answer: **D**

Excerpt from Unit 2 Tech Spec 3.6.2.2:

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be ≥ -6.25 inches with and -7.25 inches without differential pressure control and ≤ -1.0 inches.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool water level not within limits.	A.1 Restore suppression pool water level to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

Excerpt from Unit 2 Tech Spec Bases 3.6.2.2:

BASES

BACKGROUND (continued)

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the S/RV quenchers, main vents, or HPCI and RCIC turbine exhaust lines. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

If the suppression pool water level is too high, it could result in excessive clearing loads from S/RV discharges and excessive pool swell loads during a DBA LOCA. Therefore, a maximum pool water level is specified. This LCO specifies an acceptable range to prevent the suppression pool water level from being either too high or too low.

APPLICABLE SAFETY ANALYSES

Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to S/RV discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of Reference 1 remains valid.

Suppression pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement (Ref. 2).

(continued)

Excerpt from 2-ARP-9-3B: Supports Distractors (A1), (B1)

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0038 Page 19 of 39

SUPPR CHAMBER
WATER LEVEL
ABNORMAL
2-LA-64-54A

15

(Page 1 of 1)

<u>Sensor/Trip Point:</u>

LT-64-54 $\leq$ -5.5" H_2O

$\geq$ -1.75" H_2O

Sensor Location: RX Bldg, El 519'
NW corner room just inside door

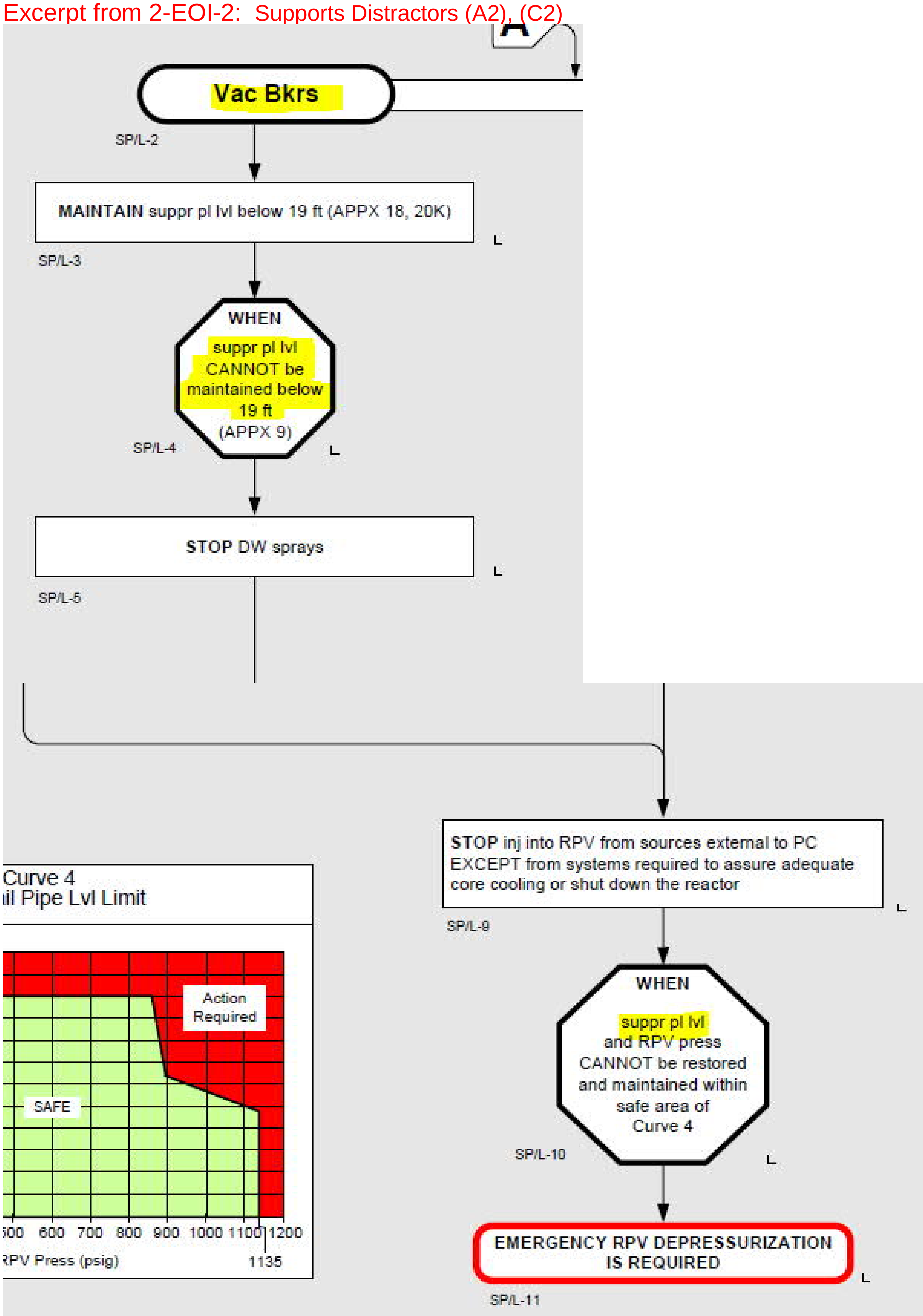
Probable Cause:
A. Suppression Chamber water level abnormal.
B. Placing Suppression Pool Cooling in service
C. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **CHECK** Suppression Pool level using multiple indications.
B. **IF** level is low, **THEN**
   DISPATCH personnel to check for leaks.
C. **IF** level is high, **THEN**
   CHECK for RCIC, HPCI, Core Spray, or RHR draining to Suppression Pool, and **CHECK** 2-TR-64-161 and -162.
D. **REFER TO** 2-OI-74, Section 8.0.
E. **REFER TO** Tech Spec 3.6.2.2.
F. **IF** level is above -1" or below -6.25" **AND NOT** in Mode 4 or Mode 5 **THEN** (otherwise N/A)
   ENTER 2-EOI-2 Flowchart.
G. **IF** level is above -1" or below -6.25" **AND** in Mode 4 or Mode 5 **THEN** (otherwise N/A)
   1. **EVALUATE** plant conditions to **DETERMINE** if 2-EOI-2 entry is appropriate.
   2. **RECORD** actions in NOMS log.

References: 2-45E620-3 2-47E610-64-1 GE 730E943-1
Technical Specifications 3.6.2.2

Excerpt from 2-EOI-2: Supports Distractors (A2), (C2)



Excerpt from EOIPM, 0-V-D: Supports Distractors (A2), (C2)

BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V-D Rev. 0002 Page 103 of 119

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: SP/L-3, SP/L-4, SP/L-5

If the suppression chamber-to-drywell vacuum breaker penetrations are submerged, the vacuum breakers cannot function as designed to relieve noncondensibles into the drywell and equalize drywell and suppression chamber pressures. Suppression pool water level must therefore be maintained below the bottom of the vacuum breaker openings to permit initiation and operation of drywell sprays.

The vacuum breakers are connected to the downcomer ring header. The drywell side of the vacuum breaker is significantly higher than the wetwell side. As drywell pressure decreases below wetwell pressure, water will be drawn up inside the downcomers until the vacuum breakers begin to open. The differential pressure required to open the vacuum breakers, however, is sufficiently low to prevent water in the downcomer from rising so high that it contacts the drywell side of the vacuum breaker, interfering with valve opening. The suppression pool water level corresponding to the bottom of the vacuum breaker openings, therefore, needs no adjustment to compensate for the valve opening differential pressure.

If suppression pool water level cannot be maintained below the specified elevation, operation of drywell sprays is terminated since post spray drywell vacuum relief cannot be ensured with the vacuum breaker openings submerged.

The flowpath proceeds to Step SP/L-9 to terminate injection into the RPV from sources external to the primary containment to prevent further increase in suppression pool water level. (Refer to the discussion Step SP/L-9.)

Examination Outline Cross-reference:	Level	RO	SRO
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray	Tier #	------	2
A2.06 (10CFR 55.43.2 - SRO Only)	Group #	------	1
Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	209001A2.06	
• Inadequate system flow	Importance Rating	------	3.2

Proposed Question: **# 86**

Unit 2 is operating at 100% RTP and **2A** Core Spray Pump has just been started for 2-SR-3.5.1.6(CS I), Core Spray Flow Rate Loop I, with the following conditions:

- 2-FCV-75-22, CORE SPRAY SYS I TEST VALVE, is currently being THROTTLED OPEN
- 2-FCV-75-25, CORE SPRAY SYS I INBD INJECT VALVE, indicates CLOSED
- 2-FI-75-21, CS SYS I FLOW, indicates 1500 gpm

Which **ONE** of the following completes the statements below?

Given the flowrate, the **2A** Core Spray Pump ___**(1)**___ operate continuously in accordance with 2-OI-75, Core Spray System.

When **2C** Core Spray Pump is started, 2-FI-75-21, CS SYS I FLOW now indicates 6350 gpm; Loop I Core Spray ___**(2)**___ Tech Spec OPERABILITY requirements.

A. (1) can
(2) meets

B. (1) can
(2) does NOT meet

C. (1) can NOT
(2) meets

D. (1) can NOT
(2) does NOT meet

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with 2-OI-75, Core Spray System, Precaution and Limitations state that Core Spray Minimum Flow Valves receive a closure signal when flow is approximately 2600 gpm rising and receives an open signal when flow lowers to approximately 2200 gpm. Given that the 2A Core Spray Pump was just started, the flowrate would be rising with the minimum flow valve still open until 2600 gpm. Core Spray Pumps may be operated continuously on minimum flow without any impact to pump performance as long as the minimum flow is OPEN. For second part, in accordance with the given SR in the question, the rated flowrate that must be achieved is 6250 gpm with two pumps in operation in order to meet Tech Spec 3.5.1.6.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible in that the stated flow rate achieved is greater than the minimum Tech Spec required. Since multiple ECCS Systems exist with numerous flowrates, the candidate could confuse the specific flowrate.

C INCORRECT: First part is incorrect but plausible in that the candidate could easily confuse the Core Spray minimum flowrates with RHR Minimum Flow valves that open (after a 10 second time delay) and close on a low flow of 5800 gpm. To prevent excessive vibration, RHR Pumps should NOT be allowed to operate for more than 3 minutes at minimum flow. Second part is correct *(See A)*.

D INCORRECT: First part is incorrect but plausible *(See C)*. Second part is incorrect *(See B)*.

SRO Level Justification: Tests the candidate's ability to recognize the low flowrate condition of the Core Spray Pumps in relation to their minimum flowrate requirements. Additionally, the SRO candidate must recognize the Tech Spec limit from surveillance requirements for Core Spray flow that coincides with Adequate Core Cooling in the event of an accident. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): (Attach if not previously provided)
- 2-OI-74, Rev. 184
- 2-OI-75, Rev. 117
- Unit 2 Tech Spec 3.5.1, Amend. 325
- Unit 2 Tech Spec Bases 3.5.1, Rev. 81

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.045 Obj. 6 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1804 #86	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	X

Comments:

Copy of Bank Question:

Proposed Question: **# 86**

2A Core Spray Pump has just been started for 2-SR-3.5.1.6(CS I), Core Spray Flow Rate Loop I, with the following conditions:

- 2-FCV-75-22, CORE SPRAY SYS I TEST VALVE, indicates FULL **OPEN**
- 2-FCV-75-25, CORE SPRAY SYS I INBD INJECT VALVE, indicates **CLOSED**
- 2-FI-75-21, CS SYS I FLOW, indicates 2300 gpm

Which of the following completes the statements below?

Given the flowrate, the 2A Core Spray Pump ___(1)____operate continuously in accordance with 2-OI-75, Core Spray System.

When **2C** Core Spray Pump is started, 2-FI-75-21, CS SYS I FLOW now indicates 5000 gpm; Loop I Core Spray___(2)____ Tech Spec OPERABILITY requirements.

A. **(1)** can
(2) meets

B. **(1)** can
(2) does not meet

C. **(1)** cannot
(2) meets

D. **(1)** cannot
(2) does not meet

Proposed Answer: **B**

Excerpts from 2-OI-75:

BFN Unit 2	Core Spray System	2-OI-75 Rev. 0117 Page 12 of 152

3.2 Operability and LCO's (continued)

D. The BFN Generic Letter (GL) 89-10 Program has excluded 2-FCV-075-0023 and -0051 from its program maintenance requirements. Consequently, administrative actions must be taken to ensure Core Spray is declared inoperable and a limiting condition for operation (LCO) entered if this valve is repositioned from its normal (open) position when the unit is in Mode 1, 2, or 3.

3.3 Equipment

A. Core Spray Pump motor full-load current is 80 amps. This corresponds to rated flow conditions.

B. Core Spray Pumps may be operated continuously on minimum flow without any impact to pump performance as long as the minimum flow is open. Minimum flow lines are orificed to pass approximately 20 % rated pump flow. When operating pumps with the minimum flow valves closed or manually isolated, the following limits must be observed:

1. Core Spray Pumps may be operated continuously for up to 5 minutes between 150 and 300 gpm per pump per loop. Operating Core Spray Pumps at this flow rate greater than this time frame may cause pump degradation.

2. Core Spray Pumps may be operated continuously for up to 4 hours between 300 and 600 gpm per pump per loop. Operating Core Spray Pumps at this flow rate greater than this time frame may cause pump degradation.

3. Core Spray Pumps may be operated continuously at greater than 600 gpm per pump per loop with no restriction on time limitations.

C. Care must be taken to insure the Core Spray suction valves are in the open position. The pumps are capable of starting with the suction valves closed.

D. Placing the Core Spray Pump(s) handswitch to the normal-after start position as soon as possible after Core Spray Pump(s) auto start will ensure the hand switch disagreement light and the pump tripped annunciator function as designed.

E. CS Motor Heaters are for humidity control in the CS Pump Motor winding insulation systems and minimize the potential of corrosion on the internal motor components. The motor heaters serve no safety-related function. Therefore, the loss of the motor heater will NOT prevent the CS Pumps from performing their intended design related function. However, due to the potential to shorten the life of the motor, and its qualified insulation system, the time period these heaters are removed from service should be minimal.

BFN Unit 2	Core Spray System	2-OI-75 Rev. 0117 Page 13 of 152

3.3 Equipment (continued)

F. Leakage of Suppression Pool quality water into the RPV may occur when Core Spray System pressure is above RPV pressure due to a 1/4 inch hole drilled into the outlet side disc face of CORE SPRAY SYS I(II) INBD INJECT VALVE, 2-FCV-75-25(53) to eliminate pressure locking concerns associated with these valves.

G. The preferred suction source for Reactor Cavity floodup is Section 8.24 which utilizes the CST Standpipe. (PER 245792).

3.4 Initiations

A. The CS System will auto initiate from the following signals:

1. RPV water level at or below -122 inches.

2. DW pressure at or above 2.45 psig and RPV pressure at or below 450 psig.

B. Manually stopping a Core Spray Pump after auto initiation will disable automatic restart of that pump until the initiation signal is clear and has been reset. The affected Core Spray Pump may still be started manually.

3.5 Isolations

A. PSC PUMP SUCTION INBD and OUTBD ISOL VALVES, 2-FCV-75-57 and 2-FCV-75-58, will close on Group II Isolation, tripping PSC Head Tank Pumps 2A and 2B.

B. The Core Spray test valve receives an auto closure signal on any CS auto initiation.

C. The Core Spray minimum flow valves receive a closure signal when flow is approximately 2600 gpm rising and receives a open signal when flow lowers to approximately 2200 gpm.

3.6 Trips

A. Electrical

Excerpt from Unit 2 Tech Spec 3.5.1:

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.1.5	-------------------------NOTES------------------------ 1. Only required to be performed when in MODE 4 > 48 hours. 2. Not required to be performed if performed within the previous 31 days. --- Verify each recirculation pump discharge valve cycles through one complete cycle of full travel.	Once prior to entering MODE 2 from MODE 3 or 4
SR 3.5.1.6	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A VESSEL TO TORUS DIFFERENTIAL PRESSURE OF Core Spray / ≥ 6250 gpm / 2 / ≥ 105 psid SYSTEM / FLOW RATE / NO. OF PUMPS / INDICATED SYSTEM PRESSURE LPCI / ≥ 12,000 gpm / 2 / ≥ 250 psig LPCI / ≥ 9,000 gpm / 1 / ≥ 125 psig	In accordance with the INSERVICE TESTING PROGRAM

(continued)

Excerpt from Tech Spec Bases 3.5.1:

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.6, SR 3.5.1.7, and SR 3.5.1.8

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 7). This periodic Surveillance is performed (in accordance with the ASME OM Code requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of References 13 and 15. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established by testing or analysis or during preoperational testing.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ 950 psig to perform SR 3.5.1.7 and ≥ 150 psig to perform SR 3.5.1.8. Adequate steam flow is represented by at least two turbine bypass valves full open for SR 3.5.1.7 and SR 3.5.1.8. Therefore, sufficient time is allowed after adequate

(continued)

Excerpts from 2-OI-74: Supports Distractors C(1), D(1)

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0184 Page 18 of 540

3.2 RHR Pumps

A. To minimize system vibration, RHR pump operation should be minimized below 7,000 gpm or above 10,000 gpm or for more than 3 minutes at minimum flow.

B. [NRC/C] The RHR pumps are considered to be operable without the seal cooler under the following conditions:

1. Always operable in the LPCI and Containment Cooling Mode.

2. During Shutdown Cooling, operable up to a suction temperature of 215°F.

3. Operable for an emergency with suction temperatures between 215°F and 400°F. Operation in this condition for more than two days requires an inspection of the seal surfaces. [NRC LER 296/83047 R1]

C. To prevent Recirculation Pump operation at shutoff head, RHR pumps must not be operated in parallel with Recirculation Pumps.

3.3 LPCI

A. LPCI will initiate on any of the following signals:

1. Reactor Vessel low low-low water level (-122 inches)(Level 1).

2. High Drywell Pressure (2.45 psig) with low Reactor Vessel Pressure (450 psig).

B. Manually stopping an RHR pump after LPCI initiation disables automatic restart of that pump until the initiation signal is reset. The affected RHR pump can still be started manually.

C. Upon an automatic LPCI initiation with normal power available, RHR Pump 2A starts immediately and 2B, 2C, 2D sequentially start at 7 second intervals. Otherwise, all RHR pumps start immediately once diesel power is available (and normal power unavailable).

D. As soon as possible after an RHR pump(s) auto start, place its corresponding control room handswitch to normal-after-start position to ensure the handswitch disagreement light(s) and pump tripped annunciator(s) function as designed.

Supports Distractors B(2), D(2)

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0184 Page 25 of 540

3.6 Interlocks (continued)

5. RHR Minimum Flow Valve Interlocks

   a. The RHR minimum flow valves auto close if both pumps in the corresponding loop are off AND either pump's SDC suction valve is open.

   b. The minimum flow valves open (after a 10 second TD) and close on a low flow of 5800 gpm. The tolerance of the flow switch may allow the setpoint of the min flow valve to be from 4500 gpm to 7000 gpm. Operation outside of this expanded range should be investigated.

   c. Placing the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 2-HS-74-148(149), in INHIBIT, simulates a high flow and the minimum flow valve will remain closed regardless of flow.

   d. Opening RHR SYSTEM I(II) MIN FLOW VALVE 2-HS-74-7A(30A) from 2-PNL-9-3, with the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 2-HS-74-148(149), in INHIBIT, causes the minimum flow valves to travel full open and full close unless the RHR SYSTEM I(II) MIN FLOW VALVE, 2-HS-74-7A(30A) is placed in closed position to break the OPEN seal in contacts.

   e. [II/C] Local operation of the RHR minimum flow valves bypasses the intended function of the Minimum Flow Inhibit switch and can cause inadvertent drainage of the Reactor vessel to the Suppression Pool. [BFPER941099]

   f. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 2-HS-74-148(149), with the respective RHR loop in standby readiness, can cause inadvertent damage to that loop RHR pump(s) should RHR pump(s) auto start. [BFA-890790003P]

   g. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT Switch, 2-HS-74-148(149), with the respective RHR loop in Shutdown Cooling, can cause inadvertent drainage of the Reactor vessel to the Suppression Pool. [BFA-890790003P]

6. The RHR Outboard LPCI injection valves, 2-FCV-74-52(66), have throttling capability. They receive an auto open signal in the presence of a LPCI initiation signal when Reactor pressure is ≤450 psig and are interlocked open under these conditions or until the appropriate LPCI SYS I (SYS II) OUTBD INJ VLV BYPASS SEL keylock Switch, 2-HS-74-155A(155B), is placed in the BYPASS position. Additionally these valves are interlocked to prevent opening when reactor pressure is greater than 450 psig if its in-line companion valve 2-FCV-74-53(67) is not fully closed.

Examination Outline Cross-reference:	Level	RO	SRO
211000 (SF1, SLCS) Standby Liquid Control	Tier #	------	2
G2.2.40 (10CFR 55.43.2 - SRO Only)	Group #	------	1
Ability to apply Technical Specifications for a system.	K/A #	211000G2.2.40	
	Importance Rating	------	4.7

Proposed Question: **# 87**

Unit 1 is operating at 100% RTP with the following conditions:

At 0800 on 5/9/20:

- The 1A SLC Pump trips during testing and will not reset

At 1000 on 5/9/20:

- 'B' EDG was declared INOPERABLE

Given the conditions above, which **ONE** of the following completes the statement below?

In accordance with Tech Spec 3.1.7, Standby Liquid Control System, entry into CONDITION C is required by __________.

[REFERENCE PROVIDED]

A. 5/9/20 at 1800

B. 5/9/20 at 2200

C. 5/16/20 at 0800

D. 5/16/20 at 1600

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT**:** Incorrect but plausible if the candidate mistakes the 'B' EDG as the power supply for the remaining 1B SLC Pump, however the correct EDG supply is actually 'C' EDG. Tech Spec 3.1.7, Condition B - 8 hours applied to 1000 results in 1800, 5/9/20. Tech Spec 3.8.1, Condition B - 4 hour statement to declare redundant equipment INOPERABLE is not applied correctly. In that case, under the assumption 'B' EDG supplies 1B SLC pump, then 1B SLC pump is required to be declared INOPERABLE 4 hours later.

B INCORRECT: Incorrect but plausible if the candidate mistakes the ‘B’ EDG as the EDG power supply for the remaining 1B SLC pump, the correct EDG supply is actually ‘C’ EDG. This applies Tech Spec 3.8.1, Condition B - 4 hour statement to declare redundant equipment INOPERABLE correctly. Under the assumption the ‘B’ EDG supplies 1B SLC pump, then 1B SLC pump is required to be declared INOPERABLE 4 hours later. This applies the 8 hours from Tech Spec 3.1.7, Condition B to 1000 equals 1800 plus 4 hours results in 2200, 5/9/20.

C **CORRECT**: In accordance with Tech Spec 3.1.7, Condition A, One SLC subsystem is INOPERABLE therefore 7 days is the required completion time added to 0800, 5/9/21 when 1A SLC Pump became INOPERABLE, resulting in 0800, 5/16/21. ‘B’ EDG being INOPERABLE doesn’t impact the other required SLC Pump since the correct EDG that supply’s the 1B SLC pump is actually ‘C’ EDG.

D INCORRECT: Incorrect but plausible if the candidate applies Tech Spec 3.1.7, Condition A adding 7 days to when 1A SLC Pump became INOPERABLE, resulting in 0800, 5/16/20 as in the correct answer. However, the candidate mistakes the ‘B’ EDG as the EDG power supply for the remaining 1B SLC pump applying 8 hours from Condition B resulting in 1600, 5/16/20.

SRO Level Justification: Tests the candidate’s ability to apply Technical Specifications for Standby Liquid Control System. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. Based on SLC System status and Technical Specification Operability, Shutdown Requirements and application of required actions will be evaluated to mitigate the consequences of the abnormal conditions. With references provided, this question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): Unit 1 Tech Spec 3.1.7, Amend. 251 (Attach if not previously provided)
Unit 1 Tech Spec 3.8.1 Amend. 249, 280

Proposed references to be provided to applicants during examination: **Unit 1 Tech Spec 3.1.7 and 3.8.1 (No Bases)**

Learning Objective: OPL 171.039 Obj. 7 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1804 #87	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41
55.43 **X**

Copy of Bank Question:

Proposed Question: **# 87**

Unit 2 is operating at 100% RTP with the following conditions.

- At 0700 on 9/9/17, 'B' EDG was declared INOPERABLE
- At 1000 on 9/9/17, 2-SHV-63-507, 2B SLC Pump Suction Shutoff Valve was found closed and failed to manually reopen

Subsequently, at 2000 on 9/9/17, 'B' EDG is declared OPERABLE.

Given the above conditions, which ONE of the following completes the statement below?

In accordance with Tech Spec 3.1.7, the entry into Condition 'C' ________.

[REFERENCE PROVIDED]

A. is required by 9/9/17 at 1800

B. is required by 9/9/17 at 1900

C. will become necessary if 2B SLC Pump cannot be restored to OPERABLE status by 9/16/17 at 2000

D. will become necessary if 2B SLC Pump cannot be restored to OPERABLE status by 9/16/17 at 1000

Proposed Answer: **D**

Excerpt from Tech Spec 3.1.7:

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SLC subsystem inoperable.	A.1	Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1	Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours

Excerpts from Tech Spec 3.8.1:

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Unit 1 and 2 diesel generators (DGs) with two divisions of 480 V load shed logic and common accident signal logic OPERABLE; and

c. Unit 3 DG(s) capable of supplying the Unit 3 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Verify power availability from the remaining OPERABLE offsite transmission network. AND	1 hour AND Once per 8 hours thereafter (continued)

AC Sources - Operating
3.8.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one shutdown board concurrent with inoperability of redundant required feature(s)
	AND		
	A.3	Restore required offsite circuit to OPERABLE status.	7 days AND 21 days from discovery of failure to meet LCO
B. One required Unit 1 and 2 DG inoperable.	B.1	Verify power availability from the offsite transmission network.	1 hour AND Once per 8 hours thereafter
	AND		

(continued)

AC Sources - Operating
3.8.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Evaluate availability of both temporary diesel generators (TDGs).	1 hour <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>		
	B.3	Declare required feature(s), supported by the inoperable Unit 1 and 2 DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	B.4.1	Determine OPERABLE Unit 1 and 2 DG(s) are not inoperable due to common cause failure.	24 hours
	<u>OR</u>		
	B.4.2	Perform SR 3.8.1.1 for OPERABLE Unit 1 and 2 DG(s).	24 hours
	<u>AND</u>		

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling	Tier #	------	2
A2.07 **(10CFR 55.43.5 – SRO Only)**	Group #	------	1
Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	217000A2.07	
• Loss of lube oil	Importance Rating	------	3.1

Proposed Question: **# 88**

Unit 2 is operating at 8% RTP with the following conditions:

- A Reactor Startup is in progress
- AUO reports that RCIC sight glass lube oil level is **NOT** visible

Given the conditions above, which **ONE** of the following completes the statements below in accordance with Tech Specs?

HPCI **OPERABILITY** is required to be verified by performing ___**(1)**___.

Under these conditions, the REACTOR MODE SWITCH ___**(2)**___ be placed in RUN.

A. (1) administrative checks
(2) can NOT

B. (1) administrative checks
(2) can

C. (1) a HPCI flow rate surveillance
(2) can NOT

D. (1) a HPCI flow rate surveillance
(2) can

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* Given that RCIC is INOPERABLE based on the AUO's report, Tech Spec 3.5.3, Condition A is applicable. Required Action A.1 states to verify by administrative means HPCI is OPERABLE immediately. In accordance with Tech Spec Bases 3.5.3, an administrative check is to be performed by examining logs or other information to determine if HPCI is out of service for maintenance or other reasons. It does NOT mean it is necessary to perform Surveillances needed to demonstrate HPCI OPERABILITY. For second part, in accordance with LCO 3.0.4 Bases, LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after the performance of a Risk Assessment. However, there is a small subset of systems/components that have been determined to be more important to risk and use of LCO 3.0.4.b allowance is prohibited. The LCOs governing such systems contain Notes stating LCO 3.0.4.b is NOT applicable, as is the case for RCIC's Tech Spec 3.5.3.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible if the candidate fails to remember (with no reference provided), that LCO 3.0.4.b does not apply to HPCI and RCIC believing the performance of a Risk Assessment would allow entry into MODE 1.

C INCORRECT: First part is incorrect but plausible if the candidate misapplies ECCS LCO Bases knowledge pertaining to HPCI and RCIC OPERABILITY definition of 'administrative check' by reviewing logs, etc…. The candidate would then incorrectly conclude that HPCI OPERABILITY can ONLY be verified by conducting a surveillance run. Second part is correct *(See A).*

D INCORRECT: First part is incorrect but plausible *(See C).* The second part is incorrect but plausible *(See B).*

SRO Level Justification: Tests the candidate's ability to predict the impact of a loss of lube oil on the RCIC System along with the correct strategy in accordance with Technical Specifications. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. Based on RCIC System status and Technical Specification INOPERABILITY, the performance of a Risk Assessment to support LCO 3.0.4.b is not applicable to RCIC. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): U2 Tech Spec 3.5.3, Amend. 286 (Attach if not previously provided)
U2 TS Bases 3.0, Amend. 286
U2 TS Bases 3.5.3, Amend. 286

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.040 Obj. 9 _ (As available)

Question Source:	Bank #	BFN 1909 #87	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	**X**
10 CFR Part 55 Content:	55.41	
	55.43	**X**

Comments:

Copy of Bank Question:

Proposed Question: **# 87**

Unit 2 is operating at 8% RTP with the following conditions:

- A Reactor Startup is in progress
- AUO reports that RCIC sight glass lube oil level is **NOT** visible

Given the conditions above, which **ONE** of the following completes the statements below in accordance with Tech Specs?

HPCI **OPERABILITY** is required to be verified by performing ____**(1)**____.

Under these conditions, the REACTOR MODE SWITCH ____**(2)**____ be placed in RUN.

A. (1) administrative checks
(2) can NOT

B. (1) administrative checks
(2) can

C. (1) HPCI flow rate surveillance
(2) can NOT

D. (1) HPCI flow rate surveillance
(2) can

Proposed Answer: **A**

Excerpt from U2 Tech Spec 3.5.3:

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.4.b is not applicable to RCIC.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RCIC System inoperable.	A.1	Verify by administrative means High Pressure Coolant Injection System is OPERABLE.	Immediately
	AND		
	A.2	Restore RCIC System to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Reduce reactor steam dome pressure to ≤ 150 psig.	36 hours

Excerpts from U2 Tech Spec Bases 3.5.3:

BASES (continued)

LCO	The OPERABILITY of the RCIC System provides adequate core cooling such that actuation of any of the low pressure ECCS subsystems is not required in the event of RPV isolation accompanied by a loss of feedwater flow. The RCIC System has sufficient capacity for maintaining RPV inventory during an isolation event.
APPLICABILITY	The RCIC System is required to be OPERABLE during MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig, since RCIC is the primary non-ECCS water source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, and in MODES 4 and 5, RCIC is not required to be OPERABLE since the low pressure ECCS injection/spray subsystems can provide sufficient flow to the RPV.
ACTIONS	A Note prohibits the application of LCO 3.0.4.b to an inoperable RCIC system. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable RCIC system and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

(continued)

ES-401 Sample Written Examination Question Worksheet Form ES-401-5

BASES

ACTIONS (continued)

A.1 and A.2

If the RCIC System is inoperable during MODE 1, or MODE 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCI System is immediately verified to be OPERABLE, the RCIC System must be restored to OPERABLE status within 14 days. In this Condition, loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of HPCI is therefore immediately verified when the RCIC System is inoperable. This may be performed as an administrative check, by examining logs or other information, to determine if HPCI is out of service for maintenance or other reasons. It does not mean it is

(continued)

ACTIONS

A.1 and A.2 (continued)

necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the HPCI System. If the OPERABILITY of the HPCI System cannot be verified, however, Condition B must be immediately entered. For certain transients and abnormal events with no LOCA, RCIC (as opposed to HPCI) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of the RPV water level. Therefore, a limited time is allowed to restore the inoperable RCIC to OPERABLE status.

The 14 day Completion Time is based on a reliability study (Ref. 3) that evaluated the impact on ECCS availability, assuming various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (AOTs). Because of similar functions of HPCI and RCIC, the AOTs (i.e., Completion Times) determined for HPCI are also applied to RCIC.

B.1 and B.2

If the RCIC System cannot be restored to OPERABLE status within the associated Completion Time, or if the HPCI System is simultaneously inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

Excerpts from U2 Tech Spec Bases for LCO 3.0.4.b:

BASES

LCO 3.0.3 (continued)	Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.6 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.6 of "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.
LCO 3.0.4	LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c. LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

(continued)

BASES

LCO 3.0.4 (continued)

management actions. For refueling and shutdown activities, the use of a key safety function defense in depth approach, as discussed in NUMARC 91-06 (and Section 11 of NUMARC 93-01) is considered an acceptable approach to satisfy LCO 3.0.4.b requirements regarding risk assessment and management. At Browns Ferry, this approach is the ORAM process. The LCO 3.0.4.b risk assessments do not have to be documented.

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these system and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., Reactor Coolant System Specific Activity), and may be applied to other Specifications based on NRC plant-specific approval.

(continued)

Examination Outline Cross-reference:

239002 (SF3 SRV) Safety Relief Valves

G2.4.2 (10CFR 55.43.5 - SRO Only)

Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	239002G2.4.2	
Importance Rating	------	4.6

Proposed Question: **# 89**

Unit 2 is operating at 100% RTP with the following conditions:

- HPCI is INOPERABLE, the applicable Tech Spec ACTION has been taken
- Engineering just reported that one (1) MSRV safety function setpoint was inadvertently adjusted to 1210 psig
- The hand switch for the respective MSRV is on the **VERTICAL** portion of Panel 2-9-3 near the MSRV Acoustic Monitor

Given the conditions above, which **ONE** of the following completes the statements below?

The reported MSRV setpoint is ___**(1)**___ the Reactor Pressure EOI entry condition.

___**(2)**___ is required in accordance with Tech Spec 3.5.1.

[REFERENCE PROVIDED]

A. (1) above
(2) NO additional ACTION

B. (1) above
(2) Entry into LCO 3.0.3

C. (1) below
(2) NO additional ACTION

D. (1) below
(2) Entry into LCO 3.0.3

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 2-EOI-1, RPV Control, the Reactor Pressure entry condition is above 1073 psig (automatic SCRAM), therefore the given setpoint of 1210 psig is above the EOI entry condition. For second part, in accordance with Tech Spec 3.5.1, ECCS – Operating, NO ACTION is currently required. Since HPCI is already given as INOPERABLE, the candidate must use the last given bullet to determine if the MSRV is an ADS valve or not based on its location on Panel 2-9-3. Since its location indicates that it is not an ADS valve, Tech Spec 3.5.1 is not applicable currently.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible if the candidate determines the REQUIRED ACTION from Tech Spec 3.5.1, H.1 is applicable for the given INOPERABLE HPCI and an ADS valve which states to enter LCO 3.0.3 Immediately.

C INCORRECT: First part is incorrect but plausible if the candidate fails to recall the Reactor Pressure EOI entry condition. Numerous Reactor Pressure values exist from setpoints to Safety Limits. Example: 13 MSRV respective lift setpoints are 1135 psig (4 MSRVs), 1145 psig (4 MSRVs), 1155 psig (5 psig), RPV Design Pressure is 1250 psig with a 10% ASME allowance of 1375 psig, Reactor Steam Dome Pressure Safety Limit is 1325 psig. Second part is correct *(See A)*.

D INCORRECT: First part is incorrect but plausible *(See C)*. Second part is incorrect but plausible *(See B)*.

SRO Level Justification: Tests the candidate's knowledge of system setpoints, interlocks and automatic actions associated with Main Steam Relief Valves as it relates to EOI entry conditions. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome especially as it relates to the Tech Specs and EOIs. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-EOI-2, Rev. 18 (Attach if not previously provided)
Unit 2 Tech Spec 3.5.1, Amend. 294
Unit 2 Tech Spec 3.4.3, Amend. 253

Proposed references to be provided to applicants during examination: **Unit 2 Tech Spec 3.5.1 (No Bases)**

Learning Objective: OPL171.009 Obj. 14d (As available)

Question Source:			
	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.009-14 009, #369	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

369. OPL171.009-14 009

Unit 2 has just completed a reactor startup with the following:

- Reactor power is currently 100%
- Report just received from Wylie industries is that two MSRVs were inadvertently adjusted to a setpoint of 1210 psig.
- **Neither** MSRV indicated in this report were ADS MSRVs.

Which ONE of the following actions is required by Technical Specifications?

[REFERENCE PROVIDED]

A. **NO** action required (information only).

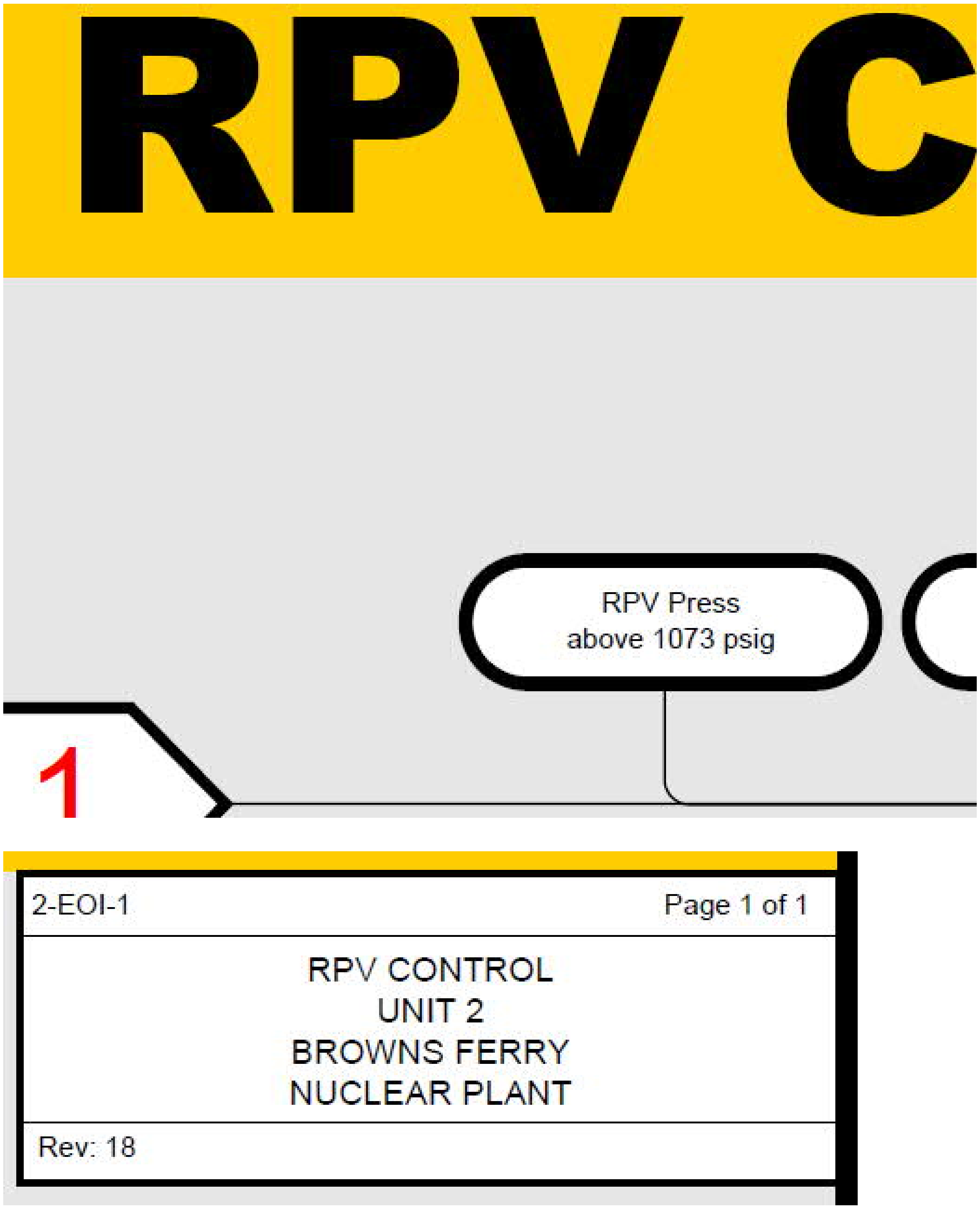
B✓ Be in Mode 3 in 12 hours, Mode 4 in 36 hours.

C. Restore operability of ONE affected MSRV within 14 days.

D. Be in Mode 3 in 12 hours, reduce reactor steam dome pressure to less than OR equal to 150 psig.

Correct Answer: B

Excerpt from 2-EOI-1:



Excerpts from Unit 2 Tech Spec 3.5.1:

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1	Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days[1]

(continued)

[1] - This Completion Time may be extended to 14 days on a one-time basis. This temporary approval expires June 1, 2005.

ECCS - Operating
3.5.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. HPCI System inoperable.	C.1	Verify by administrative means RCIC System is OPERABLE.	Immediately
	AND		
	C.2	Restore HPCI System to OPERABLE status.	14 days
D. HPCI System inoperable. AND Condition A entered.	D.1	Restore HPCI System to OPERABLE status.	72 hours
	OR		
	D.2	Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours
E. One ADS valve inoperable.	E.1	Restore ADS valve to OPERABLE status.	14 days
F. One ADS valve inoperable. AND Condition A entered.	F.1	Restore ADS valve to OPERABLE status.	72 hours
	OR		
	F.2	Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Two or more ADS valves inoperable. OR Required Action and associated Completion Time of Condition C, D, E, or F not met.	G.1	Be in MODE 3.	12 hours
	AND		
	G.2	Reduce reactor steam dome pressure to ≤ 150 psig.	36 hours
H. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. OR HPCI System and one or more ADS valves inoperable.	H.1	Enter LCO 3.0.3.	Immediately

Examination Outline Cross-reference:	Level	RO	SRO
261000 (SF9 SGTS) Standby Gas Treatment	Tier #	------	2
A2.12 **(10CFR 55.43.5 - SRO Only)**	Group #	------	1
Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	261000A2.12	
• High fuel pool ventilation radiation: Plant-Specific	Importance Rating	------	3.4

Proposed Question: **# 90**

Unit 3 is at 100% RTP, with the following conditions:

At 0600 on 5/26/21:

- Standby Gas Train A was tagged out of service

At 1000 on 5/26/21:

- A fuel bundle is dropped on the Refuel Floor
- Standby Gas Train C did not automatically start on a valid initiation signal; but, manually started

Which **ONE** of the following completes the statements below?

A Tech Spec required shutdown CONDITION must be entered at ___**(1)**___.

A ___**(2)**___ report to the NRC is required in accordance with NPG-SPP-3.5, Regulatory Reporting Requirements.

[REFERENCE PROVIDED]

A. (1) 0600 on 6/2/21
(2) 4-hour

B. (1) 0600 on 6/2/21
(2) 8-hour

C. (1) 1000 on 5/26/21
(2) 4-hour

D. (1) 1000 on 5/26/21
(2) 8-hour

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that SGT C did not start automatically, it did start manually, the candidate could believe that SGT C is still OPERABLE. SGT A is still INOPERABLE therefore Tech Spec 3.6.4.3 CONDITION A is applicable. This requires a shutdown CONDITION to be entered in 7 days if SGT A is not restored to OPERABLE status. Therefore, at 0600 on 6/02/21, CONDITION B would be entered, requiring all three Units to be in MODE 3 in 12 hours and MODE 4 in 36 hours. First part is correct *(See C).*

B INCORRECT: First part is incorrect *(See A)*. The second part is incorrect but plausible in that NPG-SPP-3.5, Regulatory Reporting Requirements, Section 3.1.D.4, an 8-hour report is required when an event or condition could have prevented the fulfillment of the safety function of systems that are needed to mitigate the consequences of an accident. In accordance with Tech Spec Bases, the SGT System's function is to ensure that radioactive materials that leak from the Primary Containment into the Secondary Containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment, and is designed to mitigate the consequences of a Loss of Coolant Accident. Therefore, the candidate could believe that a loss of SGT would require an 8-hour report.

C **CORRECT**: *(See attached)* In accordance with Tech Spec 3.6.4.3, SGT System, with two or more Standby Gas Trains INOPERABLE, all three units are required to enter LCO 3.0.3 immediately, which requires a Reactor Shutdown and cool down. This was effective once Standby Gas Train C failed to automatically start at 1000 on 5/26/21. For second part, in accordance with NPG-SPP-3.5, Regulatory Reporting Requirements, a 4-hour report to the NRC is required for the initiation of any Nuclear plant shutdown required by Technical Specifications.

D INCORRECT: The first part is correct *(See C)*. The second part is incorrect but plausible *(See B)*.

SRO Level Justification: Tests the candidate's ability to predict the impact of the Fuel Pool Ventilation Radiation effect on the Standby Gas Treatment System, as well as the Technical Specification and reporting requirements. SRO only because of link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): NPG-SPP-3.5, Rev.16 (Attach if not previously provided)

Unit 3 Tech Spec 3.6.4.3, Amend. 249

Unit 3 Tech Spec 3.6.4.3 Bases, Rev. 29

Proposed references to be provided to applicants during examination: **Unit 3 Tech Spec 3.6.4.3, NPG-SPP-3.5, Attachment 1**

Learning Objective: OPL171.033 Obj. 16 (As available)

Question Source:	Bank #	BFN 1102 #88	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2011	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	X

Comments:

Copy of Bank Question:

ILT 1102 Written Exam

88. 261000 A2.12

Unit 3 is at 100% Reactor Power. Standby Gas Treatment System (SGTS) A was tagged out of service on 1/16/11 at 0600. SGTS B has been manually started. At 1000 on 1/16/11, a container is removed from the Unit 3 Spent Fuel Pool (SFP) resulting in the following Refuel Zone Radiation Monitor indications:

- 3-RM-90-140 Detector A is reading 73 mr/hr
- 3-RM-90-140 Detector B is reading 72 mr/hr
- 3-RM-90-141 Detector A is reading 71 mr/hr
- 3-RM-90-141 Detector B is reading 71 mr/hr

SGTS C did **NOT** start. The container was placed back in the SFP **AND** Refuel Zone Radiation Monitor indications returned to normal.

Which ONE of the following completes the statements below?

A Tech Spec required shutdown condition must be entered at **__(1)__** in accordance with Tech Spec 3.6.4.3, “Standby Gas Treatment System.”

A **__(2)__** hour report to the NRC is required when the shutdown is commenced.

[REFERENCE PROVIDED]

A. **(1)** 1000 on 1/16/11
(2) four

B. **(1)** 0600 on 1/23/11
(2) four

C. **(1)** 1000 on 1/16/11
(2) one

D. **(1)** 0600 on 1/23/11
(2) one

Excerpt from Tech Spec 3.6.4.3:

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

SGT System
3.6.4.3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during OPDRVs.	C.1 Place two OPERABLE SGT subsystems in operation. OR C.2 Initiate action to suspend OPDRVs.	Immediately Immediately
D. Two or three SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

Excerpts from NPG-SPP-03.5:

Attachment 1
(Page 5 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

(2) Significant physical damage to a power reactor or any facility possessing SSNM or its equipment or carrier equipment transporting nuclear fuel or spent nuclear fuel, or to the nuclear fuel or spent nuclear fuel a facility or carrier possesses; or

NOTE

A Confirmed Cyber Attack at any TVA Nuclear site is reported to the NRC in accordance with the requirements of 10 CFR 73.77 and NPG-SPP-12.8.8.

(3) Interruption of normal operation of a licensed nuclear power reactor through the unauthorized use of or tampering with its machinery, components, or controls including the security system.

b. An actual entry of an unauthorized person into a protected area, material access area, controlled access area, vital area, or transport.

c. Any failure, degradation, or the discovered vulnerability in a safeguard system that could allow unauthorized or undetected access to a protected area, material access area, controlled access area, vital area, or transport for which compensatory measures have not been employed.

d. The actual or attempted introduction of contraband into a protected area, material access area, vital area, or transport (refer to NSDP-1 Attachment 23).

C. The following criteria require 4-hour notification:

1. §50.72(b)(2)(i) - The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

2. §50.72(b)(2)(iv)(A) - Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

3. §50.72(b)(2)(iv)(B) - Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

NPG Standard Programs and Processes	Regulatory Reporting Requirements	NPG-SPP-03.5 Rev. 0016 Page 23 of 96

Attachment 1
(Page 6 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTES

1) NPG-SPP-05.14 provides additional instructions regarding addressing and informally communicating events to outside agencies involving radiological spills and leaks.

2) Routine or day-to-day communications between TVA organizations and state agencies typically do not constitute a formal notification to other government agencies that would require a report in accordance with §50.72(b)(2)(xi).

4. §50.72(b)(2)(xi) - Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactive contaminated materials.

D. The following criteria require 8-hour notification:

NOTE

With the exception of "Events or Conditions that Could Have Prevented Fulfillment of a Safety Function," ENS notifications are required for any event that occurred within three years of discovery, even if the event was not ongoing at the time of discovery.

1. §50.72(b)(3)(ii)(A) - Any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

2. §50.72(b)(3)(ii)(B) - Any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.

3. §50.72(b)(3)(iv)(A) - Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) [see list below], except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

   a. The systems to which the requirements of paragraph §50.72(b)(3)(iv)(A) apply are:

NPG Standard Programs and Processes	Regulatory Reporting Requirements	NPG-SPP-03.5 Rev. 0016 Page 25 of 96

Attachment 1
(Page 8 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTES

1) For systems within scope, the inadvertent TS inoperability of a system in a required mode of applicability constitutes an event or condition for which there is no longer reasonable expectation that equipment can fulfill its safety function. Therefore, such events or conditions are reportable as an "Event or Condition that Could Have Prevented Fulfillment of a Safety Function."

2) According to §50.72(b)(3)(vi) events covered by §50.72(b)(3)(v) "may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant [this paragraph] if redundant equipment in the same system was operable and available to perform the required safety function."

4. §50.72(b)(3)(v) - Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:

   (A) Shut down the reactor and maintain it in a safe shutdown condition;

   (B) Remove residual heat;

   (C) Control the release of radioactive material; or

   (D) Mitigate the consequences of an accident.

5. §50.72(b)(3)(xii) - Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

NOTE

NPG-SPP-03.5.1, Reporting Requirements for Loss of Emergency Preparedness Capabilities, provides TVA site specific guidance for event notifications required by 10 CFR 50.72(b)(3)(xiii).

6. §50.72(b)(3)(xiii) - Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (for example, significant portion of control room indication, emergency notification system, or offsite notification system).

E. Follow-up Notification (§50.72(c))

With respect to the telephone notifications made under paragraphs (a) and (b) [§50.72(a) and §50.72(b), respectively] of this section [§50.72], in addition to making the required initial notification, during the course of the event:

Excerpts from Tech Spec 3.6.4.3 Bases:

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

The SGT System is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1). The function of the SGT System is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The SGT System consists of three redundant 50% capacity subsystems, each with its own dampers, charcoal filter train, and controls. The SGT subsystems share common supply and exhaust ductwork.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident (Ref. 3). For the loss of coolant accident, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).

Examination Outline Cross-reference:	Level	RO	SRO
201003 (SF1 CRDM) Control Rod and Drive Mechanism	Tier #	------	2
G2.1.7 (10CFR 55.43.5 - SRO Only)	Group #	------	2
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	K/A #	201003G2.1.7	
	Importance Rating	------	4.7

Proposed Question: **# 91**

Unit 2 is at 8% RTP. A Reactor startup is in progress following a refueling outage with the following conditions:

- Control Rod 34-27 is being notched from position 20 to position 22
- A notch withdrawal signal is given to Control Rod 34-27, the Rod settles at position 28

Which **ONE** of the following completes the statements below?

The SRO will direct the OATC to insert Control Rod 34-27 to ___**(1)**___.

Entry into Tech Spec 3.1.6, Rod Pattern Control, ___**(2)**___ required.

Note: 2-OI-85, Control Rod Drive System

2-AOI-85-7, Mispositioned Control Rod

A. (1) position 22 in accordance with 2-OI-85
(2) is

B. (1) position 22 in accordance with 2-OI-85
(2) is NOT

C. (1) position 00 in accordance with 2-AOI-85-7
(2) is

D. (1) position 00 in accordance with 2-AOI-85-7
(2) is NOT

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the procedure for a double-notched Control Rod would be to insert the rod to the intended position, after contacting the Reactor Engineer for support. The second part is correct (*See* C).

B INCORRECT**:** The first part is incorrect but plausible (*See A*). The second part is incorrect but plausible in that LCO 3.1.6 is not applicable in Mode 1 or 2 if Reactor Power is above 10%, as there is no credible Control Rod configuration that results in a Control Rod Worth that could exceed the 280 cal/gm fuel damage limit during a Control Rod Drop Accident (CRDA). However, this knowledge must be recalled from the Tech Spec LCO requirement and is infrequently referenced. Additionally, no references are provided to the candidate.

C **CORRECT**: *(See attached)* In accordance with 2-AOI-85-7, since the Control Rod is >2 notches from its intended position the correct action is to fully insert the Control Rod to position 00. For second part, the LCO for Rod Pattern Control in accordance with Tech Spec 3.1.6 is applicable less than 10% RTP. The Banked Position Withdrawal Sequence (BPWS) is applicable from the condition of all Control Rods fully inserted to 10% RTP. For the BPWS, the Control Rods are required to be moved in groups, with all Control Rods assigned to a specific group required to be within specified banked positions. The banked positions are established to minimize the maximum incremental Control Rod worth without being overly restrictive during normal plant operation. Analyses are performed using methodology demonstrating the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation.

D INCORRECT: First part is correct (*See C*). The second part is incorrect but plausible (*See B*).

SRO Level Justification: Tests the candidate's knowledge of actions required for a triple-notched Control Rod during a Reactor Startup and the Technical Specification implications. SRO only because of link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

Technical Reference(s): 2-AOI-85-7, Rev. 24 (Attach if not previously provided)
U2 Tech Spec 3.1.6, Amend. 253
U2 Tech Spec 3.1.6 Bases, Rev. 61

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.005 Obj. 32 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1006 #85	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2006	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41
55.43 **X**

Copy of Bank Question:

85. 295014G2.4.31 001

Unit 2 is starting up and reactor power is 14% power. The operator is pulling rods to achieve 2 bypass valves to roll the turbine.

Due to a previous rod being difficult to move, the CRD drive water pressure had been temporarily raised to 300 psid and not re-adjusted back down to a normal pressure. When the operator placed the rod movement control switch to the single notch out position for the next control rod, the rod quickly moved from position 16 to 22 (intended position and withdraw limit is 18). The following alarm was received:

RMCS/RWM ROD BLOCK OR SYSTEM TROUBLE (603-239)

Which ONE of the following describes the impact of this alarm condition in accordance with 34GO-OPS-065-0, Control Rod Movement, and Tech Specs?

This __________ a "mispositioned" control rod. Tech Spec 3.1.6 Rod Pattern Control ________________ a Required Action Statement applicable to these plant conditions.

A. IS NOT / contains

B. IS NOT / does NOT contain

C. IS / contains

D✓ IS / does NOT contain

Excerpts from 2-AOI-85-7:

BFN Unit 2	Mispositioned Control Rod	2-AOI-85-7 Rev. 0024 Page 7 of 10

4.2 Subsequent Actions (continued)

NOTE
If Reactor Power is less than or goes less than 22% RTP during Rod Insertion. The RWM system may impose rod blocks if not previously Bypassed.

[3] **IF** the Control Rod is > 2 notches from the intended position, **THEN**

PERFORM the following: (Otherwise N/A)

[3.1] **INSERT** the mispositioned rod to "00". ☐

[3.2] **IF** a Reactor Startup or Shutdown is not in progress, **THEN**

ENSURE 2-GOI-100-12, Power Maneuvering, has been entered if a power change occurred. (Otherwise N/A) ☐

[3.3] **IF** Reactor Power is less than 22% RTP and the Rod Worth Minimizer is not bypassed, **THEN**

MANUAL BYPASS the Rod Worth Minimizer per 2-OI-85, Manual Bypass of the Rod Worth Minimizer section as directed by the Unit Supervisor. (Otherwise N/A) ☐

BFN Unit 2	Mispositioned Control Rod	2-AOI-85-7 Rev. 0024 Page 8 of 10

4.2 Subsequent Actions (continued)

CAUTION

NRC/C] Operations outside of the allowable regions shown on the Recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. Monitoring to be performed during a power decrease and required actions are contained in 2-GOI-100-12. [NCO 940245010]

[4] **IF** the Control Rod is ≤ 2 notches from the intended position, **THEN**

PERFORM the following: (Otherwise N/A)

[4.1] **OBTAIN** recommendation from the Reactor Engineer. ☐

[4.2] **IF** a Reactor Startup or Shutdown is not in progress, **THEN**

ENSURE 2-GOI-100-12, Power Maneuvering has been entered if a power change is anticipated. (Otherwise N/A) ☐

[4.3] **IF** required to allow rod movement to correct the rod position error, **THEN**

REDUCE core thermal power as recommended by the Reactor Engineer/Reactivity Control Plan. (Otherwise N/A) ☐

[4.4] **MOVE** Control Rods with the Unit Supervisor permission and Shift Manager concurrence to recover from the rod positioning error as recommended by the Reactor Engineer/Reactivity Control Plan. ☐

Excerpt from Tech Spec 3.1.6:

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS).

APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with BPWS.	A.1 ------------NOTE----------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." -------------------------------	
	Move associated control rod(s) to correct position.	8 hours
	<u>OR</u>	
	A.2 Declare associated control rod(s) inoperable.	8 hours

(continued)

Excerpt from Tech Spec 3.1.6 Bases:

BASES

APPLICABLE SAFETY ANALYSES (continued)

depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Ref. 6 and 10) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 5).

Control rod patterns analyzed in References 10 and 11 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Analyses are performed using the Reference 10 methodology demonstrating the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The evaluation provided by the generic BPWS analysis (Ref. 8) allows a limited number (i.e., eight) and corresponding distribution of fully inserted, inoperable control rods, that are not in compliance with the sequence.

Examination Outline Cross-reference:

226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode

A2.08 (10CFR 55.43.5 - SRO Only)

Ability to (a) predict the impacts of the following on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

- Pump seal failure

	RO	SRO
Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	226001A2.08	
Importance Rating	------	2.5

Proposed Question: **# 92**

Unit 3 was operating at 100% RTP when a LOCA occurred resulting in the following conditions:

- Drywell Pressure is 28 psig
- Suppression Pool Level is 15 feet
- Suppression Chamber Pressure is 26 psig
- RHR Pumps 3B, 3C and 3D are tripped and **CANNOT** be started
- 480V RMOV Board 3A de-energizes due to an electrical fault
- RHR SYS I PUMP A SEAL LEAKAGE HIGH (2-9-3D, Window 27) alarms and the AUO reports 3A RHR Pump should be secured due to a severe leak

In accordance with 3-EOI Appendix-17B, RHR System Operation Drywell Sprays, ___**(1)**___ RHRSW Pumps provide water for Containment Sprays on Unit 3.

Given the conditions above, the SRO is required to direct entry into ___**(2)**___.

[REFERENCES PROVIDED]

A. (1) B1, B2
(2) C-2 Emergency Depressurization

B. (1) B1, B2
(2) 3-EOI Appendix-17B, Drywell Sprays

C. (1) D1, D2
(2) C-2 Emergency Depressurization

D. (1) D1, D2
(2) 3-EOI Appendix-17B, Drywell Sprays

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 3-EOI Appendix-17B, RHR System Operation Drywell Sprays, B1 and B2 RHRSW pumps are aligned to Unit **3**. D1 and D2 can provide water to Unit **1** and Unit **2** ONLY. For second part, the given conditions above indicate that there are no methods to remove heat from containment with the failure of both RHR loops. Given the loss of 480V RMOV Board 3A, the candidate must understand that even though 3A RHR Pump is running then develops a seal leakage, the Containment Spray valves are not available. The given containment parameters indicate that the PSP Curve 6 is not in the SAFE area. Since Suppression Chamber Pressure cannot be maintained within the SAFE area of PSP Curve 6, the SRO is required to direct 3-C-2, Emergency Depressurization.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible if the candidate fails to recognize that Drywell Sprays are no longer available when 480V RMOV 3A de-energizes.

C INCORRECT: First part is incorrect but plausible in that D1 and D2 RHRSW pumps can provide a source of water to Unit 1 and Unit 2 for Containment Cooling ONLY. B1 and B2 RHRSW pumps provide water to Unit 3 Loop 1 with no source available to Unit 3 Loop 2. The second part is correct *(See A).*

D INCORRECT: First part is incorrect but plausible *(See C).* Second part is incorrect but plausible *(See B).*

SRO Level Justification: Tests the candidate's ability to determine and interpret plant conditions, RHR pump failures, available equipment, and strategies when Emergency Depressurizing is required in the EOIs as it relates to Containment Spray Mode. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-EOI-2, Rev. 13 (Attach if not previously provided)
3-EOI Appendix-17B, Rev. 9
OPL171.044, Rev. 21

Proposed references to be provided to applicants during examination: **RHR SYS I PUMP A SEAL LEAKAGE HIGH (2-9-3D window 27), 3-EOI-2, Curve 6, PSP**

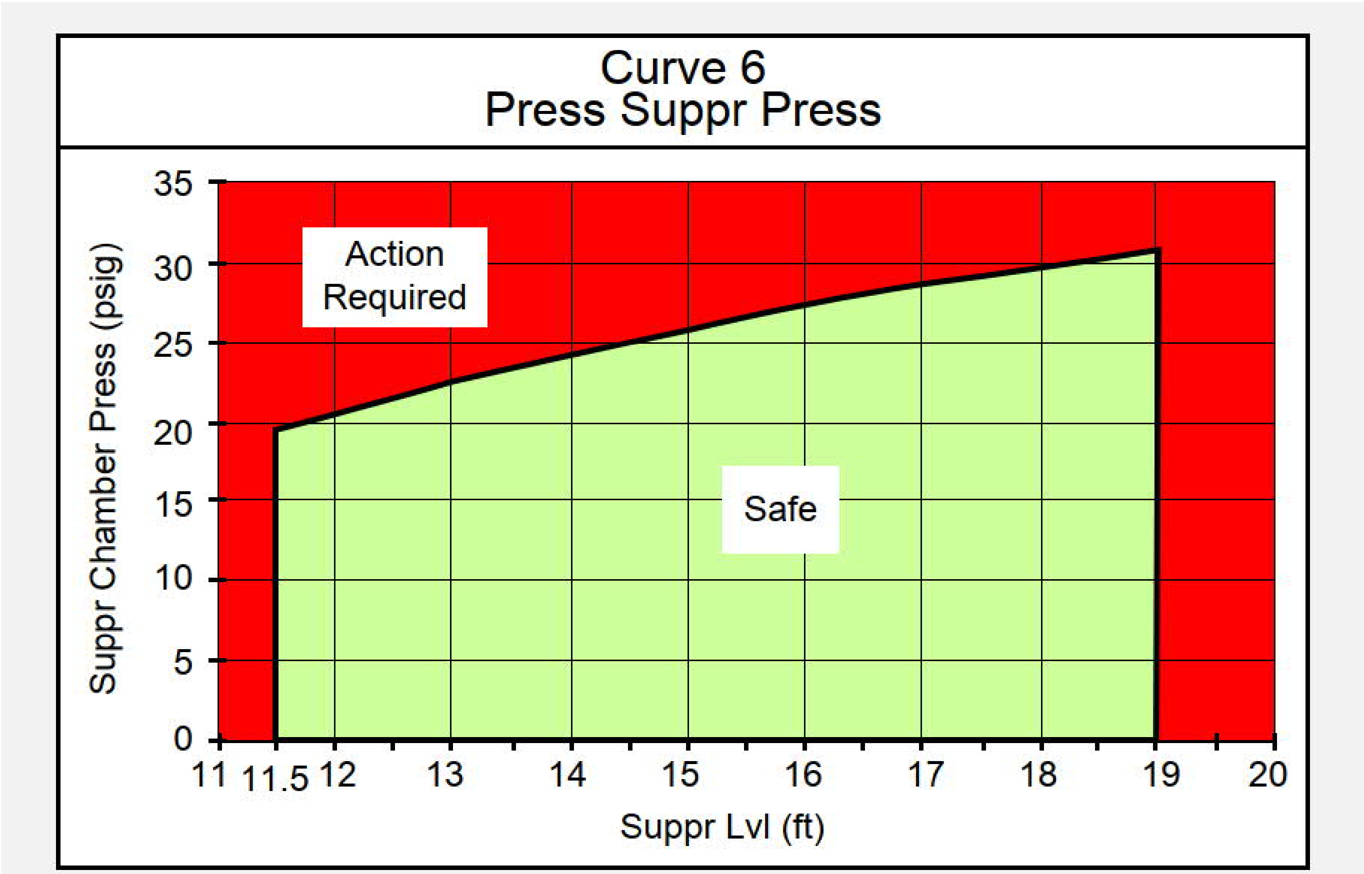
Learning Objective: OPL171.044 Obj. 13c (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

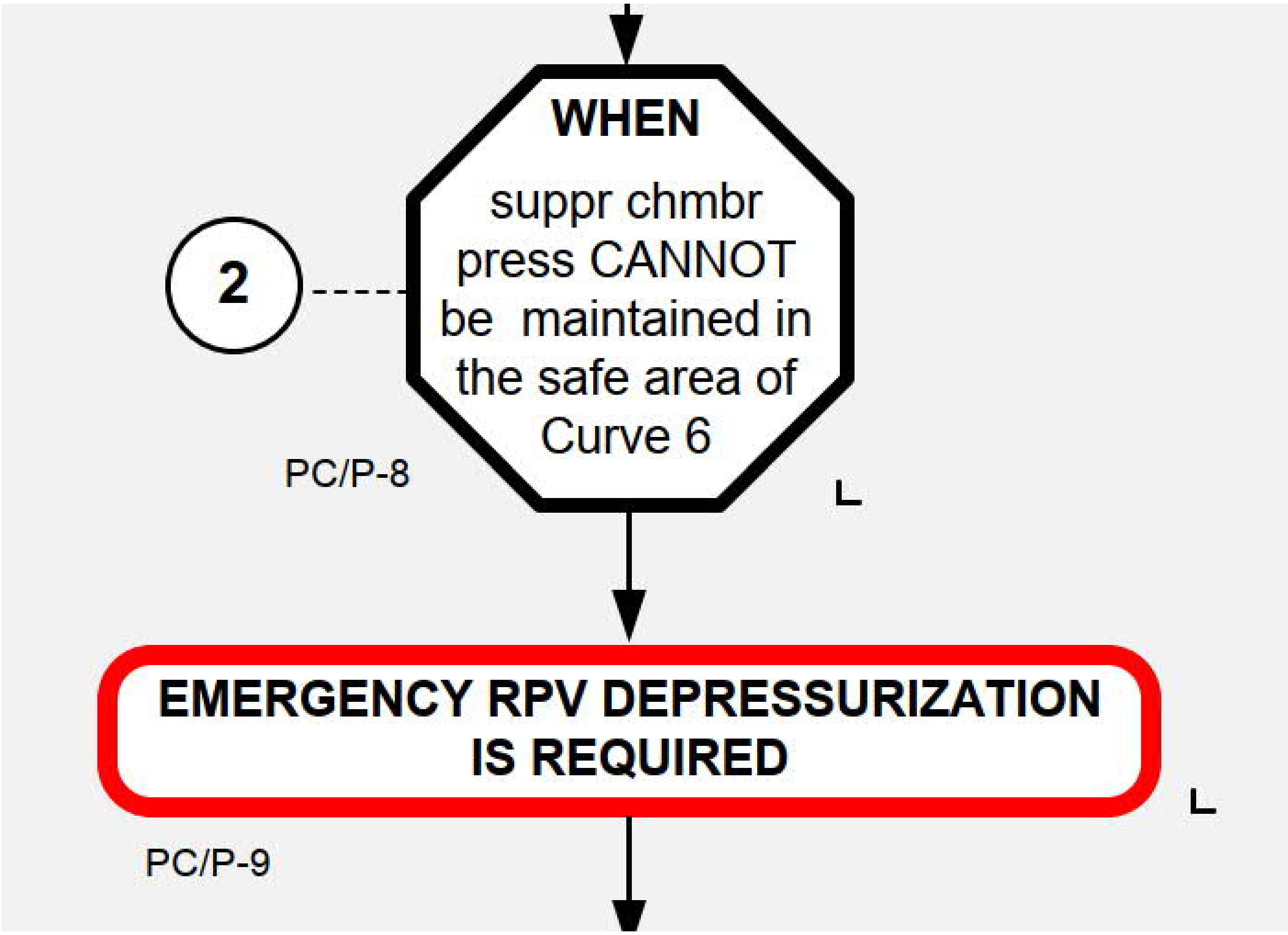
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	**X**
10 CFR Part 55 Content:	55.41	
	55.43	**X**

Comments:

3-EOI-2, Curve 6, PSP provided to candidate:



Excerpts from 3-EOI-2:



3-EOI-2 Page 1 of 1

PRIMARY CONTAINMENT CONTROL
UNIT 3
BROWNS FERRY
NUCLEAR PLANT

Rev. 13

Excerpt from 3-EOI Appendix-17B:

BFN Unit 3	RHR System Operation Drywell Sprays	3-EOI Appendix-17B Rev. 0009 Page 9 of 14

1.0 INSTRUCTIONS (continued)

[8.4] **ENSURE CLOSED** the following valves:

- 3-FCV-74-61, RHR SYS I DW SPRAY INBD VLV ☐
- 3-FCV-74-60, RHR SYS I DW SPRAY OUTBD VLV ☐
- 3-FCV-74-58, RHR SYS I SUPPR CHBR SPRAY VALVE ☐
- 3-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VLV ☐
- 3-FCV-23-46, RHR HX 3B RHRSW OUTLET VLV ☐

[8.5] **ENSURE** RHR Pumps 3A and 3C are <u>NOT</u> running. ☐

[8.6] **PLACE** 3-BKR-074-0100, RHR HTX A-C DISCH XTIE (TO U-2) VLV FCV-74-100 (MO10-171) to ON (480V RMOV Board 3B, Compartment 19A) ☐

[8.7] **START** RHRSW Pumps B1 and B2. ☐

[8.8] **NOTIFY** Unit 1 Operator to **ENSURE CLOSED** 1-FCV-23-46, RHR HX 1B RHRSW OUTLET VLV (Unit 1, Panel 1-9-3) ☐

[8.9] **NOTIFY** Unit 2 Operator to perform the following:

[8.9.1] **ENSURE CLOSED** 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV (Unit 2, Panel 2-9-3) ☐

[8.9.2] **OPEN** 2-FCV-23-57, STANDBY COOLANT VLV FROM RHRSW (Unit 2, Panel 2-9-3). ☐

[8.10] **OPEN** THE FOLLOWING VALVES:

- 3-FCV-74-100, RHR SYS I U-2 DISCH XTIE ☐
- 3-FCV-74-60, RHR SYS I DW SPRAY OUTBD VLV ☐
- 3-FCV-74-61, RHR SYS I DW SPRAY INBD VLV ☐

Excerpt from OPL171.044 Lesson Plan:

OPL171.044 , Residual Heat Removal System, Rev# 21

11. RHR Pump cross-tie valves – Reactor Building 541' elevation on top of the Suppression Pool	12k
12. RHR Pump CST suction valves – Reactor Building 541' elevation	12l
13. RHR Shutdown Cooling Suction Valves – Reactor Building 541' elevation	12m
14. RHR fuel pool cooling valves – Reactor Building 639' elevation	12n
15. RHR System vent, flush, and fill valves a) Various locations throughout the Reactor Building b) Refer to OI-74 for specific locations	12o
M. System Interrelationships	NLO – Initial Object

M. System Interrelationships

1. Reactor Recirculation System
   a) Provides LPCI mode injection path
      (1) Discharge valve of Reactor Recirculation System pumps automatically isolate at <230 psig RPV pressure with LPCI initiation signal present to ensure LPCI injection path to the core.
   b) Reactor Recirculation pumps trip off on reactor vessel low-low level of -45 inches
   c) Provides shutdown cooling suction path from RHR Recirculation Loop 'A' and return through LPCI injection paths.
2. EECW/RHRSW System
   a) Provides cooling water for RHR pump seal and RHR room coolers
   b) RHRSW Pumps A1/A2/B1/B2/C1/C2/D1/D2 assigned to RHR System for Containment Cooling, Shutdown Cooling and Standby Coolant supply.
      (1) Standby Coolant Unit 1 – D1/D2 RHRSW Pumps
      (2) Standby Coolant Unit 2
         (a) Loop I – D1/D2 RHRSW Pumps
         (b) Loop II – B1/B2 RHRSW Pumps
   c) RHRSW Pumps A1/B1/C1/D1 alternately assigned to EECW with automatic start features if manually aligned to EECW.
3. Condensate Storage and Transfer – Supplies flushing water, and can be alternately aligned to supply keep-fill pressure.
4. Main Condenser and Radwaste – provides flushing drain path
5. Fuel Pool Cross-Connects
   a) Provide fuel pool cooling augmentation capability with the RHR System.
   b) RHR Drain Pumps provide the motive force for the supplemental fuel pool cooling mode of RHR

Examination Outline Cross-reference:

239001 (SF3, SF4 MRSS) Main and Reheat Steam
A2.12 (10CFR 55.43.2 – SRO Only)
Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

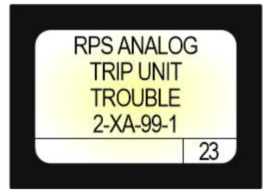
- PCIS / NSSSS actuation

Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	239001A2.12	
Importance Rating	------	4.3*

Proposed Question: **# 93**

Unit 2 is operating at 100% RTP with the following conditions:

- RPS ANALOG TRIP UNIT TROUBLE (2-9-5B, Window 23) alarms
- ATUs for 2-PDIS-001-0025B and 2-PDIS-001-0025D, MAIN STEAM LINE ‘B’ (MSL) FLOW HIGH are INOPERABLE



Given the conditions above, which **ONE** of the following is required in accordance with Tech Specs Actions?

[REFERENCE PROVIDED]

A. Isolate ‘B’ MSL in 12 hours.

B. Place channel in trip in ‘B’ PCIS logic within 24 hours.

C. Restore isolation capability in 1 hour, if not, Isolate ‘B’ MSL in 12 hours.

D. Be in MODE 3 in 12 hours AND MODE 4 in 36 hours.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that the Tech Spec 3.3.6.1, Primary Containment Isolation Instrumentation, ACTION to isolate the Main Steam Lines is directed from CONDITION D. Since both instruments are discovered as INOPERABLE, the candidate could misinterpret application of the point of discovery by going straight to CONDITION D. Tech Spec ACTIONS are always performed sequentially as applicable unless the action will result in an automatic SCRAM such as placing multiple RPS Channels in Trip at one time.

B INCORRECT: Incorrect but plausible in that the Tech Spec CONDITION A is applicable if one trip Channel is INOPERABLE. There are two Channels rendering the Trip System (B) INOPERABLE. It is plausible that the candidate could confuse Channel and Trip System. Additionally, the candidate could confuse how many of the Channels or Systems are INOPERABLE since both instruments have the same numeric designator.

C **CORRECT:** *(See attached)* In accordance with CONDITION B of Tech Spec 3.3.6.1, if PCIS isolation capability is not maintained, the REQUIRED ACTION is to restore isolation capability within 1 hour for the respective function. PCIS logic is A or C AND B or D with the given failed instruments being BOTH in B and D Channels. The B Trip System will NOT perform its function if the Main Steam Line were to experience a HIGH flow condition. CONDITION C is applicable when CONDITION A or B Completion Time is not met with a REQUIRED ACTION to reference Table 3.3.6.1-1. Function 1c, Main Steam Line Flow – HIGH states 2 per MSL for REQUIRED CHANNELS PER TRIP SYSTEM. For Function 1c, since B and D Channels in B MSL are INOPERABLE, CONDITION D is referenced from REQUIRED ACTION C.1. CONDITION D requires that B MSL is isolated within 12 hours.

D INCORRECT**:** Incorrect but plausible in that the Tech Spec ACTION to isolate the Main Steam Lines is directed from CONDITION D. Since both instruments are discovered as INOPERABLE, the candidate could misinterpret application of the point of discovery as stated in B's explanation above. However, CONDITION D has the second REQUIRED ACTION as delineated by an OR statement. Either REQUIRED ACTION delineated by the OR statement would be correct ONLY AFTER the completion time of CONDITION B was not met, but that entry first requires the 1-hour allowance to restore isolation capability in CONDITION B.

SRO Level Justification: Tests the candidate's ability to analyze plant conditions, indications, and their understanding of the impacts to equipment failures associated with the Main System as it relates to PCIS logic and Tech Specs. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the technical specifications and their bases. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): Unit 2 Tech Spec 3.3.6.1, Amend. 253 (Attach if not previously provided)
2-ARP-9-5B, Rev. 31
2-OI-64, Rev. 127
2-47E801-1, Rev. 33
2-730E927-8, Rev. 24
OPL171.017, Rev 21

Proposed references to be provided to applicants during examination: **RPS ANALOG TRIP UNIT TROUBLE (2-9-5B, Window 23), Unit 2 Tech Spec 3.3.6.1 and Table 3.3.6.1-1 (No Bases)**

Learning Objective: OPL171.017 Obj. 2a, 4 (As available)

Question Source: Bank # ILT EXAM BANK OPL171.009-14 006 #387

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	X

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

387. OPL171.009-14 006

Unit 2 is at 100% power when the ATU trouble alarm is received at Panel 9-5.

Investigation in the Auxiliary Instrument Room reveals that the ATUs for PDIS 1-25B and PDIS 1-25D (MSL B Flow) are indicating downscale and the IMs indicate that these ATUs are inoperable.

What are the required actions per Tech Specs?

[REFERENCE PROVIDED]

A. Insert a trip in "B" PCIS logic within 24 hours.

B. Reduce power and isolate "B" MSL in 12 hours.

C.✓ Restore isolation capability in 1 hour, if not Isolate "B" MSL in 12 hours.

D. Be in MODE 3 in12 hours and MODE 4 in 36 hours.

Correct Answer: C

Excerpt from 2-ARP-9-5B:

BFN Unit 2	Panel 9-5 2-XA-55-5B	2-ARP-9-5B Rev. 0031 Page 27 of 43

RPS ANALOG TRIP UNIT TROUBLE
2-XA-99-1
23

(Page 1 of 1)

Sensor/Trip Point:
Solid state analog transmitters trip unit combination.

Sensor Location: Auxiliary Instrument Room.

Probable Cause:

A. Loss of power to ATU Panel 2-9-83, -84, -85, or -86 in Aux Instrument Room (supplies form Panels 2-9-15 and 2-9-17).
B. Gross failure of trip unit (input signal outside preset high or low limits indicating loss of process signal).
C. Card pulled from Panel 2-9-83, -84, -85, or -86.
D. Trip unit calibration in progress.

Automatic Action: Loss of both power supplies to an ATU Panel causes PCIS half-isolation (inboard or outboard) and could cause a half-scram.

Operator Action:

A. **DISPATCH** personnel to Aux Instrument Room to investigate cause. ☐
B. IF a power supply light on ATU Panel 2-9-83, -84, -85, or -86 is extinguished, **THEN**
DETERMINE cause and attempt power restoration. ☐
C. IF LATCH light on Trip Unit is illuminated indicating GROSS FAIL, **THEN**
DETERMINE cause and attempt reset. ☐

NOTE
Other parameters will be masked from alarming while this alarm is sealed in.

D. **MONITOR** other input parameters. ☐
E. IF necessary, **THEN**
INITIATE CR for Instrument Group to troubleshoot and repair. ☐
F. IF alarm is NOT valid, **THEN**
REFER TO 0-OI-55. ☐
G. IF a half-scram is received, **THEN**
With SRO permission, **RESET** half-scram. **REFER TO** 2-OI-99. ☐

References:

2-45E620-6	2-45E671-28	2-45E671-34
2-45E671-40	2-45E671-46	2-730E915-1
0-OI-55		

Excerpt from 2-OI-64:

BFN Unit 2	Primary Containment System	2-OI-64 Rev. 0127 Page 109 of 151

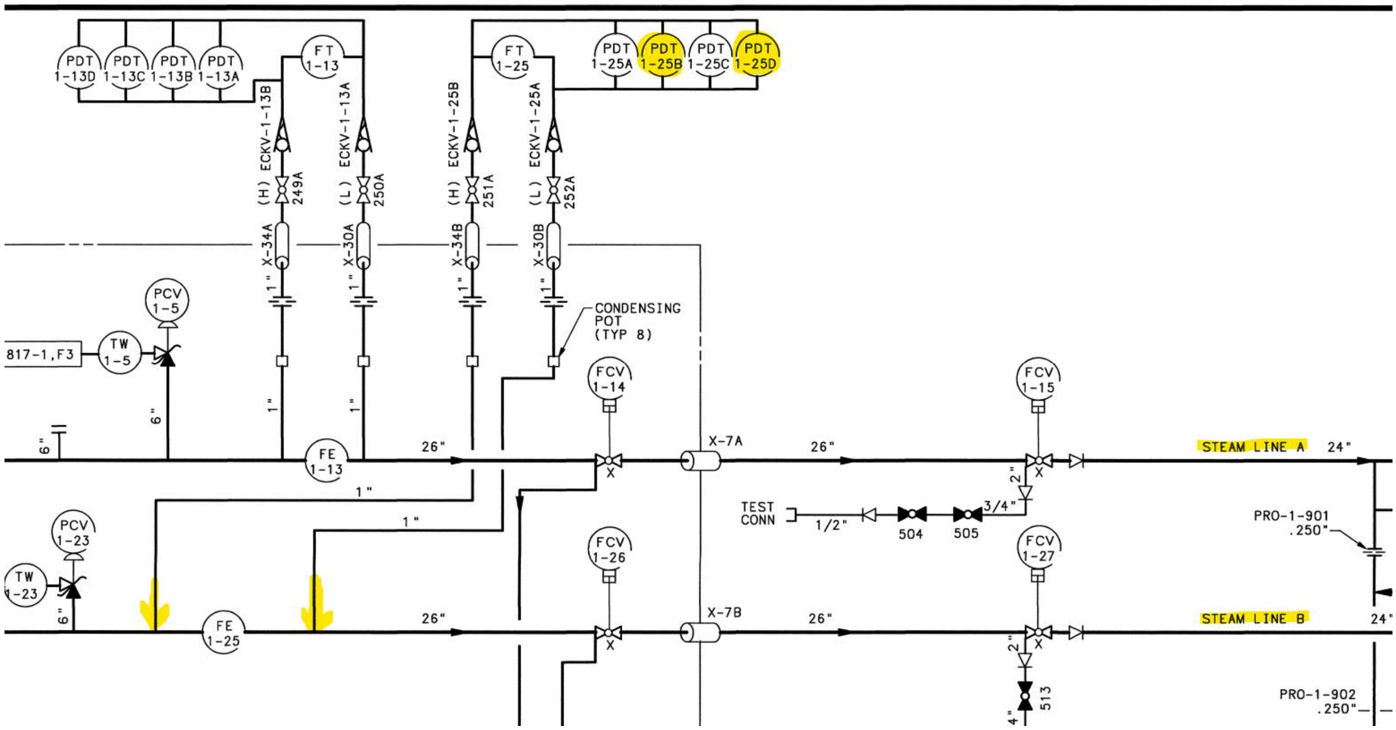
Attachment 2
(Page 4 of 10)

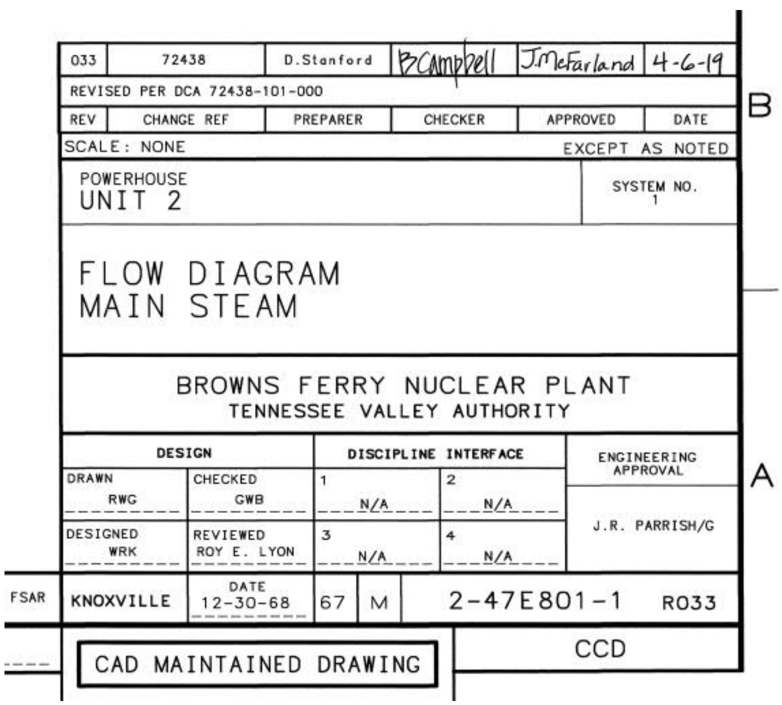
Actions to Place PCIS in Tripped Condition

(T.S. Table 3.3.6.1-1)

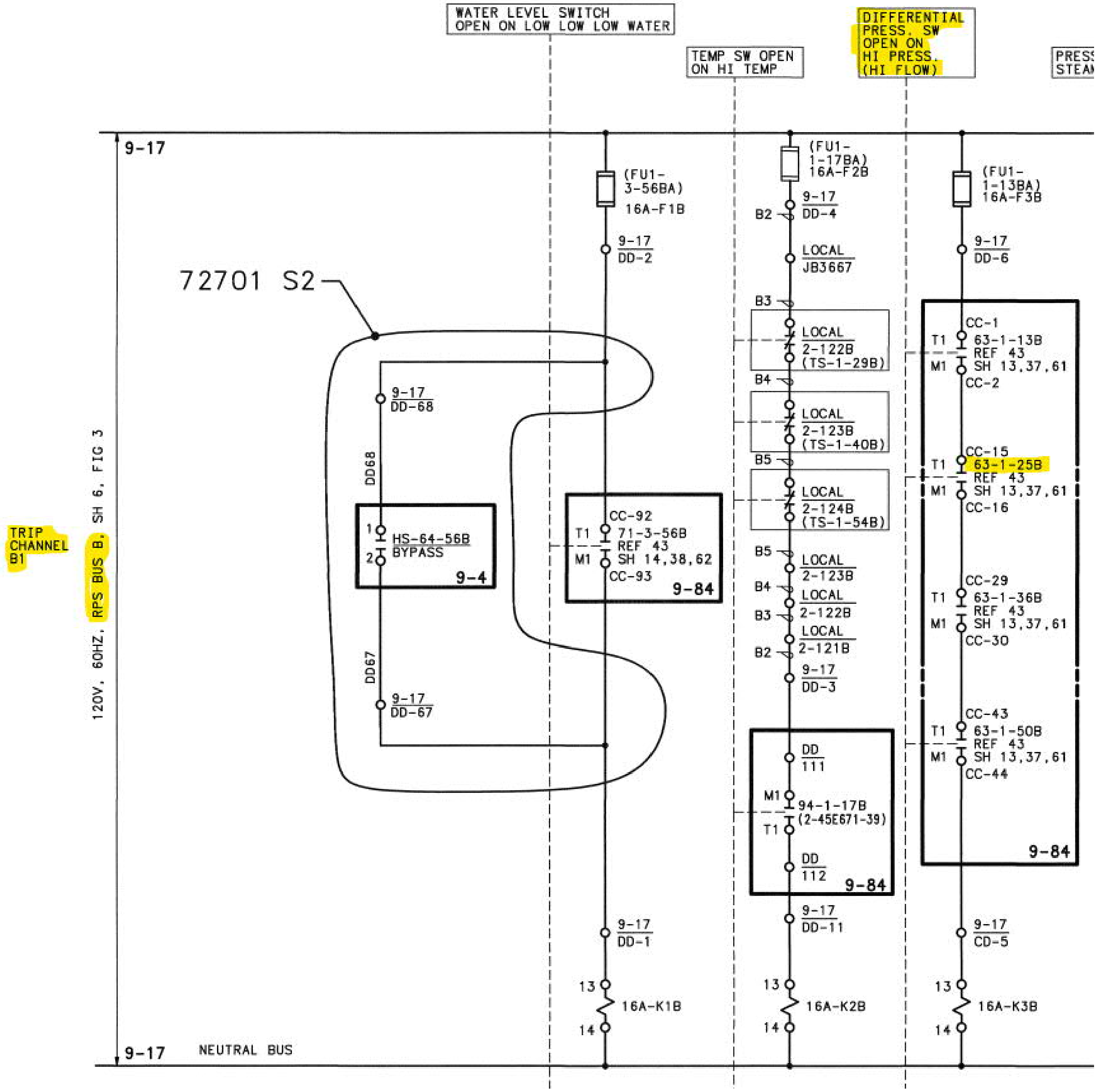
DEVICE	FUSE	RELAY	PANEL	PRINT	ALARM	REMARKS
2-PDIS-1-13A 2-PDIS-1-25A 2-PDIS-1-36A 2-PDIS-1-50A MSL FLOW HIGH **Function: 1c**	2-FU1-1-13AA (16A-F3A)	16A-K3A	9-15	2-730E927-7 2-45E671-25	2-XA-55-5B-18 MAIN STEAM LINE CH A FLOW HIGH	ALARMS AND GIVES 1/4 ISOLATION IN PCIS GROUP 1. NO PCIS DEVICES ACTUATE. A1 PCIS RED STATUS LIGHT EXTINGUISHES
2-PDIS-1-13B 2-PDIS-1-25B 2-PDIS-1-36B 2-PDIS-1-50B MSL FLOW HIGH **Function: 1c**	2-FU1-1-13BA (16A-F3B)	16A-K3B	9-17	2-730E927-8 2-45E671-37	2-XA-55-5B-19 MAIN STEAM LINE CH B FLOW HIGH	ALARMS AND GIVES 1/4 ISOLATION IN PCIS GROUP 1. NO PCIS DEVICES ACTUATE. B1 PCIS RED STATUS LIGHT EXTINGUISHES
2-PDIS-1-13C 2-PDIS-1-25C 2-PDIS-1-36C 2-PDIS-1-50C MSL FLOW HIGH **Function: 1c**	2-FU1-1-13CA (16A-F3C)	16A-K3C	9-15	2-730E927-7 2-45E671-31	2-XA-55-5B-18 MAIN STEAM LINE CH A FLOW HIGH	ALARMS AND GIVES 1/4 ISOLATION IN PCIS GROUP 1. NO PCIS DEVICES ACTUATE. A2 PCIS RED STATUS LIGHT EXTINGUISHES
2-PDIS-1-13D 2-PDIS-1-25D 2-PDIS-1-36D 2-PDIS-1-50D MSL FLOW HIGH **Function: 1c**	2-FU1-1-13DA (16A-F3D)	16A-K3D	9-17	2-730E927-8 2-45E671-43	2-XA-55-5B-19 MAIN STEAM LINE CH B FLOW HIGH.	ALARMS AND GIVES 1/4 ISOLATION IN PCIS GROUP 1. NO PCIS DEVICES ACTUATE. B2 PCIS RED STATUS LIGHT EXTINGUISHES.

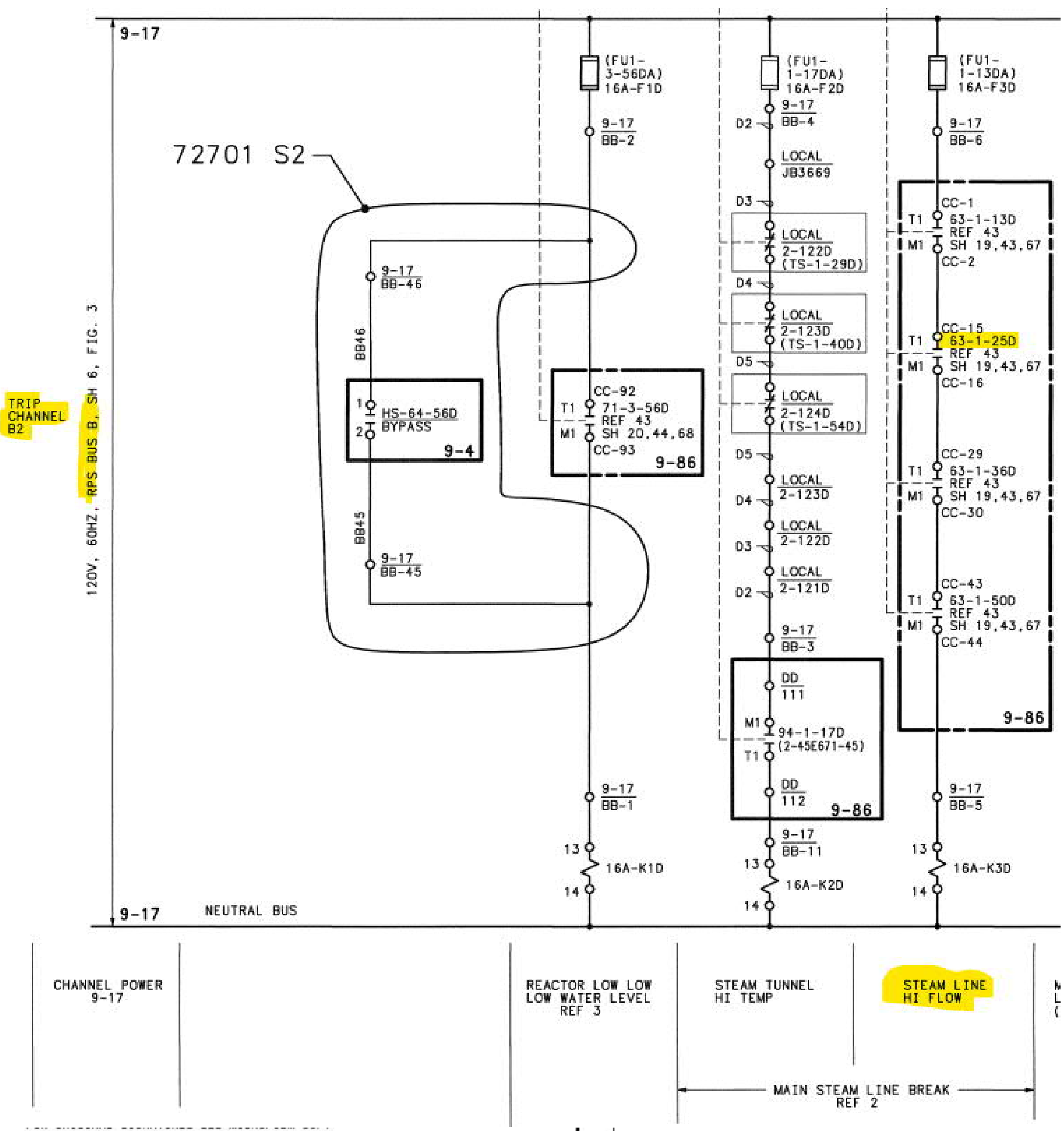
Excerpt from 2-47E801-1: Illustrates the given MSL FLOW HIGH transmitter 1-25B and 1-25D relation to 'B' MSL





Excerpts from 2-730E927-8: Illustrates the given DP Pressure Switches 1-25B and 1-25D relation to PCIS Trip Channel B1 and B2 powered by RPS Bus B





024	72701 S2	D.Stanford	J.McFarland	B.Campbell	4/5/19	
REVISED PER DCA 72701-2005-000, -2006-000						
REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE	

SCALE: NONE — EXCEPT AS NOTED

POWERHOUSE
UNIT 2 — SYSTEM NO. 64

ELEMENTARY DIAGRAM
PRIMARY CONTAINMENT
ISOLATION SYSTEM

BROWNS FERRY NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY

DESIGN		INITIAL ISSUE	ENGINEERING APPROVAL
DRAFTER *	CHECKER *	RO ISSUE PER EXCEPTION A CHANGE CONTROL, SUPERSEDES AC DWG 730E927-8 TVA REV D AND AD DWG 730E927-8 TVA REV 905	1 *
DESIGNER *	REVIEWER *		2 *
			3 *
DATE 4-1-88	67 E	2-730E927-8	R024

CAD MAINTAINED DRAWING — CCD

Excerpts from Unit 2 Tech Spec 3.3.6.1:

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

Supports Distractor (A):

ACTIONS

--NOTE--
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	--------------NOTE------------ Only applicable for Function 1.d if two or more channels are inoperable. ---------------------------------- Place channel in trip.	12 hours for Functions 2.a, 2.b, 5.h, 6.b, and 6.c AND 24 hours for Functions other than Functions 2.a, 2.b, 5.h, 6.b, and 6.c
	AND		
	A.2	--------------NOTE------------ Only applicable for Function 1.d when 15 of 16 channels are OPERABLE. ---------------------------------- Place channel in trip.	30 days

(continued)

Supports Correct Answer and Distractors (B) and (D):

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour OR 4 hours for Function 1.d when normal ventilation is not available
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated Main Steam Line (MSL). OR D.2.1 Be in MODE 3. AND D.2.2 Be in MODE 4.	12 hours 12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1 Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. OR Required Action and associated Completion Time for Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	12 hours 36 hours

(continued)

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 398 inches above vessel zero
b. Main Steam Line Pressure - Low[c]	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 825 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140% rated steam flow
d. Main Steam Tunnel Temperature - High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 528 inches above vessel zero
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5	≤ 2.5 psig

Excerpt from OPL171.017 Lesson Plan:

a) This arrangement creates trip sub-channels A1/A2 and B1/B2.

b) A trip of either sensor relay within a trip channel will cause opening of the associated contact and de- energization of the associated relay. This condition will create a "half Isolation" signal within both logic channels but NO VALVE MOVEMENT.

HPCI/RCIC are energize to actuate circuits.

(1) These are systems which are not required for post-accident mitigation.

(2) These systems are either isolated immediately upon receipt of a PCIS Isolation signal, or are provided with manual valves that are locked closed when primary containment is required.

(3) Should a trip of either sensor relay in the other trip channel occur, conditions will exist to de-energize the valve actuation relays in each logic channel, causing both Isolation valves to close.

PCIS logic is arranged as follows:

A1 OR A2
AND
B1 OR B2 **= Inboard AND Outboard valve closure**

Note: Most PCIS logic is assembled as above. The MSL drains however are an exception.

The MSL drain logic is as follows:
A1 **AND** B1 = I/B valve closure
A2 **AND** B2 = O/B valve closure

D. Group 1 (MSIV) Isolation Logic

1. Figure-2 provides a simplified diagram of the Isolation logic for the "A" main steam line inboard Isolation valve (FCV-1-14).

2. The MSIV is provided with both an AC-powered pilot solenoid (FSV-1-14C) and a DC-powered pilot solenoid (FSV-1-14B).

   Both of these pilot solenoids must be de-energized to cause the MSIV to close.

2-730E927-10
Figure-2
ILT- 2a, 3a
LOR- 2a, 3a

Examination Outline Cross-reference:

G2.1.7 **(10CFR 55.43.2 – SRO Only)**

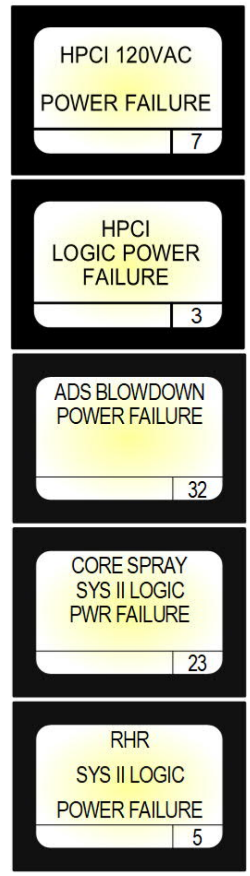
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

	RO	SRO
Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.1.7	
Importance Rating	------	4.7

Proposed Question: **# 94**

Unit 2 is operating at 100% RTP, with the following conditions:

- HPCI 120 VAC POWER FAILURE, (2-9-3F, Window 7) alarms
- HPCI LOGIC POWER FAILURE, (2-9-3F, Window 3) alarms
- ADS BLOWDOWN POWER FAILURE, (2-9-3C, Window 32) alarms
- CORE SPRAY SYS II LOGIC PWR FAILURE, (2-9-3F, Window 23) alarms
- RHR SYS II LOGIC POWER FAILURE, (2-9-3E, Window 5) alarms



Which **ONE** of the following completes the statements below?

250V DC RMOV Board ___**(1)**___ has been de-energized.

In accordance with Tech Spec Bases 3.8.7, Distribution Systems – Operating, if this RMOV Board is transferred to its alternate source, it ___**(2)**___.

A. (1) 2A
(2) remains OPERABLE

B. (1) 2A
(2) must be declared INOPERABLE

C. (1) 2B
(2) remains OPERABLE

D. (1) 2B
(2) must be declared INOPERABLE

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct *(See B).* The second part is incorrect but plausible in that the candidate is required to recall this information from memory from the Tech Spec 3.8.7 Distribution Systems - Operating Bases, and the board remaining OPERABLE is a possibility. Additionally, the alternate battery that is loaded as a result of the transfer remains OPERABLE, which adds to plausibility of the RMOV Board remaining OPERABLE when transferred to alternate power. For example, Unit 1 and 2 Diesel Auxiliary Boards can be placed on their alternate feeder breakers and considered OPERABLE as long as the restrictions on the associated drawings are met.

B **CORRECT**: *(See attached)* In accordance with Alarm Response Procedures, a loss of RMOV Board 2A is a cause for HPCI 120 VAC POWER FAILURE, (2-ARP-9-3F Window 7) and CORE SPRAY SYS II LOGIC PWR FAILURE, (2-9-3F, Window 23). For second part, in accordance with Tech Spec 3.8.7 Bases, the Unit 2 250V DC RMOV Boards 2A, 2B, and 2C have alternate power supplies from another 250V Unit DC Board. These boards are considered INOPERABLE when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

C INCORRECT: First part is incorrect but plausible given that the 250V DC ECCS power supply convention is opposite of the standard convention and often confused. For Unit 2, DIV I ECCS ATU inverters are powered from 250V DC RMOV Board 2B while DIV II is from 250V DC RMOV Board 2A. Second part is incorrect but plausible *(See A).*

D INCORRECT**:** First part is incorrect but plausible (*See C).* Second part is correct *(See B).*

SRO Level Justification: Tests the candidate's ability to evaluate plant performance and make operational judgments based on operating characteristics as it relates to ECCS power supplies versus plant conditions and Tech Specs. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-ARP-9-3C, Rev. 28 (Attach if not previously provided)

2-ARP-9-3E, Rev. 31

2-ARP-9-3F, Rev. 40

U2 Tech Spec Bases 3.8.7, Rev. 76

Proposed references to be provided to applicants during examination: **HPCI 120 VAC POWER FAILURE, (2-9-3F, Window 7), HPCI LOGIC POWER FAILURE, (2-9-3F, Window 3), ADS BLOWDOWN POWER FAILURE, (2-9-3C, Window 32), CORE SPRAY SYS II LOGIC PWR FAILURE, (2-9-3F, Window 23), RHR SYS II LOGIC POWER FAILURE, (2-9-3E, Window 5)**

Learning Objective: OPL171.037 Obj. 9_ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Excerpts from 2-ARP-9-3F:

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0040 Page 11 of 41

HPCI 120 VAC POWER FAILURE 7	Sensor/Trip Point: Relay 23A-K50	Loss of the 120 VAC from DIV II ECCS ATU inverter and Loss of power to the HPCI Flow IND Controller (2-FIC-73-33)

(Page 1 of 1)

Sensor Location: Panel 9-19
Aux Instr Rm, El 593'

Probable Cause:
A. Blown fuses, Fuse 2-FU2-073-0033C, Panel 2-9-82 AA1 & AA2,
B. DIV II ECCS ATU inverter failure.
C. Loss of 250V DC power supply to DIV II ECCS ATU inverter (RMOV BD 2A compt 11A1).

Automatic Action:
A. HPCI controller loses power. HPCI becomes inoperable.
B. If HPCI is in service, the HPCI Turbine Stop Valve, 2-FCV-73-18, closes. HPCI controller loses power. HPCI becomes inoperable.
C. 2-PI-064-67B will lose power and become inop.

Operator Action:
A. **DISPATCH** personnel to CHECK the following:
- Fuses 2-FU2-073-0033C, Panel 2-9-82, AA1 & AA2.
- DIV II ECCS ATU inverter.
- DIV II ECCS ATU inverter breaker, RMOV BD 2A, compt 11A1.

B. 2-PI-064-67B will lose power and become inop.
REFER TO TECH SPEC 3.3.3.1, Table 3.3.3.1-1, TRM 3.3.5.
C. **REFER TO**: Tech Spec 3.5.1. Tech Spec 3.3.3.1, Table 3.3.3.1-1.

References: 2-45E620-1 GE 730E928-2 and -4
Technical Specifications 3.5.1, and 3.3.3.1
Technical Specifications Bases 3.3.3.1, TRM 3.3.5

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0040 Page 6 of 41

HPCI LOGIC POWER FAILURE (3)	Sensor/Trip Point:	
	Relay 23A-K39 (Bus A)	Loss of 250V DC
	Relay 23A-K44 (Bus B)	Control Power
	Relay 23A-K44B (Bus B)	

(Page 1 of 1)

Sensor Location:
Panel 9-32, Bus A — Aux Instr Rm, El 593'
Panel 9-39, Bus B — Aux Instr Rm, El 593'

Probable Cause:
A. Cleared fuse(s).
B. Loss of 250V DC power supply to panels.

Automatic Action:
A. Logic Bus A failure renders Channel A trip and automatic isolation logic inop. HPCI continues to function.
B. Logic Bus B failure renders Channel B trip, automatic initiation, and automatic isolation logics inop. If HPCI is in service, it can only be shut down by closing the 73-16 Steam Admission valve, however the aux oil pump will not start on coastdown. HPCI becomes inoperable.

Operator Action:
A. **DETERMINE** which logic bus has failed, **REFER TO** automatic action section.
B. **DISPATCH** personnel to check source of power failure:
 1. Logic Bus A
  a. Fuses 2-FU2-073-23A-K36 (23A-F19) and 2-FU2-073-23A-K36 (23A-F20), Panel 9-32.
  b. Power supply 250V DC Rx Mov Bd 2B, Breaker 1B1.
 2. Logic Bus B
  a. Fuses 2-FU2-073-0039A and 2-FU2-073-0039B, Panel 9-39.
  b. Power supply 250V DC RMOV Bd 2A, Breaker 11D1.

C. **REFER TO** Tech Spec 3.5.1, 3.5.2, 3.3.5.1, 3.3.6.1, and TRM 3.3.3.4.

References:
2-45E620-1 — GE 730E928-2-3 and -4.
Technical Specifications 3.3.5.1, 3.3.6.1,3.5.1,3.5.2, — TRM 3.3.3.4

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0040 Page 27 of 41

CORE SPRAY SYS II LOGIC PWR FAILURE	23

(Page 1 of 1)

Sensor/Trip Point:

2-RLY-075-14A-K3B Loss of power to 250V DC Bus B

Sensor Location: Auxiliary Instrument Room
Elevation 593'
Panel 9-33

Probable Cause:

A. Fuses in logic circuit cleared.
   1. Fuse 2-FU2-75-14A/K5B (14A-F1B), 250V DC bus positive.
   2. Fuse 2-FU2-75-14A/K5B (14A-F2B), 250V bus negative.

B. Breaker 9A2 open on 250V Reactor MOV board 2A.

Automatic Action: None

Operator Action:

A. **DISPATCH** personnel to perform the following:
   - **CHECK** position of Breaker 9A2 on 250V Reactor MOV Board 2A.
   - **CHECK** fuses 2-FU2-75-14A/K5B (14A-F1B and 14A-F2B) located on Panel 9-33, elevation 593', Auxiliary Instrument Room.

B. **REFER TO** Tech Spec 3.3.5.1, 3.5.1, 3.5.2,

NOTE

1) **IF** alarm is valid, **THEN**
- 2B and 2D Core Spray Pumps will **NOT** auto start.
- SYS II Inboard Injection Valve will **NOT** auto open.
- SYS II Inboard Injection Valve will **NOT** open manually from the control room.

References: 2-45E620-1 GE 730E930-18 2-45E712-1
Technical Specifications 3.3.5.1, 3.5.1, 3.5.2

Excerpt from 2-ARP-9-3C:

BFN Unit 2	Panel 9-3 2-XA-55-3C	2-ARP-9-3C Rev. 0028 Page 39 of 42

ADS BLOWDOWN POWER FAILURE

32

(Page 1 of 1)

Sensor/Trip Point:

Relay 2E-K40	Panel 9-33	De-energized
Relay 2E-K1A	Panel 9-30	De-energized
Relay 2E-K1B	Panel 9-33	De-energized
Relay 2E-K12	Panel 9-30	De-energized
Relay 2E-K32	Panel 2-25-32	De-energized
Relay 2E-K33	Panel 2-25-32	De-energized
Relay 2E-K37	Panel 2-25-32	De-energized
Relay 2E-K38	Panel 2-25-32	De-energized

Sensor Location:
Panel 2-25-32
Backup Control Center
El 621', R-13 Q-LINE

Panel 2-9-30 and 2-9-33
Aux Instrument Rm
El 593'

Probable Cause:
A. Cleared Fuse(s)
B. Loss of 250V DC power supply to panels.
C. Auto Xfer of Logic Bus B Power Supplies.

Automatic Action:
Main steam auto relief valves, PCV-1-22 and 30, auto transfer power supply from 250V DC Rx Mov Bd A to 250 DC Rx Mov Bd B (PCV-1-22) & Bd C (PCV-1-30) on loss of normal power supply.
Logic Bus "B" transferred to Alternate supply (250V Rmov Bd 2A) upon Loss of Normal Supply (250V RMOV Bd 2B) or fuse failure.

Operator Action:
A. **CHECK** power is available to PCV-1-22 and -30.
B. IF annunciator HPCI LOGIC POWER FAILURE, XA-55-3F Window 3, is in alarm, this is indicative of loss of power to 250V DC Rx Mov Bd 2A/2B. **DISPATCH** personnel to check 250V DC Rx Mov Bd 2A Breaker 11A2 and 250V DC Rx MOV Bd 2B Breaker 1B1.
C. **DISPATCH** personnel(s) to check:
   1. Logic Bus A
      a. 250V DC Rx Mov Bd 2B, Breaker 1F1.
      b. Fuses 2-FU2-001-2E-K3 in Panel 9-30.
   2. Logic Bus B
      a. 250V DC Rx Mov Bd 2A, Breaker 9A1.
      b. Fuses 2-FU1-001-2E/K22A and 2-FU1-001-2E/K22B on Panel 9-33.
      c. Fuses 2-FU2-001-2E-K13 in Panel 9-30.
      d. Fuses 2-FU2-1-2E-K11A and 2-FU2-1-2E-K11) on Panel 9-33 (GG Block).

D. **REFER TO** Tech Spec Section 3.5.1.
E. **REFER TO** TRM 3.3.3.4.

References:
2-45N620-2 2-45E712-1, -2 and -3 GE 730E929 -1, -2 and 3
Technical Specifications 3.5.1 TRM 3.3.3.4.

Excerpt from 2-ARP-9-3E:

BFN Unit 2	Panel 9-3 2-XA-55-3E	2-ARP-9-3E Rev. 0031 Page 8 of 41

RHR SYS II LOGIC POWER FAILURE 5	Sensor/Trip Point: 2-RLY-074-10A-K1B Lose of 250V DC power.

(Page 1 of 1)

Sensor Location: Panel 9-33
Aux. Inst. Rm., El 593'

Probable Cause:
A. Fuses in logic circuit cleared:
- 2-FU2-74-10A/36B (10A-F1B) 250v DC bus positive.
- 2-FU2-74-10A/36B (10A-F2B) 250v DC bus negative.

B. Loss of 250V DC power supply.
C. Sensor failure.

Automatic Action: None

Operator Action:
A. **DISPATCH** personnel to perform the following:
- **CHECK** 250V DC Rx MOV Bd. 2A Breaker 8B1 to verify position. ☐
- **CHECK** fuses 2-FU2-74-10A/36B (10A-F1B and 10A-F2B) located on panel 9-33, El 593', Aux Instrument Room. ☐

B. **REFER TO** Tech Spec 3.3.5.1 and TRM 3.3.3.4. ☐

NOTE

1) **IF** alarm is valid, **THEN**
- 2B RHR Pump will **NOT** auto start.
- 2D RHR Pump will **NOT** auto start.
- SYS II Inboard Injection Valve will **NOT** receive an auto open signal from DIV II Logic.
- SYS II Inboard Injection Valve will **NOT** manually open from the control room due to loss of 450 psig logic from DIV II.

References: Technical Specifications 5.4 and 5.5 | Technical Requirements Manual 3.3.3.4 | FSAR Section 8.6.4.2 and 13.0
2-45E620-1 | 2-45E712-1

Excerpts from Unit 2 Tech Spec Bases 3.8.7:

BASES

LCO (continued)

When 480 V Shutdown Board 2B is aligned to the alternate supply 4.16 kV Shutdown Board C, a LOCA/LOOP with a failure of the Shutdown Board D Battery would disable the normal supply 4.16 kV Shutdown Board D, and would also prevent the 480 V Shutdown Board 2B from load shedding its 480 V loads which would overload the alternate supply Diesel Generator D. This would result in the loss of diesel generators C and D, associated 4.16 kV shutdown boards and RHRSW pumps. Therefore, the restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 2B is on its alternate supply.

The Unit 2 480 V RMOV boards 2A, 2B, 2D, and 2E have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 2 250 V DC RMOV boards 2A, 2B, and 2C have alternate power supplies from another 250 V Unit DC board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

If a 4.16 kV or 480 V shutdown board is aligned to its alternate 250 V DC control power source, a single failure of the alternate power source could affect both ECCS divisions and common equipment needed to support the other units depending on the board alignment. Therefore, the restrictions on the associated drawings shall be adhered to whenever a 4.16 kV or 480 V shutdown board is on its alternate control power supply.

(continued)

Supports Distractors A(2), C(2):

BASES

LCO (continued)

one or more of these boards become inoperable due to a failure not affecting the OPERABILITY of a board listed in Table B 3.8.7-1 (e.g., a breaker supplying a single MCC fails open), the individual loads on the board would be considered inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads would be entered. However, if one or more of these boards is inoperable due to a failure also affecting the OPERABILITY of a board listed in Table B 3.8.7-1 (e.g., loss of a 4.16 kV shutdown board, which results in de-energization of all boards powered from the 4.16 kV shutdown board), then although the individual loads are still considered inoperable, the Conditions and Required Actions of the LCO for the individual loads are not required to be entered, since LCO 3.0.6 allows this exception (i.e., the loads are inoperable due to the inoperability of a support system governed by a Technical Specification; the 4.16 kV shutdown board).

The Unit 1 and 2 diesel auxiliary boards can be placed on their alternate feeder breakers and considered OPERABLE as long as the restrictions on the associated drawings are met.

If the 480 V Shutdown Board 2A is placed on the alternate supply 4.16 kV Shutdown Board C, a LOCA/LOOP with a failure of the Shutdown Board B Battery would disable the normal supply 4.16 kV Shutdown Board B, and would also prevent the 480 V Shutdown Board 2A from load shedding its 480 V loads which would overload the alternate supply Diesel Generator C. This would result in the loss of 4.16 kV Shutdown Boards B and C which would impact both divisions ECCS in Units 1 and 2. Therefore, the time limitations and restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 2A is on its alternate supply.

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.41 (10CFR 55.43.6 – SRO Only)	Tier #	------	3
Knowledge of the refueling process.	Group #	------	------
	K/A #	G2.1.41	
	Importance Rating	------	3.7

Proposed Question: **# 95**

Unit 2 is in MODE 5 during a scheduled refueling outage. A **NON-SPIRAL** core reload is in progress with the following conditions:

- SRM B is **INOPERABLE**
- The next fuel bundle to be moved is currently located in the Spent Fuel Pool and is designated for Reactor Cavity position **10-15**
- As the respective fuel bundle is grappled, SRM C fails downscale and is declared **INOPERABLE**
- **ALL** other SRMs are **OPERABLE**

Which **ONE** of the following completes the statement below?

Given the conditions above, fuel moves ________ in accordance with Tech Specs and 0-GOI-100-3C, Fuel Movement Operations During Refueling.

[REFERENCE PROVIDED]

A. **CAN** continue since the SRM in the **AFFECTED** core quadrant **ONLY** is **OPERABLE**

B. **CANNOT** continue since the SRM in the **AFFECTED** core quadrant is **INOPERABLE**

C. **CANNOT** continue since the SRMs in the **ADJACENT** core quadrants are **INOPERABLE**

D. **CAN** continue since the SRMs in the **AFFECTED AND ADJACENT** core quadrants are **OPERABLE**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible if the candidate does not correctly apply the Tech Spec 3.3.1.2 requirement for the non-spiral operations that TWO SRMs (affected and one adjacent quadrant) are required to be OPERABLE to provide redundant monitoring of reactivity changes in the core.

B INCORRECT: Incorrect but plausible (*See A*).

C INCORRECT: Incorrect but plausible (*See D*).

D **CORRECT**: (*See attached*) In accordance with 0-GOI-100-3C, Fuel Movement Operations During Refueling, and Tech Spec 3.3.1.2, SRM Instrumentation, with one or more required SRMs INOPERABLE in MODE 5, Immediately suspend CORE ALTERATIONS except for Control Rod insertion. In accordance with Tech Spec Bases 3.3.1.2, in **NON-SPIRAL** routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes in the Reactor Core. This is accomplished by requiring one SRM (D in this case) to be OPERABLE in the quadrant (affected) where CORE ALTERATIONS are being performed and the other SRM (A in this case) to be **OPERABLE** in the **ADJACENT** quadrant containing fuel.

SRO Level Justification: Tests the candidate's knowledge of refueling procedure requirements ensuring that the reactivity of the Core will be continuously monitored during any Reactor Core Alterations. Requires knowledge of Tech Spec Bases associated with duties unique to the SRO position. SRO only because of link to 10CFR55.43 (7): Fuel handling facilities and procedures AND 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): U2 Tech Spec 3.3.1.2, Amend. 253 (Attach if not previously provided)
U2 TS Bases 3.3.1.2, Rev. 0
0-GOI-100-3C, Rev. 94
2-GOI-100-1A, Rev. 179

Proposed references to be provided to applicants during examination: **2-GOI-100-1A, Attachment 8 (Page 1 of 1) Core Quadrants/Octants**

Learning Objective: OPL171.053 Obj. 9 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1909 #95	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41
55.43 **X**

Copy of Bank Question:

Proposed Question: **# 95**

Unit 2 is in MODE 5 during a scheduled refueling outage. A **NON-SPIRAL** core reload is in progress with the following conditions:

- SRM B is **INOPERABLE**
- The next fuel bundle to be moved is currently located in the Spent Fuel Pool and is designated for Reactor Cavity position 09-42
- As the respective fuel bundle is grappled, SRM D fails downscale and is declared **INOPERABLE**
- **ALL** other SRMs are **OPERABLE**

Which **ONE** of the following completes the statement below?

Given the conditions above, fuel moves ________ in accordance with Tech Specs and 0-GOI-100-3C, Fuel Movement Operations During Refueling.

[SEE THE ATTACHED CORE QUADRANT ILLUSTRATION]

A. **CAN** continue since the SRM in the **AFFECTED** core quadrant is **OPERABLE**

B. **CAN** continue since the SRM in the **ADJACENT** core quadrant is **OPERABLE**

C. **CANNOT** continue since the SRM in the **AFFECTED** core quadrant is **INOPERABLE**

D. **CANNOT** continue since the SRM in the **ADJACENT** core quadrant is **INOPERABLE**

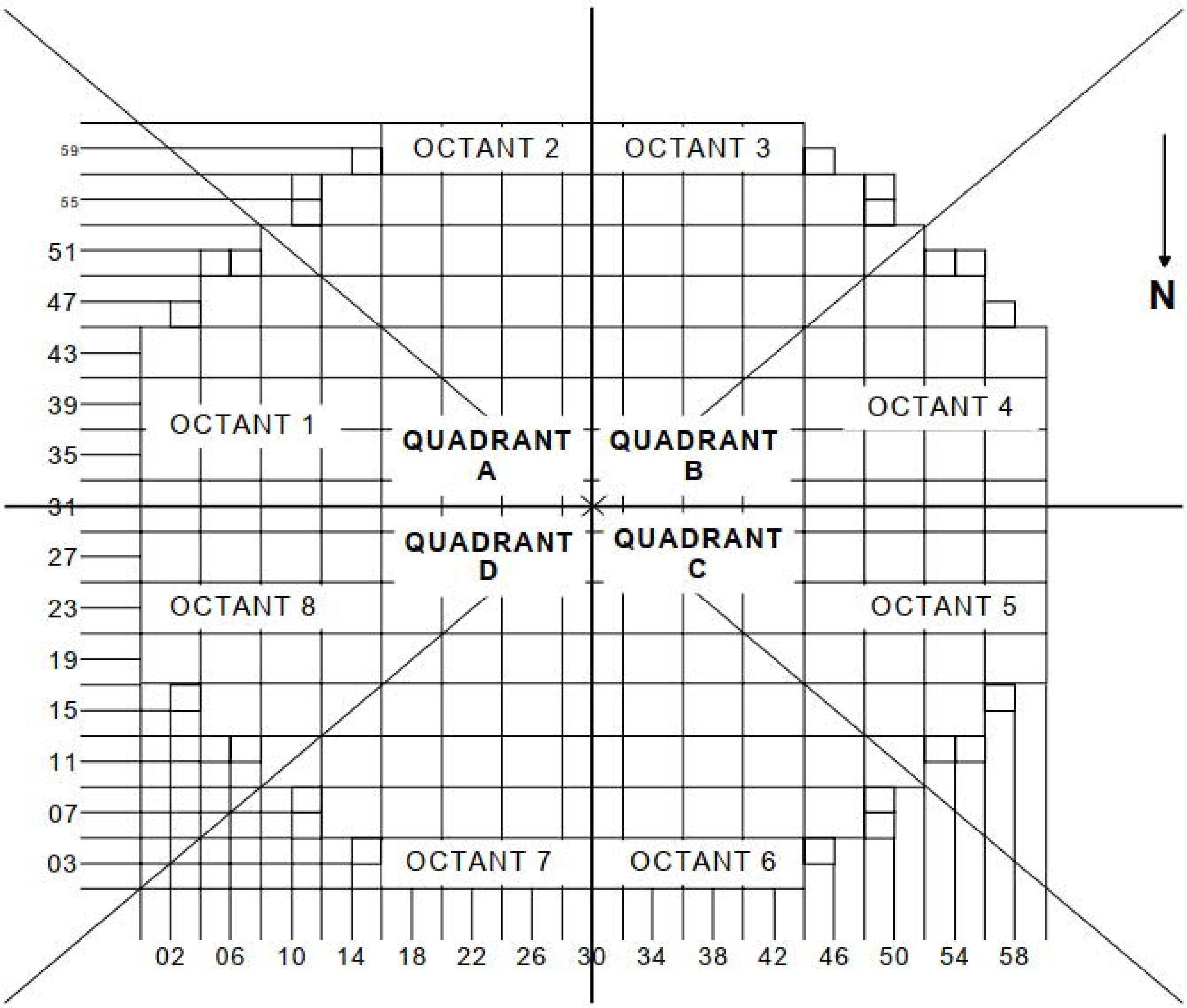
Proposed Answer: **D**

REFERENCE PROVIDED to candidate:

BFN Unit 2	Unit Startup and Power Operation	2-GOI-100-1A Rev. 0179 Page 206 of 207

Attachment 8
(Page 1 of 1)

Core Quadrants/Octants



CORE QUADRANTS / OCTANTS

Excerpts from Tech Spec 3.3.1.2: (Not provided to candidate)

3.3 INSTRUMENTATION

3.3.1.2 Source Range Monitor (SRM) Instrumentation

LCO 3.3.1.2 The SRM instrumentation in Table 3.3.1.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.2-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	A.1	Restore required SRMs to OPERABLE status.	4 hours
B. Three required SRMs inoperable in MODE 2 with IRMs on Range 2 or below.	B.1	Suspend control rod withdrawal.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	12 hours

(continued)

SRM Instrumentation
3.3.1.2

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more required SRMs inoperable in MODE 3 or 4.	D.1	Fully insert all insertable control rods.	1 hour
	AND		
	D.2	Place reactor mode switch in the shutdown position.	1 hour
E. One or more required SRMs inoperable in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

Table 3.3.1.2-1 (page 1 of 1)
Source Range Monitor Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Source Range Monitor	2(a)	3	SR 3.3.1.2.1 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	3,4	2	SR 3.3.1.2.3 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	5	2(b)(c)	SR 3.3.1.2.1 SR 3.3.1.2.2 SR 3.3.1.2.4 SR 3.3.1.2.5 SR 3.3.1.2.7

(a) With IRMs on Range 2 or below.

(b) Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector.

(c) Special movable detectors may be used in place of SRMs if connected to normal SRM circuits.

Excerpts from Tech Spec Bases 3.3.1.2:

BASES

LCO (continued)

In MODES 3 and 4, with the reactor shut down, two SRM channels provide redundant monitoring of flux levels in the core.

In MODE 5, during a spiral offload or reload, an SRM outside the fueled region will no longer be required to be OPERABLE, since it is not capable of monitoring neutron flux in the fueled region of the core. Thus, CORE ALTERATIONS are allowed in a quadrant with no OPERABLE SRM in an adjacent quadrant provided the Table 3.3.1.2-1, footnote (b), requirement that the bundles being spiral reloaded or spiral offloaded are all in a single fueled region containing at least one OPERABLE SRM is met. Spiral reloading and offloading encompass reloading or offloading a cell on the edge of a continuous fueled region (the cell can be reloaded or offloaded in any sequence).

In nonspiral routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes occurring in the reactor core. Because of the local nature of reactivity changes during refueling, adequate coverage is provided by requiring one SRM to be OPERABLE in the quadrant of the reactor core where CORE ALTERATIONS are being performed, and the other SRM to be OPERABLE in an adjacent quadrant containing fuel. These requirements ensure that the reactivity of the core will be continuously monitored during CORE ALTERATIONS.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

With one or more required SRM inoperable in MODE 5, the ability to detect local reactivity changes in the core during refueling is degraded. CORE ALTERATIONS must be immediately suspended and action must be immediately initiated to insert all insertable control rods in core cells containing one or more fuel assemblies. Suspending CORE ALTERATIONS prevents the two most probable causes of reactivity changes, fuel loading and control rod withdrawal, from occurring. Inserting all insertable control rods ensures that the reactor will be at its minimum reactivity given that fuel is present in the core. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative position.

Action (once required to be initiated) to insert control rods must continue until all insertable rods in core cells containing one or more fuel assemblies are inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each SRM Applicable MODE or other specified conditions are found in the SRs column of Table 3.3.1.2-1.

SR 3.3.1.2.1 and SR 3.3.1.2.3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on another channel. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

(continued)

Excerpts from 0-GOI-100-3C:

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3.7 Neutron Monitoring

A. With fuel in the Reactor Vessel the following neutron monitoring must be operable for refueling.

1. At least two operable SRMs, one located in the quadrant where core alterations are being performed and one adjacent to the quadrant where core alterations are being performed, except as specified in Tech Spec 3.3.1.2.

2. If a complete core off-load is being performed, the SRMs must be initially operable.

3. SRM count rate is NOT required to be greater than 3.0 CPS with four or less adjacent fuel assemblies and no other fuel assemblies in the associated quadrant.

BFN Unit 0	Fuel Movement Operations During Refueling	0-GOI-100-3C Rev. 0094 Page 35 of 141

4.0 PREREQUISITES (continued)

[7] During CORE ALTERATIONS, except as specified in Tech Spec 3.3.1.2, two operable SRMs shall be inserted to the normal level and located in and adjacent to the quadrant where core alterations are being performed. Each fueled region shall be monitored by at least one SRM/FLC.

- When four or more fuel assemblies are adjacent to an SRM (FLC), it must be reading ≥ 3 cps and have a signal-to-noise ratio of ≥ 3:1 except as specified in Tech Spec 3.3.1.2.
- Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fuel region includes only that SRM detector. Tech Spec Table 3.3.1.2-1
- The following is provided to clarify quadrant locations.

Fuel Moved in Quadrant	Required Operable SRM/FLC Quadrant Locations
A	A&B or A&D
B	A&B or B&C
C	B&C or C&D
D	C&D or A&D

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.14 (10CFR 55.43.3 – SRO Only)	Tier #	------	3
Knowledge of the process for controlling equipment configuration or status.	Group #	------	------
	K/A #	G2.2.14	
	Importance Rating	------	4.3

Proposed Question: **# 96**

Which **ONE** of the following completes the statements below concerning mispositioned components?

In accordance with NPG-SPP-10.1, System Status Control, a Mispositioned Component is any active component found out of the expected position for existing conditions when the component's required position is tracked by ___**(1)**___.

Following revision to an Equipment Alignment Checklists, the ___**(2)**___ Manager may authorize verification of only those items that were added or changed by the revision.

A. (1) procedures and clearances ONLY
(2) Shift

B. (1) procedures and clearances ONLY
(2) Engineering

C. (1) procedures, clearances, work orders, or TACFs
(2) Shift

D. (1) procedures, clearances, work orders, or TACFs
(2) Engineering

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that in accordance with NPG-SPP-10.1, System Status Control, procedures and clearances are correct, but components found out of position per work orders and the Temporary Modification process (TACFs) are also considered Mispositioned Components. The second part is correct (*See C*).

B INCORRECT: The first part is incorrect but plausible (*See A*). The second part is incorrect but plausible in that in accordance with NPG-SPP-10.1, System Status Control, the Engineering Manager is a quorum member of the Configuration Control Committee. One of the functions of this Committee is to evaluate plant activities, maintenance, and modifications that can affect status control and clearances. Therefore, it is plausible that the Engineering Manager could be responsible for authorizing verification of certain items added or changed by a revision.

C **CORRECT**: *(See attached)* The first part is correct in that in accordance with NPG-SPP-10.1, System Status Control, a Mispositioned Component is any active component found out of the expected position for existing plant conditions when the component's required position is tracked by procedures, clearances, work orders, or Temporary Modifications (TACFs). The second part is correct in that in accordance with NPG-SPP-10.1, System Status Control, the Shift Manager may authorize verification of certain items added or changed by a revision.

D INCORRECT: The first part is correct (*See C*). The second part is incorrect (*See B*).

SRO Level Justification: Tests the candidate's knowledge of the process for controlling equipment configuration or status in accordance with NPG-SPP-10.1, System Status Control. SRO only because of link to 10CFR55.43 (3): Facility licensee procedures required to obtain authority for design and operating changes in the facility. This question is rated as memory due to strictly recalling facts related to System Status Control.

Technical Reference(s): NPG-SPP-10.1, Rev.12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.113 Obj. 2,9 (As available)

Question Source: Bank # ILT EXAM BANK OPL171.113-09 002 #2286

Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

2286. OPL171.113-09 002

In accordance with NPG-SPP-10.1, System Status Control, a Mispositioned Component is any active component found out of the expected position for existing plant conditions when the component's required position is tracked by ____(1)____.

Following revision to an Equipment Alignment Checklists, the ___(2)___ may authorize verification of only those items that were added or changed by the revision.

A. 1. Procedures, Clearances, Work Orders, or TACFs
   2. Shift Manager **ONLY**

B. 1. Procedures and Clearances **ONLY**
   2. Shift Manager **ONLY**

C. 1. Procedures and Clearances **ONLY**
   2.Shift Manager or Senior Reactor Operator

D✓ 1. Procedures, Clearances, Work Orders, or TACFs
   2. Shift Manager or Senior Reactor Operator

Excerpts from NPG-SPP-10.1:

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5.0 DEFINITIONS (continued)

Equipment Alignment Checklists - Checklists used by Operations to verify initial system alignment.

Mispositioned Component - Any Active Positionable Component placed in or left out of the required position for existing plant conditions when the component's required position is tracked by one or more of the following status control tools:

- Procedures
- Clearances
- Work Orders
- Temporary Modification process

Any positionable components placed or left out of the required position for existing plant conditions due to inadequate or incorrect status control tools described above. This includes situations where a lack of process exists that should have controlled the configuration of the component.

All mispositioned components are included in the definition, regardless of cause or significance. However, since this indicator is intended as a human performance indicator, mispositionings that are due to auto or controlling equipment malfunctions are not included.

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3.2.4 Equipment Alignment Checklists (continued)

b. The extent of the revision; for example, minor changes that do not either add new components or change the position of components, do not require re-performance of the checklist.

2. The SM/SRO may authorize verification of only those items that were added or changed by the revision. Items not required to be checked may be N/A'd.

3. Form NPG-SPP-10.1-2, Checklist Log Sheet, can be used to document completion of a checklist. A status control logbook system or computer data base may be used to document checklist completion.

4. The SM/SRO shall verify the revision is properly completed before document transmittal to the Operations Superintendent.

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3.2.6 Work Documents (continued)

5. If available, eSOMs Off-Normal Equipment Alignment tracker functions similar to the LCO tracker portions of the software and provides the user a single place for entering off-normal equipment status. This method may be used in lieu of Attachment 3. The following rules apply to its use:

    a. Make an entry when equipment is aligned in an off-normal condition and is **NOT** restored before the end of the current shift.

    b. Make an entry when equipment alignment is changed under a Work Order and not otherwise flagged or tagged. For example, notching a control rod in one position to clear recurring Control Rod Drift alarms.

    c. Make an entry for conditions which the SM/SRO deems necessary.

    d. Do **NOT** make an entry for equipment alignments that are tagged or flagged, such as clearances or Temporary Modifications, because doing so would be redundant and adds no value.

3.2.7 Component Mispositions

A. Classify configuration control events in accordance with the industry standard metrics, located on the INPO website in INPO 19-002, Industry Reporting and Information System (IRIS) Reporting Requirements guideline (Tier 2).

3.2.8 Configuration Control Committees

A. A Site Configuration Control Committee (CCC) shall be established. The specific functions of the site committee are to:

1. Ensuring high level of focus on Status Control and the Clearance Program.
2. Promoting the understanding and sensitivity of Status Control and Clearances.
3. Evaluating fleet L1-L3 events for causal factors and trends at the site related to Status Control and Clearances.
4. Evaluating site L1-L5 events for trends and establish actions to correct.
5. Monitor the site configuration control performance indicator and take action to address adverse trends.
6. Evaluating significant plant activities, maintenance, and modifications that can affect status control and clearances.
7. Reviewing observations associated with status control and clearance programs.
8. Reviewing and challenging effectiveness of corrective actions from events.
9. Providing recommendations to site management for improving performance.
10. Maintaining a Focus List which contains action items, owners, and due dates.

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3.2.8 Configuration Control Committees (continued)

11. Ensuring industry benchmarking and self assessments are performed and review associated results.

12. The following stations personnel should be included as members; however, membership may be modified based upon specific need.

    a. Operations Superintendent (Chair)* (or designee)

    b. Chemistry Manager* (or designee)

    c. Maintenance Superintendents *

    d. Training Managers*

    e. Work Control Manager *

    f. Radiation Protection Manager* (or designee)

    g. Modifications Manager* (or designee)

    h. Engineering Manager*

    i. Security Manager (or designee)

    j. Primary Modifications Supplemental Contractor (or designee)

    k. PAE level performer

    * A quorum consists of four members, one of which must be the chair. The chairman should request participation from the organizations that are having the most issues with status or clearance control issues. The chairman can set minimum quorum requirements as needed to ensure adequate participation from those organizations.

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.6 **(10CFR 55.43.3 – SRO Only)**	Tier #	----	3
Knowledge of the process for making changes to procedures.	Group #	----	----
	K/A #	G2.2.6	
	Importance Rating	----	3.6

Proposed Question: **# 97**

Unit 3 is in a refueling outage with the A train of ADHR experiencing issues and must be swapped to the B train.

The AUOs note the procedure has all the correct steps, but two are out of order.

Given the conditions above, which **ONE** of the following completes the statement below in accordance with NPG-SPP-01.2, Administration of Site Technical Procedures?

The required procedure revision would be a/an ___**(1)**___ change and the ___**(2)**___ permission is required to complete this change.

A. (1) urgent
   (2) Shift Manager

B. (1) urgent
   (2) Operations Superintendent

C. (1) minor/editorial
   (2) Shift Manager

D. (1) minor/editorial
   (2) Operations Superintendent

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: In accordance with NPG-SPP-01.2 Administration of Site Technical Procedures, typographical errors do NOT change the intent of a procedure. They are therefore minor/editorial in nature and are specially listed in NPG-SPP-01.2. However, procedure steps out of order is NOT one of them especially given the conditions. Urgent changes are temporarily approved changes to procedures that are deemed necessary to maintain plant safety where inadequate time exist to make a normal revision. For second part, urgent procedure changes do require Shift Manager permission prior to being performed. Additionally, the permission is documented through signature for the urgent procedure change.

B INCORRECT**:** First part is correct (*See A*). Second part is incorrect but plausible in that the Operations Superintendent is responsible for verifying systems and components requiring status control are aligned per procedures and relaxing of status control in accordance with NPG-SPP-10.1.

C INCORRECT: First part is incorrect but plausible in that typographical errors do NOT change the intent of a procedure. They are therefore minor/editorial in nature and are specially listed in NPG-SPP-01.2. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

SRO Level Justification: This is a generic example testing the candidate's knowledge of procedure changes and the process to perform different procedure revisions based on the complexity of that revision. SROs are tested to the highest position of their license and a knowledge of the process is required since Shift managers are required to sign and authorize it's execution as an urgent procedure change. SRO only because of link to 10CFR55.43 (3): Facility licensee procedures required to obtain authority for design and operating changes to that facility. SROs are a part of the evaluation process with a few former ROs performing the revision only, not the evaluation. This is evaluated as Memory or fundamental knowledge due to the question requiring retention of memory knowledge to discern what level of change is being performed and what the evaluation process requires in accordance with NPG-SPPs.

Technical Reference(s): NPG-SPP-.01.2, Rev. 23 (Attach if not previously provided)
NPG-SPP-.10.1, Rev. 13

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.087, Obj. 1, 4 (As available)

Question Source:

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

Question History:

Last NRC Exam	

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Excerpts from NPG-SPP-10.1
:

NPG Standard Programs and Processes	System Status Control	NPG-SPP-10.1 Rev. 0013 Page 5 of 29

1.0 PURPOSE

This procedure establishes the responsibilities and programmatic methods for obtaining, maintaining and documenting control of equipment and system status in accordance with design, license and regulatory requirements.

2.0 SCOPE

A. This procedure applies to all TVA Nuclear Power Group (NPG) personnel and contractors performing activities affecting:

1. Nuclear safety related and quality related systems and equipment.

2. Non-safety related systems and equipment necessary to support the production of electricity.

3. Fire Protection systems and equipment.

B. **Review Cadence:** In accordance with NPG-SPP-01.1, this procedure is required to be reviewed at least once every four years (+3 months grace period), with the review documented in the Validation Date and Validated By fields on the SPP cover sheet.

3.0 PROCESS

3.1 **Roles and Responsibilities**

3.1.1 **Responsible Managers**

Ensure status control for equipment within their areas of responsibility.

Ensure procedures and processes for equipment within their area of responsibility meets the status control requirements of this procedure.

3.1.2 **Operations Superintendent**

Verify systems and components requiring status control are aligned per this procedure.

Authorize relaxation of status control.

3.1.3 **Shift Manager (SM) or Senior Reactor Operator (SRO)**

Determines when re-performance of all or part of an Equipment Alignment Checklist is needed for procedure revisions or minor outages.

Ensures status control is maintained.

Authorizes relaxation of status control within a clearance boundary when necessary.

Ensures procedures restore system and equipment to the correct status.

Ensures all activities that change the status of plant equipment are authorized by an approved plant procedure or work document.

Excerpts from NPG-SPP-01.2:

3.2.7 Contractor Procedure Requirements (continued)

D. Contractor procedures which are required to support operation of the plant and do not meet the criteria above may be developed and approved under contractor's Tennessee Valley Authority Nuclear approved Quality Assurance Plan. Examples of these are: the Architect Engineer design procedures, Power Service Shop procedures, Central Laboratory.

3.2.8 Minor/Editorial Changes

A. Minor changes, are inconsequential corrections, that do not affect the intent or performance of the procedure instructions. Minor changes do not require an IQR, AOR, Licensing Compliance Review, or PORC review.

B. Procedure changes that meet any of the following criteria are considered minor changes:

1. Correction of punctuation, style changes
2. Redundant or insignificant word or title changes
3. Correction of typographical errors including capitalization
4. Adding a notation that an existing step is now a critical step
5. Correction of reference errors due to title or number change
6. Omitted symbols that do not alter results
7. Incorrect units of measure due to editorial error
8. Misplaced decimals that are neither setpoint values nor tolerances
9. Page number discrepancies
10. Separating a single section into multiple sections
11. Modifications of sign-offs, signatures, or date lines that already exist
12. Corrections to attachment identifiers
13. Corrections to titles of plant organizations, position titles, department/section/unit names when there is no change in authority, responsibility, or reporting relationships
14. Corrections to addresses, telephone numbers, or computer application names
15. Corrections to equipment nomenclature or locations in procedures that already exist, and will not modify procedure scope or intent
16. Changes to equipment unique identifier information (UNID) in procedures to be consistent with design output document changes, and does not alter what component is operated

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3.2.8 **Minor/Editorial Changes (continued)**

17. Corrections to or clarification of a note or precaution which does not alter the method of accomplishing a task

18. Changes which are administrative and non-technical in nature and do not change the intent or outcome of an activity (such as adding a step requiring a log entry, a plant announcement, informational notifications, or initiation of a CR)

C. The revision description will describe the reason for change.

D. CFAMs/PO can eliminate site reviews for minor changes.

E. Minor changes are processed as a revision, except for the reduced reviews discussed in this section.

F. The procedure package is transmitted to Management Services for distribution and ECM archival.

3.2.9 **Urgent Procedure Changes**

Urgent changes are temporarily approved changes to procedures that are deemed necessary by plant management to maintain plant safety, operability or critical schedules and inadequate time exists to make a normal revision using ECM. This should not be viewed as a convenience to circumvent the normal electronic ECM process.

Urgent changes require the following:

A. Procedure User notifies supervision or Plant Management a urgent procedure change is required.

B. Responsible department personnel initiate the urgent procedure change and performs the following changes to a handwritten, or electronic copy and documents as follows:

1. Mark the changes in a clearly legible manner with black ink and initial/date the change.

2. Ensure a revision bar is marked for each change in the right hand margin, and include the Urgent Change Number (UC1 or UC2, for example), for each change.

3. List the Urgent Change Number in the Revision column of the Revision Log, with the description of the change.

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3.2.9 Urgent Procedure Changes (continued)

C. Obtain applicable reviews by signature on the Urgent Change Request Form, as follows:

1. Perform an Affected Organization/Cross Discipline Review when other organizations are directly affected by the urgent procedure change.

2. Review to determine if the Urgent Change is within the scope of 10 CFR 50.59/72.48 using Attachment 4 of NPG-SPP-03.14, Licensing Compliance Review, Form TVA 41549. Document the results of this determination and any screening/evaluations on Form TVA 41783 (NPG-SPP-01.2-7, Urgent Procedure Change Request Form), as required in Attachment 11. Reference the following for guidance:

   a. NPG-SPP-09.4, 10 CFR 50.59 Evaluations of Changes, Tests, and Experiments

   b. NPG-SPP-09.9, 10 CFR 72.48 Evaluations of Changes, Tests, and Experiments for Independent Spent Fuel Storage Installation.

3. An IQR for the urgent procedure change is not required.

4. If a PORC review is required, referencing NPG-SPP-10.5 Plant Operations Review Committee, performance of the PORC review is documented on Form TVA 41783 (NPG-SPP-01.2-7, Urgent Change Request Form), as required in Attachment 11.

5. If the Urgent Change meets the description of a Minor/Editorial Change, review requirements are reduced. Refer to Section 3.2.8 Minor/Editorial Change for requirements.

D. Obtain approvals of two members of Plant Management staff that have knowledge of the areas affected by the procedure change, one of which is the Operations Shift Manager. Document approvals by signature on Form TVA 41783, NPG SPP-01.2-7, Urgent Change Request Form.

E. The procedure can now be used to perform work.

F. The department responsible for the urgent procedure change documents the need for the urgent change in a CR and submits the urgent procedure change to Management Services by next working day. Ensure to include the following completed documents when submitting the paperwork to Management Services:

1. TVA Form 41549, Licensing Compliance Review for Technical Procedures

2. TVA Form 41783, Urgent Change Request Form

3. 10 CFR 72.48 Evaluation paperwork, if 72.48 review was marked "Yes" on TVA Form 41783.

4. 10 CFR 50.59 Evaluation paperwork, if 50.59 review was marked "Yes" on TVA Form 41783.

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Attachment 11
(Page 1 of 1)

Urgent Change Request Form

Urgent Change Request Form
GENERAL INFORMATION

Procedure No. Rev Title Tracking #

Requested By Organization

☐ BFN ☐ COC ☐ SQN ☐ WBN

☐ Yes ☐ No Minor/Editorial Change

Brief Description of Change

Check as applicable:
PORC Review ☐ Yes ☐ No Quality Related ☐ Yes ☐ No

10 CFR REVIEWS

10 CFR 50.59 Review Required ☐ Yes ☐ No
If yes, perform 10 CFR 50.59 review in accordance with NPG-SPP-09.4.

10 CFR 50.59 Reviewer

10 CFR 72.48 Review Required ☐ Yes ☐ No
If yes, obtain 10 CFR 72.48 Review

10 CFR 72.48 Reviewer

REVIEWS

PORC Chairman Signature (if required) Date

APPROVALS

Approval Authority/Site Sponsor Signature Date

Plant Manager Signature (if PORC review required) Date

Ops Shift Manager Signature Date

Examination Outline Cross-reference:

G2.3.11 (10CFR 55.43.4 – SRO Only)

Ability to control radiation releases.

	RO	SRO
Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.3.11	
Importance Rating	------	4.3

Proposed Question: **# 98**

0-EOI-4, Radioactivity Release Control, step RR-1 states:

'**IF** Turbine Building Ventilation is shutdown **THEN** RESTART Turbine Building Ventilation Fans.'

Which **ONE** of the following completes the statements below in accordance with the EOI Program Manual Bases?

Operation of Turbine Building Ventilation preserves ___**(1)**___.

Radioactivity in Turbine Building areas will be discharged through an elevated, ___**(2)**___ release point.

A. (1) equipment operability
(2) monitored

B. (1) equipment operability
(2) unmonitored

C. (1) building accessibility
(2) monitored

D. (1) building accessibility
(2) unmonitored

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that in accordance with EOI-3, Secondary Containment Control, equipment operability is a concern in the Reactor Building. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that the Turbine Building is NOT an air-tight structure and a radioactive release inside the Turbine Building would NOT only limit personnel access, but would eventually lead to an **unmonitored** ground level release.

C **CORRECT**: *(See attached)* In accordance with EOIPM Section 0-V-F, Radioactivity Release Control Bases, Operation of the Turbine Building Ventilation System preserves Turbine Building accessibility. For second part, radioactivity in Turbine Building areas is discharged through an elevated, **monitored** release point. Continued personnel access to the Turbine Building may be essential for responding to emergencies or transients which may degrade into emergencies.

D INCORRECT**:** First part is correct *(See C).* Second part is incorrect but plausible (*See B*).

SRO Level Justification: Tests the candidate's knowledge concerning control of radiation releases related to why it is desirable to restore Turbine Building ventilation during the execution of EOI-4. Requires knowledge of the bases for procedural requirements associated with decision making duties unique to the SRO position. SRO only because of link to 10CFR55.43 (4): Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. This question is rated as Memory due the strict recall of specific select procedural requirements related to specific EOI Program Manual Bases.

Technical Reference(s): EOIPM 0-V-D, Rev. 2 (Attach if not previously provided)
EOIPM 0-V-F, Rev. 4
OPL171.204, Rev. 10

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203, Obj. 9 (As available)

Question Source: Bank # ILT EXAM BANK OPL171.204-11 006 #2807

Modified Bank #		(Note changes or attach parent)
New		
Last NRC Exam		

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Copy of Bank Question:

2807. OPL171.204-11 006

0-EOI-4, Radioactivity Release Control, step RR-1 states:

'**IF** Turbine Building ventilation is shutdown **THEN** RESTART Turbine Building ventilation fans.'

Which one of the following completes the statement below in accordance with EOIPM Section 0-V-F, Radioactivity Release Control Bases?

Operation of Turbine Building ventilation preserves ___**(1)**___ and assures radioactivity in Turbine Building areas is discharged through an elevated, ___**(2)**___ release point.

A. **(1)** equipment operability
(2) unmonitored

B. **(1)** equipment operability
(2) monitored

C. **(1)** building accessibility
(2) unmonitored

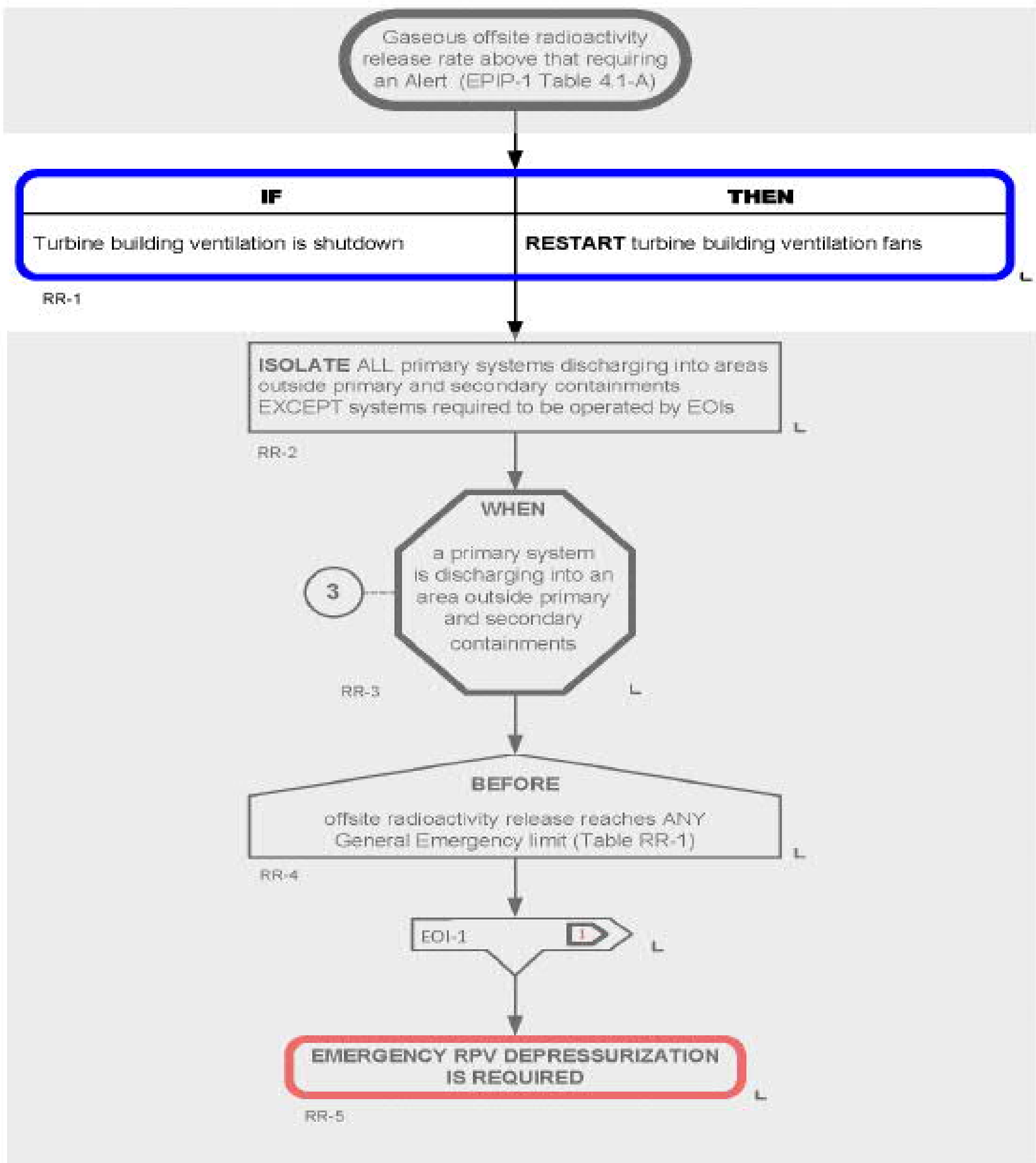
D.✓ **(1)** building accessibility
(2) monitored

Excerpts from EOIPM 0-V-F:

BFN Unit 0	EOI-4, Radioactivity Release Control Bases	EOIPM Section 0-V-F Rev. 0003 Page 8 of 17

1.0 EOI-4, RADIOACTIVITY RELEASE CONTROL BASES (continued)

RR-1



BFN Unit 0	EOI-4, Radioactivity Release Control Bases	EOIPM Section 0-V-F Rev. 0004 Page 9 of 17

1.0 EOI-4, RADIOACTIVITY RELEASE CONTROL BASES (continued)

DISCUSSION: RR-1

This retainment override step applies to all steps of this flowchart.

Operation of the Turbine Building Ventilation System preserves turbine building accessibility, and assures that radioactivity in turbine building areas is discharged through an elevated, monitored release point. Continued personnel access to the turbine building may be essential for responding to emergencies or transients which may degrade into emergencies. Since the turbine building is not an air-tight structure, a radioactive release inside the turbine building would not only limit personnel access, but would eventually lead to an unmonitored ground level release.

Excerpt from OPL171.204 Lesson Plan:

OPL171.204 , EOI-3 Secondary Containment Control and EOI-4 Radioactivity Release Control, Rev# 10

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods

II. General Introduction

A. Introduce self and/or guest(s)

B. Take Attendance

C. Handout trainee feedback forms

D. Introduce Topic / Goal

1. Upon completion of this lesson, the student will be able to implement the requirements of EOI-3 to control Secondary Containment Parameters and EOI-4 to control Radioactive Releases.

E. Learning Objectives

F. Description of how class will be conducted

G. Evaluation Method

1. During this presentation, mastery of the subject matter will be evaluated through oral questioning and/or exercises.

2. After this presentation, written exams testing this subject-matter will be completed on a weekly, bi-weekly, cyclic, and/or biennial basis (relative to the specific program).

3. System-specific knowledge will also be evaluated in the Simulator, as applicable.

H. What's In It For Me?

III. Presentation

A. Introduction

1. This lesson is designed to provide a detailed classroom discussion of EOI-3, Secondary Control, and EOI-4, Radioactivity Release Control.

a) Each step of the flow chart, including associated tables, cautions, and notes, will be discussed.

b) The basis for each step will also be discussed as it applies to anticipated plant conditions.

2. The purpose of EOI-3 is:

a) Protect equipment in secondary containment,

b) Limit radioactivity release to secondary containment, and either:

(1) Maintain secondary containment integrity, or

(2) Limit radioactive release from secondary containment.

Examination Outline Cross-reference:

G2.4.40 (10CFR 55.43.5 – SRO Only)

Knowledge of SRO responsibilities in emergency plan implementation.

	RO	SRO
Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.4.40	
Importance Rating	------	4.5

Proposed Question: **# 99**

The Shift Manager / Site Emergency Director (SM/SED) has declared a General Emergency.

The Central Emergency Control Center (CECC) is **NOT** staffed.

Besides classification, which **ONE** of the following duties can **NOT** be delegated to another emergency team member by the SM/SED?

A. Make Notifications to the State.

B. Direct the Shutdown of the Plant.

C. Conduct Site Accountability Actions.

D. Determine Protective Action Recommendations.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that while the SM/SED cannot delegate the Protective Action Recommendations (PARs) on Appendix A of EPIP-5, in accordance with OPDP-1, Conduct of Operations he can delegate some other duties, including having an SRO fax the form to the State of Alabama.

B INCORRECT: Incorrect but plausible in that in accordance with OPDP-1, the Shift Manager is responsible for safe plant operation; however, in the absence of the Shift Manager, this becomes the responsibility of the Nuclear Unit Senior Operator (NUSO) in the Control Room. Additionally, it is the responsibiity of each licensed operator to place the plant in a safe condition, and are allowed to shut down the Reactor (SCRAM) if they deem it necessary.

C INCORRECT: Incorrect but plausible in that in accordance with EPIP-8, Personnel Evacuation and Accountability, the SM/SED shall make the decision to activate Assembly and Accountability process and cannot delegate that decision. However, the actions carried out as a result of this decision can be delegated.

D **CORRECT**: In accordance with OPDP-1, Conduct of Operations, the Shift Manager functions as the Site Emergency Director until relieved, and cannot delegate classification of an emergency, protective action recommendations, or authorization of emergency exposure. All other duties may be delegated to another qualified SRO as allowed by site specific procedures. Per EPIP-5, "General Emergency," the SM/SED cannot delegate protective action reommendations.

SRO Level Justification: Requires knowledge of the Shift Manager's, whom is a licensed SRO, responsibilities in Emergency Plan implementation. The duties that can be delegated are performed by a licensed SRO. The question requires knowledge of duties that are unique to the SM/SRO Position and is therefore an SRO Question. 10 CFR 55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This is a memory or fundamental knowledge question since it requires memory of specific duties during an emergency.

Technical Reference(s): OPDP-1, Rev. 46 (Attach if not previously provided)
EPIP-1, Rev. 45
EPIP-5, Rev. 55
EPIP-8, Rev. 32

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.075, Obj. 4 (As available)

Question Source:	Bank #	BFN 1404 #99	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2014	

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Copy of Bank Question:

QUESTION 99

The Shift Manager / Site Emergency Director (SM/SED) has declared a General Emergency. The Central Emergency Control Center (CECC) is NOT staffed.

Besides classification, which ONE of the following duties can NOT be delegated to another emergency team member by the SM/SED?

A. Making notifications to the state

B. Directing the shutdown of the plant

C. Conducting site accountability actions

D. Determining Protective Action Recommendations

Correct Answer: **D**

Excerpt from OPDP-1:

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0046 Page 11 of 71

3.1.7 **Shift Manager (SM)**

A. As the senior management representative on shift, the SM is in direct charge of plant operations and is responsible through the Operations Shift Superintendent and Operations Director to the Plant Manager, for safe and reliable operation of the nuclear plant.

B. The SM is responsible for on shift management and oversight in the control room and all plant group activities.

C. The SM is responsible for the oversight function. In his absence from the control room, the oversight function must be turned over to the NUSO for each unit, as applicable.

D. The SM is responsible for the control room command function. In his absence from the control room, the control and command function automatically reverts to the NUSO for each unit.

E. The SM has the authority to take action necessary to ensure compliance with TS, operating license requirements, and approved plant procedures to protect the health and safety of employees and the public, to ensure adequate security, and to protect the plant from damage.

F. The SM shall hold an active SRO license.

G. The SM is responsible for overall reactor operations and maintains the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times.

H. Ensures field oversight conducted for work activities that are being performed during the shift, as a priority for operational focus.

I. The SM should not become involved in any single operation that distracts him when multiple operations are required in the control room, such as during plant transients or an emergency.

J. The SM is responsible for ensuring a professional atmosphere is maintained in the control room at all times.

K. During emergency situations the SM shall remain in the affected control room unless properly relieved.

1. The SM functions as Site Emergency Director and implements Emergency Plan procedures, until relieved in accordance with TVA Nuclear Emergency Plan. He or she cannot delegate:

    a. Classification of an emergency

    b. Protective Action Recommendations

    c. Authorization of Emergency Exposure

    d. All other duties may be delegated to another qualified SRO, as allowed by site-specific procedures.

Excerpt from EPIP-1:

BFN Unit 0	Emergency Classification Procedure	EPIP-1 Revision 0059 Page 7 of 137

1.0 PURPOSE

This Procedure provides guidance in determining the classification and declaration of an emergency based on plant conditions.

2.0 RESPONSIBILITY

The responsibility of declaring an Emergency based on the guidance within this procedure belongs to the Shift Manager/Site Emergency Director (SM/SED) or designated Unit Supervisor (US) when acting as the SM or the TSC Site Emergency Director (SED).

The following duties CANNOT be delegated: Emergency Classification, Emergency Dose Approval and PAR development prior to CECC Director ownership for PAR development.

3.0 INSTRUCTIONS

3.1 Precautions/Limitations

A. The criteria in EPIP-1 are given for guidance only: knowledge of actual plant conditions or the extent of the emergency may require that additional steps be taken. In all cases, this logic procedure should be combined with the sound judgment of the SM/SED and/or the TSC SED to arrive at a classification for a particular set of circumstances.

B. The Nuclear Power (NP) Radiological Emergency Plan (REP) will be activated when any one of the conditions listed in this logic is detected and declared.

C. The SM/SED shall assess, classify, and declare an emergency condition within 15 minutes after information is first available to plant operators to recognize that an EAL has been exceeded and to make the declaration promptly upon identification of the appropriate Emergency Classification Level (ECL).

Excerpts from EPIP-5:

BFN Unit 0	GENERAL EMERGENCY	EPIP-5 Rev. 0055 Page 7 of 38

3.0 EMERGENCY CLASSIFICATION ACTIONS

NOTES

- Procedure steps can be performed concurrently.
- All procedure steps must be completed.
- All procedure appendices must be returned to the SED.
- Section 3.1 (as soon as possible, within 15 Minutes from the classification declaration) and Section 3.4 (as soon as possible, not to exceed 60 minutes from classification declaration) are time critical.
- A Shift Technical Advisor or Senior Reactor Operator (SRO) should peer review Appendix A completion.

CAUTION

- Ongoing or anticipated security events or severe weather may present a danger to normal staffing and other Emergency Plan implementation processes. Observe all procedural steps carefully during severe weather and security related events.
- Appendix A, Step 7 and Appendix G, Steps 1.1[1] and 1.1[5] CANNOT be delegated.

[1] **WHEN** the Technical Support Center (TSC) SED has assumed the responsibilities from the SM/SED, **THEN**

CONTINUE in this procedure at Appendix G.

Otherwise continue in this procedure.

BFN Unit 0	GENERAL EMERGENCY	EPIP-5 Rev. 0055 Page 14 of 38

Appendix A
(Page 2 of 2)

General Emergency Initial Notification Form

1. ☐ This is a Drill ☐ This is an Actual Event - Repeat - This is an Actual Event

2. The Site Emergency Director at **BROWNS FERRY** has declared a **GENERAL EMERGENCY**

3. **Initiating Condition (IC) Designator:**______________________(Use only one IC)

4. **Radiological Conditions:** (Check one under both Airborne and Liquid column.)

Airborne Releases Offsite	Liquid Releases Offsite
☐ Minor releases within federally approved limits[1]	☐ Minor releases within federally approved limits[1]
☐ Releases above federally approved limits[1]	☐ Releases above federally approved limits[1]
☐ Release information not known	☐ Release information not known

[1]-Technical Specifications/Offsite Dose Calculation Manual

5. **Event Declared:** Time:__________________ Date:____________________

6. **The Meteorological Conditions are:** (Use 91 meter data from the Met Tower. If data is not available from the MET tower, contact the National Weather Service by dialing 1-256-890-8507 or 1-205-621-5650. The National Weather Service will provide wind direction and wind speed.)

Wind Direction is FROM:____________ degrees (15 minute average) Wind Speed: ____________ mph (15 minute average)

7. Provide Protective Action Recommendation utilizing Appendix H: (Check all Sectors as appropriate)

BFN Unit 0	GENERAL EMERGENCY	EPIP-5 Rev. 0055 Page 27 of 38

Appendix G
(Page 1 of 5)

Technical Support Center General Emergency Classification Instruction

NOTES

- Procedure steps can be performed concurrently.
- All procedure steps must be completed and remain under the direct oversight of the Site Emergency Director (SED).
- Section 1.0 (as soon as possible, within 15 Minutes) and Section 5.0 (as soon as possible, not to exceed 60 minutes from classification declaration) are time critical.

CAUTION

- Ongoing or anticipated security events or severe weather may present a danger to normal staffing and other Emergency Plan implementation processes. Observe all procedural steps carefully during security related events.
- Procedure Steps 1.1[1] and 1.1[5] of this appendix CANNOT be delegated.

1.0 NOTIFICATION OF BFN RISK COUNTIES AND STATE OF ALABAMA

NOTE

Notification of the Risk Counties/State of Alabama is required to be completed as soon as possible, not to exceed 15 minutes from the time of emergency classification declaration.

1.1 CECC Notification

[1] RECORD the following information:

- General Emergency Classification IC Designator: ____________
- General Emergency Classification declared at time: ____________
- Site Emergency Director: (Name) ________________________

[2] CONTACT the Central Emergency Control Center (CECC) Director utilizing the CECC "Direct Ring-Down" telephone or at extension 1-423-751-1614.

[3] COMMUNICATE the information recorded in step 1.1[1].

BFN Unit 0	GENERAL EMERGENCY	EPIP-5 Rev. 0055 Page 28 of 38

Appendix G
(Page 2 of 5)

Technical Support Center General Emergency Classification Instruction

1.1 CECC Notification (continued)

[4] IF the CECC Director was contacted, consider the State of Alabama notification complete, **THEN**

CONTINUE in this appendix at Section 2.0.

[5] IF the CECC Director was not contacted, **THEN**

COMPLETE Appendix A, "General Emergency Initial Notification Form," and **DIRECT** a member of the Technical Support Center (TSC) Staff (Ops Specialist/Ops Manager/EP Manager) to complete Appendix B, "State of Alabama and Operations Duty Specialist (ODS) Notification."

2.0 NOTIFICATION OF SITE PERSONNEL

> **CAUTION**
>
> Ongoing or anticipated security events may present a danger to site personnel. Do not conduct the notification of site personnel Public Address (PA) message during an ongoing or anticipated security event. All pertinent site personnel PA messages will be conducted per 0-AOI-100-8, "Security Event Response," for security events.

[1] DIRECT a member of the TSC to CONDUCT a Plant PA announcement similar to the following: (Dial 7-687 to obtain the Plant PA)

"Attention All Personnel. Attention All Personnel.

A General Emergency Classification has been declared.

The Browns Ferry Emergency Plan is being implemented at this time.

Further updates will follow."

(Repeat Message)

Excerpt from OPDP-1:

3.1.1 Site Vice President (continued)

A. The Site Vice President shall discuss with all operators potential external influences that could adversely affect operator decision-making and also reinforce the expectation to perform similarly in the plant as in training.

1. The key message to operators and operating crews is to maintain their line of sight to the core and to focus on core parameters during off-normal plant conditions, just as they experience in the simulator.

2. The discussion should discuss possible influences on operator decision-making and reinforce that operators have executive and management support to place the plant in a safe, reduced power or shutdown condition when conditions warrant.

3. The discussion is targeted for an operator training segment kickoff or during Operations Director blocked time, but can be conducted at any time.

4. This discussion should be conducted, at the SVP discretion, commensurate with operations department performance or identification of influences that could affect decision making (for example, financial factors, extended shutdowns, aggregate equipment challenges, cultural warning signs, etc). The discussion frequency will not exceed one refueling cycle."

3.1.2 All Operations Personnel

A. Adhere to all TVA procedures, processes, and standards.

B. Responsible to ensure individual qualification for all required watch stations they stand.

C. The SROs in an oversight position (SM and NUSO) shall not manipulate plant equipment. In some cases, shortages in training and manning require SROs to perform a limited number of manipulations. These will be approved by the Operations Director until required training and manning are established. Each site will, as necessary, document this issue in a CR to drive actions to meet this standard.

D. Immediate operator actions required to place the plant in a stable condition during a transient shall be performed from memory. "Immediate operator actions" are designated by procedures. All operators are required to know their immediate actions and perform them from memory upon recognizing initial conditions.

E. Whenever an activity or evolution is interrupted, ensure affected equipment is placed in a stable condition as soon as practicable.

F. Licensed Operators will recognize time critical decisions based on degrading conditions that threaten operating margin and respond as trained.

G. During normal (non-transient) operation, plant announcements shall be made before changing the status of any major equipment such as starting or stopping pumps. In addition, a field operator shall be dispatched to monitor associated major equipment startup and shutdown (non-transient), notifying the control room of any abnormalities.

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0046 Page 11 of 71

3.1.7 **Shift Manager (SM)**

A. As the senior management representative on shift, the SM is in direct charge of plant operations and is responsible through the Operations Shift Superintendent and Operations Director to the Plant Manager, for safe and reliable operation of the nuclear plant.

B. The SM is responsible for on shift management and oversight in the control room and all plant group activities.

C. The SM is responsible for the oversight function. In his absence from the control room, the oversight function must be turned over to the NUSO for each unit, as applicable.

D. The SM is responsible for the control room command function. In his absence from the control room, the control and command function automatically reverts to the NUSO for each unit.

E. The SM has the authority to take action necessary to ensure compliance with TS, operating license requirements, and approved plant procedures to protect the health and safety of employees and the public, to ensure adequate security, and to protect the plant from damage.

F. The SM shall hold an active SRO license.

G. The SM is responsible for overall reactor operations and maintains the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times.

H. Ensures field oversight conducted for work activities that are being performed during the shift, as a priority for operational focus.

I. The SM should not become involved in any single operation that distracts him when multiple operations are required in the control room, such as during plant transients or an emergency.

J. The SM is responsible for ensuring a professional atmosphere is maintained in the control room at all times.

K. During emergency situations the SM shall remain in the affected control room unless properly relieved.

1. The SM functions as Site Emergency Director and implements Emergency Plan procedures, until relieved in accordance with TVA Nuclear Emergency Plan. He or she cannot delegate:

    a. Classification of an emergency

    b. Protective Action Recommendations

    c. Authorization of Emergency Exposure

    d. All other duties may be delegated to another qualified SRO, as allowed by site-specific procedures.

Excerpt from OPDP-8:

BFN Unit 0	PERSONNEL ACCOUNTABILITY AND EVACUATION	EPIP-8 Rev. 0032 Page 6 of 33

3.2 Particular Plant Area Evacuation (continued)

C. Personnel in the evacuated area(s), upon hearing the public address announcement or being notified of the particular plant area evacuation, shall:

1. If working in a contaminated area, exit the area in accordance with Radiation Protection (RP) procedures unless instructed otherwise by RP.

2. Exit the area in an orderly manner.

D. Personnel not in the evacuated area(s) should continue assigned tasks unless instructed otherwise and should not enter the evacuated area(s) until the "All Clear" has been announced or entry into the evacuated area(s) has been approved through emergency response processes.

3.3 Activation of the Assembly and Accountability Process

A. SM/SED Responsibilities

The SM or SED shall make the decision to activate the Assembly and Accountability process. The actions carried out as a result of this decision can be delegated but the decision itself cannot be delegated.

B. Nuclear Security Responsibilities

When notified that conditions have been met that require activation of the Assembly and Accountability process or upon indications that Assembly and Accountability has been initiated, Nuclear Security will implement Appendix C of this procedure.

C. Radiation Protection (RP) Responsibilities

When notified that conditions have been met that require the activation of the Assembly and Accountability process or upon indications that Assembly and Accountability has been initiated, RP will implement Appendix E of this procedure.

D. All Employees/Supervisors Responsibilities

Upon initiation of the Assembly and Accountability process, all employees/supervisors should identify any co-workers who may be working in high noise remote locations and ensure that they are alerted that the Assembly and Accountability process has been initiated. In particular, consideration should be given to employees working in areas where hearing of the siren is affected by operating equipment and/or extensive hearing protection being worn by co-workers.

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.43 **(10CFR 55.43.5 – SRO Only)**	Tier #	------	3
Knowledge of emergency communications systems and techniques.	Group #	------	------
	K/A #	G2.4.43	
	Importance Rating	------	3.8

Proposed Question: **# 100**

Given that a NOUE has been declared by the Shift Manager, which **ONE** of the following completes the statements below in accordance with EPIPs?

The Technical Support Center (TSC) ___**(1)**___ required to be staffed.

Assembly and Accountability, ___**(2)**___ required to be performed.

A. (1) is
(2) is

B. (1) is
(2) is NOT

C. (1) is NOT
(2) is

D. (1) is NOT
(2) is NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that in accordance with the EPIPs, the lowest classifcation in which the TSC is required to be staffed is an Alert (EPIP-3). The second part is incorrect but plausible in that EPIP-2, Notification of Unusual Event Appendix C, Assembly and Accountability states that assembly/accountability can be performed at the Site Emergency Director's discretion, but it is not required. The lowest classification in which assembly/accountability is required is a Site Area Emergency (EPIP-4).

B INCORRECT: Part 1 is incorrect but plausible (*See A*). The second part is correct (*See D*).

C INCORRECT**:** First part is correct (*See D*). Second part is Incorrect but plausible (*See A*).

D **CORRECT**: The TSC is not required to be staffed in accordance with EPIP-2. The lowest classifcation in which the TSC is required to be staffed is an Alert (EPIP-3). For second part, in accordance with EPIP-2 assembly and accountability is not required, but may be performed at the Site Emergency Director's discretion. The lowest classification in which assembly/accountability is required is a Site Area Emergency (EPIP-4).

SRO Level Justification: Test the candidate's knowledge of communication techniques during a REP event in regards to staffing the TSC and directing onsite personnel assembly and accountability. SRO only because of the link to 10CFR55.43 (5): Assessment of required actions in appropriate procedures during normal, abnormal, and emergency situations. This question is rated as memory or fundamental knowledge based on it requires the candidate to recall which appendix is required for the given conditions.

Technical Reference(s): EPIP-2, Rev. 39 (Attach if not previously provided)
EPIP-3, Rev. 42
EPIP-4, Rev. 41

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.075 Obj. 13 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Excerpts from EPIP-2:

BFN Unit 0	NOTIFICATION OF UNUSUAL EVENT	EPIP-2 Rev. 0039 Page 9 of 25

3.6 **Notification of Site Personnel**

NOTE
The Emergency Response Organization (ERO) is not required to be activated for a Notification of Unusual Event, but SED judgment may require activation.

[1] IF SED judgment requires activation of the ERO, **THEN**

DIRECT completion of Appendix H, "Activation of ERO using TEENS."

[2] **IMPLEMENT** Appendix D, "Notification of Unusual Event Site Notifications."

BFN Unit 0	NOTIFICATION OF UNUSUAL EVENT	EPIP-2 Rev. 0039 Page 20 of 25

Appendix E
(Page 2 of 2)

Monitor/Re-evaluate the Event

2.0 ASSEMBLY/ACCOUNTABILITY

[1] **IF** emergency circumstances warrant Assembly/Accountability, **THEN**

CONTINUE in this procedure.

Otherwise re-enter this appendix at Step 3.0

[2] **IF** any of the following conditions exists:

A. A severe weather condition, such as a tornado, is currently in progress or is projected on-site,

OR

B. An on-site security risk condition exists that may present a danger to site personnel during the Assembly/Accountability process as determined by Site Emergency Director/Nuclear Security,

OR

C. Rapid Evacuation of the Protected Area has been conducted,

THEN

DO NOT initiate the Assembly/Accountability Process and **CONTINUE** in this appendix at Step 3.0.

[3] **CONTACT** Nuclear Security at 729-3238 or 729-2219 to initiate Assembly/Accountability, utilizing EPIP-8, Appendix C, "Nuclear Security - Assembly and Accountability Actions."

Excerpt from EPIP-3:

BFN Unit 0	ALERT	EPIP-3 Rev. 0042 Page 6 of 29

3.0 EMERGENCY CLASSIFICATION ACTIONS

NOTES

- Procedure steps can be performed concurrently.
- All procedure steps must be completed.
- All procedure appendices must be returned to the SED.
- Section 3.1 (as soon as possible, within 15 Minutes from classification declaration) and Section 3.4 (as soon as possible, not to exceed 60 minutes from classification declaration) are time critical.
- A Shift Technical Advisor or Senior Reactor Operator (SRO) should peer review Appendix A completion.

CAUTION

- Ongoing or anticipated security events or severe weather may present a danger to normal staffing and other Emergency Plan implementation processes. Observe all procedural steps carefully during severe weather and security related events.
- Step 3.1[2] of the Main Body and Appendix G, Steps 1.1[1] and 1.1[4] CANNOT be delegated.

[1] **WHEN** the TSC SED has assumed the responsibilities from the SM SED, **THEN**

CONTINUE in this procedure at Appendix G.

Otherwise continue in this procedure.

Excerpt from EPIP-4:

BFN Unit 0	SITE AREA EMERGENCY	EPIP-4 Rev. 0041 Page 8 of 29

3.3 **Evacuation of Non-Emergency Responders**

NOTE
In the event of an unplanned significant release of radioactivity or sudden increase in radiation levels, it is the responsibility of the SED to make the decision concerning the necessity for building or area evacuation. In arriving at this decision, the primary consideration is personnel safety. The assembly/accountability alarm is used to initiate the assembly of all site personnel. When specific areas are to be evacuated, only use the public address system.

[1] **IF** any of the following conditions exists:

A. A severe weather condition, such as a tornado, is currently in progress or is projected on-site,

OR

B. An on-site security risk condition exists that may present a danger to site personnel during the Assembly/Accountability process as determined by SED/Nuclear Security,

OR

C. Rapid Evacuation of the Protected Area has been conducted,

THEN

DO NOT initiate the Assembly/Accountability Process and **CONTINUE** in this procedure at Section 3.4.

[2] **DIRECT** Nuclear Security at 729-3238 or 729-2219 to commence Assembly/Accountability by utilizing EPIP-8, Appendix C, "Nuclear Security Assembly and Accountability Actions."

Facility: BFN Scenario Number: NRC- 1 Op-Test Number: 21-04

Examiners: ____________________ Operators: SRO: ____________________

____________________ ATC: ____________________

____________________ BOP: ____________________

Initial Conditions: 80% Reactor Power.

Turnover: Reactor Shutdown in progress. SRV 1-22 is INOPERABLE (ADS Valve). EHPM tagged for motor bearing inspection.

Critical Tasks:

1. With the Reactor at power and with a Primary System discharging into the Secondary Containment, manually SCRAM the Reactor before any area exceeds the Maximum Safe Temperature operating value.
2. With a Primary System discharging into the Secondary Containment, when two or more areas are greater than their Maximum Safe operating values for the same parameter, the Unit Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator (NUSO).

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Return Reactor Water Cleanup (RWCU) to Operation
2.	N/A	R-OATC R-NUSO	Reduce Reactor Power to 75% using Core Flow
3.#S	XA-55-4C_13	C-BOP C-NUSO	Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level
4.	SCHED RWCU	C-OATC TS-NUSO	Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close
5.	N/A	TS-NUSO	Core Spray Loop I Room Cooler EECW Leak
6.	ED08C	C-OATC C-NUSO	2C 4KV Unit Board Trip
7.	RC09 FCV-71-2 FCV-71-3	M-ALL	Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak
8.#	TH23	M-ALL	Fuel Damage
9.	FCV-73-16	C-BOP C-NUSO	Reactor Feedwater Pumps (RFPTs) Trip / HPCI Fails to Automatically Start and Inject

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

\# Event on previous two NRC Exams

\#S Event on previous two NRC Exams Spare Scenario

Events

1. The crew will return Reactor Water Cleanup (RWCU) to service in accordance with 2-OI-69, Reactor Water Cleanup System.

2. The crew will continue the Reactor shutdown and reduce Reactor Power using Core Flow to 75% in accordance with 2-OI-68, Reactor Recirculation System, and 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations.

3. The Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level alarm will be received, requiring the crew to take action to fill the RBCCW Surge Tank in accordance with Alarm Response Procedure 2-ARP-9-4C, Window 13, RBCCW SURGE TANK LEVEL LOW.

4. A leak will develop in the Reactor Water Cleanup System (RWCU), requiring the crew to respond in accordance with 2-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation. 2-FCV-69-1, RWCU INBOARD SUCTION ISOLATION VALVE, will fail to automatically close on an isolation signal, requiring manual action by the crew to isolate the RWCU System. The Nuclear Unit Senior Operator (NUSO) will address Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs), Condition A.

5. The Reactor Building AUO will report that an Emergency Equipment Cooling Water (EECW) leak was found in the Loop I Core Spray Room Cooler and that the leak has been isolated, requiring the NUSO to address Technical Requirements Manual 3.5.3, Equipment Area Coolers, Condition A and Technical Specification 3.5.1, Emergency Core Cooling Systems – Operating, Condition F.

6. 4KV Unit Board 2C will trip, resulting in a loss of the following electrical loads: 2C Condensate Pump, 2C Condensate Booster Pump, 2C Raw Cooling Water (RCW) Pump, 2C Condenser Cooling Water (CCW) Pump, and 2A Control Rod Drive (CRD) Pump. Action will be required to restore CRD Flow in accordance with 2-AOI-85-3, CRD System Failure.

7. Reactor Core Isolation Cooling (RCIC) will develop an un-isolable leak, causing high temperatures and radiation levels in the Reactor Building. The SRO will respond in accordance with 2-EOI-3, Secondary Containment Control.

8. Fuel damage will occur when the Reactor SCRAMs, requiring the crew to Emergency Depressurize the Reactor due to two Area Radiation Levels exceeding their Maximum Safe Values in Secondary Containment.

9. Reactor Feedwater Pumps (RFPTs) will trip when the Reactor MODE SWITCH is placed in SHUTDOWN. When Reactor Water Level lowers to the initiation setpoint, the High Pressure Coolant Injection (HPCI) System will not automatically start, requiring the crew to take action to manually start HPCI for Reactor Water Level control.

The Scenario ends when the crew has Emergency Depressurized the Reactor and has control of Reactor Water level above the Top of Active Fuel ((-) 162 inches) using low pressure systems.

Critical Tasks 2

1. With the Reactor at power and with a Primary System discharging into the Secondary Containment, before any area exceeds the Maximum Safe operating value, the Operator at the Controls inserts a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator.

a. **Safety Significance**

SCRAM reduces to decay heat the energy that the RPV may be discharging into the Secondary Containment.

b. **Cues**

Procedural Compliance.
Secondary Containment Area Temperature, Level and/or radiation indication.

c. **Measured by**

Observation - With a Primary System discharging into Secondary Containment a Reactor SCRAM is initiated before a Maximum Safe condition is reached.

d. **Feedback**

Control Rod positions
Reactor Power reduction

e. **Critical Task Failure Criteria**

The operating crew fails to proceed without delay and in a controlled manner to initiate a Reactor SCRAM from the time it is announced that one Area Temperature is approaching the Maximum Safe value.

2. With a Primary System discharging into the Secondary Containment, when two or more areas are greater than their Maximum Safe operating values for the same parameter, the Unit Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator.

a. **Safety Significance**

Places the Primary System in the lowest possible energy state, rejects heat to the Suppression Pool in preference to outside the Primary Containment, and reduces driving head and flow of system discharging into the Secondary Containment.

b. **Cues**

Procedural Compliance.
Secondary Containment Area Temperature, Level, or Radiation indication.

c. **Measured by**

Observation - NUSO transitions to 2-EOI-C2, Emergency RPV Depressurization, and directs opening at least 6 MSRVs when two or more areas are greater than their Maximum Safe operating values for the same parameter.

d. **Feedback**

Reactor Pressure Trend.
MSRV status indications.

d. **Critical Task Failure Criteria**

The operating crew fails to proceed without delay and in a controlled manner to initiate Emergency Depressurization from the time it is announced that two Area Temperature Levels exceed Maximum Safe value.

Facility: BFN Scenario Number: NRC-1 Op-Test Number: 21-04

Examiners: ____________________________ Operators: SRO: ______________________

____________________________ ATC: ______________________

____________________________ BOP: ______________________

Initial Conditions: 80% Reactor Power.

Turnover: Reactor Shutdown in progress. SRV 1-22 is INOPERABLE (ADS Valve). EHPM tagged for motor bearing inspection.

Critical Tasks:

1. With the Reactor at power and with a Primary System discharging into the Secondary Containment, manually SCRAM the Reactor before any area exceeds the Maximum Safe Temperature operating value.
2. With a Primary System discharging into the Secondary Containment, when two or more areas are greater than their Maximum Safe operating values for the same parameter, the Unit Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator (NUSO).

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Return Reactor Water Cleanup (RWCU) to Operation
2.	N/A	R-OATC R-NUSO	Reduce Reactor Power to 75% using Core Flow
3.#S	XA-55-4C_13	C-BOP C-NUSO	Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level
4.	SCHED RWCU	C-OATC TS-NUSO	Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close
5.	N/A	TS-NUSO	Core Spray Loop I Room Cooler EECW Leak
6.	ED08C	C-OATC C-NUSO	3C 4KV Unit Board Trip
7.	RC09 FCV-71-2 FCV-71-3	M-ALL	Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak
8.#	TH23	M-ALL	Fuel Damage
9.	FCV-73-16	C-BOP C-NUSO	Reactor Feedwater Pumps (RFPTs) Trip / HPCI Fails to Automatically Start and Inject

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

\# Event on previous two NRC Exams

#S Event on previous two NRC Exams Spare Scenario

Events

1. The crew will return Reactor Water Cleanup (RWCU) to service in accordance with 3-OI-69, Reactor Water Cleanup System.

2. The crew will continue the Reactor shutdown and reduce Reactor Power using Core Flow to 75% in accordance with 3-OI-68, Reactor Recirculation System, and 3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations.

3. The Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level alarm will be received, requiring the crew to take action to fill the RBCCW Surge Tank in accordance with Alarm Response Procedure 3-ARP-9-4C, Window 13, RBCCW SURGE TANK LEVEL LOW.

4. A leak will develop in the Reactor Water Cleanup System (RWCU), requiring the crew to respond in accordance with 3-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation. 3-FCV-69-1, RWCU INBOARD SUCTION ISOLATION VALVE, will fail to automatically close on an isolation signal, requiring manual action by the crew to isolate the RWCU System. The Nuclear Unit Senior Operator (NUSO) will address Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs), Condition A.

5. The Reactor Building AUO will report that an Emergency Equipment Cooling Water (EECW) leak was found in the Loop I Core Spray Room Cooler and that the leak has been isolated, requiring the NUSO to address Technical Requirements Manual 3.5.3, Equipment Area Coolers, Condition A and Technical Specification 3.5.1, Emergency Core Cooling Systems – Operating, Condition F.

6. 4KV Unit Board 3C will trip, resulting in a loss of the following electrical loads: 3C Condensate Pump, 3C Condensate Booster Pump, 3C Raw Cooling Water (RCW) Pump, 3C Condenser Cooling Water (CCW) Pump, and 3A Control Rod Drive (CRD) Pump. Manual action will be required to restore CRD Flow in accordance with 3-AOI-85-3, CRD System Failure.

7. Reactor Core Isolation Cooling (RCIC) will develop an un-isolable leak, causing high temperatures and radiation levels in the Reactor Building. The SRO will respond in accordance with 3-EOI-3, Secondary Containment Control.

8. Fuel damage will occur when the Reactor SCRAMs, requiring the crew to Emergency Depressurize the Reactor due to two Area Radiation Levels exceeding their Maximum Safe Values in Secondary Containment.

9. Reactor Feedwater Pumps (RFPTs) will trip when the Reactor MODE SWITCH is placed in SHUTDOWN. When Reactor Water Level lowers to the initiation setpoint, the High Pressure Coolant Injection (HPCI) System will not automatically start, requiring the crew to take action to manually start HPCI for Reactor Water Level control.

The Scenario ends when the crew has Emergency Depressurized the Reactor and has control of Reactor Water level above the Top of Active Fuel ((-) 162 inches) using low pressure systems.

Critical Tasks 2

1. With the Reactor at power and with a Primary System discharging into the Secondary Containment, before any area exceeds the Maximum Safe operating value, the Operator at the Controls inserts a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator.

a. **Safety Significance**

SCRAM reduces to decay heat the energy that the RPV may be discharging into the Secondary Containment.

b. **Cues**

Procedural Compliance.
Secondary Containment Area Temperature, Level and/or radiation indication.

c. **Measured by**

Observation - With a Primary System discharging into Secondary Containment a Reactor SCRAM is initiated before a Maximum Safe condition is reached.

d. **Feedback**

Control Rod positions
Reactor Power reduction

e. **Critical Task Failure Criteria**

The operating crew fails to proceed without delay and in a controlled manner to initiate a Reactor SCRAM from the time it is announced that one Area Temperature is approaching the Maximum Safe value.

2. With a Primary System discharging into the Secondary Containment, when two or more areas are greater than their Maximum Safe operating values for the same parameter, the Unit Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator.

a. **Safety Significance**

Places the Primary System in the lowest possible energy state, rejects heat to the Suppression Pool in preference to outside the Primary Containment, and reduces driving head and flow of system discharging into the Secondary Containment.

b. **Cues**

Procedural Compliance.
Secondary Containment Area Temperature, Level, or Radiation indication.

c. **Measured by**

Observation - NUSO transitions to 3-EOI-C2, Emergency RPV Depressurization, and directs opening at least 6 MSRVs when two or more areas are greater than their Maximum Safe operating values for the same parameter.

d. **Feedback**

Reactor Pressure Trend.
MSRV status indications.

d. **Critical Task Failure Criteria**

The operating crew fails to proceed without delay and in a controlled manner to initiate Emergency Depressurization from the time it is announced that two Area Temperature Levels exceed Maximum Safe value.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 1 of 10

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not start Event 1, Return Reactor Water Cleanup (RWCU) to Operation, request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to return RWCU to operation.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to return RWCU to operation.** **If contacted by the crew as the Reactor Building Assistant Unit Operator (AUO) acknowledge any direction given.**
	NRC	**If Panel 2-9-4B, Window 17, RWCU NON-REGENERATIVE HX DISCH TEMP HIGH, is received, see page 9 of 58 for Alarm Response Procedure actions.**
	NUSO	Directs the Balance of Plant Operator (BOP) to return RWCU to service.
	BOP	2-OI-69, Reactor Water Cleanup System Section 5.1 RWCU Pump Startup **NOTES** 1) All controls and indications are located on Panel 2-9-4 unless noted otherwise. 2) RWCU is required to be operated within the following restrictions with Reactor Pressure ≤50 psig (MODES 2 or 3), or any time the unit is in MODE 4, MODE 5, or de-fueled: • One pump in operation, pump can be operated to its maximum flow capacity • Two pumps in operation, maximum flow limited to ≤100 gpm per pump (200 gpm total) [1] **NOTIFY** Radiation Protection that an RPHP exists for impending RWCU Pump return to service. RECORD name of Radiation Protection representative notified in narrative log.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 2 of 10

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as Radiation Protection concerning an RPHP, acknowledge that RWCU is being returned to service and provide a name for the NOMS narrative log.**
	BOP	[1.1] **ENSURE** appropriate data and signatures recorded on Attachment 6 per Attachment 6 instructions. [2] **REVIEW** Precautions and Limitations in Section 3.0. [3] **ENSURE** RWCU pre-startup requirements in Section 4.0 have been completed. [4] **ENSURE RESET** the RWCU Group 3 Isolation using 2-HS-64-16A-S32, PCIS DIVISION I RESET and 2-HS-64-16A-S33 PCIS DIVISION II RESET, at Panel 2-9-4. [5] **CHECK** the following on Panel 2-LPNL-925-0003, Unit 2 Reactor Building Elevation 621': [5.1] Demin 2A and/or 2B Holding Pumps are running (2-HS-069-6015 and 2-HS-069-6005). [5.2] Demin 2A and/or 2B Outlet Valves are closed (2-HS-069-0035 and/or 2-HS-069-0060).
	Driver	**If contacted as the Reactor Building AUO to perform Step [5], acknowledge the direction and report that the Holding Pumps are running with the Demineralizer Outlet Valves closed.**
	BOP	[6] N/A [7] **ENSURE** 2-TIC-069-0010A, RWCU HEAT EXCHANGERS RBCCW FLOW CONTROL, is in MANUAL, and FULL OPEN demand is on 2-TCV-70-49, RWCU NON-REGENERATIVE HEAT EXCHANGER OUTLET TCV.
	Driver	**When directed to place 2-TIC-069-0010A in manual, insert Event 11. Inform the crew that 2-TIC-069-0010A is in manual and is fully open.**
	BOP	[8] **ENSURE CLOSED** the following: • 2-HC-69-15, RWCU BLOWDOWN PRESSURE CONTROL VALVE • 2-HS-69-16, RWCU BLOWDOWN TO MAIN CONDENSER • 2-HS-69-17, RWCU BLOWDOWN TO RADWASTE

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 3 of 10

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	[9] **ENSURE** the DEFEAT/OPERATE SWITCH FOR 2-HC-069-0015 in the DEFEAT position, using 2-HS-069-0015A. [10] N/A [11] **NOTIFY** Chemistry that RWCU is being placed in service and to check the durability monitor.
	Driver	**When contacted as Chemistry acknowledge the direction to check the durability monitor.**
	BOP	[12] **ENSURE OPEN** the following: [12.1] 2-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE. [12.2] 2-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE. [12.3] 2-FCV-69-8, RWCU DEMIN BYPASS VALVE. [13] **OPEN** 2-FCV-069-0012, RWCU RETURN ISOLATION VALVE by one of the two methods described below. • **THROTTLE OPEN** 2-FCV-069-0012, RWCU RETURN ISOLATION VALVE as follows: **PLACE** 2-HS-69-12A in the OPEN position, **THEN WHEN** intermediate position (red and green light) is indicated, **THEN RETURN** 2-HS-69-12A to the NORM position • **FULLY OPEN** RWCU RETURN ISOLATION VALVE, 2-FCV-069-0012 as follows: **ENSURE** 2-FCV-069-0012 is OPEN **NOTES** 1) Too high a flow on startup after isolation could cause 2-TIS-69-11 to actuate due to a high Non-Regenerative Heat Exchanger Outlet Temperature (2-XS-69-6, RWCU TEMP SELECT, Position 3, WATER TO RWCU DEMINS). 2) The RWCU Pump trips on low flow at 56 gpm, after a 30 second time delay. Failure to immediately raise flow to greater than 56 gpm in the following steps results in a pump trip.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 4 of 10

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	[14] **PLACE** seal purge in operation to pump(s) to be placed in service. (REFER TO Section 8.2)
	Driver	**When directed to place seal purge in service, acknowledge the direction and inform the crew that seal purge has been placed in service in accordance with 2-OI-69, Section 8.2**
	BOP	[15] **START** RWCU PUMP 2A(2B) using 2-HS-69-4A(B)-A, **AND RAISE** flow, using 2-HS-69-12A, RWCU RETURN ISOLATION VALVE, to prevent low flow trip. [16] **IF** two pump operation is desired, **THEN START** the second RWCU PUMP 2B(2A) using 2-HS-69-4B(A)-A, **AND RAISE** flow using RWCU RETURN ISOLATION VALVE, 2-HS-69-12A, to prevent low flow trip. [17] **WHEN** desired, **THEN PLACE** RWCU filter-demineralizers in service. (REFER TO Section 6.2)
	BOP	2-OI-69, Reactor Water Cleanup System Section 6.2, Placing Filter-Demineralizers in Service **NOTES** 1) To prevent resin intrusion into the Reactor Vessel, the resin trap (Filter Effluent Strainer) on a vessel that was removed from service due to high D/P should be backwashed prior to returning the demin vessel to service. 2) The flow rate in the mechanical seal areas can be to a maximum of 8.0 gpm for a 24 to 36 hour period. Based on the layout design showing the flush coming in over the retainer at the back of the mechanical seal, the rise in flush gpm should NOT cause any operational problems with the mechanical seal when conditions are returned to normal.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 5 of 10

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	**CAUTION** When initially placing a filter-demineralizer into service, it is desirable that most RWCU Discharge Flow be returned to the Main Condenser. If the Reactor is pressurized, however, failure to follow temperature restrictions could result in thermal shocking the Regenerative Heat Exchanger. [1] **REVIEW** Precautions and Limitations in Section 3.0. [2] – [10] Performed in the Field by an AUO.
	Driver	**When contacted as the Reactor Building AUO to prepare to roll in RWCU Demineralizer, acknowledge the direction and report that you are standing by with Steps complete Steps [2] through [10] of 2-OI-69, Reactor Water Cleanup System are complete.** **When directed to place filter-demineralizers in service, acknowledge the direction. Insert Event 1 to perform AUO actions to place demineralizers in service, and inform the crew that Demin Flow is rising.** **Demineralizers will roll in over a 1-minute time frame – when complete inform the crew that RWCU filter-demineralizers have been placed in service.**
	BOP	**NOTE** RWCU is required to be operated within the following restrictions with Reactor Pressure ≤ 50 psig (Modes 2 or 3), or any time the unit is in Mode 4, Mode 5, or defueled: • One pump in operation, pump can be operated to its maximum flow capacity • Two pumps in operation, maximum flow limited to ≤ 100 gpm per pump (200 gpm total)

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 1	Page 6 of 10
Event Description: Return Reactor Water Cleanup (RWCU) to Operation			

Time	Position	Applicant's Actions or Behavior
	BOP	[11] **PERFORM** the following simultaneously: • **CLOSE** 2-HS-69-8, RWCU DEMIN BYPASS VALVE on Panel 2-9-4
	Driver	**Verify that the crew is able to clear RWCU Demineralizer Alarm (Panel 2-9-4B, Window 24) – it will automatically reset on Event 1 after 15 seconds. If the crew cannot get the alarm to reset, insert remote function AN01E to RESET again).**
	BOP	[12] **RAISE** flow through the Demin until the desired flow has been established. [13] **ENSURE** 2-HS-069-6015(6005), DEMIN 2A(2B) HOLDING PUMP, in the AUTO position. [14] **CHECK** that Holding Pump 2A(2B), on the Demin being placed in service, has STOPPED. [15] **CHECK** 2-FCV-069-0035B(0060B), DEMIN 2A(2B) HOLDING PUMP DISCH VLV H, has CLOSED.
	Driver	**When directed to perform Steps [12], [13], and [14] acknowledge the direction and inform the crew that Steps [12], [13], and [14] are complete.**
	BOP	[16] **NOTIFY** Chemistry that the filter-demineralizer is in service **AND REQUEST** a sample for conductivity and silica of the effluent. [17] **CHECK** that the results of the filter-demineralizer effluent sample taken by Chemistry are within the limits of CI-13.1, Chemistry Manual, for return to the Reactor Vessel. [17.1] **CHECK** RWCU Flows on ICS per Section 8.16.
	Driver	**When contacted as Chemistry, acknowledge any information or direction given.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 7 of 10

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	Continuing 2-OI-69, Reactor Water Cleanup System Section 5.1 RWCU Pump Startup [18] **ADJUST** 2-HS-69-12A, RWCU RETURN ISOLATION VALVE, using as required to maintain system parameters within limits specified in this procedure. **CAUTION** Failure to maintain RWCU Non-Regenerative Heat Exchanger tube side outlet temperature below 130 ºF will reduce resin efficiency and may result in resin damage. [19] **THROTTLE** blowdown flow as required to maintain the following parameters (REFER TO Section 6.5): • Desired Reactor Water Level • Non-regenerative Heat Exchanger Tube Outlet Temperature less than 130 ºF
	BOP	[20] **IF** at Operations Management discretion it is desired to place 2-TIC-069-0010A, RWCU HEAT EXCHANGERS RBCCW FLOW CONTROL in AUTO, **THEN PERFORM** the following: [20.1] **PLACE** 2-TIC-069-0010A, RWCU HEAT EXCHANGERS RBCCW FLOW CONTROL, in AUTO (REFER TO Section 8.14).
	Driver	**When contacted by the crew to place 2-TIC-069-0010A in automatic acknowledge the direction and insert Event 21. Inform the crew that 2-TIC-069-0010A has been placed in automatic.**

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 1	Page 8 of 10

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	**NOTES** • Seal water to the RWCU Pumps has been observed to slightly lower after pump(s) are placed in service • When the Reactor Vessel is at atmospheric pressure and RWCU Pump seal water is being supplied by CS&S system, RWCU Pump seal water flow may decrease to 0 gpm after the RWCU Pump has started. See PRECAUTION P&L 3.6E [21] **ENSURE** SEAL WATER TO RWCU PUMPS, at Panel 2-25-314, is within 1.8 to 2.0 gpm (REFER TO Section 8.2). • 2-FI-085-0075, RWCU PUMP 2A PURGE WATER FLOW INDICATOR • 2-FI-085-0077, RWCU PUMP 2B SEAL WATER
	Driver	**When contacted as the Reactor Building AUO to perform Step [21], inform the crew that seal water flow is 1.9 gpm.**
	BOP	2-OI-69, Reactor Water Cleanup System Section 5.2, RWCU ICS Temperature Point Restoration [1] **TYPE** RTP in the yellow block at the top of the ICS display and **DEPRESS** Enter key to cause the RESTORE TO PROCESSING/RETURN TO SCAN screen to be displayed.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 9 of 10

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	[2] **ENTER** the following information in the entry blocks of the RESTORE TO PROCESSING/RETURN TO SCAN screen to restore point 69-6DSEL to normal processing using the Tab key to switch to each entry block: A. In the Point ID field, **TYPE** 69-6DSEL. B. In the Modified By field, **TYPE** your initials. C. In the Reason field, **TYPE** short description of reason (like "system started"). D. After all above entries made, then **DEPRESS** the F3 key (Execute) to implement the substitution. E. The INSERT VALUE screen will continue to be displayed. [3] **ENTER** the following information in the entry blocks of the RESTORE TO PROCESSING/RETURN TO SCAN screen to restore point 69-6ASEL to normal processing using the Tab key to switch to each entry block: A. In the Point ID field, **TYPE** 69-6ASEL. B. In the Modified By field, **TYPE** your initials. C. In the Reason field, **TYPE** short description of reason (like "system started"). D. After all above entries made, then **DEPRESS** the F3 key (Execute) to implement the substitution. [4] **DEPRESS** Esc key to exit the RESTORE TO PROCESSING/RETURN TO SCAN screen.
	BOP	2-9-ARP-4B, Alarm Response Procedure RWCU NON-REGENERATIVE HX DISCHARGE TEMPERATURE HIGH, Window 17 Operator Action: A. **CHECK** 2-XS-69-6, RWCU NRHX Discharge Temperature, on Panel 2-9-4. B. **CHECK** RBCCW System Temperature indication normal, Panel 2-9-4.

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 1	Page 10 of 10

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	C. **IF** temperature continues to rise, **THEN PERFORM** the following, otherwise, **MARK** steps N/A: • **REDUCE** system flow or reject flow as necessary to control temperature. • **REFER** TO 2-OI-69, Reactor Water Cleanup System D. **DISPATCH** personnel to check the following: • RWCU Heat Exchangers RBCCW Flow Controller (normally in auto with setpoint at approximately 110 ºF), located on Panel 25-2 Rx Bldg 593' • 2-TCV-70-49, RWCU NON-REGENERATIVE HEAT EXCHANGER OUTLET TCV operating properly (RBCCW to NRHX), located in RWCU HX room E. N/A
	Driver	**If contacted by the crew to check equipment in Step D (see above), acknowledge the direction and report the following as required:** • **RWCU Heat Exchangers RBCCW Flow Controller is set at 110 ºF and is in automatic** • **2-TCV-70-49 is operating properly**
	NRC	**End of Event 1. Proceed to Event 2, Reduce Reactor Power to 75% using Core Flow.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 2 Page 1 of 4

Event Description: Reduce Reactor Power to 75% using Core Flow

Time	Position	Applicant's Actions or Behavior
	NRC	**If the crew does not proceed to Event 2, Reduce Reactor Power to 75% using Core Flow, request that the Driver contact the NUSO as the Shift Manager and direct the crew to reduce Reactor Power to 75%.**
	Driver	**If requested by the Chief Examiner, contact the crew as the Shift Manager and direct the crew to continue with Step [2] of the Reactivity Control Plan (RCP) for the Reactor Shutdown and reduce Reactor Power to 75% using Core Flow.**
	NRC	**The crew may elect to conduct a re-focus reactivity brief prior to lowering Reactor Power.**
	NUSO	Directs the Operator at the Controls (OATC) to lower Reactor Power to 75% in accordance with Step [2] of RCP U2-2104NRC1 and in accordance with 2-OI-68, Reactor Recirculation System, 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations and 2-GOI-100-12, Power Maneuvering.
	OATC	2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations Section 5.3, Power Reduction 5.3.1 Reducing Reactor Power to 40% [1] **ENSURE** the operators are using Attachment 9, Operations Down Power Monitoring. [2] **REDUCE** Reactor Power by combination of Control Rod insertions and Core Flow changes, as recommended by Reactor Engineer.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 2 Page 2 of 4

Event Description: Reduce Reactor Power to 75% using Core Flow

Time	Position	Applicant's Actions or Behavior
	OATC	2-GOI-100-12, Power Maneuvering Section 5.0, Instruction Steps [7] **REDUCE** Reactor Power by combination of Control Rod insertions and Core Flow changes, as recommended by Reactor Engineer. **REFER TO** 2-SR-3.1.3.5(A), Control Rod Coupling Integrity Check and 2-OI-68, Reactor Recirculation System.
	NRC	**2-OI-68, Reactor Recirculation System** **3.0 Precautions and Limitations** **Section 3.5.3, Dual Pump Operation** **E. When raising (lowering) Reactor Power per the Reactivity Plan, the following guideline should be used to establish the 60 rpm mismatch between the Recirc Pumps.** **1. When the first Recirc Pump reaches 1200 rpm (1300 rpm) or both Recirc Pumps are at 1200 rpm (1300 rpm) then individual controls will be used.** **2. While following the Reactivity Plan establish the 60 rpm mismatch using the individual controls for the leading Recirc Pump.** **3. While maintaining the 60 rpm mismatch and using the Reactivity Plan, raise (lower) the Recirc Pump speeds using either the Master Controllers or Individual Controllers.** **4. Once a Recirc Pump reaches 1300 rpm (1200 rpm) the 60 rpm mismatch is no longer required. While following the Reactivity Plan raise (lower) the lagging Recirc Pump using the individual controller match Recirc Pump speeds.** **5. Once the Recirc Pumps are matched, then the speeds may be adjusted as required using the Master Controller or Individual Controllers while maintaining the requirements of Attachment 1.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 2 Page 3 of 4

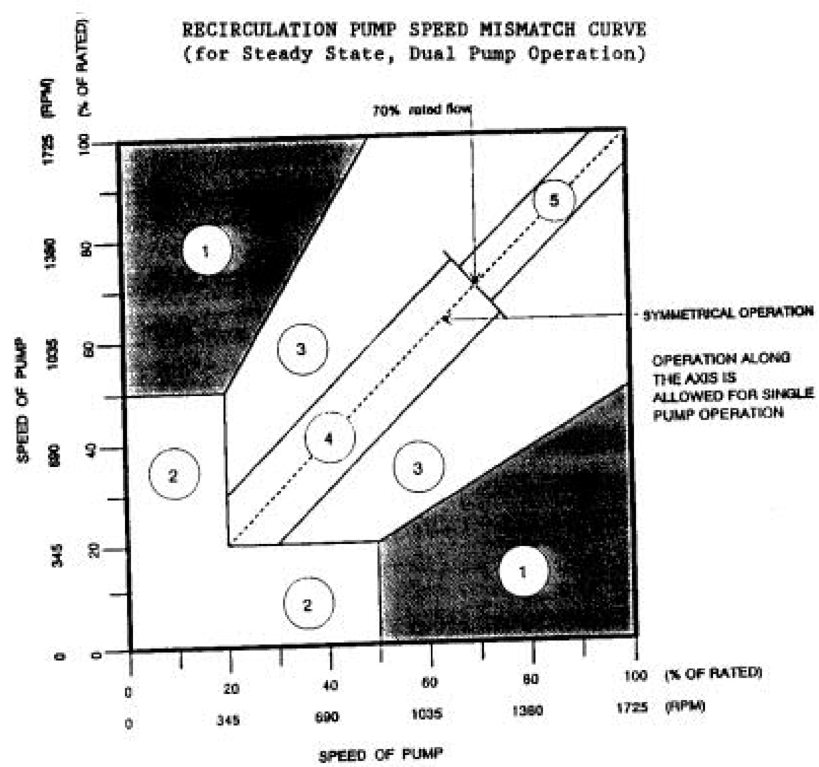
Event Description: Reduce Reactor Power to 75% using Core Flow

Time	Position	Applicant's Actions or Behavior
	NRC	See reproduced procedure page below

BFN Unit 2	Reactor Recirculation System	2-OI-68 Rev. 0159 Page 181 of 209

Attachment 1
(Page 1 of 1)

Recirculation Pump Speed Mismatch Curve



1. Avoid Region To Prevent Excessive Jet Pump Vibration.
2. Below Recirc Drive Minimum speed.
3. Operation allowed if reactor subcritical or during transient periods.
4. Limited Operation for Core flow ≤ 70% rated (mismatch ≤ 10% rated speed).
5. Limited operation for Core flow > 70% rated (mismatch ≤ 5% rated speed).

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 2 Page 4 of 4

Event Description: Reduce Reactor Power to 75% using Core Flow

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown in 0-TI-248, Station Reactor Engineer and 2-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the later part of the operating cycle (Coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the power to flow map. Instruments used to monitor pump speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing Core Flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area. [1] **IF** desired to control Recirc Pumps 2A and/or 2B speed with Recirc Individual Control, **THEN PERFORM** the following: • **LOWER** Recirc Pump 2A using 2-HS-96-17A(17B)(17C), SLOW(MEDIUM)(FAST), (Otherwise N/A) AND/OR • **LOWER** Recirc Pump 2B using 2-HS-96-18A(18B)(18C), SLOW(MEDIUM)(FAST). (Otherwise N/A) [2] **WHEN** desired to control Recirc Pumps 2A and/or 2B speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump Speed 2A & 2B using the following pushbuttons as required. 2-HS-96-33, LOWER SLOW 2-HS-96-34, LOWER MEDIUM 2-HS-96-35, LOWER FAST
	NRC	**When satisfied with power reduction, end of Event 2. Request that the Driver insert Event 3, Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 3 Page 1 of 1

Event Description: Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3, Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level**
	BOP	Acknowledges and reports the following alarm to the NUSO: • RBCCW SURGE TANK LEVEL LOW, 2-9-4C, Window 13
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 2-ARP-9-4C RBCCW SURGE TANK LEVEL LOW, Window 13 A. **ADD** water to the RBCCW Surge Tank for approximately one minute or until low level alarm resets using the following: • 2-FCV-70-1, RBCCW SYS SURGE TANK FILL VALVE (Panel 2-9-4) OR • 2-HCV-2-1369, FCV-70-1 BYPASS VALVE (locally) B. **IF** alarm does NOT reset, **THEN CHECK** the tank locally. C. **IF** unable to maintain RBCCW Surge Tank level, **THEN REFER TO** 2-AOI-70-1, Loss of Reactor Building Closed Cooling Water. D. **IF** necessary to add water more than once per shift, **THEN CHECK** Drywell floor drain system for excessive operation AND **INSPECT** system outside the Drywell for leakage.
	NRC	**The RBCCW Surge Tank Low Level alarm can be cleared 15 seconds after the fill valve is opened.**
	BOP	Opens 2-FCV-70-1, RBCCW SYS SURGE TANK FILL VALVE for approximately one minute and checks that RBCCW SURGE TANK LEVEL LOW 2-9-4C, Window 13 clears.
	Driver	**If contacted as the Reactor Building AUO to check for leaks or check RBCCW Surge Tank Level locally, acknowledge the direction. Wait 2 minutes and report that Surge Tank Level is normal.**
	NRC	**End of Event 3. Request that the Driver insert Event 4, Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 1 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4, Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close.**
	NRC	**2-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION VALVE, will not automatically close on the Primary Containment System Isolation (PCIS) signal, but can be closed manually.**
	OATC/ BOP	Acknowledges and reports the following alarms: • RWCU LEAK DETECTION TEMP HIGH, 2-9-3D, Window 17 • RWCU ISOL LOGIC CHANNEL A TEMP HIGH, 2-9-5B, Window 32 • RWCU ISOL LOGIC CHANNEL B TEMP HIGH, 2-9-5B, Window 33
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures.
	BOP	2-ARP-9-3D, Alarm Response Procedure RWCU LEAK DETECTION TEMP HIGH, Window 17 Operator Action: A. **IF** this alarm is received in conjunction with RWCU ISOL LOGIC CHANNEL A TEMP HIGH [2-XA-55-5B, Window 32] and RWCU ISOL LOGIC CHANNEL B TEMP HIGH [2-XA-55-5B, Window 33], **THEN EXIT** this procedure and **GO TO** 2-ARP-9-5B. Otherwise, **CONTINUE** in this procedure.
	BOP	Exits 2-ARP-9-3D, Alarm Response Procedure, and enters 2-ARP-9-5B, Alarm Response Procedure.

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 4	Page 2 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	BOP	2-ARP-9-5B, Alarm Response Procedure RWCU ISOL LOGIC CHANNEL A TEMP HIGH, Window 32 Operator Action: A. **CHECK** alarm by checking: 1. Analog Trip Units (ATUs) on Panel 2-9-83 and 2-9-85. 2. RWCU LEAK DETECTION TEMP HIGH annunciator in alarm (2-XA-55-3D, Window 17). 3. Area temperature indications on 2-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE, on Panel 2-9-22. 4. Area Radiation Monitors (ARMs) 2-RR-90-1, 2-RR-90-50B on Panel 2-9-2 and 2-RR-90-250 on Panel 1-9-44. 5. ICS 'HPTURB' & 'RWCU' mimics for the 834 and 835 temperature loops. B. **IF** leak is suspected, **THEN MANUALLY ISOLATE** RWCU.
	Driver	**If contacted as Unit 1 Operator to check Area Radiation Monitors or Radiation Recorders, acknowledge the request.**
	BOP	Determines that 2-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION VALVE, failed to automatically close and manually closes 2-FCV-69-2. Informs the NUSO.
	BOP	C. **IF** RWCU automatically isolates, **THEN REFER TO** 2-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation. D. **IF** TIS-69-835A(C) indicates greater than 131 ºF, **THEN ENTER** 2-EOI-3, Secondary Containment Control.
	NUSO	E. **REFER TO** Tech Spec Table 3.3.6.1-1, Primary Containment Isolation Instrumentation. F. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 3 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	NUSO	As required by 2-ARP-9-5B, Window 32 and 2-ARP-9-5B, Window 33, references NPG-SPP-18.3.5, Equipment Important to Emergency Response. No actions are required in accordance with this procedure.
	NRC	**Technical Specifications are covered starting on page 21 of 58.**
	BOP	2-ARP-9-5B, Alarm Response Procedure RWCU ISOL LOGIC CHANNEL B TEMP HIGH, Window 33 Operator Action: A. **CHECK** alarm by checking: 1. ATUs on Panel 2-9-84 and 2-9-86. 2. RWCU LEAK DETECTION TEMP HIGH annunciator in ALARM (2-XA-55-3D, Window 17). 3. Area temperature indications on 2-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE, on Panel 2-9-22. 4. ARMs 2-RR-90-1, 2-RR-90-50B on Panel 2-9-2 and 2-RR-90-250 on Panel 1-9-44. 5. ICS 'HPTURB' & 'RWCU' mimics for the 834 and 835 temperature loops. B. **IF** a leak is suspected, **THEN MANUALLY ISOLATE** RWCU**.** C. **IF** RWCU automatically isolates, **THEN REFER TO** 2-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation. D. **IF** TIS-69-835B(D) indicates greater than 131 ºF, **THEN ENTER** 2-EOI-3, Secondary Containment Control. E. **REFER TO** Technical Specification Table 3.3.6.1-1, Primary Containment Isolation Instrumentation.
	NRC	**No actions are required in accordance with Technical Specification 3.3.6.1.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 4 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	BOP	2-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation Immediate Actions [1] **PERFORM** the following: • **ENSURE CLOSED** 2-FCV-69-1, RWCU INBD SUCTION ISOLATION VALVE • **ENSURE CLOSED** 2-FCV-69-2, RWCU OUTBD SUCTION ISOLATION VALVE • **ENSURE CLOSED** 2-FCV-69-12, RWCU RETURN ISOLATION VALVE • **ENSURE TRIPPED** Reactor Water Cleanup Recirc Pumps 2A and 2B Subsequent Actions [1] **IF** any EOI entry condition is met, **THEN ENTER** appropriate EOI(s).
	NRC	**The NUSO may enter 2-EOI-3, Secondary Containment Control, if Area Temperature or Radiation exceeds the Maximum Normal value. See page 24 of 58 for 2-EOI-2 actions.**
	Driver	**If contacted as an AUO to check ATUs, acknowledge the direction. Wait 3 minutes and report that ATUs 2-TIS-69-835A-D indicate 160 degrees and lowering.**
	BOP	[2] **CHECK** the following to confirm high area temperature condition exists: • 2-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE, (Panel 2-9-22) • ATUs in Auxiliary Instrument Room [3] **IF** isolation is caused by high area temperature, **THEN DETERMINE** if a line break exists by: • RWCU ARMs 2-RI-90-9A, 13A, and 14A • Visual Observation • Rx Zone Exhaust Rad Monitors 2-RE-90-142A, 142B, 143A, and 143B

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 5 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **PERFORM** necessary Heat Balance adjustments. **REFER TO** 2-OI-69, Reactor Water Cleanup System. [5] **CHECK** the following monitors for a rise in activity: A. 2-RR-90-1, AREA RADIATION, Points 9, 13, and 14 (Panel 2-9-2). B. AIR PARTICULATE MONITOR CONSOLE, 2-MON-90-50, 2-RM-90-55 and 57 (Panel 2-9-2). C. Reactor Building, Turbine Building, and Refuel Zone Exh Rad on 0-MON-90-361, CHEMISTRY CAM MONITOR CONTROLLER (Panel 1-9-2). [6] **IF** it has been determined that leakage is the cause of the isolation, **THEN NOTIFY** RADCON of RWCU status. [7] **NOTIFY** Chemistry that RWCU has been removed from service and to perform the following evaluations. • The need to begin sampling Reactor Water • The need to remove the Durability Monitor from service [8] **IF** the isolation cannot be reset, **THEN PERFORM** the following: [8.1] **ISOLATE** seal water from the CRD System by closing 2-69-592(A) and 2-69-614(B) (R-12T, El. 593, Unit 2 Reactor Building). [8.2] **REFER TO** 2-OI-68, Recirculation System, for Recirc System operating restrictions while RWCU is isolated.
	Driver	**If contacted as Radiation Protection or Chemistry acknowledge any directions or reports given.** **If contacted as Unit 1 to check Reactor Building, Turbine Building, and Refuel Zone Exhaust Radiation on 0-MON-90-361, CHEMISTRY CAM MONITOR CONTROLLER (Panel 1-9-2), acknowledge the direction.**
	NUSO	[9] **EVALUATE** Technical Requirements Manual 3.4.1, Coolant Chemistry, for limiting conditions for operation (Required Action).

<table>
<tr><td colspan="3">Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 6 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close</td></tr>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>Technical Requirements Manual 3.4.1, Coolant Chemistry.
LCO 3.4.1 Reactor Coolant Chemistry shall be maintained within the limits of Table 3.4.1-1.
APPLICABILITY: According to Table 3.4.1-1</td></tr>
<tr><td></td><td>NUSO</td><td><table><tr><th>REQUIRED ACTION:</th><th>COMPLETION TIME:</th></tr><tr><td>A.1 – Verify by administrative means that conductivity has not been > 1.0 µmho/cm at 25°C for > 2 weeks in the past year.</td><td>A.1 – Immediately</td></tr><tr><td>B.1 – Verify by administrative means that chloride concentration has not been > 0.2 ppm for > 2 weeks in the past year.</td><td>B.1 – Immediately</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>Table 3.4.1-1
Coolant Chemistry Limits[1]
<table><tr><th>CHEMISTRY PARAMETERS</th><th>COLUMN A
APPLICABLE CONDITION
Prior To Startup And At Steaming Rates < 100,000 lb/hr</th><th>COLUMN B
APPLICABLE CONDITION
Steaming Rates > 100,000 lb/hr</th><th>COLUMN C
APPLICABLE CONDITION
Reactor Not Pressurized With Fuel In Reactor Vessel, Except During Startup Condition</th><th>COLUMN D[2]
APPLICABLE CONDITION
Noble Metal Chemical Application and Subsequent Reactor Coolant Cleanup</th><th>COLUMN E[3]
APPLICABLE CONDITION
Operation of HWC Following Noble Metal Chemical Application</th></tr>
<tr><td>CHLORIDE (ppm)</td><td>≤ 0.1</td><td>≤ 0.2</td><td>≤ 0.5</td><td>≤ 0.1</td><td>≤ 0.2</td></tr>
<tr><td>CONDUCTIVITY (µmho/cm at 25°C)</td><td>≤ 2.0</td><td>≤ 1.0</td><td>≤ 10.0</td><td>≤ 20.0</td><td>≤ 2.0</td></tr>
<tr><td>pH</td><td>5.6-8.6</td><td>5.6-8.6</td><td>5.3-8.6</td><td>4.3-9.9</td><td>5.6-8.8</td></tr></table>
(1) When there is no fuel in the reactor vessel, Technical Requirement reactor coolant chemistry limits do not apply.
(2) During the Noble Metal Chemical Application and subsequent reactor coolant cleanup, CONDITIONS A, B, C, and D (including Required Actions and Completion Times) do not apply.
(3) During operation of HWC following the Noble Metal Chemical Application, CONDITION A (including Required Action and Completion Time) does not apply.</td></tr>
<tr><td></td><td>**Driver**</td><td>**If contacted as Chemistry to verify by administrative means that conductivity and chloride concentration have not exceeded Table 3.4.1-1 limits for >2 weeks in the past year, inform the NUSO that chemistry limits have not exceeded Table 3.4.1-1 limits in the past year.**</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 7 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	NUSO	Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs) LCO 3.6.1.3 Each PCIV, except Reactor Building-to-Suppression Chamber Vacuum Breakers, shall be OPERABLE APPLICABILITY: MODES 1, 2, and 3 When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation." NOTES 1. Penetration flow paths except for 18 and 20 inch purge valve penetration flow paths may be un-isolated intermittently under administrative controls. 2. Separate Condition entry is allowed for each penetration flow path. 3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs. 4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3. **CONDITION:** NOTE: Only applicable to penetration flow paths with two PCIVs. A. – One or more penetration flow paths with one PCIV inoperable except due to MSIV leakage not within limits.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 8 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
A.1 – Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured

<u>AND</u>

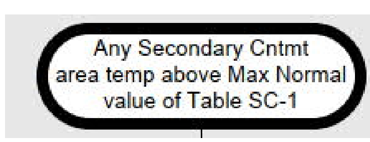
NOTE: Isolation devices in High Radiation Areas may be verified by use of administrative means.

A.2 – Verify the affected penetration flow path is isolated</td><td>COMPLETION TIME:
A.1 – 4 hours except for Main Steam Line

A.2 – Once per 31 days for isolation devices outside Primary Containment

<u>AND</u>

Prior to entering MODE 2 or 3 from MODE 4, if Primary Containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside Primary Containment</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">If RWCU Room Temperature exceeds the Maximum Normal 2-EOI-3, Secondary Containment Control


</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 9 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

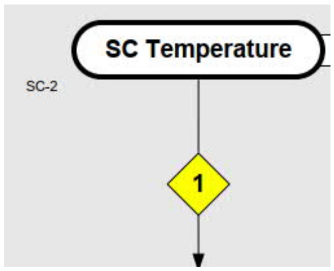
Time	Position	Applicant's Actions or Behavior
	NUSO	2-EOI-3, Secondary Containment Control SC-1

IF	THEN
Reactor Zone Ventilation Exhaust Radiation level is above 72 mR/hr	**NO ACTION REQUIRED**
Refuel Zone Ventilation Exhaust Radiation level is above 72 mR/hr	**NO ACTION REQUIRED**
Reactor Zone Ventilation is isolated AND Reactor Zone Ventilation Exhaust Radiation level is below 72 mR/hr	**NO ACTION REQUIRED**
Refuel Zone Ventilation is isolated AND Refuel Zone Ventilation Exhaust Radiation level is below 72 mR/hr	**NO ACTION REQUIRED**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 10 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	NUSO	(see below)



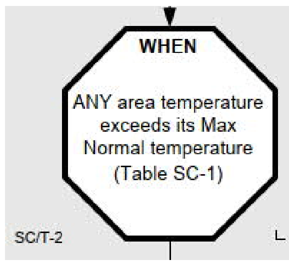
1

- An RPV water lvl instrument may be used to determine or trend lvl only when it reads above the Minimum Indicated Lvl associated with the highest max DW or SC run temp
- If DW temps or SC area temps (Table 6), as applicable, are outside the safe region of Curve 8, the associated instrument may be unreliable due to boiling in the run

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A/B	Emergency -155 to +60	on scale	N/A	below 100
		-150	N/A	101 to 150
		-145	N/A	151 to 200
		-140	N/A	201 to 250
		-130	N/A	251 to 300
		-120	N/A	301 to 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	Normal 0 to +60	on scale	N/A	below 150
		+5	N/A	151 to 200
		+15	N/A	201 to 250
		+20	N/A	251 to 300
		+30	N/A	301 to 350
LI-3-52 LI-3-62A	Post Accident -268 to +32	on scale	N/A	N/A
LI-3-55	Shutdown Floodup 0 to +500	+10	Below 100	N/A
		+15	100 to 150	N/A
		+20	151 to 200	N/A
		+30	201 to 250	N/A
		+40	251 to 300	N/A
		+50	301 to 350	N/A
		+65	351 to 400	N/A

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 11 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	NUSO	SC/T-1 **IF** Reactor Zone or Refuel Zone Ventilation Exhaust Radiation Level is below 72 mr/hr **THEN** operate available Reactor Zone or Refuel Zone Ventilation  3 SC-3 ISOLATE all systems that are discharging into the area EXCEPT systems required: • For damage control OR • To be operated by EOIs NOTE 3 — Tables SC-1 and SC-2 contain information that may be used to determine if a primary system is discharging into Secondary Containment (Emergency Depressurization will reduce discharge).
	NRC	**When the RWCU Leak has been isolated and Area Temperature and Radiation is below the Maximum Safe value, the NUSO may contact the Shift Manager and recommend exiting 2-EOI-2, Secondary Containment Control, as an emergency no longer exists.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 12 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Shift Manager by the NUSO to discuss exiting 2-EOI-3, Secondary Containment Control, agree with any recommendation given.**
	NRC	**End of Event 4. Request that the Driver insert Event 5, Core Spray Loop I Room Cooler EECW Leak.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 5 Page 1 of 2

Event Description: Core Spray Loop I Room Cooler EECW Leak

Time	Position	Applicant's Actions or Behavior	
	Driver	**When requested by the Chief Examiner, insert Event 5, Core Spray Loop I Room Cooler EECW Leak. Contact the NUSO as the Reactor Building Assistant Unit Operator (AUO) and report that you discovered and isolated a water leak in the Core Loop 1 Room Cooler. Report that the following valves were closed to isolate the leak:** • **2-SHV-67-550, NW Core Spray Room Cooler Supply Shutoff** • **2-SHV-67-553, NW Core Spray Room Cooler Outlet** **If asked, the water seems to have stopped leaking.**	
	Driver	**If contacted as Work Control or Mechanical Maintenance, acknowledge any direction concerning the Core Spray Loop I Room Cooler.**	
	NUSO	Technical Requirements Manual 3.5.3, Equipment Area Coolers LCO 3.5.3 The Equipment Area Cooler associated with each RHR Pump and the Equipment Area Cooler associated with each set of Core Spray Pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE. APPLICABILITY: Whenever the associated subsystem is required to be OPERABLE **CONDITION:** A. – One or more Equipment Area Cooler inoperable	
	NUSO	**REQUIRED ACTION:** A.1 – Declare the pump(s) served by that cooler INOPERABLE (Refer to applicable Tech Spec and TRM LCOs)	**COMPLETION TIME:** A.1 – Immediately

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 5 Page 2 of 2

Event Description: Core Spray Loop I Room Cooler EECW Leak

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td colspan="2">Technical Specification 3.5.1, ECCS – Operating
LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE
APPLICABILITY: MODE 1,
MODES 2 and 3, except High Pressure Coolant Injection (HPCI) and ADS valves are not required to be OPERABLE with Reactor Steam Dome Pressure ≤150 psig

CONDITION:
A. – One low pressure ECCS injection/spray subsystem inoperable.
<u>OR</u>
One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.</td></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
See Condition F</td><td>COMPLETION TIME:
See Condition F</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">CONDITION:
F. – One ADS Valve inoperable
<u>AND</u>
Condition A entered</td></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
F.1 – Restore ADS Valve to OPERABLE status
<u>OR</u>
F.2 – Restore Low Pressure ECCS Injection / Spray subsystem to OPERABLE status</td><td>COMPLETION TIME:
F.1 – 72 hours

F.2 – 72 hours</td></tr>
<tr><td></td><td>NRC</td><td colspan="2">End of Event 5. Request that the Driver insert Event 6, 2C 4KV Unit Board Trip.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 6 Page 1 of 3

Event Description: 2C 4KV Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6, 2C 4KV Unit Board Trip.**
	BOP	Acknowledges and reports the following alarms: • 4KV UNIT BOARD 2C UNDERVOLTAGE, 2-9-8B, Window 14 • CONDENSATE BOOSTER PUMP C AUX OIL PRESS LOW, 2-9-6A, Window 14 • MOTOR TRIPOUT, 2-9-8C, Window 33
	CREW	Monitors Reactor Water Level.
	OATC	Reports a loss of Control Rod Drive (CRD) System Flow due to 2A CRD Pump being de-energized.
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures and direct the OATC to respond in accordance with 2-AOI-85-3, CRD System Failure.
	OATC	2-AOI-85-3, CRD System Failure Immediate Actions [1] **IF** operating CRD pump has failed <u>AND</u> standby CRD pump is available, **THEN PERFORM** the following at Panel 2-9-5: (Otherwise N/A) [1.1] **PLACE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, in MAN at minimum setting. [1.2] **START** associated standby CRD Pump using 2-HS-85-2A, CRD Pump 1B.
	Driver	**If contacted as Unit 1 concerning 1B CRD Pump being started for Unit 2, acknowledge any report given. If the crew requests to use 1B CRD Pump for Unit 2, inform the crew that 1B CRD Pump is not needed for Unit 1.**
	OATC	[1.3] **IF** CRD Pump 1B was started, **THEN OPEN** 2-HS-85-8A, CRD PUMP 1B DISCH TO U2.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 6 Page 2 of 3

Event Description: 2C 4KV Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	OATC	[1.4] **ADJUST** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, to establish the following conditions: • 2-PDI-85-18A, CRD COOLING WATER HEADER DP, between 10 psid and 20 psid • 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, between 40 and 65 gpm [1.5] **BALANCE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, AND **PLACE** in AUTO or BALANCE.
	BOP	2-ARP-9-8B, Alarm Response Procedure 4KV UNIT BOARD 2C UNDERVOLTAGE, Window 14 Operator Action: A. **CHECK** Unit in stable condition by checking: • Condensate Pump C • Condensate Booster Pump C • RCW Pump C • CCW Pump C • CRD Pump 2A B. **IF** undervoltage has occurred, **THEN** 1. **CLEAR** disagreement lights on breakers. 2. **REDUCE** load as necessary to maintain stable operating conditions. 3. Condenser discharge may need to be throttled for two CCW Pump operation. **REFER TO** 2-OI-27, Condenser Circulating Water System. C. **CHECK** Unit Bd C for abnormal conditions: relay targets, smoke, burned paint, etc. D. **REFER TO** 0-OI-57A, Switchyard and 4160V AC Electrical System, to re-energize board. E. **REFER TO** appropriate OI for recovery or realignment of equipment.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 6 Page 3 of 3

Event Description: 2C 4KV Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as an AUO, Work Control, or Electrical Maintenance to investigate, acknowledge the direction. If directed to prepare protected equipment tags acknowledge the direction.** **Wait 3 minutes and report that 2C 4KV Unit Board has an overcurrent trip flag.**
	BOP	2-ARP-9-8C, Alarm Response Procedure MOTOR TRIPOUT, Window 33 Operator Action: A. **CHECK** Control Room for white disagreement light illuminated for affected equipment. B. **CLEAR** disagreement light. C. **DISPATCH** personnel to check: • Relays at associated electrical board • Equipment for abnormal conditions • Safe-stop locally reset, if necessary D. **REFER** TO 0-GOI-300-2, Electrical, if relay targets are present or for motor starting limits. E. **REFER** TO appropriate OI for recovery or realignment of equipment
	BOP	2-ARP-9-6A, Alarm Response Procedure CONDENSATE BOOSTER PUMP C AUX OIL PUMP PRESS LOW, Window 14 Operator Action: A. **DISPATCH** personnel to check Booster Pump Lube Oil system: 1. **ENSURE** running or start Aux Oil Pump. 2. **CHECK** for leaks. 3. **CHECK** oil level and temperature at reservoir.
	Driver	**If contacted as the Turbine Building AUO to start 2C Condensate Booster Pump Aux Oil Pump, insert Event 16 and report that the Aux Oil Pump is running.**
	NRC	**End of Event 6. Request that the Driver insert Event 7 Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 1 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7, Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak.**
	NRC	**Event 8, Fuel Damage, and Event 9, Reactor Feedwater Pumps Trip / HPCI Fails to Automatically Start and Inject, are automatically entered by the simulator setup. No action is required by the Driver to Insert Event 8 or Event 9. See page 44 of 58 for Event 8 actions and page 51 of 58 for Event 9 actions.**
	BOP	Acknowledges and reports the following alarms as they are received: • REACTOR BUILDING RADIATION HIGH, 2-9-3A, Window 22 • REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH, 2-9-3A, Window 4 • RCIC STEAM LINE LEAK DETECTION TEMPERATURE HIGH, 2-9-3D, Window 10
	NRC	**See Event 8 (page 44 of 58) for actions for Radiation Alarms.**
	NUSO	Directs the BOP to respond in accordance with the applicable Alarm Response Procedure.
	BOP	2-9-ARP-3D, Alarm Response Procedure RCIC STEAM LINE LEAK DETECTION TEMPERATURE HIGH, Window 10 A. **CHECK** RCIC Temperature elements on 2-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE recorder (points 9-12) on Panel 2-9-22.
	BOP	Checks Area Temperatures on Panel 2-9-22.
	BOP	B. **IF** RCIC is **NOT** in service AND 2-FI-71-1A(B), RCIC STEAM FLOW indicates flow, **THEN ISOLATE** RCIC AND **VERIFY** temperatures lowering.
	BOP	Determines that RCIC failed to automatically isolate, and attempts to manually isolate RCIC. Informs the NUSO that RCIC will not isolate.
	BOP	C. **IF** high temperature is confirmed, **THEN ENTER** 2-EOI-3, Secondary Containment Control.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 2 of 11

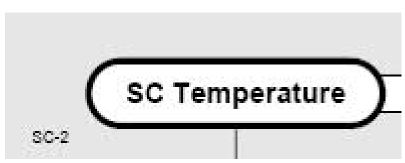
Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	BOP	Confirms that Area Temperatures are rising and informs the NUSO.
	NUSO	Enters 2-EOI-3, Secondary Containment Control. Directs the BOP to monitor Secondary Containment parameters.
	BOP	D. **CHECK** 2-RI-90-26A, CS/RCIC ROOM EL 519 RX BLDG radiation indicator on Panel 2-9-11 and **NOTIFY RADCON** if rising radiation levels are observed. E. **DISPATCH** personnel to investigate.
	Driver	**If contacted as Radiation Protection that radiation levels are rising, acknowledge the report.** **If contacted as the Reactor Building AUO to investigate, acknowledge the direction.**
	NUSO	F. **REFER TO** Tech Specs 3.3.6.1, Primary Containment Isolation Instrumentation and 3.5.3, RCIC System.
	NRC	**Technical Specification evaluation for this event is not required and should not be used to evaluate the candidate's Technical Specification competency.**
		G. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	NRC	**It is not expected that the SRO reference NPG-SPP-18.3.5, Equipment Important to Emergency Response, during this event.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 3 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
2-EOI-3, Secondary Containment Control

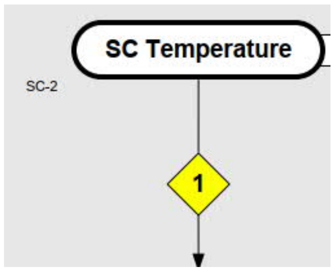
SC-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Reactor Zone Ventilation Exhaust Radiation level is above 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Refuel Zone Ventilation Exhaust Radiation level is above 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Reactor Zone Ventilation is isolated
AND
Reactor Zone Ventilation Exhaust Radiation level is below 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Refuel Zone Ventilation is isolated
AND
Refuel Zone Ventilation Exhaust Radiation level is below 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
</table>


</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 4 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	(see below)



1

- An RPV water lvl instrument may be used to determine or trend lvl only when it reads above the Minimum Indicated Lvl associated with the highest max DW or SC run temp
- If DW temps or SC area temps (Table 6), as applicable, are outside the safe region of Curve 8, the associated instrument may be unreliable due to boiling in the run

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A/B	Emergency -155 to +60	on scale	N/A	below 100
		-150	N/A	101 to 150
		-145	N/A	151 to 200
		-140	N/A	201 to 250
		-130	N/A	251 to 300
		-120	N/A	301 to 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	Normal 0 to +60	on scale	N/A	below 150
		+5	N/A	151 to 200
		+15	N/A	201 to 250
		+20	N/A	251 to 300
		+30	N/A	301 to 350
LI-3-52 LI-3-62A	Post Accident -268 to +32	on scale	N/A	N/A
LI-3-55	Shutdown Floodup 0 to +500	+10	Below 100	N/A
		+15	100 to 150	N/A
		+20	151 to 200	N/A
		+30	201 to 250	N/A
		+40	251 to 300	N/A
		+50	301 to 350	N/A
		+65	351 to 400	N/A

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 5 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

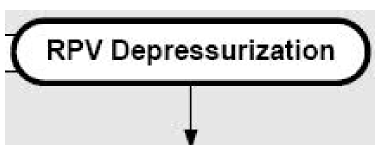
Time	Position	Applicant's Actions or Behavior
	NUSO	SC/T-1 **IF** Reactor Zone or Refuel Zone Ventilation Exhaust Radiation level is below 72 mR/hr **THEN** Operate available Reactor Zone or Refuel Zone Ventilation SC/T-2 **WHEN** ANY Area Temperature exceeds its Max Normal Temperature (Table SC-1)

Table SC-1
Secondary Cntmt Area Temp

Area	Panel 9-3 Alarm Window (unless noted)	Panel 9-22 Temp Element (unless noted)	Max Normal Value °F	Max Safe Value °F	Potential Isolation Sources
RHR sys I pumps	XA-55-3E-4	74-95A	Alarmed	150	FCV-74-47, 48
RHR sys II pumps	XA-55-3E-4	74-95B	Alarmed	210	FCV-74-47, 48
HPCI room	XA-55-3F-10	73-55A	Alarmed	270	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	XA-55-3D-10	71-41A	Alarmed	190	FCV-71-2, 3, 39
Top of torus	XA-55-3D-10 XA-55-3F-10 XA-55-3E-4	71-41B, C, D 73-55B, C, D 74-95G, H	Alarmed Alarmed Alarmed	200 240 240	FCV-71, 2, 3 FCV-73-2, 3, 81 FCV-74-47, 48
Steam tunnel (RB)	XA-55-3D-24	1-60A (Panel 9-3)	Alarmed	315	MSIVs FCV-71-2, 3, FCV-69-1, 2, 12
DW access	XA-55-3E-4	74-95E	Alarmed	170	FCV-74-47, 48
RB el 565 W (RWCU pipe trench)	XA-55-5B-32 (Panel 9-5) XA-55-5B-33 (Panel 9-5)	69-835A, B, C, D (Aux Inst room)	Alarmed	170	FCV-69-1, 2, 12
RWCU hx room	XA-55-3D-17	69-29F, G, H	Alarmed	220	FCV-69-1, 2, 12
RWCU pump A	XA-55-3D-17	69-29D	Alarmed	215	FCV-69-1, 2, 12
RWCU pump B	XA-55-3D-17	69-29E	Alarmed	215	FCV-69-1, 2, 12
RB el 593	XA-55-3E-4	74-95C, D	Alarmed	195	FCV-74-47, 48
RB el 621	XA-55-3E-4	74-95F	Alarmed	155	FCV-43-13, 14

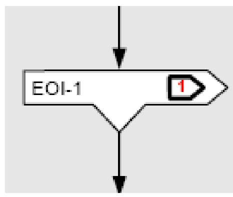
Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 6 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	SC-3 3 ISOLATE all systems that are discharging into the area EXCEPT systems required: • For damage control OR • To be operated by EOIs NOTE 3 — Tables SC-1 and SC-2 contain information that may be used to determine if a primary system is discharging into Secondary Containment (Emergency Depressurization will reduce discharge). SC-4  SC-7 3 **WHEN** A Primary System is discharging into Secondary Containment

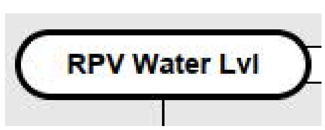
Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 7	Page 7 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	SC-8 **BEFORE** ANY Secondary Containment parameter reaches its Max Safe Value (Tables SC-1, SC-2, and SC-3) 
	CREW	**Critical Task:** **With the Reactor at power and with a Primary System discharging into the Secondary Containment, manually SCRAM the Reactor before any area exceeds the Maximum Safe Temperature operating value.** **Critical Task Failure Criteria:** **The operating crew fails to proceed without delay and in a controlled manner to initiate a Reactor SCRAM from the time it is announced that one Area Temperature is approaching the Maximum Safe value.**
	NUSO	Enters 2-EOI-1, RPV Control. Directs the crew to enter 2-AOI-100-1, Reactor SCRAM, and directs the OATC to insert a manual Reactor SCRAM.
	NRC	**Event 8, Fuel Damage, and Event 9, Reactor Feedwater Pumps Trip / HPCI Fails to Automatically Initiate, are inserted when the Reactor MODE SWITCH is placed in SHUTDOWN.**
	OATC	Inserts a manual Reactor SCRAM.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 8 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	OATC	2-AOI-100-1, Reactor SCRAM Immediate Actions [1] **DEPRESS** 2-HS-99-5A/S3A, REACTOR SCRAM A and 2-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 2-9-5. [2] **PLACE** 2-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] N/A [4] **IF** Reactor Power is 5% or BELOW, **THEN**: (otherwise **MARK** N/A) **REPORT** the following to the US: • Reactor Scram • Mode Switch is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Power level [5] N/A
	OATC	Determines that all Reactor Feedwater Pumps (RFPTs) have tripped and informs the NUSO (See Event 9).
	NUSO	2-EOI-1, RPV Control  RC/L-1 **ENSURE** each as required: • PCIS isolations (Groups 1, 2, and 3) • ECCS • RCIC

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 9 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	RC/L-2

IF	THEN
RPV Water Level can be restored and maintained above (-)162 in. AND The ADS timer has initiated	**INHIBIT** ADS
Loss of available injection systems is anticipated OR Raising RPV Water Level above (+)51 in. will facilitate use of shutdown cooling, steam-driven injection systems, or Alternate Injection Subsystems (Table L-2)	**NO ACTION REQUIRED**

RC/L-3

RESTORE and **MAINTAIN** RPV Water Level between (+)2 in. and (+)51 in. with ANY Preferred Injection Systems (Table L-1)	
IF	**THEN**
RPV Water Level cannot be restored and maintained between (+)2 in. and (+)51 in.	**NO ACTION REQUIRED**
RPV Water Level cannot be restored and maintained above (-)162 in.	**NO ACTION REQUIRED**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 10 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs the OATC/BOP to maintain Reactor Water Level using HPCI in accordance with 2-EOI-Appendix-5D, Injection System Lineup HPCI.
	NRC	**2-EOI-Appendix-5D, Injection System Lineup HPCI actions are covered in Event 9. See page 51 of 58.**
	NUSO	Table L-1 (below)
	NUSO	RPV Press, RC/P-1, RC-P/2 (below)

Table L-1
Preferred Injection Systems

SOURCES	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available 2 3 6	5C, 20M	1200 psig
HPCI with CST suction if available 2 6 7	5D, 20N	1200 psig
CNDS	6A	480 psig
CS 2	6D, 6E	330 psig
LPCI 2	6B, 6C	320 psig

RPV Press

RC/P-1

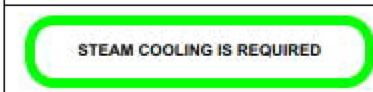
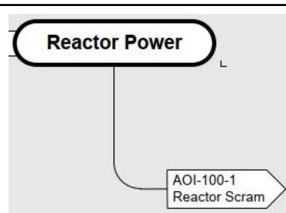
IF	THEN
A high Drywell Pressure ECCS signal exists (2.45 psig)	NO ACTION REQUIRED
EMERGENCY RPV DEPRESSURIZATION is REQUIRED or has been required	2 C2 Emergency RPV Depressurization
Emergency RPV Depressurization is anticipated	NO ACTION REQUIRED

RC-P/2

IF ANY MSRV is cycling
THEN NO ACTION REQUIRED

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 11 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>RC/P-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Suppression Pool Temperature and Level CANNOT be maintained in a safe area of Curve 3 at the existing RPV Pressure</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Pool Level CANNOT be maintained in the safe area of Curve 4</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>
</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-4
<table>
<tr><td colspan="2">STABILIZE RPV Pressure below 1073 psig using the Main Turbine Bypass Valves (APPX 8B)
➢ OK to use ANY Alternate RPV Pressure Control Systems (Table P-1)
➢ Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>DW Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to maintain Reactor Pressure using the Main Turbine Bypass Valves.</td></tr>
<tr><td></td><td>NUSO</td><td>
</td></tr>
<tr><td></td><td>NRC</td><td>End of Event 7. Continue to Event 8.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 1 of 7

Event Description: Fuel Damage

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8, Fuel Damage, is inserted when the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required by the driver to insert Event 8.**
	BOP	Acknowledges and reports the following alarms to the NUSO as they are received: • REACTOR BUILDING RADIATION HIGH, 2-9-3A, Window 22 • REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH, 2-9-3A, Window 4
	BOP	2-ARP-9-3A, Alarm Response Procedure REACTOR BUILDING RADIATION HIGH, Window 22 Operator Action: A. **DETERMINE** area with high radiation level on Panel 2-9-11. (Alarm on Panel 2-9-11 will automatically reset if radiation level lowers below setpoint.)
	BOP	Monitors Radiation Levels on Panel 2-9-11. Keeps the NUSO informed on instruments which are alarming and which are approaching Maximum Safe Values.
	BOP	B. N/A C. N/A D. **NOTIFY** Radiation Protection.
	Driver	**If contacted as Radiation Protection, acknowledge any reports or direction given.**
	BOP	E. **IF** the TSC is NOT manned and a "VALID" radiological condition exists, **THEN USE** public address system to evacuate area where high radiological conditions exist.
	BOP	Makes a plant announcement to evacuate the Reactor Building due to high radiation.

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 8	Page 2 of 7
Event Description: Fuel Damage			

Time	Position	Applicant's Actions or Behavior
	BOP	F. N/A G. **MONITOR** other parameters providing input to this annunciator frequently as these parameters will be masked from alarming while this alarm is sealed in. H. **IF** a CREV initiation is received, **THEN** 1. **CHECK** CREV A(B) Flow is ≥ 2700 CFM, and ≤ 3300 CFM as indicated on 0-FI-031-7214(7213) within 5 hours of the CREV initiation. 2. **IF** CREV A(B) Flow is NOT ≥2700 CFM, and ≤ 3300 CFM as indicated on 0-FI-031-7214(7213), **THEN PERFORM** the following: (Otherwise N/A) a. **STOP** the operating CREV per 0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System. b. **START** the standby CREV per 0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System.
	Driver	**If contacted as an AUO to monitor CREV operation, acknowledge the direction.**
	BOP	I. N/A J. For all radiation indicators except FUEL STORAGE POOL radiation indicator, 2-RI-90-30, **ENTER** 2-EOI-3, Secondary Containment Control Flowchart.
	NUSO	Re-enters 2-EOI-3, Secondary Containment Control (if not already entered on Secondary Containment Radiation).
	BOP	K. N/A L. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	NRC	**It is not expected that the SRO reference NPG-SPP-18.3.5, Equipment Important to Emergency Response, during this event.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 3 of 7

Event Description: Fuel Damage

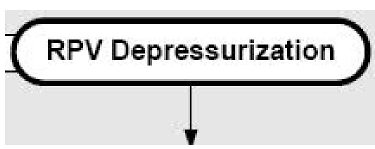
Time	Position	Applicant's Actions or Behavior
	NUSO	**SC Radiation** SC/R-1 **WHEN** ANY Area Radiation Level exceeds its Max Normal Radiation Level (Table SC-2)

Table SC-2
Secondary Cntmt Area Radiation

Area	Applicable Radiation Indicators	Max Normal Value mR/hr	Max Safe Value mR/hr	Potential Isolation Sources
RHR sys I pumps	90-25A	Alarmed	1000	FCV-74-47, 48
RHR sys II pumps	90-28A	Alarmed	1000	FCV-74-47, 48
HPCI room	90-24A	Alarmed	1000	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	90-26A	Alarmed	1000	FCV-71-2, 3, 39
CS sys II pumps	90-27A	Alarmed	1000	None
Top of torus General area	90-29A	Alarmed	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB el 565 W	90-20A	Alarmed	1000	FCV-69-1, 2, 12 SDV vents & drains
RB el 565 E	90-21A	Alarmed	1000	SDV vents & drains
RB el 565 NE	90-23A	Alarmed	1000	None
TIP room	90-22A	Alarmed	100,000	TIP ball valve
RB el 593	90-13A, 14A	Alarmed	1000	FCV-74-47, 48
RB el 621	90-9A	Alarmed	1000	FCV-43-13, 14
Recirc MG sets	90-4A	Alarmed	1000	None
Refuel floor	90-1A, 2A 3A	Alarmed	1000	None

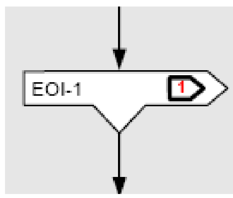
Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 4 of 7

Event Description: Fuel Damage

Time	Position	Applicant's Actions or Behavior
	NUSO	SC-3 3 ISOLATE all systems that are discharging into the area EXCEPT systems required: • For damage control OR • To be operated by EOIs NOTE 3 — Tables SC-1 and SC-2 contain information that may be used to determine if a primary system is discharging into Secondary Containment (Emergency Depressurization will reduce discharge). SC-4  SC-7 3 — **WHEN** A Primary System is discharging into Secondary Containment

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 5 of 7

Event Description: Fuel Damage

Time	Position	Applicant's Actions or Behavior
	NUSO	SC-8 **BEFORE** ANY Secondary Containment parameter reaches its Max Safe Value (Tables SC-1, SC-2, and SC-3) 
	BOP	Monitors Area Radiation levels and informs the NUSO when two areas are at Maximum Safe.
	NUSO	SC-9 **WHEN** Any Secondary Containment parameter exceeds its Max Safe value in two (2) or more areas for the same parameter (Tables SC-1, SC-2, SC-3) SC-10 **EMERGENCY DEPRESSURIZATION IS REQUIRED**
	NUSO	Updates the crew that Emergency Depressurization is required. Enters 2-C-2, Emergency RPV Depressurization.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 6 of 7

Event Description: Fuel Damage

Time	Position	Applicant's Actions or Behavior
	Crew	**Critical Task:** **With a Primary System discharging into the Secondary Containment when two or more areas are greater than their maximum safe operating values for the same parameter, the Balance of Plant Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator.** **Critical Task Failure Criteria:** **The operating crew fails to proceed with without delay and in a controlled manner to initiate Emergency Depressurization from the time it is announced that two Area Radiation Levels exceed Maximum Safe value.**
	NUSO	2-C-2, Emergency RPV Depressurization C2-1 <table><tr><th>IF</th><th>THEN</th></tr><tr><td>Reactor Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr><tr><td>It is anticipated that Reactor depressurization will result in loss of injection required for Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr><tr><td>Containment Water Level CANNOT be maintained below 44 feet</td><td>NO ACTION REQUIRED</td></tr></table>
	NUSO	C2-2 **IF** Drywell Pressure is above 2.45 PSIG **THEN PREVENT** injection from ONLY those Core Spray and LPCI pumps NOT required to assure Adequate Core Cooling (Appendix 4)

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 7 of 7

Event Description: Fuel Damage

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>C2-3
EMERGENCY DEPRESSURIZE the Reactor

IF Suppression Pool Water Level is above 5.5 feet
THEN OPEN 6 MSRVs (ADS Valves preferred)
➢ OK to exceed 100ºF/hr cooldown rate
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Less than 4 MSRVs can be opened
AND
Reactor Pressure is 80 PSIG or more above Suppression Chamber Pressure</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
<tr><td></td><td>BOP</td><td>Opens 5 SRVs and one additional SRV (due to ADS Valve 1-22 being out of service).</td></tr>
<tr><td></td><td>NUSO</td><td>C2-4
WHEN
Shutdown Cooling RPV Pressure interlock clears AND further cooldown is required</td></tr>
<tr><td></td><td>NRC</td><td>End of Event 8. When the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 9 Page 1 of 3

Event Description: Reactor Feedwater Pumps (RFPTs) Trip / HPCI Fails to Automatically Start and Inject

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9, Reactor Feedwater Pumps Trip / HPCI Fails to Automatically Start and Inject, is inserted when the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required by the Driver to insert Event 9.**
	OATC	Reports that all Reactor Feedwater Pumps (RFPTs) have tripped.
	BOP	When Reactor Water Level reaches the High Pressure Coolant Injection (HPCI) initiation setpoint (-45"), determines that HPCI did not automatically start and manually starts HPCI. Informs the NUSO of the actions required to start HPCI.
	NUSO	Directs the BOP to maintain Reactor Water Level using 2-EOI-Appendix-5D, Injection System Lineup HPCI.
	BOP	2-EOI-Appendix-5D, Injection System Lineup HPCI [1] **IF** Suppression Pool Level drops below 12.75 ft. during HPCI operation, **THEN TRIP** HPCI and **CONTROL** injection using other options. [2] **IF** Suppression Pool level CANNOT be maintained below 5.25 in, **THEN EXECUTE** EOI Appendix 16E, Bypassing HPCI High Suppression Pool Water Level Suction Transfer Interlock, concurrently with this procedure to bypass HPCI High Suppression Pool Water Level Suction Transfer Interlock. [3] **IF** BOTH of the following exist: • High temperature exists in the HPCI area, **AND** • SRO directs bypass of HPCI High Temperature Isolation Interlocks, **THEN PERFORM** the following: [3.1] **EXECUTE** EOI Appendix 16L, Bypassing HPCI High Temperature Isolation, concurrently with this procedure. [3.2] **RESET** auto isolation logic using HPCI AUTO-ISOL LOGIC A (B) RESET pushbuttons. [4] **VERIFY** at least one SGTS train in operation.

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 9	Page 2 of 3

Event Description: Reactor Feedwater Pumps (RFPTs) Trip / HPCI Fails to Automatically Start and Inject

Time	Position	Applicant's Actions or Behavior
	BOP	**CAUTIONS** 1) Operating HPCI Turbine below 2400 rpm may result in unstable system operation and equipment damage. 2) Operating HPCI Turbine with Suction Temperatures above 140 ºF may result in equipment damage. [5] **VERIFY** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, is in one of the following configurations, as desired: • in AUTO and set for 5300 gpm for rapid injection • in AUTO and set for approximately 2500 gpm for slower injection • in MANUAL with output at approximately 50% for slower injection **NOTE** HPCI Auxiliary Oil Pump will NOT start UNTIL 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, starts to open. [6] **IF** high Reactor Water Level trip logic is actuated, **THEN** [6.1] **DEPRESS** HPCI TURBINE TRIP RX LVL HIGH RESET pushbutton. [6.2] **CHECK** HPCI TURBINE TRIP LVL HIGH amber light has extinguished. [7] **PLACE** 2-HS-73-47A, HPCI AUXILIARY OIL PUMP handswitch in START. [8] **PLACE** 2-HS-73-10A, HPCI STEAM PACKING EXHAUSTER handswitch in START. [9] **OPEN** 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE. [10] **OPEN** 2-FCV-73-44, HPCI PUMP INJECTION VALVE. [11] **OPEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start the HPCI Turbine.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 9 Page 3 of 3

Event Description: Reactor Feedwater Pumps (RFPTs) Trip / HPCI Fails to Automatically Start and Inject

Time	Position	Applicant's Actions or Behavior
	BOP	[12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine speed accelerates. B. 2-CKV-73-45, HPCI SYSTEM CHECK VLV, opens by observing 2-ZI-73-45A, DISC POSITION, red light illuminated. C. HPCI flow to RPV stabilizes and is controlled automatically at the setpoint. (N/A if controller in manual). D. 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds approximately 1200 gpm. **CAUTION** HPCI PUMP MIN FLOW VALVE, 2-FCV-073-0030, automatically opens when system flow is at or below 900 gpm (lowering) only if a system initiation signal is present. Manually opening the Min Flow Valve may be required for pump min flow protection. [13] **ADJUST** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller as necessary to control injection. [14] **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. [15] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN PLACE** HPCI AUXILIARY OIL PUMP handswitch in AUTO. [16] N/A [17] N/A
	NRC	**End of Event 9. When the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.**

Scenario Setup
UNIT 2

IC	54
Exam IC	276

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-68	159	EOI-3	17	ARP 6A	34
OI-69	139	2-C-2	11	ARP 8B	17
GOI-12	48	APPX 5D	10	ARP 8C	19
GOI-12A	118	ARP 3A	55	TRM 3.4.1	21
AOI-64-2A	27	ARP 3D	34	TS 3.5.1	A294
AOI-85-3	26	ARP 4B	51	TS 3.5.3	0
AOI-100-1	116	ARP 4C	35	TS 3.6.1.3	A253
EOI-1	18	ARP 5B	31		

Simulator Setup	• Verify camera system is powered down (admin password = abcd1234) • Start CPERF **PRIOR** to placing the Simulator in RUN • Ensure Danger Tags are placed on SRV 1-22 and the Emergency High Pressure Makeup Pump
Schedule Files(s):	2104 NRC Scenario 1 UNIT 2.sch RWCU.sch
Event Files(s):	2104 NRC Scenario 1 UNIT 2.evt

Schedule File – 2104 NRC Scenario 1 Unit 2.sch

Event	Action	Description
	2104 NRC Scenario 1 Unit 2.evt	Event File
1	Insert remote CU01 to 55.00000 in 60	RWCU DEMIN FILTER A FRC-69-35
1	Insert remote CU02 to 55.00000 in 60	RWCU DEMIN FILTER B FRC-69-60
1	Insert remote AN01E after 15 to RESET	CU LOCAL RESET (2-XA-55-4B W24)
11	Insert remote CU05 to MANUAL	RWCU HX RBCCW FLOW CONTROL TIC-69-10A
21	Insert remote CU05 to AUTO	RWCU HX RBCCW FLOW CONTROL TIC-69-10A

Schedule File

Event	Action	Description
3	Insert malfunction XA-55-4C_13 to ON	RBCCW SURGE TANK LEVEL LOW 2-LA-70-2B
13	Insert malfunction XA-55-4C_13 after 10 to NORMAL delete in 1	RBCCW SURGE TANK LEVEL LOW 2-LA-70-2B
4	Schedule F:\2104\NRC\Scenarios\U2\Scenario 1\RWCU.sch	
6	Insert malfunction ED08C	4KV UNIT BOARD 2C FAILURE (RELAY 86-316 AND 86-532)
16	Insert remote FW06 to START	CONDENSATE BOOSTER PUMP C AUX OIL PUMP
7	Insert malfunction RC09 to 100.00000 in 900	RCIC STEAM LEAK INTO RCIC ROOM
	Insert malfunction FCV-71-2 to FAIL_NOW	MOTOR_OPERATED_VALVE RCIC STEAM LINE INBD ISOL VLV
	Insert override ZLOHS712A_1 to Off	HS-71-2A-GREEN RCIC STEAM LINE INBD ISOLATION VLV
	Insert override ZLOHS712A_2 to On	HS-71-2A-Red* RCIC STEAM LINE INBD ISOLATION VLV
17	Delete override ZLOHS712A_2	HS-71-2A-Red* RCIC STEAM LINE INBD ISOLATION VLV
	Insert malfunction FCV-71-3 to FAIL_NOW	MOTOR_OPERATED_VALVE RCIC STEAM LINE OUTBD ISOL VLV
	Insert override ZLOHS713A_1 to Off	HS-71-3A-GREEN RCIC STEAM LINE OUTBD ISOLATION VLV
	Insert override ZLOHS713A_2 to On	HS-71-3A-Red* RCIC STEAM LINE OUTBD ISOLATION VLV
8	Insert malfunction FW14A	RFPT 2A TRIP ON RFPT 2A BEARING LOW OIL PRESSURE (PS-3-123B)
8	Insert malfunction FW14B	RFPT 2B TRIP ON RFPT 2A BEARING LOW OIL PRESSURE (PS-3-149B)
8	Insert malfunction FW14C	RFPT 2C TRIP ON RFPT 2A BEARING LOW OIL PRESSURE (PS-3-174B)

Schedule File – 2104 NRC Scenario 1 Unit 2.sch

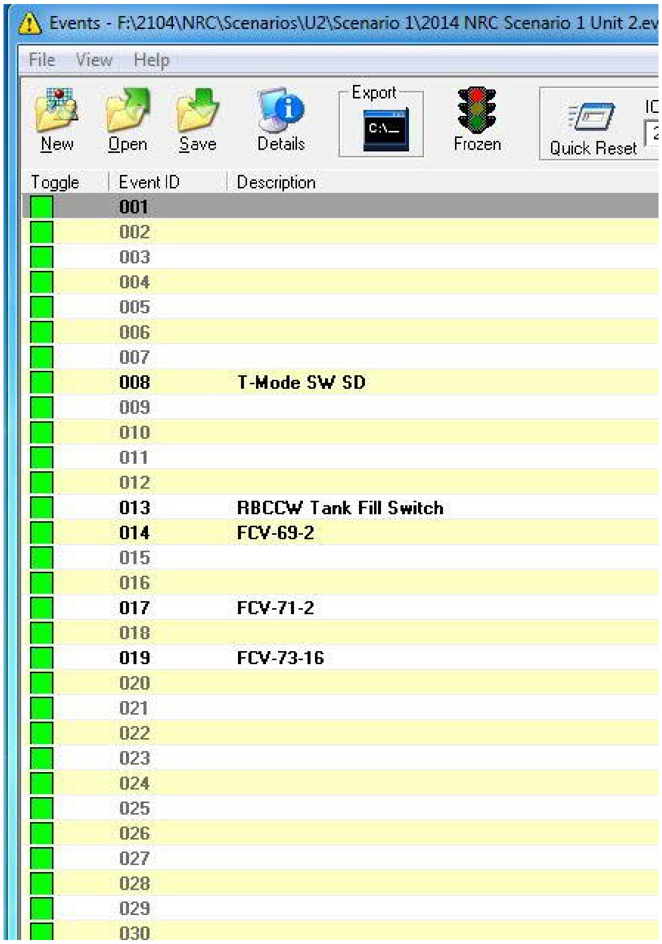
Event	Action	Description
8	Insert malfunction TH23 to 10.00000 in 900	FUEL CLADDING DAMAGE
	Insert malfunction FCV-73-16 to FAIL_NOW	MOTOR_OPERATED_VALVE HPCI TURBINE STEAM SUPPLY VLV
	Insert override ZLOHS7316A_1 to On	HS-73-16A-Green* HPCI TURBINE STEAM SUPPLY VLV
	Delete malfunction FCV-73-16	MOTOR_OPERATED_VALVE HPCI TURBINE STEAM SUPPLY VLV
	Delete override ZLOHS7316A_1	HS-73-16A-Green* HPCI TURBINE STEAM SUPPLY VLV

Schedule File – RWCU.sch

Event	Action	Description
	Insert malfunction CU04 to 25.00000	RWCU SYSTEM SUCTION BREAK
	Insert malfunction FCV-69-2 to FAIL_NOW	MOTOR_OPERATED_VALVE RWCU OUTBOARD ISOLATION VLV
	Insert override ZLOHS692A_1 to Off	HS-69-2A-GREEN RWCU OUTBD SUCT ISOLATION VALVE
	Insert override ZLOHS692A_2 to On	HS-69-2A-Red* RWCU OUTBD SUCT ISOLATION VALVE
14	Delete malfunction FCV-69-2	MOTOR_OPERATED_VALVE RWCU OUTBOARD ISOLATION VLV
14	Delete override ZLOHS692A_1	HS-69-2A-GREEN RWCU OUTBD SUCT ISOLATION VALVE
14	Delete override ZLOHS692A_2	HS-69-2A-Red* RWCU OUTBD SUCT ISOLATION VALVE

Event File

List



Details

<table>
<tr><td colspan="3">UNIT 2 SHIFT TURNOVER MEETING</td><td>Today</td></tr>
<tr><td rowspan="2">MODE 1</td><td><u>DAYS ON LINE</u>
208</td><td rowspan="2"><u>Total Drywell Leakage (gpm)</u>
1.55</td><td><u>Protected Equipment</u></td></tr>
<tr><td rowspan="2">PRA (EOOS) -GREEN
500Kv GRID - Qualified
161Kv Grid -Qualified</td><td></td></tr>
<tr><td><u>Rx Power</u>
80%</td><td><u>Floor Drain (gpm)</u>
0.11</td><td></td></tr>
<tr><td><u>MWe</u></td><td><u>Last breaker closure</u>
4/10/19 4:31</td><td><u>Equipment Drain (gpm)</u>
1.44</td><td></td></tr>
</table>

☐ Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐ CR Reviews Complete ◻ Leadership and Team Effectiveness

CHANGES IN LCOs

SRV 1-22 is INOPERABLE (ADS Valve). Tech Spec 3.5.1, Condition E (Day 4 of 14 Day LCO)

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

Reactor Shutdown. Maintain RFPTs, Condensate, and Condensate Booster Pumps running until Reactor Power is <70%

EHPM tagged for motor bearing inspection

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Continue the Reactor Shutdown. Reduce Reactor Power to 75% using Core Flow, then wait for further guidance from RE.

OPERATOR WORK AROUNDS **OWAs - 0** **Burdens - 0** **Challenges - 6**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

FPRM Attachment A, FPLCO A.2.2.1, Unit 2 Emergency High Pressure Makeup Pump – Fire Watch established

SCHEDULED ITEMS NOT COMPLETED

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 1 of 9

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not start Event 1, Return Reactor Water Cleanup (RWCU) to Operation, request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to return RWCU to operation.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to return RWCU to operation.** **If contacted by the crew as the Reactor Building Assistant Unit Operator (AUO) acknowledge any direction given.**
	NRC	**If Panel 3-9-4B, Window 17, RWCU NON-REGENERATIVE HX DISCH TEMP HIGH, is received, see page 9 of xx for Alarm Response Procedure actions.**
	NUSO	Directs the Balance of Plant Operator (BOP) to return RWCU to service.
	BOP	3-OI-69, Reactor Water Cleanup System Section 5.1 RWCU Pump Startup **NOTES** 1) All controls and indications are located on Panel 3-9-4 unless noted otherwise. 2) RWCU is required to be operated with the following restrictions with Reactor Pressure ≤ 50 psig (MODES 2 or 3) or any time the unit is in MODE 4, MODE 5, or defueled: • One pump in operation, pump can be operated to its maximum flow capacity. • Two pumps operation, max flow limit of ≤ 100 gpm per pump (200 gpm total). [1] RPHP [1.1] **NOTIFY** Radiation Protection that an RPHP exists for impending RWCU Pump return to service. RECORD name of Radiation Protection representative notified in narrative log.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 2 of 9

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as Radiation Protection concerning an RPHP, acknowledge that RWCU is being returned to service and provide a name for the NOMS narrative log.**
	BOP	[1.2] **ENSURE** appropriate data and signatures recorded on Attachment 7 per Attachment 7 instructions. [2] **REVIEW** Precautions and Limitations in Section 3.0. [3] **ENSURE** RWCU pre-startup requirements in Section 4.0 have been completed. [4] **ENSURE RESET** the RWCU Group 3 Isolation using 3-HS-64-16A-S32, PCIS DIV I RESET and 3-HS-64-16A-S33, PCIS DIV II RESET at Panel 3-9-4. [5] **CHECK** the following on Panel 3-LPNL-925-0003, Unit 3 Reactor Building, Elevation.621': [5.1] Demin 3A and/or 3B Holding Pumps are running (3-HS-069-6015 and 3-HS-069-6005). [5.2] Demin 3A and/or 3B Outlet Valves are closed (3-HS-069-0035 and/or 3-HS-069-0060).
	Driver	**If contacted as the Reactor Building AUO to perform Step [5], acknowledge the direction and report that the Holding Pumps are running with the Demineralizer Outlet Valves closed.**
	BOP	[6] N/A [7] **ENSURE** 3-TIC-069-0010A, RWCU HEAT EXCHANGERS RBCCW FLOW CONTROL is in MANUAL, and FULL OPEN demand is on 3-TCV-70-49, RWCU NON-REGENERATIVE HEAT EXCHANGER OUTLET TCV. (REFER TO Attachment 5)
	Driver	**When directed to place 3-TIC-069-0010A in manual, insert Event 11. Inform the crew that 3-TIC-069-0010A is in manual and is fully open.**
	BOP	[8] **ENSURE CLOSED** the following: • 3-HC-69-15, RWCU BLOWDOWN PRESSURE CONTROL VALVE • 3-HS-69-16A, RWCU BLOWDOWN TO MAIN CONDENSER • 3-HS-69-17A, RWCU BLOWDOWN TO RADWASTE

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 3 of 9

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	[9] **ENSURE** 3-HS-69-15A, DEFEAT/OPERATE (FOR 3-HC-69-15) in the **DEFEAT** position. [10] N/A [11] **NOTIFY** Chemistry that RWCU is being placed in service and to check the durability monitor.
	Driver	**When contacted as Chemistry acknowledge the direction to check the durability monitor.**
	BOP	[12] **ENSURE OPEN** the following valves: • 3-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE • 3-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE • 3-FCV-69-8, RWCU DEMIN BYPASS VALVE [13] **OPEN** 3-FCV-069-0012, RWCU RETURN ISOLATION VALVE by one of the two methods described below. • **PLACE** 3-HS-69-12A in the OPEN position, **THEN** **RETURN** 3-HS-69-12A to the NORM position when intermediate position (red and green light) is indicated • **PLACE** 3-HS-69-12A in the OPEN position, **THEN** **RETURN** 3-HS-69-12A to the NORM position when **FULL OPEN** position (red light only) is indicated **NOTES** 1) Too high a flow on startup after isolation could cause 3-TIS-69-11 to actuate due to a high Non-Regenerative Heat Exchanger Outlet Temperature (3-XS-69-6, RWCU TEMP SELECT, Position 3, WATER TO RWCU DEMINS). 2) The RWCU Pump will trip on low flow at 56 gpm, after a 30 second time delay. Failure to immediately raise flow to greater than 56 gpm in the following steps will result in a pump trip. 14] **PLACE** seal purge in operation to pump(s) to be placed in service. **REFER TO** Section 8.3.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 4 of 9

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed to place seal purge in service, acknowledge the direction and inform the crew that seal purge has been placed in service in accordance with 3-OI-69, Section 8.2**
	BOP	[15] N/A [16] **START** RWCU RECIRC PUMP 3A(3B) using control switch 3-HS-69-4A(4B)-A, **AND RAISE** flow, using RWCU RETURN ISOLATION VALVE, 3-HS-69-12A, to prevent low flow trip. [17] **IF** two pump operation is desired, **THEN START** the second RWCU RECIRC PUMP 3B(3A) using control switch 3-HS-69-4B(4A)-A, AND RAISE flow using RWCU RETURN ISOLATION VALVE, 3-HS-69-12A, to prevent low flow trip. [18] **IF** the RWCU filter-demineralizers are to be placed in service, **THEN REFER TO** Section 6.2.
	BOP	3-OI-69, Reactor Water Cleanup System Section 6.2, Placing Filter-Demineralizers in Service **CAUTION** When initially placing a filter-demineralizer into service, it is desirable that most RWCU Discharge Flow be returned to the Main Condenser. If the Reactor is pressurized, however, failure to follow temperature restrictions could result in thermal shocking the Regenerative Heat Exchanger. [1] **REVIEW** Precautions and Limitations in Section 3.0. [2] – [10] Performed in the Field by an AUO.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 5 of 9

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	Driver	**When contacted as the Reactor Building AUO to prepare to roll in RWCU Demineralizer, acknowledge the direction and report that you are standing by with Steps complete Steps [2] through [10] of 3-OI-69, Reactor Water Cleanup System are complete.** **When directed to place filter-demineralizers in service, acknowledge the direction. Insert Event 1 to perform AUO actions to place demineralizers in service, and inform the crew that Demin Flow is rising.** **Demineralizers will roll in over a 1-minute time frame – when complete inform the crew that RWCU filter-demineralizers have been placed in service.**
	BOP	**NOTE** RWCU is required to be operated with the following restrictions with Reactor Pressure ≤ 50 psig (MODES 2 or 3) or any time the unit is in MODE 4, MODE 5, or defueled: • One pump in operation, pump can be operated to its maximum flow capacity. • Two pumps in operation, maximum flow limited to ≤ 100 gpm per pump (200 gpm total) [11] **PERFORM** the following simultaneously: • **CLOSE** 3-HS-69-8A, RWCU DEMIN BYPASS VALVE on Panel 3-9-4
	Driver	**Verify that the crew is able to clear RWCU Demineralizer Alarm (Panel 3-9-4B, Window 24) – it will automatically reset on Event 1 after 15 seconds. If the crew cannot get the alarm to reset, insert remote function AN01E to RESET again).**
	BOP	[12] **RAISE** flow through the Demin until the desired flow has been established. [13] **ENSURE** DEMIN 3A(3B) HOLDING PUMP, 3-HS-069-6015(6005), in the AUTO position.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 6 of 9

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	[14] **CHECK** that Holding Pump 3A(3B), on the Demin being placed in service, has STOPPED. [15] **CHECK** DEMIN 3A(3B) HOLDING PUMP DISCH VALVE H, 3-HS-069-0035B(0060B), has CLOSED.
	Driver	**When directed to perform Steps [12], [13], and [14] acknowledge the direction and inform the crew that Steps [12], [13], and [14] are complete.**
	BOP	[16] **NOTIFY** Chemistry that the filter-demineralizer has been placed in service and **REQUEST** a sample for conductivity and silica of the effluent.
	Driver	**When contacted as Chemistry, acknowledge any information or direction given.**
	BOP	Continuing 3-OI-69, Reactor Water Cleanup System Section 5.1 RWCU Pump Startup [19] **ADJUST** the RWCU RETURN ISOLATION VALVE, using 3-HS-69-12A, as required to obtain desired system flow. **CAUTIONS** 1) Failure to maintain RWCU Non-Regenerative Heat Exchanger Tube Side Outlet Temperature below 130 ºF will reduce resin efficiency and may result in resin damage. 2) Exercise care when making RWCU System Flow adjustments to values greater than 270 gpm to ensure temperature limits are not exceeded. [20] **THROTTLE** blowdown flow as required to maintain the following parameters. (REFER TO Section 6.5). • Desired Reactor Water Level • Non-Regenerative Heat Exchanger Tube Outlet Temperature less than 130 ºF

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 1 Page 7 of 9

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	[21] **IF** at Operations Management discretion it is desired to place 3-TIC-069-0010A, RWCU HEAT EXCHANGERS RBCCW FLOW CONTROL in AUTO, **THEN** PERFORM the following: (Otherwise N/A) [21.1] **PLACE** 3-TIC-069-0010A, RWCU HEAT EXCHANGERS RBCCW FLOW CONTROL in AUTO. (REFER TO Section 8.15.)
	Driver	**When contacted by the crew to place 3-TIC-069-0010A in automatic acknowledge the direction and insert Event 21. Inform the crew that 3-TIC-069-0010A has been placed in automatic.**
	BOP	**NOTES** 1) Seal water to the RWCU Pumps has been observed to slightly lower after pump(s) are placed in service. 2) When the Reactor Vessel is at atmospheric pressure and RWCU Pump seal water is being supplied by CS&S system, RWCU Pump seal water flow may decrease to 0 gpm after the RWCU Pump has started. See PRECAUTION P&L 3.6E. [22] **ENSURE** PURGE (SEAL) WATER TO RWCU PUMPS at Panel 3-25-314 (1.8 to 2.0 gpm). (REFER TO Section 8.3.) • 3-FI-085-0075, RWCU PUMP 3A PURGE WATER FLOW INDICATOR • 3-FI-085-0077, RWCU PUMP 3B SEAL WATER
	Driver	**When contacted as the Reactor Building AUO to perform Step [22], inform the crew that seal water flow is 1.9 gpm.**
	BOP	[23] **AFTER** RWCU system is in service perform section 5.2 RWCU ICS TEMPERATURE POINT RESTORATION if necessary. Otherwise N/A

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 1	Page 8 of 9

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	3-OI-69, Reactor Water Cleanup System Section 5.2, RWCU ICS Temperature Point Restoration [1] **ENTER** RTP in the yellow block at the top of the ICS display and depress Enter key to cause the RESTORE TO PROCESSING/RETURN TO SCAN screen to be displayed. [2] **ENTER** the following information in the entry blocks of the RESTORE TO PROCESSING/RETURN TO SCAN screen to restore point 69-6D to normal processing using the Tab key to switch to each entry block: A. In the Point ID field, **ENTER** 69-6D. B. In the Modified By field, **ENTER** your initials. C. In the Reason field, **ENTER** short description of reason (like "system started"). D. After all above entries made, then **DEPRESS** the F3 key (Execute) to implement the substitution. E. The INSERT VALUE screen will continue to be displayed. [3] **ENTER** the following information in the entry blocks of the RESTORE TO PROCESSING/RETURN TO SCAN screen to restore point 69-6A to normal processing using the Tab key to switch to each entry block: A. In the Point ID field, **ENTER** 69-6A. B. In the Modified By field, **ENTER** your initials. C. In the Reason field, **ENTER** the short description of reason (like "system started"). D. After all above entries made, then **DEPRESS** the F3 key (Execute) to implement the substitution. [4] **DEPRESS** Esc key to exit the RESTORE TO PROCESSING/RETURN TO SCAN screen.

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 1	Page 9 of 9

Event Description: Return Reactor Water Cleanup (RWCU) to Operation

Time	Position	Applicant's Actions or Behavior
	BOP	3-9-ARP-4B, Alarm Response Procedure RWCU NON-REGENERATIVE HX DISCHARGE TEMPERATURE HIGH, Window 17 Operator Action: A. **CHECK** RWCU NRHX Discharge Temperature, 3-XS-69-6, on Panel 3-9-4. B. **CHECK** RBCCW System Temperature indication normal, Panel 3-9-4. C. **IF** temperature continues to rise, **THEN PERFORM** the following, otherwise, **MARK** steps N/A: • **REDUCE** system flow or reject flow as necessary to control temperature • **REFER TO** 3-OI-69, Reactor Water Cleanup System D. **DISPATCH** personnel to CHECK the following: • RWCU Heat Exchangers RBCCW Flow Controller (normally in auto with setpoint at approx. 110 ºF), located on 3-LPNL-925-0002 Reactor Bldg 593' • 3-TCV-70-49 operating properly (RBCCW to NRHX), located in RWCU HX room
	Driver	**If contacted by the crew to check equipment in Step D (see above), acknowledge the direction and report the following as required:** • **RWCU Heat Exchangers RBCCW Flow Controller is set at 110 ºF and is in automatic** • **3-TCV-70-49 is operating properly**
	NRC	**End of Event 1. Proceed to Event 2, Reduce Reactor Power to 75% using Core Flow.**

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 2	Page 1 of 5

Event Description: Reduce Reactor Power to 75% using Core Flow

Time	Position	Applicant's Actions or Behavior
	NRC	**If the crew does not proceed to Event 2, Reduce Reactor Power to 75% using Core Flow, request that the Driver contact the NUSO as the Shift Manager and direct the crew to reduce Reactor Power to 75%.**
	Driver	**If requested by the Chief Examiner, contact the crew as the Shift Manager and direct the crew to continue with Step [2] of the Reactivity Control Plan (RCP) for the Reactor Shutdown and reduce Reactor Power to 75% using Core Flow.**
	NRC	**The crew may elect to conduct a re-focus reactivity brief prior to lowering Reactor Power.**
	NUSO	Directs the Operator at the Controls (OATC) to lower Reactor Power to 75% in accordance with Step [2] of RCP U3-2104NRC1 and in accordance with 3-OI-68, Reactor Recirculation System, 3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations and 3-GOI-100-12, Power Maneuvering.
	OATC	3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations Section 5.3, Power Reduction 5.3.1 Reducing Reactor Power to 40% [1] **ENSURE** the operators are using Attachment 9, Operations Down Power Monitoring. [2] **REDUCE** Reactor Power by a combination of Control Rod insertions and core flow changes, as recommended by Reactor Engineer.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 2 Page 2 of 5

Event Description: Reduce Reactor Power to 75% using Core Flow

Time	Position	Applicant's Actions or Behavior
	OATC	3-GOI-100-12, Power Maneuvering Section 5.0, Instruction Steps [7] **REDUCE** Reactor Power by a combination of Control Rod insertions and Core Flow changes, as recommended by Reactor Engineer. **REFER TO** 3-SR-3.1.3.5(A) and 3-OI-68. (N/A if entering 3-GOI-100-12, Power Maneuvering, to recover from Recirc Pump Trip).
	NRC	**3-OI-68, Reactor Recirculation System** **3.0 Precautions and Limitations** **Section 3.5.3, Dual Pump Operation** E. When raising (lowering) Reactor Power per the Reactivity Plan, the following guideline should be used to establish the 60 rpm mismatch between the Recirc Pumps. 1. When the first Recirc Pump reaches 1200 rpm (1300 rpm) or both Recirc Pumps are at 1200 rpm (1300 rpm) then individual controls will be used. 2. While following the Reactivity Plan establish the 60 rpm mismatch using the individual controls for the leading Recirc Pump. 3. While maintaining the 60 rpm mismatch and using the Reactivity Plan, raise (lower) the Recirc Pump speeds using either the Master Controllers or Individual Controllers. 4. Once a Recirc Pump reaches 1300 rpm (1200 rpm) the 60 rpm mismatch is no longer required. While following the Reactivity Plan raise (lower) the lagging Recirc Pump using the individual controller match Recirc Pump speeds. 5. Once the Recirc Pumps are matched, then the speeds may be adjusted as required using the Master Controller or Individual Controllers while maintaining the requirements of Attachment 1.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 2 Page 3 of 5

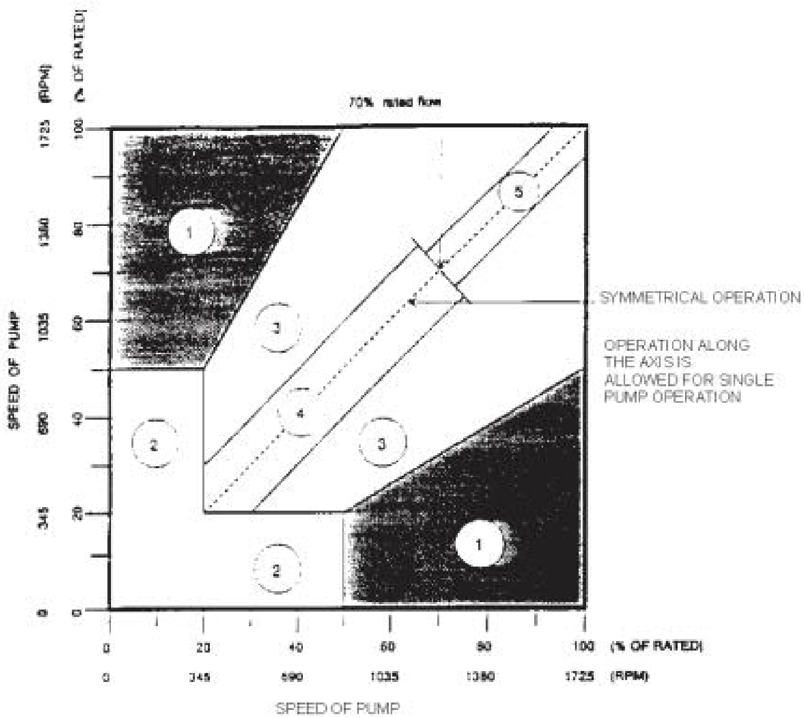
Event Description: Reduce Reactor Power to 75% using Core Flow

Time	Position	Applicant's Actions or Behavior
	NRC	

BFN Unit 3	Reactor Recirculation System	3-OI-68 Rev. 0099 Page 181 of 210

Attachment 1
(Page 1 of 1)
Recirculation Pump Speed Mismatch Curve

(for Steady State, Dual Pump Operation)



1. Avoid Region To Prevent Excessive Jet Pump Vibration.
2. Below Recirc Drive Minimum speed.
3. Operation allowed if reactor subcritical or during transient periods.
4. Limited Operation for Core flow ≤ 70% rated (mismatch ≤ 10% rated speed).
5. Limited operation for Core flow > 70% rated (mismatch ≤ 5% rated speed).

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 2	Page 4 of 5

Event Description: Reduce Reactor Power to 75% using Core Flow

Time	Position	Applicant's Actions or Behavior
		3-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown on 0-TI-248, Station Reactor Engineer and 3-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the later part of the operating cycle (Coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the Power to Flow Map. Instruments used to monitor pump speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing Core Flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area.
	OATC	[1] **IF** desired to control Recirc Pump 3A speed with Recirc Individual Control, **THEN PERFORM** the following; (Otherwise N/A) [1.1] N/A [1.2] Lower Recirc Pump 3A using 3-HS-96-17A(17B)(17C), SLOW (MEDIUM) (FAST). (Otherwise N/A) [2] **IF** desired to control Recirc Pump 3B speed with Recirc Individual Control, **THEN PERFORM** the following; (Otherwise N/A) [2.1] N/A [2.2] Lower Recirc Pump 3B using 3-HS-96-18A(18B)(18C), SLOW (MEDIUM) (FAST). (Otherwise N/A) [3] **WHEN** desired to control Recirc Pumps 3A and/or 3B speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump speed 3A & 3B using the following push buttons as required: 3-HS-96-33, LOWER SLOW 3-HS-96-34, LOWER MEDIUM 3-HS-96-35, LOWER FAST

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 2 Page 5 of 5

Event Description: Reduce Reactor Power to 75% using Core Flow

Time	Position	Applicant's Actions or Behavior
	NRC	**When satisfied with power reduction, end of Event 2. Request that the Driver insert Event 3, Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 3 Page 1 of 1

Event Description: Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3, Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low Level**
	BOP	Acknowledges and reports the following alarm to the NUSO: RBCCW SURGE TANK LEVEL LOW, 3-9-4C, Window 13
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 3-ARP-9-4C RBCCW SURGE TANK LEVEL LOW, Window 13 A. ADD water to the RBCCW Surge Tank for approximately one minute or until low level alarm resets using the following: • 3-FCV-70-1, RBCCW SYS SURGE TANK FILL VALVE (Panel 3-9-4) OR • 3-BYV-002-1369, FCV-70-1 BYPASS VALVE (locally) B. IF alarm does NOT reset, THEN CHECK tank locally. C. IF unable to maintain RBCCW Surge Tank level, THEN REFER TO 3-AOI-70-1, Loss of Reactor Building Closed Cooling Water. D. IF necessary to add water more than once per shift, THEN CHECK Drywell floor drain system for excessive operation AND INSPECT system outside Drywell for leakage.
	NRC	**The RBCCW Surge Tank Low Level alarm can be cleared 15 seconds after the fill valve is opened.**
	BOP	Opens 3-FCV-70-1, RBCCW SYS SURGE TANK FILL VALVE for approximately one minute and checks that RBCCW SURGE TANK LEVEL LOW 3-9-4C, Window 13 clears.
	Driver	**If contacted as the Reactor Building AUO to check for leaks or check RBCCW Surge Tank Level locally, acknowledge the direction. Wait 2 minutes and report that Surge Tank Level is normal.**
	NRC	**End of Event 3. Request that the Driver insert Event 4, Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close.**

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 4	Page 1 of x
Event Description:			

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4, Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close.**
	NRC	**3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION VALVE, will not automatically close on the Primary Containment System Isolation (PCIS) signal, but can be closed manually.**
	OATC/ BOP	Acknowledges and reports the following alarms: • RWCU LEAK DETECTION TEMP HIGH, 3-9-3D, Window 17 • RWCU ISOL LOGIC CHANNEL A TEMP HIGH, 3-9-5B, Window 32 • RWCU ISOL LOGIC CHANNEL B TEMP HIGH, 3-9-5B, Window 33
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures.
	BOP	3-ARP-9-3D, Alarm Response Procedure RWCU LEAK DETECTION TEMP HIGH, Window 17 Operator Action: A. IF this alarm is received in conjunction with RWCU ISOL LOGIC CHANNEL A TEMP HIGH [3-XA-55-5B, window 32] and RWCU ISOL LOGIC CHANNEL B TEMP HIGH [3-XA-55-5B, window 33], THEN EXIT this procedure and GO TO 3-ARP-9-5B. Otherwise, CONTINUE in this procedure.
	BOP	Exits 3-ARP-9-3D, Alarm Response Procedure, and enters 3-ARP-9-5B, Alarm Response Procedure.
	BOP	3-ARP-9-5B, Alarm Response Procedure RWCU ISOL LOGIC CHANNEL A TEMP HIGH, Window 32 Operator Action: A. **CHECK** alarm by checking: 1. ATUs on Panel 3-9-83 and 3-9-85.

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 4	Page 2 of x
Event Description:			

Time	Position	Applicant's Actions or Behavior
	BOP	2. RWCU LEAK DETECTION TEMP HIGH annunciator in alarm (3-XA-55-3D, Window 17). 3. Area temperature indication on 3-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE, on Panel 3-9-21. 4. Area Radiation Monitors (ARMs) 3-RR-90-1, 3-CONS-90-50A on Panel 3-9-2 and 0-CONS-90-361 on Panel 1-9-2. 5. ICS 'HPTURB' & 'RWCU' mimics for the 834 and 835 temperature loops. B. **IF** leak is suspected, **THEN MANUALLY ISOLATE** RWCU or if RWCU automatically isolates, **REFER TO** 3-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation.
	Driver	**If contacted as Unit 1 Operator to check Area Radiation Monitors or Radiation Recorders, acknowledge the request.**
	BOP	Determines that 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION VALVE, failed to automatically isolate and manually closes 3-FCV-69-2. Informs the NUSO.
	BOP	C. **IF** TIS-69-835A(C) indicates greater than 131 ºF, **THEN ENTER** 3-EOI-3, Secondary Containment Control. D. **REFER TO** Tech. Spec. Table 3.3.6.1-1, Primary Containment Isolation Instrumentation. E. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	NUSO	As required by 3-ARP-9-5B, Window 32 and 3-ARP-9-5B, Window 33, references NPG-SPP-18.3.5, Equipment Important to Emergency Response. No actions are required in accordance with this procedure.
	NRC	**Technical Specifications are covered starting on page xx of xx.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 3 of x

Event Description:

Time	Position	Applicant's Actions or Behavior
	BOP	3-ARP-9-5B, Alarm Response Procedure RWCU ISOL LOGIC CHANNEL B TEMP HIGH, Window 33 Operator Action: A. **CHECK** alarm by checking: 1. ATUs on Panel 3-9-84 and 3-9-86. 2. RWCU LEAK DETECTION TEMP HIGH annunciator in alarm (3-XA-55-3D, Window 17). 3. Area temperature indications on LEAK DETECTION SYSTEM TEMPERATURE, 3-TR-69-29, on Panel 3-9-21. 4. ARMs 3-RR-90-1, 3-CONS-90-50A on Panel 3-9-2 and 0-MON-90-361 on Panel 1-9-2. 5. ICS 'HPTURB' & 'RWCU' mimics for the 834 and 835 temperature loops. B. **IF** a leak is suspected, **THEN MANUALLY ISOLATE** RWCU or if RWCU automatically isolates, **REFER TO** 3-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation. C. **IF** TIS-69-835B(D) indicates greater than 131 ºF, **THEN ENTER** 3-EOI-3, Secondary Containment Control. D. **REFER TO** Tech. Spec. Table 3.3.6.1-1, Primary Containment Isolation Instrumentation. E. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	NRC	**No actions are required in accordance with Technical Specification 3.3.6.1.**
	BOP	3- AOI-64-2A, Group 3 Reactor Water Cleanup Isolation Immediate Actions [1] **ENSURE** automatic actions occur.

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 4	Page 4 of x
Event Description:			

Time	Position	Applicant's Actions or Behavior
	BOP	Automatic Actions: • 3-FCV-69-1, RWCU INBD SUCTION ISOLATION VALVE CLOSES • 3-FCV-69-2, RWCU OUTBD SUCTION ISOLATION VALVE CLOSES • 3-FCV-69-12, RWCU RETURN ISOLATION VALVE CLOSES • Reactor Water Cleanup Recirc Pumps 3A and 3B TRIP Subsequent Actions: [1] **IF** any EOI entry condition is met, **THEN ENTER** appropriate EOI(s).
	NRC	**The NUSO may enter 3-EOI-3, Secondary Containment Control, if Area Temperature or Radiation exceeds the Maximum Normal value. See page xx of xx for 3-EOI-2 actions.**
	Driver	**If contacted as an AUO to check ATUs, acknowledge the direction. Wait 3 minutes and report that ATUs 3-TIS-69-835A-D indicate 160 degrees and lowering.**
	BOP	[2] **CHECK** the following to confirm high area temperature condition exists: • 3-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE (Panel 3-9-21) • ATUs in Auxiliary Instrument Room [3] **IF** isolation is caused by high area temperature, **THEN DETERMINE** if a line break exists by: • RWCU ARMs 3-RI-90-9A, 13A, and 14A • Visual Observation • Rx Zone Exhaust Rad Monitors 3-RE-90-142A, 142B, 143A, and 143B [4] **PERFORM** necessary Heat Balance adjustments. **REFER TO** 3-OI-69, Reactor Water Cleanup System.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 5 of x

Event Description:

Time	Position	Applicant's Actions or Behavior
	BOP	[5] **CHECK** the following monitors for a rise in activity: • 3-RR-90-1, AREA RADIATION, Points 9, 13, and 14 (Panel 3-9-2) • 3-MON-90-50, AIR PARTICULATE MONITOR CONSOLE, 3-RM-90-55 and 57 (Panel 3-9-2) • RB, TB, and Refuel Zone Exhaust Rad on 0-MON-90-361, CHEMISTRY CAM, MONITOR CONTROLLER, (Panel 1-9-2) [6] **IF** it has been determined that leakage is the cause of the isolation, **THEN NOTIFY** RADCON of RWCU status. **NOTE** Based on recent On-Shift Analysis from the License Amendment, chemistry sampling may be delayed following plant events. In the case of a REP event, Chemistry will be required to assist with dose assessments. Dose assessments will have precedence over sampling actions in REP events. [7] **NOTIFY** Chemistry that RWCU has been removed from service for the following evaluations: • The need to begin sampling Reactor Water • The need to remove the Durability Monitor from service [8] **IF** the isolation cannot be reset, **THEN PERFORM** the following: [8.1] **ISOLATE** the CRD System by closing the following seal water valves in the Unit 3 Reactor Building Elevation 593: • 3-SHV-069-0592 (A pump) • 3-SHV-069-0614 (B pump) [8.2] **REFER TO** 3-OI-68, Reactor Recirculation System for Recirc System operating restrictions while RWCU is isolated.
	Driver	**If contacted as Radiation Protection or Chemistry acknowledge any directions or reports given.** **If contacted as Unit 1 to check Reactor Building, Turbine Building, and Refuel Zone Exhaust Radiation on 0-MON-90-361, CHEMISTRY CAM MONITOR CONTROLLER (Panel 1-9-2), acknowledge the direction.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 6 of x

Event Description:

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td colspan="2">[9] EVALUATE Technical Requirements Manual Section 3.4.1, Coolant Chemistry, for limiting conditions for operation.</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">Technical Requirements Manual 3.4.1, Coolant Chemistry.
LCO 3.4.1 Reactor Coolant Chemistry shall be maintained within the limits of Table 3.4.1-1.
APPLICABILITY: According to Table 3.4.1-1</td></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
A.1 – Verify by administrative means that conductivity has not been > 1.0 μmho/cm at 25°C for > 2 weeks in the past year.

B.1 – Verify by administrative means that chloride concentration has not been > 0.2 ppm for > 2 weeks in the past year.</td><td>COMPLETION TIME:
A.1 – Immediately

B.1 – Immediately</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">Table 3.4.1-1
Coolant Chemistry Limits(1)
<table>
<tr><th>CHEMISTRY PARAMETERS</th><th>COLUMN A
APPLICABLE CONDITION
Prior To Startup And At Steaming Rates < 100,000 lb/hr</th><th>COLUMN B
APPLICABLE CONDITION
Steaming Rates > 100,000 lb/hr</th><th>COLUMN C
APPLICABLE CONDITION
Reactor Not Pressurized With Fuel In Reactor Vessel, Except During Startup Condition</th><th>COLUMN D(2)
APPLICABLE CONDITION
Noble Metal Chemical Application and Subsequent Reactor Coolant Cleanup</th><th>COLUMN E(3)
APPLICABLE CONDITION
Operation of HWC Following Noble Metal Chemical Application</th></tr>
<tr><td>CHLORIDE (ppm)</td><td>≤ 0.1</td><td>≤ 0.2</td><td>≤ 0.5</td><td>≤ 0.1</td><td>≤ 0.2</td></tr>
<tr><td>CONDUCTIVITY (μmho/cm at 25°C)</td><td>≤ 2.0</td><td>≤ 1.0</td><td>≤ 10.0</td><td>≤ 20.0</td><td>≤ 2.0</td></tr>
<tr><td>pH</td><td>5.6-8.6</td><td>5.6-8.6</td><td>5.3-8.6</td><td>4.3-9.9</td><td>5.6-8.8</td></tr>
</table>
(1) When there is no fuel in the reactor vessel, Technical Requirement reactor coolant chemistry limits do not apply.
(2) During the Noble Metal Chemical Application and subsequent reactor coolant cleanup, CONDITIONS A, B, C, and D (including Required Actions and Completion Times) do not apply.
(3) During operation of HWC following the Noble Metal Chemical Application, CONDITION A (including Required Action and Completion Time) does not apply.</td></tr>
<tr><td></td><td>Driver</td><td colspan="2">If contacted as Chemistry to verify by administrative means that conductivity and chloride concentration have not exceeded Table 3.4.1-1 limits for >2 weeks in the past year, inform the NUSO that chemistry limits have not exceeded Table 3.4.1-1 limits in the past year.</td></tr>
</table>

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 4	Page 7 of x
Event Description:			

Time	Position	Applicant's Actions or Behavior
	NUSO	Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs) LCO 3.6.1.3 Each PCIV, except Reactor Building-to-Suppression Chamber Vacuum Breakers, shall be OPERABLE APPLICABILITY: MODES 1, 2, and 3 When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation." NOTES 1. Penetration flow paths except for 18 and 20 inch purge valve penetration flow paths may be un-isolated intermittently under administrative controls. 2. Separate Condition entry is allowed for each penetration flow path. 3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs. 4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3. **CONDITION:** NOTE: Only applicable to penetration flow paths with two PCIVs. A. – One or more penetration flow paths with one PCIV inoperable except due to MSIV leakage not within limits.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: x Page 8 of x

Event Description:

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
A.1 – Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured

<u>AND</u>

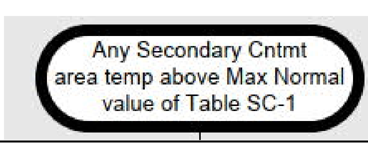
NOTE: Isolation devices in High Radiation Areas may be verified by use of administrative means.

A.2 – Verify the affected penetration flow path is isolated</td><td>COMPLETION TIME:
A.1 – 4 hours except for Main Steam Line

A.2 – Once per 31 days for isolation devices outside Primary Containment

<u>AND</u>

Prior to entering MODE 2 or 3 from MODE 4, if Primary Containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside Primary Containment</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">If RWCU Room Temperature exceeds the Maximum Normal 3-EOI-3, Secondary Containment Control


</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 9 of x

Event Description:

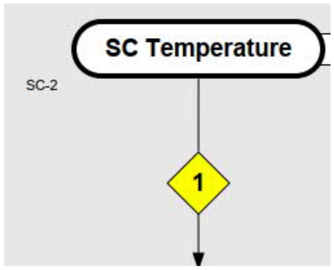
Time	Position	Applicant's Actions or Behavior
	NUSO	3-EOI-3, Secondary Containment Control SC-1

IF	THEN
Reactor Zone Ventilation Exhaust Radiation level is above 72 mR/hr	**NO ACTION REQUIRED**
Refuel Zone Ventilation Exhaust Radiation level is above 72 mR/hr	**NO ACTION REQUIRED**
Reactor Zone Ventilation is isolated AND Reactor Zone Ventilation Exhaust Radiation level is below 72 mR/hr	**NO ACTION REQUIRED**
Refuel Zone Ventilation is isolated AND Refuel Zone Ventilation Exhaust Radiation level is below 72 mR/hr	**NO ACTION REQUIRED**

Op Test No.: 21-04   Scenario No. NRC-1   Event No.: 4   Page 10 of xx

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	NUSO	(see below)



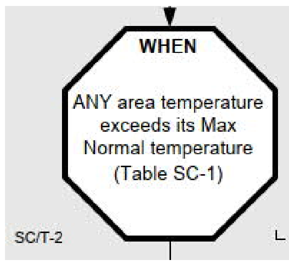
1

- An RPV water lvl instrument may be used to determine or trend lvl only when it reads above the Minimum Indicated Lvl associated with the highest max DW or SC run temp
- If DW temps or SC area temps (Table 6), as applicable, are outside the safe region of Curve 8, the associated instrument may be unreliable due to boiling in the run

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A/B	Emergency -155 to +60	on scale	N/A	below 100
		-150	N/A	101 to 150
		-145	N/A	151 to 200
		-140	N/A	201 to 250
		-130	N/A	251 to 300
		-120	N/A	301 to 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	Normal 0 to +60	on scale	N/A	below 150
		+5	N/A	151 to 200
		+15	N/A	201 to 250
		+20	N/A	251 to 300
		+30	N/A	301 to 350
LI-3-52 LI-3-62A	Post Accident -268 to +32	on scale	N/A	N/A
LI-3-55	Shutdown Floodup 0 to +500	+10	Below 100	N/A
		+15	100 to 150	N/A
		+20	151 to 200	N/A
		+30	201 to 250	N/A
		+40	251 to 300	N/A
		+50	301 to 350	N/A
		+65	351 to 400	N/A

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 11 of xx

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	NUSO	SC/T-1 **IF** Reactor Zone or Refuel Zone Ventilation Exhaust Radiation Level is below 72 mr/hr **THEN** operate available Reactor Zone or Refuel Zone Ventilation  3 SC-3 ISOLATE all systems that are discharging into the area EXCEPT systems required: • For damage control OR • To be operated by EOIs NOTE 3 — Tables SC-1 and SC-2 contain information that may be used to determine if a primary system is discharging into Secondary Containment (Emergency Depressurization will reduce discharge).
	NRC	**When the RWCU Leak has been isolated and Area Temperature and Radiation is below the Maximum Safe value, the NUSO may contact the Shift Manager and recommend exiting 3-EOI-2, Secondary Containment Control, as an emergency no longer exists.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 4 Page 12 of 12

Event Description: Reactor Water Cleanup (RWCU) Leak / One PCIV Fails to Close

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Shift Manager by the NUSO to discuss exiting 3-EOI-3, Secondary Containment Control, agree with any recommendation given.**
	NRC	**End of Event 4. Request that the Driver insert Event 5, Core Spray Loop I Room Cooler EECW Leak.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 5 Page 1 of 2

Event Description: Core Spray Loop I Room Cooler EECW Leak

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>Driver</td><td colspan="2">When requested by the Chief Examiner, insert Event 5, Core Spray Loop I Room Cooler EECW Leak. Contact the NUSO as the Reactor Building Assistant Unit Operator (AUO) and report that you discovered and isolated a water leak in the Core Loop 1 Room Cooler. Report that the following valves were closed to isolate the leak:
• 3-SHV-67-550, NW Core Spray Room Cooler Supply Shutoff
• 3-SHV-67-553, NW Core Spray Room Cooler Outlet

If asked, the water seems to have stopped leaking.</td></tr>
<tr><td></td><td>Driver</td><td colspan="2">If contacted as Work Control or Mechanical Maintenance, acknowledge any direction concerning the Core Spray Loop I Room Cooler.</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">Technical Requirements Manual 3.5.3, Equipment Area Coolers
LCO 3.5.3 The Equipment Area Cooler associated with each RHR Pump and the Equipment Area Cooler associated with each set of Core Spray Pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
APPLICABILITY: Whenever the associated subsystem is required to be OPERABLE

CONDITION:
A. – One or more Equipment Area Cooler inoperable</td></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
A.1 – Declare the pump(s) served by that cooler INOPERABLE (Refer to applicable Tech Spec and TRM LCOs)</td><td>COMPLETION TIME:
A.1 – Immediately</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 5 Page 2 of 2

Event Description: Core Spray Loop I Room Cooler EECW Leak

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.5.1, ECCS – Operating LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE APPLICABILITY: MODE 1, MODES 2 and 3, except High Pressure Coolant Injection (HPCI) and ADS valves are not required to be OPERABLE with Reactor Steam Dome Pressure ≤150 psig **CONDITION:** A. – One low pressure ECCS injection/spray subsystem inoperable. <u>OR</u> One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	
	NUSO	**REQUIRED ACTION:** See Condition F	**COMPLETION TIME:** See Condition F
	NUSO	**CONDITION:** F. – One ADS Valve inoperable <u>AND</u> Condition A entered	
	NUSO	**REQUIRED ACTION:** F.1 – Restore ADS Valve to OPERABLE status <u>OR</u> F.2 – Restore Low Pressure ECCS Injection / Spray subsystem to OPERABLE status	**COMPLETION TIME:** F.1 – 72 hours F.2 – 72 hours
	NRC	**End of Event 5. Request that the Driver insert Event 6, 3C 4KV Unit Board Trip.**	

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 6	Page 1 of 3

Event Description: 3C 4KV Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6, 3C 4KV Unit Board Trip.**
	BOP	Acknowledges and reports the following alarms: • 4KV UNIT BOARD 3C UNDERVOLTAGE, 3-9-8B, Window 14 • CONDENSATE BOOSTER PUMP C AUX OIL PRESSURE LOW, 3-9-6A, Window 14 • MOTOR TRIPOUT, 3-9-8C, Window 33
	CREW	Monitors Reactor Water Level.
	OATC	Reports a loss of Control Rod Drive (CRD) System Flow due to 3A CRD Pump being de-energized.
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures and direct the OATC to respond in accordance with 3-AOI-85-3, CRD System Failure.
	OATC	3-AOI-85-3, CRD System Failure Immediate Actions [1] **IF** operating CRD PUMP has failed **AND** the standby CRD Pump is available, **THEN PERFORM** the following at Panel 3-9-5: (Otherwise N/A) [1.1] **PLACE** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL in MAN at minimum setting. [1.2] **START** associated standby CRD Pump using the following: • 3-HS-85-2A, CRD PUMP 3B [1.3] **ADJUST** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, to establish the following conditions: • 3-PDI-85-18A, CRD COOLING WATER HEADER DP, between 10 psid and 20 psid • 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, between 40 and 65 gpm [1.4] **BALANCE** CRD SYSTEM FLOW CONTROL, 3-FIC-85-11, and **PLACE** in AUTO or BALANCE.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 6 Page 2 of 3

Event Description: 3C 4KV Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	3-ARP-9-8B, Alarm Response Procedure 4KV UNIT BOARD 3C UNDERVOLTAGE, Window 14 Operator Action: A. **CHECK** Unit in stable condition by checking: • Condensate Pump 3C • Condensate Booster Pump 3C • RCW Pump 3C • CCW Pump 3C • CRD Pump 3A B. **IF** undervoltage has occurred, **THEN** 1. **CLEAR** disagreement lights on breakers. 2. **REDUCE** load as necessary to maintain stable operating conditions. 3. Condenser discharge may need to be throttled for two CCW pump operation. **REFER TO** 3-OI-27, Condenser Circulating Water System. C. **CHECK** Unit Bd 3C for abnormal conditions: relay targets, smoke, burned paint, etc. D. **REFER TO** 0-OI-57A, Switchyard and 4160V AC Electrical System, to re-energize board. E. **REFER TO** appropriate OI for recovery or realignment of equipment.
	Driver	**If contacted as an AUO, Work Control, or Electrical Maintenance to investigate, acknowledge the direction. If directed to prepare protected equipment tags acknowledge the direction.** **Wait 3 minutes and report that 3C 4KV Unit Board has an overcurrent trip flag.**

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 6	Page 3 of 3

Event Description: 3C 4KV Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	3-ARP-9-8C, Alarm Response Procedure MOTOR TRIPOUT, Window 33 Operator Action: A. **CHECK** Control Room for white disagreement light illuminated for affected equipment. B. **CLEAR** disagreement light. C. **DISPATCH** personnel to CHECK: 1. Relays at associated electrical board. 2. Equipment for abnormal conditions. 3. Safe-stop locally reset, if necessary. D. **REFER TO** 0-GOI-300-2, Electrical, if relay targets are present or for motor starting limits. E. **REFER TO** appropriate OI for recovery or realignment of equipment.
	BOP	3-ARP-9-6A, Alarm Response Procedure CONDENSATE BOOSTER PUMP C AUX OIL PUMP PRESS LOW, Window 14 Operator Action: A. **DISPATCH** personnel to check booster pump lube oil system: 1. **ENSURE** running or start Aux Oil Pump. 2. **CHECK** for leaks. 3. **CHECK** oil level and temperature at reservoir. 4. **ROTATE** Cuno Filter.
	Driver	**If contacted as the Turbine Building AUO to start 3C Condensate Booster Pump Aux Oil Pump, insert Event 16 and report that the Aux Oil Pump is running.**
	NRC	**End of Event 6. Request that the Driver insert Event 7 Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 1 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7, Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak.**
	NRC	**Event 8, Fuel Damage, and Event 9, Reactor Feedwater Pumps Trip / HPCI Fails to Automatically Start and Inject, are automatically entered by the simulator setup. No action is required by the Driver to Insert Event 8 or Event 9. See page xx of xx for Event 8 actions and page xx of xx for Event 9 actions.**
	BOP	Acknowledges and reports the following alarms as they are received: • REACTOR BUILDING RADIATION HIGH, 3-9-3A, Window 22 • REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH, 3-9-3A, Window 4 • RCIC STEAM LINE LEAK DETECTION TEMPERATURE HIGH, 3-9-3D, Window 10
	NRC	**See Event 8 (page xx of xx) for actions for Radiation Alarms.**
	NUSO	Directs the BOP to respond in accordance with the applicable Alarm Response Procedure.
	BOP	3-9-ARP-3D, Alarm Response Procedure RCIC STEAM LINE LEAK DETECTION TEMPERATURE HIGH, Window 10 Operator Action: A. **CHECK** RCIC temperature elements on LEAK DETECTION SYSTEM TEMPERATURE recorder, 3-TR-69-29 on Panel 3-9-21.
	BOP	Checks Area Temperatures on Panel 3-9-22.
	BOP	B. **IF** RCIC is **NOT** in service **AND** 3-FI-71-1A(B), RCIC STEAM FLOW indicates flow, **THEN ISOLATE** RCIC and **CHECK** temperatures lowering.
	BOP	Determines that RCIC failed to automatically isolate, and attempts to manually isolate RCIC. Informs the NUSO that RCIC will not isolate.
	BOP	C. IF high temperature is confirmed, THEN ENTER 3-EOI-3, Secondary Containment Control.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 2 of 11

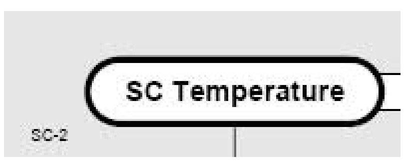
Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	BOP	Confirms that Area Temperatures are rising and informs the NUSO.
	NUSO	Enters 3-EOI-3, Secondary Containment Control. Directs the BOP to monitor Secondary Containment parameters.
	BOP	D. CHECK CS/RCIC ROOM El 519 RX BLDG radiation indicator, 3-RI-90-26A on Panel 3-9-11 and NOTIFY RADCON if rising radiation levels are observed. E. **DISPATCH** personnel to investigate.
	Driver	**If contacted as Radiation Protection that radiation levels are rising, acknowledge the report.** **If contacted as the Reactor Building AUO to investigate, acknowledge the direction.**
	NUSO	F. **REFER TO** Tech Specs 3.3.6.1, Primary Containment Isolation Instrumentation and 3.5.3, RCIC System.
	NRC	**Technical Specification evaluation for this event is not required and should not be used to evaluate the candidate's Technical Specification competency.**
		G. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	NRC	**It is not expected that the SRO reference NPG-SPP-18.3.5, Equipment Important to Emergency Response, during this event.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 3 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
3-EOI-3, Secondary Containment Control

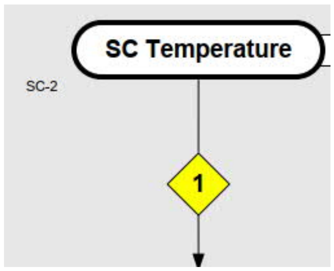
SC-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Reactor Zone Ventilation Exhaust Radiation level is above 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Refuel Zone Ventilation Exhaust Radiation level is above 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Reactor Zone Ventilation is isolated
AND
Reactor Zone Ventilation Exhaust Radiation level is below 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Refuel Zone Ventilation is isolated
AND
Refuel Zone Ventilation Exhaust Radiation level is below 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
</table>


</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 4 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	



1

- An RPV water lvl instrument may be used to determine or trend lvl only when it reads above the Minimum Indicated Lvl associated with the highest max DW or SC run temp
- If DW temps or SC area temps (Table 6), as applicable, are outside the safe region of Curve 8, the associated instrument may be unreliable due to boiling in the run

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A/B	Emergency -155 to +60	on scale	N/A	below 100
		-150	N/A	101 to 150
		-145	N/A	151 to 200
		-140	N/A	201 to 250
		-130	N/A	251 to 300
		-120	N/A	301 to 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	Normal 0 to +60	on scale	N/A	below 150
		+5	N/A	151 to 200
		+15	N/A	201 to 250
		+20	N/A	251 to 300
		+30	N/A	301 to 350
LI-3-52 LI-3-62A	Post Accident -268 to +32	on scale	N/A	N/A
LI-3-55	Shutdown Floodup 0 to +500	+10	Below 100	N/A
		+15	100 to 150	N/A
		+20	151 to 200	N/A
		+30	201 to 250	N/A
		+40	251 to 300	N/A
		+50	301 to 350	N/A
		+65	351 to 400	N/A

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 5 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	SC/T-1

IF	Reactor Zone or Refuel Zone Ventilation Exhaust Radiation level is below 72 mR/hr
THEN	Operate available Reactor Zone or Refuel Zone Ventilation

SC/T-2

WHEN

ANY Area Temperature exceeds its Max Normal Temperature (Table SC-1)

Table SC-1
Secondary Cntmt Area Temp

Area	Panel 9-3 Alarm Window (unless noted)	Panel 9-22 Temp Element (unless noted)	Max Normal Value °F	Max Safe Value °F	Potential Isolation Sources
RHR sys I pumps	XA-55-3E-4	74-95A	Alarmed	150	FCV-74-47, 48
RHR sys II pumps	XA-55-3E-4	74-95B	Alarmed	210	FCV-74-47, 48
HPCI room	XA-55-3F-10	73-55A	Alarmed	270	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	XA-55-3D-10	71-41A	Alarmed	190	FCV-71-2, 3, 39
Top of torus	XA-55-3D-10 XA-55-3F-10 XA-55-3E-4	71-41B, C, D 73-55B, C, D 74-95G, H	Alarmed Alarmed Alarmed	200 240 240	FCV-71, 2, 3 FCV-73-2, 3, 81 FCV-74-47, 48
Steam tunnel (RB)	XA-55-3D-24	1-60A (Panel 9-3)	Alarmed	315	MSIVs FCV-71-2, 3, FCV-69-1, 2, 12
DW access	XA-55-3E-4	74-95E	Alarmed	170	FCV-74-47, 48
RB el 565 W (RWCU pipe trench)	XA-55-5B-32 (Panel 9-5) XA-55-5B-33 (Panel 9-5)	69-835A, B, C, D (Aux Inst room)	Alarmed	170	FCV-69-1, 2, 12
RWCU hx room	XA-55-3D-17	69-29F, G, H	Alarmed	220	FCV-69-1, 2, 12
RWCU pump A	XA-55-3D-17	69-29D	Alarmed	215	FCV-69-1, 2, 12
RWCU pump B	XA-55-3D-17	69-29E	Alarmed	215	FCV-69-1, 2, 12
RB el 593	XA-55-3E-4	74-95C, D	Alarmed	195	FCV-74-47, 48
RB el 621	XA-55-3E-4	74-95F	Alarmed	155	FCV-43-13, 14

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 6 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	SC-3 (see below)

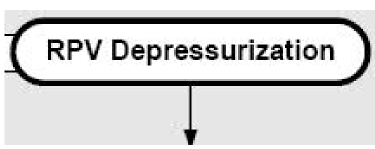
3

SC-3

ISOLATE all systems that are discharging into the area EXCEPT systems required: • For damage control OR • To be operated by EOIs

NOTE	
3	Tables SC-1 and SC-2 contain information that may be used to determine if a primary system is discharging into Secondary Containment (Emergency Depressurization will reduce discharge).

SC-4



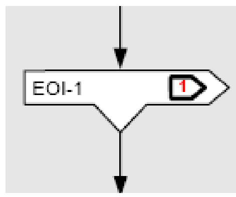
SC-7

3

WHEN
A Primary System is discharging into Secondary Containment

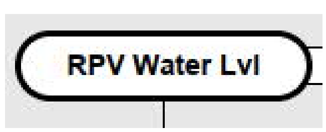
Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 7 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	SC-8 **BEFORE** ANY Secondary Containment parameter reaches its Max Safe Value (Tables SC-1, SC-2, and SC-3) 
	CREW	**Critical Task:** **With the Reactor at power and with a Primary System discharging into the Secondary Containment, manually SCRAM the Reactor before any area exceeds the Maximum Safe Temperature operating value.** **Critical Task Failure Criteria:** **The operating crew fails to proceed without delay and in a controlled manner to initiate a Reactor SCRAM from the time it is announced that one Area Temperature is approaching the Maximum Safe value.**
	NUSO	Enters 3-EOI-1, RPV Control. Directs the crew to enter 3-AOI-100-1, Reactor SCRAM, and directs the OATC to insert a manual Reactor SCRAM.
	NRC	**Event 8, Fuel Damage, and Event 9, Reactor Feedwater Pumps Trip / HPCI Fails to Automatically Initiate, are inserted when the Reactor MODE SWITCH is placed in SHUTDOWN.**
	OATC	Inserts a manual Reactor SCRAM.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 8 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	OATC	3-AOI-100-1, Reactor SCRAM Immediate Actions [1] **DEPRESS** 3-HS-99-5A/S3A, REACTOR SCRAM A and 3-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 3-9-5. [2] **PLACE** 3-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] **IF** all Control Rods can NOT be verified fully inserted, **THEN** **INITIATE** ARI. (Otherwise MARK N/A). [4] **IF** Reactor Power is 5% or BELOW, **THEN**: (Otherwise **MARK** N/A) **REPORT** the following to the UNIT SRO: • Reactor Scram • Mode Switch is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Power level [5] N/A
	OATC	Determines that all Reactor Feedwater Pumps (RFPTs) have tripped and informs the NUSO (See Event 9).
	NUSO	3-EOI-1, RPV Control  RC/L-1 **ENSURE** each as required: • PCIS isolations (Groups 1, 2, and 3) • ECCS • RCIC

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 9 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	RC/L-2

IF	THEN
RPV Water Level can be restored and maintained above (-)162 in. AND The ADS timer has initiated	**INHIBIT** ADS
Loss of available injection systems is anticipated OR Raising RPV Water Level above (+)51 in. will facilitate use of shutdown cooling, steam-driven injection systems, or Alternate Injection Subsystems (Table L-2)	**NO ACTION REQUIRED**

RC/L-3

RESTORE and **MAINTAIN** RPV Water Level between (+)2 in. and (+)51 in. with ANY Preferred Injection Systems (Table L-1)

IF	THEN
RPV Water Level cannot be restored and maintained between (+)2 in. and (+)51 in.	**NO ACTION REQUIRED**
RPV Water Level cannot be restored and maintained above (-)162 in.	**NO ACTION REQUIRED**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 10 of 11

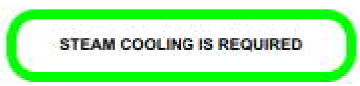
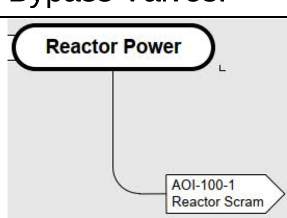
Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>Directs the OATC/BOP to maintain Reactor Water Level using HPCI in accordance with 3-EOI-Appendix-5D, Injection System Lineup HPCI.</td></tr>
<tr><td></td><td>NRC</td><td>3- EOI-Appendix-5D, Injection System Lineup HPCI actions are covered in Event 9. See page xx of xx.</td></tr>
<tr><td></td><td>NUSO</td><td>
<table>
<tr><th colspan="3">Table L-1
Preferred Injection Systems</th></tr>
<tr><th>SOURCES</th><th>APPX</th><th>INJ PRESS</th></tr>
<tr><td>CNDS and FW</td><td>5A</td><td>1210 psig</td></tr>
<tr><td>CRD</td><td>5B</td><td>1640 psig</td></tr>
<tr><td>RCIC with CST suction if available 2 3 6</td><td>5C, 20M</td><td>1200 psig</td></tr>
<tr><td>HPCI with CST suction if available 2 6 7</td><td>5D, 20N</td><td>1200 psig</td></tr>
<tr><td>CNDS</td><td>6A</td><td>480 psig</td></tr>
<tr><td>CS 2</td><td>6D, 6E</td><td>330 psig</td></tr>
<tr><td>LPCI 2</td><td>6B, 6C</td><td>320 psig</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>
RPV Press

RC/P-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>A high Drywell Pressure ECCS signal exists (2.45 psig)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>EMERGENCY RPV DEPRESSURIZATION is REQUIRED or has been required</td><td>2 C2 Emergency RPV Depressurization</td></tr>
<tr><td>Emergency RPV Depressurization is anticipated</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC-P/2
<table>
<tr><td>IF ANY MSRV is cycling
THEN NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 7 Page 11 of 11

Event Description: Un-isolable Reactor Core Isolation Cooling (RCIC) Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
RC/P-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Suppression Pool Temperature and Level CANNOT be maintained in a safe area of Curve 3 at the existing RPV Pressure</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Pool Level CANNOT be maintained in the safe area of Curve 4</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>
</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-4
<table>
<tr><td colspan="2">STABILIZE RPV Pressure below 1073 psig using the Main Turbine Bypass Valves (APPX 8B)
➢ OK to use ANY Alternate RPV Pressure Control Systems (Table P-1)
➢ Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>DW Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to maintain Reactor Pressure using the Main Turbine Bypass Valves.</td></tr>
<tr><td></td><td>NUSO</td><td>
</td></tr>
<tr><td></td><td>NRC</td><td>End of Event 7. Continue to Event 8.</td></tr>
</table>

Op Test No.: 21-04	Scenario No. NRC-1	Event No.: 8	Page 1 of 7
Event Description: Fuel Damage			

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8, Fuel Damage, is inserted when the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required by the driver to insert Event 8.**
	BOP	Acknowledges and reports the following alarms to the NUSO as they are received: • REACTOR BUILDING RADIATION HIGH, 3-9-3A, Window 22 • REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH, 3-9-3A, Window 4
	BOP	3-ARP-9-3A, Alarm Response Procedure REACTOR BUILDING RADIATION HIGH, Window 22 Operator Action: A. **DETERMINE** area with high radiation level on Panel 3-9-11. (Alarm on Panel 3-9-11 will automatically reset if radiation level lowers below setpoint.)
	BOP	Monitors Radiation Levels on Panel 3-9-11. Keeps the NUSO informed on instruments which are alarming and which are approaching Maximum Safe Values.
	BOP	B. N/A C. **NOTIFY** Radiation Protection.
	Driver	**If contacted as Radiation Protection, acknowledge any reports or direction given.**
	BOP	D. **IF** the TSC is **NOT** manned and a "VALID" radiological condition exists., **THEN USE** public address system to evacuate area where high radiological conditions exist
	BOP	Makes a plant announcement to evacuate the Reactor Building due to high radiation.
	BOP	F. **MONITOR** other parameters providing input to this annunciator frequently as these parameters will be masked from alarming while this alarm is sealed in.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 2 of 7

Event Description: Fuel Damage

Time	Position	Applicant's Actions or Behavior
	BOP	G. **IF** a CREV initiation is received, **THEN** 1. **CHECK** CREV A(B) Flow is ≥ 2700 CFM, and ≤ 3300 CFM as indicated on 0-FI-031-7214(7213) within 5 hours of the CREV initiation. 2. **IF** CREV A(B) Flow is NOT ≥2700 CFM, and ≤ 3300 CFM as indicated on 0-FI-031-7214(7213), **THEN PERFORM** the following: (Otherwise N/A) a. **STOP** the operating CREV per 0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System. b. **START** the standby CREV per 0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System.
	Driver	**If contacted as an AUO to monitor CREV operation, acknowledge the direction.**
	BOP	H. N/A I. **ENTER** 3-EOI-3, Secondary Containment Control.
	NUSO	Re-enters 3-EOI-3, Secondary Containment Control (if not already entered on Secondary Containment Radiation).
	BOP	K. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	NRC	**It is not expected that the SRO reference NPG-SPP-18.3.5, Equipment Important to Emergency Response, during this event.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 3 of 7

Event Description: Fuel Damage

Time	Position	Applicant's Actions or Behavior
	NUSO	**SC Radiation**

SC/R-1

WHEN

ANY Area Radiation Level exceeds its Max Normal Radiation Level (Table SC-2)

Table SC-2
Secondary Cntmt Area Radiation

Area	Applicable Radiation Indicators	Max Normal Value mR/hr	Max Safe Value mR/hr	Potential Isolation Sources
RHR sys I pumps	90-25A	Alarmed	1000	FCV-74-47, 48
RHR sys II pumps	90-28A	Alarmed	1000	FCV-74-47, 48
HPCI room	90-24A	Alarmed	1000	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	90-26A	Alarmed	1000	FCV-71-2, 3, 39
CS sys II pumps	90-27A	Alarmed	1000	None
Top of torus General area	90-29A	Alarmed	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB el 565 W	90-20A	Alarmed	1000	FCV-69-1, 2, 12 SDV vents & drains
RB el 565 E	90-21A	Alarmed	1000	SDV vents & drains
RB el 565 NE	90-23A	Alarmed	1000	None
TIP room	90-22A	Alarmed	100,000	TIP ball valve
RB el 593	90-13A, 14A	Alarmed	1000	FCV-74-47, 48
RB el 621	90-9A	Alarmed	1000	FCV-43-13, 14
Recirc MG sets	90-4A	Alarmed	1000	None
Refuel floor	90-1A, 2A 3A	Alarmed	1000	None

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 4 of 7

Event Description: Fuel Damage

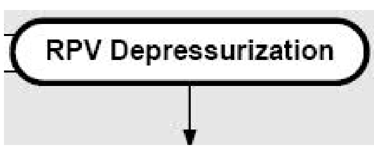
Time	Position	Applicant's Actions or Behavior
	NUSO	SC-3

3

ISOLATE all systems that are discharging into the area EXCEPT systems required: • For damage control OR • To be operated by EOIs

NOTE	
3	Tables SC-1 and SC-2 contain information that may be used to determine if a primary system is discharging into Secondary Containment (Emergency Depressurization will reduce discharge).

SC-4



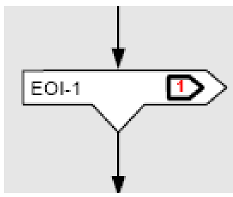
SC-7

3

WHEN
A Primary System is discharging into Secondary Containment

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 5 of 7

Event Description: Fuel Damage

Time	Position	Applicant's Actions or Behavior
	NUSO	SC-8 **BEFORE** ANY Secondary Containment parameter reaches its Max Safe Value (Tables SC-1, SC-2, and SC-3) 
	BOP	Monitors Area Radiation levels and informs the NUSO when two areas are at Maximum Safe.
	NUSO	SC-9 **WHEN** Any Secondary Containment parameter exceeds its Max Safe value in two (2) or more areas for the same parameter (Tables SC-1, SC-2, SC-3) SC-10 **EMERGENCY DEPRESSURIZATION IS REQUIRED**
	NUSO	Updates the crew that Emergency Depressurization is required. Enters 3-C-2, Emergency RPV Depressurization.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 6 of 7

Event Description: Fuel Damage

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>Crew</td><td>Critical Task:
With a Primary System discharging into the Secondary Containment when two or more areas are greater than their maximum safe operating values for the same parameter, the Balance of Plant Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator.

Critical Task Failure Criteria:
The operating crew fails to proceed with without delay and in a controlled manner to initiate Emergency Depressurization from the time it is announced that two Area Radiation Levels exceed Maximum Safe value.</td></tr>
<tr><td></td><td>NUSO</td><td>3-C-2, Emergency RPV Depressurization
C2-1
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Reactor Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr><tr><td>It is anticipated that Reactor depressurization will result in loss of injection required for Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr><tr><td>Containment Water Level CANNOT be maintained below 44 feet</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>C2-2
IF Drywell Pressure is above 2.45 PSIG
THEN PREVENT injection from ONLY those Core Spray and LPCI pumps NOT required to assure Adequate Core Cooling (Appendix 4)</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 8 Page 7 of 7

Event Description: Fuel Damage

Time	Position	Applicant's Actions or Behavior
	NUSO	C2-3 **EMERGENCY DEPRESSURIZE** the Reactor **IF** Suppression Pool Water Level is above 5.5 feet **THEN OPEN** 6 MSRVs (ADS Valves preferred) ➢ OK to exceed 100ºF/hr cooldown rate **IF** / **THEN**: Drywell Control Air is or becomes unavailable — **NO ACTION REQUIRED** Less than 4 MSRVs can be opened AND Reactor Pressure is 80 PSIG or more above Suppression Chamber Pressure — **NO ACTION REQUIRED**
	BOP	Opens 5 SRVs and one additional SRV (due to ADS Valve 1-22 being out of service).
	NUSO	C2-4 **WHEN** Shutdown Cooling RPV Pressure interlock clears AND further cooldown is required
	NRC	**End of Event 8. When the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.**

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 9 Page 1 of 3

Event Description: Reactor Feedwater Pumps (RFPTs) Trip / HPCI Fails to Automatically Start and Inject

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9, Reactor Feedwater Pumps Trip / HPCI Fails to Automatically Start and Inject, is inserted when the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required by the Driver to insert Event 9.**
	OATC	Reports that all Reactor Feedwater Pumps (RFPTs) have tripped.
	BOP	When Reactor Water Level reaches the High Pressure Coolant Injection (HPCI) initiation setpoint (-45"), determines that HPCI did not automatically start and manually starts HPCI. Informs the NUSO of the actions required to start HPCI.
	NUSO	Directs the OATC/BOP to maintain Reactor Water Level using 3-EOI-Appendix-5D, Injection System Lineup HPCI.
	BOP	3-EOI-Appendix-5D, Injection System Lineup HPCI [1] **IF** Suppression Pool level drops below 12.75 ft. during HPCI operation, **THEN** TRIP HPCI and CONTROL injection using other options. [2] **IF** Suppression Pool level CANNOT be maintained below 5.25 in, **THEN EXECUTE** EOI Appendix 16E concurrently with this procedure to bypass HPCI High Suppression Pool Water Level Suction Transfer Interlock. [3] **IF** BOTH of the following exist: • High temperature exists in the HPCI area, AND • SRO directs bypass of HPCI High Temperature Isolation Interlocks, **THEN PERFORM** the following: [3.1] **EXECUTE** EOI Appendix 16L, Bypassing HPCI High Temperature Isolation concurrently with this procedure. [3.2] **RESET** auto isolation logic using HPCI AUTO-ISOL LOGIC A (B) RESET pushbuttons. [4] **VERIFY** at least one SGTS train in operation.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 9 Page 2 of 3

Event Description: Reactor Feedwater Pumps (RFPTs) Trip / HPCI Fails to Automatically Start and Inject

Time	Position	Applicant's Actions or Behavior
	BOP	**CAUTIONS** 1) Operating HPCI Turbine below 2400 rpm may result in unstable system operation and equipment damage. 2) Operating HPCI Turbine with suction temperatures above 140°F may result in equipment damage. [5] **VERIFY** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller is in one of the following configurations, as desired: • in AUTO and set for 5300 gpm for rapid injection • in AUTO and set for approximately 2500 gpm for slower injection • in MANUAL with output at approximately 50% for slower injection **NOTE** HPCI Auxiliary Oil Pump will NOT start UNTIL 3-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, starts to open. [6] **IF** high Reactor Water Level Trip logic is actuated, THEN [6.1] **DEPRESS** HPCI TURBINE TRIP RX LVL HIGH RESET pushbutton. [6.2] **CHECK** HPCI TURBINE TRIP LVL HIGH amber light has extinguished. [7] **PLACE** HPCI AUXILIARY OIL PUMP handswitch in START. [8] **PLACE** HPCI STEAM PACKING EXHAUSTER handswitch in START. [9] **OPEN** 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE. [10] **OPEN** 3-FCV-73-44, HPCI PUMP INJECTION VALVE. [11] **OPEN** 3-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start HPCI Turbine.

Op Test No.: 21-04 Scenario No. NRC-1 Event No.: 9 Page 3 of 3

Event Description: Reactor Feedwater Pumps (RFPTs) Trip / HPCI Fails to Automatically Start and Inject

Time	Position	Applicant's Actions or Behavior
	BOP	[12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine speed accelerates. B. 3-CKV-73-45, HPCI SYSTEM CHECK VALVE, opens by observing 3-ZI-73-45A, DISC POSITION, red light illuminated. C. HPCI flow to RPV stabilizes and is controlled automatically at the setpoint. (N/A if controller in manual). D. 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds approximately 1200 gpm. **CAUTION** 3-FCV-073-0030, HPCI PUMP MIN FLOW VALVE, automatically opens when system flow is at or below 900 gpm (lowering) only if a system initiation signal is present. Manually opening the min flow valve may be required for pump min flow protection. [13] **ADJUST** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller as necessary to control injection. [14] **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. [15] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN PLACE** HPCI AUXILIARY OIL PUMP handswitch in AUTO. [16] N/A [17] N/A
	NRC	**End of Event 9. When the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.**

Scenario Setup
UNIT 3

IC	54
Exam IC	251

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-68	99	EOI-3	12	ARP 6A	27
OI-69	109	3-C-2	13	ARP 8B	24
GOI-12	49	APPX 5D	8	ARP 8C	20
GOI-12A	72	ARP 3A	57	TRM 3.4.1	21
AOI-64-2A	12	ARP 3D	32	TS 3.5.1	229
AOI-85-3	13	ARP 4B	53	TS 3.5.3	244
AOI-100-1	74	ARP 4C	42	TS 3.6.1.3	212
EOI-1	13	ARP 5B	32		

Simulator Setup	• Verify camera system is powered down (admin password = abcd1234) • Start CPERF **PRIOR** to placing the Simulator in RUN • Ensure Danger Tags are placed on SRV 1-22 and the Emergency High Pressure Makeup Pump
Schedule Files(s):	2104 NRC Scenario 1 UNIT 3.sch RWCU.sch
Event Files(s):	2104 NRC Scenario 1 UNIT 3.evt

Schedule File: 2104 NRC Scenario 1 UNIT 3.sch

Event	Action	Description
	2104 NRC Scenario 1 Unit 3.evt	Event File
1	Insert remote CU01 to 45.00000 in 60	RWCU DEMIN FILTER A FRC-69-35
1	Insert remote CU02 to 45.00000 in 60	RWCU DEMIN FILTER B FRC-69-60
1	Insert remote AN01E after 15 to RESET	CU LOCAL RESET (3-XA-55-4B W24)
11	Insert remote CU05 to MANUAL	RWCU HX RBCCW FLOW CONTROL TIC-69-10A
21	Insert remote CU05 to AUTO	RWCU HX RBCCW FLOW CONTROL TIC-69-10A

Schedule File: 2104 NRC Scenario 1 UNIT 3.sch

Event	Action	Description
3	Insert malfunction XA-55-4C_13 to ON	RBCCW SURGE TANK LEVEL LOW 3-LA-70-2B
13	Insert malfunction XA-55-4C_13 after 10 to NORMAL delete in 1	RBCCW SURGE TANK LEVEL LOW 3-LA-70-2B
4	Schedule F:\2104\NRC\Scenarios\U3\Scenario 1\RWCU.sch	RWCU Leak
6	Insert malfunction ED08C	4KV UNIT BOARD 3C FAILURE (RELAY 86-316 AND 86-532)
16	Insert remote FW06 to START	CONDENSATE BOOSTER PUMP C AUX OIL PUMP
16	Insert remote AN01D to RESET in 5 on	FW LOCAL RESET 121 (3-XA-55-6B W6)
7	Insert malfunction RC09 to 100.00000 in 900	RCIC STEAM LEAK INTO RCIC ROOM
	Insert malfunction FCV-71-2 to FAIL_NOW	MOTOR_OPERATED_VALVE RCIC STEAM LINE INBD ISOL VALVE
	Insert override ZLOHS712A_1 to Off	HS-71-2A RCIC STEAM LINE INBD ISOL VALVE
	Insert override ZLOHS712A_2 to On	HS-71-2A RCIC STEAM LINE INBD ISOL VALVE
17	Delete override ZLOHS712A_2	HS-71-2A RCIC STEAM LINE INBD ISOL VALVE
	Insert malfunction FCV-71-3 to FAIL_NOW	MOTOR_OPERATED_VALVE RCIC STEAM LINE OUTBD ISOL VALVE
	Insert override ZLOHS713A_1 to Off	HS-71-3A RCIC STEAM LINE OUTBD ISOL VALVE
	Insert override ZLOHS713A_2 to On	HS-71-3A RCIC STEAM LINE OUTBD ISOL VALVE

Schedule File: 2104 NRC Scenario 1 UNIT 3.sch

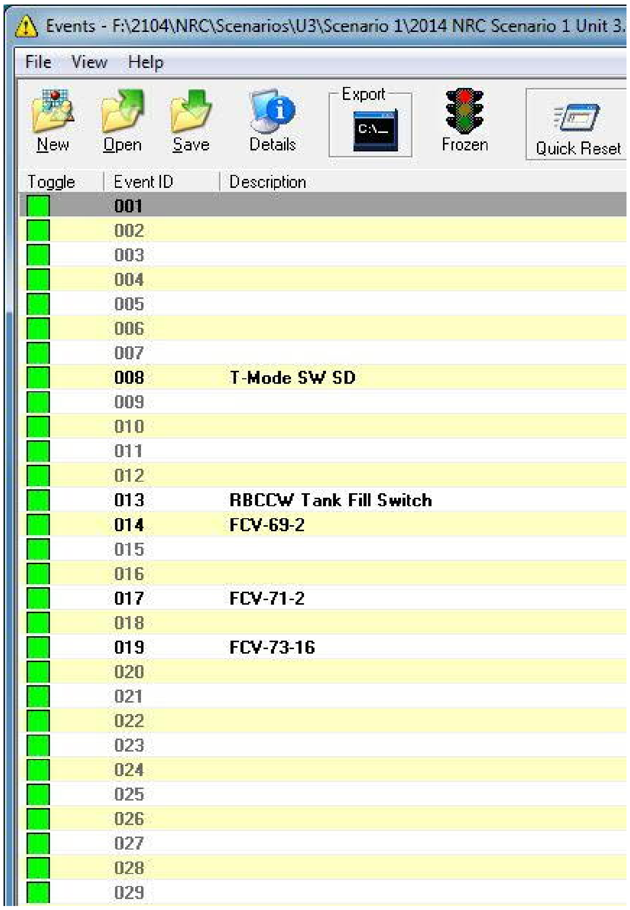
Event	Action	Description
8	Insert malfunction TH23 to 10.00000 in 900	FUEL CLADDING DAMAGE
8	Insert malfunction FW14A	RFPT 3A TRIP ON RFPT 3A BEARING LOW OIL PRESSURE (PS-3-123B)
8	Insert malfunction FW14B	RFPT 3B TRIP ON RFPT 3B BEARING LOW OIL PRESSURE (PS-3-149B)
8	Insert malfunction FW14C	RFPT 3C TRIP ON RFPT 3C BEARING LOW OIL PRESSURE (PS-3-174B)
	Insert malfunction FCV-73-16 to FAIL_NOW	MOTOR_OPERATED_VALVE HPCI TURBINE STEAM SUPPLY VALVE
	Insert override ZLOHS7316A_1 to On	HS-73-16A HPCI TURBINE STEAM SUPPLY VALVE
19	Delete malfunction FCV-73-16	MOTOR_OPERATED_VALVE HPCI TURBINE STEAM SUPPLY VALVE
19	Delete override ZLOHS7316A_1	HS-73-16A HPCI TURBINE STEAM SUPPLY VALVE

Schedule File: RWCU.sch

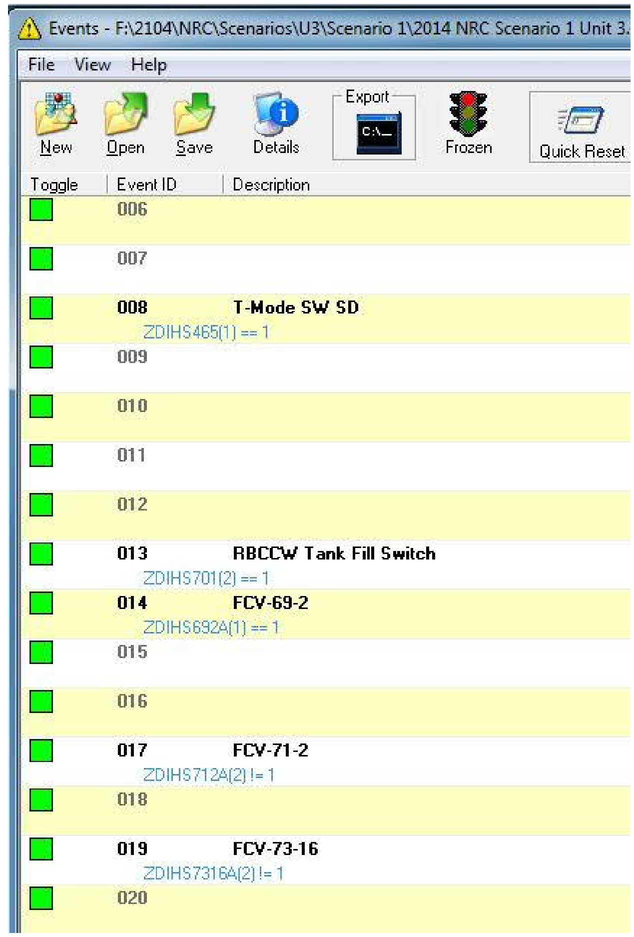
Event	Action	Description
	Insert malfunction CU04 to 25.00000	RWCU SYSTEM SUCTION BREAK
	Insert malfunction FCV-69-2 to FAIL_NOW	MOTOR_OPERATED_VALVE RWCU OUTBOARD ISOLATION VLV
	Insert override ZLOHS692A_1 to Off	HS-69-2A RWCU OUTBD SUCT ISOLATION VALVE
	Insert override ZLOHS692A_2 to On	HS-69-2A RWCU OUTBD SUCT ISOLATION VALVE
14	Delete malfunction FCV-69-2	MOTOR_OPERATED_VALVE RWCU OUTBOARD ISOLATION VALVE
14	Delete override ZLOHS692A_1	HS-69-2A RWCU OUTBD SUCT ISOLATION VALVE
14	Delete override ZLOHS692A_2	HS-69-2A RWCU OUTBD SUCT ISOLATION VALVE

Event File

List



Details



<table>
<tr><th colspan="3">UNIT 3 SHIFT TURNOVER MEETING</th><th>Today</th></tr>
<tr><td rowspan="2">MODE 1</td><td>DAYS ON LINE
234</td><td rowspan="2">Drywell Leakage (GPM)
1.81</td><td>Protected Equipment</td></tr>
<tr><td rowspan="3">PRA (EOOS) -Green
500Kv GRID - Qualified
161Kv Grid -Qualified</td><td></td></tr>
<tr><td>Rx Power
80.0%</td><td>Floor Drain (GPM)
0.27</td><td></td></tr>
<tr><td>MWe
1303</td><td>Last breaker closure
3/15/19 5:41</td><td>Equipment Drain (GPM)
1.54</td><td></td></tr>
</table>

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

SRV 1-22 is INOPERABLE – ADS Valve. Tech Spec 3.5.1.E, Condition E (Day 4 of 14 day LCO)

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

Reactor Shutdown. Maintain RFPTs, Condensate, and Condensate Booster Pumps running until Reactor Power is <70%

EHPM tagged for bearing inspection.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Continue the Reactor Shutdown. Reduce Reactor Power to 75% using Core Flow, then wait for further guidance from RE.

OPERATOR WORK AROUNDS **OWAs - 0** **Burdens - 0** **Challenges - 6**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Facility: BFN Scenario Number: NRC-2 Op-Test Number: 21-04

Examiners: ____________________ Operators: SRO: ____________________

____________________ ATC: ____________________

____________________ BOP: ____________________

Initial Conditions: 100% Reactor Power.

Turnover: 100% Reactor Power. A3 EECW Pump is tagged for oil change. 2B EHC Pump is tagged for maintenance. 2A CCW Pump ready for restart.

Critical Tasks:

1. When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the loss of forced recirculation to prevent possible fuel damage.
2. With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Swap Recirc Drive Cooling Water Pumps
2.	SW03F	C-BOP TS-NUSO	EECW Pump Trip
3.	NMAPRMGAIN(1)	C-OATC C-NUSO	APRM 1 Fails Downscale
4.#	RD04R3023	C-OATC TS-NUSO	Control Rod Drifts Out
5.	MC05	C-BOP C-NUSO	Clogged Traveling Screens / Lowering Condenser Vacuum
6.#	N/A	R-OATC R-NUSO	Reactor Power Reduction for Lowering Condenser Vacuum
7.#	RD09A RD09B RD17A RD17B	M-ALL	Hydraulic Anticipated Transient Without SCRAM (ATWS)
8.	HS-47-1A	C-BOP C-NUSO	2A EHC Pump Trip
9.	SL01A	C-BOP C-NUSO	SLC Pump Trip

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

Event on previous two NRC Exams

#S Event on previous two NRC Exams Spare Scenario

Events

1. The crew will swap Recirc Drive Cooling Water Pumps in accordance with 2-OI-68, Reactor Recirculation System, Section 6.3.

2. D3 Emergency Equipment Cooling Water (EECW) Pump will trip. The crew will start another pump to support the South EECW Header in accordance with 0-OI-67, Emergency Equipment Cooling Water, Section 5.3. Depending on the course of action taken to restore EECW Flow, the Nuclear Unit Senior Operator (NUSO) will reference either Technical Specification 3.7.1, Residual Heat Removal Service Water (RHRSW) System, Condition A or Technical Specification 3.7.2, Emergency equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS), Condition A.

3. APRM 1 will fail downscale. The crew will respond in accordance with Alarm Response Procedures and 2-OI-92B, Average Power Range Monitoring to bypass the faulty instrument. The NUSO will review Technical Specification 3.3.1.1, RPS Instrumentation, Table 3.3.1.1.

4. A Control Rod will drift out. The crew will take actions to insert the Control Rod in accordance with 2-AOI-85-6, Rod Drift Out. The drifting Control Rod will latch into position “00” and the NUSO will address Technical Specification 3.1.3, Control Rod OPERABILITY, Condition C.

5. Due to eel grass intrusion, the intake traveling screens will become clogged, resulting in lowering Condenser Vacuum. The crew will respond to lowering Condenser Vacuum in accordance with 2-AOI-47-3, Loss of Condenser Vacuum.

6. In response to lowering Condenser Vacuum, the crew will reduce Reactor Power in an attempt to maintain Condenser Vacuum accordance with 2-AOI-47-3, Loss of Condenser Vacuum.

7. When a manual Reactor SCRAM is attempted, Control Rods will not insert due to a Hydraulic Anticipated Transient Without SCRAM (ATWS). The crew will respond in accordance with 2-EOI-1A, ATWS RPV Control.

8. When a Manual Reactor SCRAM is inserted, 2A EHC Pump will trip, resulting in Main Turbine Bypass Valves failing CLOSED, requiring the crew to take action to control Reactor Pressure.

9. When the crew attempts to inject SLC for Reactor Power Control, the first SLC Pump will trip, requiring the Balance of Plant Operator (BOP) to start the alternate SLC Pump in accordance with 2-EOI Appendix-3A, SLC Injection.

The Scenario ends when the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems.

Critical Tasks 2

1. **When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the loss of forced recirculation to prevent possible fuel damage.**

   a. **Safety Significance**

   With thermal power being produced in the Reactor and actions to lower Reactor Power have not brought power out of the heating range, power oscillations and subsequent fuel damage may be generated.

   b. **Cues**

   The Reactor is SCRAMMED and ATWS actions are taken by the OATC and Reactor Power is greater than 5% or unknown.

   c. **Measured by:**

   Reactor Power on APRM displays.

   d. **Feedback**

   Lowering Reactor Water Level.
   Lowering Reactor Power.

   e. **Critical Task Failure Criteria:**

   The operating crew fails to STOP and PREVENT injection and lower Reactor Water Level for Reactor Power control during an ATWS within 130 seconds after Recirculation Flow is stopped.

2. **With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor Depressurization and subsequent power excursion, inhibit ADS.**

   a. **Safety Significance**

   Precludes core damage due to an uncontrolled reactivity addition.

   b. **Cues**

   Procedural Compliance.

   c. **Measured by**

   ADS Logic inhibited prior to an automatic initiation.

   d. **Feedback**

   Reactor Pressure trend.
   Reactor Water Level trend.
   ADS LOGIC BUS A/B INHIBITED (2-9-3C, Window 18/31) annunciator status.

   e. **Critical Task Failure Criteria**

   ADS automatic initiation with Control Rods out.

Facility: BFN Scenario Number: NRC-2 Op-Test Number:21-04

Examiners: ____________________________ Operators: SRO: ______________________

____________________________ ATC: ______________________

____________________________ BOP: ______________________

Initial Conditions: 100% Reactor Power.

Turnover: 100% Reactor Power. B3 EECW Pump is tagged for oil change. 3B EHC Pump is tagged for maintenance. 3A CCW Pump ready for restart.

Critical Tasks:

1. When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the loss of forced recirculation to prevent possible fuel damage.
2. With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Swap Recirc Drive Cooling Water Pumps
2.	SW03F	C-BOP TS-NUSO	EECW Pump Trip
3.	NMAPRMGAIN(1)	C-OATC C-NUSO	APRM 1 Fails Downscale
4.#	RD04R3023	C-OATC TS-NUSO	Control Rod Drifts Out
5.	MC05	C-BOP C-NUSO	Clogged Traveling Screens / Lowering Condenser Vacuum
6.#	N/A	R-OATC R-NUSO	Reactor Power Reduction for Lowering Condenser Vacuum
7.#	RD09A RD09B RD17A RD17B	M-ALL	Hydraulic Anticipated Transient Without SCRAM (ATWS)
8.#	HS-47-1A	C-BOP C-NUSO	3A EHC Pump Trip
9.	SL01A	C-BOP C-NUSO	SLC Pump Trip

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

Event on previous two NRC Exams

#S Event on previous two NRC Exams Spare Scenario

Events

1. The crew will swap Recirc Drive Cooling Water Pumps in accordance with 3-OI-68, Reactor Recirculation System, Section 6.3.

2. C3 Emergency Equipment Cooling Water (EECW) Pump will trip. The crew will start another pump to support the South EECW Header in accordance with 0-OI-67, Emergency Equipment Cooling Water, Section 5.3. Depending on the course of action taken to restore EECW Flow, the Nuclear Unit Senior Operator (NUSO) will reference either Technical Specification 3.7.1, Residual Heat Removal Service Water (RHRSW) System, Condition A or Technical Specification 3.7.2, Emergency equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS), Condition A.

3. APRM 1 will fail downscale. The crew will respond in accordance with Alarm Response Procedures and 3-OI-92B, Average Power Range Monitoring to bypass the faulty instrument. The NUSO will review Technical Specification 3.3.1.1, RPS Instrumentation, Table 3.3.1.1.

4. A Control Rod will drift out. The crew will take actions to insert the Control Rod in accordance with 3-AOI-85-6, Rod Drift Out. The drifting Control Rod will latch into position "00" and the NUSO will address Technical Specification 3.1.3, Control Rod OPERABILITY, Condition C.

5. Due to eel grass intrusion, the intake traveling screens will become clogged, resulting in lowering of Condenser Vacuum. The crew will respond to lowering Condenser Vacuum in accordance with 3-AOI-47-3, Loss of Condenser Vacuum.

6. In response to lowering Condenser Vacuum, the crew will reduce Reactor Power in an attempt to maintain Condenser Vacuum accordance with 3-AOI-47-3, Loss of Condenser Vacuum.

7. When a manual Reactor SCRAM is attempted, Control Rods will not insert due to a Hydraulic Anticipated Transient Without Scram (ATWS). The crew will respond in accordance with 3-EOI-1A, ATWS RPV Control.

8. When a Manual Reactor SCRAM is inserted, 3A EHC Pump will trip, resulting in Main Turbine Bypass Valves failing CLOSED, requiring the crew to take action to control Reactor Pressure.

9. When the crew attempts to inject SLC for Reactor Power Control, the first SLC Pump will trip, requiring the Balance of Plant Operator (BOP) to start the alternate SLC Pump in accordance with 3-EOI Appendix-3A, SLC Injection.

The Scenario ends when the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems.

Critical Tasks 2

1. **When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the loss of forced recirculation to prevent possible fuel damage.**

   a. **Safety Significance**
   With thermal power being produced in the Reactor and actions to lower Reactor Power have not brought power out of the heating range, power oscillations and subsequent fuel damage may be generated.

   b. **Cues**
   The Reactor is SCRAMMED and ATWS actions are taken by the OATC and Reactor Power is greater than 5% or unknown.

   c. **Measured by:**
   Reactor Power on APRM displays.

   d. **Feedback**
   Lowering Reactor Water Level.
   Lowering Reactor Power.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to STOP and PREVENT injection and lower Reactor Water Level for Reactor Power control during an ATWS within 130 seconds of the loss of forced recirculation.

2. **With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor Depressurization and subsequent power excursion, inhibit ADS.**

   a. **Safety Significance**
   Precludes core damage due to an uncontrolled reactivity addition.

   b. **Cues**
   Procedural Compliance.

   c. **Measured by**
   ADS Logic inhibited prior to an automatic initiation.

   d. **Feedback**
   Reactor Pressure trend.
   Reactor Water Level trend.
   ADS LOGIC BUS A/B INHIBITED (3-9-3C, Window 18/31) annunciator status.

   e. **Critical Task Failure Criteria**
   ADS automatic initiation with Control Rods out.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 1 Page 1 of 2

Event Description: Swap Recirc Drive Cooling Water Pumps

Time	Position	Applicant's Actions or Behavior
	Driver	**Prior to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not start Event 1, Swap Recirc Drive Cooling Water Pumps, after assuming the shift, request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to swap Recirc Drive Cooling Water Pumps.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to swap Recirc Drive Cooling Water Pumps.**
	NUSO	Directs the Balance of Plant Operator (BOP) to swap Recirc Drive Cooling Water Pumps in accordance with 2-OI-68, Reactor Recirculation System.
	BOP	2-OI-68, Reactor Recirculation System Section 6.3, Swapping Recirc Drive Cooling Water Pumps **NOTES** 1) Perform these steps, as required, to swap the Recirc Drive Cooling Water Pumps. 2) Placing the standby pump in RUN will cause the running pump to shut down after ~2 seconds if the running pump is in AUTO. 3) The red light indication above the MCR handswitch only indicates that the motor starter has been energized. A successful pump start should be verified locally or by ICS flow indication. 4) ICS screen VFDPMPA(VFDPMPB) may be referred to observe Recirc Drive Cooling Water System parameters. 5) The time both Cooling Water Pumps are running should be minimized. The pump being placed in standby should be placed in AUTO as soon as possible after placing the lead pump in RUN. [1] **IF** it is desired to place Recirc Drive Cooling Water Pump 2A2 in service, and place 2A1 pump in standby, **THEN PERFORM** the following: (Otherwise N/A): [1.1] **DEPRESS** 2-HS-96-13, FAULT RESET, on Panel 2-9-4.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 1 Page 2 of 2

Event Description: Swap Recirc Drive Cooling Water Pumps

Time	Position	Applicant's Actions or Behavior
	BOP	[1.2] **PLACE** in RUN 2-HS-68-2A2/A, RECIRC DRIVE 2A COOLING PUMP 2A2. [1.3] **CHECK** RECIRC DRIVE 2A COOLING PUMP 2A2, STARTS. [1.4] **PLACE** in AUTO 2-HS-68-2A1/A, RECIRC DRIVE 2A COOLING PUMP 2A1. [1.5] **CHECK** RECIRC DRIVE 2A COOLING PUMP 2A1 STOPS. [2] N/A [3] **IF** it is desired to place Recirc Drive Cooling Water Pump 2B2 in service, and place the B1 pump in standby, **THEN PERFORM** the following: (Otherwise N/A) [3.1] **DEPRESS** 2-HS-96-14, FAULT RESET, on Panel 2-9-4. [3.2] **PLACE** in RUN 2-HS-68-2B2/A, RECIRC DRIVE 2B COOLING PUMP 2B2. [3.3] **CHECK** RECIRC DRIVE 2B COOLING PUMP 2B2 STARTS. [3.4] **PLACE** in AUTO 2-HS-68-2B1/A, RECIRC DRIVE 2B COOLING PUMP 2B1. [3.5] **CHECK** RECIRC DRIVE 2B COOLING PUMP 2B1 STOPS. [4] N/A
	BOP	Informs the Nuclear Unit Senior Operator (NUSO) that Recirc Drive Cooling Water Pumps have been swapped.
	NRC	**End of Event 1. Request that the driver insert Event 2, D3 EECW Pump Trip.**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 2 Page 1 of 3

Event Description: EECW Pump Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 2, EECW Pump Trip to trip D3 EECW Pump.**
	BOP	Acknowledges and reports the following alarms as received to the NUSO: • MOTOR TRIPOUT, 2-9-8C, Window 33 • EECW NORTH HEADER DG SECTION PRESSURE LOW, 2-9-20A, Window 21 Informs the NUSO that D3 EECW Pump has tripped.
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 2-ARP-9-8C MOTOR TRIPOUT, Window 33 Operator Action: A. **CHECK** Control Room for white disagreement light illuminated for affected equipment. B. **CLEAR** disagreement light. C. **DISPATCH** personnel to check: • Relays at associated electrical board • Equipment for abnormal conditions • Safe-stop locally reset, if necessary
	Driver	**If contacted as the Outside NUSO, Assistant Unit Operator (AUO), or Electrical Maintenance to investigate the trip of D3 EECW Pump, acknowledge the direction.**
	BOP	D. **REFER TO** 0-GOI-300-2, Electrical, if relay targets are present or for motor starting limits. E. **REFER TO** appropriate Operating Instruction for recovery or realignment of equipment.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 2 Page 2 of 3

Event Description: EECW Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 2-ARP-9-20A EECW NORTH HEADER DG SECTION PRESSURE LOW, Window 21 Operator Action: A. **CHECK** indications on Panel 2-9-20. 1. 0-PI-67-19/2, Unit 1-2 NORTH HEADER PRESSURE. 2. 0-FI-67-3A/2, EECW NORTH HEADER PUMP A FLOW. 3. 0-FI-67-9A/2, EECW NORTH HEADER PUMP C FLOW. B. **CHECK** Panel 2-9-3 for status of North Header Pump(s) breaker lights and Pump Motor Amps normal. C. **NOTIFY** UNIT SUPERVISOR, Unit 1 and Unit 3.
	Driver	**If contacted as the Unit 1 and/or Unit 3 NUSO, acknowledge any information given.**
	BOP	D. **START** standby pump for affected header. **REFER TO** 0-OI-67, Emergency Equipment Cooling Water System. E. **DISPATCH** personnel to check affected pump room and header for abnormal conditions. F. N/A G. N/A H. **IF** pump failure is cause of alarm, **THEN REFER TO** Technical Specification 3.7.2, Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS).
	NUSO	Directs the BOP to start B3 EECW Pump.
	BOP	0-OI-67, Emergency Equipment Cooling Water System Precautions and Limitations C. The EECW System is aligned as follows: 1. At least one RHRSW Pump, assigned to the EECW System, should be running on each header to maintain the header charged at all times. If no pumps are running on a header and header pressure lowers to ≤ 0 psig, the header shall be declared inoperable and appropriate actions taken, as required by Technical Specifications.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 2 Page 3 of 3

Event Description: EECW Pump Trip

Time	Position	Applicant's Actions or Behavior	
	BOP	Starts B3 EECW Pump.	
	NUSO	References Technical Specification 3.7.2, Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS). LCO 3.7.2: The EECW System with three pumps and UHS shall be OPERABLE. APPLICABILITY: MODES 1, 2, and 3. **CONDITION:** A. One required EECW Pump INOPERABLE.	
	NUSO	**REQUIRED ACTION:** A.1 Restore the required EECW Pump to OPERABLE status.	**COMPLETION TIME:** A.1 – 7 days
	NRC	**End of Event 2. Request that the Driver insert Event 3, APRM 1 Fails Downscale.**	

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 3 Page 1 of 3

Event Description: APRM 1 Fails Downscale

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3, APRM 1 Fails Downscale.**
	OATC	Acknowledges and reports the following alarms: • APRM DOWNSCALE / OPRM INOPERABLE, 2-9-5A, Window 4 • CONTROL ROD WITHDRAWAL BLOCK, 2-9-5A, Window 7
	NUSO	Directs the Operator at the Controls (OATC) to respond in accordance with the appropriate Alarm Response Procedures.
	OATC	Alarm Response Procedure, 2-ARP-9-5A APRM DOWNSCALE / OPRM INOP, Window 4 Operator Action: A. **DETERMINE** which APRM/OPRM channel is downscale/inoperable. B. **IF** APRM failed downscale, **THEN BYPASS** channel. **REFER** TO 2-OI-92B, Average Power Range Monitoring. C. N/A D. N/A E. **REFER TO** Technical Specification (Tech Spec) Tables 3.3.1.1-1, Reactor Protection System Instrumentation, and Technical Requirements Manual (TRM) Table 3.3.4-1, Control Rod Block Instrumentation.
	OATC	Recommends to the NUSO that APRM 1 be bypassed.
	NRC	**EXAMINER NOTE: The NUSO may reference Tech Specs and the TRM. There are no required Tech Spec or TRM actions, but the NUSO may enter an Information Only LCO.**
	NUSO	Directs the OATC to bypass APRM 1 in accordance with the appropriate Operating Instruction.

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 3	Page 2 of 3

Event Description: APRM 1 Fails Downscale

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-92B, Average Power Range Monitoring Section 6.0, System Operations **NOTES** 1) Only one APRM/OPRM can be bypassed at a time. 2) All operations are performed on Panel 2-9-5 unless specifically stated otherwise. 3) In order to prevent inadvertent Rod Withdrawal Block or Reactor SCRAM while operating APRM BYPASS Selector Switch, always ensure the previously bypassed channel returns to normal status by observing the BLUE bypassed lights on Panel 2-9-14 Voters are extinguished prior to selecting any other channel to be bypassed. After bypassing a channel, the applicable BLUE BYPASSED status lights on Panel 2-9-14 Voters should be illuminated prior to testing, operating, or working on that channel.
	OATC	Section 6.1, Bypassing APRM / OPRM Channel **CAUTION** NPG-SPP-10.4, Reactivity Management Program, requires approval of the Plant Manager or his designee prior to any planned operation with APRMs bypassed unless bypassing is specifically allowed within approved procedures. [1] **REVIEW** all Precautions and Limitations. **REFER** TO Section 3.0. [2] **PLACE** 2-HS-92-7B/S3, APRM BYPASS, to desired channel to be bypassed. [3] **CHECK** BLUE BYPASSED lights illuminated on Panel 2-9-14 Voters. [4] **CHECK** white bypass light on Panel 2-9-5 is illuminated.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 3 Page 3 of 3

Event Description: APRM 1 Fails Downscale

Time	Position	Applicant's Actions or Behavior
	OATC	Alarm Response Procedure, 2-ARP-9-5A CONTROL ROD WITHDRAWAL BLOCK, Window 7 Operator Action: A. **DETERMINE** initiating condition from corresponding rod withdrawal block alarm(s) and **REFER TO** Operator Action for alarm(s). B. N/A C. N/A D. N/A E. N/A
	NRC	**End of Event 3. Request that the driver insert Event 4, Control Rod Drifts Out.**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 4 Page 1 of 3

Event Description: Control Rod Drifts Out

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4, Control Rod Drift Out.**
	NRC	**Control Rod 30-11 will drift out.**
	OATC	Acknowledges and reports the following alarm to the NUSO: • CONTROL ROD DRIFT, 2-9-5A, Window 28
	NUSO	Directs the OATC to respond in accordance with the appropriate Alarm Response and Abnormal Operating Procedures.
	OATC	Alarm Response Procedure, 2-ARP-9-5A CONTROL ROD DRIFT, Window 28 Operator Action: A. **DETERMINE** which rod is drifting from Full Core Display. B. **IF** no Control Rod motion is observed, **THEN RESET** rod drift as follows. 1. **PLACE** 2-HS-85-3A-S7, ROD DRIFT ALARM TEST switch, in RESET and **RELEASE**. 2. **RESET** annunciator. C. N/A D. **IF** rod drifting out, **THEN REFER** TO 2-AOI-85-6, Rod Drift Out and 2-AOI-85-7, Mispositioned Control Rod. E. **REFER TO** Tech Spec 3.1.3, Control Rod Operability and 3.10.8, Shutdown Margin (SDM) Test – Refueling.
	NRC	**The Control Rod Drift condition will clear when the Control Rod is driven to Position 0.**
	OATC	2-AOI-85-6, Rod Drift Out Immediate Actions: [1] N/A Subsequent Actions: [1] N/A

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 4 Page 2 of 3

Event Description: Control Rod Drifts Out

Time	Position	Applicant's Actions or Behavior
	OATC	[2] **IF** a Control Rod is moving from its intended position without operator actions, **THEN SELECT** the drifting Control Rod and **INSERT** to the FULL IN (00) position. [3] **IF** a Control Rod Block occurs during rod insertion due to Rod Worth Minimizer, **THEN BYPASS** the RWM per step 4.2[1]. (Otherwise N/A) [4] N/A [5] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern.
	Driver	**When contacted as the Reactor Engineer acknowledge any direction given.**
	OATC	[6] **IF** another Control Rod Drift occurs before Reactor Engineering completes the evaluation, **THEN MANUALLY SCRAM** Reactor and enter 2-AOI-100-1, Reactor SCRAM. [7] N/A [8] **IF** the Control Rod is latched into position "00", **THEN REMOVE** associated Hydraulic Control Unit (HCU) from service per 2-OI-85, Control Rod Drive System. (N/A if Control will not latch at "00") [9] **EVALUATE** Tech Spec 3.1.3, Control Rod OPERABILITY. [10] **INITIATE** Service Request/Work Order. [11] **NOTIFY** Reactor Engineer to perform the following: • **EVALUATE** condition of the Core to assure no resultant fuel damage has occurred, and • **EVALUATION** of impact on Thermal Limits and PCIOMOR restraints. (N/A if SCRAM was initiated.) • **DETERMINE** if other Control Rods need to be repositioned in order to safely restore Core symmetry to prevent local fuel damage. (N/A if scram was initiated.) [12] **NOTIFY** System Engineering to **PERFORM** 0-TI-20, Control Rod Drive System Testing and Troubleshooting, to determine problem with faulty Control Rod. [13] **IF** a manual SCRAM was not inserted and Reactor Startup or Shutdown is not in progress, **THEN ENSURE** 2-GOI-100-12, Power Maneuvering, has been entered if a power change occurred. (Otherwise N/A)

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 4	Page 3 of 3
Event Description: Control Rod Drifts Out	

Time	Position	Applicant's Actions or Behavior	
	OATC	[14] N/A [15] N/A [16] **NOTIFY** the Reactor Engineer to **EVALUATE** impact on preconditioning envelope, prior to returning to normal power operation.	
	NUSO	Technical Specification 3.1.3, Control Rod OPERABILITY LCO 3.1.3 Each Control Rod shall be OPERABLE Applicability: Modes 1 and 2 --- NOTE: Separate Condition entry is allowed for each Control Rod. --- **CONDITION:** C. One or more Control Rods INOPERABLE for reasons other than Condition A or B.	
	NUSO	**REQUIRED ACTION:** C.1 --------------NOTE------------------ RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of INOPERABLE Control Rod and continued operation. --------------------------------------- Fully insert INOPERABLE Control Rod <u>AND</u> C.2 Disarm the associated CRD.	**COMPLETION TIME:** C.1 – 3 hours C.2 – 4 hours
	NRC	**Tech Spec 3.10.8, Shutdown Margin (SDM) Test – Refueling is not applicable to current plant conditions.**	
	NRC	**End of Event 4. Request that the Driver insert Event 5, Clogged Traveling Screens / Lowering Condenser Vacuum.**	

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 1 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5, Clogged Traveling Screens / Lowering Condenser Vacuum.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • TRAVELING SCEEN DP HIGH, 2-9-20A, Window 18
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 2-ARP-9-20A TRAVELING SCEEN DP HIGH, Window 18 Operator Action: A. **VERIFY** alarm on 2-PDI-27-1A, TRAVELING SCREEN DIFFERENTIAL WATER LEVEL, on Panel 2-9-20. B. **DISPATCH** personnel to VERIFY the traveling screens are in service. Refer to 2-OI-27A, Screen Wash System. C. **MONITOR** Traveling Screens for carryover. D. **MONITOR** Turbine Backpressure. E. **IF** debris is being carried over, **THEN** • **MONITOR** 0-PDIS-067-0001(0005)(0008)(0011), EECW SUPPLY STRAINER DIFF PRESS, locally in RHRSW Pump Rooms • **MONITOR** Waterbox D/P for indications of fouling (<160" H2O (does not apply to 2C2 waterbox) with 3 CCW pumps in service) F. **IF** TRAVELING SCREEN DIFF WTR LVL, 2-PDI-27-1A, does NOT lower, **THEN** • **REFER** TO 2-OI-27A, Screen Wash System • **REFER** TO 0-AOI-27-1, Component Biofouling • **NOTIFY** Mechanical Maintenance to SCRAPE the trash racks and/or operate Milfoil Harvester as needed

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 2 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	BOP	G. **IF** Divers are required to clear the trash racks, **THEN REMOVE** the Amertap system from service per 2-OI-27B, Amertap Condenser Tube Cleaning System.
	BOP	0-AOI-27-1, Component Biofouling Immediate Actions: NONE Subsequent Actions: **NOTES** 1) Procedure is written in a logical order but due to changing plant conditions and operator experience, steps may be performed in parallel or out of sequence as required. 2) The most common cause of degraded cooling water system performance is the fouling of intake screens for the CCW Pumps. When intake screens begin to foul they are required to be placed in service as soon as possible using 1(2,3)-OI-27A, Screen Wash System. The timely response to this condition will keep screens from becoming over burdened with foreign material and collapsing. 3) If CCW Intake Screens cannot be cleaned with associated CCW Pump running, the pump may have to be removed from service in order to clean screens. After screens are cleaned the affected CCW Pump may be returned to service (1(2,3)-OI-27, Condenser Circulating Water System) if desired. 4) Entry into this procedure requires evaluation of situation per EOOS Management NPG-SPP-09.11.1, Equipment Out of Service Management. CAUTION Debris Filter may cycle repeatedly when total CCW flow is throttled excessively or due to heavy debris carryover from the intake. Debris Filter Flush Valve Motor is not rated for heavy repeated cycling. CCW flow should be maximized by throttling open available waterboxes until pump head is as low as possible (>20" H2O), Debris Filter should be run in manual and checked often. Expedite returning CCW System to normal three pump alignment with Debris Filter in AUTO.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 3 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	BOP	[1] **CONTACT** Maintenance to **PERFORM** attachment 7.
	Driver	**When contacted as Maintenance to perform Attachment 7, acknowledge the direction.**
	BOP	[2] **CHECK** CCW Intake Screens for fouling. [3] **IF** CCW Intake Screens are fouled, **THEN ENSURE** in service per 2-OI-27A, Screen Wash System. (Otherwise N/A).
	Driver	**If contacted as the Outside NUSO, Maintenance, or an Assistant Unit Operator (AUO) to check intake screens for fouling, acknowledge the direction. Wait 2 minutes and report that the intake screens are becoming fouled by Eel Grass.**
	BOP	[3] **IF** CCW Intake Screens are fouled, **THEN ENSURE** in service per 2-OI-27A, Screen Wash System. (Otherwise N/A). [4] **INITIATE** Attachment 1, Continuous Action Summary.
	NUSO / BOP	0-AOI-27-1, Component Biofouling Attachment 1, Continuous Action Summary Action Summary [1] **IF** at any time any of the following condition occurs: • Unexpected fouling indication of more than one river water supplied heat exchanger. **THEN** Action II is applicable, GO TO Step 4.2[9]. (Otherwise N/A) [2] **IF** at any time any of the following conditions occur: • Any indications of abnormal operation of the Circulating Water System OR • Any removal of two or more Circulating Water Pumps from service **THEN GO TO** 1(2)(3)-OI-27, Circulating Water System (Otherwise N/A)

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 4 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **IF** at any time a low reservoir level (<550') OR High Traveling Screen DP occur **THEN**: (Otherwise N/A) • **MONITOR** CCW Pumps for loss of suction/cavitation. [4] **IF** at any time the CCW Pumps indicate a loss of suction or cavitation **THEN**: (Otherwise N/A) • **SHUTDOWN** the CCW Pumps. **REFER TO** 2-OI-27, Circulating Water System.
	NRC	**As the malfunction ramps in, Condenser Vacuum will continue to lower. The crew may enter 2-AOI-47-3, Loss of Condenser Vacuum before an alarm is received.**
	BOP	2-AOI-47-3, Loss of Condenser Vacuum Immediate Actions: NONE Subsequent Actions: [1] **IF** ANY EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). **CAUTION** Operations outside of the allowable regions shown on the Recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. **REFER TO** 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, for required actions and monitoring to be performed during a power reduction. [2] **MONITOR** Condenser Vacuum (Turb Exhaust) Margin to Trip using 2-XR-002-0026, CONDENSATE, Channel 7. [3] **IF** Condenser Vacuum (Turb Exhaust) Margin to Trip as indicated on 2-XR-002-0026, CONDENSATE, approaches 0 inches Hg, with Reactor Power less than 26%, **THEN TRIP** the Main Turbine.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 5 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **IF** Condenser Vacuum is lost, **THEN OPEN** 2-DRV-043-1019, HOTWELL SAMPLE TO FLOOR DRAIN, (557'@ T-10 C-Line) and 2-DRV-043-1020, CONDENSATE DEMIN SAMPLE TO FLOOR DRAIN, (557'@ T-6 G-Line), to establish flow through the sample lines.
	Driver	**If contacted as an AUO to perform any outside steps of this procedure, acknowledge any direction given.**
	BOP	[5] **REDUCE** Reactor Power in an attempt to maintain Condenser Vacuum.
	NRC	**See Event 6, Reactor Power Reduction for Lowering Condenser Vacuum for Reactor Power reduction actions.**
	BOP	[6] **ENSURE** automatic actions. [7] **CHECK** CCW Pumps for proper operation. **IF** available, **THEN START** additional CCW Pumps. [8] – [15] N/A
	NUSO	Directs the crew to monitor Condenser Vacuum and sets a trigger value of 1.0" to Trip Condenser Vacuum for a Reactor SCRAM.
	NRC	**In accordance with the ILT Simulator Expectations, the NUSO will set target values for Condenser Vacuum as follows:** • **Low Condenser Vacuum alarm (Panel 2-9-7B, Window 17) – Core Flow Runback** • **1" Margin to Trip Condenser Vacuum – Reactor SCRAM** **However, the NUSO may conservatively direct these actions based on the rate of vacuum degradation and information from the field.**
	BOP	Acknowledges and reports the following alarm when received: • CONDENSER A, B, OR C VACUUM LOW, 2-9-7B, Window 17

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 6 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 2-ARP-9-7B CONDENSER A, B, OR C VACUUM LOW, 2-9-7B, Window 17 Operator Action: A. **CHECK** alarm by checking Condenser Vacuum lowering, MWe lowering, and Exhaust Hood Temperature rise. B. **IF** alarm is valid, **THEN REFER** TO 2-AOI-47-3, Loss of Condenser Vacuum.
	NUSO	Provides a Condenser Vacuum trigger value for the OATC to insert a manual Reactor SCRAM.
	OATC	When the trigger value for Condenser Vacuum is reached or when directed by the NUSO, inserts a manual Reactor SCRAM.
	NRC	**Condenser Vacuum will continue to lower, requiring the crew to insert a manual Reactor SCRAM.** **Two minutes after the crew inserts a manual SCRAM, the clogged Traveling Screen malfunction will be set to 50% to provide cooling water for the crew to place Suppression Pool Cooling in service if necessary.**
	Driver	**Two minutes after the Reactor SCRAM report to the crew that there has been some success in clearing Traveling Screens to allow some water flow.**
	NRC	**End of Event 5. Event 6, Reactor Power Reduction for Lowering Condenser Vacuum will be inserted by the crew as a response to lowering vacuum. No action is required by the Driver to insert Event 6.** **When the crew inserts a manual Reactor SCRAM due to lowering Condenser Vacuum, the following Events are automatically inserted by Simulator Setup:** • **Event 7, Hydraulic Anticipated Transient Without SCRAM (ATWS)** • **Event 8, 2A EHC Pump Trip** • **Event 9, SLC Pump Trip** **No action is required by the Driver to insert Events 7, 8, or 9.**

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 6	Page 1 of 5

Event Description: Reactor Power Reduction for Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	NRC	**The crew may elect to reduce power by any method or combination of runbacks in an attempt to maintain Condenser Vacuum.**
	NUSO	Directs the OATC to reduce Reactor Power by using any one or any combination of the following: • Manually using Recirc Master Control pushbuttons on Panel 2-9-5 • Upper Power Runback • Mid Power Runback • Core Flow Runback
	NRC	**2-OI-68, Reactor Recirculation System** **3.0 Precautions and Limitations** **Section 3.5.3, Dual Pump Operation** **E. When raising (lowering) Reactor Power per the Reactivity Plan, the following guideline should be used to establish the 60 rpm mismatch between the Recirc Pumps.** **1. When the first Recirc Pump reaches 1200 rpm (1300 rpm) or both Recirc Pumps are at 1200 rpm (1300 rpm) then individual controls will be used.** **2. While following the Reactivity Plan establish the 60 rpm mismatch using the individual controls for the leading Recirc Pump.** **3. While maintaining the 60 rpm mismatch and using the Reactivity Plan, raise (lower) the Recirc Pump speeds using either the Master Controllers or Individual Controllers.** **4. Once a Recirc Pump reaches 1300 rpm (1200 rpm) the 60 rpm mismatch is no longer required. While following the Reactivity Plan raise (lower) the lagging Recirc Pump using the individual controller match Recirc Pump speeds.** **5. Once the Recirc Pumps are matched, then the speeds may be adjusted as required using the Master Controller or Individual Controllers while maintaining the requirements of Attachment 1.**

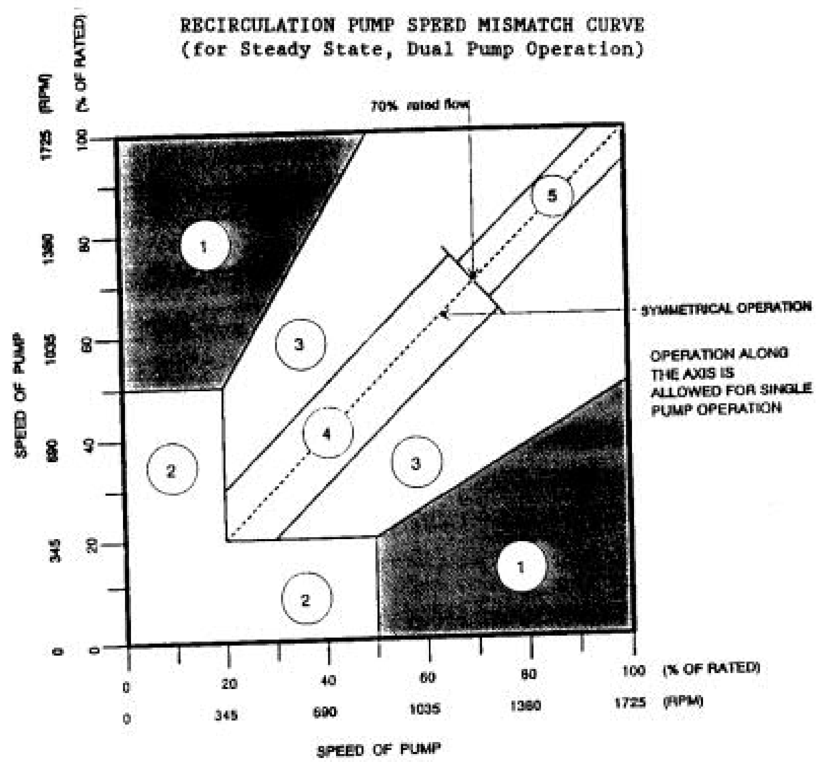
Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 6 Page 2 of 5

Event Description: Reactor Power Reduction for Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	NRC	(see below)

BFN Unit 2	Reactor Recirculation System	2-OI-68 Rev. 0159 Page 181 of 209

Attachment 1
(Page 1 of 1)
Recirculation Pump Speed Mismatch Curve



1. Avoid Region To Prevent Excessive Jet Pump Vibration.
2. Below Recirc Drive Minimum speed.
3. Operation allowed if reactor subcritical or during transient periods.
4. Limited Operation for Core flow ≤ 70% rated (mismatch ≤ 10% rated speed).
5. Limited operation for Core flow > 70% rated (mismatch ≤ 5% rated speed).

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 6	Page 3 of 5
Event Description: Reactor Power Reduction for Lowering Condenser Vacuum			

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown in 0-TI-248 and 2-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the later part of the operating cycle (Coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the Power to Flow Map. Instruments used to monitor pump speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing Core Flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area. [1] **IF** desired to control Recirc Pumps 2A and/or 2B speed with Recirc Individual Control, **THEN PERFORM** the following: • **LOWER** Recirc Pump 2A using 2-HS-96-17A(17B)(17C), SLOW(MEDIUM)(FAST), (Otherwise N/A) AND/OR • **LOWER** Recirc Pump 2B using 2-HS-96-18A(18B)(18C), SLOW(MEDIUM)(FAST). (Otherwise N/A) [2] **WHEN** desired to control Recirc Pumps 2A and/or 2B speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump Speed 2A & 2B using the following pushbuttons as required. 2-HS-96-33, LOWER SLOW 2-HS-96-34, LOWER MEDIUM 2-HS-96-35, LOWER FAST

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 6	Page 4 of 5

Event Description: Reactor Power Reduction for Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-68, Reactor Recirculation System Section 8.12, Initiating Manual Runbacks **NOTES** 1) Manual runback controls are utilized when it becomes necessary to reduce Reactor Power and Core Flow during abnormal plant conditions. 2) This section is performed at Panel 2-9-5. 3) Depressing a manual runback pushbutton initiates a runback of both Recirc Pumps until the setpoint is reached. Depressing the pushbutton a second time stops the manual runback. The pushbutton can be depressed a third and fourth time to reinitiate and stop the manual runback. This pattern can be repeated until the applicable setpoint is reached. 4) Attachment 2 can be referred to for additional information on manual runback controls. 5) When initiating manual runbacks, the appropriate manual pushbutton must be depressed until the backlight is blinking, then the pushbutton can be released. 6) If ≥ 25 rpm mismatch in the lower direction exists between Speed Demand and Calculated Speed, the Manual Runback pushbuttons are disabled. 7) RECIRC PUMPS MID POWER RUNBACK is to be used any time a Condensate Pump trips and Reactor Power is greater than or equal to 90%. [1] **IF** time permits, **THEN REVIEW** Precautions and Limitations. (REFER TO Section 3.0).

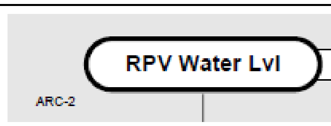
Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 6 Page 5 of 5

Event Description: Reactor Power Reduction for Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	OATC	[2] **IF** desired to reduce Reactor Power to approximately 90%, **THEN** (Otherwise N/A): [2.1] **DEPRESS** 2-HS-68-42, RECIRC PUMPS UPPER POWER RUNBACK Pushbutton. [2.2] **CHECK** the following: • Pushbutton backlight blinks until setpoint is reached • Reactor Power lowers to approximately 90% [3] **IF** desired to reduce Reactor Power to 66.3%, **THEN** (Otherwise N/A): [3.1] **DEPRESS** 2-HS-68-43, RECIRC PUMPS MID POWER RUNBACK pushbutton. [3.2] **CHECK** the following: • Pushbutton backlight blinks until setpoint is reached • Reactor Power lowers to 66.3% [4] **IF** desired to reduce Core Flow to approximately 60%, **THEN** (Otherwise N/A): [4.1] **DEPRESS** 2-HS-68-44, RECIRC PUMPS CORE FLOW RUNBACK Pushbutton. [4.2] **CHECK** the following: • Pushbutton backlight blinks until setpoint is reached • Core Flow lowers to approximately 60%
	NRC	**End of Event 6. Proceed to Event 7, Hydraulic Anticipated Transient Without SCRAM (ATWS).**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 1 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 7, Hydraulic Anticipated Transient Without SCRAM (ATWS) is inserted on Simulator Setup. No action is required by the Driver to insert Event 7.**
	Driver	**During Event 7, when contacted as the Outside NUSO acknowledge direction to perform EOI Appendices and enter events as necessary:** • **Event 17 – 2-EOI-Appendix-1D, Insert Control Rods Using Reactor Manual Control System (Close 2-FCV-85-586, CHARGING WATER ISOLATION)** • **Event 18 – Open 2-FCV-85-586, CHARGING WATER ISOLATION** • **Event 19 – 2-EOI-Appendix-1F, Manual SCRAM** • **Event 20 – 2-EOI-Appendix-2, Defeating ARI Logic Trips** • **Event 21 – 2-EOI-Appendix-8A, Bypassing Group RPV Low Low Low Level Isolation Interlocks** • **Event 22 – 2-EOI-Appendix-8E, Bypassing Group 6 RPV Low Level and High Drywell Pressure Isolation Interlocks** **Once the event(s) requested have finished their time delay, report completion of the various EOI Appendices to the Control Room.**
	NUSO	Enters 2-EOI-1A, ATWS RPV Control, and updates the crew.
	NUSO	 ARC/L-1 **ENSURE** each as required: • PCIS isolations (Groups 1, 2, and 3) • ECCS • RCIC ARC/L-2 **INHIBIT** ADS

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 2 of 13

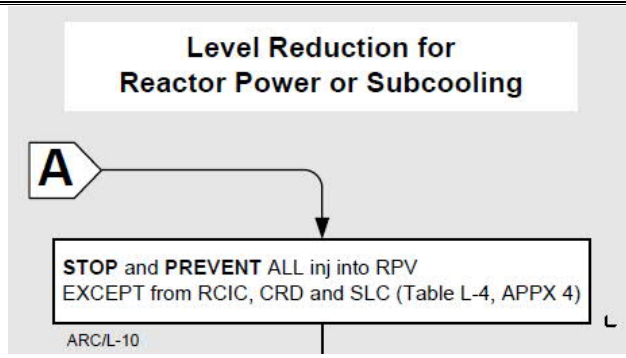
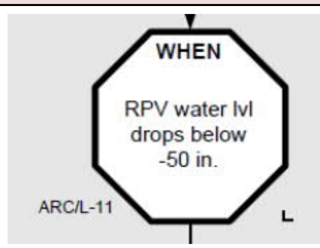
Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>CREW</td><td>Critical Task:
With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS.

Critical Task Failure Criteria:
ADS automatic initiation with Control Rods out.</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/L-3
IF ANY Main Steam Line is open
THEN START defeating the following isolations:
• MSIV Low Low Low RPV Water Level (2- EOI-Appendix 8A)
• Reactor Building Ventilation Low RPV Water Level (2-EOI-Appendix-8E)</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/L-4
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Reactor Power is above 5% or unknown
AND
Reactor Water Level is above (-) 50 inches</td><td>A</td></tr>
<tr><td>ALL Level/Power conditions exist (Table Q-1)</td><td>A</td></tr>
</table></td></tr>
<tr><td></td><td>NUSO</td><td>
<table>
<tr><th colspan="2">Table Q-1
Level/Power Conditions</th></tr>
<tr><td>• Suppression Pool Temperature is above 110ºF</td><td>☐</td></tr>
<tr><td>• Reactor Power above 5% OR unknown</td><td>☐</td></tr>
<tr><td>• RPV Level above -162 in.</td><td>☐</td></tr>
<tr><td>• MSRV open/cycling OR DW pressure above 2.4 psig</td><td>☐</td></tr>
</table></td></tr>
</table>

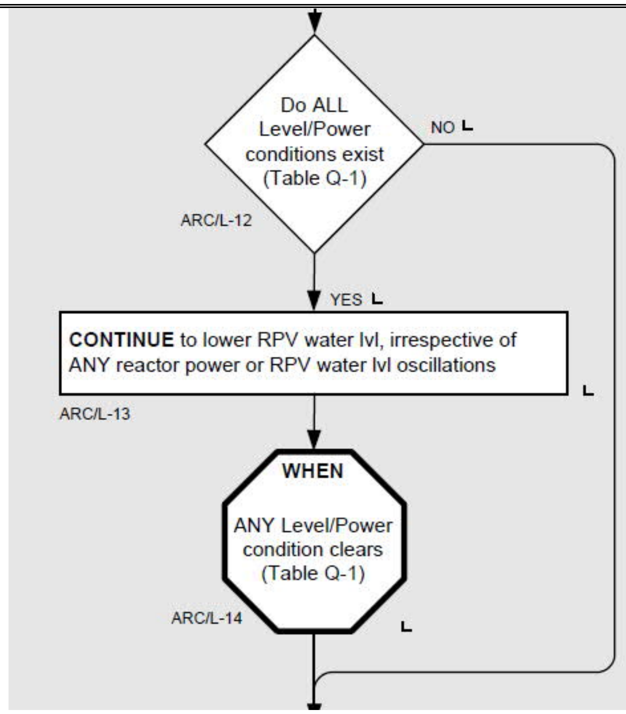
Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 3 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	
	Crew	**Critical Task:** **When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the loss of forced recirculation to prevent possible fuel damage.** **Critical Task Failure Criteria: Evaluate wording-EJL** **The operating crew fails to STOP and PREVENT injection and lower Reactor Water Level for Reactor Power control during an ATWS within 130 seconds after Recirculation Flow is stopped.**
	NUSO	

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 4 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
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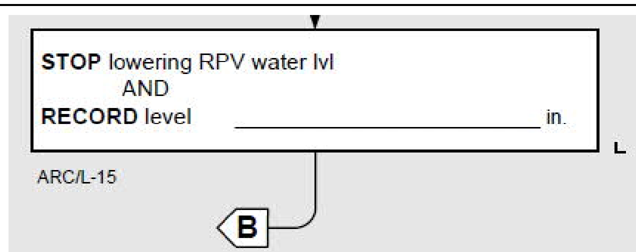
<tr><td></td><td>NUSO</td><td>
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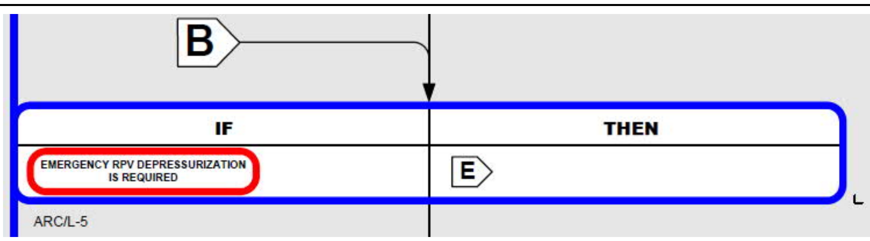
<tr><td></td><td>NUSO</td><td>
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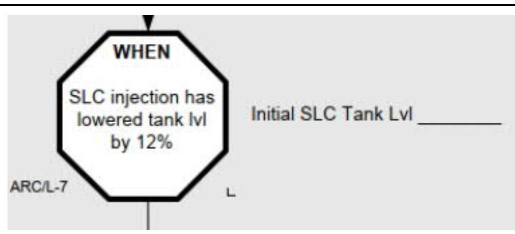
Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 5 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
ARC/L-6
<table>
<tr><td colspan="2">USE ANY Preferred ATWS Injection System (Table L-3) to maintain RPV Water Level between (-) 180 inches and:

Lowered level (if level was deliberately lowered in flowpath A)
OR
+51 inches (if level was NOT deliberately lowered)
➢ Ok to use Core Spray (2-EOI-Appendix-6D or 6E) or Alternate Injection Subsystems (Table L-2) if previously required by flowpath E or C4A</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Reactor Water Level CANNOT be restored and maintained above (-) 180 inches
AND
Core Steam Flow remains below MCSF (Table L-5)</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>


</td></tr>
<tr><td></td><td>NUSO</td><td>
<table>
<tr><th colspan="3">Table L-3
Preferred ATWS Injection Systems</th></tr>
<tr><th>SOURCES</th><th>APPX</th><th>INJ PRESS</th></tr>
<tr><td>CNDS and FW</td><td>5A</td><td>1210 psig</td></tr>
<tr><td>CRD</td><td>5B</td><td>1640 psig</td></tr>
<tr><td>RCIC with CST suction if available 2 3 6</td><td>5C, 20M</td><td>1200 psig</td></tr>
<tr><td>HPCI with CST suction if available 2 6 7</td><td>5D, 20N</td><td>1200 psig</td></tr>
<tr><td>CNDS</td><td>6A</td><td>480 psig</td></tr>
<tr><td>LPCI 2</td><td>6B, 6C</td><td>320 psig</td></tr>
<tr><td>SLC (boron tank)</td><td>7B</td><td>1450 psig</td></tr>
<tr><td>Table L-2 systems or CS
ONLY IF Step ARC/L-19 has been previously performed</td><td>----</td><td>----</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 6 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	Table L-2 Alternate Injection Subsystems (below)

Table L-2
Alternate Injection Subsystems

SOURCE	APPX	INJ PRESS
EHPM Pump	7L	1210 psig
SLC (test tank)	7B	1450 psig
SLC (boron tank)	7B	1450 psig
CNDS transfer pumps to RHR and CS	7A	110 psig
RHR crosstie to other units	7C	320 psig
Stby coolant	7D	160 psig
RHR drain pumps	7E, 7F	50 psig
PSC head tank pumps	7G	30 psig
RCIC (aux boiler steam) with CST suction if available 2 3 6	7H	1200 psig
RCIC manual start	20A	1200 psig
HPCI (aux boiler steam) with CST suction if available 2 6 7	7J	780 psig
Fire Protection system	7K	150 psig
FLEX Pump Sys (CILRT/CS)	20D	150 psig
FLEX Pump Sys (Standby Coolant)	20B	150 psig
FLEX Pump Sys (CILRT/CRD)	20C	150 psig

Time	Position	Applicant's Actions or Behavior
	NUSO	Table L-5 Minimum Core Steam Flow (below)

Table L-5
Minimum Core Steam Flow

MCSF is 1,100,000 lbm/hr and indicated by ANY:

- MSCP (Table P-3)
- Open TBPVs and RPV pressure above the following:

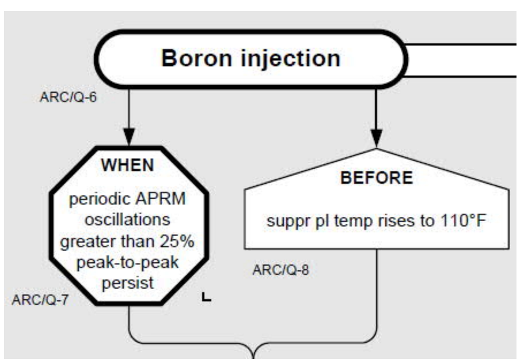
TBPV	RPV Pressure (psig)
2.1	1,100
2.3	1,000
2.6	900
2.8	800
3.2	700

- RPV injection flow greater than 2200 gpm
- Reactor power above 9.1%

Time	Position	Applicant's Actions or Behavior
	NUSO	**Reactor Power**

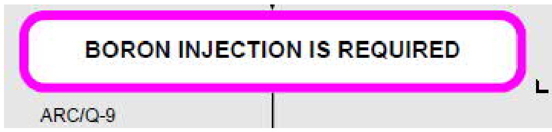
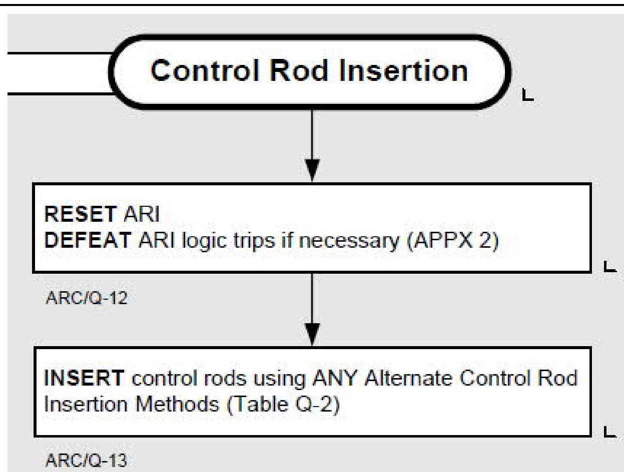
Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 7 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>The Reactor is subcritical
AND
NO boron has been injected</td><td>AOI-100-1
Reactor Scram</td></tr>
</table>
ARC/Q-2
ENSURE Reactor Mode Switch in SHUTDOWN
ARC/Q-3
INITIATE ARI
ARC/Q-4
IF tripping Recirc Pumps will cause loss of Main Turbine, RFPT, HPCI, or RCIC
THEN ENSURE Recirc Runback (pump speed 480 RPM or less)</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-5
IF Reactor Power is above 5% or unknown
THEN TRIP Recirc Pumps

</td></tr>
<tr><td></td><td>NUSO</td><td>In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, Section 4.8.4.C, when EOI-1A, ATWS RPV Control, Step ARC/Q-8 is reached, IF Reactor Power is greater than APRM downscale, THEN INITIATE SLC.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 8 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
</td></tr>
<tr><td></td><td>NRC</td><td>See Event 9 on page 43 of 50.</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-10
<table>
<tr><td colspan="2">1. INITIATE SLC (2-EOI-Appendix-3A)
2. INHIBIT ADS</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Boron CANNOT be injected using SLC</td><td>INJECT boron into RPV using CRD (2-EOI-Appendix-3B)</td></tr>
<tr><td>SLC Tank Water Level drops to 0%</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-11
<table><tr><td>ENSURE RWCU System Isolation</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>
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Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 9 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
<table>
<tr><th colspan="3">Table Q-2
Alternate Control Rod Insertion Methods</th></tr>
<tr><th>CONDITIONS</th><th>METHODS</th><th>APPX</th></tr>
<tr><td rowspan="2">Scram valves failed to open</td><td>DEENERGIZE scram solenoids</td><td>1A</td></tr>
<tr><td>VENT scram air header</td><td>1B</td></tr>
<tr><td>Scram valves opened but SDV is full</td><td>1. RESET scram DEFEAT RPS logic if necessary
2. DRAIN SDV
3. RECHARGE accumulators
4. INITIATE scram</td><td>1F</td></tr>
<tr><td rowspan="4">Manual control rod insertion methods</td><td>DRIVE control rods BYPASS RWM and RAISE CRD drive water differential pressure if necessary</td><td>1D</td></tr>
<tr><td>RAISE CRD cooling water header dp</td><td>1G</td></tr>
<tr><td>SCRAM individual control rods</td><td>1C</td></tr>
<tr><td>VENT control rod over piston volumes</td><td>1E</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>RPV Press</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/P-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>A high Drywell Pressure ECCS signal exists (2.45 PSIG)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>EMERGENCY DEPRESSURIZATION is required or has been required</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 10 of 13

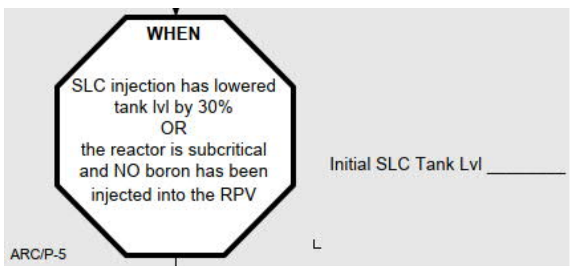
Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>ARC/P-2

IF ANY MSRV is cycling
THEN MANUALLY OPEN MSRVs UNTIL RPV press drops to the pressure at which all Main Turbine Bypass Valves are fully open (APPX 11A)</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/P-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Suppression Pool Temperature and Water Level CANNOT be maintained in the safe area of Curve 3 at the existing RPV Pressure</td><td>**NO ACTION REQUIRED**</td></tr>
<tr><td>Suppression Pool Water Level CANNOT be maintained in the safe area of Curve 4</td><td>**NO ACTION REQUIRED**</td></tr>
<tr><td>BORON INJECTION IS REQUIRED
AND
The Main Condenser is available
AND
There has been no indication of a steam line break</td><td>**NO ACTION REQUIRED**</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 11 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>ARC/P-4
<table><tr><td colspan="2">STABILIZE RPV Pressure below 1073 psig using the Main Turbine Bypass Valves (2-EOI-Appendix-8B)
➢ Use Alternate RPV Pressure Control Systems (Table P-1), if necessary
➢ Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary</td></tr><tr><th>IF</th><th>THEN</th></tr><tr><td>Drywell Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>
</td></tr>
<tr><td></td><td>OATC</td><td>2-EOI-Appendix-1F, Manual SCRAM

[1] VERIFY Reactor SCRAM and ARI reset.
[1.1] IF ARI <u>CANNOT</u> be reset, THEN EXECUTE EOI Appendix 2, Defeating ARI Logic Trips, concurrently with Step 1.0[1.2] of this procedure.
[1.2] IF Reactor SCRAM <u>CANNOT</u> be reset, THEN DISPATCH personnel to Unit 2 Auxiliary Instrument Room to defeat ALL RPS Logic trips.
[2] WHEN RPS Logic has been defeated, THEN RESET Reactor SCRAM.
[3] VERIFY OPEN SCRAM Discharge Volume Vent and Drain Valves.</td></tr>
<tr><td></td><td>OATC</td><td>Dispatches personnel to perform outside portions of 2-EOI-Appendix-1F, Manual SCRAM.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 12 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	[4] **DRAIN** SCRAM Discharge Volume (SDV) UNTIL the following annunciators clear: • WEST CRD DISCHARGE VOLUME WATER LEVEL HIGH HALF SCRAM (Panel 2-9-4, 2-XA-55-4A, Window 1). • EAST CRD DISCHARGE VOLUME WATER LEVEL HIGH HALF SCRAM (Panel 2-9-4, 2-XA-55-4A, Window 29).
	NRC	**The accumulators will drain in approximately 7 minutes, and the alarms at Panel 2-9-4, Windows 1 and 29, will clear. The OATC may then attempt a Reactor SCRAM.**
	OATC	[5] **DISPATCH** personnel to **VERIFY OPEN** 2-SHV-085-0586, CHARGING WATER ISOLATION. **NOTES** 1) If EOI Appendix 2 has been executed, ARI initiation or reset will NOT be possible or necessary in Step 1.0[6]. 2) If Reactor Pressure is greater than 600 PSIG, NUSO may direct performance of step 1.0[6] prior to accumulators being fully recharged. [6] **WHEN** CRD Accumulators are recharged, **THEN INITIATE** manual Reactor SCRAM and ARI.
	NRC	**Control Rods will insert the first time the OATC attempts a Reactor SCRAM after the ATWS.**
	OATC	[7] **CONTINUE** to perform Steps 1.0[1] through 1.0[6] UNTIL ANY of the following exists: • ALL Control Rods are fully inserted, **OR** • NO inward movement of Control Rods is observed, **OR** • NUSO directs otherwise. END OF EOI APPENDIX 1F

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 13 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	2--EOI-8A, Bypassing Group 1 RPV Low Low Low Level Isolation Interlocks [1] **BYPASS** Group 1 RPV Low-Low-Low Level Isolation Interlocks as follows (Unit 2 Control Room, Panel 9-4): [1.1] **PLACE** keylock switch 2-HS-064-0056A, GROUP 1 RPV LOW LEVEL BYPASS (SYS A1), in BYPASS. [1.2] **PLACE** keylock switch 2-HS-064-0056B, GROUP 1 RPV LOW LEVEL BYPASS (SYS B1), in BYPASS. [1.3] **PLACE** keylock switch 2-HS-064-0056C, GROUP 1 RPV LOW LEVEL BYPASS (SYS A2), in BYPASS. [1.4] **PLACE** keylock switch 2-HS-064-0056D, GROUP 1 RPV LOW LEVEL BYPASS (SYS B2), in BYPASS. [1.5] **ENSURE** closed the following valves (Unit 2 Control Room, Panel 9-3): • 2-FCV-43-13, RX RECIRC SAMPLE INBOARD ISOLATION VALVE • 2-FCV-43-14, RX RECIRC SAMPLE OUTBOARD ISOLATION VALVE [2] N/A [3] **NOTIFY** Unit Operator to ensure closed the following valves (Unit 2 Control Room, Panel 9-3): • 2-FCV-43-13, RX RECIRC SAMPLE INBOARD ISOLATION VALVE • 2-FCV-43-14, RX RECIRC SAMPLE OUTBOARD ISOLATION VALVE END OF EOI APPENDIX 8A
	NRC	**End of Event 7. When the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 8	Page 1 of 7

Event Description: 2A EHC Pump Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8, EHC Pump Trip, is inserted on Simulator Setup. No action is required by the Driver to insert this event.** **Thirty (30) seconds after the MODE SWITCH is placed in RUN, 2A EHC pump will be stopped.**
	BOP	Acknowledges and reports the following alarms when received: • STANDBY EHC PUMP FAILED, 2-9-7B, Window 15 • EHC HYDRAULIC FLUID HEADER PRESSURE LOW, 2-9-7B, Window 1
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure(s).
	BOP	Alarm Response Procedure, 2-ARP-9-7B STANDBY EHC PUMP FAILED, Window 15 Operator Action: A. **PERFORM** the following on Panel 2-9-7: 1. **CHECK** alarm by checking 2-PI-47-7, EHC HEADER PRESSURE. 2. **CHECK** 2-HS-47-2A, EHC PUMP 2B and/or 2-HS-47-1A, EHC PUMP 2A running. 3. **CHECK** 2-EI-47-2, EHC PUMP 2B PUMP MTR CURRENT and/or 2-EI-47-1, EHC PUMP 2A PUMP MTR CURRENT. **NOTE** Lights extinguish at 1300 psig lowering and illuminate at 1500 psig rising. 4. **CHECK** lights above 2-HS-47-4A, EHC PUMP 2A TEST pushbutton and 2-HS-47-5A, EHC PUMP 2B TEST pushbutton. B. **DISPATCH** personnel to pumping unit to check for abnormal conditions.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 8 Page 2 of 7

Event Description: 2A EHC Pump Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the outside NUSO, Work Control, Maintenance, or an AUO to investigate the cause for 2A EHC Pump Trip or to check for abnormal conditions, acknowledge the direction.**
	BOP	**NOTE** On EHC Hydraulic System failure accumulator and check valve arrangement will provide approximately one minute Bypass Valve operation. C. **IF** EHC Hydraulic System fails, **THEN ENSURE** Turbine trips at or below 1100 psig.
	BOP	Alarm Response Procedure, 2-ARP-9-7B EHC HYDRAULIC FLUID HEADER PRESSURE LOW, Window 1 Operator Action: A. N/A. B. **CHECK** EHC HEADER PRESSURE indicator, 2-PI-47-7 between 1550 and 1650 psig. C. **DISPATCH** personnel to inspect EHC Pump unit. **NOTE** On EHC Hydraulic System failure, accumulator and check valve arrangement will provide approximately one minute Bypass Valve operation. D. **IF** EHC Hydraulic system fails, **THEN ENSURE** Turbine trips at or below 1100 psig.
	NUSO	Directs the BOP to maintain Reactor Pressure with Main Steam Relief Valves (MSRVs) using 2-EOI-Appendix-11A, Alternate RPV Pressure Control Systems MSRVs.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 8 Page 3 of 7

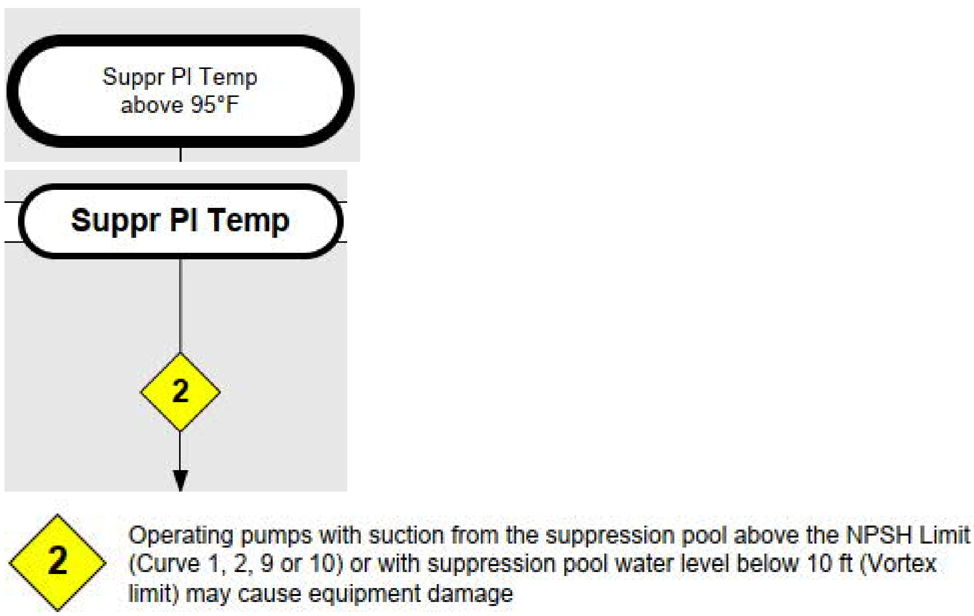
Event Description: 2A EHC Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	2-EOI-Appendix-11A, Alternate RPV Pressure Control Systems MSRVs. [1] N/A [2] N/A [3] **OPEN** MSRVs using the following sequence to control RPV pressure as directed by the SRO: 1 - 2-PCV-1-179 - MN STM LINE A RELIEF VALVE 2 - 2-PCV-1-180 - MN STM LINE D RELIEF VALVE 3 - 2-PCV-1-4 - MN STM LINE A RELIEF VALVE 4 - 2-PCV-1-31 - MN STM LINE C RELIEF VALVE 5 - 2-PCV-1-23 - MN STM LINE B RELIEF VALVE 6 - 2-PCV-1-42 - MN STM LINE D RELIEF VALVE 7 - 2-PCV-1-30 - MN STM LINE C RELIEF VALVE 8 - 2-PCV-1-19 - MN STM LINE B RELIEF VALVE 9 - 2-PCV-1-5 - MN STM LINE A RELIEF VALVE 10 - 2-PCV-1-41 - MN STM LINE D RELIEF VALVE 11 - 2-PCV-1-22 - MN STM LINE B RELIEF VALVE 12 - 2-PCV-1-18 - MN STM LINE B RELIEF VALVE 13 - 2-PCV-1-34 - MN STM LINE C RELIEF VALVE [4] N/A [5] N/A [6] N/A END OF EOI APPENDIX 11A
	BOP	Acknowledges and reports the following alarm when received: • SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, 2-9-3E, Window 12

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 8 Page 4 of 7

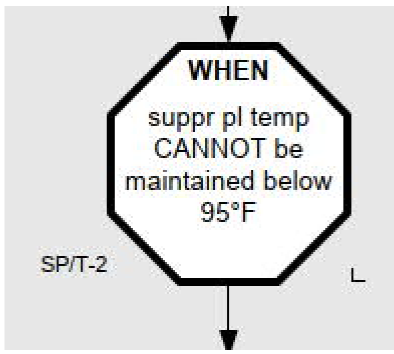
Event Description: 2A EHC Pump Trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.</td></tr>
<tr><td></td><td>BOP</td><td>(If received)
Alarm Response Procedure, 2-ARP-9-3E
SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, Window 12

Operator Action:
A. IF alarm is valid, THEN ENTER 2-EOI-2, Primary Containment Control.</td></tr>
<tr><td></td><td>NUSO</td><td>Enters 2-EOI-2, Primary Containment Control.</td></tr>
<tr><td></td><td>NUSO</td><td>
</td></tr>
</table>

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 8	Page 5 of 7

Event Description: 2A EHC Pump Trip

Time	Position	Applicant's Actions or Behavior
	NUSO	SP/T-1 **MONITOR** and **CONTROL** Suppr Pool Temperature below 95°F using available Suppr Pool Cooling (APPX 17A) SP/T-2  SP/T-3 **OPERATE** all available Suppression Pool Cooling using only RHR Pumps NOT required to assure adequate Core Cooling by continuous injection (APPX 17A)
	NUSO	Directs the BOP to place Suppression Pool Cooling in service in accordance with 2-EOI-Appendix-17A, RHR System Operation Suppression Pool Cooling.
	BOP	2-EOI-Appendix-17A, RHR System Operation Suppression Pool Cooling **NOTE** Placing a BYPASS SEL switch in BYPASS in Step 1.0[1] below prevents automatic opening of the affected RHR loop's Outboard Injection Valve. This makes LPCI Mode of that RHR Loop inoperable.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 8 Page 6 of 7

Event Description: 2A EHC Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[1] **IF** Adequate Core Cooling is assured OR directed to cool the Suppression Pool irrespective of Adequate Core Cooling, **THEN BYPASS** LPCI Injection Valve open interlock AS NECESSARY: • PLACE 2-HS-74-155A, LPCI SYSTEM I OUTBOARD INJECTION VALVE BYPASS SELECT in BYPASS • PLACE 2-HS-74-155B, LPCI SYSTEM II OUTBOARD INJECTION VALVE BYPASS SELECT in BYPASS [2] **PLACE** RHR SYSTEM I(II) in Suppression Pool Cooling as follows: [2.1] **ENSURE** at least one RHRSW Pump supplying each EECW Header. [2.2] **ENSURE** RHRSW Pump supplying desired RHR Heat Exchanger(s). [2.3] **THROTTLE** the following in service RHRSW Outlet Valves to obtain required RHRSW Flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE (Required flow is 1350 to 4500 gpm for B1 pump) (Required flow is 1700 to 4500 gpm for B2 pump) • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) [2.4] **IF** Directed by SRO, **THEN PLACE** 2-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE. [2.5] **IF** LPCI Initiation signal exists, **THEN MOMENTARILY PLACE** 2-XS-74-121(129), RHR SYS I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, in SELECT. [2.6] **IF** 2-FCV-74-53(67), RHR SYS I(II) LPCI INBOARD INJECTION VALVE, is OPEN, **THEN ENSURE CLOSED** 2-FCV-74-52(66), RHR SYSTEM I(II) LPCI OUTBOARD INJECTION VALVE.

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 8	Page 7 of 7
Event Description: 2A EHC Pump Trip			

Time	Position	Applicant's Actions or Behavior
	BOP	[2.7] **OPEN** 2-FCV-74-57(71), RHR SYS I(II) SUPPRESSION CHAMBER/POOL ISOLATION VALVE. [2.8] **ENSURE** desired RHR Pump(s) for Suppression Pool Cooling are operating. **CAUTION** RHR System Flows below 7,000 gpm or above 10,000 gpm for one pump operation may result in excessive vibration and equipment damage. [2.9] **THROTTLE OPEN** 2-FCV-74-59(73), RHR SYS I(II) SUPPRESSION POOL COOLING/TEST VALVE, to maintain EITHER of the following as indicated on 2-FI-74-50(64), RHR SYS I(II) FLOW: • Between 7,000 and 10,000 gpm for one pump operation **OR** • At or below 13,000 gpm for two pump operation [2.10] **ENSURE** CLOSED 2-FCV-74-7(30), RHR SYS I(II) MIN FLOW VALVE. [2.11] **MONITOR** RHR Pump NPSH using Attachment 1. [2.12] **NOTIFY** Chemistry that RHRSW is aligned to in service RHR Heat Exchangers.
	Driver	**When contacted as Chemistry, acknowledge any information given.**
	BOP	[2.13] **IF** Additional Suppression Pool Cooling Flow is necessary, **THEN PLACE** additional RHR and RHRSW Pumps in service using Steps 1.0[2.2] through 1.0[2.12]. [3] N/A END OF EOI APPENDIX 17A
	NRC	**End of Event 8. Once the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 9 Page 1 of 2

Event Description: SLC Pump Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9, SLC Pump Trip, is inserted on Simulator Setup. No action is required by the Driver to insert this event.** **NOTE: The first SLC Pump that is started will trip.**
	BOP	2-EOI-Appendix-3A, SLC Injection [1] **UNLOCK** and **PLACE** 2-HS-63-6A, SLC PUMP 2A/2B, control switch in START-A or START-B position. [2] **CHECK** SLC System for injection by observing the following: • Selected pump starts, as indicated by red light illuminated above pump control switch • Squib valves fire, as indicated by SQUIB VALVE A and B CONTINUITY blue lights extinguished • SLC SQUIB VALVE CONTINUITY LOST Annunciator in alarm on Panel 9-5 (2-XA-55-5B, Window 20) • 2-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV Pressure • System flow, as indicated by 2-IL-63-11, SLC FLOW, red light illuminated on Panel 9-5 • SLC INJECTION FLOW TO REACTOR Annunciator in alarm on Panel 9-5 (2-XA-55-5B, Window 14). [3] **IF** proper system operation CANNOT be verified, **THEN RETURN** to Step 1.0[1] and **START** other SLC pump.
	BOP	Determines that the first SLC Pump that was started trips, and starts the alternate SLC Pump.
	BOP	[4] **VERIFY** RWCU isolation by observing the following: • RWCU Pumps 2A and 2B tripped • 2-FCV-69-1, RWCU INBOARD SUCT ISOLATION VALVE closed • 2-FCV-69-2, RWCU OUTBOARD SUCT ISOLATION VALVE closed • 2-FCV-69-12, RWCU RETURN ISOLATION VALVE closed

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 9 Page 2 of 2

Event Description: SLC Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[5] **VERIFY** ADS inhibited. [6] **MONITOR** Reactor Power for downward trend. [7] **MONITOR** 2-LI-63-1A, SLC STORAGE TANK LEVEL, and **CHECK** that level is dropping approximately 1% per minute. [8] **WHEN** EITHER of the following exists: • SLC Tank Level drops to 0%, **OR** • As directed by SRO, **THEN** **STOP** SLC Pump 2A or 2B. [9] **NOTIFY** Chemistry to mix additional solution to compensate for dilution as directed by the SRO. [10] **WHEN** directed by the SRO to perform system flush, **THEN REFER** to OI-63, Standby Liquid Control System, Section 8.1, for system flush. END OF 2-EOI-APPENDIX-3A
	Driver	**If contacted as Chemistry, acknowledge any direction given.**
	NRC	**End of Event 9. Once the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Scenario Setup
UNIT 2

IC	28
Exam IC	277

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-68	159	ARP-3E	31	APPX-17A	18
OI-67	123	ARP-5A	60	APPX-1F	6
OI-92B	43	ARP-7B	36	APPX-8A	5
GOI-100-12A	118	ARP-8C	19		
0-AOI-27-1	12	ARP-20A	30		
AOI-47-3	22	EOI-1A	2		
AOI-85-6	21	EOI-2	16		
ODM 4.20	7	EOI-3	6		

Simulator Setup	Verify camera system is powered down (admin password = abcd1234) Start CPERF **PRIOR** to placing the Simulator in RUN Verify EECW Pump Alarm borders are properly arranged on Panels 2-9-23A / B / C / D. Hang Danger Tags on A3 EECW Pump and 2B EHC Pump. Hang Protected Equipment Tag on 2A EHC Pump.
Schedule Files(s):	2104 NRC Scenario 2 UNIT 2.sch
Event Files(s):	2104 NRC Scenario 2 UNIT 2.evt

Schedule File: 2104 NRC Scenario 2 UNIT 2.sch

Event	Action	Description
	2104 NRC Scenario 2 UNIT 2.evt	
2	Insert malfunction SW03M	RHRSW PUMP D3 TRIP
3	set nmaprmgain(1)=0.000005	Fails APRM 1 Low
4	Insert malfunction RD04R3011	DRIFT ANY CONTROL ROD OUT 30-11
14	Delete malfunction RD04R3011	DRIFT ANY CONTROL ROD OUT 30-11
5	Insert malfunction MC05 to 100.00000 in 600	INTAKE PUMPING STATION TRAVELING SCREENS CLOG

Schedule File: 2104 NRC Scenario 2 UNIT 2.sch

Event	Action	Description
7	Insert malfunction MC05 to 50.00000 in 120	INTAKE PUMPING STATION TRAVELING SCREENS CLOG
	Insert malfunction RD09A after 2 to 55.00000	ATWS (HYDRAULIC LOCK) EAST
	Insert malfunction RD09B after 2 to 55.00000	ATWS (HYDRAULIC LOCK) WEST
	Insert malfunction RD06ALLSTICK	STICK ALL CONTROL RODS
	Insert malfunction RD17A	EAST SDV LEVEL SWITCH FAILS LOW (LS-85-45E,F,G,H,K,M)
	Insert malfunction RD17B	WEST SDV LEVEL SWITCH FAILS LOW (LS-85-45A,B,C,D,J,L)
7	Insert override HS-47-1A after 30 to STOP	EHC HYD FLUID PUMP 2A
	Insert malfunction PMP-47-1 to FAIL_CCOIL	52_BREAKER EHC HYD FLUID PUMP A
	Insert malfunction SL01A	SLC PUMP 2A TRIP
	Insert malfunction SL01B	SLC PUMP 2B TRIP
9	Delete malfunction SL01A	SLC PUMP 2A TRIP
10	Delete malfunction SL01B	SLC PUMP 2B TRIP
15	Insert override ZLOXI2783A_1 to Off	XI-27-83A TRAVELING SCREEN AB
15	Insert override ZLOXI2783A_2 to On	XI-27-83A TRAVELING SCREEN AB
15	Insert override ZLOXI2782A_1 to Off	XI-27-82A TRAVELING SCREEN AA
15	Insert override ZLOXI2782A_2 to On	XI-27-82A TRAVELING SCREEN AA
5	Delete override ZLOXI2787A_1 after 480	XI-27-87A TRAVELING SCREEN CB
5	Delete override ZLOXI2787A_2 after 480	XI-27-87A TRAVELING SCREEN CB
5	Delete override ZLOXI2786A_1 after 480	XI-27-86A TRAVELING SCREEN CA
5	Delete override ZLOXI2786A_2 after 480	XI-27-86A TRAVELING SCREEN CA
17	Insert remote RD06 to CLOSE	CRD CHARGING WATER VALVE FCV-2-85-586
18	Insert remote RD06 to OPEN	CRD CHARGING WATER VALVE FCV-2-85-586

Schedule File: 2104 NRC Scenario 2 UNIT 2.sch

Event	Action	Description
22	Schedule F:\2104\NRC\Scenarios\U2\Scenario 2\App. 8E.sch	
7	Delete malfunction RD17A	EAST SDV LEVEL SWITCH FAILS LOW (LS-85-45E,F,G,H,K,M)
7	Delete malfunction RD17B	WEST SDV LEVEL SWITCH FAILS LOW (LS-85-45A,B,C,D,J,L)
27	Delete malfunction RD09A	ATWS (HYDRAULIC LOCK) EAST
27	Delete malfunction RD09B	ATWS (HYDRAULIC LOCK) WEST
27	Delete malfunction RD06ALLSTICK	STICK ALL CONTROL RODS
19	Schedule F:\2104\NRC\Scenarios\U2\Scenario 2\App. 1F.sch	
20	Schedule F:\2104\NRC\Scenarios\U2\Scenario 2\App. 2.sch	
21	Schedule F:\2104\NRC\Scenarios\U2\Scenario 2\App. 8A.sch	

Schedule File: App. 1F.sch

Event	Action	Description
	Insert remote RP13A to BYP after 300	DEFEAT CHANNEL A1 AUTO SCRAM
	Insert remote RP13B to BYP after 300	DEFEAT CHANNEL B1 AUTO SCRAM
	Insert remote RP13C to BYP after 300	DEFEAT CHANNEL A2 AUTO SCRAM
	Insert remote RP13D to BYP after 300	DEFEAT CHANNEL B2 AUTO SCRAM

Schedule File: App. 2.sch

Event	Action	Description
	Insert remote RP12A to TEST after 300	OPERATE LOCAL ATWS MODE SWITCH 2-HS-68-118A
	Insert remote RP12B to TEST after 300	OPERATE LOCAL ATWS MODE SWITCH 2-HS-68-118B

Schedule File: App. 8A.sch

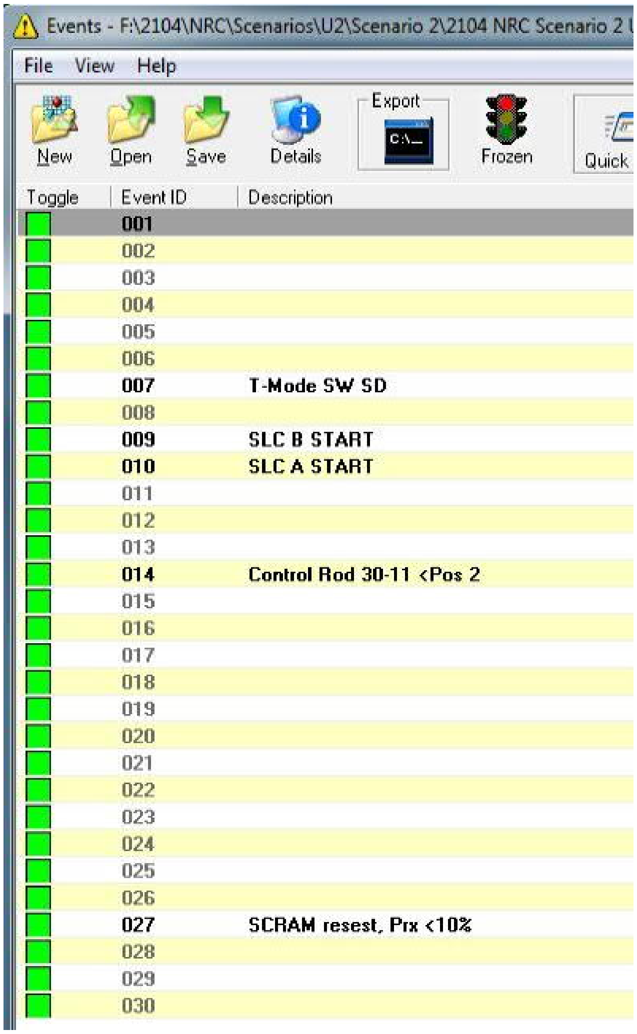
Event	Action	Description
	Insert remote RP06A to BYP after 300	BYP MSIV GP 1 LO LVL -APPNDX 8A-16A-K1A
	Insert remote RP06B to BYP after 300	BYP MSIV GP 1 LO LVL -APPNDX 8A-16A-K1B
	Insert remote RP06C to BYP after 300	BYP MSIV GP 1 LO LVL -APPNDX 8A-16A-K1
	Insert remote RP06D to BYP after 300	BYP MSIV GP 1 LO LVL -APPNDX 8A-16A-K1D

Schedule File: App. 8E.sch

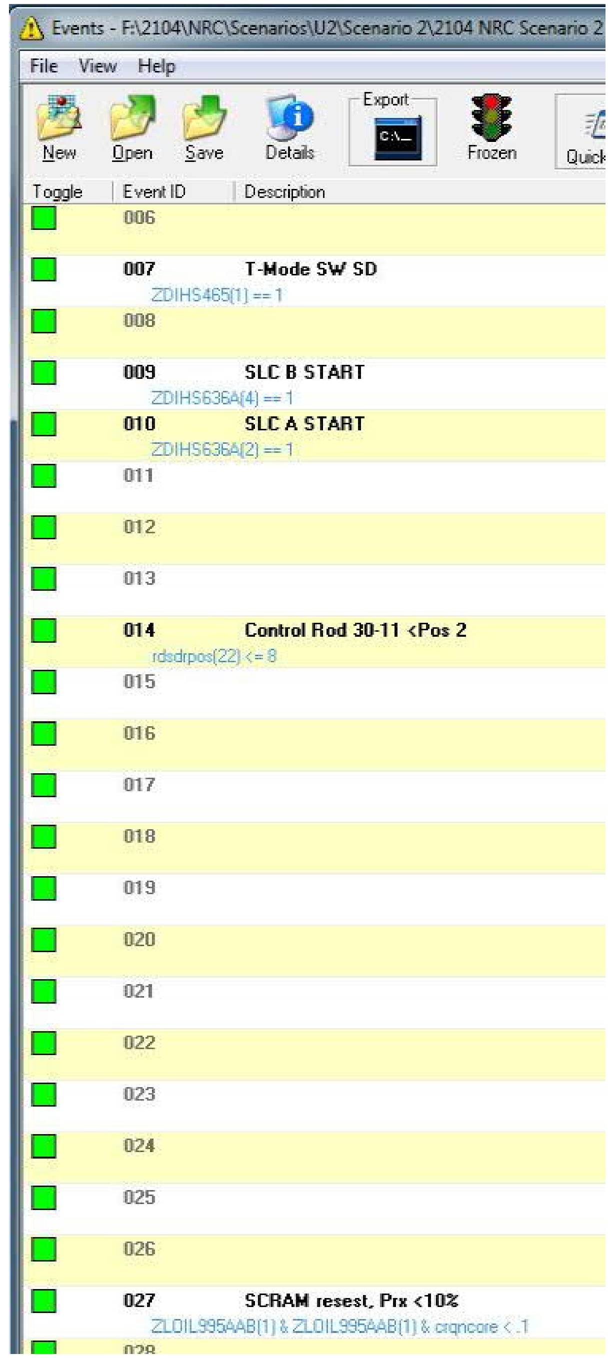
Event	Action	Description
	Insert remote RP14A to BYP after 300	BYPASS GROUP 6 INBOARD LOW LVL/HI DW PRESS INTLKS
	Insert remote RP14B to BYP after 300	BYPASS GROUP 6 OUTBOARD LOW LVL/HI DW PRESS INTLKS

Event File: 2104 NRC Scenario 2 UNIT 2.evt

List



Details



UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 1	DAYS ON LINE 227	Total Drywell Leakage (gpm) 1.55	Protected Equipment
	PRA (EOOS) -GREEN		2A EHC Pump
Rx Power 100%	500Kv GRID - Qualified 161Kv Grid -Qualified	Floor Drain (gpm) 0.11	
MWe 1308	Last breaker closure 10/01/20 4:31	Equipment Drain (gpm) 1.44	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

A3 EECW Pump is tagged for oil change (information only LCO).

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

2B EHC Pump tagged for discharge filter replacement.

2A CCW Pump repairs are complete, tags are cleared. Ready to re-start when Maintenance is ready.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Continue to support A3 EECW and 2B EHC Pump maintenance.

Alternate Recirc Drive Cooling Water Pumps.

OPERATOR WORK AROUNDS **OWAs - 1*** **Burdens - 0** **Challenges - 7**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 1 Page 1 of 2

Event Description: Swap Recirc Drive Cooling Water Pumps

Time	Position	Applicant's Actions or Behavior
	Driver	**Prior to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not start Event 1, Swap Recirc Drive Cooling Water Pumps, after assuming the shift, request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to swap Recirc Drive Cooling Water Pumps.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to swap Recirc Drive Cooling Water Pumps.**
	NUSO	Directs the Balance of Plant Operator (BOP) to swap Recirc Drive Cooling Water Pumps in accordance with 3-OI-68, Reactor Recirculation System.
	BOP	3-OI-68, Reactor Recirculation System Section 6.3, Swapping Recirc Drive Cooling Water Pumps **NOTES** 1) Perform these steps as required to swap the Recirc Drive Cooling Water Pumps. 2) Placing the standby pump in RUN will cause the running pump to shutdown after ~2 seconds if the running pump is in AUTO. 3) The red light indication above the MCR handswitch only indicates that the motor starter has been energized. A successful pump start should be verified locally or by ICS flow indication. 4) ICS screen VFDPMPA(VFDPMPB) may be referred to observe Recirc Drive cooling water system parameters. 5) The time both Cooling Water Pumps are running should be minimized. The pump being placed in standby should be placed in AUTO as soon as possible after placing the lead pump in RUN. [1] **IF** it is desired to place Recirc Drive Cooling Water Pump 3A2 in service and place 3A1 Pump in standby, **THEN PERFORM** the following: (Otherwise N/A) [1.1] 3-HS-96-13, **DEPRESS** FAULT RESET on Panel 3-9-4.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 1 Page 2 of 2

Event Description: Swap Recirc Drive Cooling Water Pumps

Time	Position	Applicant's Actions or Behavior
	BOP	[1.2] **PLACE** in RUN 3-HS-68-3A2/A, RECIRC DRIVE 3A COOLING PUMP 3A2. [1.3] **CHECK** RECIRC DRIVE 3A COOLING PUMP 3A2 starts. [1.4] **PLACE** in AUTO 3-HS-68-3A1/A, RECIRC DRIVE 3A COOLING PUMP 3A1. [1.5] **CHECK** RECIRC DRIVE 3A COOLING PUMP 3A1 stops. [2] N/A [3] **IF** it is desired to place Recirc Drive Cooling Water Pump 3B2 in service and place the B1 Pump in standby, **THEN PERFORM** the following: (Otherwise N/A) [3.1] **DEPRESS** FAULT RESET, 3-HS-96-14 on Panel 3-9-4. [3.2] **PLACE** in RUN RECIRC DRIVE 3B COOLING PUMP 3B2, 3-HS-68-3B2/A. [3.3] **CHECK** RECIRC DRIVE 3B COOLING PUMP 3B2 STARTS. [3.4] **PLACE** in AUTO 3-HS-68-3B1/A, RECIRC DRIVE 3B COOLING PUMP 3B1. [3.5] **CHECK** RECIRC DRIVE 3B COOLING PUMP 3B1 STOPS. [4] N/A
	BOP	Informs the Nuclear Unit Senior Operator (NUSO) that Recirc Drive Cooling Water Pumps have been swapped.
	NRC	**End of Event 1. Request that the driver insert Event 2, D3 EECW Pump Trip.**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 2 Page 1 of 3

Event Description: EECW Pump Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 2, EECW Pump Trip to trip C3 EECW Pump.**
	BOP	Acknowledges and reports the following alarms as received to the NUSO: • MOTOR TRIPOUT, 3-9-8C, Window 33 • EECW NORTH HEADER DG SECTION PRESSURE LOW, 3-9-20A, Window 21 Informs the NUSO that C3 EECW Pump has tripped.
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 3-ARP-9-8C MOTOR TRIPOUT, Window 33 Operator Action: A. **CHECK** Control Room for white disagreement light illuminated for affected equipment. B. **CLEAR** disagreement light. C. **DISPATCH** personnel to CHECK: 1. Relays at associated electrical board. 2. Equipment for abnormal conditions. 3. Safe-stop locally reset, if necessary.
	Driver	**If contacted as the Outside NUSO, Assistant Unit Operator (AUO), or Electrical Maintenance to investigate the trip of C3 EECW Pump, acknowledge the direction.**
	BOP	D. **REFER TO** 0-GOI-300-2, Electrical if relay targets are present or for motor starting limits. E. **REFER TO** appropriate OI for recovery or realignment of equipment.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 2 Page 2 of 3

Event Description: EECW Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 3-ARP-9-20A EECW NORTH HEADER DG SECTION PRESSURE LOW, Window 21 A. **CHECK** indications on Panel 3-9-20. 1. Unit 2-3 N HDR PRESSURE 0-PI-67-23/3 2. EECW N HDR PUMP A FLOW, 0-FI-67-3A/3 3. EECW N HDR PUMP C FLOW, 0-FI-67-9A/3 B. **CHECK** Panel 3-9-3 for status of North header pump(s) breaker lights and pump motor amps normal. C. **NOTIFY** Unit Supervisor, U1 and U2.
	Driver	**If contacted as the Unit 1 and/or Unit 2 NUSO, acknowledge any information given.**
	BOP	D. **START** standby pump for affected header. **REFER TO** 0-OI-67, Emergency Equipment Cooling Water System. E. **DISPATCH** Personnel to check affected pump room and header for abnormal conditions. F. N/A G. N/A H. **IF** A3 or C3 Pump failure is cause of alarm, **THEN REFER TO** Technical Specification 3.7.2, Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS). I. N/A
	NUSO	Directs the BOP to start C3 A3 EECW Pump.
	BOP	0-OI-67, Emergency Equipment Cooling Water System Precautions and Limitations C. The EECW System is aligned as follows: 1. At least one RHRSW Pump, assigned to the EECW System, should be running on each header to maintain the header charged at all times. If no pumps are running on a header and header pressure lowers to ≤ 0 psig, the header shall be declared inoperable and appropriate actions taken, as required by Technical Specifications.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 2 Page 3 of 3

Event Description: EECW Pump Trip

Time	Position	Applicant's Actions or Behavior	
	BOP	Starts A3 EECW Pump.	
	NUSO	References Technical Specification 3.7.2, Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS). LCO 3.7.2: The EECW System with three pumps and UHS shall be OPERABLE. APPLICABILITY: MODES 1, 2, and 3. **CONDITION:** A. One required EECW Pump INOPERABLE.	
	NUSO	**REQUIRED ACTION:** A.1 Restore the required EECW Pump to OPERABLE status.	**COMPLETION TIME:** A.1 – 7 days
	NRC	**End of Event 2. Request that the Driver insert Event 3, APRM 1 Fails Downscale.**	

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 3 Page 1 of 3

Event Description: APRM 1 Fails Downscale

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3, APRM 1 Fails Downscale.**
	OATC	Acknowledges and reports the following alarms: • APRM DOWNSCALE / OPRM INOPERABLE, 3-9-5A, Window 4 • CONTROL ROD WITHDRAWAL BLOCK, 3-9-5A, Window 7
	NUSO	Directs the Operator at the Controls (OATC) to respond in accordance with the appropriate Alarm Response Procedures.
	OATC	Alarm Response Procedure, 3-ARP-9-5A APRM DOWNSCALE / OPRM INOP, Window 4 Operator Action: A. **DETERMINE** which APRM/OPRM channel is downscale/inoperable. B. **IF** APRM failed downscale, **THEN BYPASS** channel. **REFER TO** 3-OI-92B, Average Power Range Monitoring. C. N/A D. N/A E. **REFER TO** Technical Specification (Tech Spec) Tables 3.3.1.1-1, Reactor Protection System Instrumentation, and Technical Requirements Manual (TRM) Table 3.3.4-1, Control Rod Block Instrumentation.
	OATC	Recommends to the NUSO that APRM 1 be bypassed.
	NRC	**EXAMINER NOTE: The NUSO may reference Tech Specs and the TRM. There are no required Tech Spec or TRM actions, but the NUSO may enter an Information Only LCO.**
	NUSO	Directs the OATC to bypass APRM 1 in accordance with the appropriate Operating Instruction.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 3 Page 2 of 3

Event Description: APRM 1 Fails Downscale

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-92B, Average Power Range Monitoring Section 6.0, System Operations **NOTES** 1) Only one APRM/OPRM in each trip system can be bypassed at a time. 2) All operations are performed on Panel 3-9-5 unless specifically stated otherwise. 3) In order to prevent inadvertent rod withdrawal block or Reactor SCRAM while operating APRM BYPASS selector switch, always ensure the previously bypassed channel returns to normal status by observing the BLUE bypassed lights on Panel 3-9-14. Voters are extinguished prior to selecting any other channel to be bypassed. After bypassing a channel, the applicable BLUE BYPASSED status lights on Panel 3-9-14 Voters should be illuminated prior to testing, operating, or working on that channel.
	OATC	Section 6.1, Bypassing APRM / OPRM Channel **CAUTION** NPG-SPP-10.4, Reactivity Management Program, requires approval of the Plant Manager or his designee prior to any planned operation with APRMs bypassed unless bypassing is specifically allowed within approved procedures. [1] **REVIEW** all Precautions and Limitations. **REFERENCE** Section 3.0. [2] **PLACE** APRM BYPASS, 3-HS-92-7B/S3, to desired channel to be bypassed. (APRM 1) [3] **CHECK** BLUE BYPASSED lights illuminated on Panel 3-9-14 Voters. [4] **CHECK** white bypass light on Panel 3-9-5 is illuminated.
	NRC	**EXAMINER NOTE: The blue APRM bypassed lights on Panel 3-9-14 are not modeled on the Unit 3 Simulator. Candidates can only verify that the APRM 1 white bypassed light on Panel 3-9-5 is illuminated.**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 3 Page 3 of 3

Event Description: APRM 1 Fails Downscale

Time	Position	Applicant's Actions or Behavior
	OATC	Alarm Response Procedure, 3-ARP-9-5A CONTROL ROD WITHDRAWAL BLOCK, Window 7 Operator Action: A. DETERMINE initiating condition from corresponding rod withdrawal block alarm(s) and REFER TO Operator Action for alarm(s). B. N/A C. N/A D. N/A E. N/A
	NRC	**End of Event 3. Request that the driver insert Event 4, Control Rod Drifts Out.**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 4 Page 1 of 3

Event Description: Control Rod Drifts Out

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4, Control Rod Drift Out.**
	NRC	**Control Rod 22-15 will drift out.**
	OATC	Acknowledges and reports the following alarm to the NUSO: • CONTROL ROD DRIFT, 3-9-5A, Window 28
	NUSO	Directs the OATC to respond in accordance with the appropriate Alarm Response and Abnormal Operating Procedures.
	OATC	Alarm Response Procedure, 3-ARP-9-5A CONTROL ROD DRIFT, Window 28 Operator Action: A. DETERMINE which rod is drifting from Full Core Display. B. IF no Control Rod motion is observed, THEN RESET rod drift as follows: 1. PLACE ROD DRIFT ALARM TEST switch, 3-HS-85-3A-S7, in RESET and RELEASE. 2. RESET the annunciator. C. N/A D. IF rod drifting out, THEN REFER TO 3-AOI-85-6, Rod Drift Out and 3-AOI-85-7, Mispositioned Control Rod E. REFER TO Tech Spec Section 3.1.3, Control Rod Operability and 3.10.8, Shutdown Margin (SDM) Test – Refueling.
	NRC	**The Control Rod Drift condition will clear when the Control Rod is driven to Position 0.**
	OATC	3-AOI-85-6, Rod Drift Out Immediate Actions: [1] N/A Subsequent Actions: [1] N/A

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 4 Page 2 of 3

Event Description: Control Rod Drifts Out

Time	Position	Applicant's Actions or Behavior
	OATC	[2] **IF** a Control Rod is moving from its intended position without operator actions, **THEN SELECT** the drifting Control Rod and **INSERT** to the FULL IN (00) position. [3] **IF** a Control Rod Block occurs during rod insertion due to Rod Worth Minimizer, **THEN BYPASS** the RWM per step 4.2[1]. (Otherwise N/A) [4] N/A [5] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern.
	Driver	**When contacted as the Reactor Engineer acknowledge any direction given.**
	OATC	[6] **IF** another Control Rod Drift occurs before Reactor Engineering completes the evaluation, **THEN MANUALLY SCRAM** Reactor and enter 3-AOI-100-1, Reactor SCRAM. [7] N/A [8] **IF** the Control Rod is latched into position "00", **THEN REMOVE** associated HCU from service per 3-OI-85, Control Rod Drive System. (N/A if Control will not latch at "00".) [9] **DECLARE** Control Rod INOP per Tech Spec 3.1.3. [10] **REFER** TO 3-AOI-85-7 Mispositioned Control Rod. [11] **INITIATE** Condition Report/Work Order. [12] **NOTIFY** Reactor Engineer to perform the following for current condition: • **EVALUATE** condition of core to assure no resultant fuel damage has occurred • **EVALUATION** of impact on thermal limits and PCIOMOR restraints (N/A if scram was initiated) • **DETERMINE** if other control rods need to be repositioned in order to safely restore core symmetry to prevent local fuel damage (N/A if scram was initiated) [13] **NOTIFY** System Engineering to **PERFORM** 0-TI-20, Control Rod Drive System Testing and Troubleshooting to determine problem with faulty Control Rod.

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: x	Page x of x

Event Description:

Time	Position	Applicant's Actions or Behavior	
	OATC	[14] **IF** a manual SCRAM was not inserted and Reactor Startup or Shutdown is not in progress, **THEN ENSURE** 3-GOI-100-12, Power Maneuvering, has been entered if a power change occurred. (Otherwise N/A) [17] **NOTIFY** Reactor Engineer to EVALUATE impact on preconditioning envelope, prior to returning to normal power operation.	
	NUSO	Technical Specification 3.1.3, Control Rod OPERABILITY LCO 3.1.3 Each Control Rod shall be OPERABLE Applicability: Modes 1 and 2 --- NOTE: Separate Condition entry is allowed for each Control Rod. --- **CONDITION:** C. One or more Control Rods INOPERABLE for reasons other than Condition A or B.	
	NUSO	**REQUIRED ACTION:** C.1 --------------NOTE----------------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of INOPERABLE Control Rod and continued operation. --------------------------------------- Fully insert INOPERABLE Control Rod AND C.2 Disarm the associated CRD.	**COMPLETION TIME:** C.1 – 3 hours C.2 – 4 hours
	NRC	**Tech Spec 3.10.8, Shutdown Margin (SDM) Test – Refueling is not applicable to current plant conditions.**	
	NRC	**End of Event 4. Request that the Driver insert Event 5, Clogged Traveling Screens / Lowering Condenser Vacuum.**	

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 5	Page 1 of 6
Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum			

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5, Clogged Traveling Screens / Lowering Condenser Vacuum.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • TRAVELING SCEEN DP HIGH, 3-9-20A, Window 18
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 3-ARP-9-20A TRAVELING SCEEN DP HIGH, Window 18 Operator Action: A. **CHECK** alarm on 3-PDI-27-1A, TRAVELING SCREEN DIFF WTR LEVEL, on Panel 3-9-20. B. **DISPATCH** personnel to ENSURE the traveling screens in service. Refer to 3-OI-27A, Screen Wash System. C. **MONITOR** Traveling Screens for carryover. D. **MONITOR** Turbine Backpressure. E. **IF** debris is being carried over, **THEN** • **MONITOR** 0-PDIS-067-0001(0005)(0008)(0011), EECW SUPPLY STRAINER DIFF PRESS locally in RHRSW Pump Rooms • **MONITOR** Waterbox DP for indications of fouling (< 160" H2O (does not apply to 3A1 Waterbox) with 3 CCW pumps in service). F. **IF** TRAVELING SCREEN DIFF WTR LVL, 3-PDI-27-1A, does not lower, **THEN** • **REFER** TO 3-OI-27A, Screen Wash System • **REFER** TO 0-AOI-27-1, Component Biofouling • **REQUEST** Mechanical Maintenance to SCRAPE the trash racks and/or **OPERATE** Milfoil harvester as needed. G. **IF** Divers are required to clear the trash racks, **THEN REMOVE** the Amertap System from service per OI-27B, Amertap Condenser Tube Cleaning System.

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 5	Page 2 of 6
Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum			

Time	Position	Applicant's Actions or Behavior
	BOP	0-AOI-27-1, Component Biofouling Immediate Actions: NONE Subsequent Actions: **NOTES** 1) Procedure is written in a logical order but due to changing plant conditions and operator experience, steps may be performed in parallel or out of sequence as required. 2) The most common cause of degraded cooling water system performance is the fouling of intake screens for the CCW Pumps. When intake screens begin to foul they are required to be placed in service as soon as possible using 1(2,3)-OI-27A, Screen Wash System. The timely response to this condition will keep screens from becoming over burdened with foreign material and collapsing. 3) If CCW Intake Screens cannot be cleaned with associated CCW Pump running, the pump may have to be removed from service in order to clean screens. After screens are cleaned the affected CCW Pump may be returned to service (1(2,3)-OI-27, Condenser Circulating Water System) if desired. 4) Entry into this procedure requires evaluation of situation per EOOS Management NPG-SPP-09.11.1, Equipment Out of Service Management. CAUTION Debris Filter may cycle repeatedly when total CCW flow is throttled excessively or due to heavy debris carryover from the intake. Debris Filter Flush Valve Motor is not rated for heavy repeated cycling. CCW flow should be maximized by throttling open available waterboxes until pump head is as low as possible (>20" H2O), Debris Filter should be run in manual and checked often. Expedite returning CCW System to normal three pump alignment with Debris Filter in AUTO.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 3 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	BOP	[1] **CONTACT** Maintenance to **PERFORM** attachment 7.
	Driver	**When contacted as Maintenance to perform Attachment 7, acknowledge the direction.**
	BOP	[2] **CHECK** CCW Intake Screens for fouling. [3] **IF** CCW Intake Screens are fouled, **THEN ENSURE** in service per 3-OI-27A, Screen Wash System. (Otherwise N/A).
	Driver	**If contacted as the Outside NUSO, Maintenance, or an Assistant Unit Operator (AUO) to check intake screens for fouling, acknowledge the direction. Wait 2 minutes and report that the intake screens are becoming fouled by Eel Grass.**
	BOP	[3] **IF** CCW Intake Screens are fouled, **THEN ENSURE** in service per 3-OI-27A, Screen Wash System. (Otherwise N/A). [4] **INITIATE** Attachment 1, Continuous Action Summary.
	NUSO / BOP	0-AOI-27-1, Component Biofouling Attachment 1, Continuous Action Summary Action Summary [1] **IF** at any time any of the following condition occurs: • Unexpected fouling indication of more than one river water supplied heat exchanger. **THEN** Action II is applicable, GO TO Step 4.2[9]. (Otherwise N/A) [2] **IF** at any time any of the following conditions occur: • Any indications of abnormal operation of the Circulating Water System OR • Any removal of two or more Circulating Water Pumps from service **THEN GO TO** 1(2)(3)-OI-27, Circulating Water System (Otherwise N/A)

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 4 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **IF** at any time a low reservoir level (<550') OR High Traveling Screen DP occur **THEN**: (Otherwise N/A) • **MONITOR** CCW Pumps for loss of suction/cavitation. [4] **IF** at any time the CCW Pumps indicate a loss of suction or cavitation **THEN**: (Otherwise N/A) • **SHUTDOWN** the CCW Pumps. **REFER TO** 3-OI-27, Circulating Water System.
	NRC	**As the malfunction ramps in, Condenser Vacuum will continue to lower. The crew may enter 3-AOI-47-3, Loss of Condenser Vacuum before an alarm is received.**
	BOP	3-AOI-47-3, Loss of Condenser Vacuum Immediate Actions: NONE Subsequent Actions: [1] **IF** ANY EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). **CAUTION** Operations outside of the allowable regions shown on the Recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. **REFER TO** 3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, for required actions and monitoring to be performed during a power decrease. [2] **MONITOR** Condenser Vacuum (Turb Exhaust) Margin to Trip using 3-XR-002-0026, CONDENSATE, Channel 7. [3] **IF** Condenser Vacuum (Turb Exhaust) Margin to Trip as indicated on 3-XR-002-0026, CONDENSATE, approaches 0 inches Hg, with Reactor Power less than 26%, **THEN TRIP** the Main Turbine.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 5 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **IF** Condenser Vacuum is lost, **THEN OPEN** 3-DRV-043-1019, HOTWELL SAMPLE DRAIN TO FLOOR DRAIN (565'@ T-16 D Line) and CONDENSATE DEMIN SAMPLE TO CRW VLV, 3-DRV-043-1061, (565' @T-12 G Line) to establish flow through the sample lines.
	Driver	**If contacted as an AUO to perform any outside steps of this procedure, acknowledge any direction given.**
	BOP	[5] **REDUCE** Reactor Power in an attempt to maintain Condenser Vacuum.
	NRC	**See Event 6, Reactor Power Reduction for Lowering Condenser Vacuum for Reactor Power reduction actions.**
	BOP	[6] **ENSURE** automatic actions. [7] **CHECK** CCW pumps for proper operation. If available, **START** additional CCW PUMPS. [8] **ENSURE** CLOSED 3-HS-66-1A, CONDENSER VAC BREAKERS 1A AND 1B, Panel 9-8. [9] **CHECK** 3-FR-66-20, OFF-GAS FLOW TO 6-HOUR HOLDUP VOLUME, Panel 9-8, between 20 and 180 scfm. [10] **ENSURE** OPEN, 3-FCV-66-28, OFF-GAS SYSTEM ISOLATION VALVE. [11] – [13] N/A
	NUSO	Directs the crew to monitor Condenser Vacuum and sets a trigger value of 1.0" to Trip Condenser Vacuum for a Reactor SCRAM.
	NRC	**In accordance with the ILT Simulator Expectations, the NUSO will set target values for Condenser Vacuum as follows:** • **Low Condenser Vacuum alarm (Panel 3-9-7B, Window 17) – Core Flow Runback** • **1" Margin to Trip Condenser Vacuum – Reactor SCRAM** **However, the NUSO may conservatively direct these actions based on the rate of vacuum degradation and information from the field.**
	BOP	Acknowledges and reports the following alarm when received: • CONDENSER A, B, OR C VACUUM LOW, 3-9-7B, Window 17

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 5 Page 6 of 6

Event Description: Clogged Traveling Screens / Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 3-ARP-9-7B CONDENSER A, B, OR C VACUUM LOW, 3-9-7B, Window 17 Operator Action: A. **CHECK** vacuum lowers, MWe lowers, and Exhaust Hood Temperature rises. B. **IF** alarm is valid, **THEN REFER TO** 3-AOI-47-3, Loss of Condenser Vacuum.
	NUSO	Provides a Condenser Vacuum trigger value for the OATC to insert a manual Reactor SCRAM.
	OATC	When the trigger value for Condenser Vacuum is reached or when directed by the NUSO, inserts a manual Reactor SCRAM.
	NRC	**Condenser Vacuum will continue to lower, requiring the crew to insert a manual Reactor SCRAM.** **Two minutes after the crew inserts a manual SCRAM, the clogged Traveling Screen malfunction will be set to 50% to provide cooling water for the crew to place Suppression Pool Cooling in service if necessary.**
	Driver	**Two minutes after the Reactor SCRAM report to the crew that there has been some success in clearing Traveling Screens to allow some water flow.**
	NRC	**End of Event 5. Event 6, Reactor Power Reduction for Lowering Condenser Vacuum will be inserted by the crew as a response to lowering vacuum. No action is required by the Driver to insert Event 6.** **When the crew inserts a manual Reactor SCRAM due to lowering Condenser Vacuum, the following Events are automatically inserted by Simulator Setup:** • **Event 7, Hydraulic Anticipated Transient Without SCRAM (ATWS)** • **Event 8, 3A EHC Pump Trip** • **Event 9, SLC Pump Trip** **No action is required by the Driver to insert Events 7, 8, or 9.**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 6 Page 1 of 5

Event Description: Reactor Power Reduction for Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	NRC	**The crew may elect to reduce power by any method or combination of runbacks in an attempt to maintain Condenser Vacuum.**
	NUSO	Directs the OATC to reduce Reactor Power by using any one or any combination of the following: • Manually using Recirc Master Control pushbuttons on Panel 3-9-5 • Upper Power Runback • Mid Power Runback • Core Flow Runback
	NRC	**3-OI-68, Reactor Recirculation System** **3.0 Precautions and Limitations** **Section 3.5.3, Dual Pump Operation** E. When raising (lowering) Reactor Power per the Reactivity Plan, the following guideline should be used to establish the 60 rpm mismatch between the Recirc Pumps. 1. When the first Recirc Pump reaches 1200 rpm (1300 rpm) or both Recirc Pumps are at 1200 rpm (1300 rpm) then individual controls will be used. 2. While following the Reactivity Plan establish the 60 rpm mismatch using the individual controls for the leading Recirc Pump. 3. While maintaining the 60 rpm mismatch and using the Reactivity Plan, raise (lower) the Recirc Pump speeds using either the Master Controllers or Individual Controllers. 4. Once a Recirc Pump reaches 1300 rpm (1200 rpm) the 60 rpm mismatch is no longer required. While following the Reactivity Plan raise (lower) the lagging Recirc Pump using the individual controller match Recirc Pump speeds. 5. Once the Recirc Pumps are matched, then the speeds may be adjusted as required using the Master Controller or Individual Controllers while maintaining the requirements of Attachment 1.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 6 Page 2 of 5

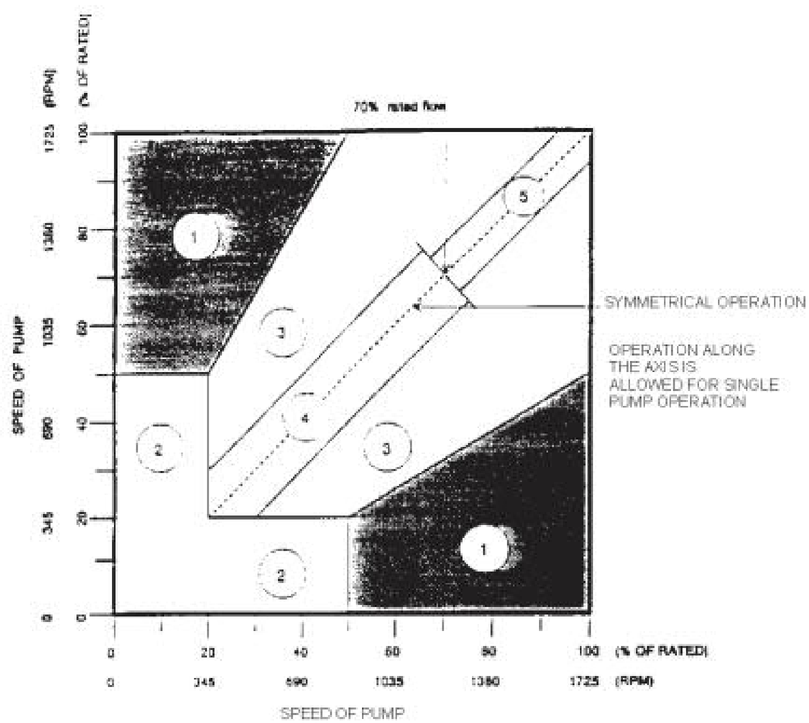
Event Description: Reactor Power Reduction for Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	NRC	(see procedure attachment below)

BFN Unit 3	Reactor Recirculation System	3-OI-68 Rev. 0099 Page 181 of 210

Attachment 1
(Page 1 of 1)
Recirculation Pump Speed Mismatch Curve

(for Steady State, Dual Pump Operation)



1. Avoid Region To Prevent Excessive Jet Pump Vibration.
2. Below Recirc Drive Minimum speed.
3. Operation allowed if reactor subcritical or during transient periods.
4. Limited Operation for Core flow ≤ 70% rated (mismatch ≤ 10% rated speed).
5. Limited operation for Core flow > 70% rated (mismatch ≤ 5% rated speed).

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 6 Page 3 of 5

Event Description: Reactor Power Reduction for Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
		3-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown on 0-TI-248, Station Reactor Engineer and 3-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the later part of the operating cycle (Coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the Power to Flow Map. Instruments used to monitor pump speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing Core Flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area.
	OATC	[1] **IF** desired to control Recirc Pump 3A speed with Recirc Individual Control, **THEN PERFORM** the following; (Otherwise N/A) [1.1] N/A [1.2] Lower Recirc Pump 3A using 3-HS-96-17A(17B)(17C), SLOW (MEDIUM) (FAST). (Otherwise N/A) [2] **IF** desired to control Recirc Pump 3B speed with Recirc Individual Control, **THEN PERFORM** the following; (Otherwise N/A) [2.1] N/A [2.2] Lower Recirc Pump 3B using 3-HS-96-18A(18B)(18C), SLOW (MEDIUM) (FAST). (Otherwise N/A) [3] **WHEN** desired to control Recirc Pumps 3A and/or 3B speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump speed 3A & 3B using the following push buttons as required: 3-HS-96-33, LOWER SLOW 3-HS-96-34, LOWER MEDIUM 3-HS-96-35, LOWER FAST

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 6 Page 4 of 5

Event Description: Reactor Power Reduction for Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	OATC	**NOTES** 1) Manual runback controls are utilized when it becomes necessary to reduce Reactor Power and Core Flow during abnormal plant conditions. 2) This section is performed at Panel 3-9-5. 3) Depressing a manual runback push-button will initiate a runback of both Recirc Pumps until the setpoint is reached. Depressing the push-button a second time will stop the manual runback. The push-button can be depressed a third and fourth time to reinitiate and stop the manual runback. This pattern can be repeated until the applicable setpoint is reached. 4) Attachment 2 can be referred to for additional information on manual runback controls. 5) When initiating manual runbacks, the appropriate manual push-button has to be depressed until the backlight is blinking, then the push-button can be released. 6) RECIRC PUMPS MID POWER RUNBACK is to be used any time a Condensate Pump trips and Reactor Power is greater than or equal to 90%. [1] **IF** time permits, **THEN REVIEW** Precautions and Limitations. **REFER TO** Section 3.0. [2] **IF** desired to reduce Reactor Power to approximately 90%, **THEN PERFORM** the following: (Otherwise N/A) [2.1] **DEPRESS** 3-HS-68-42, RECIRC PUMPS UPPER POWER RUNBACK push-button. [2.2] **CHECK** the following: • Push-button backlight blinks until setpoint is reached • Reactor power lowers to approximately 90%

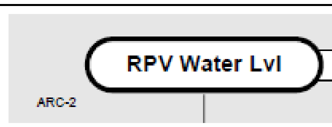
Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 6 Page 5 of 5

Event Description: Reactor Power Reduction for Lowering Condenser Vacuum

Time	Position	Applicant's Actions or Behavior
	OATC	[3] **IF** desired to reduce Reactor Power to approximately 66%, **THEN PERFORM** the following: (Otherwise N/A) [3.1] **DEPRESS** RECIRC PUMPS MID POWER RUNBACK push-button, 3-HS-68-43. [3.2] **CHECK** the following: • Push-button backlight blinks until setpoint is reached • Reactor Power lowers to approximately 66% [4] **IF** desired to reduce Core Flow to approximately 58%, **THEN PERFORM** the following: (Otherwise N/A) [4.1] **DEPRESS** 3-HS-68-44, RECIRC PUMPS CORE FLOW RUNBACK push-button. [4.2] **CHECK** the following: • Push-button backlight blinks until setpoint is reached • Core flow lowers to approximately 58%
	NRC	**End of Event 6. Proceed to Event 7, Hydraulic Anticipated Transient Without SCRAM (ATWS).**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 1 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 7, Hydraulic Anticipated Transient Without SCRAM (ATWS) is inserted on Simulator Setup. No action is required by the Driver to insert Event 7.**
	Driver	**During Event 7, when contacted as the Outside NUSO acknowledge direction to perform EOI Appendices and enter events as necessary:** • **Event 17 – 3-EOI-Appendix-1D, Insert Control Rods Using Reactor Manual Control System (Close 3-FCV-85-586, CHARGING WATER ISOLATION)** • **Event 18 – Open 3-FCV-85-586, CHARGING WATER ISOLATION** • **Event 19 – 3-EOI-Appendix-1F, Manual SCRAM** • **Event 20 – 3-EOI-Appendix-2, Defeating ARI Logic Trips** • **Event 21 – 3-EOI-Appendix-8A, Bypassing Group RPV Low Low Low Level Isolation Interlocks** • **Event 22 – 3-EOI-Appendix-8E, Bypassing Group 6 RPV Low Level and High Drywell Pressure Isolation Interlocks** **Once the event(s) requested have finished their time delay, report completion of the various EOI Appendices to the Control Room.**
	NUSO	Enters 3-EOI-1A, ATWS RPV Control, and updates the crew.
	NUSO	 ARC/L-1 **ENSURE** each as required: • PCIS isolations (Groups 1, 2, and 3) • ECCS • RCIC ARC/L-2 **INHIBIT** ADS

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 2 of 13

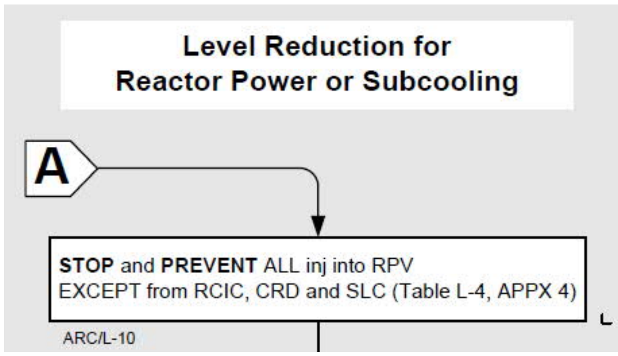
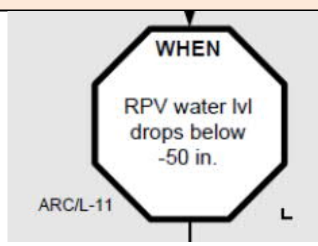
Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>CREW</td><td>Critical Task:
With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS.

Critical Task Failure Criteria:
ADS automatic initiation with Control Rods out.</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/L-3
IF ANY Main Steam Line is open
THEN START defeating the following isolations:
MSIV Low Low Low RPV Water Level (3- EOI-Appendix 8A)
Reactor Building Ventilation Low RPV Water Level (3-EOI-Appendix-8E)</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/L-4<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Reactor Power is above 5% or unknown
AND
Reactor Water Level is above (-) 50 inches</td><td>A</td></tr><tr><td>ALL Level/Power conditions exist (Table Q-1)</td><td>A</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td><table><tr><th>Table Q-1
Level/Power Conditions</th><th></th></tr><tr><td>• Suppression Pool Temperature is above 110ºF</td><td>☐</td></tr><tr><td>• Reactor Power above 5% OR unknown</td><td>☐</td></tr><tr><td>• RPV Level above -162 in.</td><td>☐</td></tr><tr><td>• MSRV open/cycling OR DW pressure above 2.4 psig</td><td>☐</td></tr></table></td></tr>
</table>

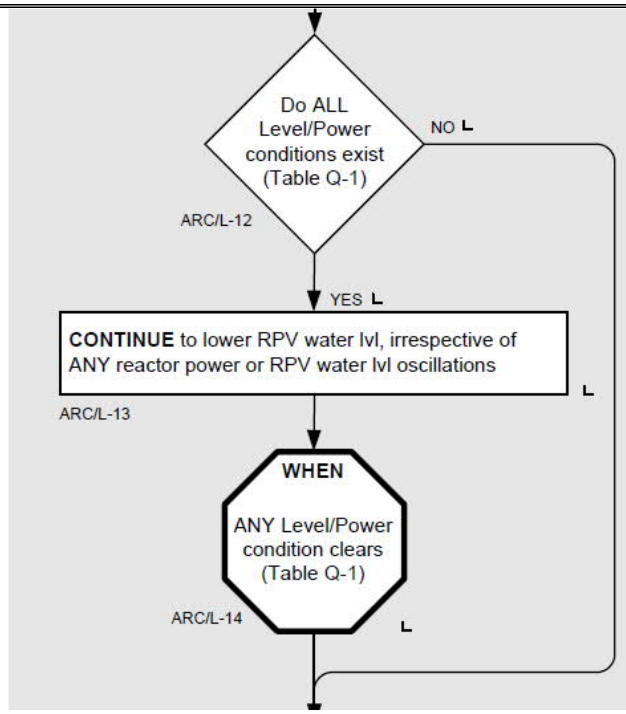
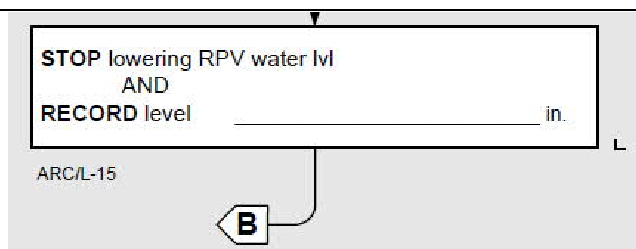
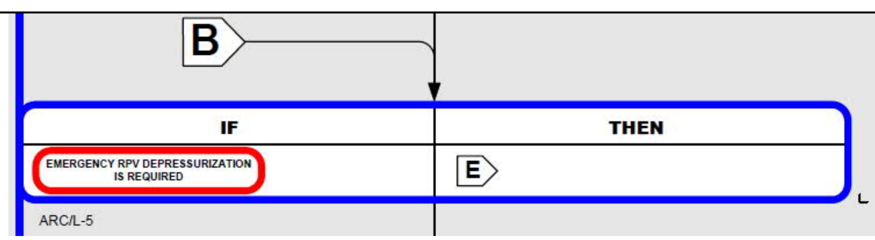
Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 3 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	
	Crew	**Critical Task:** **When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the loss of forced recirculation to prevent possible fuel damage.** **Critical Task Failure Criteria:** **The operating crew fails to STOP and PREVENT injection and lower Reactor Water Level for Reactor Power control during an ATWS within 130 seconds after Recirculation Flow is stopped.**
	NUSO	

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 4 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	
	NUSO	
	NUSO	

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 5 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	ARC/L-6 (see below)
	NUSO	ARC/L-7 (see below)
	NUSO	Table L-3 (see below)

ARC/L-6

USE ANY Preferred ATWS Injection System (Table L-3) to maintain RPV Water Level between (-) 180 inches and:

Lowered level (if level was deliberately lowered in flowpath A)

OR

+51 inches (if level was NOT deliberately lowered)

- Ok to use Core Spray (3-EOI-Appendix-6D or 6E) or Alternate Injection Subsystems (Table L-2) if previously required by flowpath E or C4A

IF	THEN
Reactor Water Level CANNOT be restored and maintained above (-) 180 inches AND Core Steam Flow remains below MCSF (Table L-5)	**NO ACTION REQUIRED**

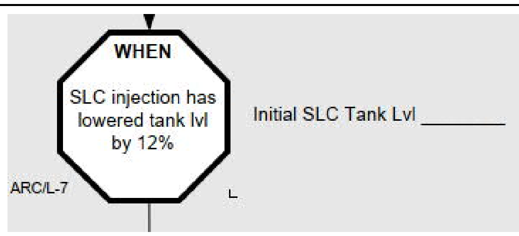


Table L-3 Preferred ATWS Injection Systems		
SOURCES	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available 2 3 6	5C, 20M	1200 psig
HPCI with CST suction if available 2 6 7	5D, 20N	1200 psig
CNDS	6A	480 psig
LPCI 2	6B, 6C	320 psig
SLC (boron tank)	7B	1450 psig
Table L-2 systems or CS ONLY IF Step ARC/L-19 has been previously performed	----	----

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 6 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
<table>
<tr><th colspan="3">Table L-2
Alternate Injection Subsystems</th></tr>
<tr><th>SOURCE</th><th>APPX</th><th>INJ PRESS</th></tr>
<tr><td>EHPM Pump</td><td>7L</td><td>1210 psig</td></tr>
<tr><td>SLC (test tank)</td><td>7B</td><td>1450 psig</td></tr>
<tr><td>SLC (boron tank)</td><td>7B</td><td>1450 psig</td></tr>
<tr><td>CNDS transfer pumps to RHR and CS</td><td>7A</td><td>110 psig</td></tr>
<tr><td>RHR crosstie to other units</td><td>7C</td><td>320 psig</td></tr>
<tr><td>Stby coolant</td><td>7D</td><td>160 psig</td></tr>
<tr><td>RHR drain pumps</td><td>7E, 7F</td><td>50 psig</td></tr>
<tr><td>PSC head tank pumps</td><td>7G</td><td>30 psig</td></tr>
<tr><td>RCIC (aux boiler steam) with CST suction if available 2 3 6</td><td>7H</td><td>1200 psig</td></tr>
<tr><td>RCIC manual start</td><td>20A</td><td>1200 psig</td></tr>
<tr><td>HPCI (aux boiler steam) with CST suction if available 2 6 7</td><td>7J</td><td>780 psig</td></tr>
<tr><td>Fire Protection system</td><td>7K</td><td>150 psig</td></tr>
<tr><td>FLEX Pump Sys (CILRT/CS)</td><td>20D</td><td>150 psig</td></tr>
<tr><td>FLEX Pump Sys (Standby Coolant)</td><td>20B</td><td>150 psig</td></tr>
<tr><td>FLEX Pump Sys (CILRT/CRD)</td><td>20C</td><td>150 psig</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>
Table L-5
Minimum Core Steam Flow

MCSF is 1,100,000 lbm/hr and indicated by ANY:

MSCP (Table P-3)
Open TBPVs and RPV pressure above the following:
<table>
<tr><th>TBPV</th><th>RPV Pressure (psig)</th></tr>
<tr><td>2.1</td><td>1,100</td></tr>
<tr><td>2.3</td><td>1,000</td></tr>
<tr><td>2.6</td><td>900</td></tr>
<tr><td>2.8</td><td>800</td></tr>
<tr><td>3.2</td><td>700</td></tr>
</table>
RPV injection flow greater than 2200 gpm
Reactor power above 9.1%

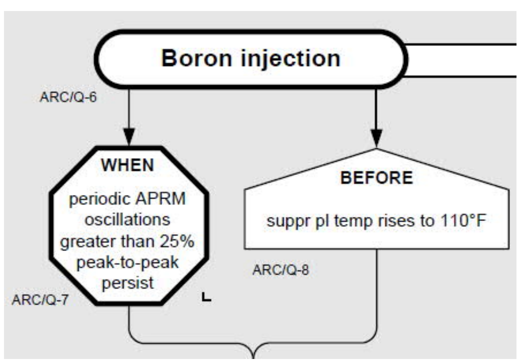
</td></tr>
<tr><td></td><td>NUSO</td><td>Reactor Power</td></tr>
</table>

<table>
<tr><td colspan="3">Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 7 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)</td></tr>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>The Reactor is subcritical
AND
NO boron has been injected</td><td>AOI-100-1
Reactor Scram</td></tr>
</table>
ARC/Q-2
ENSURE Reactor Mode Switch in SHUTDOWN

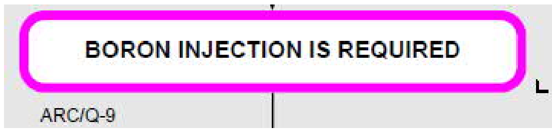
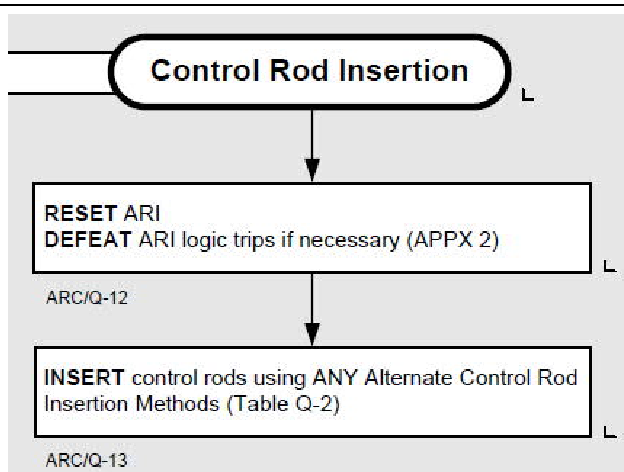
ARC/Q-3
INITIATE ARI

ARC/Q-4
IF tripping Recirc Pumps will cause loss of Main Turbine, RFPT, HPCI, or RCIC
THEN ENSURE Recirc Runback (pump speed 480 RPM or less)</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-5
IF Reactor Power is above 5% or unknown
THEN TRIP Recirc Pumps


</td></tr>
<tr><td></td><td>NUSO</td><td>In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, Section 4.8.4.C, when EOI-1A, ATWS RPV Control, Step ARC/Q-8 is reached, **IF** Reactor Power is greater than APRM downscale, **THEN INITIATE** SLC.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 8 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
</td></tr>
<tr><td></td><td>NRC</td><td>See Event 9 on page xx of xx.</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-10
<table>
<tr><td colspan="2">1. INITIATE SLC (3-EOI-Appendix-3A)
2. INHIBIT ADS</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Boron CANNOT be injected using SLC</td><td>INJECT boron into RPV using CRD (3-EOI-Appendix-3B)</td></tr>
<tr><td>SLC Tank Water Level drops to 0%</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-11
ENSURE RWCU System Isolation</td></tr>
<tr><td></td><td>NUSO</td><td>
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</table>

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 9 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
Table Q-2
Alternate Control Rod Insertion Methods
<table>
<tr><th>CONDITIONS</th><th>METHODS</th><th>APPX</th></tr>
<tr><td rowspan="2">Scram valves failed to open</td><td>DEENERGIZE scram solenoids</td><td>1A</td></tr>
<tr><td>VENT scram air header</td><td>1B</td></tr>
<tr><td>Scram valves opened but SDV is full</td><td>1. RESET scram DEFEAT RPS logic if necessary
2. DRAIN SDV
3. RECHARGE accumulators
4. INITIATE scram</td><td>1F</td></tr>
<tr><td rowspan="4">Manual control rod insertion methods</td><td>DRIVE control rods BYPASS RWM and RAISE CRD drive water differential pressure if necessary</td><td>1D</td></tr>
<tr><td>RAISE CRD cooling water header dp</td><td>1G</td></tr>
<tr><td>SCRAM individual control rods</td><td>1C</td></tr>
<tr><td>VENT control rod over piston volumes</td><td>1E</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>RPV Press</td></tr>
<tr><td></td><td>NUSO</td><td>
ARC/P-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>A high Drywell Pressure ECCS signal exists (2.45 PSIG)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>EMERGENCY DEPRESSURIZATION is required or has been required</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 10 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>ARC/P-2

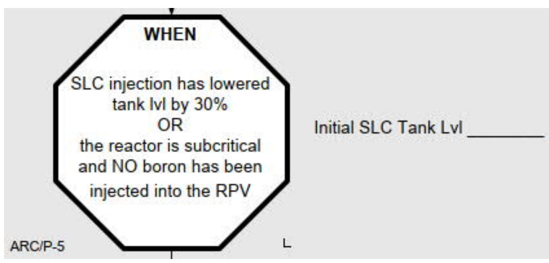
IF ANY MSRV is cycling
THEN MANUALLY OPEN MSRVs UNTIL RPV press drops to the pressure at which all Main Turbine Bypass Valves are fully open (APPX 11A)</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/P-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Suppression Pool Temperature and Water Level CANNOT be maintained in the safe area of Curve 3 at the existing RPV Pressure</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Pool Water Level CANNOT be maintained in the safe area of Curve 4</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>BORON INJECTION IS REQUIRED
AND
The Main Condenser is available
AND
There has been no indication of a steam line break</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 7 Page 11 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>ARC/P-4

STABILIZE RPV Pressure below 1073 psig using the Main Turbine Bypass Valves (3-EOI-Appendix-8B)
➢ Use Alternate RPV Pressure Control Systems (Table P-1), if necessary
➢ Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary

<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Drywell Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>
</td></tr>
<tr><td></td><td>OATC</td><td>3-EOI-Appendix-1F, Manual SCRAM

[1] VERIFY Reactor Scram and ARI reset.
[1.1] IF ARI CANNOT be reset, THEN EXECUTE EOI Appendix 2 concurrently with Step 1.0[1.2] of this procedure.
[1.2] IF Reactor Scram CANNOT be reset, THEN DISPATCH personnel to Unit 3 Auxiliary Instrument Room to defeat ALL RPS logic trips.
[2] WHEN RPS Logic has been defeated, THEN RESET Reactor SCRAM.
[3] VERIFY OPEN Scram Discharge Volume vent and drain valves</td></tr>
<tr><td></td><td>OATC</td><td>Dispatches personnel to perform outside portions of 3-EOI-Appendix-1F, Manual SCRAM.</td></tr>
</table>

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 7	Page 12 of 13

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	[4] **DRAIN** SDV UNTIL the following annunciators clear: • WEST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 3-9-4, 3-XA-55-4A, Window 1) • EAST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 3-9-4, 3-XA-55-4A, Window 29)
	NRC	**The accumulators will drain in approximately 7 minutes, and the alarms at Panel 3-9-4, Windows 1 and 29, will clear. The OATC may then attempt a Reactor SCRAM.**
	OATC	[5] **DISPATCH** personnel to **VERIFY OPEN** 3-SHV-085-0586, CHARGING WATER ISOL. **NOTES** 1) If EOI Appendix 2 has been executed, ARI initiation or reset will NOT be possible or necessary in Step 1.0[6]. 2) If Reactor Pressure is greater than 600 psig, SRO may direct performance of step 1.0[6] prior to accumulators being fully recharged. [6] **WHEN** CRD Accumulators are recharged, **THEN INITIATE** manual Reactor SCRAM and ARI.
	NRC	**Control Rods will insert the first time the OATC attempts a Reactor SCRAM after the ATWS.**
	OATC	[7] **CONTINUE** to perform Steps 1.0[1] through 1.0[6] UNTIL ANY of the following exists: • ALL Control Rods are fully inserted, **OR** • NO inward movement of Control Rods is observed, **OR** • NUSO directs otherwise. END OF EOI APPENDIX 1F

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 7	Page 13 of 13
Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)			

Time	Position	Applicant's Actions or Behavior
	OATC	3-EOI-Appendix-8A, Bypassing Group I RPV Low Low Low Level Isolation Interlocks [1] BYPASS Group 1 RPV Low-Low-Low Level Isolation Interlocks as follows (Unit 3 Control Room, Panel 9-4): [1.1] PLACE keylock switch 3-HS-64-56A, GRP 1 RPV LOW LEVEL BYPASS (SYS A1), in BYPASS. [1.2] PLACE keylock switch 3-HS-64-56B, GRP 1 RPV LOW LEVEL BYPASS (SYS B1), in BYPASS. [1.3] PLACE keylock switch 3-HS-64-56C, GRP 1 RPV LOW LEVEL BYPASS (SYS A2), in BYPASS. [1.4] PLACE keylock switch 3-HS-64-56D, GRP 1 RPV LOW LEVEL BYPASS (SYS B2), in BYPASS. [1.5] ENSURE closed the following valves (Unit 3 Control Room, Panel 9-3): • 3-FCV-43-13, RX RECIRC SAMPLE INBD ISOLATION VALVE • 3-FCV-43-14, RX RECIRC SAMPLE OUTBD ISOLATION VALVE [2] N/A END OF EOI APPENDIX 8A
	NRC	**End of Event 7. When the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 8 Page 1 of 7

Event Description: 3A EHC Pump Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8, EHC Pump Trip, is inserted on Simulator Setup. No action is required by the Driver to insert this event.** **Thirty (30) seconds after the MODE SWITCH is placed in RUN, 3A EHC pump will be stopped.**
	BOP	Acknowledges and reports the following alarms when received: • STANDBY EHC PUMP FAILED, 3-9-7B, Window 15 • EHC HYDRAULIC FLUID HEADER PRESSURE LOW, 3-9-7B, Window 1
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure(s).
	BOP	Alarm Response Procedure, 3-ARP-9-7B STANDBY EHC PUMP FAILED, Window 15 A. On Panel 3-9-7: 1. **CHECK** 3-PI-47-7, EHC HEADER PRESSURE. 2. **ENSURE** EHC PUMP 3B, 3-HS-47-2A and/or 3-HS-47-1A, EHC PUMP 3A running. 3. **CHECK** 3-EI-47-2, EHC PUMP 3B PUMP MTR CURRENT and/or 3-EI-47-1, EHC PUMP 3A PUMP MTR CURRENT. B. **DISPATCH** personnel to pumping unit to check for abnormal conditions.
	Driver	**If contacted as the outside NUSO, Work Control, Maintenance, or an AUO to investigate the cause for 3A EHC Pump Trip or to check for abnormal conditions, acknowledge the direction.**
	BOP	**NOTE** Lights extinguish at 1300 psig lowering and illuminate at 1500 psig rising. 4. **CHECK** lights above 3-HS-47-4A, EHC PUMP 3A TEST pushbutton and 3-HS-47-5A, EHC PUMP 3B TEST pushbutton.

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 8	Page 2 of 7
Event Description: 3A EHC Pump Trip			

Time	Position	Applicant's Actions or Behavior
	BOP	C. **IF** EHC Hydraulic System fails, **THEN ENSURE** turbine trips at or below 1100 psig.
	BOP	Alarm Response Procedure, 3-ARP-9-7B EHC HYDRAULIC FLUID HEADER PRESSURE LOW, Window 1 Operator Action: A. N/A B. **CHECK** 3-PI-47-7, EHC HEADER PRESSURE between 1550 and 1650 psig. C. **DISPATCH** personnel to inspect EHC pump unit. **NOTE** On EHC Hydraulic System failure, accumulator and check valve arrangement will provide approximately one minute of Bypass Valve operation. D. **IF** EHC Hydraulic system fails, **THEN ENSURE** Turbine trips at or below 1100 psig.
	NUSO	Directs the BOP to maintain Reactor Pressure with Main Steam Relief Valves (MSRVs) using 3-EOI-Appendix-11A, Alternate RPV Pressure Control Systems MSRVs.
	BOP	3-EOI-Appendix-11A, Alternate RPV Pressure Control Systems MSRVs. [1] N/A [2] N/A

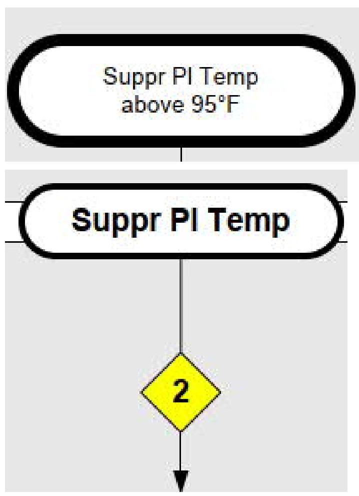
Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 8 Page 3 of 7

Event Description: 3A EHC Pump Trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>[3] OPEN MSRVs using the following sequence to control RPV pressure as directed by the SRO:
<table>
<tr><td>1</td><td>3-PCV-1-179</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>2</td><td>3-PCV-1-180</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>3</td><td>3-PCV-1-4</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>4</td><td>3-PCV-1-31</td><td>MN STM LINE C RELIEF VALVE</td></tr>
<tr><td>5</td><td>3-PCV-1-23</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>6</td><td>3-PCV-1-42</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>7</td><td>3-PCV-1-30</td><td>MN STM LINE C RELIEF VALVE</td></tr>
<tr><td>8</td><td>3-PCV-1-19</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>9</td><td>3-PCV-1-5</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>10</td><td>3-PCV-1-41</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>11</td><td>3-PCV-1-22</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>12</td><td>3-PCV-1-18</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>13</td><td>3-PCV-1-34</td><td>MN STM LINE C RELIEF VALVE</td></tr>
</table>
[4] N/A
[5] N/A
[6] N/A
END OF EOI APPENDIX 11A</td></tr>
<tr><td></td><td>BOP</td><td>Acknowledges and reports the following alarm when received:
SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, 3-9-3E, Window 12</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.</td></tr>
</table>

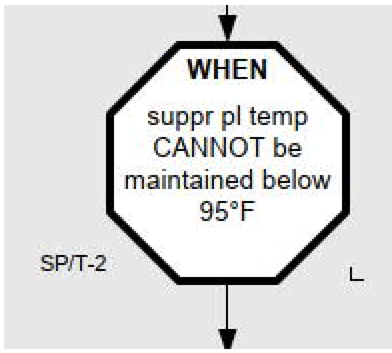
Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 8 Page 4 of 7

Event Description: 3A EHC Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	(If received) Alarm Response Procedure, 3-ARP-9-3E SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, Window 12 Operator Action: A. **IF** alarm is valid, **THEN ENTER** 3-EOI-2, Primary Containment Control.
	NUSO	Enters 3-EOI-2, Primary Containment Control.
	NUSO	  Operating pumps with suction from the suppression pool above the NPSH Limit (Curve 1, 2, 9 or 10) or with suppression pool water level below 10 ft (Vortex limit) may cause equipment damage
	NUSO	SP/T-1 **MONITOR** and **CONTROL** Suppr Pool Temperature below 95°F using available Suppr Pool Cooling (APPX 17A)

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 8 Page 5 of 7

Event Description: 3A EHC Pump Trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>SP/T-2


SP/T-3
OPERATE all available Suppression Pool Cooling using only RHR Pumps NOT required to assure adequate Core Cooling by continuous injection (APPX 17A)</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to place Suppression Pool Cooling in service in accordance with 3-EOI-Appendix-17A, RHR System Operation Suppression Pool Cooling.</td></tr>
<tr><td></td><td>BOP</td><td>3-EOI-Appendix-17A, RHR System Operation Suppression Pool Cooling

NOTE
Placing a BYPASS SEL switch in BYPASS in Step 1.0[1] below prevents automatic opening of the affected RHR loop's outboard injection valve. This makes LPCI mode of that RHR loop inoperable.

[1] IF Adequate Core Cooling is assured OR directed to cool the Suppression Pool irrespective of Adequate Core Cooling, THEN BYPASS LPCI Injection Valve open interlock AS NECESSARY:

PLACE 3-HS-74-155A, LPCI SYS I OUTBD INJECTION VALVE BYPASS SELECT in BYPASS.
PLACE 3-HS-74-155B, LPCI SYS II OUTBD INJECTION VALVE BYPASS SELECT in BYPASS.
</td></tr>
</table>

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 8	Page 6 of 7

Event Description: 3A EHC Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[2] **PLACE** RHR SYSTEM I(II) in Suppression Pool Cooling as follows: [2.1] **ENSURE** at least one RHRSW pump supplying each EECW header. [2.2] **ENSURE** RHRSW pump supplying desired RHR Heat Exchanger(s). [2.3] **THROTTLE** the following in service RHRSW outlet valves to obtain 1700 to 4500 gpm RHRSW flow: • 3-FCV-23-34, RHR HX 3A RHRSW OUTLET VLV • 3-FCV-23-46, RHR HX 3B RHRSW OUTLET VLV • 3-FCV-23-40, RHR HX 3C RHRSW OUTLET VLV • 3-FCV-23-52, RHR HX 3D RHRSW OUTLET VLV [2.4] **IF** Directed by SRO, **THEN PLACE** 3-XS-74-122(130), RHR SYSTEM I(II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE. [2.5] **IF** LPCI Initiation signal exists, **THEN MOMENTARILY PLACE** 3-XS-74-121(129), RHR SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, in SELECT. [2.6] **IF** 3-FCV-74-53(67), RHR SYS I(II) LPCI INBOARD INJECTION VALVE, is OPEN, **THEN ENSURE** CLOSED 3-FCV-74-52(66), RHR SYS I(II) LPCI OUTBOARD INJECTION VALVE. [2.7] **OPEN** 3-FCV-74-57(71), RHR SYS I(II) SUPPRRESSION CHAMBER/POOL ISOLATION VALVE. [2.8] **ENSURE** desired RHR pump(s) for Suppression Pool Cooling are operating. **CAUTION** RHR System flows below 7,000 gpm or above 10,000 gpm for one pump operation may result in excessive vibration and equipment damage.

Op Test No.: 21-04	Scenario No. NRC-2	Event No.: 8	Page 7 of 7
Event Description: 3A EHC Pump Trip			

Time	Position	Applicant's Actions or Behavior
	BOP	[2.9] **THROTTLE OPEN** 3-FCV-74-59(73), RHR SYS I(II) SUPPRESSION POOL COOLING/TEST VALVE, to maintain EITHER of the following as indicated on 3-FI-74-50(64), RHR SYSTEM I(II) FLOW: • Between 7,000 and 10,000 gpm for one pump operation OR • At or below 13,000 gpm for two pump operation [2.10] **ENSURE** CLOSED 3-FCV-74-7(30), RHR SYS I(II) MINIMUM FLOW VALVE. [2.11] **MONITOR** RHR Pump NPSH using Attachment 1. [2.12] **NOTIFY** Chemistry that RHRSW is aligned to in service RHR Heat Exchangers.
	Driver	**When contacted as Chemistry, acknowledge any information given.**
	BOP	[2.13] **IF** Additional Suppression Pool Cooling flow is necessary, **THEN PLACE** additional RHR and RHRSW pumps in service using Steps 1.0[2.2] through 1.0[2.12]. [3] N/A END OF EOI APPENDIX 17A
	NRC	**End of Event 8. Once the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 9 Page 1 of 2

Event Description: SLC Pump Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9, SLC Pump Trip, is inserted on Simulator Setup. No action is required by the Driver to insert this event.** **NOTE: The first SLC Pump that is started will trip.**
	BOP	3-EOI-Appendix-3A, SLC Injection [1] **UNLOCK** and **PLACE** 3-HS-63-6A, SLC PUMP 3A/3B, control switch in START PUMP 3A or START PUMP 3B position. [2] **CHECK** SLC System for injection by observing the following: • Selected pump starts, as indicated by red light illuminated above pump control switch • Squib valves fire, as indicated by SQUIB VALVE A and B CONTINUITY blue lights extinguished • SLC SQUIB VALVE CONTINUITY LOST Annunciator in alarm on Panel 3-9-5 (3-XA-55-5B, Window 20) • 3-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure. • System flow, as indicated by 3-IL-63-11, SLC FLOW, red light illuminated on Panel 3-9-5 • SLC INJECTION FLOW TO REACTOR Annunciator in alarm on Panel 3-9-5 (3-XA-55-5B, Window 14) [3] **IF** Proper system operation CANNOT be verified, **THEN RETURN** to Step 1.0[1] and START other SLC pump.
	BOP	Determines that the first SLC Pump that was started trips, and starts the alternate SLC Pump.
	BOP	[4] **VERIFY** RWCU isolation by observing the following: • RWCU Pumps 3A and 3B tripped • 3-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE closed • 3-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE closed • 3-FCV-69-12, RWCU RETURN ISOLATION VALVE closed [5] **VERIFY** ADS inhibited.

Op Test No.: 21-04 Scenario No. NRC-2 Event No.: 9 Page 2 of 2

Event Description: SLC Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[6] **MONITOR** Reactor Power for downward trend. [7] **MONITOR** 3-LI-63-1A, SLC STORAGE TANK LEVEL, and **CHECK** that level is dropping approximately 1% per minute. [8] **WHEN** EITHER of the following exists: • SLC tank level drops to 0%, OR • As directed by SRO, **THEN** STOP SLC Pump 3A or 3B [9] **NOTIFY** Chemistry to mix additional solution to compensate for dilution as directed by the SRO. [10] **WHEN** directed by the SRO to perform system flush, **THEN REFER** to 3-OI-63, Section 8.1, for system flush. END OF EOI APPENDIX 3A
	Driver	**If contacted as Chemistry, acknowledge any direction given.**
	NRC	**End of Event 9. Once the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Scenario Setup
UNIT 3

IC	28
Exam IC	252

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-68	99	ARP-3E	34	APPX-17A	13
OI-67	126	ARP-5A	54	APPX-1F	3
OI-92B	22	ARP-7B	30	APPX-8A	3
GOI-100-12A	72	ARP-8C	20		
0-AOI-27-1	13	ARP-20A	34		
AOI-47-3	16	EOI-1A	2		
AOI-85-6	12	EOI-2	13		
ODM 4.20	7	EOI-3	2		

Simulator Setup	• Verify camera system is powered down (admin password = abcd1234) • Start CPERF **PRIOR** to placing the Simulator in RUN • Verify EECW Pump Alarm borders are properly arranged on Panels 3-9-23A / B / C / D • Hang Danger Tags on B3 EECW Pump and 3B EHC Pump • Hang Protected Equipment Tag on 3A EHC Pump
Schedule Files(s):	2104 NRC Scenario 2 UNIT 3.sch
Event Files(s):	2104 NRC Scenario 2 UNIT 3.evt

Schedule File – 2104 NRC Scenario 2 UNIT 3.sch

Event	Action	Description
	Event F:/2104/NRC/Scenarios/U3/Scenario 2/2104 NRC Scenario 2 UNIT 3.evt	
2	Insert malfunction SW03M	RHRSW PUMP D3 TRIP
3	set nmaprmgain(1)=0.000005	
4	Insert malfunction RD04R2215	DRIFT ANY CONTROL ROD OUT 22-15
14	Delete malfunction RD04R2215	DRIFT ANY CONTROL ROD OUT 22-15
5	Insert malfunction MC05 to 100.00000 in 600	INTAKE PUMPING STATION TRAVELING SCREENS CLOG

Schedule File – 2104 NRC Scenario 2 UNIT 3.sch

Event	Action	Description
7	Insert malfunction MC05 to 50.00000 in 120	INTAKE PUMPING STATION TRAVELING SCREENS CLOG
25	Insert malfunction MC02A after 60	CCW PUMP 3A MOTOR TRIP
	Insert malfunction RD09A after 2 to 55.00000	ATWS (HYDRAULIC LOCK) EAST
	Insert malfunction RD09B after 2 to 55.00000	ATWS (HYDRAULIC LOCK) WEST
7	Insert malfunction RD06ALLSTICK after 5	STICK ALL CONTROL RODS
	Insert malfunction RD17A	EAST SDV LEVEL SWITCH FAILS LOW (LS-85-45E,F,G,H,K,M)
	Insert malfunction RD17B	WEST SDV LEVEL SWITCH FAILS LOW (LS-85-45A,B,C,D,J,L)
7	Insert override HS-47-1A after 30 to STOP	EHC PUMP 3A
	Insert malfunction PMP-47-1 to FAIL_CCOIL	52_BREAKER EHC HYD FLUID PUMP A
	Insert malfunction SL01A	SLC PUMP 3A TRIP
	Insert malfunction SL01B	SLC PUMP 3B TRIP
9	Delete malfunction SL01A	SLC PUMP 3A TRIP
10	Delete malfunction SL01B	SLC PUMP 3B TRIP
17	Insert remote RD06 to CLOSE	CRD CHARGING WATER VALVE FCV-3-85-586
18	Insert remote RD06 to OPEN	CRD CHARGING WATER VALVE FCV-3-85-586
19	Schedule F:\2104\NRC\Scenarios\U3\Scenario 2\App. 1F.sch	
20	Schedule F:\2104\NRC\Scenarios\U3\Scenario 2\App. 2.sch	

Schedule File – 2104 NRC Scenario 2 UNIT 3.sch

Event	Action	Description
21	Schedule F:\2104\NRC\Scenarios\U3\Scenario 2\App. 8A.sch	
22	Schedule F:\2104\NRC\Scenarios\U3\Scenario 2\App. 8E.sch	
7	Delete malfunction RD17A	EAST SDV LEVEL SWITCH FAILS LOW (LS-85-45E,F,G,H,K,M)
7	Delete malfunction RD17B	WEST SDV LEVEL SWITCH FAILS LOW (LS-85-45A,B,C,D,J,L)
27	Delete malfunction RD09A	ATWS (HYDRAULIC LOCK) EAST
27	Delete malfunction RD09B	ATWS (HYDRAULIC LOCK) WEST
27	Delete malfunction RD06ALLSTICK	STICK ALL CONTROL RODS

Schedule File: APP. 1F.sch

Event	Action	Description
	Insert remote RP13A to BYP after 300	DEFEAT CHANNEL A1 AUTO SCRAM
	Insert remote RP13B to BYP after 300	DEFEAT CHANNEL B1 AUTO SCRAM
	Insert remote RP13C to BYP after 300	DEFEAT CHANNEL A2 AUTO SCRAM
	Insert remote RP13D to BYP after 300	DEFEAT CHANNEL B2 AUTO SCRAM

Schedule File: APP. 2.sch

Event	Action	Description
	Insert remote RP12A to TEST after 300	OPERATE LOCAL ATWS MODE SWITCH 3-HS-68-118A
	Insert remote RP12B to TEST after 300	OPERATE LOCAL ATWS MODE SWITCH 3-HS-68-118B

Schedule File: APP. 8A.sch

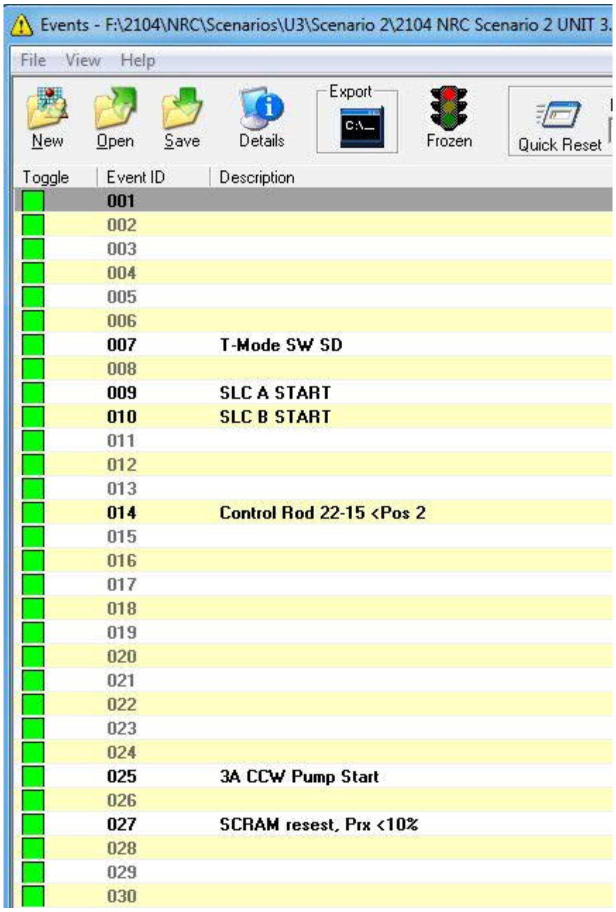
Event	Action	Description
	Insert remote RP06A to BYP after 300	BYP MSIV GP 1 LO LVL - APPNDX 8A- 16A-K1A
	Insert remote RP06B to BYP after 300	BYP MSIV GP 1 LO LVL - APPNDX 8A- 16A-K1B
	Insert remote RP06C to BYP after 300	BYP MSIV GP 1 LO LVL - APPNDX 8A- 16A-K1C
	Insert remote RP06D to BYP after 300	BYP MSIV GP 1 LO LVL - APPNDX 8A- 16A-K1D

Schedule File: APP. 8E.sch

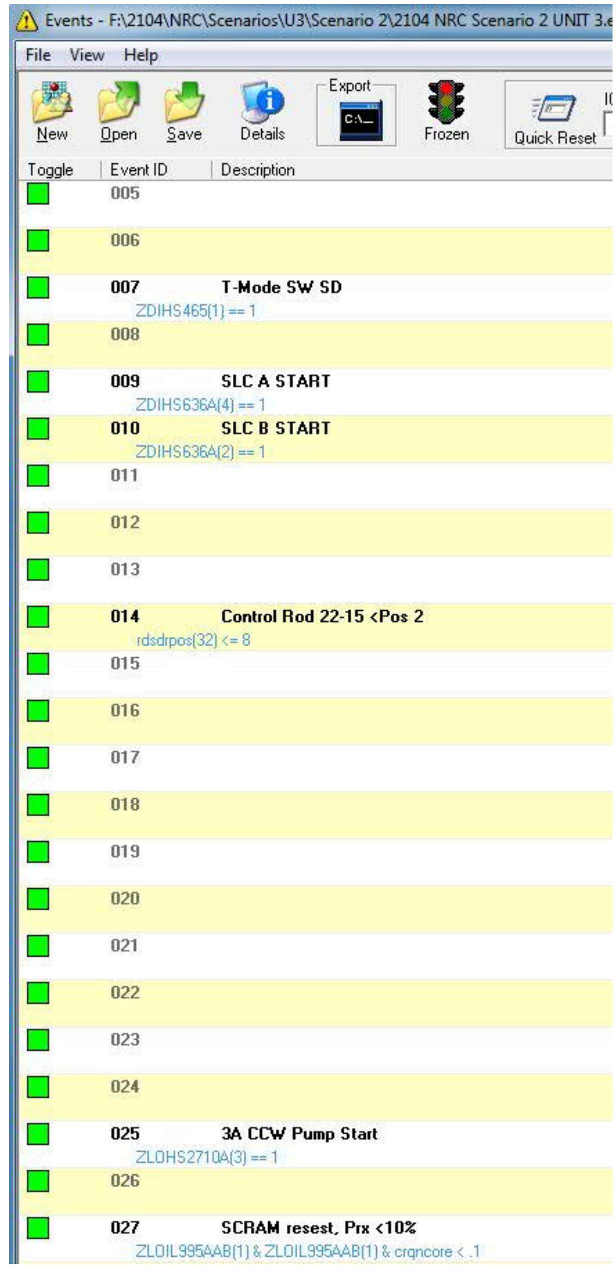
Event	Action	Description
	Insert remote RP14A to BYP after 300	BYPASS GROUP 6 INBOARD LOW LVL/HI DW PRESS INTLKS
	Insert remote RP14B to BYP after 300	BYPASS GROUP 6 OUTBOARD LOW LVL/HI DW PRESS INTLKS

Event File

List



Details



UNIT 3 SHIFT TURNOVER MEETING

MODE 1	**DAYS ON LINE** **227**	**Drywell Leakage (GPM)** **1.89**	**Today**
	PRA (EOOS) -Green		**Protected Equipment**
Rx Power **100.0%**	500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (GPM)** **0.31**	3A EHC Pump
MWe **1303**	**Last breaker closure** **10/01/20 4:31**	**Equipment Drain (GPM)** **1.58**	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

B3 EECW Pump is tagged for oil change (information only LCO).

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

B3 EECW Pump Tagged for oil change.

3B EHC Pump tagged for discharge filter replacement.

3A CCW Pump repairs complete and tags are cleared. Ready to re-start when Maintenance is ready.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Continue to support B3 EECW and 3B EHC Pump maintenance.

Alternate Recirc Drive Cooling Water Pumps.

OPERATOR WORK AROUNDS **OWAs - 1** **Burdens - 2** **Challenges - 28**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Facility: BFN Scenario Number: NRC-3 Op-Test Number: 21-04

Examiners: ____________________ Operators: SRO: ____________________

____________________ ATC: ____________________

____________________ BOP: ____________________

Initial Conditions: ~2 % Reactor Power.

Turnover: Reactor Startup in progress. Raise Reactor Power and place the Reactor MODE SWITCH in RUN. A thunderstorm watch has just been issued for Limestone County, AL. 'G' IRM is bypassed due to noise.

Critical Tasks:

1. When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, SRV Tail Pipe Level Limit, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator (NUSO).
2. When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the NUSO determines that Emergency Depressurization is required. The Operator initiates Emergency Depressurization as directed by the NUSO.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Transfer Seal Steam to Main Steam
2.	N/A	R-OATC R-NUSO	Raise Reactor Power using Control Rods
3.	XS-92-7/42B	C-OATC C-NUSO	IRM Failure
4.	RD08R2227	TS-NUSO	Control Rod Accumulator INOPERABLE
5.	SCHEDULE STACK	C-BOP C-NUSO	Failure of 2A Stack Dilution Fan, Standby Fan Fails to Automatically Start
6.	DG01B	TS-NUSO	'B' Emergency Diesel Generator (EDG) Logic Breaker Tripped
7.	SCHEDULE TORUS	M-ALL	High Suppression Pool Water Level / Emergency Depressurization
8.	RD06R3019 RD06R2615	C-OATC C-NUSO	Two Control Rods Fail to Insert
9.#	ED10A ED10B	C-BOP C-NUSO	480V Shutdown Board Trip

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

\# Event on previous two NRC Exams

#S Event on previous two NRC Exams Scenario (Spare)

Events

1. The crew will transfer Seal Steam from Auxiliary Steam to Main Steam in accordance with 2-OI-47C, Seal Steam System.

2. The crew will continue the Reactor startup by withdrawing Control Rods in accordance with 2-OI-85, Control Rod Drive System, and 2-GOI-100-1A, Unit Startup and Power Operation.

3. During Control Rod withdrawal, the IRM ‘B’ Range Switch will fail in position 8, requiring the crew to bypass IRM ‘B’ in accordance with 2-OI-92A, Intermediate Range Monitors. The Nuclear Unit Senior Operator (NUSO) will reference Technical Specification 3.3.1.1, RPS Instrumentation, Table 3.3.1.1.

4. Control Rod 22-27 Accumulator Pressure will lower below 940 psig, requiring the NUSO to address Technical Specification 3.1.5, Control Rod SCRAM Accumulators, Condition A.

5. Stack Dilution Fan 2A will fail and the standby Stack Dilution Fan will fail to automatically start. The crew will take action to manually restore Stack Dilution Flow by starting 2B Stack Dilution Fan in accordance with 2-OI-66, Off-Gas System, Section 5.1.

6. The ‘B’ Emergency Diesel Generator (EDG) Logic Breaker will trip, causing an annunciation in the Control Room and disabling ‘A’ EDG. The NUSO will address Technical Specification 3.8.1, AC Sources – Operating, Condition B.

7. The crew will respond to a rising Suppression Pool Water Level in accordance with Emergency Operating Instruction 2-EOI-2, Primary Containment Control. Suppression Pool Water Level will not be able to be maintained in the safe area of Curve 4, SRV Tail Pipe Level Limit, requiring the crew to Emergency Depressurize the Reactor in accordance with 2-C-2, Emergency RPV Depressurization.

8. When the crew inserts a manual Reactor SCRAM due to high Suppression Pool Water Level, two Control Rods will fail to insert, requiring the crew to take actions to insert the Control Rods in accordance with 2-AOI-100-1, Reactor SCRAM.

9. When the crew is Emergency Depressurizing the Reactor due to rising Suppression Pool Water Level, 480V Shutdown Power to the injection valves on the loop the crew selects to inject water into the core will be lost, requiring action to choose another source for Reactor Water injection

The Scenario ends when the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using low pressure systems.

Critical Tasks: 2

1. **When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator.**

   a. **Safety Significance**
   Prevent failure of Primary Containment from over pressurization.

   b. **Cues**
   Procedural Compliance.
   Suppression Pool Level indication.

   c. **Measured by**
   Observation – Both RPS SCRAM switches are depressed.

   d. **Feedback**
   Control Rods insert to their full in position.

   e. **Critical Task Failure Criteria**
   The operating crew fails to proceed without delay and in a controlled manner to insert a manual SCRAM when Suppression Pool Water Level cannot be maintained within the safe area of Curve 4.

2. **When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the NUSO determines that Emergency Depressurization is required. The Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator.**

   a. **Safety Significance**
   Prevent failure of Primary Containment from over pressurization.

   b. **Cues**
   Procedural Compliance.
   RPV Pressure indication.
   Suppression Pool Level indication.

   c. **Measured by**
   Observation – the Nuclear Unit Senior Operator determines (as indicated by announcement or observable transition to 2-C-2, RPV Emergency Depressurization), Emergency Depressurization is required at or before Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.
   AND
   Observation – the Nuclear Unit Senior Operator directs the Operator to open 6 ADS valves

d. **Feedback**

Suppression Pool Water Level trend.
RPV Pressure trend.
MSRV status indication.

e. **Critical Task Failure Criteria**

The operating crew fails to proceed without delay and in a controlled manner to initiate Emergency Depressurization from the time it is announced that Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.

Facility: BFN Scenario Number: NRC-3 Op-Test Number: 21-04

Examiners: ____________________ Operators: SRO: ____________________

____________________ ATC: ____________________

____________________ BOP: ____________________

Initial Conditions: ~2 % Reactor Power.

Turnover: Reactor Startup in progress. Raise Reactor Power and place the Reactor MODE SWITCH in RUN. A thunderstorm watch has just been issued for Limestone County, AL. ‘H’ IRM is bypassed due to noise.

Critical Tasks:

1. When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, SRV Tail Pipe Level Limit, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator (NUSO).
2. When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the NUSO determines that Emergency Depressurization is required. The Operator initiates Emergency Depressurization as directed by the NUSO.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Transfer Seal Steam to Main Steam
2.	N/A	R-OATC R-NUSO	Raise Reactor Power using Control Rods
3.	XS-92-7/42G	C-OATC C-NUSO	IRM Failure
4.	RD08R2227	TS-NUSO	Control Rod Accumulator INOPERABLE
5.	SCHEDULE STACK	C-BOP C-NUSO	Failure of 3A Stack Dilution Fan, Standby Fan Fails to Automatically Start
6.	DG01B	TS-NUSO	3EA Emergency Diesel Generator (EDG) Logic Breaker Tripped
7.	SCHEDULE TORUS	M-ALL	High Suppression Pool Water Level / Emergency Depressurization
8.	RD06R3019 RD06R2615	C-OATC C-NUSO	Two Control Rods Fail to Insert
9.#	ED10A ED10B	C-BOP C-NUSO	480V Shutdown Board Trip

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

Event on previous two NRC Exams

#S Event on previous two NRC Exams Scenario (Spare)

Events

1. The crew will transfer Seal Steam from Auxiliary Steam to Main Steam in accordance with 3-OI-47C, Seal Steam System.

2. The crew will continue the Reactor startup by withdrawing Control Rods in accordance with 3-OI-85, Control Rod Drive System, and 3-GOI-100-1A, Unit Startup and Power Operation.

3. During Control Rod withdrawal, the IRM ‘G’ Range Switch will fail in position 8, requiring the crew to bypass IRM ‘G’ in accordance with 3-OI-92A, Intermediate Range Monitors. The Nuclear Unit Senior Operator (NUSO) will reference Technical Specification 3.3.1.1, RPS Instrumentation, Table 3.3.1.1.

4. Control Rod 22-27 Accumulator Pressure will lower below 940 psig, requiring the NUSO to address Technical Specification 3.1.5, Control Rod SCRAM Accumulators, Condition A.

5. Stack Dilution Fan 3A will fail and the standby Stack Dilution Fan will fail to automatically start. The crew will take action to manually restore Stack Dilution Flow by starting 3B Stack Dilution Fan in accordance with 3-OI-66, Off-Gas System, Section 5.1.

6. The 3EA Emergency Diesel Generator (EDG) Logic Breaker will trip, causing an annunciation in the Control Room and disabling 3EA EDG. The NUSO will address Technical Specification 3.8.1, AC Sources – Operating, Condition B.

7. The crew will respond to a rising Suppression Pool Water Level in accordance with Emergency Operating Instruction 3-EOI-2, Primary Containment Control. Suppression Pool Water Level will not be able to be maintained in the safe area of Curve 4, SRV Tail Pipe Level Limit, requiring the crew to Emergency Depressurize the Reactor in accordance with 3-C-2, Emergency RPV Depressurization.

8. When the crew inserts a manual Reactor SCRAM due to high Suppression Pool Water Level, two Control Rods will fail to insert, requiring the crew to take actions to insert the Control Rods in accordance with 3-AOI-100-1, Reactor SCRAM.

9. When the crew is Emergency Depressurizing the Reactor due to rising Suppression Pool Water Level, 480V Shutdown Power to the injection valves on the loop the crew selects to inject water into the core will be lost, requiring action to choose another source for Reactor Water injection

The Scenario ends when the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using low pressure systems.

Critical Tasks: 2

1. **When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator.**

   a. Safety Significance
   Prevent failure of Primary Containment from over pressurization.

   b. Cues
   Procedural Compliance.
   Suppression Pool Level indication.

   c. **Measured by**
   Observation – Both RPS SCRAM switches are depressed.

   d. **Feedback**
   Control Rods insert to their full in position.

   e. Critical Task Failure Criteria
   The operating crew fails to proceed without delay and in a controlled manner to insert a manual SCRAM when Suppression Pool Water Level cannot be maintained within the safe area of Curve 4.

2. **When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the NUSO determines that Emergency Depressurization is required. The Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator.**

   a. **Safety Significance**
   Prevent failure of Primary Containment from over pressurization.

   b. **Cues**
   Procedural Compliance.
   RPV Pressure indication.
   Suppression Pool Level indication.

   c. **Measured by**
   Observation – the Nuclear Unit Senior Operator determines (as indicated by announcement or observable transition to 3-C-2, RPV Emergency Depressurization), Emergency Depressurization is required at or before Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.
   AND
   Observation – the Nuclear Unit Senior Operator directs the Operator to open 6 ADS valves

d. **Feedback**

Suppression Pool Water Level trend.
RPV Pressure trend.
MSRV status indication.

e. **Critical Task Failure Criteria**

The operating crew fails to proceed without delay and in a controlled manner to initiate Emergency Depressurization from the time it is announced that Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 1 Page 1 of 2

Event Description: Transfer Seal Steam to Main Steam

Time	Position	Applicant's Actions or Behavior
	Driver	**Prior to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not proceed to Event 1, request that the Driver insert Event 1.**
	Driver	**If contacted by the Chief Examiner to insert Event 1, contact the Nuclear Unit Supervisor Operator (NUSO) as the Shift Manager and direct the crew to transfer Seal Steam from Auxiliary Steam to Main Steam.**
	NUSO	Directs the Balance of the Plant Operator (BOP) to transfer Seal Steam from Auxiliary Steam to Main Steam in accordance with 2-OI-47C, Seal Steam System, Section 6.1.
	BOP	2-OI-47C, Seal Steam System Section 6.1, Shifting Supply from Auxiliary Steam to Main Steam **NOTES** 1) This subsection is entered with the Seal Steam supply on Auxiliary Steam. 2) Steps are performed at Panel 2-9-7 in the Control Room unless otherwise specified. 3) To seal the turbine at startup with less than 250 psig Main Steam Pressure, or with worn packings, 2-FCV-1-145, Steam Seal Reg Bypass Valve, is required to be adjusted to supplement Steam Seal Feed Valve, 2-PCV-1-147, to obtain the needed flow. 4) The Steam Seal Feed Valve, 2-PCV-1-147, is designed to handle the Steam Seal Header requirements when Main Steam Pressure exceeds 250 psig. [1] **NOTIFY** Radiation Protection that an RPHP is in effect for the impending action to place Seal Steam System on nuclear steam. **RECORD** time Radiation Protection notified in the Narrative Log.
	NRC	**An RPHP was provided to the crew at turnover.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 1 Page 2 of 2

Event Description: Transfer Seal Steam to Main Steam

Time	Position	Applicant's Actions or Behavior
	BOP	[2] **ENSURE** Reactor Pressure is greater than 200 psig. [3] **ENSURE OPEN** 2-PCV-1-147, STEAM TO STEAM SEAL REGULATOR by placing 2-HS-1-147, STEAM SEAL REGULATOR in AUTO. [4] **OPEN** 2-FCV-1-146A, MAIN STEAM SUPPLY TO STEAM SEAL Valve. [5] **CLOSE** 2-FCV-1-154A, AUX BOILER SUPPLY TO STEAM SEAL VALVE. [6] **CHECK** steam seal header pressure, as indicated on 2-PI-1-148A, STEAM SEAL HDR PRESS, is between 2 1/2 and 9 psig. [7] **CLOSE** 2-12-638, TURBINE SEAL STM VALVE. (TB EL 586', T10 J-Line near the EHC Unit behind Panel 25-111)
	Driver	**When directed as the Turbine Building AUO to close 2-12-638, TURBINE SEAL STM VALVE, acknowledge the direction and inform the crew that 2-12-638 is closed.**
	BOP	[8] **ENSURE** CLOSED 2-FCV-001-0149 using 2-HS-1-149A, STEAM SEAL UNLOADING MANUAL BYPASS VALVE. **CAUTION** Throttling STEAM SEAL REG BYPASS VALVE, 2-BYV-001-0145, with Main Steam Pressure above 250 psig could result in excessive vibration of the Steam Seal Header. [9] **THROTTLE** 2-FCV-1-145, STEAM SEAL REG BYPASS VALVE, to keep Steam Seal Header Pressure, as indicated on 2-PI-1-148A, STEAM SEAL HEADER PRESSURE, between 2 1/2 and 9 psig, if necessary. [10] **CHECK** 2-PI-66-54, STEAM PACKING EXHUAST VACUUM, is between 10 and 12 in H_2O vacuum.
	NRC	**End of Event 1. Proceed to Event 2, Raise Reactor Power Using Control Rods.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 2 Page 1 of 4

Event Description: Raise Reactor Power using Control Rods

Time	Position	Applicant's Actions or Behavior
	NRC	**If the crew does not proceed to Control Rod withdrawal, request that the Driver insert Event 2.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to continue the Reactor Startup.**
	NRC	**During Control Rod withdrawal, Event 3, IRM Failure will automatically be inserted. No action is required by the driver to insert Event 3.**
	NUSO	(The crew may elect to conduct a reactivity re-focus brief) Assumes the Reactivity Manager position. Directs the Operator at the Controls (OATC) to raise Reactor Power to 6-7% in preparation for placing the Reactor MODE SWITCH in RUN, in accordance with 2-GOI-100-1A, Unit Startup and Power Operation, 2-OI-85, Control Rod Drive System, and 2-SR-3.1.3.5(A), Control Rod Coupling Integrity Check.
	OATC	2-GOI-100-1A, Unit Startup and Power Operation Section 5.4, Withdrawal of Control Rods while in Mode 2 **NOTE** 6% to 7% RTP is the target power level to prevent rod blocks below 5% RTP or above 8% Rated Thermal Power (RTP). [73] **CONTINUE** to withdraw Control Rods to raise Reactor Power to 6% to 7% per 2-OI-85, Control Rod Drive System and 2-SR-3.1.3.5(A), Control Rod Coupling Integrity Check. [74] **ENSURE** all operable APRM downscale alarms are reset and no rod blocks exist.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 2 Page 2 of 4

Event Description: Raise Reactor Power using Control Rods

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-85, Control Rod Drive System Section 6.6, Control Rod Withdrawal 6.6.1 Initial Conditions prior to withdrawing Control Rods [1] **REVIEW** Precautions and Limitations in Section 3.7 and 3.8. [2] **ENSURE** the following prior to Control Rod movement: • CRD POWER, 2-HS-85-46 in ON • ROD WORTH MINIMIZER operable and LATCHED in to correct ROD GROUP when Rod Worth Minimizer is enforcing
	OATC	2-OI-85, Control Rod Drive System Section 6.6.2, Actions Required during and Following Control Rod Withdrawal [1] **IF** Control Rod fails to withdraw, **THEN** Refer to Section 8.15 for additional methods to reposition the Control Rod. [2] **IF** Control Rod double notches, or withdraws past its correct / desired position, **THEN** Refer to Section 6.7 for inserting Control Rod to its correct / desired position. [3] **IF** at any time while driving a selected rod during the performance of this section, the Control Rod moves more than one notch from its intended position, **THEN** refer to 2-AOI-85-7, Mispositioned Control Rod. [4] **OBSERVE** the following during Control Rod repositioning: • Control Rod Reed Switch Position Indicators (four rod display) agree with indication on Full Core Display • Nuclear Instrumentation responds as Control Rods move through the Core (This ensures Control Rod is following drive during Control Rod movement.) [5] **ATTEMPT** to minimize Automatic RBM Rod Block as follows: • **STOP** Control Rod Withdrawal (if possible) prior to reaching any RBM Rod Block using the RBM Displays on Panel 9-5 and perform step 6.6.2[6].

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 2 Page 3 of 4

Event Description: Raise Reactor Power using Control Rods

Time	Position	Applicant's Actions or Behavior
	OATC	[6] **IF** Control Rod movement was stopped to keep from exceeding a RBM Setpoint or was caused by a RBM Rod Block, **THEN** **PERFORM** the following at the Unit SROs discretion to "REINITIALIZE" the RBM: [6.1] **PLACE** 2-HS-85-46, CRD POWER, to the OFF position to deselect the Control Rod. [6.2] **PLACE** 2-HS-85-46, CRD POWER, to the ON position. [6.3] **IF** desired, **THEN CONTINUE** to withdraw Control Rods and **PERFORM** applicable section for Control Rod withdraw.
	NRC	**Order of Control Rod withdrawal in accordance with 2-SR-3.1.3.5(A), Control Rod Coupling Integrity Check:**

BFN Unit 2	Control Rod Coupling Integrity Check	2-SR-3.1.3.5(A) Rev. 0025 Page 121 of 363

Attachment 5
(Page 20 of 39)
A2 Startup Sequence Control Rod Movement Data Sheet

Date ________

RWM GP	ROD NUMBER	FROM	TO	Rod Movement Completed Signoffs UO (AC) [1]	Peer Check [2]
23	10-35	04	06		
23	26-51	04	06		
23	34-51	04	06		
23	50-35	04	06		
23	50-27	04	06		
23	34-11	04	06		
23	26-11	04	06		
23	10-27	04	06		
24	18-43	04	06		SRM-A
24	42-43	04	06		SRM-B
24	42-19	04	06		SRM-C
24	18-19	04	06		SRM-D
25	26-35	04	06		
25	34-35	04	06		
25	34-27	04	06		
25	26-27	04	06		

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 2 Page 4 of 4

Event Description: Raise Reactor Power using Control Rods

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-85, Control Rod Drive System Section 6.6.3, Control Rod Notch Withdrawal [1] **SELECT** the desired Control Rod by depressing the appropriate CRD ROD SELECT pushbutton, 2-XS-85-40. [2] **ENSURE** 2-PDI-85-17A, CRD DRIVE WATER HEADER DP is between 250 -270 psid by throttling 2-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE, as necessary. [3] N/A [4] **OBSERVE** the following for selected Control Rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED • White light on the Full Core Display ILLUMINATED • Rod Out Permit light ILLUMINATED [5] **ENSURE** ROD WORTH MINIMIZER operable and LATCHED in to correct ROD GROUP when Rod Worth Minimizer is enforcing. [6] **PLACE** 2-HS-85-48, CRD CONTROL SWITCH, in ROD OUT NOTCH and **RELEASE**. [7] **OBSERVE** Control Rod settles into desired position **AND** ROD SETTLE light extinguishes. [8] N/A [9] N/A
	OATC	2-OI-85, Control Rod Drive System Section 6.6.5, Return to Normal after Completion of Control Rod Withdrawal [1] **WHEN** Control Rod movement is no longer desired **AND** deselecting Control Rods is desired, **THEN**: [1.1] **PLACE** 2-HS-85-46, CRD POWER, in OFF. [1.2] **PLACE** 2-HS-85-46, CRD POWER, in ON.
	NRC	**End of Event 2. Event 3, IRM Failure, is automatically inserted on simulator setup. No action is required by the Driver to insert Event 3.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 3 Page 1 of 2

Event Description: IRM Failure

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 3, IRM 'B' Failure, is automatically inserted on Simulator setup. No action is required by the Driver to insert Event 3.**
	NRC	**IRM 'B' will need to be ranged up after approximately 13 Control Rods are pulled (about Control Rod 42-11 in Group 25).**
	OATC	During Control Rod withdrawal will be ranging IRM Switches to prevent a Control Rod Block or Reactor SCRAM Signal. When the OATC attempts to range IRM 'B' UP, the reading will not change. Stops Control Rod withdrawal. Notifies the NUSO.
	NUSO	Directs the OATC to bypass IRM 'B' in accordance with 2-OI-92A, Intermediate Range Monitors.
	OATC	2-OI-92A, Intermediate Range Monitors Section 6.1, Bypassing an IRM Channel **CAUTION** NPG-SPP 10.4 requires approval of the Plant Manager or his designee prior to any planned operation with IRMs bypassed unless bypassing is specifically allowed within approved procedures. **NOTES** 1) It is not necessary for a bypassed IRM channel to have its detector inserted into the Core. 2) Only one IRM in each trip system can be bypassed at a time. 3) All operations are performed on Panel 2-9-5 unless specifically stated otherwise. [1] **REVIEW** all precautions and limitations in Section 3.0. [2] **PLACE** the appropriate IRM Bypass selector switch to the BYPASS position: • 2-HS-92-7A/S4A, IRM BYPASS • 2-HS-92-7A/S4B, IRM BYPASS [3] **CHECK** that the Bypassed light is illuminated.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 3 Page 2 of 2

Event Description: IRM Failure

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>Driver</td><td>If contacted by the crew as the Shift Manager, acknowledge any report given.
If contacted as the Plant Manager / Shift Manager for permission to continue Control Rod withdrawal with IRM 'B' bypassed, acknowledge the request and ask for their recommendation. Concur with any recommendation given.</td></tr>
<tr><td></td><td>OATC</td><td>Informs the NUSO that IRM 'B' is bypassed.</td></tr>
<tr><td></td><td>NUSO</td><td>References Technical Specification 3.3.1.1, RPS Instrumentation.
Declares an Information Only LCO based on only three IRM channels being required per trip system in accordance with Table 3.3.1.1-1 (page 1 of 3).

RPS Instrumentation
3.3.1.1

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation
<table>
<tr><th>FUNCTION</th><th>APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS</th><th>REQUIRED CHANNELS PER TRIP SYSTEM</th><th>CONDITIONS REFERENCED FROM REQUIRED ACTION D.1</th><th>SURVEILLANCE REQUIREMENTS</th><th>ALLOWABLE VALUE</th></tr>
<tr><td>1. Intermediate Range Monitors</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>a. Neutron Flux - High</td><td>2</td><td>3</td><td>G</td><td>SR 3.3.1.1.1
SR 3.3.1.1.3
SR 3.3.1.1.5
SR 3.3.1.1.6
SR 3.3.1.1.9
SR 3.3.1.1.14</td><td>≤ 120/125 divisions of full scale</td></tr>
<tr><td></td><td>5(a)</td><td>3</td><td>H</td><td>SR 3.3.1.1.1
SR 3.3.1.1.4
SR 3.3.1.1.9
SR 3.3.1.1.14</td><td>≤ 120/125 divisions of full scale</td></tr>
<tr><td>b. Inop</td><td>2</td><td>3</td><td>G</td><td>SR 3.3.1.1.3
SR 3.3.1.1.14</td><td>NA</td></tr>
<tr><td></td><td>5(a)</td><td>3</td><td>H</td><td>SR 3.3.1.1.4
SR 3.3.1.1.14</td><td>NA</td></tr>
</table></td></tr>
<tr><td></td><td>NRC</td><td>End of Event 3. Request that the driver insert Event 4, Control Rod Accumulator Inoperable.</td></tr>
</table>

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 4	Page 1 of 4
Event Description: Control Rod Accumulator Inoperable			

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5, Control Rod Accumulator Inoperable.**
	NRC	**The alarm will occur on Control Rod 22-27 Accumulator.**
	OATC	Acknowledges and reports the following alarm to the NUSO: • CRD ACCUMULATOR PRESSURE LOW / LEVEL HIGH, 2-9-5A, Window 29
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the appropriate Alarm Response Procedure.
	OATC	Alarm Response Procedure, 2-ARP-9-5A CRD ACCUMULATOR PRESSURE LOW / LEVEL HIGH, 2-9-5A, Window 29 Operator Action: A. **CHECK** alarm by amber background light illuminated on Full Core Display. B. **LOG IN** the Narrative Log the Control Rod number, and time alarm was received. **NOTE** If any of the following fuses is/are cleared, the local indications at Panel 25-4 and 25-22 will NOT illuminate. C. **IF** multiple accumulator lights are lit on Panel 9-5, **THEN CHECK** Fuses 2-FU1-085-25-004G, -004H in Panel 25-4 and 2-FU1-085-25-022G, -022H in Panel 25-22. D. **DISPATCH** personnel to Panel 25-4 (east side), Panel 25-22 (west side) El 565', to determine if level high or pressure low. E. **DEPRESS** pushbutton for associated HCU to determine if alarm is caused by level high or pressure low as follows: • If alarm is due to high level, the red light will extinguish • If light stays illuminated, alarm is due to low N2 pressure

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 4 Page 2 of 4

Event Description: Control Rod Accumulator Inoperable

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Reactor Building AUO to respond to the CRD Accumulator alarm, acknowledge the direction. Wait two minutes and report to the crew that CRD 22-27 Accumulator Pressure is 900 psig and lowering. You are unable to raise Accumulator Nitrogen Pressure by recharging.**
	OATC	F. IF alarm is valid, THEN REFER TO 2-OI-85, Control Rod Drive System and 2-AOI-85-3, CRD System Failure.
	NRC	**The actions 2-OI-85, Control Rod Drive System and 2-AOI-85-3, CRD System Failure are covered starting on page xx of xx.**
	OATC	G. **IF** accumulator pressure is less than or equal to 940 psig, **THEN DECLARE** Control Rod HCU "INOPERABLE". H. **IF** associated HCU's nitrogen pressure is found less than 940 psi, **THEN INITIATE** a Condition Report (CR) to calibrate the pressure switch. The HCU will NOT be declared operable until the switch has been calibrated.
	NUSO	I. **IF** alarm is due to low pressure with pressure greater than 940 psig and accumulator CANNOT be recharged within one hour, **THEN EVALUATE** per Tech Spec 3.1.5. 1. **IF** the Control Rod is declared SLOW, **THEN REFER** TO TECH SPEC 3.1.4. Currently no more than 13 OPERABLE Control Rods are to be slow and no more than 2 OPERABLE Control Rods that are slow occupy adjacent locations. 2. **IF** Control Rod is declared INOPERABLE, **THEN REFER TO** Tech Spec 3.1.3. J. **RECORD** this evaluation in narrative log. K. N/A

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 4 Page 3 of 4

Event Description: Control Rod Accumulator Inoperable

Time	Position	Applicant's Actions or Behavior	
	NUSO	If the Control Rod is declared INOPERABLE, references Tech Spec 3.1.3, Control Rod OPERABILITY. LCO 3.1.3 Each Control Rod shall be OPERABLE APPLICABILITY: MODES 1 and 2 NOTE: Separate Condition entry is allowed for each Control Rod. **CONDITION:** C. One or more Control Rods INOPERABLE for reasons other than Condition A or B.	
	NUSO	**REQUIRED ACTION:** NOTE: RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of INOPERABLE Control Rod and continued operation. C.1 – Fully insert INOPERABLE Control Rod. AND C.2 – Disarm the associated CRD.	**COMPLETION TIME:** C.1 – 3 hours C.2 – 4 hours
	NUSO	If the Control Rod is declared SLOW, references Tech Spec 3.1.4, Control Rod SCRAM Times. Currently, there are less than 13 OPERABLE Control Rods that are slow, and there are no OPERABLE Control Rods that are slow that occupy adjacent locations. Therefore, there are no actions required for Tech Spec 3.1.4, Control Rod SCRAM Times.	

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 4 Page 4 of 4

Event Description: Control Rod Accumulator Inoperable

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.1.5, Control Rod SCRAM Accumulators LCO 3.1.5: Each Control Rod SCRAM Accumulator shall be OPERABLE APPLICABILITY: Modes 1 and 2 NOTE: Separate Condition entry is allowed for each Control Rod SCRAM Accumulator. **CONDITION**: A.1 One Control Rod SCRAM Accumulator inoperable with Reactor Steam Dome Pressure ≥ 900 psig.	
	NUSO	**REQUIRED ACTION:** A.1 NOTE: Only applicable if the associated Control Rod SCRAM time was within the limits of Table 3.1.4-1 during the last SCRAM time Surveillance. Declare the associated Control Rod SCRAM Time "slow". <u>OR</u> A.2- Declare the associated Control Rod INOPERABLE.	**COMPLETION TIME:** A.1 – 8 hours A.2 – 8 hours
	NRC	**End of Event 4. Request that the driver insert Event 5, Failure of 2A Stack Dilution Fan, Standby Fan Fails to Automatically Start.**	

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 5 Page 1 of 1

Event Description: Failure of 2A Stack Dilution Fan, Standby Fan Fails to Automatically Start

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4, Failure of 2A Stack Dilution Fan, Standby Fan Fails to Automatically Start.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • STACK GAS DILUTION AIR FLOW LOW, 2-9-7A, Window 3
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 2-ARP-9-7A STACK GAS DILUTION AIR FLOW LOW, Window 3 Operator Action: A. **ENSURE** alternate fan ON and damper open, (red light illuminated) on Panel 2-9-7.
	BOP	Determines that the standby Stack Dilution Fan did not automatically start and manually starts 2B Stack Dilution Fan.
	BOP	B. **DISPATCH** personnel to stack to check and report status of the following for both fans: 1. Fan motor. 2. Fan belts. 3. Damper stuck closed. C. **CHECK** breaker 5C on 480V Diesel Aux Bd A and B.
	Driver	**If contacted as an AUO, acknowledge any direction given.**
	NRC	**End of Event 5. Request that the Driver insert Event 6, 'A' Emergency Diesel Generator (EDG) Logic Breaker Tripped.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 6 Page 1 of 4

Event Description: 'B' Emergency Diesel Generator (EDG) Logic Breaker Tripped.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6, 'B' Emergency Diesel Generator (EDG) Logic Breaker Tripped.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • DIESEL GENERATOR B CONTROL POWER OFF, 0-9-23-7, Window 14
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 1/2-ARP-9-23B DIESEL GENERATOR B CONTROL POWER OFF, Window 14 Operator Action: A. **CHECK** any other alarms on Panels 9-8 or 0-9-23-7 which may indicate problem area. B. **DISPATCH** personnel to check the Probable Causes listed above. C. **IF** loss of normal power has occurred, **THEN TRANSFER** to alternate power source per 0-OI-57D, DC Electrical System. D. **IF** unable to restore power or to correct problem, **THEN REFER TO** Technical Specification 3.8.1, 3.3.8.1, 3.8.4, & 3.8.7. E. **LOG** valid events and actions taken in narrative log.
	Driver	**If contacted as the Outside NUSO, Work Control, AUO, or Electrical Maintenance, acknowledge any direction given.** **Wait 2 minutes and report that the Logic Breaker for EDG 'B' is tripped. If directed to attempt to close the Logic Breaker, report that the breaker will not close.**

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 6	Page 2 of 4

Event Description: ‘B’ Emergency Diesel Generator (EDG) Logic Breaker Tripped.

Time	Position	Applicant’s Actions or Behavior	
	NRC	**There are no required actions for Tech Spec 3.3.8.1, LOP Instrumentation or Tech Spec 3.8.7, Distribution Systems – Operating.**	
	NUSO	References Technical Specification 3.8.4, DC Sources – Operating. LCO 3.8.4 The following DC electrical power sources shall be OPERABLE: a. Unit DC subsystems 1, 2, and 3; b. Shutdown Board DC subsystems A, B, C, and D; c. Unit 1 and 2 Diesel Generator (DG) DC subsystems; d. Unit 3 DG DC subsystem(s) supporting DG(s) required to be OPERABLE by LCO 3.8.1, "AC Sources - Operating"; and e. Unit 3 Shutdown Board DC subsystem 3EB needed to support equipment required to be OPERABLE by LCO 3.7.3, "Control Room Emergency Ventilation (CREV) System." APPLICABILITY: MODES 1, 2, and 3 **CONDITION:** C. One or more DG DC electrical power subsystem(s) INOPERABLE	
	NUSO	**REQUIRED ACTION:** C.1 – One or more DG DC subsystem(s) inoperable.	**COMPLETION TIME:** C.1 – Immediately

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 6 Page 3 of 4

Event Description: 'B' Emergency Diesel Generator (EDG) Logic Breaker Tripped.

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td colspan="2">Declares 'B' EDG INOPERABLE.
References Technical Specification 3.8.1, AC Sources – Operating.
LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:
a. Two qualified circuits between the Offsite Transmission Network and the onsite Class 1E AC Electrical Power Distribution System;
b. Unit 1 and 2 Diesel Generators (DGs) with two divisions of 480 V load shed logic and Common Accident Signal Logic OPERABLE;
and
c. Unit 3 DG(s) capable of supplying the Unit 3 4.16 kV Shutdown Board(s) required by LCO 3.8.7, "Distribution Systems - Operating."
APPLICABILITY: MODES 1, 2, and 3
CONDITION: One required Unit 1 and 2 DG INOPERABLE</td></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
B.1 – Verify power availability from the offsite transmission network.

<u>AND</u>

B.2 – Evaluate availability of both temporary diesel generators (TDGs).

<u>AND</u>

B.3. – Declare required feature(s), supported by the inoperable Unit 1 and 2 DG, inoperable when the redundant required feature(s) are inoperable.</td><td>COMPLETION TIME:
B.1 – 1 hour
<u>AND</u>
Once per 8 hours thereafter

B.2 – 1 hour
<u>AND</u>
Once per 12 hours thereafter

B.3 – 4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 6 Page 4 of 4

Event Description: 'B' Emergency Diesel Generator (EDG) Logic Breaker Tripped.

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION: (continued)** AND B.4.1 Determine OPERABLE Unit 1 and 2 DG(s) are not inoperable due to common cause failure. OR B.4.2 – Perform SR 3.8.1.1 for OPERABLE Unit 1 and 2 DG(s). AND B.5 – Restore Unit 1 and 2 DG to OPERABLE status	**COMPLETION TIME:** B.4.1 – 24 hours B.4.2 – 24 hours B.5 – 7 days AND 24 hours from discovery of Condition B entry ≥ 6 days concurrent with unavailability of TDG(s) AND 14 days AND 21 days from discovery of failure to meet LCO
	NRC	**End of Event 6. Request that the driver insert Event 7, High Suppression Pool Water Level / Emergency Depressurization.**	

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 1 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7, High Suppression Pool Water Level / Emergency Depressurization.**
	NRC	**Event 8, Two Control Rods Fail to Insert and Event 9, 480V Shutdown Board Trip, will occur during Event 7 and are automatically entered by Simulator Setup. No action is required by the driver to insert Event 8 or Event 9.**
	BOP	Acknowledges and reports the following alarms as they are received: • DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL, 2-9-3B, Window 26 • SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, 2-9-3B, Window 15
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedures.
	BOP	Alarm Response Procedure, 2-ARP-9-3B DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL, Window 26 Operator Action: A. **CHECK** alarm by checking Drywell to Suppression Chamber DP. B. **REFER TO** 2-OI-64, Primary Containment System. C. **REFER** TO Tech Spec Section 3.6.2.6, Drywell-to-Suppression Chamber Differential Pressure.
	NRC	**Due to time constraints, Tech Spec evaluation for this event is not required and should not be used to evaluate the applicant's Tech Spec competency.**

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 7	Page 2 of 10

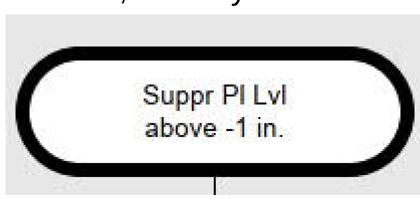
Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 2-ARP-9-3B SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, Window 15 Operator Action: A. **CHECK** Suppression Pool Water Level using multiple indications. B. **IF** level is low, **THEN DISPATCH** personnel to check for leaks. C. **IF** level is high, **THEN CHECK** for RCIC, HPCI, Core Spray, or RHR draining to the Suppression Pool, and **CHECK** 2-TR-64-161, SUPPRESSION POOL WATER TEMPERATURE and 2-TR-64-162, SUPPRESSION POOL WATER TEMPERATURE. D. **REFER TO** 2-OI-74, Residual Heat Removal System, Section 8.0. E. **REFER TO** Tech Spec 3.6.2.2, Suppression Pool Water Level. F. **IF** level is above -1" or below -6.25" **AND NOT** in Mode 4 or Mode 5 **THEN** (otherwise N/A) **ENTER** 2-EOI-2, Primary Containment Control. G. **IF** level is above -1" or below -6.25" **AND** in Mode 4 or Mode 5 **THEN** (otherwise N/A) 1. **EVALUATE** plant conditions to **DETERMINE** if 2-EOI-2, Primary Containment Control entry is appropriate. 2. **RECORD** actions in NOMS log.
	BOP	Determines that the cause for the Drywell to Suppression Chamber DP alarm is rising Suppression Pool Water Level, and informs the NUSO.
	NUSO	Directs the BOP to monitor Suppression Pool Water Level and to provide an update when level reaches (-) 1 inch.
	BOP	Acknowledges and reports the following alarm to the NUSO when received: • SUPPRESSION POOL LEVEL HIGH, 2-9-3F, Window 12
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 3 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>Alarm Response Procedure, 2-ARP-9-3F
SUPPRESSION POOL LEVEL HIGH, Window 12

A. CHECK CST 2 and Suppression Pool Water Level using multiple indications.
B. ENSURE HPCI Suction automatically transfers to the Suppression Pool.
C. IF automatic transfer fails, THEN REFER TO 2-OI-73, High Pressure Coolant Injection System, Section 6.1.
D. REFER TO Tech Spec 3.5.1, ECCS – Operating.</td></tr>
<tr><td></td><td>NRC</td><td>Due to time constraints, Tech Spec evaluation for this event is not required and should not be used to evaluate the applicant's Tech Spec competency.</td></tr>
<tr><td></td><td>NUSO</td><td>When appropriate, enters 2-EOI-2, Primary Containment Control on high Suppression Pool Water Level (level above (-) 1 inch).</td></tr>
<tr><td></td><td>NUSO</td><td>2-EOI-2, Primary Containment Control

</td></tr>
<tr><td></td><td>NUSO</td><td>SP/L-1
<table>
<tr><td colspan="2">MONITOR and CONTROL Suppression Pool Water Level (-) 6 inches to (-) 1 inch.</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Suppression Pool Water Level CANNOT be maintained below (-) 1 inch.</td><td>A</td></tr>
<tr><td>Suppression Pool Water Level CANNOT be maintained above (-) 6 inches.</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 4 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	 SP/L-3 **MAINTAIN** Suppression Pool Water Level below 19 ft. (APPX 18, 20K) SP/L-4 **WHEN** Suppression Pool Level CANNOT be maintained below (APPX 9) 19 feet **STOP** DW Sprays
	NUSO	Directs the BOP to control Suppression Pool Water Level in accordance with 2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup.
	Driver	**If contacted as an AUO to perform any steps locally per 2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup, acknowledge any direction given.**
	BOP	2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup [1] N/A [2] N/A [3] **IF** directed by the NUSO, **THEN REMOVE** water from the Suppression Pool as follows: [3.1] **DISPATCH** personnel to perform the following (Unit 2 RB, El 519 ft, Torus Area): [3.1.1] **ENSURE OPEN** 2-SHV-074-0786A (B), RHR DRAIN PUMP 2A(2B) DISCH TO MAIN CONDENSER/RADWASTE VALVE.

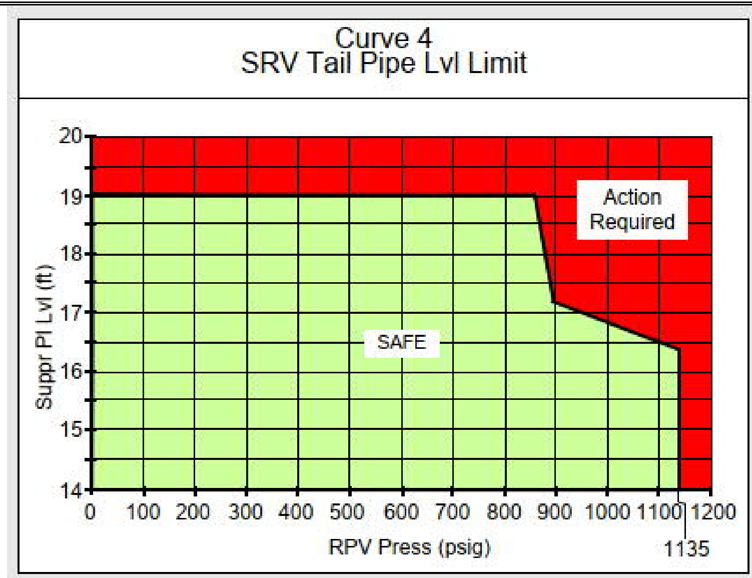
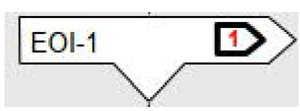
Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 5 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	BOP	[3.2] **IF** Main Condenser is the desired drain path, **THEN OPEN** 2-FCV-74-62, RHR MAIN CONDENSER FLUSH VALVE. [3.3] **IF** Radwaste is the desired drain path, **THEN PERFORM** the following: [3.3.1] **ESTABLISH** communications with Radwaste. [3.3.2] **OPEN** 2-FCV-74-63, RHR RADWASTE SYSTEM FLUSH VALVE.
	Driver	**After 2 minutes, report that the outside portions of 2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup are complete. If directed to start the RHR Drain Pump, report that the RHR Drain Pump has been started.** **If contacted as the Rad Waste Operator, acknowledge any reports or direction given.**
	NUSO	 SP/L-6 **MAINTAIN** Suppression Pool Level within the safe area of Curve 4 (APPX 18, 20K)

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 6 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	
	NUSO	SP/L-7 **WHEN** Suppression Pool Level CANNOT be maintained within the safe area of Curve 4 (APPX 9) 
	NUSO	Directs the OATC to insert a manual Reactor SCRAM and directs the crew to enter 2-AOI-100-1, Reactor SCRAM.
	NRC	**Event 8, 2 Control Rods Fail to Insert, is automatically entered on Simulator Setup. No action is required by the driver to insert Event 8. See page xx of xx for Event 8 actions.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 7 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	OATC	Inserts a manual Reactor SCRAM. 2-AOI-100-1, Reactor SCRAM Immediate Actions [1] **DEPRESS** 2-HS-99-5A/S3A, REACTOR SCRAM A and 2-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 2-9-5. [2] **PLACE** 2-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] **IF** all Control Rods can NOT be verified fully inserted, **THEN INITIATE** ARI.
	OATC	Determines that there are two (2) rods out.
	NRC	**When the Reactor MODE SWITCH is placed in SHUTDOWN, the Feedwater Heater Outlet Isolation Valves will close.** **See page xx of xx for actions for Event 8, Two Control Rods Fail to Insert and page xx of xx for actions for Event 9, 480V Shutdown Board Trip.**
	OATC	[4] **IF** Reactor Power is 5% or BELOW, **THEN REPORT** the following to the NUSO: • Reactor SCRAM • Mode Switch is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Reactor Power level [5] N/A

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 8 of 10

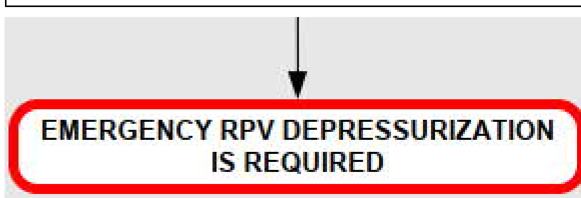
Event Description: High Suppression Pool Water Level / Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>Following the Reactor SCRAM, enters 2-EOI-1A, ATWS RPV Control and directs the crew to perform the following:

Maintain Reactor Pressure to ensure that Suppression Pool Level is maintained within the safe area of Curve 4 in accordance with 2-EOI-Appendix-8B, Reopening MSIVs/Bypass Valve Operation
Maintain Reactor Water Level using in accordance with 2-EOI-Appendix-5D, Injection System Lineup HPCI or 2-EOI-Appendix-5C, Injection System Lineup RCIC
Insert Control Rods
</td></tr>
<tr><td></td><td>NUSO</td><td>(Continuing actions of 2-EOI-2, Primary Containment Control)

SP/L-8
WHEN Suppression Pool Level and RPV Pressure CANNOT be maintained within the safe area of Curve 4 (APPX 9).

SP/L-9
STOP injection into RPV from sources external to Primary Containment EXCEPT from systems required to assure Adequate Core Cooling or shut down the Reactor

SP/L-10
WHEN Suppression Pool Level and RPV Pressure CANNOT be restored and maintained within the safe area of Curve 4

</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 9 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>Enters 2-C-2, Emergency RPV Depressurization

C2-1
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr><tr><td>It is anticipated that RPV depressurization will result in loss of injection required for Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr><tr><td>Containment Water Level CANNOT be maintained below 44 feet</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>C2-2
<table><tr><td>IF</td><td>Drywell Pressure is above 2.45 psig</td></tr><tr><td>THEN</td><td>PREVENT injection from ONLY those Core Spray and LPCI pumps NOT required to assure Adequate Core Cooling (APPX 4)</td></tr></table></td></tr>
</table>

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 7	Page 10 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	C2-3 **EMERGENCY DEPRESSURIZE** the RPV **IF** Suppression Pool Water Level is above 5.5 feet **THEN OPEN** 6 MSRVs (ADS Valves preferred) • OK to exceed 100 F/hr Cooldown Rate **IF** / **THEN**: Drywell Control Air becomes unavailable — **NO ACTION REQUIRED** Less than 4 MSRVs can be opened AND RPV Pressure is 80 psi or more above Suppression Chamber Pressure — **NO ACTION REQUIRED**
	NRC	**The first loop of Low Pressure Injection (Core Spray/RHR) that the crew attempts to use will result in a loss of the 480V Shutdown Board for that loop. See Event 9 on page xx of xx.**
	NRC	**End of Event 7. When the crew has inserted all Control Rods, Emergency Depressurized the Reactor, and has control of Reactor Water level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.**

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 8	Page 1 of 3
Event Description: Two Control Rods Fail to Insert			

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8 is automatically entered by simulator setup. No action is required by the Driver to insert Event 8.**
	OATC	Following the Reactor SCRAM and after initiating Alternate Rod Insertion (ARI), determines that all Control Rods are not in. Informs the NUSO that two rods are out, and that Reactor Power is less than 5%.
	NUSO	Directs the OATC to insert Control Rods in accordance with 2-AOI-100-1, Reactor SCRAM.
	NRC	**Not all Subsequent Actions of 2-AOI-100-1, Reactor SCRAM, are listed below.**
	OATC	2-AOI-100-1, Reactor SCRAM Subseqent Actions [16] **IF** all rods are **NOT** inserted to Position 02 or beyond, **THEN DIRECT** Reactor Engineer to commence determination that Reactor will remain subcritical under all conditions without boron. (Otherwise N/A)
	Driver	**If contacted as the Reactor Engineer, acknowledge any direction or report given.**
	OATC	[17] **IF** any Control Rod fails to fully insert and is required to be re-SCRAMMED, **THEN PERFORM** the following, as required: [17.1] **RESET** the SCRAM per Steps 4.2[24] thru 4.2[24.12]. [17.2] **CHECK** WEST and EAST CRD DISCH VOL WTR LVL HIGH HALF SCRAM annunciators (2-XA-55-4A-1 and 4A-29) reset. [17.3] **INITIATE** manual SCRAM. [17.4] **REPEAT** Step 4.2[17], as necessary, as long as rod motion is observed. [18] **IF** any Control Rod fails to fully insert and it is required to Drive Control Rods, **THEN REFER TO** 2-OI-85, Control Rod Drive System.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 8 Page 2 of 3

Event Description: Two (2) Control Rods Fail to Insert

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-85, Control Rod Drive System Section 6.7, Control Rod Insertion [1] **REVIEW** Precautions and Limitations in Section 3.7 and 3.8. [2] **OBSERVE** the following during Control Rod repositioning: • Control Rod reed switch position indicators (four rod display) agree with indication on Full Core Display • Nuclear Instrumentation responds as Control Rods move through the Core (This ensures Control Rod is following drive during Control Rod movement.) [3] **ENSURE** the following prior to Control Rod movement: • CRD POWER, 2-HS-85-46 in ON. • When Rod Worth Minimizer is enforcing, the ROD WORTH MINIMIZER is operable and LATCHED in to the correct ROD GROUP [4] **PERFORM** the following to insert the Control Rod as appropriate. • Control Rod Notch Insertion per Section 6.7.2 • Control Rod Continuous Insertion per Section 6.7.3
	OATC	2-OI-85, Control Rod Drive System Section 6.7.3, Continuous Insertion of Control Rod [1] **ENSURE** Section 6.7.1 has been performed. [2] **SELECT** desired Control Rod by depressing appropriate CRD ROD SELECT pushbutton, 2-XS-85-40. [3] **OBSERVE** the following for selected Control Rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED • White light on Full Core Display ILLUMINATED [4] **PLACE** AND **HOLD** 2-HS-85-48, CRD CONTROL SWITCH, to ROD IN. [5] **WHEN** Control Rod notch reaches even rod notch position prior to desired final Control Rod notch position, **THEN RELEASE** 2-HS-85-48, CRD CONTROL SWITCH.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 8 Page 3 of 3

Event Description: Two (2) Control Rods Fail to Insert

Time	Position	Applicant's Actions or Behavior
	NRC	**When the OATC has selected and each stuck Control Rod and begins to drive the rod in, the malfunction will clear to allow rod insertion. As a result, the Control Rod will drive in at a much faster rate than normal.**
	OATC	[6] **OBSERVE** the Control Rod settles into desired position **AND** the ROD SETTLE light extinguishes. [7] **IF** Control Rod settles one notch past its intended position, **THEN** with Unit SROs permission return the Control Rod to the intended position per Section 6.6. [8] **IF** the Control Rod moves more than one notch from its intended position, **THEN** refer to 2-AOI-85-7 MISPOSITIONED CONTROL ROD. [9] **WHEN** Control Rod movement is no longer required **AND** deselecting Control Rods is desired, **THEN**: [9.1] **PLACE** 2-HS-85-46, CRD POWER, in OFF. [9.2] **PLACE** 2-HS-85-46, CRD POWER, in ON.
	NRC	**End of Event 8. When the crew has inserted all Control Rods, Emergency Depressurized the Reactor, and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using low pressure systems, end of Scenario.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 9 Page 1 of 4

Event Description: 480V Shutdown Board Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9, 480V Shutdown Board Trip, is automatically entered by simulator setup. No action is required by the Driver to insert Event 9.** **The first loop of Low Pressure Injection (Core Spray/RHR) that the crew attempts to use will result in a loss of the 480V Shutdown Board for that loop.**
	NUSO	During Emergency Depressurization, directs the BOP to maintain Reactor Water Level using Core Spray or RHR in accordance with any of the following EOI Appendices: • 2-EOI-Appendix-6B, Injection Subsystems Lineup – RHR System I LPCI Mode (see below) • 2-EOI-Appendix-6C, Injection Subsystems Lineup – RHR System II LPCI Mode (see page xx of xx) • 2-EOI-Appendix-6D, Injection Subsystems Lineup –Core Spray System (see page xx of xx) • 2-EOI-Appendix-6E, Injection Subsystems Lineup – Core Spray System II (see page xx of xx)
	NRC	**If the crew selects Loop II of RHR or Core Spray to maintain Reactor Water Level, proceed to page xx of xx for the procedure(s) for injection. If the crew selects Loop I of RHR see below for the procedure(s) for injection.**
	BOP	**IF USING LOOP I OF RHR FOR INJECTION:** 2-EOI-Appendix-6B, Injection Subsystems Lineup RHR System I LPCI Mode [1] **IF** Adequate Core Cooling is assured **AND** It becomes necessary to bypass the LPCI Injection Valve auto open signal to control injection, **THEN PLACE** 2-HS-74-155A, LPCI SYS-I OUTBD INJECTION VALVE BYPASS SELECT, in **BYPASS**. [2] **ENSURE OPEN** 2-FCV-74-7, RHR SYSTEM I MINIMUM FLOW VALVE.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 9 Page 2 of 4

Event Description: 480V Shutdown Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **ENSURE OPEN** the following valves: • 2-FCV-74-1, RHR PUMP 2A SUPPRESSION POOL SUCTION VALVE • 2-FCV-74-12, RHR PUMP 2C SUPPRESSION POOL SUCTION VALVE [4] ENSURE CLOSED the following valves: • 2-FCV-74-61, RHR SYS I DRYWELL SPRAY INBOARD VALVE • 2-FCV-74-60, RHR SYS I DRYWELL SPRAY OUTBOARD VALVE • 2-FCV-74-57, RHR SYS I SUPPRESSION CHAMBER/POOL ISOLATION VALVE • 2-FCV-74-58, RHR SYS I SUPPRESSION CHAMBER SPRAY VALVE • 2-FCV-74-59, RHR SYS I SUPPRESSION POOL COOLING/TEST VALVE [5] ENSURE RHR Pump 2A and / or 2C running.
	CREW	Determines that 2A 480V Shutdown Board has tripped. Proceeds to use the appropriate EOI Appendix for injection with low pressure systems for the opposite loop.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 9 Page 3 of 4

Event Description: 480V Shutdown Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	**IF USING LOOP I OF CORE SPRAY FOR INJECTION:** 2-EOI-Appendix-6D, Injection Subsystems Lineup Core Spray System I [1] **VERIFY OPEN** the following valves: • 2-FCV-75-2, CORE SPRAY PUMP 2A SUPPRESSION POOL SUCTION VALVE • 2-FCV-75-11, CORE SPRAY PUMP 2C SUPPRESSION POOL SUCTION VALVE • 2-FCV-75-23, CORE SPRAY SYS I OUTBOARD INJECTION VALVE [2] **VERIFY CLOSED** 2-FCV-75-22, CORE SPRAY SYSTEM I TEST VALVE. [3] **VERIFY** Core Spray Pump 2A and/or 2C running.
	CREW	Determines that 2A 480V Shutdown Board has tripped. Proceeds to use the appropriate EOI Appendix for injection with low pressure systems for the opposite loop.
	BOP	**IF USING LOOP II OF RHR FOR INJECTION:** 2-EOI-Appendix-6C, Injection Subsystems Lineup RHR System II LPCI Mode [1] **IF** Adequate Core Cooling is assured **AND**, it becomes necessary to bypass LPCI Injection Valve auto open signal to control injection, **THEN PLACE** 2-HS-74-155B, LPCI SYS II OUTBOARD INJECTION VALVE BYPASS SELECT IN **BYPASS**. [2] **ENSURE OPEN** 2-FCV-74-30, RHR SYSTEM II MINIMUM FLOW VALVE. [3] **ENSURE OPEN** the following valves: • 2-FCV-74-24, RHR PUMP 2B SUPPR POOL SUCT VALVE • 2-FCV-74-35, RHR PUMP 2D SUPPR POOL SUCT VALVE

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 9 Page 4 of 4

Event Description: 480V Shutdown Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **ENSURE CLOSED** the following valves: • 2-FCV-74-75, RHR SYS II DW SPRAY INBOARD VALVE • 2-FCV-74-74, RHR SYS II DW SPRAY OUTBOARD VALVE • 2-FCV-74-71, RHR SYS II SUPPR CHAMBER/POOL ISOLATION VALVE • 2-FCV-74-72, RHR SYS II SUPPRESSION CHAMBER SPRAY VALVE • 2-FCV-74-73, RHR SYS II SUPPRESSION POOL COOLING/TEST VALVE [5] **ENSURE** RHR Pump 2B and/or 2D running.
	BOP	Determines that 2B 480V Shutdown Board has tripped. Proceeds to use the appropriate EOI Appendix for injection with low pressure systems for the opposite loop.
	BOP	**IF USING LOOP II OF CORE SPRAY FOR INJECTION:** 2-EOI-Appendix-6E, Injection Subsystems Lineup Core Spray System II [1] **VERIFY OPEN** the following valves: • 2-FCV-75-30, CORE SPRAY PUMP 2B SUPPR POOL SUCT VALVE • 2-FCV-75-39, CORE SPRAY PUMP 2D SUPPR POOL SUCT VALVE • 2-FCV-75-51, CORE SPRAY SYS II OUTBD INJECT VALVE [2] **VERIFY CLOSED** 2-FCV-75-50, CORE SPRAY SYS II TEST VALVE. [3] **VERIFY** Core Spray Pump 2B and/or 2D running.
	BOP	Determines that 2B 480V Shutdown Board has tripped. Proceeds to use the appropriate EOI Appendix for injection with low pressure systems for the opposite loop.
	NRC	**End of Event 9. When the crew has inserted all Control Rods, Emergency Depressurized the Reactor, and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using low pressure systems, end of Scenario.**

Scenario Setup
UNIT 2

IC	38
Exam IC	278

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-47C	30	ARP-7A	35	APPX 18	10
OI-85	147	ARP-23B	31	TS 3.1.3	253
OI-92A	29	EOI-1	18	TS 3.1.5	253
AOI-85-3	26	EOI-2	16	TS 3.1.4	253
AOI-100-1	116	APPX-6B	12	SR 3.1.3.5(A)	25
ARP-3B	38	APPX-6C	12		
ARP-3F	40	APPX-6D	8		
ARP-5A	60	APPX-6E	8		

Simulator Setup	Verify camera system is powered down (admin password = abcd1234) Start CPERF **PRIOR** to placing the Simulator in RUN
Schedule Files(s):	2104 NRC Scenario 3 UNIT 2.sch
Event Files(s):	2104 NRC Scenario 3 UNIT 2.evt

Schedule File – 2104 NRC Scenario 3 ES-D-2 UNIT 2.sch

Event	Action	Description
	2104 NRC Scenario 3 UNIT 2.evt	EVENT FILE
	Insert override XS-92-7/42B to 7	CH B IRM RANGE
	Insert malfunction RD08R2227 to 95.00000 on event 4	CRD ACCUMULATOR LOW PRESSURE 22-27
	Insert malfunction PMP-66-31A to FAIL_CONTROL_POWER	42_CONTACTOR STACK DILUTION AIR FAN B
5	Insert override HS-66-29A to STOP	STACK DILUTION FAN 2A
15	Delete malfunction PMP-66-31A	42_CONTACTOR STACK DILUTION AIR FAN B

Schedule File – 2104 NRC Scenario 3 ES-D-2 UNIT 2.sch

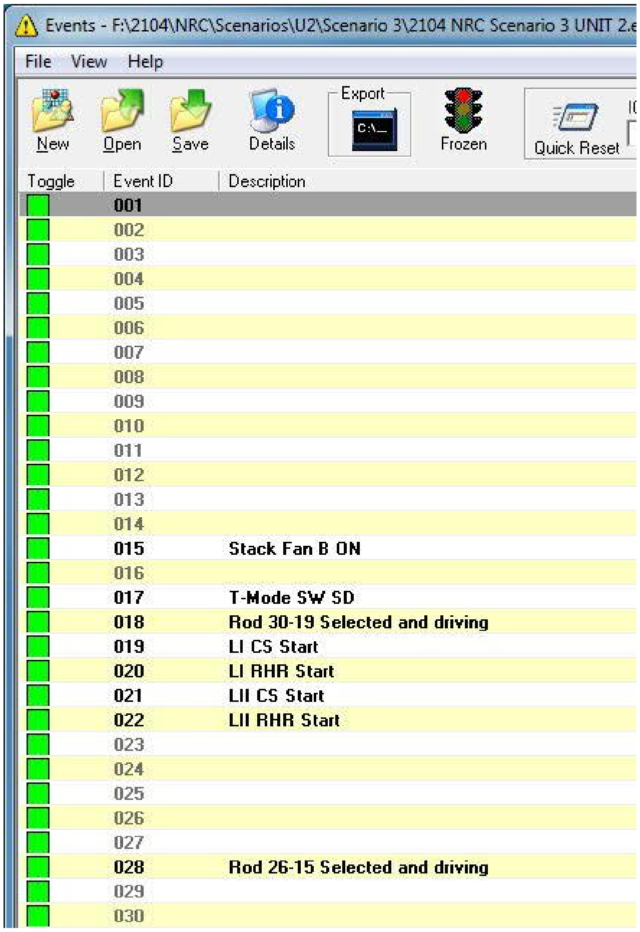
Event	Action	Description
	Insert override ZLOHS6631A_1 to On	HS-66-31A-Green* STACK DILUTION FAN 2B
	Insert override ZLOHS6631A_2 to Off	HS-66-31A-RED STACK DILUTION FAN 2B
15	Delete override ZLOHS6631A_1	HS-66-31A-Green* STACK DILUTION FAN 2B
15	Delete override ZLOHS6631A_2	HS-66-31A-RED STACK DILUTION FAN 2B
5	Insert override ZLOHS6629A_1 to Off	HS-66-29A-GREEN STACK DILUTION FAN 2A
6	Insert remote DG01B to OPEN	UNIT 1/2 DIESEL GENERATOR B LOGIC BREAKERS
	Insert override ZLOZI7434_1 to On	ZI-74-34 RHR PUMP B CST SUCTION VLV POSN
	Insert override ZLOZI7445_1 to On	ZI-74-45 RHR PUMP D CST SUCTION VLV POSN
	Insert override ZLOZI7411_1 to On	ZI-74-11 RHR PUMP A CST SUCTION VLV POSN
	Insert override ZLOZI7531_1 to On	ZI-75-31 CS PUMP B CST SUCTION VLV POSN
	Insert override ZLOZI7540_1 to On	ZI-75-40 CS PUMP D CST SUCTION VLV POSN
	Insert override ZLOZI7512_1 to On	ZI-75-12 CS PUMP CST SUCTION VLV POSN
	Insert override ZLOZI753_1 to On	ZI-75-3 CS PUMP A CST SUCTION VLV POSN
7	Insert remote RH07 to OPEN	RHR PUMP B CONDENSATE SUCTION VALVE HCV-74-34
7	Insert remote RH08 to OPEN	RHR PUMP D CONDENSATE SUCTION VALVE HCV-74-45
7	Insert remote RH05 after 180 to OPEN	RHR PUMP A CONDENSATE SUCTION VALVE HCV-74-11
7	Insert remote RH06 after 180 to OPEN	RHR PUMP C CONDENSATE SUCTION VALVE HCV-74-23
7	Insert remote CS06A to ALIGN	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 1

Schedule File – 2104 NRC Scenario 3 ES-D-2 UNIT 2.sch

Event	Action	Description
7	Insert remote CS06B to ALIGN	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 2
7	Insert remote FW11 to XCON	CROSS CONNECT CSTS
17	Insert remote CS06A to NORM	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 1
17	Insert remote CS06B to NORM	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 2
17	Insert remote CS06A after 120 to ALIGN	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 1
17	Insert remote CS06B after 120 to ALIGN	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 2
	Insert malfunction RD06R3019	STICK ANY CONTROL ROD 30-19
	Insert malfunction RD06R2615	STICK ANY CONTROL ROD 26-15
18	Delete malfunction RD06R3019	STICK ANY CONTROL ROD 30-19
28	Delete malfunction RD06R2615	STICK ANY CONTROL ROD 26-15
19	Insert malfunction ED10A after 3	480V SHUTDOWN BOARD 2A FAILURE
20	Insert malfunction ED10A after 3	480V SHUTDOWN BOARD 2A FAILURE
21	Insert malfunction ED10B after 3	480V SHUTDOWN BOARD 2B FAILURE
22	Insert malfunction ED10B after 3	480V SHUTDOWN BOARD 2B FAILURE
17	Insert override HS-3-75A after 30 to CLOSE	RFW FROM HTR A1 ISOL
17	Insert override HS-3-76A after 30 to CLOSE	RFW FROM HTR B1 ISOL
17	Insert override HS-3-77A after 30 to CLOSE	RFW FROM HTR C1 ISOL

Event File

List



Details

<table>
<tr><td colspan="3">UNIT 2 SHIFT TURNOVER MEETING</td><td>Today</td></tr>
<tr><td rowspan="2">MODE
2</td><td><u>DAYS ON LINE</u>
208</td><td rowspan="2"><u>Total Drywell Leakage (gpm)</u>
1.55</td><td><u>Protected Equipment</u></td></tr>
<tr><td rowspan="3">PRA (EOOS) -GREEN
500Kv GRID - Qualified
161Kv Grid -Qualified</td><td></td></tr>
<tr><td rowspan="2"><u>Rx Power</u>
~2%</td><td rowspan="2"><u>Floor Drain (gpm)</u>
0.11</td><td></td></tr>
<tr><td></td></tr>
<tr><td><u>MWe</u>
0</td><td><u>Last breaker closure</u>
N/A</td><td><u>Equipment Drain (gpm)</u>
1.44</td><td></td></tr>
</table>

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

IRM ‘G’ bypassed due to noise. Tech Spec 3.3.1.1 (Information only)

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

Reactor Startup

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Continue the Reactor Startup. Contact the OPS Superintendent prior to placing the MODE SWITCH in RUN.

Thunderstorm watch was just issued for counties in North Alabama.

Transfer Seal Steam from Auxiliary Steam to Main Steam in accordance with 2-OI-47C, Seal Steam System.

OPERATOR WORK AROUNDS **OWAs - 0** **Burdens - 0** **Challenges - 0**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 1 Page 1 of 2

Event Description: Transfer Seal Steam to Main Steam

Time	Position	Applicant's Actions or Behavior
	Driver	**Prior to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not proceed to Event 1, request that the Driver insert Event 1.**
	Driver	**If contacted by the Chief Examiner to insert Event 1, contact the Nuclear Unit Supervisor Operator (NUSO) as the Shift Manager and direct the crew to transfer Seal Steam from Auxiliary Steam to Main Steam.**
	NUSO	Directs the Balance of the Plant Operator (BOP) to transfer Seal Steam from Auxiliary Steam to Main Steam in accordance with 3-OI-47C, Seal Steam System, Section 6.1.
	BOP	3-OI-47C, Seal Steam System Section 6.1, Shifting Supply from Auxiliary Steam to Main Steam **NOTES** 1) Section 6.1 is entered with the Seal Steam supply on Auxiliary Steam. 2) Section 6.1 is performed at Panel 3-9-7 in the Control Room unless otherwise specified. 3) To seal the Turbine at startup with less than 250 psig Main Steam Pressure, or with worn packings, the 3-FCV-1-145, STEAM SEAL REGULATOR BYPASS VALVE, is required to be adjusted to supplement 3-PCV-1-147, STEAM SEAL FEED VALVE, to obtain the needed flow. 4) The 3-PCV-1-147, STEAM SEAL FEED VALVE, is designed to handle the Steam Seal Header requirements when Main Steam Pressure exceeds 250 psig. [1] **BEFORE** placing Seal Steam System on Nuclear Steam, **PERFORM** the following: [1.1] **NOTIFY** Radiation Protection that an RPHP is in effect for the impending action to place Seal Steam System on nuclear steam. **RECORD** time Radiation Protection notified in the Narrative Log.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 1 Page 2 of 2

Event Description: Transfer Seal Steam to Main Steam

Time	Position	Applicant's Actions or Behavior
	NRC	**An RPHP was provided to the crew at turnover.**
	BOP	[2] **CHECK** Reactor Pressure is greater than 200 psig. [3] **ENSURE** OPEN 3-PCV-1-147, STEAM TO STEAM SEAL PRESS REGULATOR by placing 3-HS-1-147, STEAM SEAL REGULATOR, in AUTO. [4] **OPEN** 3-FCV-1-146, MAIN STEAM SUPPLY TO STEAM SEAL VALVE. [5] **CLOSE** 3-FCV-1-154, AUX BOILER SUPPLY TO STEAM SEAL VALVE. [6] **CHECK** Steam Seal Header Pressure, as indicated on 3-PI-1-148A, STEAM SEAL HEADER PRESSURE, is between 2 1/2 and 9 psig. [7] **CLOSE** 3-SHV-012-0638, TURBINE SEAL STM VALVE. (Turbine Building Elevation 586', T16 J-Line near the EHC Unit behind Panel 25-111)
	Driver	**When directed as the Turbine Building AUO to close 3-12-638, TURBINE SEAL STM VALVE, acknowledge the direction and inform the crew that 3-12-638 is closed.**
	BOP	[8] **ENSURE** CLOSED 3-FCV-001-0149, STEAM SEAL UNLOADING MANUAL BYPASS VALVE. **CAUTION** Throttling 3-BYV-001-0145, STEAM SEAL REG BYPASS VALVE, with Main Steam Pressure above 250 psig could result in excessive vibration of the Steam Seal Header. [9] **THROTTLE** 3-FCV-1-145, STEAM SEAL REGULATOR BYPASS VALVE, to keep Steam Seal Header Pressure, as indicated on 3-PI-1-148A, STEAM SEAL HEADER PRESSURE, between 2 1/2 and 9 psig, if necessary. [10] **CHECK** SPE Vacuum, as indicated on 3-PI-66-54, STEAM PACKING EXHAUST VACUUM, is between 10 and 12 in H2O vacuum.
	NRC	**End of Event 1. Proceed to Event 2, Raise Reactor Power Using Control Rods.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 2 Page 1 of 4

Event Description: Raise Reactor Power using Control Rods

Time	Position	Applicant's Actions or Behavior
	NRC	**If the crew does not proceed to Control Rod withdrawal, request that the Driver insert Event 2.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to continue the Reactor Startup.**
	NRC	**During Control Rod withdrawal, Event 3, IRM Failure will automatically be inserted. No action is required by the driver to insert Event 3.**
	NUSO	(The crew may elect to conduct a reactivity re-focus brief) Assumes the Reactivity Manager position. Directs the Operator at the Controls (OATC) to raise Reactor Power to 6-7% in preparation for placing the Reactor MODE SWITCH in RUN, in accordance with 3-GOI-100-1A, Unit Startup and Power Operation, 3-OI-85, Control Rod Drive System, and 3-SR-3.1.3.5(A), Control Rod Coupling Integrity Check.
	OATC	3-GOI-100-1A, Unit Startup Section 5.4, Withdrawal of Control Rods while in Mode 2 [83] **CONTINUE** to withdraw Control Rods to raise Reactor Power to 6% to 7% per 3-OI-85, Control Rod Drive System, and 3-SR-3.1.3.5(A), Control Rod Coupling Integrity Check. [84] **ENSURE** all operable APRM downscale alarms are reset and no rod blocks exist.
	OATC	3-OI-85, Control Rod Drive System Section 6.6, Control Rod Withdrawal 6.6.1 Initial Conditions Prior to Withdrawing Control Rods [1] **REVIEW** Precautions and Limitations in Section 3.7 and Section 3.8. [2] **CHECK** the following prior to Control Rod movement: • CRD POWER, 3-HS-85-46 in ON. • Rod Worth Minimizer is operable and LATCHED into the correct ROD GROUP when Rod Worth Minimizer is enforcing (not required with no fuel in RPV)

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 2 Page 2 of 4

Event Description: Raise Reactor Power using Control Rods

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-85, Control Rod Drive System Section 6.6.2, Actions Required during and Following Control Rod Withdrawal [1] **IF** the Control Rod fails to withdraw, **THEN** Refer to Section 8.15 for additional methods to reposition Control Rod. [2] **IF** the Control Rod double notches, or withdraws past its correct/desired position, **THEN** Refer to Section 6.7 for inserting Control Rod to its correct/desired position. [NRC IR 84-02] [3] **IF** at any time while driving a selected rod during the performance of this section, the Control Rod moves more than one notch from its intended position, **THEN** Refer to 3-AOI-85-7, Mispositioned Control Rod. [4] **OBSERVE** the following during Control Rod repositioning: • Control Rod reed switch position indicators (four rod display) agree with the indication on the Full Core Display. • Nuclear Instrumentation responds as Control Rods move through the core (This ensures Control Rod is following drive during Control Rod movement.) [5] **ATTEMPT** to minimize automatic RBM Rod Block as follows: • **STOP** Control Rod withdrawal (if possible) prior to reaching any RBM Rod Block using the RBM displays on Panel 3-9-5 and PERFORM Step 6.6.2[6]. [6] **IF** Control Rod movement was stopped to keep from exceeding a RBM setpoint or was caused by a RBM Rod Block, **THEN PERFORM** the following at the Unit SRO's discretion to "REINITIALIZE" the RBM: [6.1] **PLACE** 3-HS-85-46, CRD POWER in the OFF position to deselect the Control Rod. [6.2] **PLACE** 3-HS-85-46, CRD POWER, in the ON position. [6.3] **IF** desired, **THEN CONTINUE** to withdrawal Control Rods and **PERFORM** applicable section for Control Rod withdrawal.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 2 Page 3 of 4

Event Description: Raise Reactor Power using Control Rods

Time	Position	Applicant's Actions or Behavior
	NRC	**Order of Control Rod withdrawal in accordance with 3-SR-3.1.3.5(A), Control Rod Coupling Integrity Check:**

BFN Unit 3	Control Rod Coupling Integrity Check	3-SR-3.1.3.5(A) Rev. 0027 Page 121 of 363

Attachment 5
(Page 20 of 39)
A2 Startup Sequence Control Rod Movement Data Sheet

Date ________

RWM GP	ROD NUMBER	FROM	TO	Rod Movement Completed Signoffs	
				UO (AC) [1]	Peer Check [2]
25	26-35	04	06		
25	34-35	04	06		
25	34-27	04	06		
25	26-27	04	06		
26	10-43	04	06		
26	18-51	04	06		
26	42-51	04	06		
26	50-43	04	06		
26	50-19	04	06		
26	42-11	04	06		
26	18-11	04	06		
26	10-19	04	06		
27	18-35	04	06		
27	26-43	04	06		
27	34-43	04	06		
27	42-35	04	06		
27	42-27	04	06		
27	34-19	04	06		
27	26-19	04	06		
27	18-27	04	06		

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 2 Page 4 of 4

Event Description: Raise Reactor Power using Control Rods

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-85, Control Rod Drive System Section 6.6.3, Control Rod Notch Withdrawal [1] **SELECT** the desired Control Rod by depressing the appropriate 3-XS-85-40, CRD ROD SELECT pushbutton. [2] **ENSURE** CRD DRIVE WATER HEADER DP is between 250 -270 psid on 3-PDI-85-17A by throttling 3-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE, as necessary. [3] N/A [4] **OBSERVE** the following for the selected Control Rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED • White light on the Full Core Display ILLUMINATED • Rod Out Permit light ILLUMINATED [5] **CHECK** Rod Worth Minimizer is operable and LATCHED into the correct ROD GROUP when the Rod Worth Minimizer is enforcing. [6] **PLACE** CRD CONTROL SWITCH, 3-HS-85-48, in ROD OUT NOTCH, and **RELEASE**. [7] **OBSERVE** the Control Rod settles into the desired position and the ROD SETTLE light extinguishes. [8] N/A [9] N/A
	OATC	3-OI-85, Control Rod Drive System 6.6.5 Return to Normal After Completion of Control Rod Withdrawal [1] **WHEN** Control Rod movement is no longer desired AND deselecting Control Rods is desired, **THEN**: [1.1] **PLACE** 3-HS-85-46, CRD POWER in OFF. [1.2] **PLACE** 3-HS-85-46, CRD POWER in ON.
	NRC	**End of Event 2. Event 3, IRM Failure is automatically inserted on simulator setup. No action is required by the Driver to insert Event 3.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 3 Page 1 of 2

Event Description: IRM Failure

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 3, IRM Failure, is automatically inserted on Simulator setup. No action is required by the Driver to insert Event 3.**
	NRC	**IRM 'G' will need to be ranged up after approximately 13 Control Rods are pulled (about Control Rod 42-11 in Group 25).**
	OATC	During Control Rod withdrawal will be ranging IRM Switches to prevent a Control Rod Block or Reactor SCRAM Signal. When the OATC attempts to range IRM 'B' UP, the reading will not change. Stops Control Rod withdrawal. Notifies the NUSO.
	NUSO	Directs the OATC to bypass IRM 'G' in accordance with 3-OI-92A, Intermediate Range Monitors.
	OATC	3-OI-92A, Intermediate Range Monitors Section 6.1, Bypassing an IRM Channel **NOTES** 1) It is not necessary for a bypassed IRM channel to have its detector inserted into the Core. 2) Only one IRM in each trip system can be bypassed at a time. 3) All operations are performed on Panel 3-9-5 unless specifically stated otherwise. **CAUTION** NPG-SPP 10.4 requires approval of the Plant Manager or his designee prior to any planned operation with IRMs bypassed unless bypassing is specifically allowed within approved procedures. [1] **REVIEW** all precautions and limitations in Section 3.0. [2] **PLACE** the appropriate IRM Bypass selector switch to the BYPASS position: • 3-HS-92-7A/S4A, IRM BYPASS • 3-HS-92-7A/S4B, IRM BYPASS [3] **CHECK** that the Bypassed light is illuminated.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 3 Page 2 of 2

Event Description: IRM Failure

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>Driver</td><td>If contacted by the crew as the Shift Manager, acknowledge any report given.
If contacted as the Plant Manager / Shift Manager for permission to continue Control Rod withdrawal with IRM 'B' bypassed, acknowledge the request and ask for their recommendation. Concur with any recommendation given.</td></tr>
<tr><td></td><td>OATC</td><td>Informs the NUSO that IRM 'G' is bypassed.</td></tr>
<tr><td></td><td>NUSO</td><td>References Technical Specification 3.3.1.1, RPS Instrumentation.
Declares an Information Only LCO based on only three IRM channels being required per trip system in accordance with Table 3.3.1.1-1 (page 1 of 3).

RPS Instrumentation
3.3.1.1

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation
<table>
<tr><th>FUNCTION</th><th>APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS</th><th>REQUIRED CHANNELS PER TRIP SYSTEM</th><th>CONDITIONS REFERENCED FROM REQUIRED ACTION D.1</th><th>SURVEILLANCE REQUIREMENTS</th><th>ALLOWABLE VALUE</th></tr>
<tr><td>1. Intermediate Range Monitors</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>a. Neutron Flux - High</td><td>2</td><td>3</td><td>G</td><td>SR 3.3.1.1.1
SR 3.3.1.1.3
SR 3.3.1.1.5
SR 3.3.1.1.6
SR 3.3.1.1.9
SR 3.3.1.1.14</td><td>≤ 120/125 divisions of full scale</td></tr>
<tr><td></td><td>5(a)</td><td>3</td><td>H</td><td>SR 3.3.1.1.1
SR 3.3.1.1.4
SR 3.3.1.1.9
SR 3.3.1.1.14</td><td>≤ 120/125 divisions of full scale</td></tr>
<tr><td>b. Inop</td><td>2</td><td>3</td><td>G</td><td>SR 3.3.1.1.3
SR 3.3.1.1.14</td><td>NA</td></tr>
<tr><td></td><td>5(a)</td><td>3</td><td>H</td><td>SR 3.3.1.1.4
SR 3.3.1.1.14</td><td>NA</td></tr>
</table></td></tr>
<tr><td></td><td>NRC</td><td>End of Event 3. Request that the driver insert Event 4, Control Rod Accumulator Inoperable.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 4 Page 1 of 4

Event Description: Control Rod Accumulator Inoperable

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5, Control Rod Accumulator Inoperable.**
	NRC	**The alarm will occur on Control Rod 22-27 Accumulator.**
	OATC	Acknowledges and reports the following alarm to the NUSO: • CRD ACCUMULATOR PRESSURE LOW / LEVEL HIGH, 3-9-5A, Window 29
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the appropriate Alarm Response Procedure.
	OATC	Alarm Response Procedure, 3-ARP-9-5A CRD ACCUMULATOR PRESSURE LOW / LEVEL HIGH, 3-9-5A, Window 29 Operator Action: A. **CHECK** alarm by amber background light illuminated on Full Core Display. B. **LOG** in the narrative log the Control Rod number and time alarm was received. **NOTE** If any of the following fuses is/are cleared, the local indications at Panel 25-4 and 25-22 will NOT illuminate. C. **IF** multiple accumulator lights are lit on Panel 3-9-5, **THEN CHECK** for cleared fuses 3-FU1-085-25-004G, -004H in Panel 25-4 and 3-FU1-085-25-022G, -022H in Panel 25-22. D. **DISPATCH** personnel to Panel 25-4 (east side), Panel 25-22 (west side) El 565', to determine if level high or pressure low. E. **DEPRESS** push-button for associated HCU to determine if alarm is caused by level high or pressure low as follows: 1. If alarm is due to high level, the red light will extinguish. 2. If light stays illuminated, alarm is due to low N2 pressure.

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 4	Page 2 of 4

Event Description: Control Rod Accumulator Inoperable

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Reactor Building AUO to respond to the CRD Accumulator alarm, acknowledge the direction. Wait two minutes and report to the crew that CRD 22-27 Accumulator Pressure is 900 psig and lowering. You are unable to raise Accumulator Nitrogen Pressure by recharging.**
	OATC	F. **IF** alarm is valid, **THEN REFER TO** 3-OI-85, Control Rod Drive System and 3-AOI-85-3, CRD System Failure.
	NRC	**The actions 3-OI-85, Control Rod Drive System and 3-AOI-85-3, CRD System Failure are covered starting on page xx of xx.**
	OATC	**NOTE** If accumulator pressure is greater than 940 psig the accumulator is not required to be declared Inoperable when the "star" valve is CLOSED, unless accumulator is unattended. G. **IF** Accumulator Pressure is less than or equal to 940 psig, **THEN DECLARE** Control Rod HCU "INOPERABLE". H. **IF** the associated HCU's nitrogen pressure is found less than 940 psi, **THEN INITIATE** a Condition Report (CR) to calibrate the pressure switch. The HCU will NOT be declared operable until the switch has been calibrated. I. **IF** alarm is due to low pressure with pressure greater than 940 psig and accumulator CANNOT be recharged within one hour, **THEN EVALUATE** per Tech Spec 3.1.5. 1. **IF** the Control Rod is declared SLOW **REFER TO** TECH SPEC 3.1.4. Currently no more than 13 OPERABLE Control Rods shall be slow and no more than 2 OPERABLE Control Rods that are slow shall occupy adjacent locations. 2. **IF** the Control Rod is declared INOPERABLE, **THEN REFER** TO TECH SPEC 3.1.3. J. **RECORD** this evaluation in narrative log. K. N/A

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: x Page x of x

Event Description: IRM Failure

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td colspan="2">If the Control Rod is declared INOPERABLE, references Tech Spec 3.1.3, Control Rod OPERABILITY.

LCO 3.1.3 Each Control Rod shall be OPERABLE

APPLICABILITY: MODES 1 and 2

NOTE: Separate Condition entry is allowed for each Control Rod.

CONDITION:
C. One or more Control Rods INOPERABLE for reasons other than Condition A or B.</td></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
NOTE: RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of INOPERABLE Control Rod and continued operation.

C.1 – Fully insert INOPERABLE Control Rod.
<u>AND</u>
C.2 – Disarm the associated CRD.</td><td>COMPLETION TIME:

C.1 – 3 hours

C.2 – 4 hours</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">If the Control Rod is declared SLOW, references Tech Spec 3.1.4, Control Rod SCRAM Times.
Currently, there are less than 13 OPERABLE Control Rods that are slow, and there are no OPERABLE Control Rods that are slow that occupy adjacent locations. Therefore, there are no actions required for Tech Spec 3.1.4, Control Rod SCRAM Times.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 4 Page 4 of 4

Event Description: Control Rod Accumulator Inoperable

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td colspan="2">Technical Specification 3.1.5, Control Rod SCRAM Accumulators

LCO 3.1.5: Each Control Rod SCRAM Accumulator shall be OPERABLE

APPLICABILITY: Modes 1 and 2

NOTE: Separate Condition entry is allowed for each Control Rod SCRAM Accumulator.

CONDITION:
A.1 One Control Rod SCRAM Accumulator inoperable with Reactor Steam Dome Pressure ≥ 900 psig.</td></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
A.1
NOTE: Only applicable if the associated Control Rod SCRAM time was within the limits of Table 3.1.4-1 during the last SCRAM time Surveillance.

Declare the associated Control Rod SCRAM Time “slow”.

<u>OR</u>

A.2- Declare the associated Control Rod INOPERABLE.</td><td>COMPLETION TIME:
A.1 – 8 hours

A.2 – 8 hours</td></tr>
<tr><td></td><td>NRC</td><td colspan="2">End of Event 4. Request that the driver insert Event 5, Failure of 3A Stack Dilution Fan, Standby Fan Fails to Automatically Start.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 5 Page 1 of 1

Event Description: Failure of 3A Stack Dilution Fan, Standby Fan Fails to Automatically Start

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4, Failure of 3A Stack Dilution Fan, Standby Fan Fails to Automatically Start.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • STACK GAS DILUTION AIR FLOW LOW, 3-9-7A, Window 3
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 3-ARP-9-7A STACK GAS DILUTION AIR FLOW LOW, Window 3 Operator Action: A. **CHECK** alternate fan **ON** and damper open, (red light illuminated) on Panel 3-9-7.
	BOP	Determines that the standby Stack Dilution Fan did not automatically start and manually starts 3B Stack Dilution Fan.
	BOP	B. **DISPATCH** personnel to stack to check and report status of the following for both fans: 1. Fan motor. 2. Fan belts. 3. Damper stuck closed. C. **CHECK** Breaker 5E on 480V Diesel Aux Bd A and B.
	Driver	**If contacted as an AUO, acknowledge any direction given.**
	NRC	**End of Event 5. Request that the Driver insert Event 6, 3A Emergency Diesel Generator (EDG) Logic Breaker Tripped.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: x Page 1 of 4

Event Description: 3EA Emergency Diesel Generator (EDG) Logic Breaker Tripped

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6, 3EA Emergency Diesel Generator (EDG) Logic Breaker Tripped.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • DIESEL GENERATOR 3A CONTROL POWER OFF, 3-9-23A, Window 14
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	Alarm Response Procedure, 3-ARP-9-23A DIESEL GENERATOR B CONTROL POWER OFF, Window 14 Operator Action: A. **OBSERVE** any other alarms on panels 9-8 or 9-23 which may indicate problem area. B. **CHECK** panels, breakers and batteries. IF necessary, **THEN CHECK** fuses and relays. C. **IF** loss of normal power has occurred, **THEN TRANSFER** to alternate power source. **REFER TO** 0-OI-57D DC Electrical System. D. **REFER TO** TS 3.8.1, 3.8.2, 3.8.4, and 3.8.5.
	Driver	**If contacted as the Outside NUSO, Work Control, AUO, or Electrical Maintenance, acknowledge any direction given.** **Wait 2 minutes and report that the Logic Breaker for EDG '3EA' is tripped. If directed to attempt to close the Logic Breaker, report that the breaker will not close.**
	NRC	**There are no required actions for Tech Spec 3.3.8.1, LOP Instrumentation or Tech Spec 3.8.7, Distribution Systems – Operating.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: x Page 2 of 4

Event Description: 3EA Emergency Diesel Generator (EDG) Logic Breaker Tripped

Time	Position	Applicant's Actions or Behavior	
	NUSO	References Technical Specification 3.8.4, DC Sources – Operating. LCO 3.8.4 The following DC electrical power sources shall be OPERABLE: a. Unit DC subsystems 1, 2, and 3; b. Shutdown Board DC subsystems 3EB; c. Unit 3 Diesel Generator (DG) DC subsystems; d. Unit 1 and 2 DG DC subsystem(s) supporting DG(s) required to be OPERABLE by LCO 3.8.1, "AC Sources - Operating"; and e. Unit 1 and 2 Shutdown Board DC subsystems needed to support equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," and LCO 3.7.3, "Control Room Emergency Ventilation (CREV) System." APPLICABILITY: MODES 1, 2, and 3 **CONDITION:** C. One or more DG DC electrical power subsystem(s) INOPERABLE	
	NUSO	**REQUIRED ACTION:** C.1 – One or more DG DC subsystem(s) inoperable.	**COMPLETION TIME:** C.1 – Immediately

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: x Page 3 of 4

Event Description: 3EA Emergency Diesel Generator (EDG) Logic Breaker Tripped

Time	Position	Applicant's Actions or Behavior	
	NUSO	Declares 3EA EDG INOPERABLE References Technical Specification 3.8.1, AC Sources – Operating. LCO 3.8.1 The following AC electrical power sources shall be OPERABLE: a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; b. Unit 3 diesel generators (DGs) with two divisions of 480 V load shed logic and common accident signal logic OPERABLE; and c. Unit 1 and 2 DG(s) capable of supplying the Unit 1 and 2 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating." APPLICABILITY: MODES 1, 2, and 3. **CONDITION:** One required Unit 3 DG INOPERABLE	
	NUSO	**REQUIRED ACTION:** B.1 – Verify power availability from the offsite transmission network. AND B.2 – Evaluate availability of both temporary diesel generators (TDGs). AND B.3. – Declare required feature(s), supported by the inoperable Unit 3 DG, inoperable when the redundant required feature(s) are inoperable.	**COMPLETION TIME:** B.1 – 1 hour AND Once per 8 hours thereafter B.2 – 1 hour AND Once per 12 hours thereafter B.3 – 4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: x Page 4 of 4

Event Description: 3EA Emergency Diesel Generator (EDG) Logic Breaker Tripped

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION: (continued)** AND B.4.1 Determine OPERABLE Unit 3 DG(s) are not inoperable due to common cause failure. OR B.4.2 – Perform SR 3.8.1.1 for OPERABLE Unit 3 DG(s). AND B.5 – Restore Unit 1 and 2 DG to OPERABLE status	**COMPLETION TIME:** B.4.1 – 24 hours B.4.2 – 24 hours B.5 – 7 days from discovery of unavailability of TDG(s) AND 24 hours from discovery of Condition B entry ≥ 6 days concurrent with unavailability of TDG(s) AND 14 days AND 21 days from discovery of failure to meet LCO
	NRC	**End of Event 6. Request that the driver insert Event 7, High Suppression Pool Water Level / Emergency Depressurization.**	

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 1 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7, High Suppression Pool Water Level / Emergency Depressurization.**
	NRC	**Event 8, Two Control Rods Fail to Insert and Event 9, 480V Shutdown Board Trip, will occur during Event 7 and are automatically entered by Simulator Setup. No action is required by the driver to insert Event 8 or Event 9.**
	BOP	Acknowledges and reports the following alarms as they are received: • DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL, 3-9-3B, Window 26 • SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, 3-9-3B, Window 15
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedures.
	BOP	Alarm Response Procedure, 3-ARP-9-3B DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL, Window 26 Operator Action: A. **CHECK** alarm by checking Drywell to Suppression Chamber DP. B. **REFER TO** 3-OI-64, Primary Containment System. C. **REFER TO** Tech Spec Section 3.6.2.6, Drywell-to-Suppression Chamber Differential Pressure.
	NRC	**Due to time constraints, Tech Spec evaluation for this event is not required and should not be used to evaluate the applicant's Tech Spec competency.**

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 7	Page 2 of 10

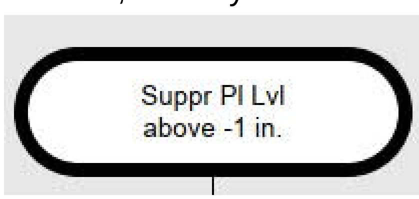
Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 3-ARP-9-3B SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, Window 15 Operator Action: A. **CHECK** level using multiple indications. B. **IF** level is low, **THEN DISPATCH** personnel to check for leaks. C. **IF** level is high, **THEN CHECK** for RCIC, HPCI, Core Spray, or RHR draining to Suppression Pool, and CHECK 3-TR-64-161, SUPPRESSION POOL WATER TEMPERATURE and 3-TR-64-162, SUPPRESSION POOL WATER TEMPERATURE. D. **REFER TO** 3-OI-74, Residual Heat Removal System, Sections 8.2, 8.3, and 8.4. E. **REFER TO** Tech Spec Section 3.6.2.2, Suppression Pool Water Level. F. **IF** level is above -1" or below -6.25" **AND NOT** in Mode 4 or Mode 5 **THEN** (otherwise N/A) **ENTER** 3-EOI-2, Primary Containment Control. G. **IF** level is above -1" or below -6.25" **AND** in Mode 4 or Mode 5 **THEN** (otherwise N/A) 1. **EVALUATE** plant conditions to **DETERMINE** if 3-EOI-2 entry is appropriate. 2. **RECORD** actions in NOMS log.
	BOP	Determines that the cause for the Drywell to Suppression Chamber DP alarm is rising Suppression Pool Water Level, and informs the NUSO.
	NUSO	Directs the BOP to monitor Suppression Pool Water Level and to provide an update when level reaches (-) 1 inch.
	BOP	Acknowledges and reports the following alarm to the NUSO when received: • SUPPRESSION POOL LEVEL HIGH, 3-9-3F, Window 12
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 3 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>Alarm Response Procedure, 3-ARP-9-3F
SUPPRESSION POOL LEVEL HIGH, Window 12

A. CHECK CST 3 and Suppression Pool level using multiple indications.
B. ENSURE HPCI Suction automatically transfers to the Suppression Pool.
C. IF automatic transfer fails, THEN REFER TO 3-OI-73, High Pressure Coolant Injection System.
D. REFER TO Tech Spec 3.5.1, ECCS - Operating, 3.5.2 and 3.6.2.2, Suppression Pool Water Level.</td></tr>
<tr><td></td><td>NRC</td><td>Due to time constraints, Tech Spec evaluation for this event is not required and should not be used to evaluate the applicant’s Tech Spec competency.</td></tr>
<tr><td></td><td>NUSO</td><td>When appropriate, enters 3-EOI-2, Primary Containment Control on high Suppression Pool Water Level (level above (-) 1 inch).</td></tr>
<tr><td></td><td>NUSO</td><td>3-EOI-2, Primary Containment Control

</td></tr>
<tr><td></td><td>NUSO</td><td>SP/L-1
<table>
<tr><td colspan="2">MONITOR and CONTROL Suppression Pool Water Level (-) 6 inches to (-) 1 inch.</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Suppression Pool Water Level CANNOT be maintained below (-) 1 inch.</td><td>A</td></tr>
<tr><td>Suppression Pool Water Level CANNOT be maintained above (-) 6 inches.</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 4 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	 SP/L-3 **MAINTAIN** Suppression Pool Water Level below 19 ft. (APPX 18, 20K) SP/L-4 **WHEN** Suppression Pool Level CANNOT be maintained below (APPX 9) 19 feet **STOP** DW Sprays
	NUSO	Directs the BOP to control Suppression Pool Water Level in accordance with 3-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup.
	Driver	**If contacted as an AUO to perform any steps locally per 3-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup, acknowledge any direction given.**
	BOP	3-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup [1] N/A [2] N/A [3] **IF** Directed by the NUSO, **THEN REMOVE** water from Suppression Pool as follows: [3.1] **DISPATCH** personnel to perform the following (Unit 3 RB, Elevation 519 ft, Torus Area): [3.1.1] **VERIFY OPEN** 3-SHV-074-0786A (B), RHR DRAIN PUMP A (B) DISCHARGE SHUTOFF VALVE.

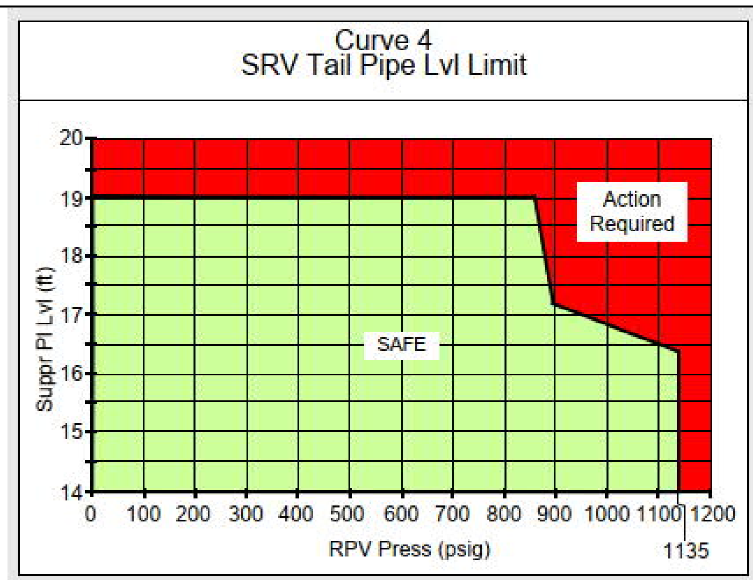
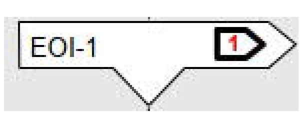
Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 5 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	BOP	[3.1.2] **OPEN** the following valves: • 3-SHV-074-0564A(B), RHR DRAIN PUMP A(B) SEAL WATER SUPPLY • 3-SHV-074-0529A (B), RHR DRAIN PUMP A (B) SHUTOFF VALVE [3.1.3] **UNLOCK** and **OPEN** 3-SHV-074-0765A (B), RHR DRAIN PUMP A(B) DISCHARGE. [3.1.4] **NOTIFY** Unit Operator that RHR Drain Pump 3A (3B) is lined up to remove water from Suppression Pool. [3.1.5] **REMAIN** at torus area UNTIL Unit 3 Operator directs starting of RHR Drain Pump 3A (3B). [3.2] **IF** Main Condenser is desired drain path, **THEN OPEN** 3-FCV-74-62, RHR MAIN CONDENSER FLUSH VALVE. [3.3] **IF** Radwaste is desired drain path, **THEN PERFORM** the following: [3.3.1] **ESTABLISH** communications with Radwaste [3.3.2] **OPEN** 3-FCV-74-63, RHR RADWASTE SYSTEM FLUSH VALVE. [3.4] **NOTIFY** personnel in Unit 3 RB, El 519 ft, Torus Area to start RHR Drain Pump 3A(3B). [3.5] **THROTTLE** 3-FCV-74-108, RHR DRAIN PUMP 3A/B DISCHARGE HEADER VALVE, as necessary.
	Driver	**After 2 minutes, report that the outside portions of 3-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup are complete. If directed to start the RHR Drain Pump, report that the RHR Drain Pump has been started.** **If contacted as the Rad Waste Operator, acknowledge any reports or direction given.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 6 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	 SP/L-6 **MAINTAIN** Suppression Pool Level within the safe area of Curve 4 (APPX 18, 20K)
	NUSO	
	NUSO	SP/L-7 **WHEN** Suppression Pool Level CANNOT be maintained within the safe area of Curve 4 (APPX 9) 

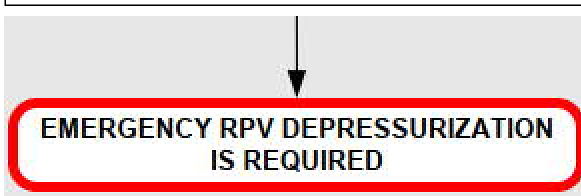
Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 7 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs the OATC to insert a manual Reactor SCRAM and directs the crew to enter 3-AOI-100-1, Reactor SCRAM.
	NRC	**Event 8, 2 Control Rods Fail to Insert, is automatically entered on Simulator Setup. No action is required by the driver to insert Event 8. See page xx of xx for Event 8 actions.**
	OATC	Inserts a manual Reactor SCRAM 3-AOI-100-1, Reactor SCRAM Immediate Actions [1] **DEPRESS** 3-HS-99-5A/S3A, REACTOR SCRAM A and 3-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 3-9-5. [2] **PLACE** REACTOR MODE SWITCH, 3-HS-99-5A/S1, in SHUTDOWN. [3] **IF** all Control Rods can NOT be verified fully inserted, **THEN INITIATE** ARI. (Otherwise MARK N/A).
	OATC	Determines that there are two (2) rods out.
	NRC	**When the Reactor MODE SWITCH is placed in SHUTDOWN, the Feedwater Heater Outlet Isolation Valves will close.** **See page xx of xx for actions for Event 8, Two Control Rods Fail to Insert and page xx of xx for actions for Event 9, 480V Shutdown Board Trip.**
	OATC	[4] **IF** Reactor Power is 5% or BELOW, **THEN** (Otherwise MARK N/A) **REPORT** the following to the UNIT SRO: • Reactor SCRAM • Mode Switch is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Power level

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 8 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	Following the Reactor SCRAM, enters 3-EOI-1A, ATWS RPV Control and directs the crew to perform the following: • Maintain Reactor Pressure to ensure that Suppression Pool Level is maintained within the safe area of Curve 4 in accordance with 3-EOI-Appendix-8B, Reopening MSIVs/Bypass Valve Operation • Maintain Reactor Water Level using in accordance with 3-EOI-Appendix-5D, Injection System Lineup HPCI or 3-EOI-Appendix-5C, Injection System Lineup RCIC • Insert Control Rods
	NUSO	(Continuing actions of 3-EOI-2, Primary Containment Control) SP/L-8 **WHEN** Suppression Pool Level and RPV Pressure CANNOT be maintained within the safe area of Curve 4 (APPX 9). SP/L-9 **STOP** injection into RPV from sources external to Primary Containment **EXCEPT** from systems required to assure Adequate Core Cooling or shut down the Reactor SP/L-10 **WHEN** Suppression Pool Level and RPV Pressure CANNOT be restored and maintained within the safe area of Curve 4 

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 7	Page 9 of 10
Event Description: High Suppression Pool Water Level / Emergency Depressurization			

Time	Position	Applicant's Actions or Behavior
	NUSO	Enters 2-C-2, Emergency RPV Depressurization C2-1 **IF:** RPV Water Level CANNOT be determined — **THEN:** **NO ACTION REQUIRED** **IF:** It is anticipated that RPV depressurization will result in loss of injection required for Adequate Core Cooling — **THEN:** **NO ACTION REQUIRED** **IF:** Containment Water Level CANNOT be maintained below 44 feet — **THEN:** **NO ACTION REQUIRED**
	NUSO	C2-2 **IF** Drywell Pressure is above 2.45 psig **THEN** **PREVENT** injection from ONLY those Core Spray and LPCI pumps NOT required to assure Adequate Core Cooling (APPX 4)

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 7 Page 10 of 10

Event Description: High Suppression Pool Water Level / Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>C2-3
EMERGENCY DEPRESSURIZE the RPV
IF Suppression Pool Water Level is above 5.5 feet
THEN OPEN 6 MSRVs (ADS Valves preferred)
• OK to exceed 100 F/hr Cooldown Rate
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Drywell Control Air becomes unavailable</td><td>NO ACTION REQUIRED</td></tr><tr><td>Less than 4 MSRVs can be opened
AND
RPV Pressure is 80 psi or more above Suppression Chamber Pressure</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NRC</td><td>The first loop of Low Pressure Injection (Core Spray/RHR) that the crew attempts to use will result in a loss of the 480V Shutdown Board for that loop. See Event 9 on page xx of xx.</td></tr>
<tr><td></td><td>NRC</td><td>End of Event 7. When the crew has inserted all Control Rods, Emergency Depressurized the Reactor, and has control of Reactor Water level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.</td></tr>
</table>

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 8	Page 1 of 3

Event Description: Two Control Rods Fail to Insert

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8 is automatically entered by simulator setup. No action is required by the Driver to insert Event 8.**
	OATC	Following the Reactor SCRAM and after initiating Alternate Rod Insertion (ARI), determines that all Control Rods are not in. Informs the NUSO that two rods are out, and that Reactor Power is less than 5%.
	NUSO	Directs the OATC to insert Control Rods in accordance with 3-AOI-100-1, Reactor SCRAM.
	NRC	**Not all Subsequent Actions of 3-AOI-100-1, Reactor SCRAM, are listed below.**
	OATC	3-AOI-100-1, Reactor SCRAM [16] **IF** all rods are **NOT** inserted to Position 02 or beyond, **THEN DIRECT** Reactor Engineer to commence determination that the Reactor will remain subcritical under all conditions without boron.
	Driver	**If contacted as the Reactor Engineer, acknowledge any direction or report given.**
	OATC	[17] **IF** any Control Rod fails to fully insert and it is required to Re-SCRAM, **THEN PERFORM** the following, as required. (Otherwise N/A) [17.1] **RESET** the SCRAM per Steps 4.2[24] thru 4.2[24.12]. [17.2] **CHECK** WEST and EAST CRD DISCH VOL WTR LVL HIGH HALF SCRAM annunciators (3-XA-55-4A, Window 1 and 3-XA-55-4A, Window 29) are reset. [17.3] **INITIATE** a manual SCRAM. [17.4] **REPEAT** Step 4.2[17], as necessary, as long as rod motion is observed. [18] **IF** any Control Rod fails to fully insert and it is required to Drive Control Rods, **THEN REFER** TO 3-OI-85, Control Rod Drive System.
	OATC	3-OI-85, Control Rod Drive System Section 6.7, Control Rod Insertion [1] **REVIEW** Precautions and Limitations in Sections 3.7 and 3.8.

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 8	Page 2 of 3
Event Description: Two Control Rods Fail to Insert			

Time	Position	Applicant's Actions or Behavior
	OATC	[2] **ENSURE** the following prior to Control Rod movement: • 3-HS-85-46, CRD POWER in ON • ROD WORTH MINIMIZER is operable and LATCHED in to the correct ROD GROUP, when Rod Worth Minimizer is enforcing [3] **OBSERVE** the following during Control Rod repositioning: • Control Rod reed switch position indicators (four rod display) agree with the indication on the Full Core Display. • Nuclear Instrumentation responds as Control Rods move through the core (This ensures Control Rod is following drive during Control Rod movement.) [4] **PERFORM** the following to insert the Control Rod as appropriate. • Control Rod Notch Insertion per Section 6.7.2 • Control Rod Continuous Insertion per Section 6.7.3
	OATC	3-OI-85, Control Rod Drive System Section 6.7.3, Continuous Insertion of Control Rod [1] **CHECK** Section 6.7.1 has been performed. [2] **SELECT** the desired Control Rod by depressing the appropriate 3-XS-85-40, CRD ROD SELECT pushbutton. [3] **OBSERVE** the following for the selected Control Rod: CRD ROD SELECT pushbutton is brightly ILLUMINATED White light on the Full Core Display ILLUMINATED [4] **PLACE** and HOLD CRD CONTROL SWITCH, 3-HS-85-48, in ROD IN. [5] **WHEN** Control Rod notch reaches the even rod notch position prior to the desired final Control Rod notch position, THEN RELEASE 3-HS-85-48, CRD CONTROL SWITCH.
	NRC	**When the OATC has selected and each stuck Control Rod and begins to drive the rod in, the malfunction will clear to allow rod insertion. As a result, the Control Rod will drive in at a much faster rate than normal.**

Op Test No.: 21-04	Scenario No. NRC-3	Event No.: 8	Page 3 of 3

Event Description: Two Control Rods Fail to Insert

Time	Position	Applicant's Actions or Behavior
	OATC	[6] **OBSERVE** the Control Rod settles into desired position and the ROD SETTLE light extinguishes. [7] **IF** the Control Rod double notches or inserts past its correct/desired position, **THEN** with Unit SROs permission return the Control Rod to the intended position per Section 6.6. (Otherwise N/A) [8] **IF** the Control Rod moves more than one notch from its intended position, **THEN PERFORM** 3-AOI-85-7, Mispositioned Control Rod. (Otherwise N/A) [9] **WHEN** Control Rod movement is no longer desired AND deselecting Control Rods is desired, **THEN**: [9.1] PLACE 3-HS-85-46, CRD POWER in OFF. [9.2] PLACE 3-HS-85-46, CRD POWER in ON.
	NRC	**End of Event 8. When the crew has inserted all Control Rods, Emergency Depressurized the Reactor, and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using low pressure systems, end of Scenario.**

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 9 Page 1 of x

Event Description: 480V Shutdown Board Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9, 489V Shutdown Board Trip, is automatically entered by Simulator Setup. No action is required by the driver to insert Event 9.** **The first loop of Low Pressure Injection (Core Spray/RHR) that the crew attempts to use will result in a loss of the 480V Shutdown Board for that loop.**
	NUSO	During Emergency Depressurization, directs the BOP to maintain Reactor Water Level using Core Spray or RHR in accordance with any of the following EOI Appendices: • 3-EOI-Appendix-6B, Injection Subsystems Lineup – RHR System I LPCI Mode (see below) • 3-EOI-Appendix-6C, Injection Subsystems Lineup – RHR System II LPCI Mode (see page xx of xx) • 3-EOI-Appendix-6D, Injection Subsystems Lineup –Core Spray System (see page xx of xx) • 3-EOI-Appendix-6E, Injection Subsystems Lineup – Core Spray System II (see page xx of xx)
	NRC	**If the crew selects Loop II of RHR or Core Spray to maintain Reactor Water Level, proceed to page xx of xx for the procedure(s) for injection. If the crew selects Loop I of RHR see below for the procedure(s) for injection.**
	BOP	**IF USING LOOP I OF RHR FOR INJECTION:** 3-EOI-Appendix-6B, Injection Subsystems Lineup RHR System I LPCI Mode [1] **IF** Adequate core cooling is assured **AND** it becomes necessary to bypass LPCI Injection Valve auto open signal to control injection, **THEN PLACE** 3-HS-74-155A, LPCI SYSTEM I OUTBD INJECTION VALVE BYPASS SELECT, in BYPASS. [2] **ENSURE OPEN** 3-FCV-74-7, RHR SYSTEM I MINIMUM FLOW VALVE.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 9 Page x of x

Event Description: 480V Shutdown Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **ENSURE OPEN** the following valves: • 3-FCV-74-1, RHR PUMP 3A SUPPRESSION POOL SUCTION VALVE • 3-FCV-74-12, RHR PUMP 3C SUPPRESSION POOL SUCTION VALVE [4] **ENSURE CLOSED** the following valves: • 3-FCV-74-61, RHR SYSTEM I DRYWELL SPRAY INBOARD VALVE • 3-FCV-74-60, RHR SYSTEM I DRYWELL SPRAY OUTBOARD VALVE • 3-FCV-74-57, RHR SYSTEM I SUPPRESSION CHAMBER/POOL ISOLATION VALVE • 3-FCV-74-58, RHR SYSTEM I SUPPRESSION CHAMBER SPRAY VALVE • 3-FCV-74-59, RHR SYSTEM I SUPPRESSION POOL COOLING/TEST VALVE [5] **ENSURE** RHR Pump 3A and / or 3C running
	CREW	Determines that 3A 480V Shutdown Board has tripped. Proceeds to use the appropriate EOI Appendix for injection with low pressure systems for the opposite loop.
	BOP	**IF USING LOOP I OF CORE SPRAY FOR INJECTION:** 3-EOI-Appendix-6D, Injection Subsystems Lineup Core Spray System II [1] **VERIFY OPEN** the following valves: • 3-FCV-75-2, CORE SPRAY PUMP 3A SUPPRESSION POOL SUCTION VALVE • 3-FCV-75-11, CORE SPRAY PUMP 3C SUPPRESSION POOL SUCTION VALVE • 3-FCV-75-23, CORE SPRAY SYS I OUTBD INJECTION VALVE [2] **VERIFY CLOSED** 3-FCV-75-22, CORE SPRAY SYS I TEST VALVE. [3] **VERIFY** Core Spray Pump 3A and/or 3C running.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 9 Page x of x

Event Description: 480V Shutdown Board Trip

Time	Position	Applicant's Actions or Behavior
	CREW	Determines that 3A 480V Shutdown Board has tripped. Proceeds to use the appropriate EOI Appendix for injection with low pressure systems for the opposite loop.
	BOP	**IF USING LOOP II OF RHR FOR INJECTION:** 3-EOI-Appendix-6C, Injection Subsystems Lineup RHR System II LPCI Mode [1] **IF** Adequate Core Cooling is assured **AND**, it becomes necessary to bypass LPCI Injection Valve auto open signal to control injection, **THEN PLACE** 3-HS-74-155B, LPCI SYS II OUTBOARD INJECTION VALVE BYPASS SELECT IN BYPASS. [2] **ENSURE OPEN** 3-FCV-74-30, RHR SYSTEM II MINIMUM FLOW VALVE. [3] **ENSURE OPEN** the following valves: • 3-FCV-74-24, RHR PUMP 3B SUPPRESSION POOL SUCTION VALVE. • 3-FCV-74-35, RHR PUMP 3D SUPPRESSION POOL SUCTION VALVE. [4] **ENSURE CLOSED** the following valves: • 3-FCV-74-75, RHR SYSTEM II DRYWELL SPRAY INBOARD VALVE • 3-FCV-74-74, RHR SYSTEM II DRYWELL SPRAY OUTBOARD VALVE • 3-FCV-74-71, RHR SYSTEM II SUPPR CHAMBER/POOL ISOLATION VALVE • 3-FCV-74-72, RHR SYSTEM II SUPPR CHAMBER SPRAY VALVE • 3-FCV-74-73, RHR SYSTEM II SUPPR POOL COOLING/TEST VALVE [5] **ENSURE** RHR Pump 3B and/or 3D running.

Op Test No.: 21-04 Scenario No. NRC-3 Event No.: 9 Page x of x

Event Description: 480V Shutdown Board Trip

Time	Position	Applicant's Actions or Behavior
	CREW	Determines that 3B 480V Shutdown Board has tripped. Proceeds to use the appropriate EOI Appendix for injection with low pressure systems for the opposite loop.
	BOP	**IF USING LOOP II OF CORE SPRAY FOR INJECTION:** 3-EOI-Appendix-6E, Injection Subsystems Lineup Core Spray System II [1] **VERIFY OPEN** the following valves: • 3-FCV-75-30, CORE SPRAY PUMP 3B SUPPRESSION POOL SUCTION VALVE • 3-FCV-75-39, CORE SPRAY PUMP 3D SUPPRESSION POOL SUCTION VALVE • 3-FCV-75-51, CORE SPRAY SYS II OUTBD INJECTION VALVE [2] **VERIFY CLOSED** 3-FCV-75-50, CORE SPRAY SYSTEM II TEST VALVE. [3] **VERIFY** CS Pump 3B and/or 3D running.
	CREW	Determines that 3B 480V Shutdown Board has tripped. Proceeds to use the appropriate EOI Appendix for injection with low pressure systems for the opposite loop.
	NRC	**End of Event 9. When the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using low pressure systems, end of Scenario.**

Scenario Setup
UNIT 3

IC	38
Exam IC	263

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-47C	23	ARP-7A	30	APPX 18	3
OI-85	91	ARP-23B	29	TS 3.1.3	212
OI-92A	18	EOI-1	13	TS 3.1.5	212
AOI-85-3	13	EOI-2	13	TS 3.1.4	212
AOI-100-1	74	APPX-6B	7	SR 3.1.3.5(A)	27
ARP-3B	23	APPX-6C	8		
ARP-3F	36	APPX-6D	4		
ARP-5A	54	APPX-6E	4		

Simulator Setup	Verify camera system is powered down (admin password = abcd1234) Start CPERF **PRIOR** to placing the Simulator in RUN
Schedule Files(s):	2104 NRC Scenario 3 UNIT 3.sch
Event Files(s):	2104 NRC Scenario 3 UNIT 3.evt

Schedule File – 2104 NRC Scenario 3 ES-D-2 UNIT 3.sch

Event	Action	Description
	2104 NRC Scenario 3 UNIT 3.evt	EVENT FILE
	Insert override XS-92-7/42G to 8	CHANNEL G IRM RANGE SWITCH
4	Insert malfunction RD08R2227 to 95.00000	CRD ACCUMULATOR LOW PRESSURE 22-27
	Insert malfunction PMP-66-31A to FAIL_CONTROL_POWER	42_CONTACTOR STACK DILUTION AIR FAN B
5	Insert override HS-66-29A to STOP	STACK DILUTION FAN 3A
15	Delete malfunction PMP-66-31A	42_CONTACTOR STACK DILUTION AIR FAN B
	Insert override ZLOHS6631A_1 to On	HS-66-31A-Green* STACK DILUTION FAN 3B

Schedule File – 2104 NRC Scenario 3 ES-D-2 UNIT 2.sch

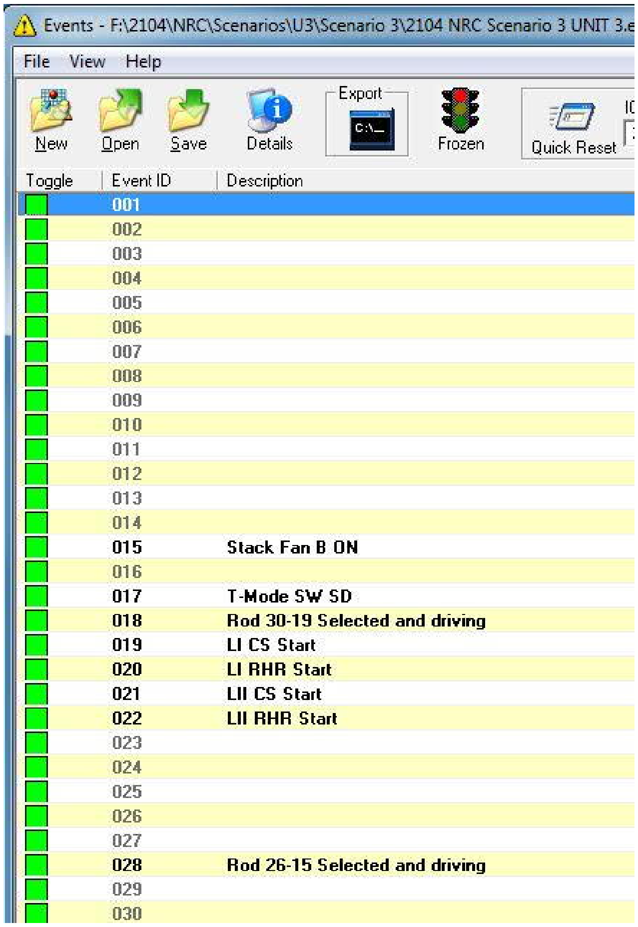
Event	Action	Description
	Insert override ZLOHS6631A_2 to Off	HS-66-31A-RED STACK DILUTION FAN 3B
15	Delete override ZLOHS6631A_1	HS-66-31A-Green* STACK DILUTION FAN 3B
15	Delete override ZLOHS6631A_2	HS-66-31A-RED STACK DILUTION FAN 3B
5	Insert override ZLOHS6629A_1 to Off	HS-66-29A-GREEN STACK DILUTION FAN 3A
6	Insert remote DG01A to OPEN	UNIT 3 DIESEL GENERATOR 3A LOGIC BREAKER
	Insert override ZLOZI7434_1 to On	ZI-74-34-Green* RHR PUMP 3B CST SUCTION VALVE
	Insert override ZLOZI7445_1 to On	ZI-74-45-Green* RHR PUMP 3D CST SUCTION VALVE
	Insert override ZLOZI7411_1 to On	ZI-74-11-Green* RHR PUMP 3A CST SUCTION VLV
	Insert override ZLOZI7531_1 to On	ZI-75-31-Green* CORE SPRAY PUMP 3B CST SUCTION VLV
	Insert override ZLOZI7540_1 to On	ZI-75-40-Green* CORE SPRAY PUMP 3D CST SUCTION VLV
	Insert override ZLOZI7512_1 to On	ZI-75-12-Green* CORE SPRAY PUMP 3C CST SUCTION VALVE
	Insert override ZLOZI753_1 to On	ZI-75-3-Green* CORE SPRAY PUMP 3A CST SUCTION VALVE
7	Insert remote RH07 to OPEN	RHR PUMP B CONDENSATE SUCTION VALVE HCV-74-34
7	Insert remote RH08 to OPEN	RHR PUMP D CONDENSATE SUCTION VALVE HCV-74-45
7	Insert remote RH05 after 180 to OPEN	RHR PUMP A CONDENSATE SUCTION VALVE HCV-74-11
7	Insert remote RH06 after 180 to OPEN	RHR PUMP C CONDENSATE SUCTION VALVE HCV-74-23
7	Insert remote CS06A to ALIGN	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 1

Schedule File – 2104 NRC Scenario 3 ES-D-2 UNIT 2.sch

Event	Action	Description
7	Insert remote CS06B to ALIGN	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 2
7	Insert remote FW11 to XCON	CROSS CONNECT CSTS
17	Insert remote CS06A to NORM	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 1
17	Insert remote CS06B to NORM	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 2
17	Insert remote CS06A after 120 to ALIGN	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 1
17	Insert remote CS06B after 120 to ALIGN	ALIGN CONDENSATE STORAGE TANK TO CORE SPRAY LOOP 2
	Insert malfunction RD06R3019	STICK ANY CONTROL ROD 30-19
	Insert malfunction RD06R2615	STICK ANY CONTROL ROD 26-15
18	Delete malfunction RD06R3019	STICK ANY CONTROL ROD 30-19
28	Delete malfunction RD06R2615	STICK ANY CONTROL ROD 26-15
19	Insert malfunction ED10A after 3	480V SHUTDOWN BOARD 3A FAILURE
20	Insert malfunction ED10A after 3	480V SHUTDOWN BOARD 3A FAILURE
21	Insert malfunction ED10B after 3	480V SHUTDOWN BOARD 3B FAILURE
22	Insert malfunction ED10B after 3	480V SHUTDOWN BOARD 3B FAILURE
17	Insert override HS-3-75A after 30 to CLOSE	HP HTR 3A1 FW OUTLET ISOL VLV
17	Insert override HS-3-76A after 30 to CLOSE	HP HTR 3B1 FW OUTLET ISOL VLV
17	Insert override HS-3-77A after 30 to CLOSE	HP HTR 3C1 FW OUTLET ISOL VLV

Event File

List



Details

UNIT 3 SHIFT TURNOVER MEETING

UNIT 3 SHIFT TURNOVER MEETING			Today
MODE 1	**DAYS ON LINE** 234	**Drywell Leakage (GPM)** 1.89	**Protected Equipment**
	PRA (EOOS) -Green		
Rx Power 2%	500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (GPM)** 0.31	
MWe 0	**Last breaker closure** N/A	**Equipment Drain (GPM)** 1.58	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

IRM ‘H’ bypassed due to noise.

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

Reactor Startup

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Continue the Reactor Startup. Contact the Ops Superintendent prior to placing the MODE SWITCH in RUN.

Thunderstorm watch was just issued for counties in North Alabama.

Transfer Seal Steam from Auxiliary Steam to Main Steam in accordance with 3-OI-47C, Seal Steam System.

OPERATOR WORK AROUNDS **OWAs - 0** **Burdens - 0** **Challenges - 0**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Facility: BFN Scenario Number: NRC-4 Op-Test Number:21-04

Examiners: ______________________ Operators: SRO: ______________________

______________________ ATC: ______________________

______________________ BOP: ______________________

Initial Conditions: 95% Reactor Power.

Turnover: Core Spray Loop I outage with MOVATs testing in progress. Remove 2C Condensate Booster Pump (CBP) from service for maintenance. APRM 1 is bypassed due to a critical fault.

Critical Tasks:

1. Start the Standby Stator Cooling Water Pump before the Turbine Trip Timer times out and trips the Main Turbine, resulting in a Reactor SCRAM.
2. Initiate Drywell Sprays while in the safe area of the Drywell Spray Initiation Limit Curve and before Drywell Temperature rises to 280 °F.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Remove 2C Condensate Booster Pump (CBP) from Service for Maintenance
2. #S	RD22	C-OATC C-NUSO	Control Rod Drive (CRD) Flow Controller Fails High
3.	FW33C FW33D	C-OATC C-NUSO	Reactor Feedwater Pump (RFPT) Vibration Alarm
4.	N/A	R-OATC R-NUSO	Power Reduction for RFPT Shutdown
5.	RM08A RM08B	TS-NUSO	Refuel Zone Radiation Monitors Fail Upscale
6.#S	PC01C	C-BOP TS-NUSO	Standby Gas Train ‘C’ Fails to Auto Start
7.#	HS-35-35A PMP-35-36	C-BOP C-NUSO	2A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start
8.#	TH33A	M-ALL	Steam Leak in the Drywell
9.	XS-74-121 XS-74-129	C-BOP C-NUSO	Drywell Spray Failure

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

Event on previous two NRC Exams

#S Event on previous two NRC Exams Spare Scenario

Events

1. The crew will remove 2C Condensate Booster Pump from service in accordance with 2-OI-2, Condensate System, Section 8.19.

2. The CRD Flow Controller will fail high, requiring action by the crew to take manual control of the CRD Flow Controller in accordance with OPDP-1, Conduct of Operations, 2-AOI-85-3, CRD System Failure, and 2-OI-85, Control Rod Drive System to restore system flow.

3. A vibration alarm on 2C Reactor Feedwater Pump (RFPT) will be received, and the crew will respond in accordance with Alarm Response Procedures and 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low. The vibration readings on 2C RFPT will force the crew to remove it from service.

4. As required by 2-OI-3, Reactor Feedwater System, the crew will reduce Reactor Power in order to remove 2C RFPT from service.

5. Refueling Zone Radiation Monitors 2-RE-90-140A, CH 0A REFUEL ZONE DET A, and 2-RE-90-140B, CH0B REFUEL ZONE DET B, will fail high resulting in a Group 6 Primary Containment Isolation. The Nuclear Unit Senior Operator (NUSO) will reference Technical Specifications 3.3.6.2, Secondary Containment Isolation Instrumentation, Condition A; 3.3.7.1, CREV System Instrumentation, and Technical Requirements Manual, Condition A; 3.3.10, Reactor Coolant Leakage Detection Instrumentation, Condition A.

6. When the Refueling Zone Radiation Monitoring Channels fail, ‘C’ Standby Gas Train (SGT) will fail to automatically start, requiring the crew to manually start ‘C’ SGT. The NUSO will address Technical Specification 3.6.4.3, Standby Gas Treatment System, Condition A.

7. 2A Stator Cooling Water (SCW) Pump will trip and the Standby Pump will not automatically start. The crew will respond in accordance with Alarm Response Procedure (ARP) GEN STATOR COOLANT SYS ABNORMAL, 2-ARP-9-7A, Window 22, and start the Standby SCW Pump in order to prevent the Turbine Trip Timer from timing out and tripping the Main Turbine.

8. An un-isolable Steam Leak will develop in the Drywell and gradually worsen, resulting in rising Drywell Temperature and Pressure. The crew will respond in accordance with 2-AOI-64-1, Drywell Pressure and/or Temperature High or Excessive Leakage into the Drywell, and will be forced to insert a manual Reactor SCRAM. The NUSO will respond in accordance with 2-EOI-2, Secondary Containment Control.

9. The first loop of Drywell Sprays will fail when the crew attempts to spray Containment to reduce Containment Pressure and Temperature, requiring action to use the other loop of Drywell Spray.

The Scenario ends when the crew has sprayed Containment and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using either high or low pressure systems.

Critical Tasks 2

1. **Start the Standby Stator Cooling Water Pump before the Turbine Trip Timer times out and trips the Main Turbine, resulting in a Reactor SCRAM.**

   a. **Safety Significance**
   Prevents an unnecessary Reactor SCRAM.

   b. **Cues**
   Turbine Trip Timer Initiated Alarm.
   Standby Stator Cooling Water Pump fails to auto start.

   c. **Measured by:**
   Observation – the crew takes action to start the standby Stator Cooling Water Pump.

   d. **Feedback**
   The standby Stator Cooling Water Pump starts.
   The Turbine Trip Timer Initiated alarm clears.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to start the standby Stator Cooling Water Pump and an automatic Reactor SCRAM occurs after the Turbine Trip Timer times out.

2. **Initiate Drywell Sprays while in the safe area of the Drywell Spray Initiation Limit Curve and before Drywell Temperature rises to 280 °F.**

   a. **Safety Significance:**
   Precludes failure of containment.

   b. **Cues:**
   Procedural compliance.
   High Drywell Pressure and Suppression Chamber Pressure.

   c. **Measured by:**
   Observation - NUSO directs Drywell Sprays in accordance with 2-EOI-Appendix-17B, RHR System Operation Drywell Sprays.
   AND
   Observation - RO initiates Drywell Sprays.

   d. **Feedback:**
   Drywell and Suppression Pressure lowering.
   RHR flow to containment.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to spray the Drywell before Drywell Temperature reaches 280 ºF.

Facility: BFN Scenario Number: NRC-4 Op-Test Number: 21-04

Examiners: ______________________ Operators: SRO: ______________________

______________________ ATC: ______________________

______________________ BOP: ______________________

Initial Conditions: 95% Reactor Power.

Turnover: Core Spray Loop I outage with MOVATs testing in progress. Remove 3C Condensate Booster Pump (CBP) from service for maintenance. APRM 1 is bypassed due to a critical fault.

Critical Tasks:

1. Start the Standby Stator Cooling Water Pump before the Turbine Trip Timer times out and trips the Main Turbine, resulting in a Reactor SCRAM.
2. Initiate Drywell Sprays while in the safe area of the Drywell Spray Initiation Limit Curve and before Drywell Temperature rises to 280 °F.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Remove 3C Condensate Booster Pump (CBP) from Service for Maintenance
2. #S	RD22	C-OATC C-NUSO	Control Rod Drive (CRD) Flow Controller Fails High
3.	FW33C FW33D	C-OATC C-NUSO	Reactor Feedwater Pump (RFPT) Vibration Alarm
4.	N/A	R-OATC R-NUSO	Power Reduction for RFPT Shutdown
5.	RM08A RM08B	TS-NUSO	Refuel Zone Radiation Monitors Fail Upscale
6.#S	PC01C	C-BOP TS-NUSO	Standby Gas Train ‘C’ Fails to Auto Start
7.#	HS-35-35A PMP-35-36	C-BOP C-NUSO	3A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start
8.#	TH33A	M-ALL	Steam Leak in the Drywell
9.	XS-74-121 XS-74-129	C-BOP C-NUSO	Drywell Spray Failure

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

Event on previous two NRC Exams

#S Event on previous two NRC Exams Spare Scenario

Events

1. The crew will remove 3C Condensate Booster Pump from service in accordance with 3-OI-2, Condensate System, Section 8.19.

2. The CRD Flow Controller will fail high, requiring action by the crew to take manual control of the CRD Flow Controller in accordance with OPDP-1, Conduct of Operations, 3-AOI-85-3, CRD System Failure, and 3-OI-85, Control Rod Drive System to restore system flow.

3. A vibration alarm on 3C Reactor Feedwater Pump (RFPT) will be received, and the crew will respond in accordance with Alarm Response Procedures and 3-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low. The vibration readings on 3C RFPT will force the crew to remove it from service.

4. As required by 3-OI-3, Reactor Feedwater System, the crew will reduce Reactor Power in order to remove 3C RFPT from service.

5. Refueling Zone Radiation Monitors 3-RE-90-140A, CH 0A REFUEL ZONE DET A, and 3-RE-90-140B, CH0B REFUEL ZONE DET B, will fail high resulting in a Group 6 Primary Containment Isolation. The Nuclear Unit Senior Operator (NUSO) will reference Technical Specifications 3.3.6.2, Secondary Containment Isolation Instrumentation, Condition A; 3.3.7.1, CREV System Instrumentation, and Technical Requirements Manual, Condition A; 3.3.10, Reactor Coolant Leakage Detection Instrumentation, Condition A.

6. When the Refueling Zone Radiation Monitoring Channels fail, ‘C’ Standby Gas Train (SGT) will fail to automatically start, requiring the crew to manually start ‘C’ SGT. The NUSO will address Technical Specification 3.6.4.3, Standby Gas Treatment System, Condition A.

7. 3A Stator Cooling Water (SCW) Pump will trip and the Standby Pump will not automatically start. The crew will respond in accordance with Alarm Response Procedure (ARP) GEN STATOR COOLANT SYS ABNORMAL, 3-ARP-9-7A, Window 22, and start the Standby SCW Pump in order to prevent the Turbine Trip Timer from timing out and tripping the Main Turbine.

8. An un-isolable Steam Leak will develop in the Drywell and gradually worsen, resulting in rising Drywell Temperature and Pressure. The crew will respond in accordance with 3-AOI-64-1, Drywell Pressure and/or Temperature High or Excessive Leakage into the Drywell, and will be forced to insert a manual Reactor SCRAM. The NUSO will respond in accordance with 3-EOI-2, Secondary Containment Control.

9. The first loop of Drywell Sprays will fail when the crew attempts to spray Containment to reduce Containment Pressure and Temperature, requiring action to use the other loop of Drywell Spray.

The Scenario ends when the crew has sprayed Containment and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using either high or low pressure systems.

Critical Tasks 2

1. **Start the Standby Stator Cooling Water Pump before the Turbine Trip Timer times out and trips the Main Turbine, resulting in a Reactor SCRAM.**

   a. **Safety Significance**
   Prevents an unnecessary Reactor SCRAM.

   b. **Cues**
   Turbine Trip Timer Initiated Alarm.
   Standby Stator Cooling Water Pump fails to auto start.

   c. **Measured by:**
   Observation – the crew takes action to start the standby Stator Cooling Water Pump.

   d. **Feedback**
   The standby Stator Cooling Water Pump starts.
   The Turbine Trip Timer Initiated alarm clears.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to start the standby Stator Cooling Water Pump and an automatic Reactor SCRAM occurs after the Turbine Trip Timer times out.

2. **Initiate Drywell Sprays while in the safe area of the Drywell Spray Initiation Limit Curve and before Drywell Temperature rises to 280 °F.**

   a. **Safety Significance:**
   Precludes failure of containment.

   b. **Cues:**
   Procedural compliance.
   High Drywell Pressure and Suppression Chamber Pressure.

   c. **Measured by:**
   Observation - NUSO directs Drywell Sprays in accordance with 3-EOI-Appendix-17B, RHR System Operation Drywell Sprays.
   <u>**AND**</u>
   Observation - RO initiates Drywell Sprays.

   d. **Feedback:**
   Drywell and Suppression Pressure lowering.
   RHR flow to containment.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to spray the Drywell before Drywell Temperature reaches 280 ºF.

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Event Description: Remove ‘C’ Condensate Booster Pump (CBP) from Service for Maintenance

Time	Position	Applicant’s Actions or Behavior
	Driver	**Prior to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not proceed to Event 1, Remove ‘C’ Condensate Booster Pump (CBP) from Service for Maintenance, after assuming the shift request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to secure 2C Condensate Booster Pump.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to secure 2C Condensate Booster Pump.**
	NUSO	Directs the Balance of Plant Operator (BOP) to remove 2C Condensate Booster Pumps from service in accordance with 2-OI-2 Condensate System,Section 8.18, Removing a Condensate Booster Pump from service at High Power

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 1 Page 2 of 3

Event Description: Remove ‘C’ Condensate Booster Pump (CBP) from Service for Maintenance

Time	Position	Applicant’s Actions or Behavior
	BOP	2-OI-2, Condensate System Section 8.19, Removing a Condensate Booster Pump from service at High Power **NOTES** 1) There is adequate FLOW / NPSH to maintain 100% power when one Condensate Booster Pump is taken out of service. 2) During operation with only two Condensate Booster Pumps in service (3-2-3): • Condensate flow above 14.1 MLBM/HR (Approximately 87.5% power) can result in the Condensate Booster Pump motors operating above their rated horsepower but within rated service factor • While operating within rated service factor, the Condensate Booster Pump motor winding temperatures are not to exceed 266 degrees F and the motor amps are not to exceed 427.8 amps 3) There is adequate FLOW / NPSH to maintain 100% power when one Reactor Feedwater Pump and one Condensate Booster Pump (3-2-2) are taken out of service and the following conditions apply: • Three Condensate Pumps are in service • The Reactor Feedwater Pump is removed from service and secured (no flow thru the minimum flow valve) prior to removing the Condensate Booster Pump from service • Note 2 above is applicable
	BOP	[1] **REVIEW** Precautions and Limitations in Section 3.4. **Completed during pre-shift brief.** [2] **IF** time permits, **THEN REVIEW** Drawing 2-47E800-3 Notes regarding operational guidelines for Condensate and Feedwater system. (Otherwise N/A)

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 1 Page 3 of 3

Event Description: Remove ‘C’ Condensate Booster Pump (CBP) from Service for Maintenance

Time	Position	Applicant’s Actions or Behavior
	BOP	[3] **ENSURE** Reactor Power is ≤ 95% If testing in accordance with 0-TI-704, Condensate Pump, Condensate Booster Pump and Feedwater Pump Testing, then adjust Reactor Power as required to support 0-TI-704 prior to stopping a Condensate Booster Pump. (Ref P&L 3.1 B) [4] **ENSURE** hydrogen injection is secured to the Condensate Booster Pump to be stopped. **REFER TO** 2-OI-4, Hydrogen Water Chemistry System.
	Driver	**If directed as the Turbine Building Assistant Unit Operator (AUO) to perform 2-OI-4, Hydrogen Water Chemistry System, Section 8.10 [2.3] Shut down Hydrogen Injection to 2C Condensate Booster Pump, acknowledge the direction. Inform the crew that Hydrogen Water Injection is secured to 2C Condensate Booster Pump.**
	BOP	[5] **N/A** [6] **WHEN** directed by the Unit 2 Unit SRO, **THEN STOP** CONDENSATE BOOSTER PUMP using one of the following: • 2-HS-2-68A, CONDENSATE BOOSTER PUMP 2C [7] **ENSURE** limiting conditions for Condensate Booster Pump operation in 0-OI-57A, Switchyard and 4160V AC Electrical System are met. [8] **N/A**
	NRC	**End of Event 1. Request that the Driver insert Event 2, Control Rod Drive (CRD) Flow Controller Fails High.**

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 2	Page 1 of 2
Event Description: Control Rod Drive (CRD) Flow Controller Fails High			

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, insert Event 2, Control Rod Drive (CRD) Flow Controller Fails High.**
	OATC	Acknowledges and reports the following alarm to the Nuclear Unit Senior Operator (NUSO): • CRD ACCUMULATOR CHARGING WATER HEADER PRESSURE HIGH, 2-9-5A, Window 10
	NUSO	Acknowledges the Operator at the Controls' (OATC) alarm report. Directs the OATC to respond in accordance with applicable Alarm Response Procedures and subsequently 2-OI-85, Control Rod Drive System.
	OATC	Alarm Response Procedure, 2-ARP-9-5A CRD ACCUM CHG WTR HDR PRESS HIGH, 2-9-5A, Window 10 A. **CHECK** pressure high on 2-PI-85-13A, CRD ACCUMULATOR CHARGING WATER HEADER on Panel 2-9-5. B. **CHECK** 2-FCV-85-11A (B), CRD LINE A(B) FLOW CONTROL VALVE, in service.
	NRC	**The crew may attempt to switch Flow Control Valves. However, as long as the Flow Controller is failed High, neither set of Flow Control Valves will operate in automatic. 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, must be placed in MANUAL.**
	OATC	C. **IF** in-service controller has failed, **THEN REFER TO** 2-OI-85, Control Rod Drive System. D. **N/A**
	OATC	Determines that the CRD Flow Controller has failed High, causing 2-FCV-85-11A, CRD LINE A FLOW CONTROL VALVE to CLOSE. In accordance with OPDP-1, Conduct of Operations, Section 3.5.3, Manual Control of Automatic Systems, takes manual control of 2-FIC-85-11, CRD SYSTEM FLOW CONTROL to restore CRD Parameters back to normal.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 2 Page 2 of 2

Event Description: Control Rod Drive (CRD) Flow Controller Fails High

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-85, Control Rod Drive System Section 8.33, AUTOMATIC/MANUAL operation of 2-FIC-85-11 [1] **REVIEW** all Precautions and Limitations in Section 3.6. [2] **IF** transferring 2-FIC-85-11 from AUTO to MANUAL **THEN**: [2.1] **PLACE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL in BALANCE. [2.2] **BALANCE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL by turning Manual Control Pot inside Control Selector Wheel until red deviation pointer is in the Green Band. [2.3] **PLACE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL in MANUAL. [2.4] **ADJUST** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL manual potentiometer to establish the desired system flow. Refer to Section 5.1 or 6.10.
	NRC	**End of Event 2. Request that the Driver insert Event 3, Reactor Feedwater Pump (RFPT) Vibration Alarm.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 1 of 8

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3, Reactor Feedwater Pump (RFPT) Vibration Alarm.**
	OATC	Acknowledges and reports the following alarms: • RFPT C ABNORMAL, 2-9-6C, Window 15 • RFPT VIBRATION OR AXIAL POSITION HIGH-HIGH, 2-9-6C, Window 17
	NUSO	Acknowledges the Operator at the Controls' (OATC) alarm report. Directs the OATC to respond in accordance with applicable Alarm Response Procedures.
	NRC	**Given the degrading condition of 2C RFPT, the crew may elect one and/or both of the following paths:** **(1) Respond per 2-9-6C, Window 17 then remove 2C RFPT from service in accordance with 2-OI-3, Reactor Feedwater System (see page 8).** **(2) Respond per 2-9-6C, Window 17 then conservatively trip 2C RFPT in accordance with 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low (see page 13).**
	OATC	2-ARP-9-6C, Alarm Response Procedure RFPT C ABNORMAL, Window 15 Operator Action: A. **CHECK** other RFPT alarms on Panel 2-9-6 to determine problem area. B. **REFER TO** appropriate alarm response procedure. C. **IF** NO other annunciator on Panel 2-9-6 is in alarm, **THEN PERFORM** an alarm summary on alarm types.
	OATC	2-ARP-9-6C, Alarm Response Procedure RFPT VIB OR AXIAL POSITION HIGH-HIGH, 2-9-6C, Window 17 Operator Action: A. **CHECK** RFPT/RFP vibration readings on 2-XR-3-177 on Panel 2-9-6 AND RFPT and RFP Vibration display (RFPTV) on ICS.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 2 of 8

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	B. **DISPATCH** personnel to Panel 2-LPNL-025-0673, VIBRATION MONITORING PANEL, located outside of RFPT Room 2A, Elevation 617' to PERFORM the following: • REPORT vibration data for affected RFPT/RFP • REPORT all alarm/alert conditions on panel • Advise the Unit Operator of any changes in vibration data
	Driver	**If directed as the Turbine Building AUO to REPORT 2C RFPT vibration data, all alarms/conditions for Vibration Monitoring Panel, acknowledge direction. Wait 2 minutes, inform the OATC/BOP that the alarm is valid on the 2C RFPT. Vibration can be felt in the 2C RFPT room and ALERT lights are illuminated on the Vibration Monitoring Panel.**
	OATC	C. **IF** a sustained vibration of exceeding the DANGER setpoints (**REFER TO** setpoints on the next page) is confirmed on both pump inboard and outboard bearings OR any turbine bearing, **THEN REMOVE** the RFPT from service. **REFER TO** 2-OI-3, Reactor Feedwater System. D. **ADJUST** load on pump if necessary.
	NRC	**The crew will verify that Condensate System Flow measured on 2-XR-002-0026, CONDENSATE, is less than 12 Mlbm/hr (75% Reactor Power) prior to removing a 2C Reactor Feedwater Pump from service. See Event 4 below.**
	OATC	E. **IF** alarm does NOT reset, **THEN REMOVE** pump from service. **REFER TO** 2-OI-3, Reactor Feedwater System. RFP 2C Turbine Axial Thrust — 2-XM-3-1062-1 — 21 Mils 2-XM-3-1062-2 — 21 Mils Turbine Outbd Bearing — 2-XM-3-199A1 — 4.5 Mils 2-XM-3-199A2 — 4.5 Mils Turbine Inbd Bearing — 2-XM-3-199B1 — 4.5 Mils 2-XM-3-199B2 — 4.5 Mils Pump Inbd Bearing — 2-XM-3-0169A1 — 4.5 Mils 2-XM-3-0169A2 — 4.5 Mils Pump Outbd Bearing — 2-XM-3-0170A1 — 4.5 Mils 2-XM-3-0170A2 — 4.5 Mils Pump Axial Thrust — 2-XM-3-0171A1 — 20.0 Mils 2-XM-3-0171A2 — 20.0 Mils

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 3 of 8

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	NRC	**2C RFPT Axial Thrust vibration will ramp up to 15 mils to exceed DANGER setpoint on 2-XR-3-177 on Panel 2-9-6.**
	OATC	2-OI-3, Feedwater System Section 7.1, RFP/RFPT Shutdown **NOTES** 1) Reactor Power should be verified less than or equal to 95% prior to starting or stopping a Condensate Booster Pump or Reactor Feedwater Pump. REFER TO P&L 3.0VV if performing 0-TI-704. 2) There is NOT adequate flow/NPSH to maintain 100% CLTP when one Condensate Pump is out of service. To maintain adequate NPSH available to the Condensate Booster Pumps, Condensate flow is limited to 14.1 M lbm/hr. 3) When operating with one Condensate and/or Condensate Booster Pump out of service, verify that condensate system flow measured on 2-XR-002-0026, CONDENSATE (Point 4), is less than 12 mlbm/hr prior to removing a Reactor Feedwater Pump from service. 4) If Feedwater Control becomes unstable, it may be necessary to switch to SINGLE ELEMENT mode from the THREE ELEMENT mode earlier than recommended in this instruction.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 4 of 8

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	**CAUTIONS** 1) FAILURE to monitor SJAE/OG CNDR CNDS FLOW, 2-FI-2-42, on Panel 2-9-6 for proper flow (between 2 x 106 and 3 x 106 lbm/hr) may only result in SJAE poor performance. The SJAE's will NOT trip on Condensate System low pressure. 2) Changes in Condensate System flow may require adjustment to 2-FCV-002-0190, SPE CNDS BYPASS 3) When isolating the Reactor Feedwater Pump(s) for maintenance, the associated injection water should also be isolated to prevent high seal differential pressure and allow the RFW Pump shafts to rotate freely. 4) When a Reactor Feed Pump is isolated (suction, discharge, and minimum flow valve closed) with injection water aligned to the pump, there is a potential of rising pump casing pressure and seal water leak off flows reaching the point where seal water drains are overcome and seal water is forced into the oil system through the bearing housings. Therefore, the time that an RFPT is isolated with injection water aligned to the pump should be minimized. [1] **REFER TO** Section 3.0 and **REVIEW** Precautions and Limitations. [2] **N/A** [3] **IF** any Condensate or Condensate Booster Pump is NOT in service, **THEN ENSURE** that Condensate System Flow measured on 2-XR-002-0026, CONDENSATE (Point 4), is less than 12 Mlbm/hr (approximately 85% power) prior to removing a Reactor Feedwater Pump from service. [4] **N/A** [5] **ENSURE** in AUTO, RFPT Turning Gear Motor. • RFPT 2C TURNING GEAR MOTOR, 2-HS-3-152A

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 5 of 8

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	**NOTES** 1) When selected, Column 1 on individual RFPT Speed Control Panel Display Stations (PDS) displays actual pump speed and is not controlled in any mode. 2) When selected, Column 2 on individual RFPT Speed Control PDS displays pump flow bias and is changed with the Ramp Up/Ramp Down pushbuttons with the controller in AUTO. 3) When selected, Column 3 on individual RFPT Speed Control PDS displays RFPT speed demand and is changed with the Ramp Up/Ramp Down pushbuttons with the controller in MANUAL. 4) Attachment 2 can be referred to for additional information on the RFPT Speed Control PDSs. [6] **LOWER** speed of RFPT/RFP being removed from service by either of the following methods: • **IF** Using individual RFPT Manual Governor switch, **THEN GO TO** Step 7.1[7]. • **IF** Using individual RFPT Speed Control PDS in MANUAL, **THEN GO TO** Step 7.1[8]. [7] **SLOWLY LOWER** speed of individual RFPT on Panel 2-9-5, by performing the following: [7.1] **N/A** [7.2] **N/A** [7.3] **CONTROL** RFPT 2C [7.3.1] **DEPRESS** RFPT 2C SPEED CONT RAISE/LOWER switch, 2-HS-46-10A to MANUAL GOVERNOR. • **CHECK** amber light at switch illuminated. [7.3.2] **SLOWLY LOWER** RFPT speed, by placing RFPT Speed Control switch in RAISE and LOWER positions, as necessary. [7.3.3] **IF** this is NOT the last operating feed pump, **THEN OBSERVE** rise in speed of any remaining RFPT operating in AUTO.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 6 of 8

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	[8] SLOWLY LOWER RFPT Speed Control PDS, on Panel 2-9-5 by performing the following: [8.1] **N/A** [8.2] **N/A** [8.3] **CONTROL** RFPT 2C [8.3.1] **PLACE** PDS, 2-SIC-46-10 in MANUAL AND ENSURE Column 3 is selected. [8.3.2] **SLOWLY LOWER** RFPT speed, using Ramp Up/Ramp Down pushbuttons as necessary. [8.3.3] **IF** this is NOT the last operating feed pump, **THEN OBSERVE** rise in speed of any remaining RFPT operating in AUTO. [9] **N/A** [10] **CONTINUE** to slowly lower RFPT speed to minimum speed setting (approximately 600 rpm). **NOTES** 1) One RFPT may be allowed to remain as a running standby pump at minimum speed setting (approximately 600 rpm). 2) With Reactor Feed Pump running at ~600 rpm, adjusting 2-FIC-2-29A, CNDS FLOW CONTROL SHORT CYCLE, will supply vessel inventory as needed by raising and lowering the header pressure. Typically, a Feed Pump running at ~600 rpm will build 20 to 25 psig across the pump. 3) 2-LIC-3-53, RFW START-UP LEVEL CONTROL does not respond linearly with Controller Demand. The design is to respond slowly to dampen level swings. 4) This evolution has better results when the Condensate and Condensate Booster Pumps are in a two and two configuration. It should be noted that a two and one configuration will establish a lower header pressure and more attention will be needed to ensure Condensate minimum flow requirements are met. Conversely with a three and three configuration, pressure is higher with higher potential to overfeed the vessel.

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Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	[11] **IF** RFPT/RFP being Shut Down is NOT the last operating RFP, **OR IF** Unit is to be maintained > 400 psig while shut down, **THEN**: [11.1] MARK Step 7.1[12] as N/A. [11.2] GO TO Step 7.1[13] [12] **N/A** [13] **WHEN** RFPT is ready to be shut down, **THEN DEPRESS** RFPT TRIP, to trip RFPT being removed from service. (N/A IF Step 7.1[12] was performed.) • RFPT 2C TRIP, 2-HS-3-176A **NOTES** 1) Reverse flow through the RFPT Minimum Flow Valve could occur if the RFPT Discharge Check Valve is not properly seated. 2) The check valve position indicator should not be relied upon for positive valve closure indication. 3) Step 7.1[15] is performed only if RFPT Discharge Check Valve failure occurs. [14] **ENSURE CLOSED**, RFP DISCH TESTABLE CHECK VALVE, by one of the following: • Observing RFPT Discharge Flow indicator • Locally listening to check valve [15] **N/A** [16] **N/A** [17] **IF** RFP is NOT rolling on minimum flow, **THEN ENSURE** Turning Gear motor starts and engages when RFPT coasts down to zero speed.
	NRC	**The crew may elect to conservatively trip 2C RFPT in accordance with 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 8 of 8

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low Section 5.0 [9] **IF** a RFPT has tripped and will not be required to maintain level, **THEN REFER TO** 2-OI-3, Reactor Feedwater System and **SHUT DOWN** the tripped RFPT.
	NRC	**End of Event 3. Proceed to Event 4, Power Reduction for RFPT Shutdown.**

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Event Description: Power Reduction for RFPT Shutdown

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 4, Power Reduction for RFPT Shutdown, is entered by the crew. No action is required by the Driver to insert Event 4.**
	NUSO	Prior to removing 2C Reactor Feedwater Pump from service, directs the OATC to verify that Condensate system flow measured on 2-XR-002-0026, CONDENSATE, is less than 12 Mlbm/hr (approximately 85% Reactor Power) in accordance with 2-OI-3, Reactor Feedwater System, Section 7.1 RFP/RFPT Shutdown.
	OATC	Lowers Reactor Power to ensure that Condensate System Flow measured on CONDENSATE, 2-XR-002-0026, is less than 12 Mlbm/hr prior to removing 2C Reactor Feedwater Pump from service. May elect to use either the Master Recirc Speed Control or a Recirc System Runback (or a combination of both methods) in accordance with 2-OI-68, Reactor Recirculation System. 2-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow [1] **N/A** [2] **WHEN** desired to control Recirc Pumps 2A and/or 2B speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump Speed 2A & 2B using the following pushbuttons as required. • 2-HS-96-31, RAISE SLOW • 2-HS-96-32, RAISE MEDIUM • 2-HS-96-33, LOWER SLOW • 2-HS-96-34, LOWER MEDIUM • 2-HS-96-35, LOWER FAST

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 4 Page 2 of 2

Event Description: Power Reduction for RFPT Shutdown

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-68, Reactor Recirculation System Section 8.12, Initiating Manual Runbacks [1] **IF** time permits, **THEN REVIEW** Precautions and Limitations. (REFER TO Section 3.0). [2] **IF** desired to reduce Reactor Power to approximately 90%, **THEN** (Otherwise N/A): [2.1] **DEPRESS** 2-HS-68-42, RECIRC PUMPS UPPER POWER RUNBACK Pushbutton. [2.2] **CHECK** the following: • Pushbutton backlight blinks until setpoint is reached • Reactor Power lowers to approximately 90% [3] **IF** desired to reduce Reactor Power to 66.3%, **THEN** (Otherwise N/A): [3.1] **DEPRESS** 2-HS-68-43, RECIRC PUMPS MID POWER RUNBACK pushbutton. [3.2] **CHECK** the following: • Pushbutton backlight blinks until setpoint is reached • Reactor Power lowers to 66.3% [4] **IF** desired to reduce Core Flow to approximately 60%, **THEN** (Otherwise N/A): [4.1] **DEPRESS** 2-HS-68-44, RECIRC PUMPS CORE FLOW RUNBACK. [4.2] **CHECK** the following: • Pushbutton backlight blinks until setpoint is reached • Core Flow lowers to approximately 60%
	NRC	**End of Event 4. Request that the driver insert Event 5, Refuel Zone Radiation Monitors Fail Upscale.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 1 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner insert Event 2, Refuel Zone Radiation Monitors Fail Upscale.**
	BOP	Acknowledges and reports the following alarms: • REFUELING ZONE EXHAUST RADIATION HIGH, 2-9-3A, Window 34 • DRYWELL LEAK DETECTION RADIATION HIGH, 2-9-3D, Window 19 • DRYWELL/SUPPR CHAMBER H2O2 ANALYZER FAILURE, 2-9-7C, Window 22
	BOP	Alarm Response Procedure, 2-ARP-9-3A REFUELING ZONE EXHAUST RADIATION HIGH, Window 34 Operator Actions: A. CHECK alarm condition on the following: 1. 2-RR-90-144, REACTOR & REFUEL ZONE EXHAUST RADIATION recorder on Panel 2-9-2. 2. 2-RM-90-140/142, RX & REFUEL ZONE EXH CH A RAD MON RTMR radiation monitor on Panel 2-9-10. 3. 2-RM-90-141/143, RX & REFUEL ZONE EXH CH BRAD MON RTMR radiation monitor on Panel 2-9-10. B. **N/A** C. **NOTIFY** Unit SRO, Unit 1 and Unit 3.
	Driver	**If contacted as Unit 1, or Unit 3 acknowledge any information given.**
	BOP	D. **N/A** E. **N/A** F. **ENTER** 2-EOI-3, Secondary Containment Control. G. **REFER TO** 2-AOI-64-2D, Group 6 Ventilation System Isolation and, for loss of power to NUMAC drawer, to 2-OI-90, Radiation Monitoring System, Section 6.0. H. **N/A** I. **REFER TO** EPIP-1, Emergency Classification Procedure.

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 5	Page 2 of 12
Event Description: Refuel Zone Radiation Monitors Fail Upscale			

Time	Position	Applicant's Actions or Behavior
	NUSO	J. **REFER TO** Technical Specification Section 3.3.6.2, Secondary Containment Isolation instrumentation and 3.3.7.1, CREV System Instrumentation.
	NUSO	Enters 2-EOI-3, Secondary Containment Control.
	NRC	**2-AOI-64-2D, Group 6 Ventilation System Isolation is covered starting on page 18.**
	BOP	Alarm Response Procedure, 2-ARP-9-3D DRYWELL LEAK DETECTION RADIATION DOWNSCALE, Window 19 Operator Action: A. **DETERMINE** cause of alarm by performing the following: 1. **CHECK** AIR PARTICULATE MONITOR CONTROLLER, 2-MON-90-50 on Panel 2-9-2 for condition bringing in alarm 2. **N/A** B. **N/A** C. **N/A** D. **IF** corrective maintenance is required, **THEN NOTIFY** Chemistry to commence its sampling procedure. E. **REFER TO** Tech Specs 3.4.4, RCS Operational Leakage, 3.4.5, RCS Leakage Detection System, and TRM 3.3.10, Reactor Coolant Leakage Detection Instrumentation for CAM LCO requirements and **IMPLEMENT** appropriate TS/TRM actions as required. F**. N/A**
	Driver	**If notified as Chemistry to begin sampling, acknowledge the direction.** **If notified as the Work Control/Outside NUSO to investigate, acknowledge the direction.**
	NRC	**The NUSO may enter 2-EOI-3, Secondary Containment Control, based on the receipt of the REFUELING ZONE EXHAUST RADIATION HIGH Alarm (2-9-3A, Window 34).**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 3 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Shift Manager concerning 2-EOI-3, Secondary Containment Control, acknowledge any reports given and concur with any recommendations.**
	BOP	2-AOI-64-2D, Group 6 Ventilation System Isolation 4.1 Immediate Actions: None 4.2 Subsequent Actions [1] Using Panel 2-9-3 mimic or Containment Isolation Status System on Panel 2-9-4, ENSURE Group 6 isolation valves penetrating Primary Containment are CLOSED. [2] IF Refuel Zone Isolation is due to high radiation as indicated on 2-RM-90-140/142, RX & REFUEL ZONE EXH CH A RAD MON RTMR, or 2-RM-90-141/143, RX REFUEL ZONE EXH CHA A RAD MON RTMR, (Panel 2-9-10) or 2-RR-90-144, REACTOR & REFUEL ZONE EXHAUST RADIATION, (Panel 2-9-2) or associated recorder on Panel 2-9-2, THEN PERFORM the following, otherwise, MARK steps N/A: [2.1] **N/A** [2.2] **N/A** **CAUTION** Main Steam Isolation Valves (MSIV's) may isolate on a Group I isolation if the time the Reactor Zone fans are removed from service is NOT minimized during Reactor Power operation and the Steam Vault Exhaust Booster Fan is NOT in-service. Steam tunnel Temperature is to be closely monitored while Reactor Zone fans are out-of-service. [3] **MONITOR** Steam tunnel temperature closely while Reactor Zone fans are out of service. [4] **IF** Steam tunnel temperature rises, **THEN** Using 2-OI-30B, Reactor Zone Ventilation System, **ENSURE** STEAM VAULT EXH BOOSTER FAN is in-service. Otherwise, MARK N/A.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 4 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>[5] N/A
[6] N/A
[7] MONITOR the following to aid in determining location of problem:
2-RR-90-1, AREA RADIATION (Panel 2-9-2)
2-MON-90-50, AIR PARTICULATE MONITOR CONSOLE (Panel 2-9-2)
2-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE (Panel 2-9-22)
NOTE
Based on recent On-Shift Analysis from the License Amendment, chemistry sampling may be delayed following plant events. In the case of a REP event, Chemistry will be required to assist with dose assessments. Dose assessments will have precedence over sampling actions in REP events.

[8] N/A
[9] N/A
[10] IF isolation is the result of invalid radiation signal, THEN REFER TO 2-OI-90, Radiation Monitoring System, Section 6.6, NUMAC Radiation Monitor Operation, Immediate Clearing of Group 6 Isolation, to inhibit trip. Otherwise, MARK N/A.</td></tr>
<tr><td></td><td>BOP</td><td>2-OI-90, Radiation Monitoring System, Section 6.6, NUMAC Radiation Monitor Operation

NOTES
1) This section is applicable to Main Steam Line Radiation monitors 2-RM-90-136, 137 and Reactor Zone/Refuel Zone Radiation monitors 2-RM-90-140/142 and 2-RM-90-141/143.
2) A screen saver activates on the monitor after 30 minutes of constant display.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 5 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	BOP	[1] **IF** the screen saver is activated, **THEN DEPRESS** any of the prompt keys at the bottom of the screen to display the monitored channels. **NOTES** 1) There are two detectors for each channel of the Reactor Zone/Refuel Zone Monitors and are indicated on each monitor as follows: 2-RM-90-140/142 Display — Description 2A 2-RE-90-142A — Reactor Zone channel A detector A 2B 2-RE-90-142B — Reactor Zone channel A detector B 0A 2-RE-90-140A — Refuel Zone channel A detector A 0B 2-RE-90-140B — Refuel zone channel A detector B 2-RM-90-141/143 Display — Description 3A 2-RE-90-143A — Reactor Zone channel B detector A 3B 2-RE-90-143B — Reactor Zone channel B detector B 1A 2-RE-90-141A — Refuel Zone channel B detector A 1B 2-RE-90-141B — Refuel Zone channel B detector B 2) Only the "A" detector of each channel described above has input to radiation recorder 2-RR-90-144 Reactor & Refuel Zone Exhaust Radiation. 3) Any active trip condition will be indicated by a highlighted "TRIP" at the top of the screen. A non-highlighted "TRIP" at the top of the screen indicates that there are one or more past trip conditions that have not been acknowledged. 4) Trips on the Reactor Zone/Refuel Zone Radiation monitors will automatically reset when the alarming condition resets.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 6 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	BOP	[2] Immediate Resetting of Group 6 Isolation Due to Reactor Zone Radiation Monitors **NOTES** 1) This section is to be performed in the event of a trip signal that will not reset in order to prevent further impact to plant operation due to reactor zone isolation. This is only considered appropriate when the signal is believed to be invalid. 2) Technical Specifications only allow one trip channel at a time to be out of service. This section provides directions for removing both trip channels from service but should only be performed on one channel at a time. Reference Technical Specification 3.3.6.2 for limiting conditions. 3) This section places jumpers to inhibit the upscale trips for a monitor. 4) Downscale trips take more than one trip channel to activate the logic. **CAUTION** A Reactor Zone Isolation can cause a Unit scram in less than five minutes due to high temperature in the Steam Tunnel. [2.1] **PLACE** affected monitor Key-lock switch to INOP position. [2.2] **IF** the affected monitor is 2-RM-90-140/142, **THEN PLACE** jumper across the following terminals in the back of Panel 2-9-10 to inhibit the upscale trip: TB HH terminals 49 and 50 [2.3] **N/A**
	Driver	**If contacted as the Work Control/Outside NUSO or Instrument Mechanics to install the jumper on Terminal Board HH Terminals 49 and 50, acknowledge the direction.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 7 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: When the Key-Lock switch on the Refuel Monitors on Panel 2-9-10 is placed in the INOP Position, the Monitor will display message: "INOP IS NOT SUPPORTED ON THE SIMULATOR. RETURN THE INOP/OPER KEY TO OPER TO RETURN THE NUMAC TO OPERABILITY". This message is normal, and no further action is required by the candidates with respect to placing the Radiation Monitor in an inoperable status.**
	NUSO	Technical Specification 3.3.6.2, Secondary Containment Isolation Instrumentation LCO 3.3.6.2 The secondary containment isolation instrumentation for each Function in Table 3.3.6.2-1 shall be OPERABLE. APPLICABILITY: According to Table 3.3.6.2-1

Secondary Containment Isolation Instrumentation
3.3.6.2

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3, (a)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr

(a) During operations with a potential for draining the reactor vessel.

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 5	Page 8 of 12
Event Description: Refuel Zone Radiation Monitors Fail Upscale			

Time	Position	Applicant's Actions or Behavior	
	NUSO	**CONDITION:** A. – One or more channels INOPERABLE.	
	NUSO	**REQUIRED ACTION:** A.1 Place channel in trip	**COMPLETION TIME:** 12 hours for Functions 1 and 2 <u>AND</u> 24 hours for Functions other than Functions 1 and 2
	NUSO	Technical Specification 3.3.7.1, CREV System Instrumentation LCO 3.3.7.1 The CREV System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE Applicability: According to Table 3.3.7.1-1	

CREV System Instrumentation
3.3.7.1

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3,(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3 (a)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
5. Control Room Air Supply Duct Radiation - High	1,2,3, (a)	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 270 cpm above background

(a) During operations with a potential for draining the reactor vessel.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 9 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	**CONDITION:** A. – One or more required channels INOPERABLE.	
	NUSO	**REQUIRED ACTION:** A.1 Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	**COMPLETION TIME:** Immediately
	NUSO	**CONDITION:** C. – As required by Action A.1 and referenced in Table 3.3.7.1-1.	
	NUSO	**REQUIRED ACTION:** C.1 Declare associated CREV subsystem inoperable. AND C.2 Place channel in trip.	**COMPLETION TIME:** C.1 – 1 hour from discovery of loss of CREV initiation capability C.2 – 24 hours
	NRC	**It is acceptable for the candidate to enter Technical Specification 3.4.5, RCS Leakage Detection Instrumentation, (see page 26) without first entering Technical Requirements Manual 3.3.10, Reactor Coolant Leakage Detection, (see next page) first.**	
	NUSO	Tech Req Manual 3.3.10, Reactor Coolant Leakage Detection LCO 3.3.10 The Reactor Coolant Leakage Detection Instrumentation for each function in Table 3.3.10-1 shall be OPERABLE Applicability: Modes 1,2,3 **CONDITION**: A. – Required instrumentation INOPERABLE.	
	NUSO	**REQUIRED ACTION:** A.1 Enter the Condition referenced in Table 3.3.10-1 for the Function.	**COMPLETION TIME:** Immediately

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 10 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	NUSO	(Table 3.3.10-1 reproduced below)
	NUSO	**CONDITION:** D. – As Required by Required Action A.1 and referenced in Table 3.3.10-1

Reactor Coolant Leakage Detection
TR 3.3.10

Table 3.3.10-1
Reactor Coolant Leakage Detection Instrumentation

FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	TECHNICAL SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Drywell Equipment Drain Flow Integrating Recorder (a)	B	TSR 3.3.10.1 TSR 3.3.10.3 TSR 3.3.10.4	N/A
2. Deleted			
3. Deleted			
4. Drywell Floor Drain Flow Integrating Recorder (b)	C	TSR 3.3.10.1 TSR 3.3.10.4 (c)	N/A
5. Drywell Floor Drain Sump Fill Rate Timer (b)	B	TSR 3.3.10.1 TSR 3.3.10.2 TSR 3.3.10.4	≥ 80.4 min
6. Drywell Floor Drain Sump Pump Out Rate Timer (b)	B	TSR 3.3.10.1 TSR 3.3.10.2 TSR 3.3.10.4 TSR 3.3.10.5	≤ 8.9 min
7. Drywell Air Sampling (Gas)	D	(d)	3 X Average Background
8. Drywell Air Sampling (Particulate)	E	(d)	3 X Average Background

(a) Used to determine identifiable reactor coolant LEAKAGE. Considered part of sump system.
(b) Used to determine unidentifiable reactor coolant LEAKAGE. Considered part of sump system.
(c) The channel calibration will be performed in accordance with SR 3.4.5.3.
(d) Surveillances will be performed in accordance with SR 3.4.5.1, 3.4.5.2 and 3.4.5.4.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 11 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** D.1 – Verify the primary containment atmospheric monitoring system particulate channel is OPERABLE. OR D.2 – Declare the primary containment atmospheric monitoring system inoperable. (TS LCO 3.4.5)	**COMPLETION TIME:** D.1 – Immediately D.2 – Immediately
	NUSO	**CONDITION:** E. – As Required by Required Action A.1 and referenced in Table 3.3.10-1	
	NUSO	**REQUIRED ACTION:** E.1 – Verify the primary containment atmospheric monitoring system gas channel is OPERABLE. OR E.2 – Declare the primary containment atmospheric monitoring system inoperable. (TS LCO 3.4.5)	**COMPLETION TIME:** E.1 – Immediately E.2 – Immediately
	NUSO	Technical Specification 3.4.5, RCS Leakage Detection Instrumentation LCO 3.4.5 The following RCS Leakage Detection Instrumentation shall be OPERABLE: a. Drywell Floor Drain Sump monitoring system; and b. One channel of either Primary Containment atmospheric particulate or atmospheric gaseous monitoring system Applicability: Modes 1, 2, and 3. **CONDITION:** B – Required Primary Containment atmospheric monitoring system inoperable.	

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 12 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** B.1 – Analyze grab samples of Primary Containment atmosphere. AND B.2 – Restore required Primary Containment atmospheric monitoring system to OPERABLE status.	**COMPLETION TIME:** B.1 – Once per 12 hours B.2 – 30 days
	NRC	**NOTE: No action is required within Technical Specification 3.4.4, Reactor Coolant System (RCS) Operational Leakage.**	
	NRC	**End of Event 5. Proceed to Event 6, Standby Gas Train 'C' Fails to Auto Start.**	

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 6 Page 1 of 1

Event Description: Standby Gas Train 'C' Fails to Auto Start

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 6, Standby Gas Train 'C' Fails to Auto Start, is automatically entered on Simulator setup. No action is required by the driver to insert Event 6.**
	Driver	**If contacted as the U1 or U3 operator to start SBGT C, state that U1 / U3 operators cannot leave the horse shoe area at this time.**
	BOP	Determines that 'C' Standby Gas Train (SGT) did not automatically start. In accordance with OPDP-1, Conduct of Operations, Section 3.5.3, Manual Control of Automatic Systems, manually starts 'C' SGT.
	BOP	0-OI-65, Standby Gas Treatment System Section 5.2, Standby Gas Treatment System Manual Initiation [4] **START** SGT FAN A(B)(C) as follows: [4.2] **IF** starting SGT FAN C from Panel 2-9-25, **THEN PLACE** SGTS FAN C, 0-HS-65-69A/2 in START.
	BOP	Informs the NUSO that 'C' SGT failed to automatically start, but is started manually and is running normally.
	Driver	**If contacted as the Work Control/Outside NUSO to investigate the cause for 'C' SGT not automatically starting, acknowledge the direction.**
	NUSO	Technical Specification 3.6.4.3, SGT System LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE Applicability: Modes 1, 2, and 3. During operations with a potential for draining the Reactor Vessel (OPDRVs) **CONDITION:** A. – One SGT System inoperable.

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Event Description: Standby Gas Train 'C' Fails to Auto Start

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** A.1 – Restore SGT subsystem to OPERABLE status.	**COMPLETION TIME:** A.1 – 7 days
	NRC	**End of Event 6. Request that the Driver insert Event 7, 2A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start.**	

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 7	Page 1 of 3

Event Description: 2A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: The Unit 2 Main Turbine will trip and the Reactor will SCRAM in approximately 1 minute from the loss of 2A Stator Cooling Water Pump if the crew does not manually start the standby pump (it will fail to automatically start).**
	Driver	**When requested by the Chief Examiner, insert Event 7, 3A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start.**
	Crew	**Critical Task:** **Start the Standby Stator Cooling Water Pump before the Turbine Trip Timer times out and trips the Main Turbine, resulting in a Reactor SCRAM.** **Critical Task Failure Criteria:** **The operating crew fails to start the standby Stator Cooling Water Pump and an automatic Reactor SCRAM occurs after the Turbine Trip Timer times out.**
	BOP	Acknowledges and reports the following alarms: • GEN STATOR COOLANT SYS ABNORMAL, 2-9-7A, Window 22 • TURBINE TRIP TIMER INITIATED, 2-9-8A, Window 1
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures
	BOP	Alarm Response Procedure, 2-ARP-9-7A GEN STATOR COOLANT SYS ABNORMAL, Window 22 **NOTE** The control room alarm typer can be used to confirm this alarm.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 7 Page 2 of 3

Event Description: 2A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start

Time	Position	Applicant's Actions or Behavior
	BOP	A. **IF** while performing the action of this ARP 2-XA-55-9-8A window 1 alarms, **THEN** 1. **ENSURE** all available Stator Cooling Water Pumps running. 2. **ATTEMPT** to RESET alarm. (2-XA-55-9-8A Window 1) 3. **IF** alarm fails to reset AND Reactor Power is above Turbine Bypass Valve capability, **THEN SCRAM** the Reactor. B. **ENSURE** a Stator Cooling Water Pump is running and **CHECK** stator temperature recorder, 2-TR-57-59, Panel 2-9-8.
	BOP	Starts 2B SCW Pump. Verifies SCW has been restored and that TURBINE TRIP TIMER INITIATED, 2-9-8A, Window 1, can be reset.
	BOP	B. **ENSURE** a Stator Cooling Water Pump is running and **CHECK** stator temperature recorder, 2-TR-57-59, Panel 2-9-8. C. **CHECK** alarm and **MONITOR** stator cooling system parameters using ICS "STATCWA" or "MAINGEN". D. **REQUEST** personnel to **REFER TO** Local Panel ARP for correct alarm response actions to be taken. E. **N/A**
	BOP	Alarm Response Procedure, 2-ARP-9-8A TURBINE TRIP TIMER INITIATED, Window 1 **NOTE** The control room alarm typer can be used to confirm this alarm. Operator Action: A. **CHECK** Stator Cooling Water Flow and Temperature and Generator Stator temperatures using ICS. B. **ENSURE** all available Stator Cooling Water Pumps running. **NOTE** The full capacity of the Turbine Bypass Valves with all nine valves open is 25% Reactor Power. To determine the capacity of the Bypass Valves, subtract 3% for each out of service Bypass Valve from the 25%. (Example, one Bypass Valve out of service, [25% - 3% = 22%], therefore, the capacity of the Bypass Valves with one Bypass Valve out of service is 22%.)

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 7	Page 3 of 3

Event Description: 2A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start

Time	Position	Applicant's Actions or Behavior
		C. **IF** all of the following conditions exist: • Alarm fails to reset, • Low Stator Cooling Water flow OR High Generator or Stator Cooling temperatures are observed on ICS, • Reactor Power is above Turbine Bypass Valve capability, **THEN**, SCRAM the Reactor (Otherwise N/A) D. **DISPATCH** personnel to Stator Coolant Unit to investigate.
	Driver	**If contacted as the Turbine Building AUO to investigate the cause for 2A SCW Pump tripping, acknowledge the direction. After 3 minutes, report that 2A SCW Pump is hot to the touch.** **If contacted as Work Control/Outside SRO to write a clearance for 2A SCW Pump and/or protect 2B SCW Pump, acknowledge the direction.**
	NRC	**End of Event 7. Request that the Driver insert Event 8, Steam Leak in the Drywell.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 1 of 12

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5, Reactor Feedwater Pump (RFPT) Vibration Alarm.**
	BOP	Acknowledges and reports the following alarms as they are received: • DRYWELL TO SUPPR CHAMBER DIFF PRESS ABNORMAL, 2-9-3B, Window 26 • PRI CONTAINMENT N2 PRESS HIGH, 2-9-3B, Window 10 • DRYWELL NORM OPERATING PRESS HIGH, 2-9-3B, Window 19 • DRYWELL ATMOSPHERIC TEMP HIGH, 2-9-3B, Window 3 • DRYWELL PRESSURE ABNORMAL, 2-9-5B, Window 31 • DRYWELL PRESS APPROACHING SCRAM, 2-9-3B, Window 30
	NUSO	Acknowledges alarm report and directs the BOP to respond in accordance with appropriate Alarm Response Procedures. Directs the crew to monitor Drywell Pressure and Temperature, and provides critical parameters and set points for further action.
	BOP	Alarm Response Procedure, 2-ARP-9-3B PRI CONTAINMENT N_2 PRESS HIGH, Window 10 A. CHECK Containment Pressure using multiple indications: B. CHECK Containment Temperature. C. REFER TO 2-OI-64, Primary Containment System, Section 6.1, Venting the Drywell with Standby Gas Treatment fan.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 2 of 12

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 2-ARP-9-3B DRYWELL NORM OPERATING PRESS HIGH, Window 19 Operator Action: A. **CHECK** drywell pressure and temperature for rise. B. **CHECK** weather report for atmospheric pressure. C. **IF** Drywell DP Compressor is running, **THEN STOP** compressor. D. **CHECK** N2 makeup valves to Suppression Chamber and Drywell closed. E. **CHECK** Drywell Control Air System Flow Elements 2-FIQ-032-00092 (Rx Bldg 565' R10-S) and 2-FIQ-032-0075 (Rx Building 565' R20-T0) < 0.5 SCFM. F. **IF** pressure rise is due to normal startup, **THEN REFER TO** 2-OI-64, Primary Containment System for normal venting instructions. G. **IF** Drywell Pressure is high, **THEN REFER TO** 2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell.
	NRC	**Due to the rate of rise of Drywell Pressure, the crew may not have time to address rising Drywell Pressure using 2-OI-64, Primary Containment System, or 2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell. 2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell, actions start on page 41.**
	OATC	When the Drywell Pressure/Temperature trigger point is reached, inserts a Core Flow Runback and Reactor SCRAM.
	OATC	2-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 2-HS-99-5A/S3A and 2-HS-99-5A/S3B, REACTOR SCRAM A and B, on Panel 2-9-5. [2] **PLACE** 2-HS-99-5A-S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] **N/A**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 3 of 12

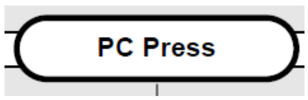
Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	OATC	[4] **IF** Reactor Power is 5% or BELOW, **THEN**: (Otherwise **MARK** N/A) **REPORT** the following to the UNIT SRO: • Reactor SCRAM • MODE Switch is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Power level
	Driver	**If contacted as any AUO to perform the following, acknowledge the direction:** • **Monitor Diesels** • **Perform the Gas Log**
	OATC	2-AOI-100-1, Reactor SCRAM 4.2 Subsequent Actions [1] **ANNOUNCE** Reactor SCRAM over PA system. [2] **DRIVE** in all IRMs and SRMs from Panel 2-9-5 as time and conditions permit. [2.1] **DOWNRANGE** IRMs as necessary to follow power as it lowers. [3] **ENSURE** SCRAM DISCH VOLUME VENT & DRAIN VALVES CLOSED by green indicating lights at SDV Display on Panel 2-9-5.
	OATC	Informs the NUSO when Drywell Pressure reaches 2.45 psig.
	NUSO	When Drywell Pressure reaches 2.45 psig, enters the following EOIs and informs the crew: • 2-EOI-1, RPV Control • 2-EOI-2, Primary Containment Control

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 4 of 12

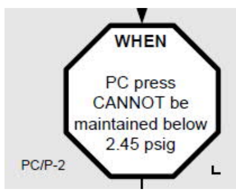
Event Description: Steam Leak in the Drywell

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NRC</td><td>Candidate may elect to first spray the Drywell based on the Drywell Temperature leg of 2-EOI-2, Primary Containment Control (See page 39)</td></tr>
<tr><td></td><td>NUSO</td><td>
2-EOI-2, Primary Containment Control



PC/P-1

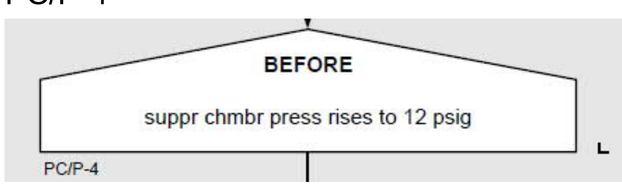
MONITOR and CONTROL Primary Containment Pressure below 2.45 PSIG using the vent system (2-AOI-64-1)



PC/P-3

<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Primary Containment Pressure reduction is required to restore and maintain Adequate Core Cooling or reduce total offsite radiation dose</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Chamber Pressure drops to 0 PSIG</td><td>STOP Suppression Chamber Sprays</td></tr>
<tr><td>Drywell Pressure drops to 0 PSIG</td><td>STOP Drywell Sprays</td></tr>
</table>

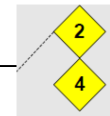
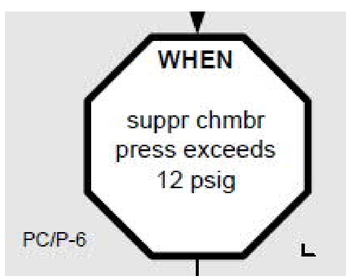
PC/P-4



</td></tr>
</table>

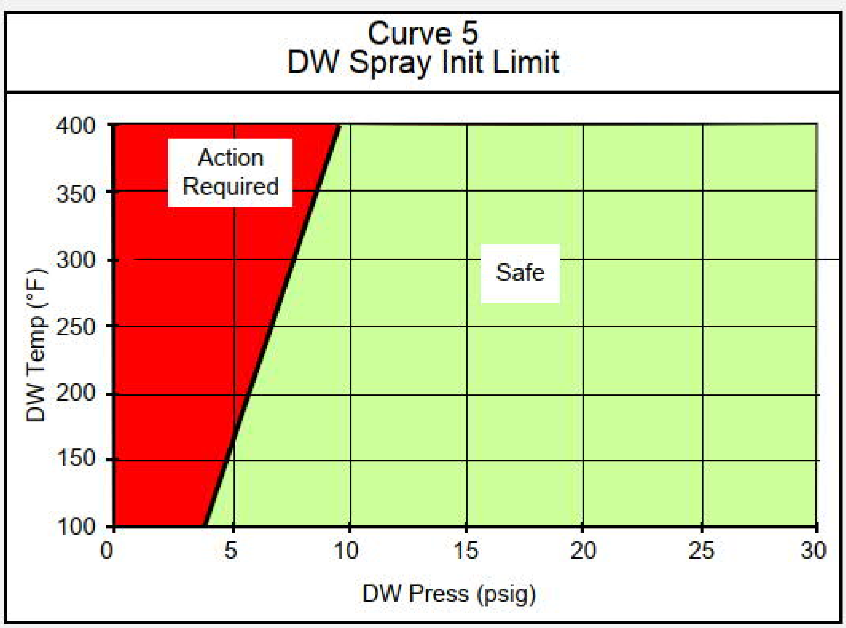
Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 5 of 12

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
		PC/P-5  **INITIATE** Suppression Chamber Sprays ➢ Use only source NOT required to assure Adequate Core Cooling by continuous injection (2-EOI-Appendix-17C) **IF**: Needed to augment Suppression Chamber Sprays — **THEN**: **NO ACTION REQUIRED**  Operating pumps with suction from the suppression pool above the NPSH Limit (Curve 1, 2, 9 or 10) or with suppression pool water level below 10 ft (Vortex limit) may cause equipment damage  Reducing PC press will reduce the available NPSH for pumps taking suction from the suppr pl
	NUSO	
	NRC	**2-EOI-Appendix-17B, RHR System Operation Drywell Sprays – See Attachment 1, starting on page 47.** **2-EOI-Appendix-17C, RHR System Operation Suppression Chamber Sprays – See Attachment 2, starting on page 45.**

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 8	Page 6 of 12

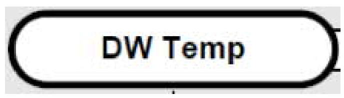
Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	NUSO	**IF** Suppression Pool Water Level is below 19 feet AND Drywell Temperature is in the safe area of Curve 5 **THEN** 1. **SHUT DOWN** Recirc Pumps 2. **SHUT DOWN** Drywell Blowers 3. **INITIATE** Drywell Sprays ➢ Use only sources NOT required to assure Adequate Core Cooling by continuous inj (APPX17B) **IF**: Needed to augment Drywell Sprays — **THEN**: **NO ACTION REQUIRED** PC/P-7 

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 7 of 12

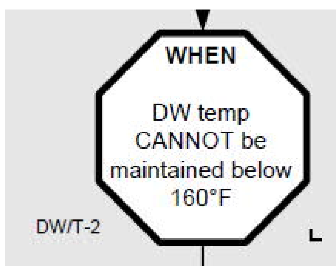
Event Description: Steam Leak in the Drywell

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NRC</td><td>Candidate may elect to first Spray the Drywell based on the Drywell Temp leg of 2-EOI-2, Primary Containment Control.</td></tr>
<tr><td></td><td>NUSO</td><td>
2-EOI-2, Primary Containment Control



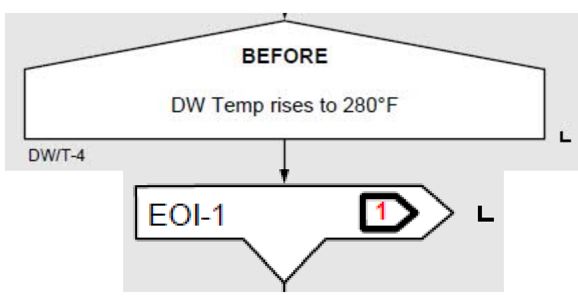
DW/T-1

MONITOR and CONTROL Drywell Temperature below 160ºF using available Drywell Cooling



DW/T-3

OPERATE all available Drywell Cooling

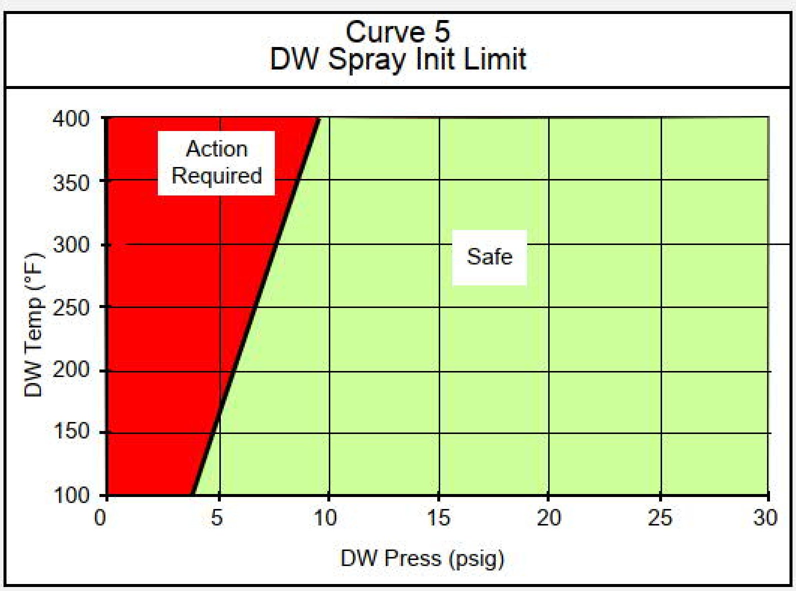


</td></tr>
<tr><td></td><td>NUSO</td><td>
DW/T-5

<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Pressure drops to 0 PSIG</td><td>STOP Drywell Sprays (2-EOI-Appendix-17B)</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 8 of 12

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	NUSO	DW/T-6 **IF** Suppression Pool Water Level is below 19 feet AND Drywell Temperature is in the safe area of Curve 5 **THEN** 1. **SHUT DOWN** Recirc Pumps 2. **SHUT DOWN** Drywell Blowers 3. **INITIATE** Drywell Sprays ➢ Use only sources NOT required to assure Adequate Core Cooling by continuous inj (APPX17B) **IF**: Needed to augment Drywell Sprays — **THEN**: **NO ACTION REQUIRED** 

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 9 of 12

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	BOP	2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell **NOTE** This procedure covers possible multiple symptoms of a problem within Primary Containment. Any or all of the symptoms may exist. The SRO will direct actions based on symptoms and experience. 4.1 Immediate Actions: None 4.2 Subsequent Actions [1] **IF** any EOI entry condition is met, **THEN ENTER** appropriate EOI(s). (Otherwise N/A) [2] **IF** Drywell Pressure is High, **THEN PERFORM** the following: (Otherwise N/A) [2.1] **CHECK** Drywell Pressure using multiple indications. [2.2] **IF** Drywell pressure rising rate indicates Reactor Scram at 2.45 psig is imminent, **THEN REDUCE** Reactor Power via Recirc Flow to minimize the impact of a SCRAM from high power. (Otherwise N/A) [2.3] **CHECK** Drywell pressure using multiple indications. [2.4] **ALIGN** and START additional Drywell coolers and fans as necessary. **REFER** TO 2-OI-64, Primary Containment System. **CAUTION** Stack release rates exceeding 1.4x10^{7}µci/sec, or a 0-SI-4.8.B.1.a.1, Airborne Effluent Release Rate, release fraction above one will result in ODCM release limits being exceeded. [2.5] **VENT** Drywell as follows: [2.5.1] **CLOSE** 2-FCV-64-34, SUPPRESSION CHAMBER INBOARD ISOLATION VALVE (Panel 2-9-3). [2.5.2] **ENSURE** OPEN 2-FCV-64-31, DRYWELL INBOARD ISOLATION VALVE (Panel 2-9-3).

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 10 of 12

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	BOP	[2.5.3] **ENSURE** 2-FIC-84-20 is in AUTO and SET at 100 scfm (Panel 2-9-55). [2.5.4] **ENSURE** RUNNING a Standby Gas Treatment Fan STGTS TRAIN C(A)(B) (Panel 2-9-25). [2.5.5] **IF** required, **THEN REQUEST** Unit 1 Operator to START Standby Gas Treatment Fans A or B. (Otherwise N/A) **CAUTION** If 2-FCV-84-20 closes after 2-HS-64-35 is opened, the reason for valve closure must be cleared and 2-HS-64-35 must be returned to OPEN in order for 2-FCV-84-20 to re-open. [2.5.6] **N/A** [2.5.7] **PLACE** 2-FCV-84-20 2-HS-64-35, CONTROL DRYWELL/SUPPRESSION CHAMBER VENT, in OPEN (Panel 2-9-3). [2.5.8] **MONITOR** stack release rates to prevent exceeding ODCM limits. [2.5.9] **WHEN** Drywell pressure has been reduced as required, **THEN STOP** SGT Train(s). [2.5.10] **ENSURE** 2-HS-64-35, in AUTO and 2-FCV-84-20 CLOSED (Panel 2-9-3). [2.5.11] **OPEN** 2-FCV-64-34, SUPPRESSION CHAMBER INBOARD ISOLATION VALVE (Panel 2-9-3). [2.5.12] **ENSURE** Drywell DP compressor operates correctly to maintain required Drywell to Suppression Chamber DP. [2.5.13] **RECORD** SGTS Train(s) run time in appropriate Control Room Reactor narrative log for transfer to 1-SR-2, Instrument Checks and Observations. [2.6] **N/A** [2.7] **ENSURE** CLOSED, N2 makeup valves to Drywell and Suppression Chamber. [2.8] **CHECK** Suppression Chamber Pressure.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 11 of 12

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
		[2.9] **CHECK** Suppression Pool Water Level. [2.10] **CHECK** Suppression Pool temp for indication of a leaking or stuck open relief valve. [2.11] **N/A** [2.12] **N/A** [2.13] **N/A** **NOTE** Based on recent On-Shift Analysis from the License Amendment, chemistry sampling may be delayed following plant events. In the case of a REP event, Chemistry will be required to assist with dose assessments. Dose assessments will have precedence over sampling actions in REP events.
	BOP	[2.14] NOTIFY Chemistry to sample Drywell atmosphere for radioactivity. [2.15] NOTIFY Radwaste that fluids being discharged from Drywell may be highly radioactive. [3] **IF** Drywell Temperature is High, **THEN PERFORM** the following: (Otherwise N/A) [3.1] **IF** Reactor is at power AND Drywell cooling is lost and can NOT be immediately restored, **THEN PERFORM** the following: (Otherwise N/A) [3.1.1] **IF** Core Flow is above 60%, **THEN REDUCE** Core Flow to between 50-60%. [3.1.2] **MANUALLY SCRAM** the Reactor and **REFER TO** 2-AOI-100-1, Reactor SCRAM. [3.1.3] **INITIATE** a 90ºF/hr cooldown rate. **REFER TO** 2-AOI-100-1, Reactor SCRAM.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 6 Page 12 of 12

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	BOP	[3.2] **CHECK** Drywell Temperature using multiple indications. [3.3] **ALIGN** and **START** additional Drywell coolers and fans as necessary. **REFER TO** 2-OI-64, Primary Containment System. [3.4] **VENT** Drywell. **REFER TO** Section 4.2[2.5]. [3.5] **N/A**
	NRC	**Event 9, Drywell Spray Failure, is inserted during Simulator Setup. No action is required by the Driver to insert Event 9.**
	NRC	**When the crew has sprayed Containment and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using either high or low pressure systems, end of Scenario.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 1 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	NRC	**The first loop of Drywell Sprays will fail when the crew attempts to spray Containment to reduce Containment Pressure and Temperature, requiring action to use the other loop of Drywell Spray.**
	NUSO	Directs BOP to perform 2-EOI-Appendix-17C, RHR System Operation Suppression Chamber Sprays.
	BOP	2-EOI-Appendix-17C, RHR System Operation Suppression Chamber Sprays [1] **BEFORE** Suppression Chamber pressure drops below 0 psig, **CONTINUE** in this procedure at Step 1.0[6]. [2] **IF** Adequate Core Cooling is assured **OR** Directed to spray the Suppression Chamber irrespective of Adequate Core Cooling, **THEN BYPASS** LPCI injection valve auto interlock as necessary: • **PLACE** 2-HS-74-155A, LPCI SYS I OUTBD INJECTION VALVE BYPASS SELECT in BYPASS • **PLACE** 2-HS-74-155B, LPCI SYS II OUTBD INJECTION VALVE BYPASS SEL in BYPASS [3] **N/A** [4] **N/A** [5] **INITIATE** Suppression Chamber Sprays as follows: [5.1] **ENSURE** at least one RHRSW pump supplying each EECW header. [5.2] **IF** EITHER of the following exists: • LPCI Initiation signal is NOT present, OR • Directed by SRO, **THEN PLACE** keylock switch 2-XS-74-122 (130), RHR SYSTEM I (II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE [5.3] MOMENTARILY PLACE 2-XS-74-121(129), RHR SYS I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, switch in SELECT.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 2 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	BOP	[5.4] **IF** 2-FCV-74-53 (67), RHR SYSTEM I (II) LPCI INBOARD INJECTION VALVE, is OPEN, **THEN ENSURE CLOSED** 2-FCV-74-52 (66), RHR SYSTEM I (II) LPCI OUTBOARD INJECTION VALVE. [5.5] **ENSURE** OPERATING the desired RHR System I(II) pump(s) for Suppression Chamber Spray [5.6] **ENSURE** OPEN 2-FCV-74-57(71), RHR SYSTEM I (II) SUPPRESSION CHAMBER/POOL ISOL VLV. [5.7] **OPEN** 2-FCV-74-58 (72), RHR SYSTEM I (II) SUPPRESSION CHAMBER SPRAY VALVE. [5.8] **IF** RHR System I (II) is operating ONLY in Suppression Chamber Spray mode, **THEN CONTINUE** in this procedure at Step 1.0[5.11]. [5.9] **ENSURE CLOSED** 2-FCV-74-7 (30), RHR SYSTEM I (II) MINIMUM FLOW VALVE. [5.10] **RAISE** System flow by placing the second RHR System I (II) pump in service as necessary. [5.11] **MONITOR** RHR Pump NPSH using Attachment 2. [5.12] **ENSURE** RHRSW pump supplying desired RHR Heat Exchanger(s). [5.13] **THROTTLE** the following in-service RHRSW outlet valves to obtain the required flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE (Required flow is 1350 to 4500 gpm for A1 pump) (Required flow is 1700 to 4500 gpm for A2 pump) • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE (Required flow is 1350 to 4500 gpm for B1 pump) (Required flow is 1700 to 4500 gpm for B2 pump) • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE (Required flow is 1350 to 4500 gpm for D1 pump) (Required flow is 1700 to 4500 gpm for D2 pump)

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 3 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	BOP	[5.14] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers
	Driver	**If contacted as Chemistry, acknowledge any reports or direction given.**
	BOP	[6] **N/A**
	NRC	**The first loop of Drywell Sprays will fail when the crew attempts to spray Containment to reduce Containment Pressure and Temperature, requiring action to use the other loop of Drywell Spray. (Select Logic).**
	CREW	**Critical Task:** **Initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit Curve and before Drywell Temperature rises to** **280 °F.** **Critical Task Failure Criteria: The operating crew fails to spray the Drywell before Drywell Temperature reaches 280 ºF.**
	NUSO	Directs BOP to perform 3-EOI-Appendix-17B RHR System Operation Drywell Sprays.
	BOP	2-EOI-Appendix-17B RHR System Operation Drywell Sprays [1] **BEFORE** Drywell Pressure drops below 0 psig **CONTINUE** in this procedure at Step 1.0[7]. [2] **IF** Adequate Core Cooling is assured **OR** directed to spray the Drywell irrespective of Adequate Core Cooling, **THEN BYPASS** LPCI injection valve auto open signal as necessary: • **PLACE** 2-HS-74-155A, LPCI SYSTEM I OUTBOARD INJECTION VALVEBYPASS SELECT IN BYPASS • **PLACE** 2-HS-74-155B, LPCI SYSTEM II OUTBOARD INJECTION VALVE BYPASS SELECT IN BYPASS [3] **ENSURE** Recirc Pumps and Drywell Blowers are shutdown. [4] **N/A** [5] **N/A**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 4 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	BOP	[6] **INITIATE** Drywell Sprays as follows: [6.1] **ENSURE** at least one RHRSW Pump supplying each EECW header. [6.2] **IF** EITHER of the following exists: • LPCI Initiation signal is NOT present, **OR** • Directed by SRO, **THEN PLACE** keylock switch 2-XS-74-122(130), RHR SYSTEM I (II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE [6.3] **MOMENTARILY PLACE** 2-XS-74-121 (129), RHR SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, switch in SELECT. [6.4] **IF** 2-FCV-74-53 (67), RHR SYSTEM I (II) LPCI INBOARD INJECTION VALVE, is OPEN, **THEN ENSURE CLOSED** 2-FCV-74-52 (66), RHR SYSTEM I (II) LPCI OUTBOARD INJECTION VALVE. [6.5] **ENSURE OPERATING** the desired System I (II) RHR Pump(s) for Drywell Spray. [6.6] **OPEN** the following valves: • 2-FCV-74-60(74), RHR SYSTEM I(II) DW SPRAY OUTBOARD VALVE • 2-FCV-74-61 (75), RHR SYSTEM I (II) DW SPRAY INBOARD VALVE [6.7] **ENSURE CLOSED** 2-FCV-074-0007 (0030), RHR SYSTEM I (II) MINIMUM FLOW VALVE. [6.8] **IF** Additional Drywell Spray flow is necessary, **THEN PLACE** the second System I (II) RHR Pump in service. [6.9] **MONITOR** RHR Pump NPSH using Attachment 2. [6.10] **ENSURE** RHRSW Pump supplying desired RHR Heat Exchanger(s).

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 5 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	BOP	[6.11] **THROTTLE** the following in-service RHRSW Outlet Valves to obtain the required RHRSW Flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE (Required flow is 1350 to 4500 gpm for B1 pump) (Required flow is 1700 to 4500 gpm for B2 pump) • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) [6.12] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	Driver	**If contacted as Chemistry, acknowledge any reports or direction given.**
	BOP	[7] **WHEN** EITHER of the following exists: • **BEFORE** Drywell Pressure drops below 0 psig, OR • Directed by SRO to stop Drywell Sprays, **THEN STOP** Drywell Sprays as follows: [7.1] **ENSURE CLOSED** the following valves: • 2-FCV-74-100(101), RHR SYSTEM I U-1(SYS II U3) DISCH XTIE • 2-FCV-74-60(74), RHR SYSTEM I(II) DRYWELL SPRAY OUTBOARD VALVE • 2-FCV-74-61(75), RHR SYSTEM I(II) DRYWELL SPRAY INBOARD VALVE [7.2] **ENSURE OPEN** 2-FCV-74-7(30), RHR SYSTEM I (II) MINIMUM FLOW VALVE. [7.3] **IF** RHR operation is desired in ANY other mode, **THEN EXIT** this EOI Appendix. [7.4] **STOP** RHR Pumps 2A and 2C (2B and 2D).

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 6 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	NRC	**End of Event 9. Once the crew has sprayed Containment and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using either high or low pressure systems, end of Scenario.**

Scenario Setup
UNIT 2

IC	28
Exam IC	279

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-2	109	AOI-100-1	116	EOI-1	18
OI-3	161	ARP-3A	55	EOI-2	16
OI-65	55	ARP-3B	38	APPX-17B	17
OI-68	159	ARP-5A	60	APPX-17C	16
OI-85	147	ARP-5B	31	TS 3.3.6.2	253
OI-90	38	ARP-6C	25	TS 3.3.7.1	253
AOI-3-1	23	ARP-7A	35	TS 3.4.5	253
AOI-64-2D	37	ARP-8A	43	TS 3.6.4.3	290
AOI-64-1	27	OPDP-1	46	TRM 3.3.10	100

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Hang Protected Equipment Tags on the following: RHR Loop I and II, Core Spray Loop II, ‘C’ and ‘D’ EDG, HPCI, ‘C’ and ‘D’ 4KV Shutdown Boards, and 2A 250V RMOV Board
Schedule Files(s):	2104 NRC Scenario 4 UNIT 2.sch
Event Files(s):	2104 NRC Scenario 4 UNIT 2.evt

Schedule File – 2104 NRC Scenario 4 UNIT 2.sch

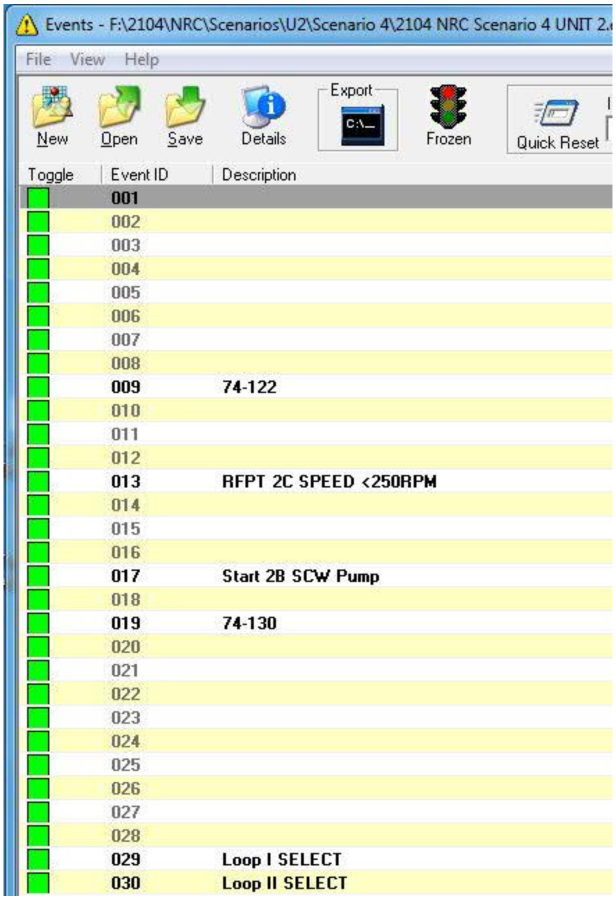
Event	Action	Description
	2104 NRC Scenario 4 UNIT 2.evt	Event File
1	Insert remote FW06 to START	CONDENSATE BOOSTER PUMP C AUX OIL PUMP
2	Insert malfunction RD22 to 100.00000 on	CRD FLOW TRANSMITTER FT-85-11 FAILURE
3	Insert malfunction FW33C to 85.00000 in 60	RFPT 2C AXIAL THRUST CH A FAILURE (XE-3-1062-1)
3	Insert malfunction FW33D to 88.00000 in 60	RFPT 2C AXIAL THRUST CH B FAILURE (XE-3-1062-2)
13	Delete malfunction FW33C	RFPT 2C AXIAL THRUST CH A FAILURE (XE-3-1062-1)
13	Delete malfunction FW33D	RFPT 2C AXIAL THRUST CH B FAILURE (XE-3-1062-2)

Schedule File – 2104 NRC Scenario 4 UNIT 2.sch

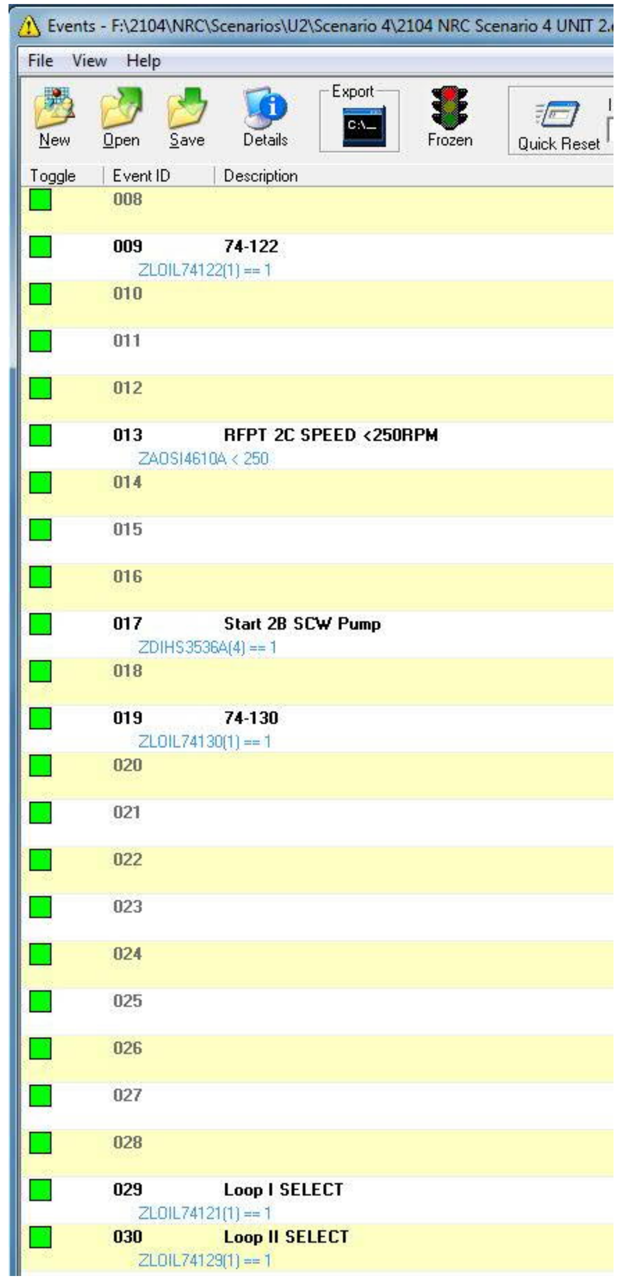
Event	Action	Description
5	Insert malfunction RM08A to 1000000.00000	REFUEL ZONE RAD CH 0A MONITOR FAILURE (RM-90-140/142)
5	Insert malfunction RM08B to 1000000.00000	REFUEL ZONE RAD CH 0B MONITOR FAILURE (RM-90-140/142)
	Insert malfunction PC01C	SBGT SYSTEM C AUTO START FAILURE (CONTACT 10 OF HS 65-69A)
7	Insert override HS-35-35A to STOP	GEN STATOR CLG WATER PUMP 2A
	Insert malfunction PMP-35-36 to FAIL_CCOIL	52_BREAKER GEN STATOR COOLING WATER PUMP B
17	Delete malfunction PMP-35-36	52_BREAKER GEN STATOR COOLING WATER PUMP B
8	Insert malfunction TH33A to 1.50000 in 600	MAIN STEAM LINE A BREAK IN CONTAINMENT (DRYWELL)
	Insert override XS-74-121 to NAR	RHR SYS I CTMT SPRAY/CLG VLV SELECT
	Insert override XS-74-129 to NAR	RHR SYS II CTMT SPRAY CLG/VLV SELECT
9	Delete override XS-74-129	RHR SYS II CTMT SPRAY CLG/VLV SELECT
19	Delete override XS-74-121	RHR SYS I CTMT SPRAY/CLG VLV SELECT
29	Insert override XS-74-129 to NAR	RHR SYS II CTMT SPRAY CLG/VLV SELECT
30	Insert override XS-74-121 to NAR	RHR SYS I CTMT SPRAY/CLG VLV SELECT

Event File

List



Details



UNIT 2 SHIFT TURNOVER MEETING			Today
MODE 1	**DAYS ON LINE** 35	**Total Drywell Leakage (gpm)** 1.55	**Protected Equipment**
	PRA (EOOS) -GREEN		RHR Loop I and II
			Core Spray Loop II
Rx Power 95%	500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (gpm)** 0.11	'C' and 'D' EDG
			HPCI
MWe 1080	**Last breaker closure** 4/12/21 4:31	**Equipment Drain (gpm)** 1.44	4KV Shutdown Boards 'C', 'D'
			250V RMOV Board 2A

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

Core Spray Loop I Outage – day 1 of 7 day LCO IAW Tech Spec 3.5.1.A

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

APRM 1 Critical Fault – APRM 1 is bypassed (Information Only LCO 3.3.1.1)

Core Spray Loop I MOVATS Testing in progress

Reactor Power lowered to 95% to secure 2C Condensate Booster Pump

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Reduce Reactor Power and remove 2C Condensate Booster Pump from service for maintenance

Reactor Engineer will brief the return to 100% power later in the shift

OPERATOR WORK AROUNDS **OWAs - 1*** **Burdens - 0** **Challenges - 7**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 1 Page 1 of 3

Event Description: Remove 3C Condensate Booster Pump (CBP) from Service for Maintenance

Time	Position	Applicant's Actions or Behavior
	Driver	**Prior to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not proceed to Event 1, Remove 3C Condensate Booster Pump (CBP) from Service for Maintenance, after assuming the shift request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to secure 3C Condensate Booster Pump (CBP).**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to secure 3C Condensate Booster Pump.**
	NUSO	Directs the Balance of Plant Operator (BOP) to remove 3C Condensate Booster Pumps from service in accordance with 3-OI-2 Condensate System,Section 8.18, Removing a Condensate Booster Pump from service at High Power

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 1 Page 2 of 3

Event Description: Remove 3C Condensate Booster Pump (CBP) from Service for Maintenance

Time	Position	Applicant's Actions or Behavior
	BOP	3-OI-2, Condensate System Section 8.19, Removing a Condensate Booster Pump from service at High Power **NOTES** 1) There is adequate FLOW / NPSH to maintain 100% power when one Condensate Booster Pumps (CBP) is taken out of service. 2) During operation with only two CBP in service (3-2-3): • Condensate flow above 14.1 MLBM/HR (Approx 87.5% power) can result in the CBP motors operating above their rated HP but within rated service factor. • While operating within rated service factor, the CBP motor winding temperatures are not to exceed 266 degrees F and the motor amps are not to exceed 427.8 amps. 3) There is adequate FLOW / NPSH to maintain 100% power when one Reactor Feedwater Pump and one CBP. (3-2-2) are taken out of service and the following conditions apply: • Three Condensate Pumps are in service • The Reactor Feedwater Pump is removed from service and secured (no flow thru the minimum flow valve) prior to removing the CBP from service • Note 2 above is applicable
	BOP	[1] **REVIEW** Precautions and Limitations in Section 3.4. **Completed during pre-shift brief.** [2] **IF** Time permits, **THEN REVIEW** Drawing 3-47E800-3 Notes regarding operational guidelines for Condensate and Feedwater system. (Otherwise N/A) [3] **ENSURE** Reactor Power is ≤ 95%. If testing in accordance with 0-TI-704, (Condensate Pump, Condensate Booster Pump and Feedwater Pump Testing), then adjust Reactor Power as required to support 0-TI-704 prior to stopping a Condensate Booster Pump. (Ref P&L 3.1 B)

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 1 Page 3 of 3

Event Description: Remove 'C' Condensate Booster Pump (CBP) from Service for Maintenance

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **ENSURE** hydrogen injection is secured to the Condensate Booster Pump to be stopped. **REFER TO** 3-OI-4, Hydrogen Water Chemistry System.
	Driver	**If directed as the Turbine Building AUO to perform 3-OI-4, Hydrogen Water Chemistry System, Section 8.10 [2.3] Shut down Hydrogen Injection to 3C Condensate Booster Pump, acknowledge direction. Inform BOP that Hydrogen Water Injection is secured to 3C Condensate Booster Pump.**
	BOP	[5] **N/A** [6] **WHEN** directed by the Unit 3 Unit SRO, **THEN STOP** CONDENSATE BOOSTER PUMP using one of the following: • CONDENSATE BOOSTER PUMP 3C, 3-HS-2-68A [7] **N/A**
	NRC	**End of Event 1. Request that the Driver insert Event 2, Control Rod Drive (CRD) Flow Controller Fails High.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 2 Page 1 of 2

Event Description: Control Rod Drive (CRD) Flow Controller Fails High

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, insert Event 2, Control Rod Drive (CRD) Flow Controller Fails High.**
	OATC	Acknowledges and reports the following alarm to the Nuclear Unit Senior Operator (NUSO): • CRD ACCUMULATOR CHARGING WATER HEADER PRESSURE HIGH, 3-9-5A, Window 10
	NUSO	Acknowledges the Operator at the Controls' (OATC) alarm report. Directs the OATC to respond in accordance with applicable Alarm Response Procedures and subsequently 3-OI-85, Control Rod Drive System.
	OATC	Alarm Response Procedure, 3-ARP-9-5A CRD ACCUM CHG WTR HDR PRESS HIGH, 3-9-5A, Window 10 A. **CHECK** pressure high on 3-PI-85-13A, CRD ACCUMULATOR CHARGING WATER HEADER on Panel 3-9-5. B. **CHECK** 3-FCV-85-11A (B), CRD LINE A(B) FLOW CONTROL VALVE, in service.
	NRC	**The crew may attempt to switch Flow Control Valves. However, as long as the Flow Controller is failed High, neither set of Flow Control Valves will operate in automatic. 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, must be placed in MANUAL.**
	OATC	C. **IF** in-service controller has failed, **THEN REFER TO** 3-OI-85, Control Rod Drive System. D. **N/A**
	OATC	Determines that the CRD Flow Controller has failed High, causing 3-FCV-85-11A, CRD LINE A FLOW CONTROL VALVE to CLOSE. In accordance with OPDP-1, Conduct of Operations, Section 3.5.3, Manual Control of Automatic Systems, takes manual control of 3-FIC-85-11, CRD SYSTEM FLOW CONTROL to restore CRD Parameters back to normal.

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 2	Page 2 of 2

Event Description: Control Rod Drive (CRD) Flow Controller Fails High

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-85, Control Rod Drive System Section 8.33, AUTOMATIC/MANUAL operation of 3-FIC-85-11 [1] **REVIEW** all Precautions and Limitations in Section 3.6. [2] **IF** transferring 3-FIC-85-11 from AUTO to MANUAL **THEN**: [2.1] **PLACE** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL in **BALANCE.** [2.2] **BALANCE** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL by turning Manual Control Pot inside Control Selector Wheel until red deviation pointer is in the Green Band. [2.3] **PLACE** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL in MANUAL. [2.4] **ADJUST** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL manual potentiometer to establish the desired system flow. Refer to Section 5.1 or 6.10.
	NRC	**End of Event 2. Request that the Driver insert Event 3, Reactor Feedwater Pump (RFPT) Vibration Alarm.**

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 3	Page 1 of 7

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3, Reactor Feedwater Pump (RFPT) Vibration Alarm.**
	OATC	Acknowledges and reports the following alarms: • RFPT C ABNORMAL, 3-9-6C, Window 15 • RFPT VIBRATION OR AXIAL POSITION HIGH-HIGH, 3-9-6C, Window 17
	NUSO	Acknowledges the Operator at the Controls' (OATC) alarm report. Directs the OATC to respond in accordance with applicable Alarm Response Procedures.
	NRC	**Given the degrading condition of 3C RFPT, the crew may elect one and/or both of the following paths:** **(1) Respond per 3-9-6C, Window 17 then remove 3C RFPT from service in accordance with 3-OI-3, Reactor Feedwater System (see page 8)** **(2) Respond per 3-9-6C, Window 17 then conservatively trip 3C RFPT in accordance with 3-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low (see page 12)**
	OATC	Alarm Response Procedure, 3-ARP-9-6C RFPT C ABNORMAL, Window 15 Operator Action: A. **CHECK** other RFP alarms on Panel 3-9-6 to determine problem area. B. **REFER TO** appropriate alarm response procedure. C. **IF** no other annunciator on Panel 3-9-6 is in alarm, **THEN PERFORM** an alarm summary on alarm types.
	OATC	Alarm Response Procedure, 3-ARP-9-6C RFPT VIB OR AXIAL POSITION HIGH-HIGH, 3-9-6C, Window 17 Operator Action: A. **CHECK** RFPT/RFP vibration readings on 3-XR-3-177 on Panel 3-9-6 AND RFPT and RFP Vibration display (RFPTV) on ICS.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 2 of 7

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td></td><td>B. DISPATCH personnel to Panel 3-LPNL-025-0673, Vibration Monitoring Panel, located outside of RFP Room 3A, EL 617', to PERFORM the following: REPORT vibration data for affected RFPT/RFP.
• REPORT all alarm/alert conditions on panel.
• Advise the Unit Operator of any changes in vibration data.
• Locally reset and request UO to reset control room annunciators.</td></tr>
<tr><td></td><td>Driver</td><td>If directed as the Turbine Building AUO to REPORT 3C RFPT vibration data, all alarms/conditions for Vibration Monitoring Panel, acknowledge direction. Wait 2 minutes, inform the OATC/BOP that the alarm is valid on the 3C RFPT. Vibration can be felt in the 3C RFPT room and ALERT lights are illuminated on the Vibration Monitoring Panel.</td></tr>
<tr><td></td><td>OATC</td><td>C. ADJUST load on pump if necessary, by: Lowering Reactor Power OR Reducing RFPT speed on affected pump(s)</td></tr>
<tr><td></td><td>NRC</td><td>The crew will verify that Condensate system flow measured on CONDENSATE, 3-XR-002-0026, is less than 12 Mlbm/hr (75% Reactor power) prior to removing a 3C Reactor Feedwater Pump from service. See Event 4 below.</td></tr>
<tr><td></td><td>OATC</td><td>D. IF a sustained vibration exceeding the DANGER setpoints
(REFER TO setpoints below) is confirmed on both pump inboard and outboard bearings or any turbine bearing, THEN REMOVE the RFPT from Service. REFER TO 3-OI-3.
E. IF alarm does NOT reset, THEN REMOVE pump from service. REFER TO 3-OI-3.
<table>
<tr><td><u>RFP 3C</u></td><td></td><td></td></tr>
<tr><td>Turbine Axial Thrust</td><td>3-XM-3-1062-1</td><td>15 Mils</td></tr>
<tr><td></td><td>3-XM-3-1062-2</td><td>15 Mils</td></tr>
<tr><td>Turbine Outbd Bearing</td><td>3-XM-3-199A1</td><td>4.5 Mils</td></tr>
<tr><td></td><td>3-XM-3-199A2</td><td>4.5 Mils</td></tr>
<tr><td>Turbine Inbd Bearing</td><td>3-XM-3-199B1</td><td>4.5 Mils</td></tr>
<tr><td></td><td>3-XM-3-199B2</td><td>4.5 Mils</td></tr>
<tr><td>Pump Inbd Bearing</td><td>3-XM-3-0169A1</td><td>4.5 Mils</td></tr>
<tr><td></td><td>3-XM-3-0169A2</td><td>4.5 Mils</td></tr>
<tr><td>Pump Outbd Bearing</td><td>3-XM-3-0170A1</td><td>4.5 Mils</td></tr>
<tr><td></td><td>3-XM-3-0170A2</td><td>4.5 Mils</td></tr>
<tr><td>Pump Axial Thrust</td><td>3-XM-3-0171A1</td><td>20.0 Mils</td></tr>
<tr><td></td><td>3-XM-3-0171A1</td><td>20.0 Mils</td></tr>
</table></td></tr>
<tr><td></td><td>NRC</td><td>3C RFPT Axial Thrust vibration will ramp up to 15 mils to exceed DANGER setpoint on 3-XR-3-177 on Panel 3-9-6.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 3 of 7

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-3, Reactor Feedwater System Section 7.1, RFP/RFPT Shutdown CAUTIONS 1) There is NOT adequate flow/NPSH to maintain 100% (3952) when one Condensate Pump is out of service. To maintain adequate NPSH available to the Condensate Booster Pumps, Condensate flow is limited to 14.1 Mlbm/hr. 2) At 100% (3952), Reactor power should be verified less than or equal to 95% prior to starting or stopping a Condensate Booster Pump or Reactor Feedwater Pump. If testing in accordance with 0-TI-704, (Condensate Pump, Condensate Booster Pump and Feedwater Pump Testing), then adjust Reactor power as required to support 0-TI-704 prior to stopping a Condensate Booster Pump or Reactor Feedwater Pump. 3) FAILURE to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow (between 2 x 106 and 3 x 106 lbm/hr) may result in SJAE poor performance. The SJAE's will NOT trip on Condensate system low pressure. 4) Changes in Condensate System flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190. Refer to P&L 3.0L as necessary. 5) When isolating the Reactor Feedwater Pump(s) for maintenance, the associated injection water should also be isolated to prevent high seal differential pressure and allow the RFW Pump shafts to rotate freely. (BFNPER123395) 6) When a Reactor Feed Pump is isolated (suction, discharge and minimum flow valve closed) with injection water aligned to the pump, there is a potential of rising pump casing pressure and seal water leakoff flows reaching the point where seal water drains are overcome and seal water is forced into the oil system through the bearing housings. Therefore, the time that a RFP is isolated with injection water aligned to the pump should be minimized. 7) When operating with one Condensate and/or Condensate Booster Pump out of service, verify that Condensate system flow measured on CONDENSATE, 3-XR-002-0026, is less than 12 Mlbm/hr (75% Reactor power) prior to removing a Reactor Feedwater Pump from service.

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 3	Page 4 of 7

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	[1] **REFER TO** Section 3.0, and **REVIEW** Precautions and Limitations. [2] **N/A** **NOTE** It may be necessary to switch to SINGLE ELEMENT mode from THREE ELEMENT mode earlier than recommended if Feedwater control becomes unstable. [3] **N/A** [4] **ENSURE** in AUTO: • RFPT 3C TURNING GEAR MOTOR, 3-HS-3-152A **NOTES** 1) Column 1 (PV) on individual RFPT Speed Control Panel Display Stations (PDS) displays actual pump speed and is NOT controlled in any mode. 2) When selected, then Column 2 (SP) on individual RFPT Speed Control PDS displays pump flow bias and is changed with Ramp RAISE/Ramp LOWER pushbuttons with controller in AUTO (blue light illuminated). 3) When selected, then Column 3 (CO) on individual RFPT Speed Control PDS displays RFPT speed demand and is changed with Ramp RAISE/Ramp LOWER pushbuttons with controller in MANUAL (amber light illuminated). 4) Attachment 2 has additional information on RFPT Speed Control PDS. [5] **LOWER** speed of RFPT/RFP being removed from service by performing either one of the following: • **IF** using individual RFPT Manual Governor switch, **THEN GO TO** Step 7.1[6]. • **IF** using individual RFPT Speed Control PDS in MANUAL, **THEN GO TO** Step 7.1[7].

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 5 of 7

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	[6] **LOWER** speed of RFPT using individual RFPT SPEED CONT RAISE/LOWER switch as follows (Panel 3-9-5): • 3-HS-46-10A, RFPT 3C SPEED CONT RAISE/LOWER [6.1] **DEPRESS** RFPT Speed Control Raise/Lower switch to MANUAL GOVERNOR. [6.2] **ENSURE** illuminated amber light at switch. [6.3] **SLOWLY LOWER** RFPT speed by placing RFPT Speed Control switch in RAISE or LOWER positions as necessary. [6.4] **IF** this is NOT the last operating RFP, **THEN OBSERVE** rise in speed of any operating RFPT in auto as RFW Control System maintains Reactor Water Level. [7] **LOWER** speed of RFPT using individual RFPT SPEED CONTROL PDS as follows (Panel 3-9-5): • RFPT 3C SPEED CONTROL (PDS), 3-SIC-46-10. [7.1] **PLACE** PDS in MANUAL (amber Light illuminated) and ENSURE Column 3 (CO) selected. [7.2] **SLOWLY LOWER** RFPT speed using Ramp RAISE/Ramp LOWER pushbuttons as necessary. [7.3] **IF** this is NOT the last operating RFP, **THEN OBSERVE** rise in speed of any operating RFPT in auto as RFW Control System maintains Reactor Water Level. **CAUTION** RFP Discharge Check Valve failure may be experienced while removing RFP from service. [8] **N/A** [9] **CONTINUE** to slowly lower RFPT speed to minimum speed setting (approximately 600 rpm) using the manual governor in step 7.1[6] or the Ramp RAISE / Ramp LOWER pushbuttons in step 7.1[7] as necessary.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 3 Page 6 of 7

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	**NOTES** 1) One RFPT may be allowed to remain as a running standby pump at minimum speed setting (approximately 600 rpm). 2) With Reactor Feed Pump running at ~600 RPM, adjusting CNDS FLOW CONTROL SHORT CYCLE, 3-FIC-2-29, will supply vessel inventory as needed by raising and lowering header pressure. Typically, a Feed Pump running at ~600 RPM will build 20 to 25 psig across the pump. 3) RFW START-UP LEVEL CONTROL, 3-LIC-3-53 does NOT respond linearly with Controller Demand. The design is to respond slowly to dampen level swings. 4) This evolution has better results when Condensate and Condensate Booster pumps are in a two and two configuration. It should be noted that a two and one configuration will establish a lower header pressure and more attention will be needed to ensure Condensate minimum flow requirements are met. Conversely with a three and three configuration, pressure is higher with higher potential to overfeed the vessel. [10] **IF** RFPT/RFP being removed from service is NOT the last operating RFPT, **THEN** GO TO Step 7.1[12]. [11] **N/A** [12] **WHEN** RFPT is ready to be shutdown, **THEN DEPRESS** RFPT TRIP, to trip RFPT being removed from service. (N/A if step 7.1[11] was performed) • RFPT 3C TRIP, 3-HS-3-176A **NOTES** 1) Check valve position indicator should NOT be relied upon for positive valve closure indication. 2) Step 7.1[14] is performed only if RFP Discharge Check Valve failure occurs.

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 3	Page 7 of 7

Event Description: Reactor Feedwater Pump (RFPT) Vibration Alarm

Time	Position	Applicant's Actions or Behavior
	OATC	[13] **ENSURE** CLOSED, RFP DISCH TESTABLE CHECK VLV, by one of the following: • RFP 3C DISCH TESTABLE CHECK VLV, 3-FCV-3-92. [13.1] Observe RFP discharge flow indicator. [13.2] Locally listening to check valve. [14] **N/A** **NOTE** Turning Gear motor will lockout if Turning Gear does NOT engage within five seconds of reaching zero speed. Lockout can be reset by placing control switch to OFF and pulling switch out (in OFF position). [15] **IF** RFPT is NOT rolling on minimum flow AND RFPT coasts down to zero speed, **THEN** ENSURE Turning Gear motor starts and engages. (Otherwise N/A) [16] **CLOSE** RFP Discharge Valve. • RFP 3C DISCHARGE VALVE, 3-FCV-3-5 [17] **PLACE** RFP MIN FLOW VALVE, in CLOSE. • RFP 3C MIN FLOW VALVE, 3-HS-3-6 [18] **ENSURE** Turning Gear engaged.
	NRC	**The crew may elect to conservatively trip 3C RFPT in accordance with 3-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low**
	OATC	3-AOI-1, Loss of Reactor Feedwater or Reactor Water Level High/Low Section 4.2 [12] **IF** a RFPT has tripped and is NOT required to maintain level, **THEN SECURE** tripped RFPT. **REFER TO** 3-OI-3, Reactor Feedwater System.
	NRC	**End of Event 3. Proceed to Event 4, Power Reduction for RFPT Shutdown.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 4 Page 1 of 2

Event Description: Power Reduction for RFPT Shutdown

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 4, Power Reduction for RFPT Shutdown, is entered by the crew. No action is required by the Driver to insert Event 4.**
	NUSO	Prior to removing 3C Reactor Feedwater Pump from service, directs the OATC to verify that Condensate system flow measured on 3-XR-002-0026, CONDENSATE, is less than 12 Mlbm/hr (75% Reactor Power) in accordance with 3-OI-3, Reactor Feedwater System, Section 7.1 RFP/RFPT Shutdown.
	OATC	Lowers Reactor Power to ensure that Condensate System Flow measured on CONDENSATE, 3-XR-002-0026, is less than 12 Mlbm/hr prior to removing 3C Reactor Feedwater Pump from service. May elect to use either the Master Recirc Speed Control or a Recirc System Runback (or a combination of both methods) in accordance with 3-OI-68, Reactor Recirculation System. 3-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow [1] **N/A** [2] **N/A** [3] **WHEN** desired to control Recirc Pumps 3A and/or 3B speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump speed 3A & 3B using the following push buttons as required: • 3-HS-96-31, RAISE SLOW • 3-HS-96-32, RAISE MEDIUM • 3-HS-96-33LOWER SLOW, • 3-HS-96-34, LOWER MEDIUM • 3-HS-96-35, LOWER FAST

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 4 Page 2 of 2

Event Description: Power Reduction for RFPT Shutdown

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-68, Reactor Recirculation System Section 8.12, Initiating Manual Runbacks [1] **IF** time permits, **THEN REVIEW** Precautions and Limitations. **REFER TO** Section 3.0. [2] **IF** desired to reduce Reactor Power to approximately 90%, **THEN PERFORM** the following: (Otherwise N/A) [2.1] **DEPRESS** 3-HS-68-42, RECIRC PUMPS UPPER POWER RUNBACK. [2.2] **CHECK** the following: • Push-button backlight blinks until setpoint is reached • Reactor Power lowers to approximately 90% [3] **IF** desired to reduce Reactor Power to approximately 66%, **THEN** PERFORM the following: (Otherwise N/A) [3.1] **DEPRESS** 3-HS-68-43, RECIRC PUMPS MID POWER RUNBACK. [3.2] **CHECK** the following: • Push-button backlight blinks until setpoint is reached • Reactor Power lowers to approximately 66% [4] **IF** desired to reduce Core Flow to approximately 58%, **THEN PERFORM** the following: (Otherwise N/A) [4.1] **DEPRESS** 3-HS-68-44, RECIRC PUMPS CORE FLOW RUNBACK. [4.2] **CHECK** the following: • Push-button backlight blinks until setpoint is reached • Core Flow lowers to approximately 58%
	NRC	**End of Event 4. Request that the driver insert Event 5, Refuel Zone Radiation Monitors Fail Upscale.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 1 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 2, Refuel Zone Radiation Monitors Fail Upscale.**
	BOP	Acknowledges and reports the following alarms: • REFUELING ZONE EXHAUST RADIATION HIGH, 3-9-3A, Window 34 • DRYWELL LEAK DETECTION RADIATION HIGH, 3-9-3D, Window 19 • DRYWELL/SUPPR CHAMBER H2O2 ANALYZER FAILURE, 3-9-7C, Window 22
	BOP	Alarm Response Procedure, 3-ARP-9-3A REFUELING ZONE EXHAUST RADIATION HIGH, Window 34 Operator Actions: A. **CHECK** alarm condition on the following: 1. REACTOR & REFUEL ZONE EXHAUST RADIATION recorder, 3-RR-90-144 points 3 and 4 on Panel 3-9-2. 2. RX & REFUEL ZONE EXH CH A RAD MON RTMR, 3-RM-90-140/142 on Panel 3-9-10. 3. RX & REFUEL ZONE EXH CH B RAD MON RTMR, 3-RM-90-141/143 on Panel 3-9-10. B. **N/A** C. **NOTIFY** Shift Manager, Unit 1 and Unit 2.
	Driver	**If contacted as the Shift Manager, Unit 1, or Unit 2 acknowledge any information given.**
	BOP	D. **N/A** E. **N/A** F. **ENTER** 3-EOI-3, Secondary Containment Control. G. **REFER TO** 3-AOI-64-2D, Group 6 Ventilation System Isolation and, for loss of power to NUMAC drawer, to 3-OI-90, Radiation Monitoring System. H. **N/A** I. **REFER TO** EPIP-1, Emergency Classification Procedure.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 2 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	NUSO	J. **REFER TO** Technical Specification Section 3.3.6.2, Secondary Containment Isolation instrumentation and 3.3.7.1, CREV System Instrumentation.
	NUSO	Enters 3-EOI-3, Secondary Containment Control.
	NRC	**3-AOI-64-2D, Group 6 Ventilation System Isolation is covered starting on page 17.**
	BOP	Alarm Response Procedure, 3-ARP-9-3D DRYWELL LEAK DETECTION RADIATION DOWNSCALE, Window 19 Operator Action: A. **DETERMINE** cause of alarm by performing the following: 1. **CHECK** AIR PARTICULATE MONITOR CONTROLLER, 3-MON-90-50 on Panel 3-9-2 for condition bringing in alarm 2. **N/A** B. **N/A** C. **N/A** D. **IF** corrective maintenance is required, **THEN NOTIFY** Chemistry to commence its sampling procedure. E. **REFER TO** Tech Specs 3.4.4, RCS Operational Leakage, 3.4.5, RCS Leakage Detection System, and TRM 3.3.10, Reactor Coolant Leakage Detection Instrumentation for CAM LCO requirements and **IMPLEMENT** appropriate TS/TRM actions as required. F. **N/A**
	Driver	**If notified as Chemistry to begin sampling, acknowledge the direction.** **If notified as the Work Control/Outside NUSO to investigate, acknowledge the direction.**
	NRC	**The NUSO may enter 3-EOI-3, Secondary Containment Control, based on the receipt of the REFUELING ZONE EXHAUST RADIATION HIGH Alarm (3-9-3A, Window 34).**
	Driver	**If contacted as the Shift Manager concerning 3-EOI-3, Secondary Containment Control, acknowledge any reports given and concur with any recommendations.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 3 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	BOP	3-AOI-64-2D, Group 6 Ventilation System Isolation 4.1 Immediate Actions: None 4.2 Subsequent Actions [1] IF any Emergency Operating Instruction (EOI) entry condition is met, **THEN ENTER** appropriate EOI(s). Otherwise, **MARK** N/A. [2] Using Panel 3-9-3 mimic or Containment Isolation Status System on Panel 3-9-4, **ENSURE** Group 6 isolation valves penetrating Primary Containment are CLOSED. [3] **IF** Refuel Zone Isolation is due to high radiation, as indicated on 3-RM-90-140/142, RX & REFUEL ZONE EXH CH A RAD MON RTMR, or 3-RM-90-141/143, RX & REFUEL ZONE EXH CH B RAD MON RTMR, Panel 3-9-10, or associated recorder on Panel 3-9-2, **THEN PERFORM** the following, otherwise, **MARK** steps N/A: [3.1] **N/A** [3.2] **N/A** **CAUTION** MSIV's may isolate on a Group I isolation if the time the Reactor Zone fans are removed from service is NOT minimized during Reactor Power operation and the Steam Vault Exhaust Booster Fan is NOT in service. Steam Tunnel Temperature is to be closely monitored while Reactor Zone fans are out of service. [4] Using 3-OI-30B, Reactor Zone Ventilation System, **ENSURE** STEAM VAULT EXH BOOSTER FAN in service. [5] **N/A** [6] **N/A** [7] **CHECK** the following to confirm condition: • 3-RR-90-144, REACTOR & REFUEL ZONE EXHAUST RADIATION (Panel 3-9-2)

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 4 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	BOP	• 3-RM-90-140/142, RX & REFUEL ZONE EXH CH A RAD MON RTMR (Panel 3-9-10) • 3-RM-90-141/143, RX & REFUEL ZONE EXH CH B RAD MON RTMR (Panel 3-9-10) [8] **CHECK** Reactor and Refueling Zone radiation detectors' power supplies reading 600V-DC on 3-RM-90-140/142, RX & REFUEL ZONE EXH CH A RAD MON RTMR, and 3-RM-90-141/143, RX & REFUEL ZONE EXH CH B RAD MON RTMR, as follows: [8.1] **DEPRESS** any button on touchpad to actuate screen. [8.2] Using touchpad, **SELECT** ETC. [8.3] Using touchpad, **SELECT** INPUT STATUS. [9] **MONITOR** the following to aid in determining location of problem: • 3-RR-90-1, AREA RADIATION (Panel 3-9-2) • 3-MON-90-50, AIR PARTICULATE MONITOR CONSOLE (Panel 3-9-2) • 3-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE (Panel 3-9-21) [10] **N/A** [11] **N/A** [12] **IF** isolation is result of invalid radiation signal **OR** loss of power to NUMAC drawer, **THEN REFER TO** 3-OI-90, Radiation Monitoring System, Section 6.6, NUMAC Radiation Monitor Operation, for Immediate Resetting of Group 6 Isolation Due to Reactor Zone Radiation Monitors, to inhibit trip. Otherwise, **MARK** N/A.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 5 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	BOP	3-OI-90, Radiation Monitoring System Section 6.4, NUMAC Radiation Monitor Operation **NOTES** 1) This section is applicable to Main Steam Line radiation monitors 3-RM-90-136, 137 and Reactor Zone/Refuel Zone radiation monitors 3-RM-90-140/142 and 3-RM-90-141/143. 2) A screen saver activates on the monitor after 30 minutes of constant display. [1] **IF** the screen saver is activated, **THEN DEPRESS** any of the prompt keys at the bottom of the screen to display the monitored channels. **NOTES** 1) There are two detectors for each channel of the Reactor Zone/Refuel Zone Monitors and are indicated on each monitor as follows: **3-RM-90-140/142** Display — Description 2A — 3-RE-90-142A, Reactor Zone Channel A Detector A. 2B — 3-RE-90-142B, Reactor Zone Channel A Detector B. 0A — 3-RE-90-140A, Refuel Zone Channel A Detector A. 0B — 3-RE-90-140B, Refuel Zone Channel A Detector B. **3-RM-90-141/143** Display — Description 3A — 3-RE-90-143A, Reactor Zone Channel B Detector A. 3B — 3-RE-90-143B, Reactor Zone Channel B Detector B. 1A — 3-RE-90-141A, Refuel Zone Channel B Detector A. 1B — 3-RE-90-141B, Refuel Zone Channel B Detector B. 2) Only the "A" detector of each channel described above has input to radiation recorder 3-RR-90-144. 3) Any active trip condition will be indicated by a highlighted "TRIP" at the top of the screen. A non-highlighted "TRIP" at the top of the screen indicates that there are one or more past trip conditions that have NOT been acknowledged. 4) Trips on the Reactor Zone/Refuel Zone Radiation monitors will automatically reset when the alarming condition resets.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 6 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	BOP	[2] **PERFORM** the following to immediately Reset Group 6 Isolation Due to Reactor Zone Radiation Monitors. **NOTES** 1) This section is to be performed in the event of a trip signal that will NOT reset in order to prevent further impact to plant operation due to reactor zone isolation. This is only considered appropriate when the signal is believed to be invalid. 2) Technical Specifications only allow one trip channel at a time to be out of service. This section provides directions for removing both trip channels from service but should only be performed on one channel at a time. Reference Technical Specification 3.3.6.2 for limiting conditions. 3) This section places jumpers to inhibit the upscale trips for a monitor. **CAUTION** A Reactor Zone isolation can cause a unit scram in less than five minutes due to high temperature in the steam tunnel. [2.1] **PLACE** affected monitor keylock switch to INOP position. [2.2] **IF** the affected monitor is 3-RM-90-140/142, **THEN PLACE** jumper across the following terminals in the back of Panel 3-9-10 to inhibit the upscale trip: TB HH terminals 49 and 50 [2.3] **N/A**
	Driver	**If contacted as the Work Control/Outside NUSO or Instrument Mechanics to install the jumper on Terminal Board HH Terminals 49 and 50, acknowledge the direction.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 7 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: When the Key-Lock switch on the Refuel Monitors on Panel 3-9-10 is placed in the INOP Position, the Monitor will display message: "INOP IS NOT SUPPORTED ON THE SIMULATOR. RETURN THE INOP/OPER KEY TO OPER TO RETURN THE NUMAC TO OPERABILITY". This message is normal, and no further action is required by the candidates with respect to placing the Radiation Monitor in an inoperable status.**
	NUSO	Technical Specification 3.3.6.2, Secondary Containment Isolation Instrumentation LCO 3.3.6.2 The secondary containment isolation instrumentation for each Function in Table 3.3.6.2-1 shall be OPERABLE. Applicability: According to Table 3.3.6.2-1

Secondary Containment Isolation Instrumentation
3.3.6.2

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3, (a)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr

(a) During operations with a potential for draining the reactor vessel.

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 5	Page 8 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	**CONDITION**: A. One or more channels INOPERABLE.	
	NUSO	**REQUIRED ACTION:** A.1 Place channel in trip	**COMPLETION TIME:** 12 hours for Functions 1 and 2 AND 24 hours for Functions other than Functions 1 and 2
	NUSO	Technical Specification 3.3.7.1, CREV System Instrumentation LCO 3.3.7.1 The CREV System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE Applicability: According to Table 3.3.7.1-1	

CREV System Instrumentation
3.3.7.1

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3,(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3 (a)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
5. Control Room Air Supply Duct Radiation - High	1,2,3, (a)	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 270 cpm above background

(a) During operations with a potential for draining the reactor vessel.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 9 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	**CONDITION:** A. – One or more required channels INOPERABLE.	
	NUSO	**REQUIRED ACTION:** A.1 Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	**COMPLETION TIME:** Immediately
	NUSO	**CONDITION:** C. – As required by Action A.1 and referenced in Table 3.3.7.1-1.	
	NUSO	**REQUIRED ACTION:** C.1 Declare associated CREV subsystem inoperable. AND C.2 Place channel in trip.	**COMPLETION TIME:** C.1 – 1 hour from discovery of loss of CREV initiation capability C.2 – 24 hours
	NRC	**It is acceptable for the candidate to enter Technical Specification 3.4.5, RCS Leakage Detection Instrumentation, (see page 26) without first entering Technical Requirements Manual 3.3.10, Reactor Coolant Leakage Detection first.**	
	NUSO	Tech Req Manual 3.3.10, Reactor Coolant Leakage Detection LCO 3.3.10 The Reactor Coolant Leakage Detection Instrumentation for each function in Table 3.3.10-1 shall be OPERABLE Applicability: Modes 1,2,3 **CONDITION**: A. – Required instrumentation INOPERABLE.	
	NUSO	**REQUIRED ACTION:** A.1 Enter the Condition referenced in Table 3.3.10-1 for the Function.	**COMPLETION TIME:** Immediately

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 10 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior
	NUSO	Reactor Coolant Leakage Detection TR 3.3.10 Table 3.3.10-1 Reactor Coolant Leakage Detection Instrumentation (see table below)
	NUSO	**CONDITION:** D. – As Required by Required Action A.1 and referenced in Table 3.3.10-1

Reactor Coolant Leakage Detection
TR 3.3.10

Table 3.3.10-1
Reactor Coolant Leakage Detection Instrumentation

FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	TECHNICAL SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Drywell Equipment Drain Flow Integrating Recorder (a)	B	TSR 3.3.10.1 TSR 3.3.10.3 TSR 3.3.10.4	N/A
2. Deleted			
3. Deleted			
4. Drywell Floor Drain Flow Integrating Recorder (b)	C	TSR 3.3.10.1 TSR 3.3.10.4 (c)	N/A
5. Drywell Floor Drain Sump Fill Rate Timer (b)	B	TSR 3.3.10.1 TSR 3.3.10.2 TSR 3.3.10.4	≥ 80.4 min
6. Drywell Floor Drain Sump Pump Out Rate Timer (b)	B	TSR 3.3.10.1 TSR 3.3.10.2 TSR 3.3.10.4 TSR 3.3.10.5	≤ 8.9 min
7. Drywell Air Sampling (Gas)	D	(d)	3 X Average Background
8. Drywell Air Sampling (Particulate)	E	(d)	3 X Average Background

(a) Used to determine identifiable reactor coolant LEAKAGE. Considered part of sump system.
(b) Used to determine unidentifiable reactor coolant LEAKAGE. Considered part of sump system.
(c) The channel calibration will be performed in accordance with SR 3.4.5.3.
(d) Surveillances will be performed in accordance with SR 3.4.5.1, 3.4.5.2 and 3.4.5.4.

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Event Description: Refuel Zone Radiation Monitors Fail Upscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** D.1 – Verify the primary containment atmospheric monitoring system particulate channel is OPERABLE. <u>OR</u> D.2 – Declare the primary containment atmospheric monitoring system inoperable. (TS LCO 3.4.5)	**COMPLETION TIME:** D.1 – Immediately D.2 – Immediately
	NUSO	**CONDITION:** E. As Required by Required Action A.1 and referenced in Table 3.3.10-1	
	NUSO	**REQUIRED ACTION:** E.1 – Verify the primary containment atmospheric monitoring system gas channel is OPERABLE. <u>OR</u> E.2 – Declare the primary containment atmospheric monitoring system inoperable. (TS LCO 3.4.5)	**COMPLETION TIME:** E.1 – Immediately E.2 – Immediately

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 5 Page 12 of 12

Event Description: Refuel Zone Radiation Monitors Fail Upscale

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td colspan="2">Technical Specification 3.4.5, RCS Leakage Detection Instrumentation
LCO 3.4.5 The following RCS Leakage Detection Instrumentation shall be OPERABLE:
a. Drywell Floor Drain Sump monitoring system; and
b. One channel of either Primary Containment atmospheric particulate or atmospheric gaseous monitoring system
Applicability: Modes 1, 2, and 3.

CONDITION:
B – Required Primary Containment atmospheric monitoring system inoperable.</td></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION:
B.1 – Analyze grab samples of Primary Containment atmosphere.
<u>AND</u>
B.2 – Restore required Primary Containment atmospheric monitoring system to OPERABLE status.</td><td>COMPLETION TIME:
B.1 – Once per 12 hours

B.2 – 30 days</td></tr>
<tr><td></td><td>NRC</td><td colspan="2">NOTE: No action is required within Technical Specification 3.4.4, Reactor Coolant System (RCS) Operational Leakage.</td></tr>
<tr><td></td><td>NRC</td><td colspan="2">End of Event 5. Proceed to Event 6, Standby Gas Train 'C' Fails to Auto Start.</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 6 Page 1 of 2

Event Description: Standby Gas Train 'C' Fails to Auto Start

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 6, Standby Gas Train 'C' Fails to Auto Start, is automatically entered on Simulator setup. No action is required by the driver to insert Event 6.**
	Driver	**If contacted as the U1 or U2 operator to start SBGT 'C', state that U1 / U2 operators cannot leave the horse shoe area at this time.**
	BOP	Determines that 'C' Standby Gas Train (SGT) did not automatically start. In accordance with OPDP-1, Conduct of Operations, Section 3.5.3, Manual Control of Automatic Systems, manually starts 'C' SGT.
	BOP	0-OI-65, Standby Gas Treatment System Section 5.2, Standby Gas Treatment System Manual Initiation [4] **START** SGT FAN A(B)(C) as follows: [4.3] **IF** starting SGT FAN A(B)(C) from Panel 3-9-25, **THEN DEPRESS** SGTS TRAIN A(B)(C) FAN, 0-HS-65-18A/3(40A/3)69A/3) push-button.
	BOP	Informs the NUSO that 'C' SGT failed to automatically start, but is started manually and is running normally.
	Driver	**If contacted as the Work Control/Outside NUSO to investigate the cause for 'C' SGT not automatically starting, acknowledge the direction.**
	NUSO	Technical Specification 3.6.4.3, SGT System LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE Applicability: Modes 1, 2, and 3. During operations with a potential for draining the Reactor Vessel (OPDRVs) **CONDITION:** A. – One SGT System inoperable.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 6 Page 2 of 2

Event Description: Standby Gas Train ‘C’ Fails to Auto Start

Time	Position	Applicant’s Actions or Behavior	
	NUSO	**REQUIRED ACTION:** A.1 – Restore SGT subsystem to OPERABLE status.	**COMPLETION TIME:** A.1 – 7 days
	NRC	**End of Event 6. Request that the Driver insert Event 7, 3A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start.**	

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 7	Page 1 of 3

Event Description: 3A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: The Unit 3 Main Turbine will trip and the Reactor will SCRAM in just over 1 minute from the loss of 3A Stator Cooling Water Pump if the crew does not manually start the standby pump (it will fail to automatically start).**
	Driver	**When requested by the Chief Examiner, insert Event 7, 3A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start.**
	Crew	**Critical Task:** **Start the Standby Stator Cooling Water Pump before the Turbine Trip Timer times out and trips the Main Turbine, resulting in a Reactor SCRAM.** **Critical Task Failure Criteria:** **The operating crew fails to start the standby Stator Cooling Water Pump and an automatic Reactor SCRAM occurs after the Turbine Trip Timer times out.**
	BOP	Acknowledges and reports the following alarms: • GEN STATOR COOLANT SYS ABNORMAL, 3-9-7A, Window 22 • TURBINE TRIP TIMER INITIATED, 3-9-8A, Window 1
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures
	BOP	Alarm Response Procedure, 3-ARP-9-7A GEN STATOR COOLANT SYS ABNORMAL, Window 22 **NOTE** The control room alarm typer can be used to confirm this alarm. Operator Action: A. **IF** while performing the action of this ARP, Turbine Trip Timer Initiated, **3-XA-55-9-8A window 1** alarms, **THEN** 1. **ENSURE** all available Stator Cooling Water Pumps running. 2. **ATTEMPT** to RESET alarm **3-XA-55-9-8A window 1.** 3. **IF** alarm fails to reset **AND** Reactor Power is above turbine bypass valve capability, **THEN SCRAM** the Reactor.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 7 Page 2 of 3

Event Description: 3A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start

Time	Position	Applicant's Actions or Behavior
	BOP	B. **ENSURE** a Stator Cooling Water Pump is running and **CHECK** Stator Temperature Recorder, 3-TR-57-59, Panel 3-9-8.
	BOP	Starts 3B SCW Pump. Verifies SCW has been restored and that TURBINE TRIP TIMER INITIATED, 3-9-8A, Window 1, can be reset.
	BOP	C. **CHECK** alarm by dispatching personnel to check the Stator Coolant Control Cabinet. D. **REQUEST** personnel to **REFER TO** Local Panel ARP for correct alarm response actions to be taken. E. **N/A**
	BOP	Alarm Response Procedure, 3-ARP-9-8A TURBINE TRIP TIMER INITIATED, Window 1 **NOTE** The control room alarm typer can be used to confirm this alarm. Operator Action: A. **CHECK** Stator Cooling Water Flow and Temperature and Generator Stator temperatures using ICS. B. **ENSURE** all available Stator Cooling Water Pumps running. **NOTE** The full capacity of the Turbine Bypass Valves with all nine valves open is 21.5% Reactor Power. To determine the capacity of the Bypass Valves, subtract 2.5% for each out of service Bypass Valve from the 21.5%. (Example, one Bypass Valve out of service, [21.5% - 2.5% = 19%], therefore, the capacity of the Bypass Valves with one Bypass Valve out of service is 19%.)

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 7 Page 3 of 3

Event Description: 3A Stator Cooling Water (SCW) Pump Failure, Standby Pump Fails to Auto Start

Time	Position	Applicant's Actions or Behavior
		C. **IF** all of the following conditions exist: • Alarm fails to reset, • Low Stator Cooling Water flow OR High Generator or Stator • Cooling temperatures are observed on ICS, • Reactor Power is above turbine bypass valve capability, **THEN, SCRAM** the Reactor. (Otherwise N/A) D. **DISPATCH** personnel to Stator Coolant Unit to investigate.
	Driver	**If contacted as the Turbine Building AUO to investigate the cause for 3A SCW Pump tripping, acknowledge the direction. After 3 minutes, report that 3A SCW Pump is hot to the touch.** **If contacted as Work Control/Outside SRO to write a clearance for 3A SCW Pump and/or protect 3B SCW Pump, acknowledge the direction.**
	NRC	**End of Event 7. Request that the Driver insert Event 8, Steam Leak in the Drywell.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 1 of 11

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8, Steam Leak in the Drywell.**
	BOP	Acknowledges and reports the following alarms as they are received: • DRYWELL TO SUPPR CHAMBER DIFF PRESS ABNORMAL, 3-9-3B, Window 26 • PRI CONTAINMENT N2 PRESS HIGH, 3-9-3B, Window 10 • DRYWELL NORM OPERATING PRESS HIGH, 3-9-3B, Window 19 • DRYWELL ATMOSPHERIC TEMP HIGH, 3-9-3B, Window 3 • DRYWELL PRESSURE ABNORMAL, 3-9-5B, Window 31 • DRYWELL PRESS APPROACHING SCRAM, 3-9-3B, Window 30
	NUSO	Acknowledges alarm report and directs the BOP to respond in accordance with appropriate Alarm Response Procedures. Directs the crew to monitor Drywell Pressure and Temperature, and provides critical parameters and set points for further action.
	BOP	Alarm Response Procedure, 3-ARP-9-3B PRI CONTAINMENT N_2 PRESS HIGH, Window 10 Operator Action: A. **CHECK** Containment Pressure using multiple indications: B. **CHECK** Containment Temperature. C. **REFER TO** 3-OI-64, Primary Containment System, Section 6.1, Venting the Drywell with Standby Gas Treatment Fan.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 2 of 11

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 3-ARP-9-3B DRYWELL NORM OPERATING PRESS HIGH, Window 19 Operator Action: A. **CHECK** Drywell Pressure and Temperature for rise and **CHECK** weather report for atmospheric pressure. B. **CHECK** to see if Drywell DP Compressor is running, **IF** Drywell DP Compressor is running **THEN STOP** compressor. C. **CHECK** N2 makeup valves to Suppression Chamber and Drywell closed. D. **IF** pressure rise is due to normal startup, **THEN REFER TO** 3-OI-64, Primary Containment System for normal venting instructions. E. **IF** Drywell Pressure is high, **THEN REFER TO** 3-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell.
	NRC	**Due to the rate of rise of Drywell Pressure, the crew may not have time to address rising Drywell Pressure using 3-OI-64, Primary Containment System, or 3-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell. 3-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell, actions start on page 40.**
	OATC	When the Drywell Pressure/Temperature trigger point is reached, inserts a Core Flow Runback and Reactor SCRAM.
	OATC	3-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 3-HS-99-5A/S3A and 3-HS-99-5A/S3B, REACTOR SCRAM A and B, on Panel 3-9-5. [2] **PLACE** 3-HS-99-5A-S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] **N/A**

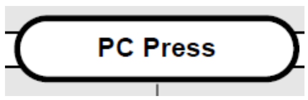
Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 3 of 11

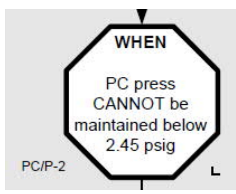
Event Description: Steam Leak in the Drywell

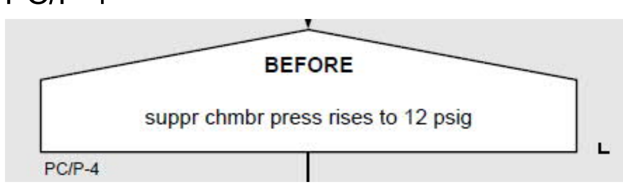
Time	Position	Applicant's Actions or Behavior
	OATC	[4] **IF** Reactor Power is 5% or BELOW, **THEN**: (Otherwise **MARK** N/A) **REPORT** the following to the UNIT SRO: • Reactor SCRAM • MODE Switch is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Power level
	Driver	**If contacted as any AUO to perform the following, acknowledge the direction:** • **Monitor Diesels** • **Perform the Gas Log**
	OATC	3-AOI-100-1, Reactor SCRAM 4.2 Subsequent Actions [1] **ANNOUNCE** Reactor SCRAM over PA system. [2] **DRIVE** in all IRMs and SRMs from Panel 3-9-5 as time and conditions permit. [2.1] **DOWNRANGE** IRMs as necessary to follow power as it lowers. [3] **ENSURE** SCRAM DISCH VOLUME VENT & DRAIN VALVES closed by green indicating lights at SDV Display on Panel 3-9-5.
	OATC	Informs the NUSO when Drywell Pressure reaches 2.45 PSIG.
	NUSO	When Drywell Pressure reaches 2.45 psig, enters the following EOIs and informs the crew: • 3-EOI-1, RPV Control • 3-EOI-2, Primary Containment Control

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 4 of 11

Event Description: Steam Leak in the Drywell

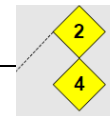
<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NRC</td><td>Candidate may elect to first spray the Drywell based on the Drywell Temperature leg of 3-EOI-2, Primary Containment Control
(See page 38)</td></tr>
<tr><td></td><td>NUSO</td><td>3-EOI-2, Primary Containment Control


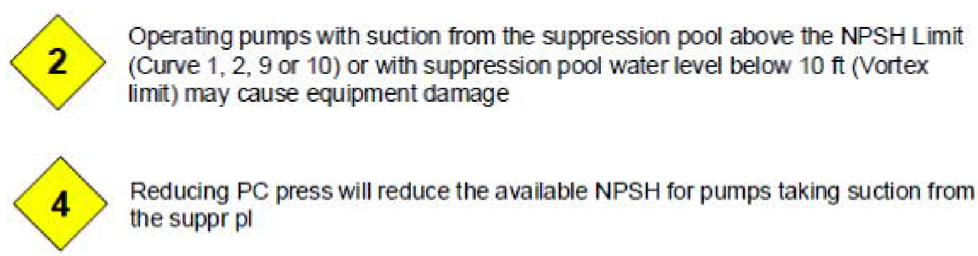
PC/P-1
MONITOR and CONTROL Primary Containment Pressure below 2.45 PSIG using the vent system (3-AOI-64-1)


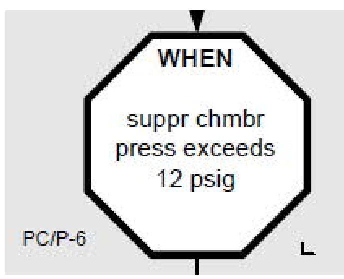
PC/P-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Primary Containment Pressure reduction is required to restore and maintain Adequate Core Cooling or reduce total offsite radiation dose</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Chamber Pressure drops to 0 PSIG</td><td>STOP Suppression Chamber Sprays</td></tr>
<tr><td>Drywell Pressure drops to 0 PSIG</td><td>STOP Drywell Sprays</td></tr>
</table>
PC/P-4

</td></tr>
</table>

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 5 of 11

Event Description: Steam Leak in the Drywell

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
PC/P-5


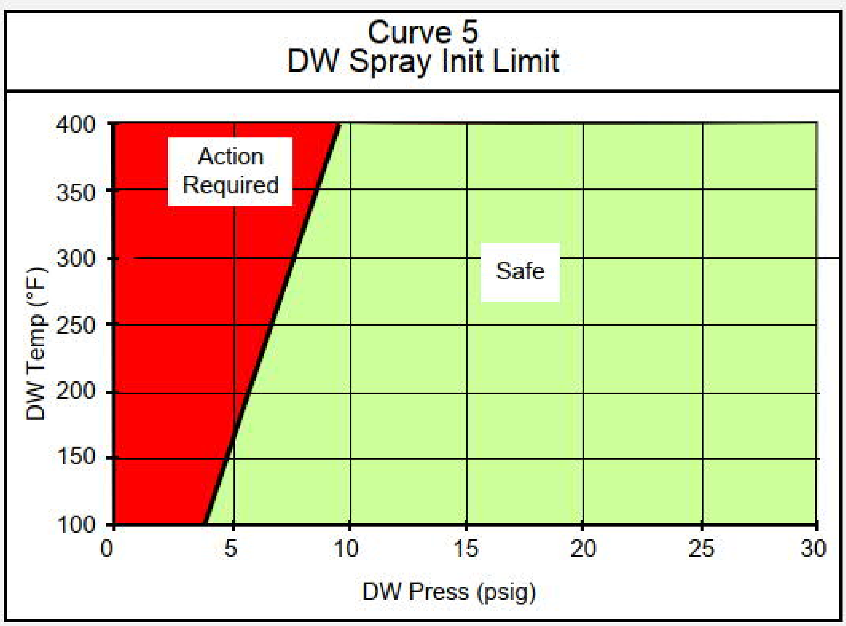
<table>
<tr><td colspan="2">INITIATE Suppression Chamber Sprays
➢ Use only source NOT required to assure Adequate Core Cooling by continuous injection (3-EOI-Appendix-17C)</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Needed to augment Suppression Chamber Sprays</td><td>NO ACTION REQUIRED</td></tr>
</table>


</td></tr>
<tr><td></td><td>NUSO</td><td>


</td></tr>
<tr><td></td><td>NRC</td><td>3-EOI-Appendix-17B, RHR System Operation Drywell Sprays – See Attachment 1, starting on page 45.
3-EOI-Appendix-17C, RHR System Operation Suppression Chamber Sprays – See Attachment 2, starting on page 43.</td></tr>
</table>

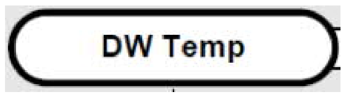
Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 8	Page 6 of 11

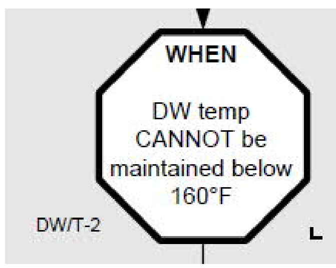
Event Description: Steam Leak in the Drywell

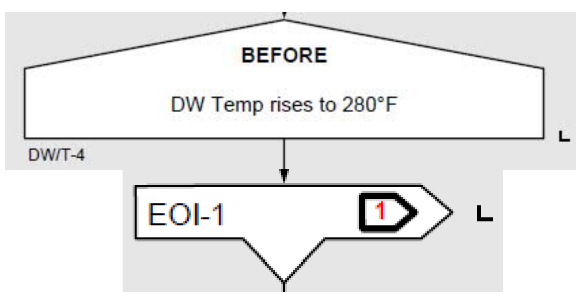
Time	Position	Applicant's Actions or Behavior
	NUSO	**IF** Suppression Pool Water Level is below 19 feet AND Drywell Temperature is in the safe area of Curve 5 **THEN** 1. **SHUT DOWN** Recirc Pumps 2. **SHUT DOWN** Drywell Blowers 3. **INITIATE** Drywell Sprays ➢ Use only sources NOT required to assure Adequate Core Cooling by continuous inj (APPX17B) IF: Needed to augment Drywell Sprays — THEN: NO ACTION REQUIRED PC/P-7 

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 7 of 11

Event Description: Steam Leak in the Drywell

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NRC</td><td>Candidate may elect to first Spray the Drywell based on the Drywell Temp leg of 3-EOI-2, Primary Containment Control.</td></tr>
<tr><td></td><td>NUSO</td><td>3-EOI-2, Primary Containment Control


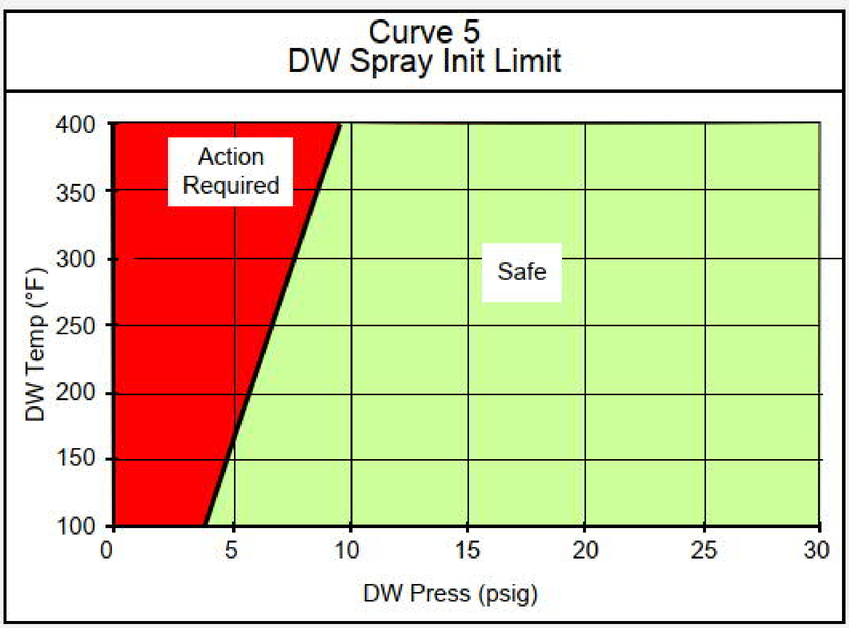
DW/T-1
MONITOR and CONTROL Drywell Temperature below 160ºF using available Drywell Cooling


DW/T-3
OPERATE all available Drywell Cooling

</td></tr>
<tr><td></td><td>NUSO</td><td>DW/T-5
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Pressure drops to 0 PSIG</td><td>STOP Drywell Sprays
(3-EOI-Appendix-17B)</td></tr>
</table></td></tr>
</table>

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 8	Page 8 of 11
Event Description: Steam Leak in the Drywell			

Time	Position	Applicant's Actions or Behavior
	NUSO	DW/T-6 **IF** Suppression Pool Water Level is below 19 feet AND Drywell Temperature is in the safe area of Curve 5 **THEN** 1. **SHUT DOWN** Recirc Pumps 2. **SHUT DOWN** Drywell Blowers 3. **INITIATE** Drywell Sprays ➢ Use only sources NOT required to assure Adequate Core Cooling by continuous inj (APPX17B)

IF	THEN
Needed to augment Drywell Sprays	NO ACTION REQUIRED



Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 8	Page 9 of 11

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	BOP	3-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell **NOTE** This procedure covers possible multiple symptoms of a problem within Primary Containment. Any or all of the symptoms may exist. The SRO will direct actions based on symptoms and experience. 4.1 Immediate Actions – None 4.2 Subsequent Actions 4.2.1 EOI Entry Conditions [1] **IF** any EOI entry condition is met, **THEN ENTER** appropriate EOI(s). (Otherwise N/A) 4.2.2 Drywell Pressure is High [1] **CHECK** Drywell Pressure using multiple indications. [2] **IF** Drywell Pressure rising rate indicates Reactor SCRAM at 2.45 psig is imminent, **THEN REDUCE** Reactor Power via Recirc Flow to minimize the impact of a SCRAM from high power. (Otherwise N/A) [3] **ALIGN** and **START** additional Drywell coolers and fans as necessary. **REFER TO** 3-OI-64, Primary Containment System. **CAUTION** Stack release rates exceeding $1.4x10^7$µci/sec, or a 0-SI-4.8.B.1.a.1, Airborne Effluent Release Rate, release fraction above one will result in ODCM release limits being exceeded. [4] **VENT** Drywell as follows: [4.1] **CLOSE** 3-FCV-64-34, SUPPR CHBR INBD ISOLATION VLV (Panel 3-9-3). [4.2] **ENSURE OPEN**, 3-FCV-64-31, DRYWELL INBD ISOLATION VLV, (Panel 3-9-3). [4.3] **ENSURE** 3-FIC-84-20, PATH A VENT FLOW CONT, is in AUTO and SET at 100 scfm (Panel 3-9-55). [4.4] **ENSURE** RUNNING, required Standby Gas Treatment Fan(s) SGTS Train(s) C(A)(B) (Panel 3-9-25). [4.5] **IF** required, **THEN REQUEST** Unit 1 Operator to START Standby Gas Treatment Fans A or B. (Otherwise **N/A**).

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 10 of 11

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	BOP	**NOTE** If 3-FCV-84-20 closes after placing 3-HS-64-35 to open, the valve's closure signal must be reset and 3-HS-64-35 must be returned to the OPEN position in order for 3-FCV-84-20 to RE-OPEN. [4.6] N/A [4.7] **PLACE** 3-FCV-84-20 3-HS-64-35, CONTROL DW/SUPPR CHBR VENT, in OPEN (Panel 3-9-3). **CAUTION** Stack release rates exceeding 1.4 X 10^7 µci/sec, or a SI-4.8.B.1.a.1 release fraction above one will result in ODCM release limits being exceeded. [4.8] **MONITOR** stack release rates to prevent exceeding ODCM limits. [4.9] **WHEN** Drywell Pressure has been reduced as required, **THEN STOP** SGT Train(s). [4.10] **ENSURE** 3-HS-64-35, in AUTO and 3-FCV-84-20 CLOSED (Panel 3-9-3). [4.11] **OPEN** 3-FCV-64-34, SUPPR CHBR INBD ISOLATION VALVE (Panel 3-9-3). [4.12] **ENSURE** Drywell DP compressor operates correctly to maintain required Drywell to Suppression Chamber DP. [4.13] **N/A** [5] **N/A** [6] **ENSURE CLOSED**, N2 makeup valves to Drywell and Suppression Chamber. [7] **CHECK** Suppression Chamber Pressure. [8] **CHECK** Suppression Pool Water Level. [9] **CHECK** Suppression Pool Temp for indication of a leaking or stuck open relief valve. [10] **N/A**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 8 Page 11 of 11

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	BOP	[11] **N/A** [12] **CHECK** DRYWELL ATMOSPHERE DEWPOINT TEMPERATURE, 3-MR-80-36, for indication of a steam or water leak in the Drywell (Panel 3-9-47). [13] **N/A** **NOTE** Based on recent On-Shift Analysis from the License Amendment, chemistry sampling may be delayed following plant events. In the case of a REP event, Chemistry will be required to assist with dose assessments. Dose assessments will have precedence over sampling actions in REP events. [14] **NOTIFY** Chemistry to sample Drywell atmosphere for radioactivity. [15] **NOTIFY** Radwaste that fluids being discharged from Drywell may be highly radioactive. 4.2.3 High Drywell Temperature [1] **IF** Reactor is at power **AND** Drywell cooling is lost and can **NOT** be immediately restored, **THEN PERFORM** the following: (Otherwise N/A) [1.1] IF Core Flow is above 60%, **THEN REDUCE** Core Flow to between 50-60%. [1.2] **MANUALLY SCRAM** the Reactor and **REFER TO** 3-AOI-100-1. **(see page 33 for 3-AOI-100-1 actions)** [1.3] **INITIATE** a 90 ºF/hr cooldown rate. **REFER TO** 3-AOI-100-1. [2] **CHECK** Drywell Temperature using multiple indications. [3] N/A [4] **VENT** the Drywell. **REFER TO** Section 4.2.2[4]. [5] **N/A**
	NRC	**Event 9, Drywell Spray Failure, is inserted during Simulator Setup. No action is required by the Driver to insert Event 9.**
	NRC	**When the crew has sprayed Containment and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using either high or low pressure systems, end of Scenario.**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 1 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	NRC	**The first loop of Drywell Sprays will fail when the crew attempts to spray Containment to reduce Containment Pressure and Temperature, requiring action to use the other loop of Drywell Spray.**
	NUSO	Directs BOP to perform 3-EOI-Appendix-17C, RHR System Operation Suppression Chamber Sprays.
	BOP	3-EOI-Appendix-17C, RHR System Operation Suppression Chamber Sprays [1] **BEFORE** Suppression Chamber pressure drops below 0 psig **CONTINUE** in this procedure at Step 1.0[6]. [2] **IF** Adequate core cooling is assured **OR** Directed to spray the Suppression Chamber irrespective of Adequate Core Cooling, **THEN BYPASS** LPCI Injection Valve auto open signal as necessary: • **PLACE** 3-HS-74-155A, LPCI SYS I OUTBOARD INJECTION VALVE BYPASS SELECT IN **BYPASS** • **PLACE** 3-HS-74-155B, LPCI SYS II OUTBOARD INJECTION VALVE BYPASS SELECT IN **BYPASS** [3] **N/A** [4] **N/A** [5] **INITIATE** Suppression Chamber Sprays as follows: [5.1] **ENSURE** at least one RHRSW Pump supplying each EECW header. [5.2] **IF EITHER** of the following exists: • LPCI Initiation signal is NOT present, OR • Directed by SRO, **THEN PLACE** keylock switch 3-XS-74-122(130), RHR SYSTEM I (II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE. [5.3] **MOMENTARILY PLACE** 3-XS-74-121 (129), RHR SYS I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, switch in SELECT.

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 9	Page 2 of 6
Event Description: Drywell Spray Failure			

Time	Position	Applicant's Actions or Behavior
	BOP	[5.4] **IF** 3-FCV-74-53 (67), RHR SYSTEM I (II) LPCI INBOARD INJECTION VALVE, is OPEN, **THEN ENSURE CLOSED** 3-FCV-74-52 (66), RHR SYSTEM I (II) LPCI OUTBOARD INJECTION VALVE. [5.5] **ENSURE OPERATING** the desired System I (II) RHR pump(s) for Suppression Chamber Spray. [5.6] **ENSURE OPEN** 3-FCV-74-57(71), RHR SYSTEM I(II) SUPPRESSION CHAMBER /POOL ISOLATION VALVE [5.7] **OPEN** 3-FCV-74-58 (72), RHR SYSTEM I (II) SUPPRESSION CHAMBER SPRAY VLV. [5.8] **IF** RHR System I(II) is operating ONLY in Suppression Chamber Spray mode, **THEN** CONTINUE in this procedure at step 1.0[5.11]. [5.9] **ENSURE CLOSED** 3-FCV-74-7 (30), RHR SYSTEM I (II) MINIMUM FLOW VALVE. [5.10] **RAISE** system flow by placing the second RHR System I(II) pump in service as necessary. [5.11] **MONITOR** RHR Pump NPSH using Attachment 2. [5.12] **ENSURE** RHRSW pump supplying desired RHR Heat Exchanger(s). [5.13] **THROTTLE** the following in-service RHRSW outlet valves to obtain the required RHRSW flow: • 3-FCV-23-34, RHR HX 3A RHRSW OUTLET VALVE (Required flow is 1700. to 4500 gpm) • 3-FCV-23-46, RHR HX 3B RHRSW OUTLET VALVE (Required flow is 1350 to 4500 gpm for B1 pump) (Required flow is 1700 to 4500 gpm for B2 pump) • 3-FCV-23-40, RHR HX 3C RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) • 3-FCV-23-52, RHR HX 3D RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) [5.14] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 3 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as Chemistry, acknowledge any reports or direction given.**
	BOP	[6] **N/A**
	CREW	**Critical Task:** **Initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit Curve and before Drywell Temperature rises to 280 °F.** **Critical Task Failure Criteria: The operating crew fails to spray the Drywell before Drywell Temperature reaches 280 ºF.**
	NUSO	Directs BOP to perform 3-EOI-Appendix-17B RHR System Operation Drywell Sprays
	BOP	3-EOI-Appendix-17B RHR System Operation Drywell Sprays [1] **BEFORE** Drywell Pressure drops below 0 psig **CONTINUE** in this procedure at Step 1.0[7]. [2] **IF** Adequate Core Cooling is assured **OR** directed to spray the Drywell irrespective of Adequate Core Cooling, **THEN BYPASS** LPCI injection valve auto open signal as necessary: • **PLACE** 3-HS-74-155A, LPCI SYSTEM I OUTBOARD INJECTION VALVE BYPASS SELECT IN **BYPASS** • **PLACE** 3-HS-74-155B, LPCI SYSTEM II OUTBOARD INJECTION VALVE BYPASS SELRCT IN **BYPASS** [3] **ENSURE** Recirc Pumps and Drywell Blowers are shutdown. [4] **N/A** [5] **N/A**

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 4 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	BOP	[6] **INITIATE** Drywell Sprays as follows: [6.1] **ENSURE** at least one RHRSW Pump supplying each EECW header. [6.2] **IF** EITHER of the following exists: • LPCI Initiation signal is NOT present, **OR** • Directed by SRO, **THEN PLACE** keylock switch 3-XS-74-122(130), RHR SYSTEM I (II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE [6.3] **MOMENTARILY PLACE** 3-XS-74-121 (129), RHR SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT switch in SELECT. [6.4] **IF** 3-FCV-74-53 (67), RHR SYS I (II) LPCI INBOARD INJECTION VALVE, is OPEN, **THEN ENSURE CLOSED** 3-FCV-74-52 (66), RHR SYSTEM I (II) LPCI OUTBOARD INJECTION VALVE. [6.5] **ENSURE OPERATING** the desired System I (II) RHR Pump(s) for Drywell Spray. [6.6] **OPEN** the following valves: • 3-FCV-74-60(74), RHR SYS I(II) DRYWELL SPRAY OUTBOARD VALVE • 3-FCV-74-61 (75), RHR SYS I (II) DRYWELL SPRAY INBOARD VALVE [6.7] **ENSURE CLOSED** 3-FCV-74-7 (30), RHR SYSTEM I (II) MINIMUM FLOW VALVE. [6.8] **IF** Additional Drywell Spray flow is necessary, **THEN PLACE** the second System I (II) RHR Pump in service. [6.9] **MONITOR** RHR Pump NPSH using Attachment 2. [6.10] **ENSURE** RHRSW Pump supplying desired RHR Heat Exchanger(s).

Op Test No.: 21-04	Scenario No. NRC-4	Event No.: 9	Page 5 of 6

Event Description: Steam Leak in the Drywell

Time	Position	Applicant's Actions or Behavior
	BOP	[6.11] **THROTTLE** the following in-service RHRSW Outlet Valves to obtain the required RHRSW Flow: • 3-FCV-23-34, RHR HX 3A RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) • 3-FCV-23-46, RHR HX 3B RHRSW OUTLET VALVE (Required flow is 1350 to 4500 gpm for B1 pump) (Required flow is 1700 to 4500 gpm for B2 pump) • 3-FCV-23-40, RHR HX 3C RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) • 3-FCV-23-52, RHR HX 3D RHRSW OUTLET VALVE (Required flow is 1700 to 4500 gpm) [6.12] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	Driver	**If contacted as Chemistry, acknowledge any reports or direction given.**
	BOP	[7] **WHEN EITHER** of the following exists: • BEFORE Drywell Pressure drops below 0 psig, **OR** • Directed by SRO to stop Drywell Sprays, **THEN STOP** Drywell Sprays as follows: [7.1] **ENSURE CLOSED** the following valves: • 3-FCV-74-100, RHR SYSTEM I U-2 DISCH XTIE • 3-FCV-74-60(74), RHR SYSTEM I(II) DW SPRAY OUTBOARD VALVE • 3-FCV-74-61(75), RHR SYSTEM I(II) DW SPRAY INBOARD VALVE [7.2] **ENSURE OPEN** 3-FCV-74-7(30), RHR SYSTEM I (II) MINIMUM FLOW VALVE. [7.3] I**F** RHR operation is desired in ANY other mode, **THEN EXIT** this EOI Appendix. [7.4] **STOP** RHR Pumps 3A and 3C (3B and 3D).

Op Test No.: 21-04 Scenario No. NRC-4 Event No.: 9 Page 6 of 6

Event Description: Drywell Spray Failure

Time	Position	Applicant's Actions or Behavior
	NRC	**End of Event 9. Once the crew has sprayed Containment and has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using either high or low pressure systems, end of Scenario.**

Scenario Setup
UNIT 3

IC	28
Exam IC	255

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-2	74	AOI-100-1	74	EOI-1	13
OI-3	112	ARP-3A	57	EOI-2	13
OI-65	55	ARP-3B	23	APPX-17B	13
OI-68	99	ARP-5A	54	APPX-17C	12
OI-85	91	ARP-5B	32	TS 3.3.6.2	213
OI-90	62	ARP-6C	29	TS 3.3.7.1	213
AOI-3-1	14	ARP-7A	30	TS 3.4.5	244
AOI-64-2D	17	ARP-8A	38	TS 3.6.4.3	249
AOI-64-1	6	OPDP-1	46	TRM 3.3.10	93

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Hang Protected Equipment Tags on the following: RHR Loop I and II, Core Spray Loop II, 3C and 3D EDG, HPCI, 3EC and 3ED 4KV Shutdown Boards, and 3A 250V RMOV Board
Schedule Files(s):	2104 NRC Scenario 4 UNIT 3.sch
Event Files(s):	2104 NRC Scenario 4 UNIT 3.evt

Schedule File – 2104 NRC Scenario 4 UNIT 3.sch

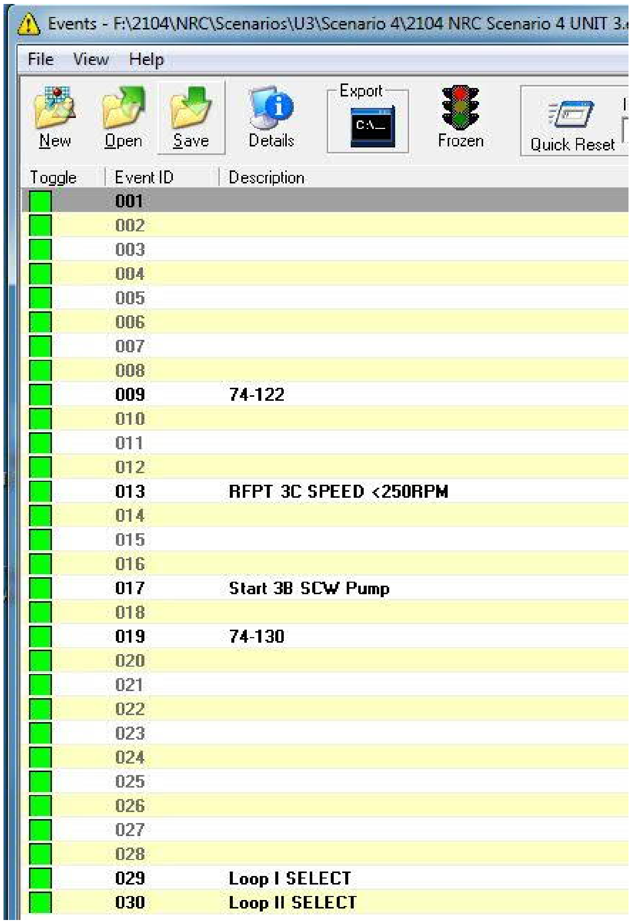
Event	Action	Description
	2104 NRC Scenario 4 UNIT 2.evt	Event File
1	Insert remote FW06 to START	CONDENSATE BOOSTER PUMP C AUX OIL PUMP
2	Insert malfunction RD22 to 100.00000 on	CRD FLOW TRANSMITTER FT-85-11 FAILURE
3	Insert malfunction FW33C to 85.00000 in 60	RFPT 2C AXIAL THRUST CH A FAILURE (XE-3-1062-1)
3	Insert malfunction FW33D to 88.00000 in 60	RFPT 2C AXIAL THRUST CH B FAILURE (XE-3-1062-2)
13	Delete malfunction FW33C	RFPT 2C AXIAL THRUST CH A FAILURE (XE-3-1062-1)
13	Delete malfunction FW33D	RFPT 2C AXIAL THRUST CH B FAILURE (XE-3-1062-2)

Schedule File – 2104 NRC Scenario 4 UNIT 3.sch

Event	Action	Description
5	Insert malfunction RM08A to 1000000.00000	REFUEL ZONE RAD CH 0A MONITOR FAILURE (RM-90-140/142)
5	Insert malfunction RM08B to 1000000.00000	REFUEL ZONE RAD CH 0B MONITOR FAILURE (RM-90-140/142)
	Insert malfunction PC01C	SBGT SYSTEM C AUTO START FAILURE (CONTACT 10 OF HS 65-69A)
7	Insert override HS-35-35A to STOP	GEN STATOR CLG WATER PUMP 3A
	Insert malfunction PMP-35-36 to FAIL_CCOIL	52_BREAKER GEN STATOR COOLING WATER PUMP B
17	Delete malfunction PMP-35-36	52_BREAKER GEN STATOR COOLING WATER PUMP B
8	Insert malfunction TH33A to 1.50000 in 600	MAIN STEAM LINE A BREAK IN CONTAINMENT (DRYWELL)
	Insert override XS-74-121 to NAR	RHR SYS I CTMT SPRAY/CLG VLV SELECT
	Insert override XS-74-129 to NAR	RHR SYS II CTMT SPRAY CLG/VLV SELECT
9	Delete override XS-74-129	RHR SYS II CTMT SPRAY CLG/VLV SELECT
19	Delete override XS-74-121	RHR SYS I CTMT SPRAY/CLG VLV SELECT
29	Insert override XS-74-129 to NAR	RHR SYS II CTMT SPRAY CLG/VLV SELECT
30	Insert override XS-74-121 to NAR	RHR SYS I CTMT SPRAY/CLG VLV SELECT

Event File

List



Details

UNIT 3 SHIFT TURNOVER MEETING			Today
MODE 1	**DAYS ON LINE** **275**	**Drywell Leakage (GPM)** **1.89**	**Protected Equipment**
			RHR Loop I and II
	PRA (EOOS) -Green		Core Spray Loop II
Rx Power	500Kv GRID - Qualified	**Floor Drain (GPM)**	3C and 3D EDG
95.0%	161Kv Grid -Qualified	**0.31**	HPCI
MWe	**Last breaker closure**	**Equipment Drain (GPM)**	4KV Shutdown Boards 3EC, 3ED
1224	**8/15/20 5:41**	**1.58**	250V RMOV Board 3A

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

Core Spray Loop I Outage – day 1 of 7 day LCO IAW Tech Spec 3.5.1.A

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

APRM 1 Critical Fault – APRM 1 is bypassed (Information Only LCO 3.3.1.1)

Core Spray Loop I MOVATS Testing in progress

Reactor Power lowered to 95% to secure 3C Condensate Booster Pump

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Reduce Reactor Power and remove 3C Condensate Booster Pump from service for maintenance

Reactor Engineer will brief the return to 100% power later in the shift

OPERATOR WORK AROUNDS **OWAs - 1** **Burdens - 2** **Challenges - 28**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

TVA

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Respond to a Control Rod Drift in accordance with 2-AOI-85-5, Rod Drift In
JPM NUMBER:	80A-U2	REVISION:	5

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-085-AB-05/ Respond to a Control Rod Drift In			
K/A RATINGS:	RO: 3.2 SRO: 3.3			
K/A No. & STATEMENT:	201002 Reactor Manual Control System A2.02; Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rod Drift Alarm			
RELATED PRA INFORMATION:	Risk Significant			
SAFETY FUNCTION:	1			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 13 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ______________
Developer Date

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator Date

Approved by: ______________________ ______________
Site Training Management Date

Approved by: ______________________ ______________
Site Training Program Owner Date

TVA | **Job Performance Measure (JPM)**

OPERATOR: ______________________________ JPM Number: 80A-U2

RO _____ SRO _____ DATE: ___________

TASK STANDARD: The Examinee is expected to exercise partially withdrawn Control Rods and respond to a Control Rod drift.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 2-SR-3.1.3.3, 2-AOI-85-5, 2-AOI-100-1

VALIDATION TIME: 13 min

PERFORMANCE TIME: __________

COMMENTS: ____
____
____
____
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Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	05/27/10	All	Initial issue
1	09/01/15	1	Incorrect task number; changed to U-085-AB-05: Incorrect malfunction numbers; rd07r0231, rd07r2223 & rd07r3031; Corrected
2	10/09/15	All	Incorporated review comments
3	10/17/16	All	Updated format
4	08/20/20	All	Procedure update
5	1/14/21	All	JPM update

Procedure Revisions

Procedure	Revision
2-SR-3.1.3.3	33
2-AOI-85-5	24
2-AOI-100-1	116

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• Reset to IC 28 • Run schedule file: 2104 NRC JPM a UNIT 2.SCH • Verify event file 2104 NRC JPM a UNIT 2.EVT loads • Place the simulator in RUN to ensure stable conditions • Provide Initial Rod Data Sheet – PRLOG • Endure the candidate has been pre-briefed on 2-SR-3.1.3.3 • Display “CRD Exercise” on ICS Screen. NOTE: CRD EXERCISE MUST BE STARTED ON THE BOOTH ICS COMPUTER (System mimics -> Ops Support -> CRD Exercise Monitor) • When prompted by the Examiner, INSERT - Event 1 to Drift Control Rod 14-31 into the Core • When prompted by the Examiner, INSERT - Event 2 to Drift Control Rod Multiple Control Rods into the Core

Malfunctions	Description	Event	Severity	Delay	Initial set
rd07r1431	Rod 14-31 drifts into core	1	NA	0 sec	NA
rd07r1831	Rod 18-31 drifts into core	2	NA	0 sec	NA
rd07r2223	Rod 22-23 drifts into core	2	NA	30 sec	NA

Remotes	Description	Event	Severity	Delay	Initial set
NONE					

Overrides	Description	Event	Severity	Delay	Initial set
NONE					

Schedule File(s): **2104 NRC JPM a UNIT 2.SCH**
Event File(s): **2104 NRC JPM a UNIT 2.EVT**

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

- You are an Operator on Unit 2
- The Reactor is at 100% Power

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to perform 2-SR-3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3 Exercising a Partially Withdrawn Control Rod.

START TIME:___________

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 2-SR-3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3, before commencing the JPM.	
Step 1: 2-SR- 3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3, Exercising a Partially Withdrawn Control Rod **NOTE** 1) Problem details of any Control Rod issues observed during the performance of this surveillance must be recorded on Attachment 3 for further review and possible corrective maintenance. Slow/fast rod movements or settle times should be noted in the remarks section. 2) Section 7.3 is performed for all Partially Withdrawn Control Rods and performance of steps is represented by initialing the appropriate CRD exercised on Attachment 1. **CAUTIONS** 1) Any mispositioned Control Rod events will be dispositioned by following the direction contained within 2-AOI-85-7. 2) If a Control Rod moves unexpectedly one notch beyond its intended position, notify Unit SRO, obtain Unit SRO concurrence and return the rod to its intended position. 3) At any time Core Thermal Power is less than or equal to 10%, entry into LCO 3.1.6 may be required. **7.3 Exercising a Partially Withdrawn Control Rod** **EXAMINER NOTE: The candidate may select any partially withdrawn Control Rod in any order.** [1] **SELECT** desired Control Rod by **DEPRESSING** appropriate 2-XS-85-40, CRD ROD SELECT pushbutton. Expected Action(s): Selects a partially withdrawn Control Rod.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: [2] **OBSERVE** the following for the selected Control Rod: • **CHECK** 2-XS-85-40, CRD ROD SELECT pushbutton is brightly ILLUMINATED • **CHECK** white light on the Full Core Display is ILLUMINATED • **CHECK** Rod Out Permit light is ILLUMINATED Expected Action(s): Verifies that the appropriate lights are illuminated.	_____SAT _____UNSAT _____N/A
Step 3: [3] **INSERT** Control Rod one notch by performing the following: [3.1] **PLACE** 2-HS-85-48, CRD CONTROL SWITCH in ROD IN and **RELEASE.** [3.2] **OBSERVE** Control Rod settles into the desired position and ROD SETTLE light extinguishes. Expected Action(s): Inserts withdrawn Control Rod one notch.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 4: [3.3] **IF** the Control Rod failed to insert, **THEN PERFORM** the following: (Otherwise N/A) [3.4] **IF** the Control Rod unexpectedly inserts one notch beyond its intended position, **THEN PERFORM** the following: (Otherwise N/A) Expected Action(s): Marks Steps [3.3] and [3.4] as N/A.	_____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 5: [4] **WITHDRAW** selected Control Rod one notch by performing the following: [4.1] **PLACE** 2-HS-85-48, CRD CONTROL SWITCH in ROD OUT NOTCH and **RELEASE**. [4.2] **OBSERVE** Control Rod settles into the desired position and ROD SETTLE light extinguishes. Expected Action(s): Withdraws withdrawn Control Rod one notch.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 6: [4.3] **IF** Control Rod failed to withdraw, **THEN PERFORM** the following: (Otherwise N/A) [4.4] **IF** Control Rod unexpectedly withdraws one notch beyond its intended position, **THEN PERFORM** the following: (Otherwise N/A) Expected Action(s): Marks Steps [4.3] and [4.4] as N/A.	_____SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [5] **DOCUMENT** completion of Control Rod test as follows: [5.1] PERFORMER • **INITIAL** Attachment 1 (Control Rod Exercise Data Sheet) in the box corresponding to the Control Rod coordinates for the Control Rod just exercised to document proper movement and CRD latching. [5.2] Concurrent Verifier (CV) • **ENSURE** rod inserted and returned to its original position. • **INITIAL** Attachment 2 (Control Rod Concurrent Verifier (CV)Check) in the box corresponding to the Control Rod coordinates for the Rod just exercised. **EXAMINER NOTE: If prompted by applicant for Concurrent Verification, state “Attachment 2 Concurrent Verification has been completed by another Operator.”** Expected Action(s): Initials Attachment 1 for exercised Control Rod and continues to exercise Rods.	_____SAT _____UNSAT _____N/A
EXAMINER NOTES: • **Perform above actions for at least two Control Rods.** • **Begin Alternate Path - when satisfied with the number of rod manipulations direct Simulator Booth Operator to insert Event 1 for Control Rod 14-31 Drift In.**	
DRIVER NOTE: **When requested by the Examiner, insert Event 1 to cause Control Rod 14-31 to drift in.**	

STEP / STANDARD	SAT / UNSAT
Step 8: Candidate recognizes Control Rod 14-31 drifting in and responds per 2-AOI-85-5, Rod Drift In. **4.2 Subsequent Actions** [2] **IF** a Control Rod is moving (or has moved) from its intended position without operator actions, **THEN INSERT** the Control Rod to position 00 using CONTINUOUS IN. (Otherwise N/A) [3] **IF** a Control Rod Block occurs during rod insertion due to Rod Worth Minimizer, **THEN BYPASS** the RWM per step 4.2[1] above. (Otherwise N/A) Expected Action(s): Responds in accordance with 2-AOI-85-5, Rod Drift In, and inserts Control Rod 14-31 to full in position as indicated by position 00 indication.	______ SAT ______UNSAT ______N/A
DRIVER NOTE: **When Control Rod 14-31 reaches position 00, verify that malfunction rd07r1431 (14-31 Control Rod Drift In) is deleted by the simulator setup so that CONTROL ROD DRIFT, (2-9-5A, WINDOW 28) can be reset.**	
EXAMINER NOTES: **Control Rod 14-31 will settle into position 00.** **The Candidate may or may not reset the drift lights and alarms.** **Expected Alarms:** • **CONTROL ROD WITHDRAWAL BLOCK, (2-9-5A, WINDOW 7)** • **ROD BLOCK MONITOR (RBM) DOWNSCALE, (2-9-5A, WINDOW 31)**	
Step 9: [4] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern. Expected Action(s): Candidate notifies Reactor Engineer to Evaluate Core Thermal limits and Preconditioning Limits for the current Control Rod Pattern.	______SAT ______UNSAT ______N/A
CUE: If contacted as the Reactor Engineer acknowledge any direction or information given.	

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td><u>Step 10</u>:

[5] IF another Control Rod Drift occurs before Reactor Engineering completes the evaluation,
• THEN MANUALLY SCRAM the Reactor and enter 2-AOI-100-1, Reactor SCRAM.

[6] CHECK Thermal Limits on ICS (RUNMON).

<u>Expected Action(s):</u>

Reviews step and may inform the Nuclear Unit Senior Operator (NUSO) of the requirement to insert a Reactor SCRAM if another Control Rod drifts.</td><td>_____SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">EXAMINER NOTE: Acknowledge candidate report.</td></tr>
<tr><td colspan="2">EXAMINER NOTE: When ready for multiple rod drifts, direct the Simulator Booth Operator to insert Event 2 (Control Rod 18-31 Rod Drift, and 30 seconds later Control Rod 22-23 Rod Drift).</td></tr>
<tr><td colspan="2">DRIVER NOTE: When requested by the Examiner, insert Event 2 (Control Rod 18-31 Drift In). 30 seconds later, Control Rod 22-23 will drift in if a Reactor SCRAM has not been inserted.</td></tr>
<tr><td><u>Step 11</u>:

4.1 Immediate Actions

[1] IF multiple Control Rods are drifting into core, THEN MANUALLY SCRAM Reactor. REFER TO 2-AOI-100-1.

<u>Expected Action(s):</u>

Recognizes that multiple Control Rods are drifting into the Core and inserts a manual Reactor SCRAM in accordance with the Immediate Actions of 2-AOI-85-5, Rod Drift In.</td><td>Critical Step

_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: When informed that multiple Control Rods are drifting, acknowledge the report. At any point following the Reactor SCRAM, request that the Driver place the Simulator in FREEZE and inform the candidate “Another Operator will continue with the Reactor SCRAM actions. This completes your task”.</td></tr>
</table>

STOP TIME: _____

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

- You are an Operator on Unit 2
- The Reactor is at 100% Power

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to perform 2-SR-3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3 Exercising a Partially Withdrawn Control Rod.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Respond to a Control Rod Drift in accordance with 3-AOI-85-5, Rod Drift In
JPM NUMBER:	80A-U3	REVISION:	4

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-085-AB-05/ Respond to a Control Rod Drift In			
K/A RATINGS:	RO: 3.2 SRO: 3.3			
K/A No. &STATEMENT:	201002 Reactor Manual Control System A2.02; Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rod Drift Alarm			
RELATED PRA INFORMATION:	Risk Significant			
SAFETY FUNCTION:	1			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 13 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

OPERATOR: ______________________________ JPM Number: 80A-U3

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to exercise partially withdrawn Control Rods and respond to a Control Rod drift.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 3-SR-3.1.3.3, 3-AOI-85-5, 3-AOI-100-1

VALIDATION TIME: 13 min

PERFORMANCE TIME: ___________

COMMENTS: ____

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Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained
THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	05/27/10	All	Initial issue
1	05/11/15	1	Incorrect task number; changed to U-085-AB-05: Incorrect malfunction numbers; rd07r0231, rd07r2223 & rd07r3031; Corrected
2	10/09/15	All	Incorporated review comments
3	11/10/20	All	Procedure update
4	1/14/21	All	JPM update

Procedure Revisions

Procedure	Revision
3-SR-3.1.3.3	34
3-AOI-85-5	15
3-AOI-100-1	74

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• Reset to IC 28 • Run schedule file: 2104 NRC JPM a UNIT 3.SCH • Verify event file 2104 NRC JPM a UNIT 3.EVT loads • Place the simulator in RUN to ensure stable conditions • Provide Initial Rod Data Sheet – PRLOG • Endure the candidate has been pre-briefed on 3-SR-3.1.3.3 • Display “CRD Exercise” on ICS Screen. NOTE: CRD EXERCISE MUST BE STARTED ON THE BOOTH ICS COMPUTER (System mimics -> Ops Support -> CRD Exercise Monitor) • When prompted by the Examiner, INSERT - Event 1 to Drift Control Rod 14-31 into the Core • When prompted by the Examiner, INSERT - Event 2 to Drift Control Rod Multiple Control Rods into the Core

Malfunctions	Description	Event	Severity	Delay	Initial set
rd07r1431	Rod 14-31 drifts into core	1	NA	0 sec	NA
rd07r1831	Rod 18-31 drifts into core	2	NA	0 sec	NA
rd07r2223	Rod 22-23 drifts into core	2	NA	30 sec	NA

Remotes	Description	Event	Severity	Delay	Initial set
NONE					

Overrides	Description	Event	Severity	Delay	Initial set
NONE					

Schedule File(s): **2104 NRC JPM a UNIT 3.SCH**
Event File(s): **2104 NRC JPM a UNIT 3.EVT**

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

- You are an Operator on Unit 3
- The Reactor is at 100% Power

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to perform 3-SR-3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3 Exercising a Partially Withdrawn Control Rod.

START TIME:__________

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 3-SR-3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3, before commencing the JPM.	
Step 1: 3-SR- 3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3, Exercising a Partially Withdrawn Control Rod **NOTE** 1) Problem details of any Control Rod issues observed during the performance of this surveillance must be recorded on Attachment 3 for further review and possible corrective maintenance. Slow/fast rod movements or settle times should be noted in the remarks section. 2) Section 7.3 is performed for all Partially Withdrawn Control Rods and performance of steps is represented by initialing the appropriate CRD exercised on Attachment 1. **CAUTIONS** 3) Any mispositioned Control Rod events will be dispositioned by following the direction contained within 3-AOI-85-7. 4) If a Control Cod moves unexpectedly one notch beyond its intended position, notify Unit Supervisor, obtain Unit Supervisor concurrence and return the Rod to its intended position. 5) At any time Core Thermal Power is less than or equal to 10%, entry into LCO 3.1.6 may be required. **7.3 Exercising a Partially Withdrawn Control Rod** **EXAMINER NOTE: The candidate may select any partially withdrawn Control Rod in any order.** [1] **SELECT** desired Control Rod by **DEPRESSING** appropriate 3-XS-85-40, CRD ROD SELECT pushbutton. Expected Action(s): Selects a partially withdrawn Control Rod.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: [2] **OBSERVE** the following for selected Control Rod: • **CHECK** CRD ROD SELECT pushbutton is brightly ILLUMINATED. • **CHECK** white light on the Full Core Display is ILLUMINATED. • **CHECK** Rod Out Permit light is ILLUMINATED. Expected Action(s): Verifies that the appropriate lights are illuminated.	______ SAT ______UNSAT ______N/A
Step 3: [3] **INSERT** Control Rod one notch by performing the following: [3.1] **PLACE** 3-HS-85-48, CRD CONTROL SWITCH in ROD IN and **RELEASE**. [3.2] **OBSERVE** Control Rod settles into the desired position and ROD SETTLE light extinguishes. Expected Action(s): Inserts withdrawn Control Rod one notch.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 4: [3.3] **IF** Control Rod failed to insert, **THEN PERFORM** the following: (Otherwise N/A) [3.4] **IF** the Control Rod unexpectedly inserts one notch beyond its intended position, **THEN PERFORM** the following: (Otherwise N/A) Expected Action(s): Marks Steps [3.3] and [3.4] as N/A.	_____SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 5: [4] **WITHDRAW** selected Control Rod one notch by performing the following: [4.1] **PLACE** 3-HS-85-48, CRD CONTROL SWITCH in ROD OUT NOTCH and **RELEASE**. [4.2] **OBSERVE** Control Rod settles into the desired position and ROD SETTLE light extinguishes. Expected Action(s): Withdraws withdrawn Control Rod one notch.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 6: [4.3] **IF** Control Rod failed to withdraw, **THEN PERFORM** the following: (Otherwise N/A) [4.4] **IF** Control Rod unexpectedly withdraws one notch beyond its intended position, **THEN PERFORM** the following: (Otherwise N/A) Expected Action(s): Marks Steps [4.3] and [4.4] as N/A.	

STEP / STANDARD	SAT / UNSAT
Step 7: [5] **DOCUMENT** completion of Control Rod test as follows: [5.1] PERFORMER • **INITIAL** Attachment 1 (Control Rod Exercise Data Sheet) in the box corresponding to the Control Rod coordinates for the Control Rod just exercised to document proper movement and CRD latching. [5.2] Concurrent Verifier (CV) • **ENSURE** rod inserted and returned to its original position. • **INITIAL** Attachment 2 (Control Rod Concurrent Verifier (CV) Check) in the box corresponding to the Control Rod coordinates for the Rod just exercised. **EXAMINER NOTE: If prompted by applicant for Concurrent Verification, state "Attachment 2 Concurrent Verification has been completed by another Operator."** Expected Action(s): Initials Attachment 1 for exercised Control Rod and continues to exercise Rods.	_____SAT _____UNSAT _____N/A

EXAMINER NOTES:

- **Perform above actions for at least two Control Rods.**
- **Begin Alternate Path - when satisfied with the number of rod manipulations direct Simulator Booth Operator to insert Event 1 for Control Rod 14-31 Drift In.**

DRIVER NOTE:

When requested by the Examiner, insert Event 1 to cause Control Rod 14-31 to drift in.

STEP / STANDARD	SAT / UNSAT
Step 8: Candidate recognizes Control Rod 14-31 drifting in and responds per 3-AOI-85-5, Rod Drift In. **4.2 Subsequent Actions** [2] **IF** a Control Rod is moving (or has moved) from its intended position without operator actions, **THEN INSERT** the Control Rod to position 00 using CONTINUOUS IN. (Otherwise N/A) [3] **IF** a Control Rod Block occurs during rod insertion due to Rod Worth Minimizer, **THEN BYPASS** the RWM per step 4.2[1] above. (Otherwise N/A) Expected Action(s): Responds in accordance with 3-AOI-85-5, Rod Drift In, and inserts Control Rod 14-31 to full in position as indicated by position 00 indication.	______ SAT ______UNSAT ______N/A
DRIVER NOTE: **When Control Rod 14-31 reaches position 00, verify that malfunction rd07r1431 (14-31 Control Rod Drift In) is deleted by the simulator setup so that CONTROL ROD DRIFT, (2-9-5A, WINDOW 28) can be reset.**	
EXAMINER NOTES: **Control Rod 14-31 will settle into position 00.** **The Candidate may or may not reset the drift lights and alarms.** **Expected Alarms:** • **CONTROL ROD WITHDRAWAL BLOCK, (2-9-5A, WINDOW 7)** • **ROD BLOCK MONITOR (RBM) DOWNSCALE, (2-9-5A, WINDOW 31)**	
Step 9: [4] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern. Expected Action(s): Candidate notifies Reactor Engineer to Evaluate Core Thermal limits and Preconditioning Limits for the current Control Rod Pattern.	_____SAT ______UNSAT ______N/A
CUE: If contacted as the Reactor Engineer acknowledge any direction or information given.	

STEP / STANDARD	SAT / UNSAT
Step 10: [5] **IF** another Control Rod Drift occurs before Reactor Engineering completes the evaluation, • **THEN MANUALLY SCRAM** the Reactor and enter 3-AOI-100-1, Reactor SCRAM [6] **CHECK** Thermal Limits on ICS (RUNMON). Expected Action(s): Reviews step and may inform the Nuclear Unit Senior Operator (NUSO) of the requirement to insert a Reactor SCRAM if another Control Rod drifts.	_____SAT _____UNSAT _____N/A
EXAMINER NOTE: Acknowledge applicant report.	
EXAMINER NOTE: When ready for multiple rod drifts, direct the Simulator Booth Operator to insert Event 2 (Control Rod 06-31 Rod Drift, and 30 seconds later Control Rod 10-39 Rod Drift).	
DRIVER NOTE: When requested by the Examiner, insert Event 2 (06-31 Control Rod Drift In). 30 seconds later, Control Rod 10-39 will drift in if a Reactor SCRAM has not been inserted.	
Step 11: **4.1 Immediate Actions** [1] **IF** multiple Control Rods are drifting into core, **THEN MANUALLY SCRAM** Reactor. **REFER TO** 3-AOI-100-1. Expected Action(s): Recognizes multiple Control Rods are drifting into the Core and inserts a manual Reactor SCRAM in accordance with 3-AOI-85-5, Rod Drift In.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: When informed that multiple Control Rods are drifting, acknowledge the report. At any point following the Reactor SCRAM, request that the Driver place the Simulator in FREEZE and inform the candidate "Another Operator will continue with the Reactor SCRAM actions. This completes your task".	

STOP TIME: ______

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

- You are an Operator on Unit 3.
- The Reactor is at 100% Power

INITIATING CUES:

The Unit Supervisor directs you to perform 3-SR-3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, Step 7.3 Exercising Partially Withdrawn Control Rod.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Inject to the Reactor in accordance with 2-EOI-Appendix-5C, Injection System Lineup - RCIC		
JPM NUMBER:		18A-U2	REVISION:	9	

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-000-EM-31 / Lineup Injection Systems-RCIC in accordance with EOI Appendix 5C			
K/A RATINGS:	RO: 3.7 SRO: 3.7			
K/A No. & STATEMENT:	217000 Reactor Core Isolation Cooling System (RCIC) A1.01; Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: RCIC Flow			
RELATED PRA INFORMATION:	CDF Contribution = 8%			
SAFETY FUNCTION:	2			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 5 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 18A-U2

RO ________ SRO ________ DATE: _____________

TASK STANDARD: Examinee is expected to inject to the Reactor using the Reactor Core Isolation (RCIC) System, recognize a failure of the Automatic Flow Controller, and take action to re-establish flow to raise Reactor Water Level.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 2-EOI-APPENDIX-5C

VALIDATION TIME: 5 min

PERFORMANCE TIME: ____________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ______________________________ DATE: ___________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
7	08/08/17	All	JPM converted to new format
8	11/09/20	All	Updated JPM
9	1/14/21	All	JPM update

Procedure Revisions

Procedure	Revision
2-EOI-APP-5C	7

SIMULATOR SETUP

IC	N/A
Exam IC	280

Console Operator Instructions	• **Reset to IC 280** • **Verify RCIC Controller set to 620 gpm and is in AUTO** • **Run schedule file ILT 2104 NRC JPM –b– 18A.SCH. Verify that Event File ILT 2104 NRC JPM –b– 18A.evt loads** • **Place the simulator in RUN once the candidate states that the task is understood** • **During the JPM, verify that the RCIC Flow Controller Fails 30 seconds after speed rises above 3500 RPM.**

Malfunctions	Description	Event	Severity	Delay	Initial set
RC04	RCIC AUTOMATIC FLOW CONTROLLER FAILURE (FIC-71-36A)	1	0	30	N/A

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 2. A Manual Reactor SCRAM was inserted, and the following conditions exist:

- The High Pressure Coolant Injection (HPCI) System is INOPERABLE
- 2C Reactor Feedwater Pump (RFPT) was controlling Reactor Water Level, but has tripped
- No RFPT can be started
- The Unit Supervisor has entered 2-EOI-1, RPV Control, on low Reactor Water Level

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to restore Reactor Water Level to (+) 2 to (+) 51 inches, using the Reactor Core Isolation Cooling (RCIC) System in accordance with 2-EOI-APPENDIX-5C, Injection System Lineup – RCIC.

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **PERFORM** the following EOI appendices, if necessary: • Appendix-16A, Bypassing RCIC Low RPV Pressure Isolation • Appendix-16K, Bypassing RCIC High Temperature Isolations Expected Action(s): Determines that neither EOI Appendix is required to run RCIC and continues in this procedure.	______ SAT ______UNSAT ______N/A
Step 2: [2] **ENSURE RESET** auto isolation logic using 2-XS-71-51A(B), RCIC AUTO-ISOL LOGIC A (B) RESET pushbuttons. Expected Action(s): Determines that no isolation signal is present and verifies that auto isolation logic is reset.	______ SAT ______UNSAT ______N/A
Step 3: [3] **ENSURE RESET** and **OPEN** 2-FCV-71-9, RCIC TURB TRIP/THROTTLE VALVE. Expected Action(s): Verifies OPEN 2-FCV-71-9, RCIC TURBINE TRIP/THROTTLE VALVE.	______ SAT ______UNSAT ______N/A
Step 4: [4] **ENSURE** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with a setpoint at 620 gpm. Expected Action(s): Verifies that 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, is in AUTO and set to 620 gpm.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 5: [5] **OPEN** 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE Expected Action(s): Opens 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE.	_____ SAT ______UNSAT ______N/A
Step 6: [6] **OPEN** 2-FCV-71-39, RCIC PUMP INJECTION VALVE Expected Action(s): Opens 2-FCV-71-39, RCIC PUMP INJECTION VALVE.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 7: [7] **OPEN** 2-FCV-71-25, RCIC LUBE OIL COOLING WTR VALVE Expected Action(s): Opens 2-FCV-71-25, RCIC LUBE OIL COOLING WTR VALVE.	_____ SAT ______UNSAT ______N/A
Step 8: [8] **PLACE** 2-HS-71-31A, RCIC VACUUM PUMP, in START. Expected Action(s): Places 2-HS-71-31A, RCIC VACUUM PUMP, in START.	_____SAT ______UNSAT ______N/A

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td><u>Step 9</u>:

CAUTIONS
1) Operating RCIC turbine below 2100 RPM may result in unstable system operation and equipment damage.
2) High Suppression Chamber pressure may trip RCIC.
3) Operating RCIC Turbine with suction temperatures above 240°F may result in equipment damage.

[9] OPEN 2-FCV-71-8, RCIC TURBINE STEAM SUPPLY VLV, to start RCIC turbine.

<u>Expected Action(s):</u>

Opens 2-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td><u>Step 10:</u>

[10] CHECK proper RCIC operation by observing the following:

A. Speed accelerates above 2100 rpm
B. Flow to RPV controlled automatically at 620 gpm
C. 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE, closes as flow rises above 120 gpm

<u>Expected Action(s):</u>

Verifies that:
A. RCIC turbine accelerates to >2100 rpm
B. RCIC flow stabilizes at 620 gpm
C. 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE closes</td><td>______SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER NOTE: Beginning of Alternate Path. Thirty (30) seconds after RCIC Speed exceeds 3500 rpm, RCIC Flow Controller automatic operation will fail.</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
<u>Step 11:</u> [11] **ADJUST** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. <u>Expected Action(s):</u> Controls injection with RCIC. After thirty (30) seconds, determines that the automatic flow controller is failed, and in accordance with OPDP-1, Conduct of Operations, Section 3.3.5, Manual Control of Automatic Systems, places 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL in manual. Adjusts the setpoint in manual as necessary to obtain the flow required to control Reactor Water Level.	**Critical Step** ______ SAT ______UNSAT ______N/A
<u>Step 12:</u> [12] **IF** <u>BOTH</u> of the following exist: • RCIC Initiation signal is <u>NOT</u> present, **AND** • RCIC flow is below 60 gpm, **THEN ENSURE OPEN** 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE. <u>Expected Action(s):</u> Verifies that a RCIC initiation signal is not present as indicated by the amber lamp 2-IL-71-52, RCIC AUTO INITIATION, being extinguished. If flow drops <60 gpm following the Flow Controller failure, the candidate verifies that the 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE, is OPEN.	______SAT ______UNSAT ______N/A
Examiner Note: It is not necessary for the candidate to obtain a Reactor Water Level of > (+) 2 inches. A rising trend in Reactor Water Level will suffice.	
CUE: Another Operator will take over Reactor Water Level Control.	

STOP TIME: ______

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 2. A Manual Reactor SCRAM was inserted, and the following conditions exist:

- The High Pressure Coolant Injection (HPCI) System is INOPERABLE
- 2C Reactor Feedwater Pump (RFPT) was controlling Reactor Water Level, but has tripped
- No RFPT can be started
- The Unit Supervisor has entered 2-EOI-1, RPV Control, on low Reactor Water Level

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to restore Reactor Water Level to (+) 2 to (+) 51 inches, using the Reactor Core Isolation Cooling (RCIC) System in accordance with 2-EOI-APPENDIX-5C, Injection System Lineup – RCIC.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Inject to the Reactor in accordance with 3-EOI-Appendix-5C, Injection System Lineup - RCIC
JPM NUMBER:	18A-U3	REVISION:	9

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO

TASK NUMBER / TASK TITLE(S):	U-000-EM-31 / Lineup Injection Systems-RCIC in accordance with EOI Appendix 5C
K/A RATINGS:	RO: 3.7 SRO: 3.7
K/A No. & STATEMENT:	217000 Reactor Core Isolation Cooling System (RCIC) A1.01; Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: RCIC Flow
RELATED PRA INFORMATION:	CDF Contribution = 4%
SAFETY FUNCTION:	2

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 5 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ______________

Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________

Validator *Date*

Approved by: ______________________ ______________

Site Training Management *Date*

Approved by: ______________________ ______________

Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 18A-U3

RO _______ SRO _______ DATE: ____________

TASK STANDARD: Examinee is expected to inject to the Reactor using the Reactor Core Isolation (RCIC) System, recognize a failure of the Automatic Flow Controller, and take action to re-establish flow to raise Reactor Water Level.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 3-EOI-APPENDIX-5C

VALIDATION TIME: 5 min

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained
THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
7	08/08/17	All	JPM converted to new format
8	11/09/20	All	Updated JPM
9	1/14/21	All	JPM update

Procedure Revisions

Procedure	Revision
3-EOI-APP-5C	5

SIMULATOR SETUP

IC	N/A
Exam IC	282

Console Operator Instructions	• **Reset to IC 282** • **Verify RCIC Controller set to 620 gpm and is in AUTO** • **Run schedule file ILT 2104 NRC JPM –b– 18A.SCH. Verify that Event File ILT 2104 NRC JPM –b– 18A.evt loads** • **Place the simulator in RUN once the candidate states that the task is understood** • **During the JPM, verify that the RCIC Flow Controller Fails 30 seconds after speed rises above 3500 rpm.**

Malfunctions	Description	Event	Severity	Delay	Initial set
RC04	RCIC AUTOMATIC FLOW CONTROLLER FAILURE (FIC-71-36A)	1	0	30	N/A

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 3. A Manual Reactor SCRAM was inserted, and the following conditions exist:

- The High Pressure Coolant Injection (HPCI) System is INOPERABLE
- 3C Reactor Feedwater Pump (RFPT) was controlling Reactor Water Level, but has tripped
- No RFPT can be started
- The Unit Supervisor has entered 3-EOI-1, RPV Control, on low Reactor Water Level

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to restore Reactor Water Level to (+) 2 to (+) 51 inches, using the Reactor Core Isolation Cooling (RCIC) System in accordance with 3-EOI-APPENDIX-5C, Injection System Lineup – RCIC.

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **PERFORM** the following EOI appendices, if necessary: • Appendix-16A, Bypassing RCIC Low RPV Pressure Isolation • Appendix-16K, Bypassing RCIC High Temperature Isolations Expected Action(s): Determines that neither EOI Appendix is required to run RCIC and continues in this procedure.	______ SAT ______UNSAT ______N/A
Step 2: [2] **ENSURE RESET** auto isolation logic using 3-XS-71-51A(B), RCIC AUTO-ISOL LOGIC A (B) RESET pushbuttons. Expected Action(s): Determines that no isolation signal is present and verifies that auto isolation logic is reset.	______ SAT ______UNSAT ______N/A
Step 3: [3] **ENSURE RESET** and **OPEN** 3-FCV-71-9, RCIC TURB TRIP/THROTTLE VALVE. Expected Action(s): Verifies OPEN 3-FCV-71-9, RCIC TURBINE TRIP/THROTTLE VALVE.	______ SAT ______UNSAT ______N/A
Step 4: [4] **ENSURE** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with a setpoint at 620 gpm. Expected Action(s): Verifies that 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, is in AUTO and set to 620 gpm.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
<u>Step 5</u>: [5] **OPEN** 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE <u>Expected Action(s):</u> Opens 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE.	______ SAT ______UNSAT ______N/A
<u>Step 6</u>: [6] **OPEN** 3-FCV-71-39, RCIC PUMP INJECTION VALVE <u>Expected Action(s):</u> Opens 3-FCV-71-39, RCIC PUMP INJECTION VALVE.	**Critical Step** ______ SAT ______UNSAT ______N/A
<u>Step 7</u>: [7] **OPEN** 3-FCV-71-25, RCIC LUBE OIL COOLING WATER VALVE <u>Expected Action(s):</u> Opens 3-FCV-71-25, RCIC LUBE OIL COOLING WATER VALVE.	______ SAT ______UNSAT ______N/A
<u>Step 8</u>: [8] **PLACE** 3-HS-71-31A, RCIC VACUUM PUMP, in START. <u>Expected Action(s):</u> Places 3-HS-71-31A, RCIC VACUUM PUMP, in START.	_____SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 9: **CAUTIONS** 1) Operating RCIC turbine below 2100 rpm may result in unstable system operation and equipment damage. 2) High Suppression Chamber pressure may trip RCIC. 3) Operating RCIC Turbine with suction temperatures above 240°F may result in equipment damage. [9] **OPEN** 3-FCV-71-8, RCIC TURBINE STEAM SUPPLY VLV, to start RCIC turbine. Expected Action(s): Opens 3-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 10: [10] **CHECK** proper RCIC operation by observing the following: A. Speed accelerates above 2100 rpm. B. Flow to RPV controlled automatically at 620 gpm. C. 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE, closes as flow rises above 120 gpm. Expected Action(s): Verifies that: A. RCIC turbine accelerates to >2100 rpm. B. RCIC flow stabilizes at 620 gpm. C. 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE closes.	_____SAT ______UNSAT ______N/A
EXAMINER NOTE: Beginning of Alternate Path. Thirty (30) seconds after RCIC Speed exceeds 3500 rpm, RCIC Flow Controller automatic operation will fail.	

STEP / STANDARD	SAT / UNSAT
Step 11: [11] **ADJUST** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. Expected Action(s): Controls injection with RCIC. Determines that the automatic flow controller is failed, and in accordance with OPDP-1, Conduct of Operations, Section 3.3.5, Manual Control of Automatic Systems, places 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL in manual. Adjusts the setpoint in manual as necessary to obtain the flow required to control Reactor Water Level.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 12: [12] **IF** BOTH of the following exist: • RCIC Initiation signal is NOT present, **AND** • RCIC flow is below 60 gpm, **THEN ENSURE OPEN** 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE. Expected Action(s): Verifies that a RCIC initiation signal is not present as indicated by the amber lamp 3-IL-71-52, RCIC AUTO INITIATION, being extinguished. If flow drops <60 GPM following the Flow Controller failure, the candidate verifies that the 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE, is OPEN.	______SAT ______UNSAT ______N/A
Examiner Note: It is not necessary for the candidate to obtain a Reactor Water Level of > (+) 2 inches. A rising trend in Reactor Water Level will suffice.	
CUE: Another Operator will take over Reactor Water Level Control.	

STOP TIME: ______

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 3. A Manual Reactor SCRAM was inserted, and the following conditions exist:

- The High Pressure Coolant Injection (HPCI) System is INOPERABLE
- 3C Reactor Feedwater Pump (RFPT) was controlling Reactor Water Level, but has tripped
- No RFPT can be started
- The Unit Supervisor has entered 3-EOI-1, RPV Control, on low Reactor Water Level

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to restore Reactor Water Level to (+) 2 to (+) 51 inches, using the Reactor Core Isolation Cooling (RCIC) System in accordance with 3-EOI-APPENDIX-5C, Injection System Lineup – RCIC.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Alternate Generator Bus Duct Fans in accordance with 2-OI-47, Turbine-Generator System
JPM NUMBER:	743A-U2	REVISION:	1

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	N/A			
K/A RATINGS:	RO: 3.1 SRO: 2.9			
K/A No. & STATEMENT:	245000 Main Turbine Generator and Auxiliary Systems A4.02: Ability to manually operate and/or monitor in the control room: Generator controls			
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A			
SAFETY FUNCTION:	4			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 5 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ____________
Developer Date
(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator Date

Approved by: ______________________ ____________
Site Training Management Date

Approved by: ______________________ ____________
Site Training Program Owner Date

TVA	**Job Performance Measure (JPM)**

OPERATOR: ________________________________ JPM Number: _743A-U2__

RO _____ SRO _____ DATE: ______________

TASK STANDARD: Examinee is expected to alternate Turbine-Generator Bus Duct Cooling Fans and respond to a subsequent loss of both Bus Duct Cooling Fans, then insert a manual Reactor SCRAM, and trip the Main Generator.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-OI-47, 2-ARP-9-7A

VALIDATION TIME: _5 min_

PERFORMANCE TIME: ___________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	09/01/20	All	Initial issue
1	1/13/21	All	JPM update

Procedure Revisions

Procedure	Revision
2-OI-47	189
2-ARP-9-7A	35

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• **Reset to IC 28** • **Run schedule file: 2104 NRC JPM c UNIT 2.SCH** • **Verify event file 2104 NRC JPM c UNIT 2.EVT loads** • **Place the Simulator in RUN to ensure stable conditions** • **Ensure the examinee has been briefed on 2-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans**

Malfunctions	Description	Event	Severity	Delay	Initial set
EG13A	MAIN GENERATOR BUS DUCT COOLING 2A FAN FAILURE, HS-262-1A	1	N/A	2	N/A
EG13B	MAIN GENERATOR BUS DUCT COOLING 2B FAN FAILURE, HS-262-2A	1	N/A	10	N/A

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following plant conditions:

- 2A Bus Duct Cooling Fan must be shut down for mechanical inspection
- All **OUTSIDE** portions of 2-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans have been completed.
- Unit 2 Assistant Unit Operator is standing by locally at 2B Bus Duct Cooling Fan

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to alternate Turbine-Generator Bus Duct Cooling Fans in accordance with 2-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans.

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: 2-OI-47, Turbine-Generator System Section 6.11.1, Alternating Operating Bus Duct Cooling Fans **NOTES** 1) GENERATOR BUS DUCT FAN FAILURE, (3-9-7A, WINDOW 31) alarm, may be received when performing the following steps, due to securing one fan before starting the next. 2) EWR19-EEB-262-015 has determined acceptability for starting a Bus Duct Cooling Fan with reverse rotation of less than 100 rpm. **CAUTION** Starting Generator Bus Duct Heat Exchanger Fan while rotating greater than 100 rpm in the reverse direction creates potential to damage the fan. U-2 GENERATOR BUS DUCT FAN FAILURE, (3-9-7A, WINDOW 31) BUS DUCT HTX FAN 2A(2B) BACKDRAFT DMPR, 2-DMP-262-0051(0052), should preclude reverse rotation of the associated fan at speeds greater than 100 rpm. It is acceptable to try restarting the fan one time immediately if the breaker trips on the first attempt. [1] **Starting Bus Duct Cooling Fan A/Stopping Fan B** Expected Action(s): Marks this step as N/A, as 2B Bus Duct Fan is being started as given in the Initial Conditions. Proceeds to Step [2].	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: [2] Starting Bus Duct Cooling Fan B/Stopping Fan A Steps [2.1] through [2.3] are complete **EXAMINER CUE: If the examinee contacts the Assistant Unit Operator (AUO) for information concerning any portion of Steps [2.1] through [2.3], inform the examinee that the step(s) is(are) complete. If requested, Fan 2B is rotating less than 100 RPM.** Expected Action(s): Marks these steps as complete as given in the Initial Conditions.	______ SAT ______UNSAT ______N/A
Step 3: **CAUTION** Dual fan operation should be limited to ≤ 5 minutes with inlet vane dampers full open ref. EWR20MEB262128 Rev. 0 [2.4] **MOMENTARILY PLACE** 2-HS-262-0002A, GENERATOR BUS DUCT HX FAN B, in START on Panel 2-9-8. • **CHECK** for proper fan operation Expected Action(s): Starts Bus Duct Fan 2B by placing 2-HS-262-0002A, GENERATOR BUS DUCT HX FAN B, in START.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 4: [2.5] **MOMENTARILY PLACE** 2-HS-262-0001A, GENERATOR BUS DUCT HX FAN A in STOP on Panel 2-9-8. Expected Action(s): Stops Bus Duct Fan 2A by placing 2-HS-262-0001A, GENERATOR BUS DUCT HX FAN, in STOP.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: (BEGIN ALTERNATE PATH) 2B Bus Duct Cooling Fan will trip 10 seconds after 2A Fan is stopped. If the examinee attempts to re-start 2A Fan, it will not start.	
Step 5: **EXAMINER NOTE: It is acceptable for the examinee to reference the Alarm Response Procedure (ARP) in response to the loss of both Bus Duct Fans. There is no Abnormal Operating Procedure (AOI) for this event.** When 2B Bus Duct Fan Trips: Alarm Response Procedure, 2-ARP-9-7A GENERATOR BUS DUCT FAN FAILURE, (2-9-7A, WINDOW 31) Operator Action: A. CHECK Main Bus Cooling Fans, 2-HS-262-1A or 2-HS-262-2A, indicates running on Panel 2-9-8. Expected Action(s): Verifies that **neither** Bus Duct Fan is running.	______ SAT ______UNSAT ______N/A
Step 6: **CAUTION** Starting Generator Bus Duct Heat Exchanger Fan while rotating greater than 100 rpm in the reverse direction creates potential to damage the fan. U-2 GENERATOR BUS DUCT HTX FAN 2A(2B) BACKDRAFT DMPR, 2-DMP-262-0051(0052), should preclude reverse rotation of the associated fan at speeds greater than 100 rpm. It is acceptable to try restarting the fan one time immediately if the breaker trips on the first attempt. B. **START** 2-HS-262-1A(2A), GENERATOR BUS DUCT HX FAN A(B) using on Panel 2-9-8 to start the standby fan. Expected Action(s): The examinee may attempt to start 2A Bus Duct Fan.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 7: C. **IF** no Fans are operating and the Generator is tied to the grid and loaded to greater than the self-cooled bus rating of 16,500 amps **THEN PERFORM** the following: 1. **IMMEDIATELY INSERT** a manual Reactor SCRAM. Expected Action(s): Verifies Generator amps are above 16,500 and inserts a manual Reactor SCRAM.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 8: 2. **TRIP** the Main Generator Expected Action(s): Trips the Main Generator.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: Following the Reactor SCRAM (a SCRAM report is not required) and Main Turbine Trip, inform the examinee "Another Operator will address plant conditions. This completes your task".	

STOP TIME: ___________

Job Performance Measure (JPM)

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following plant conditions:

- 2A Bus Duct Cooling Fan must be shut down for mechanical inspection
- All outside portions of 2-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans have been completed
- Unit 2 Assistant Unit Operator is standing by locally at 2B Bus Duct Cooling Fan

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to alternate Turbine-Generator Bus Duct Cooling Fans in accordance with 2-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Alternate Generator Bus Duct Fans in accordance with 3-OI-47, Turbine-Generator System
JPM NUMBER:	743A-U3	REVISION:	1

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	N/A			
K/A RATINGS:	RO: 3.1 SRO: 2.9			
K/A No. & STATEMENT:	245000 Main Turbine Generator and Auxiliary Systems A4.02: Ability to manually operate and/or monitor in the control room: Generator controls			
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A			
SAFETY FUNCTION:	4			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 5 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

TVA	**Job Performance Measure (JPM)**

OPERATOR: ______________________________ JPM Number: _743A-U3__

RO _____ SRO _____ DATE: ______________

TASK STANDARD: Examinee is expected to alternate Turbine-Generator Bus Duct Cooling Fans and respond to a subsequent loss of both Bus Duct Cooling Fans, then insert a manual Reactor SCRAM, and trip the Main Generator.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 3-OI-47, 3-ARP-9-7A

VALIDATION TIME: __5 min_

PERFORMANCE TIME: ___________

COMMENTS: ____

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Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ______________________________ DATE: __________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	09/01/20	All	Initial issue
1	1/13/21	All	JPM update

Procedure Revisions

Procedure	Revision
3-OI-47	126
3-ARP-9-7A	30

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• **Reset to IC 28** • **Run schedule file: 2104 NRC JPM c UNIT 3.SCH** • **Verify event file 2104 NRC JPM c UNIT 3.EVT loads** • **Place the Simulator in RUN to ensure stable conditions** • **Ensure the examinee has been briefed on 3-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans**

Malfunctions	Description	Event	Severity	Delay	Initial set
EG13A	MAIN GENERATOR BUS DUCT COOLING 3A FAN FAILURE, HS-262-1A	1	N/A	2	N/A
EG13B	MAIN GENERATOR BUS DUCT COOLING 2B FAN FAILURE, HS-262-3A	1	N/A	10	N/A

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 3 with the following plant conditions:

- 3A Bus Duct Cooling Fan must be shut down for mechanical inspection
- All **OUTSIDE** portions of 3-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans have been completed.
- Unit 3 Assistant Unit Operator is standing by locally at 3B Bus Duct Cooling Fan

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to alternate Turbine-Generator Bus Duct Cooling Fans in accordance with 3-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans.

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: 3-OI-47, Turbine-Generator System Section 6.11.1, Alternating Operating Bus Duct Cooling Fans **NOTES** 1) GENERATOR BUS DUCT FAN FAILURE, (3-9-7A, WINDOW 31) alarm, may be received when performing the following steps, due to securing one fan before starting the next. 2) EWR19-EEB-262-015 has determined acceptability for starting a Bus Duct Cooling Fan with reverse rotation of less than 100 rpm. **CAUTION** Starting Generator Bus Duct Heat Exchanger Fan while rotating greater than 100 rpm in the reverse direction creates potential to damage the fan. U-3 GENERATOR BUS DUCT FAN FAILURE, (3-9-7A, WINDOW 31) BUS DUCT HTX FAN 3A(3B) BACKDRAFT DMPR, 3-DMP-262-0051(0052), should preclude reverse rotation of the associated fan at speeds greater than 100 rpm. It is acceptable to try restarting the fan one time immediately if the breaker trips on the first attempt. [1] **Starting Bus Duct Cooling Fan A/Stopping Fan B** Expected Action(s): Marks this step as N/A, as 3B Bus Duct Fan is being started as given in the Initial Conditions. Proceeds to Stop [2].	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: [2] **PERFORM** the following to **SWAP** from Bus Duct Cooling Fan A to Fan B: (Otherwise N/A) Steps [2.1] through [2.3] are complete **EXAMINER CUE: If the examinee contacts the Assistant Unit Operator (AUO) for information concerning any portion of Steps [2.1] through [2.3], inform the examinee that the step(s) is(are) complete. If requested, Fan 3B is rotating less than 100 RPM.** Expected Action(s): Marks these steps as complete as given in the Initial Conditions.	______ SAT ______UNSAT ______N/A
Step 3: **CAUTION** Dual fan operation should be limited to ≤ 5 minutes with inlet vane dampers full open ref. EWR20MEB262128 Rev. 0 [2.4] On Panel 9-7, **MOMENTARILY PLACE** 3-HS-262-0002A, GEN BUS DUCT HX FAN B, in START • CHECK for proper fan operation Expected Action(s): STARTS Bus Duct Fan 3B by placing 3-HS-262-0002A, GENERATOR BUS DUCT HX FAN B, in START.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 4: [2.5] On Panel 9-7, **MOMENTARILY PLACE** 3-HS-262-0001A, GEN BUS DUCT HX FAN A, in STOP. Expected Action(s): STOPS Bus Duct Fan 3A by placing 3-HS-262-0001A, GENERATOR BUS DUCT HX FAN A, in STOP.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: (BEGIN ALTERNATE PATH) 3B Bus Duct Cooling Fan will trip 10 seconds after 3A Fan is stopped. If the examinee attempts to re-start 3A Fan, it will not start.	
Step 5: **EXAMINER NOTE: It is acceptable for the examinee to reference the Alarm Response Procedure (ARP) in response to the loss of both Bus Duct Fans. There is no Abnormal Operating Procedure (AOI) for this event.** When 3B Bus Duct Fan Trips: Alarm Response Procedure, 3-ARP-9-7A GENERATOR BUS DUCT FAN FAILURE, (3-9-7A, WINDOW 31) Operator Action: A. CHECK Main Bus Cooling Fans, 3-HS-262-1A or 3-HS-262-2A, indicates running on Panel 3-9-8. Expected Action(s): Verifies that **neither** Bus Duct Fan is running.	______ SAT ______UNSAT ______N/A
Step 6: **CAUTION** Starting Generator Bus Duct Heat Exchanger Fan while rotating greater than 100 rpm in the reverse direction creates potential to damage the fan. U-3 GENERATOR BUS DUCT HTX FAN 3A(3B) BACKDRAFT DMPR, 3-DMP-262-0051(0052), should preclude reverse rotation of the associated fan at speeds greater than 100 rpm. It is acceptable to try restarting the fan one time immediately if the breaker trips on the first attempt. B. **START** 3-HS-262-1A(2A), GENERATOR BUS DUCT HX FAN A(B) using, on Panel 3-9-8 to start the standby fan. Expected Action(s): The examinee may attempt to start 3A Bus Duct Fan.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
<u>Step 7:</u> C. **IF** no Fans are operating and the Generator is tied to the grid and loaded to greater than the self-cooled bus rating of 16,500 amps **THEN PERFORM** the following: 1. **IMMEDIATELY INSERT** a manual Reactor SCRAM. <u>Expected Action(s):</u> Verifies Generator amps are above 16,500 and inserts a manual Reactor SCRAM.	**Critical Step** ______ SAT ______UNSAT ______N/A
<u>Step 8:</u> 2. **TRIP** the Main Generator <u>Expected Action(s):</u> Trips the Main Generator.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: Following the Reactor SCRAM (SCRAM report is not required) and Main Turbine Trip, inform the examinee "Another Operator will address plant conditions. This completes your task".	

STOP TIME: ___________

TVA	**Job Performance Measure (JPM)**

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 3 with the following plant conditions:

- 3A Bus Duct Cooling Fan must be shut down for mechanical inspection
- All **OUTSIDE** portions of 3-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans have been completed.
- Unit 3 Assistant Unit Operator is standing by locally at 3B Bus Duct Cooling Fan

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to alternate Turbine-Generator Bus Duct Cooling Fans in accordance with 3-OI-47, Turbine-Generator System, Section 6.11.1, Alternating Operating Bus Duct Cooling Fans.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Purge the Drywell with the Primary Containment Purge Filter Fan in accordance 2-OI-64, Primary Containment System
JPM NUMBER:	747-U2	REVISION:	1

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO

TASK NUMBER / TASK TITLE(S):	U-064-NO-09 / Place Primary Containment Ventilation in Service
K/A RATINGS:	RO: 3.6 SRO: 3.6
K/A No. &STATEMENT:	223001 Primary Containment System and Auxiliaries A4.05: Ability to manually operate and/or monitor in the Control Room: Containment/Drywell oxygen concentration
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A
SAFETY FUNCTION:	5

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 15 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

OPERATOR: ______________________________ JPM Number: 747-U2

RO _______ SRO _______ DATE: ____________

TASK STANDARD: The Examinee is expected to perform Control Room operations required to air purge the Drywell for Primary Containment entry during an outage.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 2-OI-64, Section 8.2

VALIDATION TIME: 15 min

PERFORMANCE TIME: ___________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	10/27/2020	All	Initial issue
1	1/20/21	All	JPM update

Procedure Revisions

Procedure	Revision
2-OI-64	129

SIMULATOR SETUP

IC	28
Exam IC	281

Console Operator Instructions	• **Reset to IC 281** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 2-OI-64, Primary Containment System, Section 8.2** • **When requested by the candidate, insert Event 1 to start the Containment Purge Filter Fan**

Remotes	Description	Event	Severity	Delay	Initial set
PC06	CTMT PURGE FILTER FAN HS-64-131	1	START	N/A	N/A

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are the Balance of Plant Operator on Unit 2 with the following plant conditions:

- Unit 2 is currently at 100% RTP
- Unit 2 is being shut down for an outage. Reactor Power reduction will start on the next shift
- The Drywell is to be purged with air for Primary Containment entry
- Drywell Control Air has been aligned to Plant Control Air

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to air purge the Drywell for Primary Containment entry in accordance with 2-OI-64, Primary Containment System, Section 8.2 Purging the Drywell with Primary Containment Purge Filter Fan, starting at Step [8].

NOTE: All Precautions and Limitations **AND** Pre-Startup/Standby Readiness requirements have been met.

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: [8] **PLACE** 2-HS-64-142A, DRYWELL DIFFERENTIAL PRESSURE (DP) COMPRESSOR AND VALVES CONTROL, in **OFF.** Expected Action(s): Places 2-HS-64-142A, DRYWELL DIFFERENTIAL PRESSURE(DP) COMPRESSOR AND VALVES CONTROL, in OFF.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [9] **IF** Drywell Control Air (DWCA) is aligned to Containment Inerting Nitrogen Source, **THEN ALIGN** DWCA to Plant Control Air. **REFER TO** 2-OI-32A, Drywell Control Air System. Expected Action(s): Marks this step as N/A or completed. **EXAMINER CUE: If the candidate requests the status of Step [9], refer them to the Initial Conditions.**	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [10] **ENSURE CLOSED** the following valves (Panel 2-9-3): • 2-FCV-64-31, DRYWELL INBOARD ISOLATION VALVE • 2-FCV-64-34, SUPPRESSION CHAMBER INBOARD ISOLATION VALVE • 2-FCV-76-18, DRYWELL NITROGEN (N2) MAKEUP INBOARD ISOLATION VALVE • 2-FCV-76-19, SUPPRESSION CHAMBER NITROGEN (N2) MAKEUP INBOARD ISOLATION VALVE • 2-FCV-76-24, PRIMARY CONTAINMENT NITROGEN (N2) OUTBOARD ISOLATION VALVE • 2-FCV-64-32, SUPPRESSION CHAMBER VENT INBOARD ISOLATION VALVE • 2-FCV-64-33, SUPPRESSION CHAMBER VENT OUTBOARD ISOLATION VALVE • 2-FCO-64-36, DRYWELL/SUPPRESSION CHAMBER VENT TO STANDBY GAS TREATMENT (SGT) Expected Action(s): Ensures GREEN valve/operator position indicating lamps are illuminated for ALL of the above control switches in Step [10].	______ SAT ______UNSAT ______N/A

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
Step 4:

NOTES
1) If the Reactor MODE SWITCH is taken out of RUN during this procedure, the PRIMARY CONTAINMENT PURGE RUN MODE BYPASS switches shall be returned to the NORMAL position.
2) Tech Spec 3.6.1.1 shall be referred to before purging in the RUN MODE (MODE 1).
3) The following annunciators are expected when initiating Drywell purging due to gross failure on low Drywell Pressure.
• Reactor Protection System (RPS) ANALOG TRIP UNIT (ATU) TROUBLE 2-XA-99-1, (2-9-5B, WINDOW 23)
• Emergency Core Cooling System (ECCS) ANALOG TRIP UNIT TROUBLE 2-XA-71-60, (2-9-3E, WINDOW 30)

[11] IF the REACTOR MODE SWITCH is in RUN, THEN PLACE the following switches in the BYPASS position (Panel 2-9-3):
• 2-HS-64-24, PRIMARY CONTAINMENT (PC) PURGE DIVISION I RUN MODE BYPASS
• 2-HS-64-25, PRIMARY CONTAINMENT (PC) PURGE DIVISION II RUN MODE BYPASS

Expected Action(s):

Places the following switches in the BYPASS position:
• 2-HS-64-24, PRIMARY CONTAINMENT (PC) PURGE DIVISION I RUN MODE BYPASS
• 2-HS-64-25, PRIMARY CONTAINMENT (PC) PURGE DIVISION II RUN MODE BYPASS
</td><td>
Critical Step

______ SAT

______UNSAT

______N/A
</td></tr>
<tr><td>
Step 5:

[12] RECORD start time in Narrative log.

Expected Action(s):

Marks this step as completed.

EXAMINER CUE: If requested to RECORD start time, inform examinee that Step [12] is complete.
</td><td>
______ SAT

______UNSAT

______N/A
</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 6: [13] OPEN the following valves (Panel 2-9-3): • 2-FCV-64-29, DRYWELL VENT INBOARD ISOLATION VALVE, using 2-HS-64-29 • 2-FCV-64-30, DRYWELL VENT OUTBOARD ISOLATION VALVE, using 2-HS-64-30 Expected Action(s): Momentarily places 2-HS-64-29 and 2-HS-64-30 in the OPEN position and ensures the indicating lamp is illuminated RED valve position above the associated hand switch.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 7: [14] **MONITOR** Drywell Pressure (Panel 2-9-3). Expected Action(s): Monitors Drywell Pressure on various indications/recorders on Panel 2-9-3.	______ SAT ______UNSAT ______N/A
Step 8: [15] **START** 2-HS-64-131, CONTAINMENT PURGE FILTER FAN using (Reactor Building, El 621). Expected Action(s): Dispatches an Assistant Unit Operator (AUO) to start the Containment Purge Filter Fan. **DRIVER CUE: When contacted as an AUO to start the Containment Purge Filter Fan, insert Event 1, and inform the candidate that the Containment Purge Filter Fan is running.**	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
<u>Step 9</u>: [16] **OPEN** the following valves (Panel 2-9-3) A. 2-FCV-64-17, DRYWELL/SUPPRESSION CHAMBER AIR PURGE ISOLATION VLV, using 2-HS-64-17 B. 2-FCV-64-18, DRYWELL ATMOSPHERE SUPPLY INBOARD ISOLATION VLV, using 2-HS-64-18 <u>Expected Action(s):</u> Momentarily places 2-HS-64-17 and 2-HS-64-18 in the OPEN position and ensures the indicating lamp is illuminated RED valve position above the associated hand switch.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: After the completion of Step [16], inform the candidate "Another Operator will continue this procedure. This completes your task".	

STOP TIME: ______

	Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are the Balance of Plant Operator on Unit 2 with the following plant conditions:

- Unit 2 is currently at 100% RTP
- Unit 2 is being shut down for an outage. Reactor Power reduction will start on the next shift
- The Drywell is to be purged with air for Primary Containment entry
- Drywell Control Air has been aligned to Plant Control Air

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to air purge the Drywell for Primary Containment entry in accordance with 2-OI-64, Primary Containment System, Section 8.2 Purging the Drywell with Primary Containment Purge Filter Fan, starting at Step [8].

NOTE: All Precautions and Limitations **AND** Pre-Startup/Standby Readiness requirements have been met.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Purge the Drywell with the Primary Containment Purge Filter Fan in accordance 3-OI-64, Primary Containment System
JPM NUMBER:	747-U3	REVISION:	1

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-064-NO-09 / Place Primary Containment Ventilation in Service			
K/A RATINGS:	RO: 3.6 SRO: 3.6			
K/A No. &STATEMENT:	223001 Primary Containment System and Auxiliaries A4.05: Ability to manually operate and/or monitor in the Control Room: Containment/Drywell oxygen concentration			
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A			
SAFETY FUNCTION:	5			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 15 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

OPERATOR: ________________________________ JPM Number: 747-U3

RO _______ SRO _______ DATE: ____________

TASK STANDARD: The Examinee is expected to perform Control Room operations required to air purge the Drywell for Primary Containment entry during an outage.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 3-OI-64, Section 8.2

VALIDATION TIME: 15 min

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	10/27/20	All	Initial issue
1	1/20/21	All	JPM update

Procedure Revisions

Procedure	Revision
3-OI-64	67

SIMULATOR SETUP

IC	28
Exam IC	264

Console Operator Instructions	• **Reset to IC 264** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 3-OI-64, Primary Containment System, Section 8.2** • **When requested by the candidate, insert Event 1 to start the Containment Purge Filter Fan**

Remotes	Description	Event	Severity	Delay	Initial set
PC06	CTMT PURGE FILTER FAN HS-64-131	1	START	N/A	N/A

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are the Balance of Plant Operator on Unit 3 with the following plant conditions:

- Unit 3 is currently at 100% RTP
- Unit 3 is being shut down for an outage. Reactor Power reduction will start on the next shift
- The Drywell is to be purged with air for Primary Containment entry
- Drywell Control Air is aligned to Plant Control Air

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to air purge the Drywell for Primary Containment entry in accordance with 3-OI-64, Primary Containment System, Section 8.2 Purging the Drywell with Primary Containment Purge Filter Fan, starting at Step [8].

NOTE: All Precautions and Limitations **AND** Pre-Startup/Standby Readiness requirements have been met.

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: [8] **PLACE** 3-HS-64-142A, DRYWELL DIFFERENTIAL PRESSURE (DP) COMPRESSOR AND VALVES CONTROL, in **OFF.** Expected Action(s): Places 3-HS-64-142A, DRYWELL DIFFERENTIAL PRESSURE(DP) COMPRESSOR AND VALVES CONTROL, in OFF.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [9] **IF** Drywell Control Air **(**DWCA) is aligned to Containment Inerting Nitrogen Source, **THEN ALIGN** DWCA to Plant Control Air. **REFER TO** 3-OI-32A, Drywell Control Air System. Expected Action(s): Marks this step as N/A or completed. **EXAMINER CUE: If the candidate requests the status of Step [9], refer them to the Initial Conditions.**	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [10] **ENSURE CLOSED** the following valves (Panel 3-9-3): • 3-FCV-64-31, DRYWELL INBOARD ISOLATION VALVE • 3-FCV-64-34, SUPPRESSION CHAMBER INBOARD ISOLATION VALVE • 3-FCV-76-18, DRYWELL NITROGEN (N2) MAKEUP INBOARD ISOLATION VALVE • 3-FCV-76-19, SUPPRESSION CHAMBER NITROGEN (N2) MAKEUP INBOARD ISOLATION VALVE • 3-FCV-76-24, PRIMARY CONTAINMENT NITROGEN (N2) OUTBOARD ISOLATION VALVE • 3-FCV-64-32, SUPPRESSION CHAMBER VENT INBOARD ISOLATION VALVE • 3-FCV-64-33, SUPPRESSION CHAMBER VENT OUTBOARD ISOLATION VALVE • 3-FCO-64-36, DRYWELL/SUPPRESSION CHAMBER VENT TO STANDBY GAS TREATMENT (SGT) Expected Action(s): Ensures GREEN valve/operator position indicating lamps are illuminated for ALL of the above control switches in Step [10].	______ SAT ______UNSAT ______N/A

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 4</u>:

NOTES
1) If the Reactor Mode switch is taken out of RUN during this procedure, the PRIMARY CONTAINMENT PURGE RUN MODE BYPASS switches are returned to the NORMAL position.
2) Tech Spec 3.6.1.1 shall be referred to before purging in the RUN Mode (MODE 1).
3) The following annunciators are expected when initiating Drywell purging due to gross failure on low Drywell pressure.
• Reactor Protection System (RPS) ANALOG TRIP UNIT (ATU) TROUBLE 3-XA-99-1, (3-9-5B, WINDOW 23)
• Emergency Core Cooling System (ECCS) ANALOG TRIP UNIT TROUBLE 3-XA-71-60 , (3-9-3E, WINDOW 30)

[11] IF the REACTOR MODE SWITCH is in RUN, THEN PLACE the following switches in the BYPASS position (Panel 3-9-3):
• 3-HS-64-24, PRIMARY CONTAINMENT (PC) PURGE DIVISION I RUN MODE BYPASS
• 3-HS-64-25, PRIMARY CONTAINMENT (PC) PURGE DIVISION II RUN MODE BYPASS

<u>Expected Action(s):</u>

Places the following switches in the BYPASS position:
• 3-HS-64-24, PRIMARY CONTAINMENT (PC) PURGE DIVISION I RUN MODE BYPASS
• 3-HS-64-25, PRIMARY CONTAINMENT (PC) PURGE DIVISION II RUN MODE BYPASS
</td><td>
Critical Step

______ SAT

______UNSAT

______N/A
</td></tr>
<tr><td>
<u>Step 5</u>:

[12] RECORD start time in Narrative log.

<u>Expected Action(s):</u>

Marks this step as completed.

EXAMINER CUE: If requested to RECORD start time, inform examinee that Step [12] is complete.
</td><td>
______ SAT

______UNSAT

______N/A
</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 6: [13] OPEN the following valves (Panel 3-9-3): • 3-FCV-64-29, DRYWELL VENT INBOARD ISOLATION VLV, using 3-HS-64-29 • 3-FCV-64-30, DRYWELL VENT OUTBOARD ISOLATION VLV, using 3-HS-64-30 Expected Action(s): Momentarily places 3-HS-64-29 and 3-HS-64-30 in the OPEN position and ensures the indicating lamp is illuminated RED valve position above the associated hand switch.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 7: [14] **MONITOR** Drywell Pressure (Panel 3-9-3). Expected Action(s): Monitors Drywell Pressure on various indications/recorders on Panel 3-9-3 for lowering trend.	______ SAT ______UNSAT ______N/A
Step 8: [15] **START** 3-HS-64-131, CONTAINMENT PURGE FILTER FAN using (Reactor Building, El 621). Expected Action(s): Dispatches an Assistant Unit Operator (AUO) to start the Containment Purge Filter Fan. **DRIVER CUE: When contacted as an AUO to start the Containment Purge Filter Fan, insert Event 1, and inform the candidate that the Containment Purge Filter Fan is running.**	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 9: [16] **OPEN** the following valves (Panel 3-9-3) A. 3-FCV-64-17, DRYWELL/SUPPRESSION CHAMBER AIR PURGE ISOLATION VLV, using 3-HS-64-17. B. 3-FCV-64-18, DRYWELL ATMOSPHERE SUPPLY INBOARD ISOLATION VLV, using 3-HS-64-18. Expected Action(s): Momentarily places 3-HS-64-17 and 3-HS-64-18 in the OPEN position and ensures the indicating lamp is illuminated RED valve position above the associated hand switch.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: After the completion of Step [16], inform the candidate "Another Operator will continue this procedure. This completes your task".	

STOP TIME: ______

Job Performance Measure (JPM)

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are the Balance of Plant Operator on Unit 3 with the following plant conditions:

- Unit 3 is currently at 100% RTP
- Unit 3 is being shut down for an outage. Reactor Power reduction will start on the next shift
- The Drywell is to be purged with air for Primary Containment entry
- Drywell Control Air has been aligned to Plant Control Air

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to air purge the Drywell for Primary Containment entry in accordance with 3-OI-64, Primary Containment System, Section 8.2 Purging the Drywell with Primary Containment Purge Filter Fan, starting at Step [8].

NOTE: All Precautions and Limitations **AND** Pre-Startup/Standby Readiness requirements have been met.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Restore Offsite Power to 4KV Shutdown Board at Panel 0-9-23
JPM NUMBER:	631-U2	REVISION:	4

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO

TASK NUMBER / TASK TITLE(S):	U-082-NO-09 / Restore Offsite Power to 4KV Shutdown Board at Panel 9-23
K/A RATINGS:	RO: 3.4 SRO: 3.4
K/A No. &STATEMENT:	262001 A.C. Electrical Distribution; A4.02 Ability to manually operate and/or monitor in the control room: Synchroscope, including understanding of running and incoming voltages
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A
SAFETY FUNCTION:	6

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 18 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

TVA

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: _631-U2_

RO _____ SRO _____ DATE: _____________

TASK EXPECTED ACTION(S): The Examinee is expected to restore Offsite Power to 4KV Shutdown Board at Panel 0-9-23

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 0-OI-82

VALIDATION TIME: _20 minutes_

PERFORMANCE TIME: ___________

COMMENTS: ____
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Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
3	11/02/2020	All	Update JPM
4	01/25/2021	All	Procedure update

Procedure Revisions

Procedure	Revision
0-OI-82	174

SIMULATOR SETUP

IC	28
Exam IC	282

Console Operator Instructions	• Reset to IC 282 • Place the Simulator in RUN and verify stable conditions • Ensure the examinee has been briefed on 0-OI-82, Standby Diesel Generator System, Section 8.3, Restoring Offsite Power to 4KV Shutdown Board at Panel 9-23 • Ensure stopwatch is available

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A					

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following plant conditions:

- 2 hours ago, 4 KV Shutdown Board A was separated from Offsite Power and is being powered by Emergency Diesel Generator (EDG) A
- Assistant Unit Operators (AUOs) are standing by on station for support
- No inclement weather currently exists in the area

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to perform the following operation in accordance with 0-OI-82, Standby Diesel Generator System, Section 8.3, Restoring Offsite Power to 4KV Shutdown Board at Panel 9-23:

- Restore Offsite Power to 4KV Shutdown Board A using Normal Feeder Breaker 1614 while leaving EDG A in Parallel with System

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: 0-OI-82, Standby Diesel Generator (EDG) System Section 8.3 Restoring Offsite Power to 4KV Shutdown Board at Panel 9-23 **NOTES** The following list of 4KV Shutdown Board Normal and Alternate Feeder Breakers may be useful when performing this section: **Shutdown Board** — **A** Normal Feeder Breaker — 1614 Alternate Feeder Breaker — 1716 [1] **ENSURE** 4KV Shutdown Board A is being supplied power by its respective Diesel Generator as the only source of power. Expected Action(s): Ensures 4KV Shutdown Board A is being supplied by EDG A.	______ SAT ______UNSAT ______N/A
Step 2: [2] **ENSURE** the associated 4 KV Shutdown Board auto transfer lockout relay is tripped to MANUAL. **Diesel** — **Handswitch Name** — **Handswitch No.** — **Panel** A — 4KV SD BD A AUTO/LOCKOUT RESET — 0-211-A — 0-9-23-7 Expected Action(s): Ensures HS-0-211-A, 4KV SD BD A AUTO/LOCKOUT RESET, is tripped to manual on Panel 0-9-23-7.	______ SAT ______UNSAT ______N/A

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>Step 3:

[3] PLACE the synchroscope switch for the 4KV Shutdown Board feeder breaker that is to be paralleled with the Diesel Generator in ON.

Expected Action(s):

Places the synchroscope switch for the 4KV Shutdown Board feeder breaker that is to be paralleled with the Diesel Generator in ON.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td>Step 4:

[4] CHECK 4VKV Shutdown Bus 1(2) Voltage is between 3950 Volts and 4400 Volts and NOT undergoing abnormal voltage transients.

Expected Action(s):

Checks 4VKV Shutdown Bus 1 Voltage is between 3950 Volts and 4400 Volts and NOT undergoing abnormal voltage transients.</td><td>______ SAT

______UNSAT

______N/A</td></tr>
<tr><td>Step 5:

[5] CHECK associated incoming frequency is between 59 Hertz and 61 Hertz and NOT undergoing abnormal frequency transients.

<table>
<tr><th>Shutdown Bd</th><th>Instrument Name</th><th>Instrument No.</th><th>Panel</th></tr>
<tr><td>A or B</td><td>GENERATOR SYNC FREQUENCY</td><td>0-SI-82-AB</td><td>0-9-23-7</td></tr>
</table>

Expected Action(s):

Verifies incoming frequency is between 59 Hertz and 61 Hertz and NOT undergoing abnormal frequency transients using 0-SI-82-AB, GENERATOR SYNC FREQUENCY.</td><td>______ SAT

______UNSAT

______N/A</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 6: **CAUTION** **DO NOT** parallel the Diesel Generators with an unstable Offsite source or during inclement weather (e.g., lightning, heavy winds). [6] **IF** 4KV Shutdown Bus 1 (2) is experiencing abnormal voltage or frequency conditions, **THEN PERFORM** the following: Expected Action(s): Verifies that there are no abnormal voltage or frequency conditions. As given in the Initial Conditions, there is no inclement weather in the area.	______ SAT ______UNSAT ______N/A
Step 7: **CAUTION** Only one Unit 1 and 2 Diesel Generator at a time is allowed to be operated in Parallel with System. [7] **PULL and PLACE** the associated Diesel Generator mode selector switch in PARALLELED WITH SYSTEM. **Diesel** | **Handswitch Name** | **Handswitch No.** | **Panel** A | DG A MODE SELECT | 0-HS-82-A/5A | 0-9-23-7 Expected Action(s): Pulls 0-HS-82-A/5A, DG A MODE SELECT SWITCH and places it in PARALLELED WITH SYSTEM.	**Critical Step** ______ SAT ______UNSAT ______N/A

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 8:</u>

CAUTION
Failure of the PARALLELED WITH SYSTEM light to illuminate in the following step could indicate that the DG is still in SINGLE UNIT operation and result in overload when the DG output breaker is closed.

[8] RELEASE the Diesel Generator Mode Selector Switch and OBSERVE PARALLELED WITH SYSTEM light illuminated.

<u>Expected Action(s):</u>

Releases 0-HS-82-A/5A, DG A MODE SELECT SWITCH and observes PARALLELED WITH SYSTEM light is illuminated.
</td><td>
______ SAT
______UNSAT
______N/A
</td></tr>
<tr><td>
<u>Step 9:</u>

[9] PLACE the associated Diesel Generator NFPA 805 Mode Switch in ENABLE.

<table>
<tr><th>Diesel</th><th>Handswitch Name</th><th>Handswitch No.</th><th>Panel</th></tr>
<tr><td>A</td><td>NFPA 805 MODE SWITCH</td><td>0-43-211-000A/23</td><td>4kV SD BD A Compt 23</td></tr>
</table>
<u>Expected Action(s):</u>

Directs the AUO to place 0-43-211-000A/23, NFPA 805 MODE SWITCH for EDG A on Panel 4KV SD BD A Compt 23 in ENABLE.
</td><td>
______ SAT
______UNSAT
______N/A
</td></tr>
<tr><td colspan="2">DRIVER NOTE: When directed to place 0-43-211-000A/23, NFPA 805 MODE SWITCH for EDG A on Panel 4KV SD BD A Compt 23 in ENABLE, verify that Remote Function ED32A is ENABLED, and report that the NFPA MODE SWITCH is in the” enable” position.</td></tr>
</table>

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 10</u>:

[10] ADJUST Diesel Generator Frequency using the associated Diesel Generator Governor Control switch to obtain a synchroscope needle rotation of one revolution every 15 to 20 seconds in the SLOW direction.

<table>
<tr><th>Diesel</th><th>Handswitch Name</th><th>Handswitch No.</th><th>Panel</th></tr>
<tr><td>A</td><td>DG A GOVERNOR CONTROL</td><td>0-HS-82-A/3A</td><td>0-9-23-7</td></tr>
</table>

<u>Expected Action(s):</u>

Adjusts EDG A Frequency using 0-HS-82-A/3A, DG A GOVERNOR CONTROL, to obtain a synchroscope needle rotation of one revolution every 15 to 20 seconds in the SLOW direction.
</td><td>
Critical Step

______ SAT

______UNSAT

______N/A
</td></tr>
<tr><td>
<u>Step 11</u>:

[11] USE the associated Diesel Generator Voltage Regulator Control Switch to match Diesel Generator and System Voltages.

<table>
<tr><th>Diesel</th><th>Instrument Name</th><th>Inst No.</th><th>Panel</th></tr>
<tr><td rowspan="3">A</td><td>DG A VOLT REGULATOR CONT</td><td>0-HS-82-A/2A</td><td rowspan="3">0-9-23-7</td></tr>
<tr><td>GEN SYNC REF VOLTAGE</td><td>0-EI-82-AB</td></tr>
<tr><td>SYSTEM SYNC REF VOLTAGE</td><td>0-EI-211-0AB</td></tr>
</table>

<u>Expected Action(s):</u>

Matches EDG A Voltage with System Voltage using 0-HS-82-A/2A, DG A VOLT REGULATOR CONT switch.
</td><td>
______ SAT

______UNSAT

______N/A
</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 12: [12] **WHEN** the synchroscope needle is approximately 2 minutes on the right hand side of the 12 o'clock position, **THEN CLOSE** the 4KV Shutdown Board Feeder Breaker that is to be paralleled with the Diesel Generator. Expected Action(s): Closes the 4KV Shutdown Board Feeder Breaker 1614 when the synchroscope needle is approximately (+/-) 5 minutes from the 12 o'clock position.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: Following the completion of Step 12, inform the candidate "Another Operator will finish the procedure. This completes your task".	

STOP TIME: __________

Job Performance Measure (JPM)

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following plant conditions:

- 2 hours ago, 4 KV Shutdown Board A was separated from Offsite Power and is being powered by Emergency Diesel Generator (EDG) A
- Assistant Unit Operators (AUOs) are standing by on station for support
- No inclement weather currently exists in the area

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to perform the following operation in accordance with 0-OI-82, Standby Diesel Generator System, Section 8.3, Restoring Offsite Power to 4KV Shutdown Board at Panel 9-23:

- Restore Offsite Power to 4KV Shutdown Board A using Normal Feeder Breaker 1614 while leaving EDG A in Parallel with System

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Restore Offsite Power to 4KV Shutdown Board at Panel 3-9-23
JPM NUMBER:	631-U3	REVISION:	4

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-082-NO-09 / Restore Offsite Power to 4KV Shutdown Board at Panel 9-23			
K/A RATINGS:	RO: 3.4 SRO: 3.4			
K/A No. &STATEMENT:	262001 A.C. Electrical Distribution; A4.02 Ability to manually operate and/or monitor in the control room: Synchroscope, including understanding of running and incoming voltages			
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A			
SAFETY FUNCTION:	6			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 18 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ____________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ____________________ ____________
Validator *Date*

Approved by: ____________________ ____________
Site Training Management *Date*

Approved by: ____________________ ____________
Site Training Program Owner *Date*

TVA	**Job Performance Measure (JPM)**

OPERATOR: ______________________________ JPM Number: _631-U3_

RO _____ SRO _____ DATE: ______________

TASK EXPECTED ACTION(S): The Examinee is expected to restore Offsite Power to 4KV Shutdown Board at Panel 3-9-23

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 3-OI-82

VALIDATION TIME: _20 minutes_

PERFORMANCE TIME: ___________

COMMENTS: ____
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Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ______________________________ DATE: __________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
3	11/02/2020	All	Update JPM
4	01/25/2021	All	Procedure update

Procedure Revisions

Procedure	Revision
3-OI-82	152

SIMULATOR SETUP

IC	28
Exam IC	265

Console Operator Instructions	• Reset to IC 265 • Place the Simulator in RUN and verify stable conditions • Ensure the examinee has been briefed on 3-OI-82, Standby Diesel Generator System, Section 8.3, Restoring Offsite Power to 4KV Shutdown Board at Panel 3-9-23 • Ensure stopwatch is available

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A					

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 3 with the following plant conditions:

- 2 hours ago, 4 KV Shutdown Board 3EA was separated from Offsite Power and is being powered by Emergency Diesel Generator (EDG) 3A
- An Assistant Unit Operators (AUOs) are standing by on station for support
- No inclement weather currently exists in the area

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to perform the following operation in accordance with 3-OI-82, Standby Diesel Generator System, Section 8.3, Restoring Offsite Power to 4KV Shutdown Board at Panel 9-23:

- Restore Offsite Power to 4KV Shutdown Board 3EA using the Normal Feeder Breaker 1334 while leaving EDG 3A in Parallel with System

START TIME:___________

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 1</u>:

3-OI-82, Standby Diesel Generator (EDG) System
Section 8.3 Restoring Offsite Power to 4KV Shutdown Board at Panel 9-23

NOTES
The following list of 4KV Shutdown Board Normal and Alternate Feeder Breakers may be useful when performing this section:
<table>
<tr><th>Shutdown Board</th><th>3EA</th></tr>
<tr><td>Normal Feeder Breaker</td><td>1334</td></tr>
<tr><td>Alternate Feeder Breaker</td><td>1726</td></tr>
</table>
[1] CHECK 4KV Shutdown Board 3EA is being supplied power by its respective Diesel Generator as the only source of power.

<u>Expected Action(s):</u>

Verifies 4KV Shutdown Board 3EA is being supplied by EDG 3A.
</td><td>______ SAT

______UNSAT

______N/A</td></tr>
<tr><td>
<u>Step 2</u>:

[2] ENSURE the associated 4 KV Shutdown Board auto transfer lockout relay is tripped to MANUAL.
<table>
<tr><th>Diesel</th><th>Handswitch Name</th><th>Handswitch No.</th><th>Panel</th></tr>
<tr><td>3A</td><td>4KV SD BD 3EA AUTO/LOCKOUT RESET</td><td>3-43-211-3EA</td><td>3-9-23</td></tr>
</table>
<u>Expected Action(s):</u>

Ensures 3-43-211-3EA, 4KV SD BD 3EA AUTO/LOCKOUT RESET, is tripped to manual on Panel 3-9-23.
</td><td>______ SAT

______UNSAT

______N/A</td></tr>
</table>

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>Step 3:

[3] PLACE the synchroscope switch for the 4KV Shutdown Board feeder breaker that is to be paralleled with the Diesel Generator in ON.

Expected Action(s):

Places the synchroscope switch for the 4KV Shutdown Board feeder breaker that is to be paralleled with the Diesel Generator in ON.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td>Step 4:

[4] CHECK the applicable 4VKV Unit Board (4KV Bus Tie board) Voltage is between 3950 Volts and 4400 Volts and NOT undergoing abnormal voltage transients.

Expected Action(s):

Checks 4VKV 3A Unit Board (4KV Bus Tie board) Voltage is between 3950 Volts and 4400 Volts and not undergoing abnormal voltage transients.</td><td>______ SAT

______UNSAT

______N/A</td></tr>
<tr><td>Step 5:

[5] CHECK associated incoming frequency is between 59 Hertz and 61 Hertz and NOT undergoing abnormal frequency transients.

<table><tr><th>Shutdown Bd</th><th>Instrument Name</th><th>Instrument No.</th><th>Panel</th></tr><tr><td>3EA or 3EB</td><td>GENERATOR SYNC FREQUENCY</td><td>3-SI-82-3AB</td><td>3-9-23</td></tr></table>
Expected Action(s):

Verifies incoming frequency is between 59 Hertz and 61 Hertz and not undergoing abnormal frequency transients using 3-SI-82-3AB, GENERATOR SYNC FREQUENCY.</td><td>______ SAT

______UNSAT

______N/A</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 6: **CAUTION** **DO NOT** parallel the Diesel Generators with an unstable Offsite source or during inclement weather (e.g., lightning, heavy winds). [6] **IF** 4VKV Unit Board (4KV Bus Tie board) is experiencing abnormal voltage or frequency conditions, **THEN PERFORM** the following: Expected Action(s): Verifies that there are no abnormal voltage or frequency conditions. As given in the Initial Conditions, there is no inclement weather in the area.	______ SAT ______UNSAT ______N/A
Step 7: **CAUTION** Only one Unit 1 and 2 Diesel Generator at a time is allowed to be operated in Parallel with System. [7] **PULL and PLACE** the associated Diesel Generator Mode Selector Switch in PARALLELED WITH SYSTEM. Expected Action(s): Pulls 3-HS-82-3A/5A, DG 3A MODE SELECT SWITCH and places it in PARALLELED WITH SYSTEM.	**Critical Step** ______ SAT ______UNSAT ______N/A

Diesel	Handswitch Name	Handswitch No.	Panel
3A	DG 3A MODE SELECT	3-HS-82-3A/5A	3-9-23

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 8:</u>

CAUTION
Failure of the PARALLELED WITH SYSTEM light to illuminate in the following step could indicate that the DG is still in SINGLE UNIT operation and result in overload when the DG output breaker is closed.

[8] RELEASE the Diesel Generator Mode Selector Switch and OBSERVE PARALLELED WITH SYSTEM light illuminated.

<u>Expected Action(s):</u>

Releases 3-HS-82-3A/5A, DG 3A MODE SELECT switch and observes PARALLELED WITH SYSTEM light is illuminated.
</td><td>______ SAT

______UNSAT

______N/A</td></tr>
<tr><td>
<u>Step 9:</u>

[9] PLACE the associated Diesel Generator NFPA 805 MODE SWITCH in ENABLE.

<table>
<tr><th>Diesel</th><th>Handswitch Name</th><th>Handswitch No.</th><th>Panel</th></tr>
<tr><td>3A</td><td>NFPA 805 MODE SWITCH</td><td>3-43-211-03EA/03</td><td>4KV SDBD 3EA compt 03</td></tr>
</table>
<u>Expected Action(s):</u>

Directs AUO to place 3-43-211-03EA/03, NFPA 805 MODE SWITCH for EDG 3A, on Panel 4KV SDBD 3EA compt 03 in ENABLE.
</td><td>______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">DRIVER NOTE: When directed to place 3-43-211-03EA/03, NFPA 805 MODE SWITCH for EDG 3A on Panel 4KV SDBD 3EA compt 03 in ENABLE, verify that Remote Function ED32A is ENABLED, and report that the NFPA MODE SWITCH is in the ‘enable’ position.</td></tr>
</table>

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 10</u>:

[10] ADJUST Diesel Generator Frequency using the associated Diesel Generator Governor Control Switch to obtain a synchroscope needle rotation of one revolution every 15 to 20 seconds in the SLOW direction.

<table>
<tr><th>Diesel</th><th>Handswitch Name</th><th>Handswitch No.</th><th>Panel</th></tr>
<tr><td>3A</td><td>DG 3A GOVERNOR CONTROL</td><td>3-HS-82-3A/3A</td><td>3-9-23</td></tr>
</table>
<u>Expected Action(s):</u>

Adjusts EDG 3A Frequency using 3-HS-82-3A/3A, DG 3A GOVERNOR CONTROL, to obtain a synchroscope needle rotation of one revolution every 15 to 20 seconds in the SLOW direction.
</td><td>
Critical Step

______ SAT

______UNSAT

______N/A
</td></tr>
<tr><td>
<u>Step 11</u>:

[11] USE the associated Diesel Generator Voltage Regulator Control Switch to match Diesel Generator and System Voltages.

<table>
<tr><th>Diesel</th><th>Instrument Name</th><th>Inst No.</th><th>Panel</th></tr>
<tr><td rowspan="3">3A</td><td>DG 3A VOLTAGE REGULATOR CONTROL</td><td>3-HS-82-3A/2A</td><td rowspan="3">3-9-23</td></tr>
<tr><td>GEN SYNC REFERENCE VOLTAGE</td><td>3-EI-82-3AB</td></tr>
<tr><td>SYSTEM SYNC REFERENCE VOLTAGE</td><td>3-EI-211-3EAB/B</td></tr>
</table>
<u>Expected Action(s):</u>

Matches EDG 3A Voltage with System Voltage using 3-HS-82-3A/2A, DG 3A VOLT REGULATOR CONTROL SWITCH.
</td><td>
______ SAT

______UNSAT

______N/A
</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 12: [12] **WHEN** the synchroscope needle is approximately 2 minutes on the right hand side of the 12 o'clock position, **THEN** **CLOSE** the 4KV Shutdown Board Feeder Breaker that is to be paralleled with the Diesel Generator. Expected Action(s): Closes the 4kV Shutdown Board Feeder Breaker 1334 when the synchroscope needle is approximately (+/-) 5 minutes from the 12 o'clock position.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: Following the completion of Step 12, inform the examinee "Another Operator will finish the procedure. This completes your task".	

STOP TIME: ___________

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 3 with the following plant conditions:

- 2 hours ago, 4 KV Shutdown Board 3EA was separated from Offsite Power and is being powered by Emergency Diesel Generator (EDG) 3A
- An Assistant Unit Operators (AUOs) are standing by on station for support
- No inclement weather currently exists in the area

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to perform the following operation in accordance with 3-OI-82, Standby Diesel Generator System, Section 8.3, Restoring Offsite Power to 4KV Shutdown Board at Panel 9-23:

- Restore Offsite Power to 4KV Shutdown Board 3EA using the Normal Feeder Breaker 1334 while leaving EDG 3A in Parallel with System

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Respond to Loss of Power to One RPS Bus in accordance with 2-AOI-99-1, Loss of Power to One RPS Bus	
JPM NUMBER:	748-U2		REVISION:	0

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-099-AB-01			
K/A RATINGS:	RO: 3.7 SRO: 3.9			
K/A No. &STATEMENT:	212000 Reactor Protection System A2.01; Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RPS motor-generator set failure			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	7			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 12 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: 748-U2

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to restore plant conditions to normal following a Loss of Power to One RPS Bus.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 2-AOI-99-1

VALIDATIONTIME: 12 min

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	11/9/2020	All	Initial issue

Procedure Revisions

Procedure	Revision
2-AOI-99-1	30

SIMULATOR SETUP

IC	28
Exam IC	283

Console Operator Instructions	• **Reset to IC 283** • **Place the simulator in RUN to ensure stable conditions**

Malfunctions	Description	Event	Severity	Delay	Initial set
NONE					

Remotes	Description	Event	Severity	Delay	Initial set
NONE					

Overrides	Description	Event	Severity	Delay	Initial set
NONE					

Schedule File(s): **N/A**
Event File(s): **N/A**

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

- You are an Operator on Unit 2
- Unit 2 is operating at 100% Reactor Power
- 2A RPS motor generator tripped, and 2A RPS Bus has been placed on alternate

INITIATING CUES: The Unit 2 Nuclear Unit Senior Operator has directed you to continue with restoring affected systems to normal in accordance with 2-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

START TIME:__________

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 1</u>:

2-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12]

[12] RESET the RPS trip logic half SCRAM at Panel 2-9-5 as follows:

NOTE
The eight CONTROL ROD TEST SCRAM SOLENOID GROUP A and B LIGHTS SHOULD ILLUMINATE.

[12.1] MOMENTARILY PLACE 2-HS-99-5A-S5, SCRAM RESET as follows:
[12.2] RESET FIRST position. (Group 2/3).
[12.2] RESET SECOND position. (Group 1/4).
[12.4] NORMAL position.

<u>Expected Action(s):</u>

Resets the RPS trip logic half SCRAM at Panel 2-9-5 by placing 2-HS-99-5A-S5, SCRAM RESET in RESET FIRST position (Group 2/3), then RESET SECOND position (Group 1/4), and finally back to the NORMAL position.
</td><td>
Critical Step

_____ SAT

_____UNSAT

_____N/A
</td></tr>
<tr><td>
<u>Step 2</u>:

[13] VERIFY the following:
[13.1] All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED.

<u>Expected Action(s):</u>

Verifies all eight SCRAM solenoid logic reset lights are illuminated.
</td><td>
_____ SAT

_____UNSAT

_____N/A
</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 3: [13.2] The following four lights ILLUMINATED: [13.2.1] 2-IL-99-5A/AB, SYSTEM A BACKUP SCRAM VALVE. [13.2.2] 2-IL-99-5A/CD, SYSTEM B BACKUP SCRAM VALVE. Expected Action(s): Verifies that the A and B Backup SCRAM valve lights are illuminated.	_____SAT _____UNSAT _____N/A
Step 4: [13.3] Scram Discharge Volume Vent and Drain Valves indicate OPEN. Expected Action(s): Verifies the eight (8) Scram Discharge Volume Vent and Drain Valves indicate open.	_____ SAT _____UNSAT _____N/A
Step 5: [14] **RESET** PCIS trip logic at Panel 2-9-4 as follows: [14.1] **MOMENTARILY** PLACE 2-HS-64-16A-S32, PCIS DIV I RESET, to left and right RESET positions. [14.2] **VERIFY** the following red lights ILLUMINATED: [14.2.1] 2-IL-64-A1, MSIV GROUP A1. [14.2.2] 2-IL-64-B1, MSIV GROUP B1. **EXAMINER NOTE: Verifying the red lights for 2-IL-64-A1, MSIV GROUP A1 and 2-IL-64-B1, MSIV GROUP B1 are ILLUMINATED, is NOT a Critical Step.** Expected Action(s): On Panel 2-9-4, PLACES 2-HS-64-16A-S32, PCIS DIV I RESET, to left and right RESET positions.	**Critical Step** _____SAT _____UNSAT _____N/A

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>Step 6:

[14.3] MOMENTARILY PLACE 2-HS-64-16A-S33, PCIS DIV II RESET, to left and right RESET positions.
[14.2] VERIFY the following red lights ILLUMINATED:
[14.2.1] 2-IL-64-A2, MSIV GROUP A2.
[14.2.2] 2-IL-64-B2, MSIV GROUP B2.

EXAMINER NOTE: Verifying the red lights for 2-IL-64-A2, MSIV GROUP A2 and 2-IL-64-B2, MSIV GROUP B2 are ILLUMINATED, is NOT a Critical Step.

Expected Action(s):

Places 2-HS-64-16A-S33, PCIS DIV II RESET, to left and right RESET positions.</td><td>Critical Step

_____SAT

_____UNSAT

_____N/A</td></tr>
<tr><td>Step 7:

[15] IF Unit was in Shutdown Cooling prior to the loss of one RPS Bus, THEN REFER to 2-AOI-74-1, Loss of Shutdown Cooling.

Expected Action(s):

Marks this step as N/A. Initial Conditions state Unit 2 is operating at 100% Reactor Power.</td><td>_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td>Step 8:

[16] VERIFY only one Standby Gas Train operating.

Expected Action(s):

Determines that all three trains of Standby Gas are running. The candidate may contact Unit 1 to secure two Standby Gas Trains.</td><td>_____SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">DRIVER NOTE: If contacted as Unit 1 to secure two trains of Standby Gas, insert Event 1 to secure Standby Gas Trains B and C. Inform the candidate that Train A is running.</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
<u>Step 9</u>: [17] **RESET** the Secondary Containment Isolation logic at Panel 2-9-25, as follows: [17.1] **PLACE** 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS switch to OFF. [17.2] **PLACE** 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS Switch to OFF. <u>Expected Action(s):</u> Resets Secondary Containment Isolation logic by placing the Reactor and Refuel Fans in OFF on Panel 2-9-25.	**Critical Step** _____SAT _____UNSAT _____N/A
<u>Step 9</u>: [18] **START** the Refuel Zone supply and exhaust fans, at Panel 2-9-25, as follows: [18.1] **PLACE** 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS Switch in SLOW A (SLOW B) position. <u>Expected Action(s):</u> Places 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS SWITCH in SLOW A (SLOW B) position.	**Critical Step** _____SAT _____UNSAT _____N/A
<u>Step 10</u>: [19] **START** the Reactor Building supply and exhaust fans, at Panel 2-9-25, as follows: [19.1] **PLACE** 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS switch to the SLOW A(B) position. <u>Expected Action:</u> Places 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS SWITCH to the SLOW A(B) position.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: Once Step [19.1] is completed, inform the examinee "Another Operator will continue the procedure. This completes your task".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

- You are an Operator on Unit 2
- Unit 2 is operating at 100% Reactor Power
- 2A RPS motor generator tripped, and 2A RPS Bus has been placed on alternate

INITIATING CUES: The Unit 2 Nuclear Unit Senior Operator has directed you to continue with restoring affected systems to normal in accordance with 2-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Respond to Loss of Power to One RPS Bus in accordance with 3-AOI-99-1, Loss of Power to One RPS Bus	
JPM NUMBER:	748-U3	REVISION:	0	

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-099-AB-01			
K/A RATINGS:	RO: 3.7 SRO: 3.9			
K/A No. &STATEMENT:	212000 Reactor Protection System A2.01; Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RPS motor-generator set failure			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	7			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 12 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: 748-U3

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to restore plant conditions to normal following a Loss of Power to One RPS Bus.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 3-AOI-99-1

VALIDATIONTIME: 12 min

PERFORMANCE TIME: ___________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	11/9/2020	All	Initial issue

Procedure Revisions

Procedure	Revision
3-AOI-99-1	20

SIMULATOR SETUP

IC	28
Exam IC	266

Console Operator Instructions	• **Reset to IC 266** • **Place the simulator in RUN to ensure stable conditions**

Malfunctions	Description	Event	Severity	Delay	Initial set
NONE					

Remotes	Description	Event	Severity	Delay	Initial set
NONE					

Overrides	Description	Event	Severity	Delay	Initial set
NONE					

Schedule File(s): **N/A**

Event File(s): **N/A**

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

- You are an Operator on Unit 3
- Unit 3 is operating at 100% Reactor Power
- 3A RPS motor generator tripped, and 3A RPS Bus has been placed on alternate

INITIATING CUES: The Unit 3 Nuclear Unit Senior Operator has directed you to continue with restoring affected systems to normal in accordance with 3-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: 3-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12] [12] RESET the RPS trip logic half SCRAM at Panel 3-9-5 as follows: [12.1] **MOMENTARILY PLACE** 3-HS-99-5A-S5, SCRAM RESET as follows: [12.2] RESET FIRST position. (Group 2/3). [12.2] RESET SECOND position. (Group 1/4). [12.4] NORMAL. Expected Action(s): Resets the RPS trip logic half SCRAM at Panel 3-9-5 by placing 3-HS-99-5A-S5, SCRAM RESET in RESET FIRST position (Group 2/3), then RESET SECOND position (Group 1/4), and finally back to the NORMAL position	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 2: [13] **CHECK** the following conditions: [13.1] All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED. Expected Action(s): Verifies all eight SCRAM solenoid logic reset lights are illuminated.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [13.2] The following four lights ILLUMINATED: [13.2.1] 3-IL-99-5A/AB, SYSTEM A BACKUP SCRAM VALVE. [13.2.2] 3-IL-99-5A/CD, SYSTEM B BACKUP SCRAM VALVE. Expected Action(s): Verifies that the A and B Backup SCRAM valve lights are illuminated.	_____SAT _____UNSAT _____N/A
Step 4: [13.3] Scram Discharge Volume Vent and Drain Valves indicate OPEN. [13.4] Points SOE033 (Channel A3 manual scram) and SOE035 (Channels A1&A2 Auto Scram) on ICS computer or on the First Out Printer reads “NOT TRIP” for RPS “A”. [13.5] N/A Expected Action(s): Verifies the eight (8) Scram Discharge Volume Vent and Drain Valves indicate open. Checks ICS computer points SOE033 and SOE035 or the First Out Printer reads ‘NOT TRIP’ for RPS 2A.	_____ SAT _____UNSAT _____N/A

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>

<u>Step 5</u>:

[14] RESET PCIS trip logic at Panel 3-9-4 as follows:
[14.1] MOMENTARILY PLACE 3-HS-64-16A-S32, PCIS DIV I RESET, to left and right RESET positions.
[14.2] CHECK the following red lights ILLUMINATED:
[14.2.1] 3-IL-64-A1, MSIV GROUP A1.
[14.2.2] 3-IL-64-B1, MSIV GROUP B1.

EXAMINER NOTES: Verifying the red lights for 3-IL-64-A1, MSIV GROUP A1 and 3-IL-64-B1, MSIV GROUP B1 are ILLUMINATED, is NOT a Critical Step.

<u>Expected Action(s):</u>

On Panel 3-9-4, PLACES 3-HS-64-16A-S32, PCIS DIV I RESET, to left and right RESET positions.

</td><td>

Critical Step

_____SAT

_____UNSAT

_____N/A

</td></tr>
<tr><td>

<u>Step 6</u>:

[14.3] MOMENTARILY PLACE 3-HS-64-16A-S33, PCIS DIV II RESET, to left and right RESET positions.
[14.2] CHECK the following red lights ILLUMINATED:
[14.2.1] 3-IL-64-A2, MSIV GROUP A2.
[14.2.2] 3-IL-64-B2, MSIV GROUP B2.

EXAMINER NOTES: Verifying the red lights for 3-IL-64-A2, MSIV GROUP A2 and 3-IL-64-B2, MSIV GROUP B2 are ILLUMINATED, is NOT a Critical Step.

<u>Expected Action(s):</u>

On Panel 3-9-4, PLACES 3-HS-64-16A-S33, PCIS DIV II RESET, to left and right RESET positions.

</td><td>

Critical Step

_____SAT

_____UNSAT

_____N/A

</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 7: [15] **IF** Unit 3 was in Shutdown Cooling prior to the loss of one RPS Bus, **THEN REFER** to Loss of Shutdown Cooling, 3-AOI-74-1. Expected Action(s): Marks this step as N/A. Initial Conditions state Unit 3 is operating at 100% Reactor Power.	______ SAT ______UNSAT ______N/A
Step 8: [16] **RESET** the Secondary Containment Isolation logic at Panel 3-9-25, as follows: [16.1] **PLACE** 3-HS-64-11A, REACTOR ZONE FANS AND DAMPERS switch to OFF. [16.2] **PLACE** 3-HS-64-3A, REFUEL ZONE FANS AND DAMPERS Switch to OFF. [16.3] **VERIFY** only one Standby Gas Train operating. Expected Action(s): **On Panel 3-9-25, RESETS the Secondary Containment Isolation logic as follows:** • **PLACES 3-HS-64-11A, REACTOR ZONE FANS AND DAMPERS switch to OFF** • **PLACES 3-HS-64-3A, REFUEL ZONE FANS AND DAMPERS Switch to OFF** **On Panel 3-9-20 or 3-9-25, VERIFIES only one Standby Gas Train is operating (A or B or C red light ILLUMINATED).**	**Critical Step** ______SAT ______UNSAT ______N/A
Step 9: [17] **START** the Refuel Zone supply and exhaust fans, at Panel 3-9-25, as follows: [17.1] **PLACE** 3-HS-64-3A, REFUEL ZONE FANS AND DAMPERS Switch in SLOW A (SLOW B) position. Expected Action(s): **On Panel 3-9-25, PLACES 3-HS-64-3A, REFUEL ZONE FANS AND DAMPERS Switch in SLOW A (SLOW B) position.**	**Critical Step** ______SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 10: [18] **START** the Reactor Building supply and exhaust fans, at Panel 3-9-25, as follows: [18.1] **PLACE** 3-HS-64-11A, REACTOR ZONE FANS AND DAMPERS switch to the SLOW A(B) position. Expected Action: **On Panel 3-9-25, PLACES 3-HS-64-11A, REACTOR ZONE FANS AND DAMPERS switch to the SLOW A(B) position.**	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: **Once Step [18.1] is completed, inform the examinee "Another Operator will continue the procedure. This completes your task".**	

STOP TIME: ______

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

- You are an Operator on Unit 3
- Unit 3 is operating at 100% Reactor Power
- 3A RPS motor generator tripped, and 3A RPS Bus has been placed on alternate

INITIATING CUES: The Unit 3 Nuclear Unit Senior Operator has directed you to continue with restoring affected systems to normal in accordance with 3-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

	Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Respond to a loss of Reactor Building Closed Cooling Water (RBCCW) in accordance with 2-AOI-70-1, Loss of RBCCW
JPM NUMBER:	602A-U2	REVISION:	3

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-070-AB-01: Perform manipulations required for a loss of Reactor Building Closed Cooling Water .			
K/A RATINGS:	RO: 3.3 SRO: 3.4			
K/A No. &STATEMENT:	400000 Component Cooling Water System A2.01: Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Loss of CCW pump.			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	8			

EVALUATION LOCATION:	☐In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 5 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ______________
Developer Date

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator Date

Approved by: ______________________ ______________
Site Training Management Date

Approved by: ______________________ ______________
Site Training Program Owner Date

	Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: _602A-U2___

RO SRO DATE: _______

TASK STANDARD: The Examinee is expected to respond to a loss of Reactor Building Closed Cooling Water (RBCCW)

Operator Fundamental evaluated:
OF-2 Controlling plant evolutions precisely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 2-AOI-70-1

VALIDATION TIME: ___5 min____

PERFORMANCE TIME: ___________

COMMENTS: ____
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Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	06/29/2008	All	Initial issue
1	10/31/2016	All	Updated AOI required actions. Updated to new format.
2	03/29/2017	All	Incorporated NRC Prep Week Comments.
3	11/09/20	All	Updated JPM

Procedure Revisions

Procedure	Revision
2-AOI-70-1	33

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• **Reset to IC 28** • **Run Schedule File 602F Unit 2.sch** • **Ensure event file 602F Event for 70-48.evt starts** • **Place the simulator in RUN to ensure stable conditions** • **When directed by the examiner insert Event 1 to trip RBCCW Pump 2A** • **Verify that event 2 (RBCCW Pump 2B trip) is triggered when the 70-48 valve begins to close**

Malfunctions	Description	Event	Severity	Delay	Initial set
SW02A	RBCCW PUMP 2A TRIP	1	NA	NA	TRIP
SW02B	RBCCW PUMP 2B TRIP	2	NA	80	TRIP

Events	Description	Event	Severity	Delay	Initial set
2	70-48 begins to close	NA	NA	NA	NA

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS: You are a Unit 2 Operator with Reactor Power is 100%.

There is NO equipment out of service.

INITIATING CUES: Respond to plant conditions.

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: 2-AOI-70-1, Loss of Reactor Building Closed Cooling Water **4.1 Immediate Actions:** [1] **IF** RBCCW Pump(s) has tripped, **THEN** Perform the following (Otherwise N/A): • **SECURE** RWCU Pumps • **ENSURE** 2-FCV-70-48, RBCCW SECTIONALIZING VALVE **CLOSED** Expected Action(s): Secures RWCU Pumps and verifies that 2-FCV-70-48 is closing.	_____SAT _____UNSAT _____N/A
Examiner Note: (BEGIN ALTERNATE PATH) 2B RBCCW Pump will trip 80 seconds after 2-FCV-70-48 begins to close.	
Step 2: **4.2 Subsequent Actions** **CAUTION** Operations outside of the allowable regions shown on the Power to Flow Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. REFER TO 2-GOI-100-12A for required actions and monitoring to be performed during a power reduction. [1] **IF** Reactor is at power AND Drywell Cooling cannot be immediately restored, **THEN PERFORM** the following (otherwise **N/A**): [1.1] **IF** Core Flow is above 60%, **THEN REDUCE** Core Flow to between 50-60%. Expected Action(s): Reduces Core Flow to between 50-60%.	_____SAT _____UNSAT _____N/A

Step 3: [1.2] **MANUALLY SCRAM** the Reactor and **PLACE** MODE Switch in SHUTDOWN. **REFER** TO 2-AOI-100-1, Reactor SCRAM. Expected Action(s): **(Critical Steps)** Inserts a manual Reactor SCRAM and places 2-HS-99-5A-S1, REACTOR MODE SWITCH, in SHUTDOWN. (**NOT** Critical) SCRAM Report and refers to 2-AOI-100-1, Reactor SCRAM	**Critical Step** ______SAT ______UNSAT ______N/A
Examiner Note: **The candidate MUST SCRAM the Reactor before tripping the Recirc Pumps.** **The candidate may elect to insert a manual Reactor SCRAM, then shutdown both Recirc Pumps BEFORE giving the SCRAM Report.**	
Step 4: [1.3] **SHUT DOWN** both Recirc Pumps. • **DEPRESS** 2-HS-96-19, RECIRC DRIVE 2A SHUTDOWN • **DEPRESS** 2-HS-96-20, RECIRC DRIVE 2B SHUTDOWN Expected Action(s): Shutdowns both Recirc Pumps.	**Critical Step** ______SAT ______UNSAT ______N/A
EXAMINER CUE: [After the SCRAM Report is given and both Recirc Pumps are shutdown] "Another Operator will continue with this procedure. This completes your task."	

STOP TIME: ______

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS: You are a Unit 2 Operator with Reactor Power is 100%.

There is NO equipment out of service.

INITIATING CUES: Respond to plant conditions.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Respond to a loss of Reactor Building Closed Cooling Water (RBCCW) in accordance with 3-AOI-70-1, Loss of RBCCW
JPM NUMBER:	602A-U3	REVISION:	3

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-070-AB-01: Perform manipulations required for a loss of Reactor Building Closed Cooling Water.			
K/A RATINGS:	RO: 3.3 SRO: 3.4			
K/A No. &STATEMENT:	400000 Component Cooling Water System A2.01: Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Loss of CCW pump.			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	8			

EVALUATION LOCATION:	☐In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 5 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: _602A-U3_

RO SRO DATE: _______

TASK STANDARD: The Examinee is expected to respond to a loss of Reactor Building Closed Cooling Water (RBCCW)

Operator Fundamental evaluated:
OF-2 Controlling plant evolutions precisely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 3-AOI-70-1

VALIDATION TIME: ___5 min___

PERFORMANCE TIME: ___________

COMMENTS: ____

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Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	06/29/2008	All	Initial issue
1	10/31/2016	All	Updated AOI required actions. Updated to new format.
2	03/29/2017	All	Incorporated NRC Prep Week Comments.
3	11/09/20	All	Updated JPM

Procedure Revisions

Procedure	Revision
3-AOI-70-1	20

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• **Reset to IC 28** • **Run Schedule File 602F Unit 3.sch** • **Ensure event file 602F Event for 70-48.evt starts** • **Place the simulator in RUN to ensure stable conditions** • **When directed by the examiner insert Event 1 to trip RBCCW Pump 3A** • **Verify that event 2 (RBCCW Pump 3B trip) is triggered when the 70-48 valve begins to close**

Malfunctions	Description	Event	Severity	Delay	Initial set
SW02A	RBCCW PUMP 3A TRIP	1	NA	NA	TRIP
SW02B	RBCCW PUMP 3B TRIP	2	NA	80	TRIP

Events	Description	Event	Severity	Delay	Initial set
2	70-48 beings to close	NA	NA	NA	NA

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS: You are a Unit 3 Operator with Reactor Power is 100%.

There is NO equipment out of service.

INITIATING CUES: Respond to plant conditions.

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: 3-AOI-70-1, Loss of Reactor Building Closed Cooling Water **4.1 Immediate Actions:** [1] **IF** RBCCW Pump(s) has tripped, **THEN** Perform the following (Otherwise N/A): • **SECURE** RWCU Pumps. • **ENSURE** 3-FCV-70-48, RBCCW SECTIONALIZING VLV **CLOSED**. Expected Action(s): Secures RWCU Pumps and verifies that 3-FCV-70-48 is closing.	______SAT ______UNSAT ______N/A
Examiner Note: (BEGIN ALTERNATE PATH) 2B RBCCW Pump will trip 80 seconds after 3-FCV-70-48 begins to close.	
Step 2: **4.2 Subsequent Actions** **CAUTION** Operations outside of the allowable regions shown on the Power to Flow Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. REFER TO 3-GOI-100-12A for required actions and monitoring to be performed during a power reduction. [1] **IF** Reactor is at power **AND** Drywell Cooling cannot be immediately restored, **THEN PERFORM** the following (otherwise **N/A**): [1.1] **IF** core flow is above 60%, **THEN REDUCE** core flow to between 50-60%. Expected Action(s): Reduces core flow to between 50-60%.	______SAT ______UNSAT ______N/A

<u>Step 3</u>: [1.2] **MANUALLY SCRAM** the Reactor and **PLACE** MODE Switch in SHUTDOWN. **REFER** TO 3-AOI-100-1, Reactor SCRAM. <u>Expected Action(s):</u> **(Critical Steps)** Inserts a manual Reactor SCRAM and places 3-HS-99-5A-S1, REACTOR MODE SWITCH, in SHUTDOWN. (**NOT** Critical) SCRAM Report and refers to 3-AOI-100-1	**Critical Step** ______SAT ______UNSAT ______N/A
Examiner Note: **The candidate MUST SCRAM the Reactor before tripping the Recirc Pumps.** **The candidate may elect to insert a manual Reactor SCRAM, then shutdown both Recirc Pumps BEFORE giving the SCRAM Report.**	
<u>Step 4</u>: [1.3] **SHUT DOWN** both Recirc Pumps. • **DEPRESS** 3-HS-96-19, RECIRC DRIVE 3A SHUTDOWN • **DEPRESS** 3-HS-96-20, RECIRC DRIVE 3B SHUTDOWN <u>Expected Action(s):</u> Shutdowns both Recirc Pumps.	**Critical Step** ______SAT ______UNSAT ______N/A
EXAMINER CUE: [After the SCRAM Report is given and both Recirc Pumps are shutdown] "Another Operator will continue with this procedure. This completes your task."	

STOP TIME: ______

TVA	**Job Performance Measure (JPM)**

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS: You are a Unit 3 Operator with Reactor Power is 100%.

There is NO equipment out of service.

INITIATING CUES: Respond to plant conditions.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Emergency Vent Primary Containment in accordance with 2-EOI-Appendix-13, Emergency Venting Primary Containment
JPM NUMBER:	55-U2	REVISION:	2

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-000-EM-63 / Emergency Vent Primary Containment in accordance with EOI Appendix 13			
K/A RATINGS:	RO: 3.3 SRO: 3.4			
K/A No. & STATEMENT:	288000 Plant Ventilation Systems A2.01: Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High Drywell Pressure: Plant-Specific			
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A			
SAFETY FUNCTION:	9			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 5 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*
(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: __55-U2__

RO _____ SRO _____ DATE: _____________

TASK STANDARD: The Examinee is expected to Emergency Vent Primary Containment.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-EOI-Appendix-13

VALIDATION TIME: __5 minutes__

PERFORMANCE TIME: ___________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
1	09/01/2020	All	Format update
2	1/12/21	All	JPM update

Procedure Revisions

Procedure	Revision
2-EOI-Appendix-13	10

SIMULATOR SETUP

IC	28
Exam IC	284

Console Operator Instructions	• Reset to Exam IC 284 • 2-XS-74-121(129), RHR SYS I(II) CONTAINMENT SPRAY / COOLING VALVE SELECT switches must be placed in NORMAL AFTER SELECT following simulator reset • Place the simulator in RUN to ensure stable conditions

Malfunctions	Description	Event	Severity	Delay	Initial set

Overrides	Description	Event	Severity	Delay	Initial set

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following plant conditions:

- A Loss of Coolant Accident (LOCA) has occurred on Unit 2
- The Reactor has SCRAMMED
- Several Control Rods failed to insert on the SCRAM
- All Drywell Sprays have failed
- Drywell Pressure and Temperature are rising

Note: Another Operator is tasked with maintaining Reactor Water Level

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to emergency vent Primary Containment in accordance with 2-EOI-Appendix-13, Emergency Venting Primary Containment.

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: **EXAMINER NOTE: Verify the candidate has been briefed on 2-EOI-Appendix-13 prior to beginning the JPM.** 2-EOI-Appendix-13, Emergency Venting Primary Containment [1] **NOTIFY** SHIFT MANAGER/SED of the following: • Emergency Venting of Primary Containment is in progress • Off-Gas Release Rate Limits will be exceeded **NOTES** 1) HARDENED CONTAINMENT VENT VALVES 2-FCV-64-221 and 222 may be operated locally with handwheels (U2 RB el. 580, top of clean room, southwest corner). 2) If an alternate DC power source is needed for the HCVS valve solenoids, Att. 4 HCVS Battery Alignment may be performed. 3) If an alternate air supply is needed for the HCVS valves, Att. 5 HCVS Nitrogen Bottle Alignment may be performed. 4) If required, HCVS valves may be operated from the U3 DG building using Att. 6, HCVS Operation from the Remote Operating Station. Expected Action(s): Informs the Shift Manager that Emergency Venting Primary Containment is in progress and that Off-Gas Release Rate Limits will be exceeded. **EXAMINER CUE: Acknowledge any information provided by the candidate to the Shift Manager with respect to Emergency Venting.**	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 2: [2] **VENT** the Suppression Chamber as follows (Panel 2-9-3): [2.1] **IF** EITHER of the following exists: • Suppression Pool Water Level CANNOT be determined to be below 26 ft., **OR** • Suppression Chamber CANNOT be vented, **THEN CONTINUE** in this procedure at Step 1.0[3] Expected Action(s): Makes note of Step [2.1] and continues in the procedure.	______ SAT ______UNSAT ______N/A
Step 3: [2.2] **PLACE** keylock switch 2-HS-64-222B, HARDENED CONTAINMENT VENT OUTBOARD PERMISSIVE, in PERM. [2.3] **CHECK** blue indicating light above 2-HS-64-222B, HARDENED CONTAINMENT VENT OUTBOARD PERMISSIVE, illuminated. **EXAMINER NOTE: ONLY placing keylock switch 2-HS-64-222B in PERMISSIVE is critical in this step.** Expected Action(s): Places 2-HS-64-222B, HARDENED CONTAINMENT VENT OUTBOARD PERMISSIVE, in PERM (permissive) and verifies that the blue indicating light illuminates.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 4: [2.4] **OPEN** 2-FCV-64-222, HARDENED CONTAINMENT VENT OUTBOARD ISOLATION VALVE. Expected Action(s): Opens 2-FCV-64-222, HARDENED CONTAINMENT VENT OUTBOARD ISOLATION VALVE.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 5: [2.5] **PLACE** keylock switch 2-HS-64-221B, HARDENED CONTAINMENT VENT INBOARD PERMISSIVE, in PERM. [2.6] **CHECK** blue indicating light above 2-HS-64-221B, HARDENED CONTAINMENT VENT INBOARD PERMISSIVE, illuminated. **EXAMINER NOTE: ONLY placing keylock switch 2-HS-64-221B in PERMISSIVE is critical in this step.** Expected Action(s): Places keylock switch 2-HS-64-221B, HARDENED CONTAINMENT VENT INBOARD PERMISSIVE, in PERM (permissive) and verifies that the blue light illuminates.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 6: [2.7] **OPEN** 2-FCV-64-221, HARDENED CONTAINMENT VENT INBOARD ISOLATION VALVE. Expected Action(s): Opens 2-FCV-64-221, HARDENED CONTAINMENT VENT INBOARD ISOLATION VALVE.	______ SAT ______UNSAT ______N/A
Step 7: [2.8] **CHECK** Drywell and Suppression Chamber Pressure lowering. Expected Action(s): Verifies Drywell and Suppression Chamber Pressure are lowering.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: Inform the candidate "Another Operator will continue with this procedure. This completes your task".	

STOP TIME: ___________

Job Performance Measure (JPM)

Provide to Applicant

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following plant conditions:

- A Loss of Coolant Accident (LOCA) has occurred on Unit 2
- The Reactor has SCRAMMED
- Several Control Rods failed to insert on the SCRAM
- All Drywell Sprays have failed
- Drywell Pressure and Temperature are rising

Note: Another Operator is tasked with maintaining Reactor Water Level

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to emergency vent Primary Containment in accordance with 2-EOI-Appendix-13, Emergency Venting Primary Containment.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Emergency Vent Primary Containment in accordance with 3-EOI-Appendix-13, Emergency Venting Primary Containment
JPM NUMBER:	55-U3	REVISION:	3

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-000-EM-63 / Emergency Vent Primary Containment in accordance with EOI Appendix 13			
K/A RATINGS:	RO: 3.3 SRO: 3.4			
K/A No. & STATEMENT:	288000 Plant Ventilation Systems A2.01: Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High Drywell Pressure: Plant-Specific			
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A			
SAFETY FUNCTION:	9			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 5 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 55-U3

RO _____ SRO _____ DATE: ______________

TASK STANDARD: The Examinee is expected to Emergency Vent Primary Containment.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 3-EOI-Appendix-13

VALIDATION TIME: 5 min

PERFORMANCE TIME: ___________

COMMENTS: ______________________________

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
1	09/01/2020	All	Format update
2	1/12/21	All	JPM update
3	02/25/21	All	Procedure update

Procedure Revisions

Procedure	Revision
3-EOI-Appendix-13	7

SIMULATOR SETUP

IC	28
Exam IC	267

Console Operator Instructions	• Reset to Exam IC 267 • 3-XS-74-121(129), RHR SYS I(II) CONTAINMENT SPRAY / COOLING VALVE SELECT switches must be placed in NORMAL AFTER SELECT following simulator reset • Place the simulator in RUN to ensure stable conditions

Malfunctions	Description	Event	Severity	Delay	Initial set

Overrides	Description	Event	Severity	Delay	Initial set

****
IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.
****

INITIAL CONDITIONS:

You are an Operator on Unit 3 with the following plant conditions:

- A Loss of Coolant Accident (LOCA) has occurred on Unit 3
- The Reactor has SCRAMMED
- Several Control Rods failed to insert on the SCRAM
- All Drywell Sprays have failed
- Drywell Pressure and Temperature are rising

Note: Another Operator is tasked with maintaining Reactor Water Level

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to emergency vent Primary Containment in accordance with 3-EOI-Appendix-13, Emergency Venting Primary Containment.

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: **EXAMINER NOTE: Verify the candidate has been briefed on 3-EOI-Appendix-13 prior to beginning the JPM.** 3-EOI-Appendix-13, Emergency Venting Primary Containment [1] **NOTIFY** SHIFT MANAGER/SED of the following: • Emergency Venting of Primary Containment is in progress • Off-Gas Release Rate Limits will be exceeded **NOTES** 1) HARDENED CONTAINMENT VENT VALVES 3-FCV-64-221 and 222 may be operated locally with handwheels (U3 RB el. 580, west of clean room). 2) If an alternate DC power source is needed for the HCVS valve solenoids, Att. 4 HCVS Battery Alignment may be performed. 3) If an alternate air supply is needed for the HCVS valves, Att. 5 HCVS Nitrogen Bottle Alignment may be performed. 4) If required, HCVS valves may be operated from the U3 DG building using Att. 6, HCVS Operation from the Remote Operating Station. Expected Action(s): Informs the Shift Manager that Emergency Venting Primary Containment is in progress and that Off-Gas Release Rate Limits will be exceeded. **EXAMINER CUE: Acknowledge any information provided by the candidate to the Shift Manager with respect to Emergency Venting.**	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: [2] **VENT** the Suppression Chamber as follows (Panel 3-9-3): [2.1] **IF** EITHER of the following exists: • Suppression Pool Water Level CANNOT be determined to be below 26 ft., **OR** • Suppression Chamber CANNOT be vented, **THEN CONTINUE** in this procedure at Step 1.0[3] Expected Action(s): Makes note of Step [2.1] and continues in the procedure.	______ SAT ______UNSAT ______N/A
Step 3: [2.2] **PLACE** keylock switch 3-HS-64-222B, HARDENED CONTAINMENT VENT OUTBOARD PERMISSIVE, in PERM. [2.3] **CHECK** blue indicating light above 3-HS-64-222B, HARDENED CONTAINMENT VENT OUTBOARD PERMISSIVE, illuminated. **EXAMINER NOTE: ONLY placing keylock switch 3-HS-64-222B in PERMISSIVE is critical in this step.** Expected Action(s): Places 3-HS-64-222B, HARDENED CONTAINMENT VENT OUTBOARD PERMISSIVE, in PERM (permissive) and verifies that the blue indicating light illuminates.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 4: [2.4] **OPEN** 3-FCV-64-222, HARDENED CONTAINMENT VENT OUTBOARD ISOLATION VALVE. Expected Action(s): Opens 3-FCV-64-222, HARDENED CONTAINMENT VENT OUTBOARD ISOLATION VALVE.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 5: [2.5] **PLACE** keylock switch 3-HS-64-221B, HARDENED CONTAINMENT VENT INBOARD PERMISSIVE, in PERM. [2.6] **CHECK** blue indicating light above 3-HS-64-221B, HARDENED CONTAINMENT VENT INBOARD PERMISSIVE, illuminated. **EXAMINER NOTE: ONLY placing keylock switch 3-HS-64-221B in PERMISSIVE is critical in this step.** Expected Action(s): Places keylock switch 3-HS-64-221B, HARDENED CONTAINMENT VENT INBOARD PERMISSIVE, in PERM (permissive) and verifies that the blue light illuminates.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 6: [2.7] **OPEN** 3-FCV-64-221, HARDENED CONTAINMENT VENT INBOARD ISOLATION VALVE. Expected Action(s): Opens 3-FCV-64-221, HARDENED CONTAINMENT VENT INBOARD ISOLATION VALVE.	______ SAT ______UNSAT ______N/A
Step 7: [2.8] **CHECK** Drywell and Suppression Chamber Pressure lowering. Expected Action(s): Verifies Drywell and Suppression Chamber Pressure are lowering.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: Inform the candidate “Another Operator will continue with this procedure. This completes your task”.	

STOP TIME: ___________

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 3 with the following plant conditions:

- A Loss of Coolant Accident (LOCA) has occurred on Unit 3
- The Reactor has SCRAMMED
- Several Control Rods failed to insert on the SCRAM
- All Drywell Sprays have failed
- Drywell Pressure and Temperature are rising

Note: Another Operator is tasked with maintaining Reactor Water Level

INITIATING CUES:

The Nuclear Unit Senior Operator has directed you to emergency vent Primary Containment in accordance with 3-EOI-Appendix-13, Emergency Venting Primary Containment.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Perform Field Actions for a Stuck Open MSRV
JPM NUMBER:	247-U1	REVISION:	4

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-001-AB-01 / Perform actions of Main Steam Relief Valve Stuck Open 1-AOI-1-1			
K/A RATINGS:	RO: 4.1 SRO: 4.2			
K/A No. &STATEMENT:	239002 Relief/Safety Relief Valves A2.03; Ability to (a) predict the impacts of the following on the RELIEF/SAFETY RELIEF VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV.			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	3			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 10 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ______________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator *Date*

Approved by: ______________________ ______________
Site Training Management *Date*

Approved by: ______________________ ______________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 247-U1

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to perform operations to close a stuck open Main Steam Relief Valve (MSRV) from outside the Control Room.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 1-AOI-1

VALIDATIONTIME: 10 minutes

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	2/7/2011	All	Initial Issue
1	9/22/2015	All	Convert to new format
2	8/14/2019	All	Fix typographical issues, make change in one cue due to lack of photograph.
3	1/16/2020	All	JPM format update
4	11/2/2020	All	JPM format update

Procedure Revisions

Procedure	Revision
1-AOI-1-1	5

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, protected equipped……
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 1.

- Reactor Power is 85%.
- Control Room actions to close MSRV 1-PCV-1-22 have **NOT** been successful.
- The Nuclear Unit Supervisor Operator has entered 1-AOI-1-1, Relief Valve Stuck Open.

INITIATING CUES:

The Nuclear Unit Supervisor Operator directs you to attempt to close MSRV 1-PCV-1-22 from outside the Control Room in accordance with 1-AOI-1-1, Step 4.2.3[2].

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

START TIME:__________

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>Step 1:

1-AOI-1, Relief Valve Stuck Open
Section 4.2.3, Attempt to close valve from outside the Control Room:

NOTES
1) 1-PCV-1-22 is an ADS Valve.
2) 1-PCV-1-22 has two power supplies, it will auto transfer on loss of power and is Normal Seeking.
3) Attachment 1 may be addressed for fuse and breaker information.

[2] IF 1-PCV-1-22 is NOT closed, THEN PERFORM the following: (Otherwise N/A this section.)
[2.1] On Panel 1-25-32 PLACE the associated transfer switch 1-XS-1-22, MAIN STM LINE B RELIEF VALVE XFR in EMERG position.

Expected action(s):

SIMULATES placing 1-XS-1-22, MAIN STM LINE B RELIEF VALVE XFR in 1 EMERG on Panel -25-32.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE:
(When 1-XS-1-22 is simulated in EMERG): “1-XS-1-22 is in the EMERG Position.”
(When the Control Room is contacted about the position of 1-PCV-1-22): “SRV 1-22 is OPEN.”</td></tr>
<tr><td>Step 2:

[2.2] IF the SRV does NOT close, THEN PERFORM the following while OBSERVING the indications for the 1-PCV-1-22 on the Acoustic Monitor: (Otherwise N/A)
• CYCLE the 1-HS-1-22C, MAIN STM LINE B RELIEF VALVE, to the following positions several times. CLOSE/AUTO to OPEN to CLOSE/AUTO

Expected action(s):

SIMULATES cycling 1-HS-1-22C, MAIN STM LINE B RELIEF VALVE.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: **(As 1-HS-1-22C is cycled): "1-HS-1-22C is in CLOSE/AUTO or OPEN" (as needed).** **(When the Control Room is contacted about the position of 1-PCV-1-22): "SRV 1-22 is OPEN."**	
Step 3: [2.3] **IF** the SRV does **NOT** close, **THEN PERFORM** the following: (Otherwise N/A) A. **VERIFY** the 1-HS-1-22C, MAIN STM LINE B RELIEF VALVE, in the CLOSE/AUTO position. B. **PLACE** the associated transfer switch 1-XS-1-22, MAIN STM LINE B RELIEF VALVE XFR, in NORM position. Expected action(s): SIMULATES placing 1-HS-1-22C in CLOSE/AUTO then 1-XS-1-22 in NORM.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: **(As each switch is simulated to be re-positioned as required):** **"1-HS-1-22C is in CLOSE/AUTO"** **"1-XS-1-22 is in NORM"**	
EXAMINER CUE: Acknowledge the report given to the MCR.	

STEP / STANDARD	SAT / UNSAT

Step 4:

EXAMINER NOTE: *PANELS WILL NOT BE OPENED*
The candidate may elect to SIMULATE opening the breakers OR pulling fuses.

[2.4] **IF** the SRV does **NOT** close, **THEN REMOVE** the power from 1-PCV-1-22 by performing one of the following: (Opening breakers are the preferred method) (Otherwise N/A)

A. **OPEN** the following breakers: (Preferred method)

- 1A 250V RMOV, Compartment 11C2
- 1B 250V RMOV, Compartment 1C1

OR

B. In Panel 1-25-32 (Bay 3) **PULL** the following fuses as necessary:

- Fuse 1-FU1-001-0022A (Block EE, F2)
- Fuse 1-FU1-001-0022B (Block EE, F7)
- Fuse 1-FU1-001-0022C (Block EE, F12)
- Fuse 1-FU1-001-0022D (Block EE, F15)

Expected action(s):

SIMULATES either opening breakers or pulling fuses.

Critical Step

______ SAT

______UNSAT

______N/A

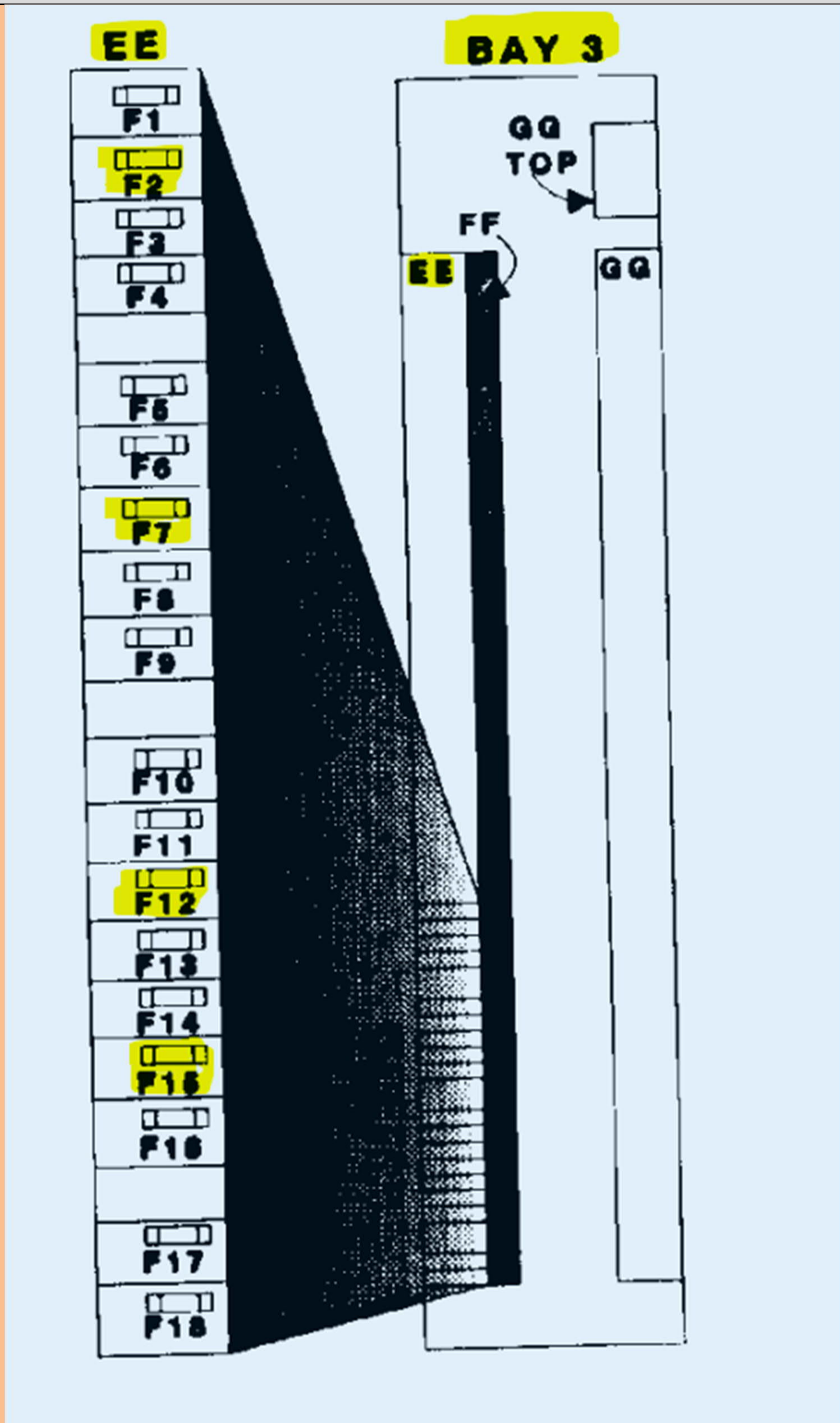
EXAMINER NOTE: If the candidate elects to pull fuses, see the attached page 8 from 1-AOI-1-1, Attachment 1 (Page 4 of 4) for Panel 1-25-32 for the respective fuses in Bay 3.

BFN Unit 1	Relief Valve Stuck Open	1-AOI-1-1 Rev. 0005 Page 34 of 34

Attachment 1
(Page 4 of 4)

Unit 1 SRV Solenoid Power Breaker/Fuse Table, Panels 1-25-32 and 1-LPNL-925-0658

Panel 1-25-32
(Rear)

STEP / STANDARD	SAT / UNSAT
	

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: (When the Control Room is contacted about the position of 1-PCV-1-22): "1-PCV-1-22 is CLOSED." **"Another Operator will continue this procedure. This completes your task"**	

STOP TIME: __________

Job Performance Measure (JPM)

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or "That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 1.

- Reactor Power is 85%.
- Control Room actions to close MSRV 1-PCV-1-22 have **NOT** been successful.
- The Nuclear Unit Supervisor Operator has entered 1-AOI-1-1, Relief Valve Stuck Open.

INITIATING CUES:

The Nuclear Unit Supervisor Operator directs you to attempt to close MSRV 1-PCV-1-22 from outside the Control Room in accordance with 1-AOI-1-1, Step 4.2.3[2].

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Perform Field Actions for a Stuck Open MSRV
JPM NUMBER:	247-U2	REVISION:	4

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-001-AB-01 / Perform actions of Main Steam Relief Valve Stuck Open 2-AOI-1-1			
K/A RATINGS:	RO: 4.1 SRO: 4.2			
K/A No. &STATEMENT:	239002 Relief/Safety Relief Valves A2.03; Ability to (a) predict the impacts of the following on the RELIEF/SAFETY RELIEF VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV.			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	3			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 10 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: _247-U2__

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to perform operations to close a stuck open Main Steam Relief Valve (MSRV) from outside the Control Room.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 2-AOI-1-1

VALIDATIONTIME: __10 minutes__

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records.(Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	2/7/2011	All	Initial Issue
1	9/22/2015	All	Convert to new format
2	8/14/2019	All	Fix typographical issues, make change in one cue due to lack of photograph.
3	1/16/2020	All	JPM format update
4	11/2/2020	All	JPM format update

Procedure Revisions

Procedure	Revision
2-AOI-1	30

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, protected equipped……
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2.

- Reactor Power is 85%.
- Control Room actions to close MSRV 2-PCV-1-22 have **NOT** been successful.
- The Nuclear Unit Supervisor Operator has entered 2-AOI-1-1, Relief Valve Stuck Open.

INITIATING CUES:

The Nuclear Unit Supervisor Operator directs you to attempt to close MSRV 2-PCV-1-22 from outside the Control Room in accordance with 2-AOI-1-1, Step 4.2.3[2].

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

START TIME:__________

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td><u>Step 1</u>:

2-AOI-1, Relief Valve Stuck Open
Section 4.2.3, Attempt to close valve from outside the Control Room:

NOTES
1) 2-PCV-1-22 is an ADS Valve.
2) 2-PCV-1-22 has two power supplies, it will auto transfer on loss of power and is Normal Seeking.
3) Attachment 1 may be addressed for fuse and breaker information.

[2] IF 2-PCV-1-22 is NOT closed, THEN PERFORM the following: (Otherwise N/A this section.)
[2.1] On Panel 2-25-32 PLACE the transfer switch associated 2-XS-1-22, MAIN STM LINE B RELIEF VALVE XFR, in EMERG position.

<u>Expected action(s):</u>

SIMULATES placing 2-XS-1-22, MAIN STM LINE B RELIEF VALVE XFR in EMERG on Panel 1-25-32.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUES:
(When 2-XS-1-22 is simulated in EMERG): “2-XS-1-22 is in the EMERG Position.”
(When the Control Room is contacted about the position of 2-PCV-1-22): “SRV 1-22 is OPEN.”</td></tr>
<tr><td><u>Step 2</u>:

[2.2] IF the SRV does NOT close, THEN PERFORM the following while OBSERVING the indications for the 2-PCV-1-22 on the Acoustic Monitor: (Otherwise N/A)
• CYCLE the 2-HS-1-22C, MAIN STM LINE B RELIEF VALVE, to the following positions several times. CLOSE/AUTO to OPEN to CLOSE/AUTO

<u>Expected action(s):</u>

SIMULATES cycling 2-HS-1-22C, MAIN STM LINE B RELIEF VALVE.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: **(As 2-HS-1-22C is cycled): "2-HS-1-22C is in CLOSE/AUTO or OPEN" (as needed).** **(When the Control Room is contacted about the position of 2-PCV-1-22): "SRV 1-22 is OPEN."**	
Step 3: [2.3] **IF** the SRV does **NOT** close, **THEN PERFORM** the following: (Otherwise N/A) A. **VERIFY** the 2-HS-1-22C, MAIN STM LINE B RELIEF VALVE, in the CLOSE/AUTO position. B. **PLACE** the associated transfer switch 2-XS-1-22, MAIN STM LINE B RELIEF VALVE XFR, in NORM position. Expected action(s): **SIMULATES placing 2-HS-1-22C in CLOSE/AUTO and 2-XS-1-22 in NORM.**	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: **(As each switch is simulated to be re-positioned as required):** **"2-HS-1-22C is in CLOSE/AUTO"** **"2-XS-1-22 is in NORM"**	
EXAMINER CUE: Acknowledge the report given to the MCR.	

STEP / STANDARD	SAT / UNSAT

Step 4:

EXAMINER NOTE: *PANELS WILL NOT BE OPENED*
The candidate may elect to SIMULATE opening the breakers OR pulling fuses.

[2.4] **IF** the SRV does **NOT** close, **THEN REMOVE** the power from 2-PCV-1-22 by performing one of the following: (Opening breakers are the preferred method) (Otherwise N/A)

A. **OPEN** the following breakers (Preferred method)

- 2A 250V RMOV, compartment 11C2
- 2B 250V RMOV, compartment 1C1

OR

B. In Panel 2-25-32 **PULL** the following fuses as necessary

- Fuse 2E-F6E (Block EE, F15)
- Fuse 2E-F4E (Block EE, F7)

Expected action(s):

SIMULATES either opening breakers or pulling fuses.

Critical Step

______ SAT

______UNSAT

______N/A

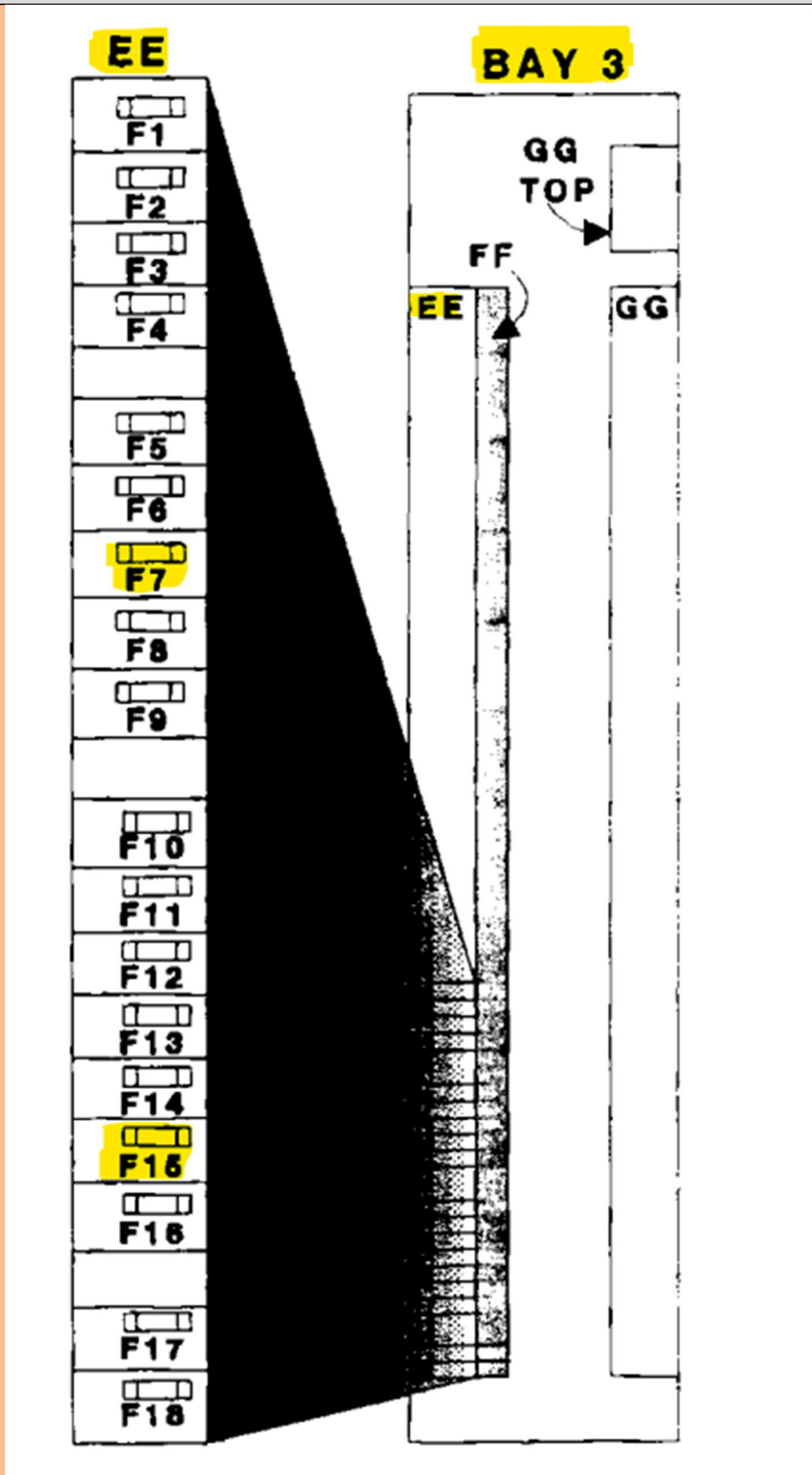
EXAMINER NOTE: If the candidate elects to pull fuses, see the attached page 8 from 2-AOI-1-1, Attachment 1 (Page 3 of 3) for Panel 2-25-32 for the respective fuses in Bay 3.

BFN Unit 2	Relief Valve Stuck Open	2-AOI-1-1 Rev. 0030 Page 28 of 28

Attachment 1
(Page 3 of 3)

UNIT 2 SRV Solenoid Power Breaker/Fuse Table, Panel 25-32

PANEL 2-25-32
(REAR)

STEP / STANDARD	SAT / UNSAT
	

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: (When the Control Room is contacted about the position of 2-PCV-1-22): “2-PCV-1-22 is CLOSED.” **“Another Operator will continue this procedure. This completes your task”**	

STOP TIME: __________

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2.

- Reactor Power is 85%.
- Control Room actions to close MSRV 2-PCV-1-22 have **NOT** been successful.
- The Nuclear Unit Supervisor Operator has entered 2-AOI-1-1, Relief Valve Stuck Open.

INITIATING CUES:

The Nuclear Unit Supervisor Operator directs you to attempt to close MSRV 2-PCV-1-22 from outside the Control Room in accordance with 2-AOI-1-1, Step 4.2.3[2].

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Perform Field Actions for a Stuck Open MSRV
JPM NUMBER:	247-U3	REVISION:	4

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-001-AB-01 / Perform actions of Main Steam Relief Valve Stuck Open 3-AOI-1-1			
K/A RATINGS:	RO: 4.1 SRO: 4.2			
K/A No. &STATEMENT:	239002 Relief/Safety Relief Valves A2.03; Ability to (a) predict the impacts of the following on the RELIEF/SAFETY RELIEF VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV.			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	3			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 10 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ______________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator *Date*

Approved by: ______________________ ______________
Site Training Management *Date*

Approved by: ______________________ ______________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 247-U3

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to perform operations to close a stuck open Main Steam Relief Valve (MSRV) from outside the Control Room.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 3-AOI-1-1

VALIDATIONTIME: 10 minutes

PERFORMANCE TIME: ___________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: ___________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	2/7/2011	All	Initial Issue
1	9/22/2015	All	Convert to new format
2	8/14/2019	All	Fix typographical issues, make change in one cue due to lack of photograph.
3	1/16/2020	All	JPM format update
4	11/2/2020	All	JPM format update

Procedure Revisions

Procedure	Revision
3-AOI-1-1	14

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, protected equipped……
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3.

- Reactor Power is 85%.
- Control Room actions to close MSRV 3-PCV-1-22 have **NOT** been successful.
- The Nuclear Unit Supervisor Operator has entered 3-AOI-1-1, Relief Valve Stuck Open.

INITIATING CUES:

The Nuclear Unit Supervisor Operator directs you to attempt to close MSRV 3-PCV-1-22 from outside the Control Room in accordance with 3-AOI-1-1, Step 4.2.3[2].

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

START TIME:___________

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 1</u>:

3-AOI-1, Relief Valve Stuck Open
Section 4.2.3, Attempt to close valve from outside the Control Room:

NOTES
1) 3-PCV-1-22 is an ADS Valve.
2) 3-PCV-1-22 has two power supplies, it will auto transfer on loss of power and is Normal Seeking.
3) Attachment 1 may be addressed for fuse and breaker information.

[2] IF 3-PCV-1-22 is NOT closed, THEN PERFORM the following: (Otherwise N/A this section.)
[2.1] On Panel 3-25-32 PLACE the transfer switch associated 3-XS-1-22, MAIN STM LINE B RELIEF VALVE XFR, in EMERG position.

<u>Expected action(s):</u>

SIMULATES placing 3-XS-1-22, MAIN STM LINE B RELIEF VALVE XFR in EMERG on Panel -25-32.
</td><td>
Critical Step

______ SAT

______UNSAT

______N/A
</td></tr>
<tr><td colspan="2">
EXAMINER CUE:
(When 3-XS-1-22 is simulated in EMERG): “3-XS-1-22 is in the EMERG Position.”
(When the Control Room is contacted about the position of 3-PCV-1-22): “SRV 1-22 is OPEN.”
</td></tr>
<tr><td>
<u>Step 2</u>:

[2.2] IF the SRV does NOT close, THEN PERFORM the following while OBSERVING the indications for the 3-PCV-1-22 on the Acoustic Monitor: (Otherwise N/A)
• CYCLE the 3-HS-1-22C, MAIN STM LINE B RELIEF VALVE, to the following positions several times. CLOSE/AUTO to OPEN to CLOSE/AUTO

<u>Expected action(s):</u>

SIMULATES cycling 3-HS-1-22C, MAIN STM LINE B RELIEF VALVE.
</td><td>
Critical Step

______ SAT

______UNSAT

______N/A
</td></tr>
</table>

<table>
<tr><td>STEP / STANDARD</td><td>SAT / UNSAT</td></tr>
<tr><td colspan="2">EXAMINER CUE:
(As 3-HS-1-22C is cycled): “3-HS-1-22C is in CLOSE/AUTO or OPEN” (as needed).
(When the Control Room is contacted about the position of 3-PCV-1-22): “SRV 1-22 is OPEN.”</td></tr>
<tr><td><u>Step 3</u>:

[2.3] IF the SRV does NOT close, THEN PERFORM the following: (Otherwise N/A)
A. VERIFY the 3-HS-1-22C, MAIN STM LINE B RELIEF VALVE, in the CLOSE/AUTO position.
B. PLACE the associated transfer switch 3-XS-1-22, MAIN STM LINE B RELIEF VALVE XFR, in NORM position.

<u>Expected action(s):</u>

SIMULATES placing 3-HS-1-22C in CLOSE/AUTO and 3-XS-1-22 in NORM.</td><td>_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE:
(As each switch is simulated to be re-positioned as required):
“3-HS-1-22C is in CLOSE/AUTO”
“3-XS-1-22 is in NORM”</td></tr>
<tr><td colspan="2">EXAMINER CUE: Acknowledge the report given to the MCR.</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT

Step 4:

EXAMINER NOTE: *PANELS WILL NOT BE OPENED*
The candidate may elect to SIMULATE opening the breakers OR pulling fuses.

[2.4] **IF** the SRV does **NOT** close, **THEN REMOVE** the power from 3-PCV-1-22 by performing one of the following: (Opening breakers are the preferred method) (Otherwise N/A)

A. **OPEN** the following breakers: (Preferred method)

- 3A 250V RMOV, Compartment 11C2
- 3B 250V RMOV, Compartment 1C1

OR

B. In Panel 3-25-32 (Bay 3) **PULL** the following fuses as necessary:

- Fuse 3-FU1-001-0022A (Block EE, F2)
- Fuse 3-FU1-001-0022B (Block EE, F7)
- Fuse 3-FU1-001-0022C (Block EE, F12)
- Fuse 3-FU1-001-0022D (Block EE, F15)

Expected action(s):

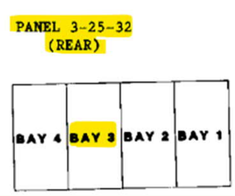
SIMULATES either opening breakers or pulling fuses.

EXAMINER NOTE: If the candidate elects to pull fuses, see the attached page 8 from 3-AOI-1-1, Attachment 1 (Page 3 of 3) for Panel 3-25-32 for the respective fuses in Bay 3.

BFN Unit 3	Relief Valve Stuck Open	3-AOI-1-1 Rev. 0014 Page 29 of 29

Attachment 1
(Page 3 of 3)

Unit 3 SRV Solenoid Power Breaker/Fuse Table, Panel 25 32

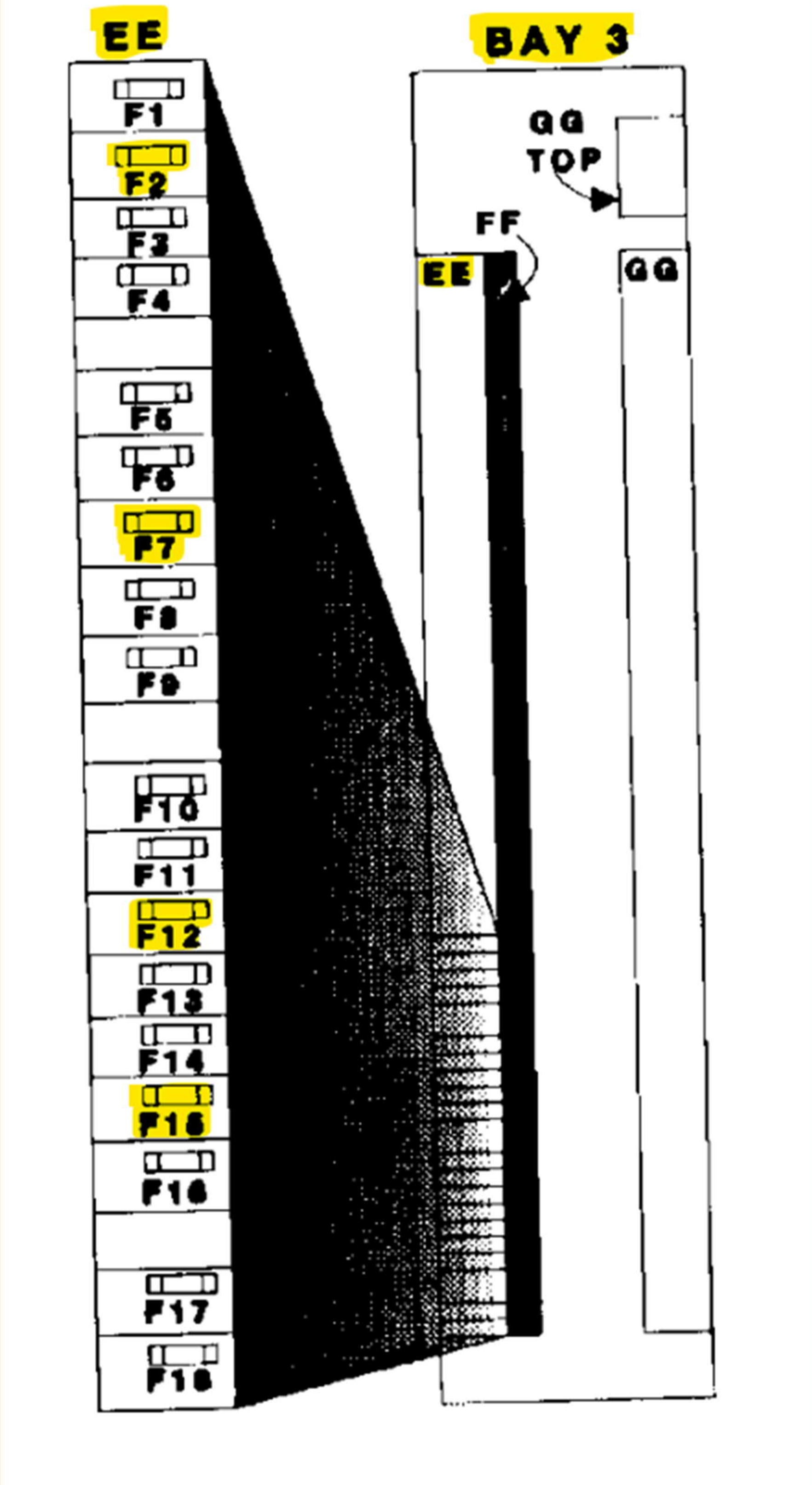


Critical Step

______ SAT

______UNSAT

______N/A

STEP / STANDARD	SAT / UNSAT
	

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: (When the Control Room is contacted about the position of 3-PCV-1-22): "3-PCV-1-22 is CLOSED." **"Another Operator will continue this procedure. This completes your task"**	

STOP TIME: __________

TVA	**Job Performance Measure (JPM)**

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3.

- Reactor Power is 85%.
- Control Room actions to close MSRV 3-PCV-1-22 have **NOT** been successful.
- The Nuclear Unit Supervisor Operator has entered 3-AOI-1-1, Relief Valve Stuck Open.

INITIATING CUES:

The Nuclear Unit Supervisor Operator directs you to attempt to close MSRV 3-PCV-1-22 from outside the Control Room in accordance with 3-AOI-1-1, Step 4.2.3[2].

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Locally Start an EHPM Pump in accordance with 1-EOI Appendix-7L, Alternate Injection System Lineup EHPM System	
JPM NUMBER:	733A-U1	REVISION:	0	

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-000-EM-114/ OPERATE THE EHPM SYSTEM			
K/A RATINGS:	RO: 4.0 SRO: 4.0			
K/A No. &STATEMENT:	295009 Low Reactor Water Level AA1.02; Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor Water Level Control.			
RELATED PRA INFORMATION:	Risk Significant			
SAFETY FUNCTION:	4			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 20 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 733A-U1

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to perform operations necessary to locally start the EHPM Pump.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 1-EOI-APPENDIX-7L

VALIDATION TIME: 10 minutes

PERFORMANCE TIME: ___________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records.(Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/19/2020	All	Initial issue

Procedure Revisions

Procedure	Revision
1-EOI-APP-7L	2

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, protected equipped……
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 1.

- A Loss of Coolant Accident (LOCA) has occurred on Unit 1
- Reactor Water Level is (-) 140 inches and lowering
- Reactor Feed Pumps, HPCI, and RCIC are **NOT** available
- Operation of the Unit 1 EHPM system is **NOT** available at Panel 1-9-21

INITIATING CUES:

The Unit Supervisor directs you to raise Reactor Water Level to (+) 2 to (+) 51 inches using 1-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System, Attachment 2, EHPM Pump Operation from Local Control Panel 1-LNPL-925-6000.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: 1-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System Attachment 2, EHPM Pump Operation from Local Control Panel (LCP) 1-LNPL-925-6000 [1] **IF** 4KV EHPM BD 1 is not energized, **THEN ENERGIZE** from a Supplemental Diesel Generator by performing either Attachment 2, 3, or 4. Expected Action(s): Candidate is required to perform Attachment 4, Supplemental Diesel Generator Operation from Local Control Panel 1-LNPL-925-6000.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Candidate may elect to verify 1-EI-007-0410, EHPM SYS 4KV BUS VOLTS indicates 0 volts on 1-LNPL-925-6000, EMERGENCY HIGH PRESSURE MAKEUP CONTROL PANEL.	
EXAMINER CUE: If asked, from Main Control Room, EHPM NORMAL SOURCE VOLTAGE, 1-EI-7-410A indicates 0 Volts, 4KV EHPM Board 1 is NOT energized.	
EXAMINER NOTE: Alternate path starts in Step 2 below.	

Attachment 4, Supplemental Diesel Generator Operation from Local Control Panel 1-LNPL-925-6000

NOTE

When the SDG Start switch is taken to Start, the Pre-lube oil pumps starts immediately to lubricate the Turbo Charger prior to the diesel starting. Therefore, the diesel will experience a time delay when the diesel start switch is taken to start.

U1 EHPM Bd Rm	
SDG A	SDG B
SUPP DG A START 0-HS-83-A/U1-B	SUPP DG B START 0-HS-83-B/U1-B

STEP / STANDARD	SAT / UNSAT
Step 2: [1] **TRANSFER** Supplemental Diesel Generator control from Main Control Room to Local Control Panel, 1-LPNL-925-6000, by placing 1-XS-083-0414, SUPPLEMENTAL DIESEL CONTROL TRANSFER switch, to LCP. Expected Action(s): Candidate **SIMULATES** placing 1-XS-083-0414, SUPPLEMENTAL DIESEL CONTROL TRANSFER switch, to LCP.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: 1-XS-083-0414, SUPPLEMENTAL DIESEL CONTROL TRANSFER switch is in the LCP position.	
Step 3: [2] **PLACE** SDG handswitch to the START position using the appropriate SDG hand switch listed in table above. (previous page) Expected Action(s): Candidate **SIMULATES** placing either 0-HS-83-A/U1-B, SUPPLEMENTAL DG A START **OR** 0-HS-83-B/U1-B, SUPP DG B START to **START.**	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: If candidate informs Main Control Room that either SUPPLEMENTAL DG A START OR SUPPLEMENTAL DG B START has been placed in START, acknowledge.	
Step 4: [3] **CHECK** EHPM ALTERNATE SOURCE VOLTAGE on 1-EI-83-413 indicates between 3950 Volts and 4400 Volts. Expected Action(s): Candidate checks 1-EI-83-413, EHPM ALTERNATE SOURCE VOLTAGE indicates between 3950 Volts and 4400 Volts.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: When Candidate checks 1-EI-83-413, EHPM ALTERNATE SOURCE VOLTAGE, state EHPM ALTERNATE SOURCE VOLTAGE reads between 3950 Volts and 4400 Volts.	

STEP / STANDARD	SAT / UNSAT
Step 5: [4] **ENSURE** Normal Power Supply, 1-HS-7-1/1B, 4KV EHPM BD NORM FDR, is OPEN. Expected Action(s): Candidate attempts to check 1-HS-7-1/1B, 4KV EHPM BD NORM FDR, is **OPEN** using 1-LI-007-0003/1J, OPEN **GREEN** light lit**, however** 1-IL-007-0003/1J, CLOSED indicates **RED**. Candidate is required to **SIMULATE** taking 1-HS-7-1/1B, 4KV EHPM BD NORM FDR to **TRIP** then verify 1-HS-7-1/1B, 4KV EHPM BD NORM FDR, is **OPEN** using 1-IL-007-0003/1J, OPEN **GREEN** light lit.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: When candidate checks 1-HS-7-1/1B, 4KV EHPM BD NORM FDR, OPEN, state 1-IL-007-0003/1J, CLOSED indicates RED. **After candidate SIMULATES taking 1-HS-7-1/1B to TRIP then verifies 1-HS-7-1/1B, is OPEN using 1-IL-007-0003/1J, OPEN GREEN light lit, state GREEN light lit, RED light is off.**	
Step 6: [5] **CLOSE** Alternate Power Supply, 1-HS-7-1/5B, 4KV EHPM BD ALT FDR, by placing to CLOSE. Expected Action(s): Candidate **SIMULATES** taking 1-HS-7-1/5B, 4KV EHPM BD ALT FDR, to CLOSE. Verifies 1-IL-007-0003/5J, CLOSED **RED** light lit and 1-IL-007-0003/5J, OPEN **GREEN** light off.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: When candidate SIMULATES taking 1-HS-007-0001/5B, 4KV EHPM BD ALT FDR, to CLOSE, state 1-IL-007-0002/5J, CLOSED RED light lit and 1-IL-007-0002/5J, OPEN GREEN light off.	
Step 7: [6] **NOTIFY** Unit 1 MCR that 4KV EHPM BD 1 is now powered by Supplemental Diesel Generator. Expected Action(s): Candidate **NOTIFIES** Unit 1 MCR that 4KV EHPM BD 1 is now powered by Supplemental Diesel Generator.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: Acknowledge as Unit 1 MCR that 4KV EHPM BD 1 is now powered by Supplemental Diesel Generator. (Step 7 above).	
EXAMINER NOTE: Step 7 above ends Alternate Path from Attachment 4, Step 8 below returns to Attachment 2, EHPM Pump Operation from Local Control Panel (LCP) 1-LNPL-925-6000.	
Step 8: [2] **TRANSFER** Unit 1 EHPM Pump control from Main Control Room to Local Control Panel, 1-LPNL-925-6000, by placing 1-XS-7-411, EHPM SYS CONTROL TRANSFER switch, to LCP. Expected Action(s): Candidate **SIMULATES** placing switch 1-XS-007-0411, EHPM SYS CONTROL TRANSFER, in the LCP position.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: 1-XS-007-0411, EHPM SYS CONTROL TRANSFER switch is in the LCP position.	
Step 9: [3] **ESTABLISH** Unit 1 RPV injection in BATCH mode as follows: [3.1] **START** EHPM PUMP by placing 1-HS-7-1B, EHPM PUMP START-STOP to START. Expected Action(s): Candidate **SIMULATES** placing 1-HS-7-1B, EHPM PUMP START-STOP to START.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: 1-HS-7-1B, EHPM PUMP START-STOP switch is in the START position.	
Step 10: [3.2] **NOTIFY** Unit 1 Main Control Room (MCR) that the next step will inject to the RPV. Expected Action(s): Candidate notifies the MCR that the next step will inject water into the RPV.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: Acknowledge the report given to the MCR.	

STEP / STANDARD	SAT / UNSAT
Step 11: [3.3] **THROTTLE** 1-FCV-007-0008, EHPM PUMP INJECTION VALVE as necessary to establish flow 950-1250 GPM as indicated on 1-FI-007-0403, EHPM SYS NORMAL FLOW. Expected Action(s): Candidate **SIMULATES** throttling 1-FCV-007-0008, EHPM PUMP INJECTION VALVE to establish flow 950-1250 GPM as indicated on 1-FI-007-0403, EHPM SYS NORMAL FLOW.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: As candidate SIMULATES throttling 1-FCV-007-0008, EHPM PUMP INJECTION VALVE, provide indication that the flow indicated on 1-FI-007-0403, EHPM SYS NORMAL FLOW is rising and eventually in the range of 950-1250 GPM.	
Step 12: [3.4] **MONITOR** Unit 1 RPV Level on 1-LI-003-0148A, RPV LEVEL 'A', and 1-LI-003-0148B, RPV LEVEL 'B'. Expected Action(s): Candidate monitors 1-LI-003-0148A, RPV LEVEL 'A', and 1-LI-003-0148B, RPV LEVEL 'B'.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: Provide indication that Reactor Water Level as indicated on 1-LI-003-0148A, RPV LEVEL 'A', and 1-LI-003-0148B, RPV LEVEL 'B' is (-) 130 inches and rising.	
EXAMINER NOTE: (Once the CUE is given for Reactor Water Level rising) "Another Operator will be tasked with completing the procedure".	

STOP TIME: __________

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 1.

- A Loss of Coolant Accident (LOCA) has occurred on Unit 1
- Reactor Water Level is (-) 140 inches and lowering
- Reactor Feed Pumps, HPCI, and RCIC are **NOT** available
- Operation of the Unit 1 EHPM system is **NOT** available at Panel 1-9-21

INITIATING CUES:

The Unit Supervisor directs you to raise Reactor Water Level to (+) 2 to (+) 51 inches using 1-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System, Attachment 2, EHPM Pump Operation from Local Control Panel 1-LNPL-925-6000.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Locally Start an EHPM Pump in accordance with 2-EOI Appendix-7L, Alternate Injection System Lineup EHPM System	
JPM NUMBER:	733A-U2	REVISION:	1	

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-000-EM-114/ OPERATE THE EHPM SYSTEM			
K/A RATINGS:	RO: 4.0 SRO: 4.0			
K/A No. &STATEMENT:	295009 Low Reactor Water Level AA1.02; Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor Water Level Control.			
RELATED PRA INFORMATION:	Risk Significant			
SAFETY FUNCTION:	4			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 20 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

TVA	**Job Performance Measure (JPM)**

OPERATOR: ______________________________ JPM Number: 733A-U2

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to perform operations necessary to locally start the EHPM Pump.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 2-EOI-APPENDIX-7L

VALIDATIONTIME: 20 minutes

PERFORMANCE TIME: ___________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records.(Otherwise just retain this page.)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/19/2020	All	Initial issue
1	02/25/2021	All	Procedure update

Procedure Revisions

Procedure	Revision
2-EOI-APP-7L	4

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, protected equipped……
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2.

- A Loss of Coolant Accident (LOCA) has occurred on Unit 2
- Reactor Water Level is (-) 140 inches and lowering
- Reactor Feed Pumps, HPCI, and RCIC are **NOT** available
- Operation of the Unit 2 EHPM system is **NOT** available at Panel 2-9-21

INITIATING CUES:

The Unit Supervisor directs you to raise Reactor Water Level to (+) 2 to (+) 51 inches using 2-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System, Attachment 2, EHPM Pump Operation from Local Control Panel 2-LNPL-925-6000.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: 2-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System Attachment 1, EHPM Pump Operation from Local Control Panel (LCP) 2-LNPL-925-6000 [1] **IF** 4KV EHPM BD 2 is not energized, **THEN ENERGIZE** from a Supplemental Diesel Generator by performing either Attachment 2, 3, or 4. Expected Action(s): Candidate is required to perform Attachment 4, Supplemental Diesel Generator Operation from Local Control Panel 2-LNPL-925-6000.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Candidate may elect to verify 2-EI-007-0410, EHPM SYS 4KV BUS VOLTS indicates 0 volts on 2-LNPL-925-6000, EMERGENCY HIGH PRESSURE MAKEUP CONTROL PANEL.	
EXAMINER CUE: If asked, from Main Control Room, EHPM NORMAL SOURCE VOLTAGE, 2-EI-7-410A indicates 0 Volts, 4KV EHPM Board 2 is NOT energized.	
EXAMINER NOTE: Alternate path starts in Step 2.	

Attachment 4, Supplemental Diesel Generator Operation from Local Control Panel 2-LNPL-925-6000

NOTE

When the SDG Start switch is taken to Start, the Pre-lube oil pumps starts immediately to lubricate the Turbo Charger prior to the diesel starting. Therefore, the diesel will experience a time delay when the diesel start switch is taken to start.

U2 EHPM Bd Rm	
SDG A	SDG B
SUPP DG A START 0-HS-83-A/U2-B	SUPP DG B START 0-HS-83-B/U2-B

STEP / STANDARD	SAT / UNSAT
Step 2: [1] **TRANSFER** Supplemental Diesel Generator control from Main Control Room to Local Control Panel, 2-LPNL-925-6000, by placing 2-XS-083-0414, SUPPLEMENTAL DIESEL CONTROL TRANSFER switch, to LCP. Expected Action(s): Candidate **SIMULATES** placing 2-XS-083-0414, SUPPLEMENTAL DIESEL CONTROL TRANSFER switch, to LCP.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: 2-XS-083-0414, SUPPLEMENTAL DIESEL CONTROL TRANSFER switch is in the LCP position.	
Step 3: [2] **PLACE** SDG handswitch to the START position using the appropriate SDG hand switch listed in table above. (previous page) Expected Action(s): Candidate **SIMULATES** placing either 0-HS-83-A/U2-B, SUPPLEMENTAL DG A START **OR** 0-HS-83-B/U2-B, SUPPLEMENTAL DG B START to **START.**	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: If candidate informs Main Control Room that either SUPPLEMENTAL DG A START OR SUPPLEMENTAL DG B START has been placed in START, acknowledge.	
Step 4: [3] **CHECK** EHPM ALTERNATE SOURCE VOLTAGE on 2-EI-83-413 indicates between 3950 Volts and 4400 Volts. Expected Action(s): Candidate checks 2-EI-83-413, EHPM ALTERNATE SOURCE VOLTAGE indicates between 3950 Volts and 4400 Volts.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: When Candidate checks 2-EI-83-413, EHPM ALTERNATE SOURCE VOLTAGE, state EHPM ALTERNATE SOURCE VOLTAGE reads between 3950 Volts and 4400 Volts.	

STEP / STANDARD	SAT / UNSAT
Step 5: [4] **ENSURE** Normal Power Supply, 2-HS-7-2/1B, 4KV EHPM BD NORM FDR, is OPEN. Expected Action(s): Candidate attempts to check 2-HS-7-2/1B, 4KV EHPM BD NORM FDR, is **OPEN** using 2-IL-007-0003/1J, OPEN **GREEN** light lit**, however** 2-IL-007-0003/1J, CLOSED indicates **RED**. Candidate is required to **SIMULATE** taking 2-HS-7-2/1B, 4KV EHPM BD NORM FDR to **TRIP** then verify 2-HS-7-2/1B, 4KV EHPM BD NORM FDR, is **OPEN** using 2-IL-007-0003/1J, OPEN **GREEN** light lit.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: When candidate checks 2-HS-7-2/1B, 4KV EHPM BD NORM FDR, OPEN, state 2-IL-007-0003/1J, CLOSED indicates RED. **After candidate SIMULATES taking 2-HS-7-2/1B to TRIP then verifies 2-HS-7-2/1B, is OPEN using 2-IL-007-0003/1J, OPEN GREEN light lit, state GREEN light lit, RED light is off.**	
Step 6: [5] **CLOSE** Alternate Power Supply, 2-HS-7-2/5B, 4KV EHPM BD ALT FDR, by placing to CLOSE. Expected Action(s): Candidate **SIMULATES** taking 2-HS-7-2/5B, 4KV EHPM BD ALT FDR, to CLOSE. Verifies 2-IL-007-0003/5J, CLOSED **RED** light lit and 2-IL-007-0003/5J, OPEN **GREEN** light off.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: When candidate SIMULATES taking 2-HS-7-2/5B, 4KV EHPM BD ALT FDR, to CLOSE, state 2-IL-007-0002/5J, CLOSED RED light lit and 2-IL-007-0002/5J, OPEN GREEN light off.	
Step 7: [6] **NOTIFY** Unit 2 MCR that 4KV EHPM BD 2 is now powered by Supplemental Diesel Generator. Expected Action(s): Candidate **NOTIFIES** Unit 2 MCR that 4KV EHPM BD 2 is now powered by Supplemental Diesel Generator.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: Acknowledge as Unit 2 MCR that 4KV EHPM BD 2 is now powered by Supplemental Diesel Generator. (Step 7 above).	
EXAMINER NOTE: Step 7 above ends Alternate Path from Attachment 3, **Step 8 below returns to Attachment 1, EHPM Pump Operation from Local Control Panel (LCP) 2-LNPL-925-6000.**	
Step 8: [2] **TRANSFER** Unit 2 EHPM Pump control from Main Control Room to Local Control Panel, 2-LPNL-925-6000, by placing 2-XS-7-411, EHPM SYS CONTROL TRANSFER switch, to LCP. Expected Action(s): Candidate **SIMULATES** placing switch 2-XS-007-0411, EHPM SYS CONTROL TRANSFER, in the LCP position.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: 2-XS-007-0411, EHPM SYS CONTROL TRANSFER switch is in the LCP position.	
Step 9: [3] **ESTABLISH** Unit 2 RPV injection in BATCH mode as follows: [3.1] **START** EHPM PUMP by placing 2-HS-7-1B, EHPM PUMP START-STOP to START. Expected Action(s): Candidate **SIMULATES** placing 2-HS-7-1B, EHPM PUMP START-STOP to START.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: 2-HS-7-1B, EHPM PUMP START-STOP switch is in the START position.	
Step 10: [3.2] **NOTIFY** Unit 2 Main Control Room (MCR) that the next step will inject to the RPV. Expected Action(s): Candidate notifies the MCR that the next step will inject water into the RPV.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: Acknowledge the report given to the MCR.	

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td><u>Step 11</u>:

[3.3] THROTTLE 2-FCV-007-0008, EHPM PUMP INJECTION VALVE as necessary to establish flow 950-1250 GPM as indicated on 2-FI-007-0403, EHPM SYS NORMAL FLOW.

<u>Expected Action(s):</u>

Candidate SIMULATES throttling 2-FCV-007-0008, EHPM PUMP INJECTION VALVE to establish flow 950-1250 GPM as indicated on 2-FI-007-0403, EHPM SYS NORMAL FLOW.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: As candidate SIMULATES throttling 2-FCV-007-0008, EHPM PUMP INJECTION VALVE, provide indication that the flow indicated on 2-FI-007-0403, EHPM SYS NORMAL FLOW is rising and eventually in the range of 950-1250 GPM.</td></tr>
<tr><td><u>Step 12</u>:

[3.4] MONITOR Unit 2 RPV Level on 2-LI-003-0148A, RPV LEVEL ‘A’, and 2-LI-003-0148B, RPV LEVEL ‘B’.

<u>Expected Action(s):</u>

Candidate monitors 2-LI-003-0148A, RPV LEVEL ‘A’, and 2-LI-003-0148B, RPV LEVEL ‘B’.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: Provide indication that Reactor Water Level as indicated on 2-LI-003-0148A, RPV LEVEL ‘A’, and 2-LI-003-0148B, RPV LEVEL ‘B’ is (-) 130 inches and rising.</td></tr>
<tr><td colspan="2">EXAMINER NOTE: (Once the CUE is given for Reactor Water Level rising) “Another Operator will be tasked with completing the procedure”.</td></tr>
</table>

STOP TIME: __________

	Job Performance Measure (JPM)
TVA	

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2.

- A Loss of Coolant Accident (LOCA) has occurred on Unit 2
- Reactor Water Level is (-) 140 inches and lowering
- Reactor Feed Pumps, HPCI, and RCIC are **NOT** available
- Operation of the Unit 2 EHPM system is **NOT** available at Panel 2-9-21

INITIATING CUES:

The Unit Supervisor directs you to raise Reactor Water Level to (+) 2 to (+) 51 inches using 2-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System, Attachment 2, EHPM Pump Operation from Local Control Panel 2-LNPL-925-6000.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Locally Start an EHPM Pump in accordance with 3-EOI Appendix-7L, Alternate Injection System Lineup EHPM System		
JPM NUMBER:		733A-U3	REVISION:	0	

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-000-EM-114/ OPERATE THE EHPM SYSTEM			
K/A RATINGS:	RO: 4.0 SRO: 4.0			
K/A No. &STATEMENT:	295009 Low Reactor Water Level AA1.02; Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor Water Level Control.			
RELATED PRA INFORMATION:	Risk Significant			
SAFETY FUNCTION:	4			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 20 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) Y

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: 733A-U3

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to perform operations necessary to locally start the EHPM Pump.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 3-EOI-APPENDIX-7L

VALIDATIONTIME: 20 minutes

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records.(Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/19/2020	All	Initial issue

Procedure Revisions

Procedure	Revision
3-EOI-APP-7L	3

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, protected equipped……
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3.

- A Loss of Coolant Accident (LOCA) has occurred on Unit 3
- Reactor Water Level is (-) 140 inches and lowering
- Reactor Feed Pumps, HPCI, and RCIC are **NOT** available
- Operation of the Unit 3 EHPM system is **NOT** available at Panel 3-9-21

INITIATING CUES:

The Unit Supervisor directs you to raise Reactor Water Level to (+) 2 to (+) 51 inches using 3-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System, Attachment 2, EHPM Pump Operation from Local Control Panel 3-LNPL-925-6000.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: 3-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System Attachment 1, EHPM Pump Operation from Local Control Panel (LCP) 3-LNPL-925-6000 [1] **IF** 4KV EHPM BD 3 is not energized, **THEN ENERGIZE** from a Supplemental Diesel Generator by performing either Attachment 2, 3, or 4. Expected Action(s): Candidate is required to perform Attachment 4, Supplemental Diesel Generator Operation from Local Control Panel 3-LNPL-925-6000.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Candidate may elect to verify 3-EI-007-0410, EHPM SYS 4KV BUS VOLTS indicates 0 volts on 3-LNPL-925-6000, EMERGENCY HIGH PRESSURE MAKEUP CONTROL PANEL.	
EXAMINER CUE: If asked, from Main Control Room, EHPM NORMAL SOURCE VOLTAGE, 3-EI-7-410A indicates 0 Volts, 4KV EHPM Board 3 is NOT energized.	
EXAMINER NOTE: Alternate path starts in Step 2.	

Attachment 4, Supplemental Diesel Generator Operation from Local Control Panel 3-LNPL-925-6000

NOTE

When the SDG Start switch is taken to Start, the Pre-lube oil pumps starts immediately to lubricate the Turbo Charger prior to the diesel starting. Therefore, the diesel will experience a time delay when the diesel start switch is taken to start.

U3 EHPM Bd Rm	
SDG A	SDG B
SUPP DG A START 3-HS-83-A/U3-B	SUPP DG B START 3-HS-83-B/U3-B

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td><u>Step 2</u>:

[1] TRANSFER Supplemental Diesel Generator control from Main Control Room to Local Control Panel, 3-LPNL-925-6000, by placing 3-XS-083-0414, SUPPLEMENTAL DIESEL CONTROL TRANSFER switch, to LCP.

<u>Expected Action(s):</u>

Candidate SIMULATES placing 3-XS-083-0414, SUPPLEMENTAL DIESEL CONTROL TRANSFER switch, to LCP.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: 3-XS-083-0414, SUPPLEMENTAL DIESEL CONTROL TRANSFER switch is in the LCP position.</td></tr>
<tr><td><u>Step 3</u>:

[2] PLACE SDG handswitch to the START position using the appropriate SDG hand switch listed in table above. (previous page)

<u>Expected Action(s):</u>

Candidate SIMULATES placing either 3-HS-83-A/U3-B, SUPPLEMENTAL DG A START OR 3-HS-83-B/U3-B, SUPPLEMENTAL DG B START to START.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: If candidate informs Main Control Room that either SUPPLEMENTAL DG A START OR SUPPLEMENTAL DG B START has been placed in START, acknowledge.</td></tr>
<tr><td><u>Step 4</u>:

[3] CHECK EHPM ALTERNATE SOURCE VOLTAGE on 3-EI-83-413 indicates between 3950 Volts and 4400 Volts.

<u>Expected Action(s):</u>

Candidate checks 3-EI-83-413, EHPM ALTERNATE SOURCE VOLTAGE indicates between 3950 Volts and 4400 Volts.</td><td>______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: When Candidate checks 3-EI-83-413, EHPM ALTERNATE SOURCE VOLTAGE, state EHPM ALTERNATE SOURCE VOLTAGE reads between 3950 Volts and 4400 Volts.</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 5: [4] **ENSURE** Normal Power Supply, 3-HS-007-0003/1B, 4KV EHPM BD NORM FDR, is OPEN. Expected Action(s): Candidate attempts to check 3-HS-7-3/1B, 4KV EHPM BD NORM FDR, is **OPEN** using 3-LI-007-0003/1J, OPEN **GREEN** light lit**, however** 3-IL-007-0003/1J, CLOSED indicates **RED**. Candidate is required to **SIMULATE** taking 3-HS-7-3/1B, 4KV EHPM BD NORM FDR to **TRIP** then verify 3-HS-7-3/1B, 4KV EHPM BD NORM FDR, is **OPEN** using 3-IL-007-0003/1J, OPEN **GREEN** light lit.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: When candidate checks 3-HS-7-3/1B, 4KV EHPM BD NORM FDR, OPEN, state 3-IL-007-0003/1J, CLOSED indicates RED. **After candidate SIMULATES taking 3-HS-7-3/1B to TRIP then verifies 3-HS-7-3/1B, is OPEN using 3-IL-007-0003/1J, OPEN GREEN light lit, state GREEN light lit, RED light is off.**	
Step 6: [5] **CLOSE** Alternate Power Supply, 3-HS-7-3/5B, 4KV EHPM BD ALT FDR, by placing to CLOSE. Expected Action(s): Candidate **SIMULATES** taking 3-HS-7-3/5B, 4KV EHPM BD ALT FDR, to CLOSE. Verifies 3-IL-007-0003/5J, CLOSED **RED** light lit and 3-IL-007-0003/5J, OPEN **GREEN** light off.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: When candidate SIMULATES taking 3-HS-7-3/5B, 4KV EHPM BD ALT FDR, to CLOSE, state 3-IL-007-0002/5J, CLOSED RED light lit and 3-IL-007-0002/5J, OPEN GREEN light off.	
Step 7: [6] **NOTIFY** Unit 3 MCR that 4KV EHPM BD 3 is now powered by Supplemental Diesel Generator. Expected Action(s): Candidate **NOTIFIES** Unit 3 MCR that 4KV EHPM BD 3 is now powered by Supplemental Diesel Generator.	_____ SAT _____UNSAT _____N/A

<table>
<tr><td>STEP / STANDARD</td><td>SAT / UNSAT</td></tr>
<tr><td colspan="2">EXAMINER CUE: Acknowledge as Unit 3 MCR that 4KV EHPM BD 3 is now powered by Supplemental Diesel Generator. (Step 7 above).</td></tr>
<tr><td colspan="2">EXAMINER NOTE: Step 7 above ends Alternate Path from Attachment 3,
Step 8 below returns to Attachment 1, EHPM Pump Operation from Local Control Panel (LCP) 3-LNPL-925-6000.</td></tr>
<tr><td><u>Step 8</u>:

[2] TRANSFER Unit 3 EHPM Pump control from Main Control Room to Local Control Panel, 3-LPNL-925-6000, by placing 3-XS-7-411, EHPM SYS CONTROL TRANSFER switch, to LCP.

<u>Expected Action(s):</u>

Candidate SIMULATES placing switch 3-XS-007-0411, EHPM SYS CONTROL TRANSFER, in the LCP position.</td><td>Critical Step

_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: 3-XS-007-0411, EHPM SYS CONTROL TRANSFER switch is in the LCP position.</td></tr>
<tr><td><u>Step 9</u>:

[3] ESTABLISH Unit 3 RPV injection in BATCH mode as follows:
[3.1] START EHPM PUMP by placing 3-HS-7-1B, EHPM PUMP START-STOP to START.

<u>Expected Action(s):</u>

Candidate SIMULATES placing 3-HS-7-1B, EHPM PUMP START-STOP to START.</td><td>Critical Step

_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: 3-HS-7-1B, EHPM PUMP START-STOP switch is in the START position.</td></tr>
<tr><td><u>Step 10</u>:

[3.2] NOTIFY Unit 3 Main Control Room (MCR) that the next step will inject to the RPV.

<u>Expected Action(s):</u>

Candidate notifies the MCR that the next step will inject water into the RPV.</td><td>_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: Acknowledge the report given to the MCR.</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 11: [3.3] **THROTTLE** 3-FCV-007-0008, EHPM PUMP INJECTION VALVE as necessary to establish flow 950-1250 GPM as indicated on 3-FI-007-0403, EHPM SYS NORMAL FLOW. Expected Action(s): Candidate **SIMULATES** throttling 3-FCV-007-0008, EHPM PUMP INJECTION VALVE to establish flow 950-1250 GPM as indicated on 3-FI-007-0403, EHPM SYS NORMAL FLOW.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: As candidate SIMULATES throttling 3-FCV-007-0008, EHPM PUMP INJECTION VALVE, provide indication that the flow indicated on 3-FI-007-0403, EHPM SYS NORMAL FLOW is rising and eventually in the range of 950-1250 GPM.	
Step 12: [3.4] **MONITOR** Unit 3 RPV Level on 3-LI-003-0148A, RPV LEVEL 'A', and 3-LI-003-0148B, RPV LEVEL 'B'. Expected Action(s): Candidate monitors 3-LI-003-0148A, RPV LEVEL 'A', and 3-LI-003-0148B, RPV LEVEL 'B'.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: Provide indication that Reactor Water Level as indicated on 3-LI-003-0148A, RPV LEVEL 'A', and 3-LI-003-0148B, RPV LEVEL 'B' **is (-) 130 inches and rising.**	
EXAMINER NOTE: (Once the CUE is given for Reactor Water Level rising) "Another Operator will be tasked with completing the procedure".	

STOP TIME: __________

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3.

- A Loss of Coolant Accident (LOCA) has occurred on Unit 3
- Reactor Water Level is (-) 140 inches and lowering
- Reactor Feed Pumps, HPCI, and RCIC are **NOT** available
- Operation of the Unit 3 EHPM system is **NOT** available at Panel 3-9-21

INITIATING CUES:

The Unit Supervisor directs you to raise Reactor Water Level to (+) 2 to (+) 51 inches using 3-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System, Attachment 2, EHPM Pump Operation from Local Control Panel 3-LNPL-925-6000.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Place Unit 1 Divisional ECCS Analog Trip Unit Inverter in Service in accordance with 0-OI-57C, 208V/120V AC Electrical System, Section 5.2.
JPM NUMBER:	306-U1	REVISION:	9

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-57C-NO-01			
K/A RATINGS:	RO: 3.2 SRO: 3.3			
K/A No. &STATEMENT:	263000 D.C. Electrical Distribution A3.01; Ability to monitor automatic operations of D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights.			
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A			
SAFETY FUNCTION:	6			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: xx min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer Date

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator Date

Approved by: ______________________ ____________
Site Training Management Date

Approved by: ______________________ ____________
Site Training Program Owner Date

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: _306-U1_

RO _____ SRO _____ DATE: ____________

TASK EXPECTED ACTION(S): Examinee is expected to place Divisional Emergency Core Cooling System (ECCS) Analog Trip Unit (ATU) Inverter in Service – Division I (DIV I)

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 0-OI-57C

VALIDATION TIME: _xx minutes_

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
8	10/29/2020	All	Update JPM
9	02/25/2021	All	Procedure update

Procedure Revisions

Procedure	Revision
0-OI-57C	134

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, protected equipped……
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 1.

- Unit 1 is in cold Shutdown
- The Unit 1 Division I ECCS Analog Trip Unit Inverter was shutdown and taken out of service for preventive maintenance
- The maintenance has been completed and the clearance released and removed

INITIATING CUES: The Shift Manager has directed you to return the Unit 1 ECCS Analog Trip Unit (ATU) Inverter - Division I to service in accordance with 0-OI-57C, 208V/120V AC Electrical System, Section 5.2.

NOTE:

All Precautions and Limitations in Section 3.0 have been reviewed.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: 0-OI-57C, 208V/120V AC Electrical System Section 5.2, Placing Unit 1 ECCS ATU Inverter Division I, 1-INVT-256-0001 in Service [1] **ENSURE** the 1-INVT-256-0001, ECCS ATU INVERTER Div I is shut down. **REFER TO** Section 7.3. **CAUTION** Unit 1, 1-INVT-256-0001(2), ECCS ATU INVERTER DIV I(II) requires a 60 second wait period prior to restart. Expected Action(s): **N/A, given in the INITIAL CONDITIONS.**	_____ SAT _____UNSAT _____N/A
Step 2: [2] **REVIEW** all Precautions and Limitations in Section 3.0. Expected Action(s): **N/A, given in the INITIAL CONDITIONS.**	_____ SAT _____UNSAT _____N/A
Step 3: [3] **CHECK CLOSED** 1-INVT-256-0001, ECCS ATU INVERTER Div I, on the following 250V Reactor MOV Boards: 1B - Compartment 8A (Div I) Expected Action(s): **Locates** 250V RMOV Board 1B - Compartment 8A (Div I) and **SIMULATES** checking CLOSED the breaker for 1-INVT-256-0001, ECCS ATU INVERTER Div I	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: After the breaker is SIMULATED check CLOSED, state the breaker for 1-INVT-256-0001, ECCS ATU INVERTER Div I, is CLOSED	

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 4:</u>

NOTE
Steps 5.2[4] through 5.2[10] are performed from 1-INVT-256-0001(2), ECCS ATU INVERTER Div I(II), located in Electrical Board Room 1B(1A) EL 593' (621').

[4] ENSURE the 1-IL-256-0001/P1, DC INPUT AVAILABLE is illuminated.

<u>Expected Action(s):</u>

ENSURES 1-IL-256-0001/P1, DC INPUT AVAILABLE is illuminated.
</td><td>
______ SAT

______UNSAT

______N/A
</td></tr>
<tr><td>
<u>Step 5:</u>

[5] DEPRESS and HOLD 1-HS-256-0001/S4, PRECHARGE.

<u>Expected Action(s):</u>

SIMULATES DEPRESSING and HOLDING 1-HS-256-0001/S4, PRECHARGE.
</td><td>
Critical Step

______ SAT

______UNSAT

______N/A
</td></tr>
<tr><td colspan="2">EXAMINER CUE: After SIMULATING DEPRESSING and HOLDING 1-HS-256-0001/S4, PRECHARGE, state you have held the PRECHARGE pushbutton for 10 seconds.</td></tr>
<tr><td>
<u>Step 6:</u>

[6] WHEN the 1-IL-256-0001/P4, PRECHARGE illuminates, THEN PERFORM the following:

[6.1] RELEASE 1-HS-256-0001/S4, PRECHARGE.

[6.2] CLOSE 1-BRK-256-0001/B1, DC INPUT.

<u>Expected Action(s):</u>

SIMULATES RELEASING 1-HS-256-0001/S4, PRECHARGE and SIMULATES CLOSES 1-BRK-256-0001/B1, DC INPUT.
</td><td>
Critical Step

______ SAT

______UNSAT

______N/A
</td></tr>
<tr><td colspan="2">EXAMINER CUE: After SIMULATED, state 1-HS-256-0001/S4, PRECHARGE has been RELEASED and 1-BRK-256-0001/B1, DC INPUT is CLOSED.</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 7: [7] CHECK AC Volts is between 117 and 123 volts on 1-EI-256-0001/V1, AC VOLTMETER(V1). Expected Action(s): **CHECKS AC Volts between 117 and 123 volts on 1-EI-256-0001/V1, AC VOLTMETER(V1).**	______ SAT ______UNSAT ______N/A
Step 8: [8] CLOSE 1-BKR-256-0001/B2, AC SYSTEM OUTPUT. Expected Action(s): **SIMULATES CLOSING 1-BKR-256-0001/B2, AC SYSTEM OUTPUT.**	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: After SIMULATED, state 1-BKR-256-0001/B2, AC SYSTEM OUTPUT is CLOSED	
Step 9: [9] **DEPRESS** 1-HS-256-0001(2)/1, ALARM RESET. Expected Action(s): **SIMULATES DEPRESSING** 1-HS-256-0001/1, ALARM RESET.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: After SIMULATED, state ALL alarms are clear.	
Step 10: [10] **CHECK the following parameters on 1-INVT-256-0001, ECCS ATU INVERTER Div I:** A. 1-IL-256-0001/P2, LOW DC VOLTAGE is OFF. B. 1-IL-256-0001/P3, AC OVERVOLTAGE is OFF. C. AC current is less than 42 Amperes on 1-II-256-0001, AC AMMETER(A1).	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
D. AC voltage is between 117 and 123 volts on 1-EI-256-0001/V1, AC VOLTMETER(V1). E. Frequency is between 59.7 and 60.3 Hz on 1-SI-256-0001/E1, FREQUENCY METER(E1). Expected Action(s): **Locates the correct parameters and after CUE (below), accepts readings as normal.**	
EXAMINER CUE: **As each parameter is checked, as applicable, state:** • **AC current is reading 5 amps** • **AC voltage is reading 125 volts** • **Frequency is reading 60.1 Hz**	
EXAMINER CUE: After the examinee repeats the parameter readings, state This completes your task.	

STOP TIME: ___________

TVA	**Job Performance Measure (JPM)**

Provide to Applicant

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 1.

- Unit 1 is in cold Shutdown
- The Unit 1 Division I ECCS Analog Trip Unit Inverter was shutdown and taken out of service for preventive maintenance
- The maintenance has been completed and the clearance released and removed

INITIATING CUES: The Shift Manager has directed you to return the Unit 1 ECCS Analog Trip Unit (ATU) Inverter - Division I to service in accordance with 0-OI-57C, 208V/120V AC Electrical System, Section 5.2.

NOTE:

All Precautions and Limitations in Section 3.0 have been reviewed.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Place Unit 2 Divisional ECCS Analog Trip Unit Inverter in Service in accordance with 0-OI-57C, 208V/120V AC Electrical System, Section 5.3.1.
JPM NUMBER:	306-U2	REVISION:	8

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-57C-NO-01			
K/A RATINGS:	RO: 3.2 SRO: 3.3			
K/A No. &STATEMENT:	263000 D.C. Electrical Distribution A3.01; Ability to monitor automatic operations of D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights.			
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A			
SAFETY FUNCTION:	6			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 20 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

TVA

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: _306-U2_

RO _____ SRO _____ DATE: _____________

TASK EXPECTED ACTION(S): The Examinee is expected to place Divisional Emergency Core Cooling System (ECCS) Analog Trip Unit (ATU) Inverter in Service – Division I (DIV I)

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 0-OI-57C

VALIDATION TIME: _20 minutes_

PERFORMANCE TIME: ___________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
8	10/29/2020	All	Update JPM

Procedure Revisions

Procedure	Revision
0-OI-57C	134

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, protected equipped……
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2.

- Unit 2 is in cold Shutdown
- The Unit 2 Division I ECCS Analog Trip Unit Inverter was shutdown and taken out of service for preventive maintenance
- The maintenance has been completed and the clearance released and removed.

INITIATING CUES: The Shift Manager has directed you to return the Unit 2 ECCS Analog Trip Unit (ATU) Inverter - Division I to service in accordance with 0-OI-57C, 208V/120V AC Electrical System, Section 5.3.1.

NOTE:

All Precautions and Limitations in Section 3.0 have been reviewed.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: 0-OI-57C, 208V/120V AC Electrical System Section 5.3.1, Placing Unit 2 ECCS ATU Inverter Division I, 1-INVT-256-0001 in Service [1] **ENSURE** the 2-INVT-256-0001, ECCS ATU INVERTER DIV I is shut down. **REFER TO** Section 7.4.1. **CAUTION** Unit 2, 2-INVT-256-0001, ECCS ATU INVERTER DIV I requires a 60 second wait period prior to restart. Expected Action(s): N/A, given in the INITIAL CONDITIONS.	______ SAT ______UNSAT ______N/A
Step 2: [2] **REVIEW** all Precautions and Limitations in Section 3.0. Expected Action(s): N/A, given in the INITIAL CONDITIONS.	______ SAT ______UNSAT ______N/A
Step 3: [3] **ENSURE CLOSED** the following breaker on 250V Reactor MOV Board 2B Compartment 8A: • 2-INVT-256-0001, DIVISION I ECCS ANALOG TRIP UNIT INVERTERS Expected Action(s): Locates 250V RMOV Board 2B - Compartment 8A (DIV I) and SIMULATES checking CLOSED the breaker for 2-INVT-256-0001, ECCS ATU INVERTER DIV I	______ SAT ______UNSAT ______N/A
EXAMINER CUE: After the breaker is SIMULATED check CLOSED, state the breaker for 2-INVT-256-0001, ECCS ATU INVERTER DIV I, is CLOSED	

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td><u>Step 4:</u>

NOTE
Steps 5.3.1[4] through 5.3.1[10] are performed at the Division I ECCS ATU Inverter located in Electrical Board Room 2B.

[4] CHECK the 2-IL-256-0001/P1, DC AVAILABLE is illuminated.

<u>Expected Action(s):</u>

ENSURES 2-IL-256-0001/P1, DC AVAILABLE is illuminated.</td><td>______ SAT

______UNSAT

______N/A</td></tr>
<tr><td><u>Step 5:</u>

[5] DEPRESS and HOLD 2-HS-256-0001/S4, PRECHARGE.

<u>Expected Action(s):</u>

SIMULATES DEPRESSING and HOLDING 2-HS-256-0001/S4, PRECHARGE.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: After SIMULATING DEPRESSING and HOLDING 2-HS-256-0001/S4, PRECHARGE, state you have held the PRECHARGE pushbutton for 10 seconds.</td></tr>
<tr><td><u>Step 6:</u>

[6] WHEN the 2-IL-256-0001/P4, PRECHARGE illuminates, THEN PERFORM the following:

[6.1] RELEASE 2-HS-256-0001/S4, PRECHARGE.

[6.2] CLOSE ECCS ATU INVERTER DIV I DC INPUT BKR, 2-BKR-256-0001A.

<u>Expected Action(s):</u>

SIMULATES RELEASING 2-HS-256-0001/S4, PRECHARGE and SIMULATES CLOSES 2-BKR-256-0001A, ECCS ATU INVERTER DIV I DC INPUT BKR.</td><td>Critical Step

______ SAT

______UNSAT

______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: After SIMULATED, state 2-HS-256-0001/S4, PRECHARGE has been RELEASED and 2-BKR-256-0001A, ECCS ATU INVERTER DIV I DC INPUT BKR is CLOSED.</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 7: [7] CHECK AC Volts is between 117 and 123 volts on 2-EI-256-0001/V1, AC OUTPUT VOLTAGE(V1). Expected Action(s): CHECKS AC Volts between 117 and 123 volts on 2-EI-256-0001/V1, AC OUTPUT VOLTAGE(V1).	______ SAT ______UNSAT ______N/A
Step 8: [8] CLOSE 2-BKR-256-0001B, ECCS ATU INVERTER DIV I AC OUTPUT. Expected Action(s): SIMULATES CLOSING 2-BKR-256-0001B, ECCS ATU INVERTER DIV I AC OUTPUT.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: After SIMULATED, state 2-BKR-256-0001B, ECCS ATU INVERTER DIV I AC OUTPUT is CLOSED	
Step 9: [9] **DEPRESS** 2-HS-256-0001/S1, ALARM RESET. Expected Action(s): SIMULATES DEPRESSING 2-HS-256-0001/S1, ALARM RESET.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: After SIMULATED, state ALL alarms are clear.	

<table>
<tr><td>
<u>Step 10:</u>

[10] CHECK the following parameters on 2-INVT-256-0001, ECCS INVERTER DIV I:

A. 2-IL-256-0001/P2, DC VOLTAGE LOW is OFF.

B. 2-IL-256-0001/P3, AC OVERVOLTAGE is OFF.

C. AC current is less than 42 Amperes on 2-II-256-0001/A1, AMMETER.

D. AC voltage is between 117 and 123 volts on 2-EI-256-0001/V1, AC OUTPUT VOLTMETER(V1).

E. Frequency is between 59.7 and 60.3 Hz on 2-SI-256-0001/E1, FREQUENCY INDICATION.

<u>Expected Action(s):</u>

Locates the correct parameters and after CUE (below), accepts readings as normal.
</td><td>
_____ SAT

_____UNSAT

_____N/A
</td></tr>
<tr><td colspan="2">
EXAMINER CUE:
As each parameter is checked, as applicable, state:

• AC current is reading 5 amps
• AC voltage is reading 125 volts
• Frequency is reading 60.1 Hz
</td></tr>
<tr><td colspan="2">
EXAMINER CUE: After the examinee repeats the parameter readings, state This completes your task.
</td></tr>
</table>

STOP TIME: __________

Job Performance Measure (JPM)

Provide to Applicant

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or "That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2.

- Unit 2 is in cold Shutdown
- The Unit 2 Division I ECCS Analog Trip Unit Inverter was shutdown and taken out of service for preventive maintenance
- The maintenance has been completed and the clearance released and removed.

INITIATING CUES: The Shift Manager has directed you to return the Unit 2 ECCS Analog Trip Unit (ATU) Inverter - Division I to service in accordance with 0-OI-57C, 208V/120V AC Electrical System, Section 5.3.1.

NOTE:

All Precautions and Limitations in Section 3.0 have been reviewed.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Place Unit 3 Divisional ECCS Analog Trip Unit Inverter in Service in accordance with 0-OI-57C, 208V/120V AC Electrical System, Section 5.4.
JPM NUMBER:		306-U3	REVISION: 8

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-57C-NO-01			
K/A RATINGS:	RO: 3.2 SRO: 3.3			
K/A No. &STATEMENT:	263000 D.C. Electrical Distribution A3.01; Ability to monitor automatic operations of D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights.			
RELATED PRA INFORMATION:	Key System Contribution to CDF = N/A			
SAFETY FUNCTION:	6			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 20 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ______________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator *Date*

Approved by: ______________________ ______________
Site Training Management *Date*

Approved by: ______________________ ______________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: _306-U3_

RO _____ SRO _____ DATE: _____________

TASK EXPECTED ACTION(S): The Examinee is expected to place Divisional Emergency Core Cooling System (ECCS) Analog Trip Unit (ATU) Inverter in Service – Division I (DIV I)

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 0-OI-57C

VALIDATION TIME: _20 minutes_

PERFORMANCE TIME: ___________

COMMENTS: ____
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Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
8	10/29/2020	All	Update JPM

Procedure Revisions

Procedure	Revision
0-OI-57C	134

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, protected equipped……
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3.

- Unit 3 is in cold Shutdown
- The Unit 3 Division I ECCS Analog Trip Unit Inverter was shutdown and taken out of service for preventive maintenance
- The maintenance has been completed and the clearance released and removed

INITIATING CUES: The Shift Manager has directed you to return the Unit 3 ECCS Analog Trip Unit (ATU) Inverter - Division I to service in accordance with 0-OI-57C, 208V/120V AC Electrical System, Section 5.4.1.

NOTE:

All Precautions and Limitations in Section 3.0 have been reviewed.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: 0-OI-57C, 208V/120V AC Electrical System Section 5.2, Placing Unit 3 Division I ECCS Analog Trip Unit Inverter in Service [1] **ENSURE** the 3-INVT-256-0001, ECCS ATU INVERTER DIV I is shut down. **REFER TO** Section 7.5.1. **CAUTION** Unit 3, 3-INVT-256-0001, ECCS ATU INVERTER DIV I requires a 60 second wait period prior to restart. Expected Action(s): N/A, given in the INITIAL CONDITIONS.	______ SAT ______UNSAT ______N/A
Step 2: [2] **REVIEW** all Precautions and Limitations. REFER TO Section 3.0. Expected Action(s): N/A, given in the INITIAL CONDITIONS.	______ SAT ______UNSAT ______N/A
Step 3: [3] **CHECK CLOSED** the following breaker on 250V DC RMOV Board 3B Compartment 8A: 3-BKR-281-03B/008A, DIV I ECCS ATU INVERTER Expected Action(s): Locates 250V RMOV Board 3B - Compartment 8A (DIV I) and SIMULATES checking CLOSED 3-BKR-281-03B/008A, DIV I ECCS ATU INVERTER	______ SAT ______UNSAT ______N/A
EXAMINER CUE: After the breaker is SIMULATED check CLOSED, state 3-BKR-281-03B/008A, DIV I ECCS ATU INVERTER is CLOSED	

Step 4: NOTE Steps 5.4.1[4] through 5.4.1[10] are performed from 3-INVT-256-0001, ECCS ATU INVERTER DIV I, located in Electrical Board Room 3B EL 593'. [4] **ENSURE** the 3-IL-256-0001/P1, DC INPUT AVAILABLE is illuminated. Expected Action(s): ENSURES 3-IL-256-0001/P1, DC INPUT AVAILABLE is illuminated.	______ SAT ______UNSAT ______N/A
Step 5: [5] **DEPRESS** and **HOLD** 3-HS-256-0001/S4, PRECHARGE. Expected Action(s): SIMULATES DEPRESSING and HOLDING 3-HS-256-0001/S4, PRECHARGE.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: After SIMULATING DEPRESSING and HOLDING 3-HS-256-0001/S4, PRECHARGE, state you have held the PRECHARGE pushbutton for 10 seconds.	
Step 6: [6] **WHEN** 3-IL-256-0001/P4, PRECHARGE illuminates, **THEN PERFORM** the following: [6.1] **RELEASE** 3-HS-256-0001/S4, PRECHARGE. [6.2] **CLOSE** 3-BRK-256-0001/B1, DC INPUT. Expected Action(s): SIMULATES RELEASING 3-HS-256-0001/S4, PRECHARGE and SIMULATES CLOSES 3-BRK-256-0001/B1, DC INPUT.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: After SIMULATED, state 3-HS-256-0001/S4, PRECHARGE has been RELEASED and 3-BRK-256-0001/B1, DC INPUT is CLOSED.	

Step 7: [7] CHECK AC Volts is between 117 and 123 volts on 3-EI-256-0001/V1, AC VOLTMETER(V1). Expected Action(s): CHECKS AC Volts between 117 and 123 volts on 3-EI-256-0001/V1, AC VOLTMETER(V1).	______ SAT ______UNSAT ______N/A
Step 8: [8] CLOSE 3-BKR-256-0001/B2, AC SYSTEM OUTPUT. Expected Action(s): SIMULATES CLOSING 3-BKR-256-0001/B2, AC SYSTEM OUTPUT.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: After SIMULATED, state 3-BKR-256-0001/B2, AC SYSTEM OUTPUT is CLOSED	
Step 9: [9] **DEPRESS** 3-HS-256-0001/S1, ALARM RESET. Expected Action(s): SIMULATES DEPRESSING 3-HS-256-0001/S1, ALARM RESET.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: After SIMULATED, state ALL alarms are clear.	

Step 10: [10] **CHECK the following parameters on 3-INVT-256-0001, ECCS ATU INVERTER DIV I:** A. 3-IL-256-0001/P2, LOW DC VOLTAGE is OFF. B. 3-IL-256-0001/P3, AC OVERVOLTAGE is OFF. C. AC current is less than 42 Amperes on 3-II-256-0001/A1, AC AMMETER(A1). D. AC voltage is between 117 and 123 volts on 3-EI-256-0001/V1, AC VOLTMETER(V1). E. Frequency is between 59.7 and 60.3 Hz on 3-SI-256-0001/E1, FREQUENCY METER(E1). Expected Action(s): Locates the correct parameters and after CUE (below), accepts readings as normal.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: **As each parameter is checked, as applicable, state:** • **AC current is reading 5 amps** • **AC voltage is reading 125 volts** • **Frequency is reading 60.1 Hz**	
EXAMINER CUE: After the examinee repeats the parameter readings, state This completes your task.	

STOP TIME: ___________

TVA	**Job Performance Measure (JPM)**

Provide to Applicant

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3.

- Unit 3 is in cold Shutdown
- The Unit 3 Division I ECCS Analog Trip Unit Inverter was shutdown and taken out of service for preventive maintenance
- The maintenance has been completed and the clearance released and removed

INITIATING CUES: The Shift Manager has directed you to return the Unit 3 ECCS Analog Trip Unit (ATU) Inverter - Division I to service in accordance with 0-OI-57C, 208V/120V AC Electrical System, Section 5.4.1.

NOTE:

All Precautions and Limitations in Section 3.0 have been reviewed.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Determine Control Rod Withdrawal Requirements
JPM NUMBER:	516	REVISION:	2

TASK APPLICABILITY:	☐ SRO	☐ STA	☒ UO	☐ NAUO
TASK NUMBER / TASK TITLE(S):	N/A			
K/A RATINGS:	RO 4.3			
K/A STATEMENT:	2.1.37 Knowledge of procedures, guidelines, or limitations associated with reactivity management.			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	CONDUCT OF OPERATIONS - ADMIN			

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☒ Other - List	Classroom		

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 15 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ______________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator *Date*

Approved by: ______________________ ______________
Site Training Management *Date*

Approved by: ______________________ ______________
Site Training Program Owner *Date*

OPERATOR: ________________________________ JPM Number __516__.

RO ____ DATE: _______

TASK STANDARD: The Examinee is expected to determine Control Rod withdrawal requirements based on Source Range Monitor (SRMs) readings.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 2-GOI-100-1A, Unit Startup and Power Operation

VALIDATION TIME: 15 minutes

PERFORMANCE TIME: ________

COMMENTS: ____

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____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
2	12/2/20	All	Updated JPM

Procedure Revisions

Procedure	Revision
2-GOI-100-1A	181

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS: You are the Unit 2 Operator performing a Reactor Startup with the initial Source Range Monitor (SRM) counts as follows:

A – 19

B – 14

C – 19

D – 18

INITIATING CUES: The current indication for SRM counts are as follows:

A – 298

B – 235

C – 330

D – 278

Given the conditions above, the Nuclear Unit Senior Operator (NUSO) has directed you to determine how Control Rods will be withdrawn in accordance with 2-GOI-100-1A, Unit Startup and Power Operation, Step 5.4 - Withdrawal of Control Rods while in MODE 2.

Note: Show all work to support determination

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START TIME __________

STEP / STANDARD	SAT / UNSAT
Step 1: 2-GOI-100-1A, Unit Startup and Power Operation, Step 5.4 - Withdrawal of Control Rods while in MODE 2 **NOTE** Source Range Data should be taken just prior to withdrawing Control Rods for Startup. This will minimize a difference in Source Range counts caused by a change in plant conditions. [1] **PERFORM** the following for SRMs on Panel 2-9-5: [1.1] **RECORD** SOURCE RANGE MONITORS reading: CHANNEL A LEVEL 19 cps CHANNEL C LEVEL 19 cps CHANNEL B LEVEL 14 cps CHANNEL D LEVEL 18 cps (R) _______ _________ _________ Initials Date Time Expected Action(s): Step [1.1] given from the Initial Conditions and already completed in candidate's copy of 2-GOI-100-1A, Unit Startup and Power Operation, Step 5.4	___SAT ___UNSAT ___N/A
Examiner Note: Filling in Initials/Date/Time is NOT required in Steps 5.4 [1], [3], [4], [14]	
Step 2: [2] **RECORD** SOURCE RANGE MONITORS readings in Step 5.4[1.1] on PIP 95-119 on Panel 2-9-5: Expected Action(s): N/A, [2] given as completed on the candidate's handout for 2-GOI-100-1A, Step 5.4	___SAT ___UNSAT ___N/A

STEP / STANDARD	SAT / UNSAT
Step 3: **NOTE** A review of startup data has revealed that when count rate doubles five times, criticality is imminent. As an added precaution, the fourth count rate doubling has been chosen as a starting point to limit rod withdrawal to single notch movement. This requirement, along with close monitoring of neutron monitoring instrumentation, should assure a slow controlled approach to criticality. Criticality should be expected at all times. [3] **CALCULATE** SRM count rate at which notch withdrawal limitations is to be imposed by multiplying pre-startup count rate recorded in Step 5.4[1] by a factor of 16. Expected Action(s): Calculates initial SRM count rate recorded in Step 5.4[1] by a factor of 16 (2^4 or four doublings) and fills in the Initials/Date/Time CHANNEL A LEVEL - 19 cps X 16 = 304 cps CHANNEL B LEVEL - 14 cps X 16 = 224 cps CHANNEL C LEVEL - 19 cps X 16 = 304 cps CHANNEL D LEVEL - 18 cps X 16 = 288 cps (R) ________ ________ ________ Initials Date Time 1st (R) ________ ________ ________ Initials Date Time Reactor Engineer	**Critical Step** ___SAT ___UNSAT ___N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **RECORD** results below and at Step 5.4[14]: Expected Action(s): Records calculated SRM count rate results below from Step 5.4[3] and at Step 5.4[14] CHANNEL A LEVEL 304 cps CHANNEL B LEVEL 224 cps CHANNEL C LEVEL 304 cps CHANNEL D LEVEL 288 cps	___SAT ___UNSAT ___N/A

Step 5:

NOTE

Once required, Control Rod withdrawal is limited to single-notch withdrawal until Reactor power is in the heating range.

CAUTIONS

1) Near end of core life, criticality may occur before five doublings due to a stronger top peak flux and buildup of plutonium.

2) When rod movement is restricted to notch withdrawal, failure to stop at each notch position may result in high notch worth.

Step 5 (con't):

[14] **WHEN** SRMs indicate the calculated values recorded below,

CHANNEL A LEVEL ___ cps

CHANNEL B LEVEL ___ cps

CHANNEL C LEVEL ___ cps

CHANNEL D LEVEL ___ cps

THEN START single-notch withdrawal of Control Rods.

Expected Action(s):

Records calculated SRM count rate results below as directed in Step 5.4[4]

CHANNEL A LEVEL 304 cps

CHANNEL B LEVEL 224 cps

CHANNEL C LEVEL 304 cps

CHANNEL D LEVEL 288 cps

Since at least one of the four SRM CHANNEL LEVELs indicates the calculated value, candidate determines that single-notch withdrawal of Control Rods is required.

Critical Step

___SAT

___UNSAT

___N/A

Examiner Cue: Once Step 5.4[14] in complete and the candidate reports to the NUSO that single-notch withdrawal of Control Rods is required, acknowledge report.
END OF TASK

STOP TIME ________

Provide to Applicant

****

Classroom

****

INITIAL CONDITIONS: You are the Unit 2 Operator performing a Reactor Startup with the initial Source Range Monitor (SRM) counts as follows:

A – 19

B – 14

C – 19

D – 18

INITIATING CUES: The current indication for SRM counts are as follows:

A – 298

B – 235

C – 330

D – 278

Given the conditions above, the Nuclear Unit Senior Operator (NUSO) has directed you to determine how Control Rods will be withdrawn in accordance with 2-GOI-100-1A, Unit Startup and Power Operation, Step 5.4 - Withdrawal of Control Rods while in MODE 2.

Note: Show all work to support determination

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TVA | # Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Placing an RPS Channel in Trip
JPM NUMBER:	745	REVISION:	0

TASK APPLICABILITY:	☐SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-099-SU-02, Perform MSIV Closure – RPS Trip Functional Test			
K/A RATINGS:	RO 3.9			
K/A STATEMENT:	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.			
RELATED PRA INFORMATION:	Risk Significant RPS Scram Reduction			
SAFETY FUNCTION:	CONDUCT OF OPERATIONS - ADMIN			

EVALUATION LOCATION:	☐In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☒ Other - List	Classroom		

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 10 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________

Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)

(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________

Validator *Date*

Approved by: ______________________ ____________

Site Training Management *Date*

Approved by: ______________________ ____________

Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: __745__

RO ____ SRO ____ DATE: _______

TASK STANDARD: For the failed RPS instrument 2-PIS-3-22AA, Reactor High Pressure A1 Channel, the Examinee is expected to determine the procedure(s) and describe how to place the RPS instrument channel in trip

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 2-OI-99, Reactor Protection System
Print 2-730E915-9

VERIFICATION TIME: __10 min_

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	9/24/20	All	Initial issue

Procedure Revisions

Procedure	Revision
2-OI-99	93
Print 2-730E915-9	29

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS: 2-PIS-3-22AA, Reactor High Pressure A1 Channel was being tested by the Instrument Mechanics to support a surveillance. It failed to meet its required AC steps, resulting in it being declared INOPERABLE.

INITIATING CUES: As the Unit 2 Reactor Operator, the Nuclear Unit Senior Operator (NUSO) has directed you to perform the Tech Spec Required Action to place the 2-PIS-3-22AA, Reactor High Pressure A1 Channel in trip in accordance with plant procedures.

Determine **ALL** of the following:

- What plant procedure(s)/document(s) is(are) used to perform the Tech Spec Required Action?
- Identify how the Required Action is performed in accordance with the respective plant procedure(s)/document(s) from above?

Answer:

____

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START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: Refers to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9 (next page) for 2-PIS-3-22AA, Reactor High Pressure A1 Channel.	**Critical Step** ______ SAT ______UNSAT ______N/A

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0093 Page 100 of 106

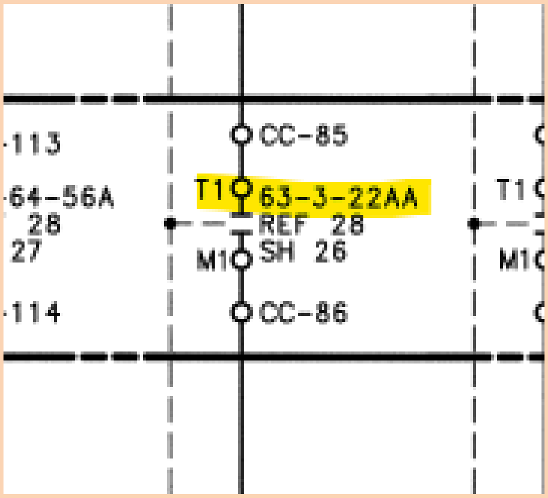
Attachment 3
(Page 5 of 11)
Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REM
2-PIS-3-22AA RX HIGH PRESS A1 CHANNEL Function: 3	2-FU1-3-22AA (5AF5A)	2-RLY-099-05AK05A	9-15	2-730E915-9 2-45E671-26	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCR
2-PIS-3-22BB RX HIGH PRESS B1 CHANNEL Function: 3	2-FU1-3-22BA (5AF5B)	2-RLY-099-05AK05B	9-17	2-730E915-10 2-45E671-38	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCR
2-PIS-3-22C RX HIGH PRESS A2 CHANNEL Function: 3	2-FU1-3-22CA (5AF5C)	2-RLY-099-05AK05C	9-15	2-730E915-9 2-45E671-32	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCR
2-PIS-3-22D RX HIGH PRESS B2 CHANNEL Function: 3	2-FU1-3-22DA (5AF5D)	2-RLY-099-05AK05D	9-17	2-730E915-10 2-45E671-44	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCR

NOTE:
Device Function corresponds to the TS Table 3.3.1.1 Functions.

Expected Action(s):

Examinee refers to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9 (next page) to reference the respective failed instrument 2-PIS-3-22AA, Reactor High Pressure A1 Channel.

STEP / STANDARD	SAT / UNSAT
<u>Step 1 (continued):</u> Print 2-730E915-9 (2-PIS-3-22AA is located between A-3 and E-3 coordinates)  	

Step 2: Determine how the Required Action is performed to place 2-PIS-3-22AA, Reactor High Pressure A1 Channel in trip in accordance with 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9. Expected Action(s): In accordance with 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9, determines fuse 2-FU1-3-22AA (5A-F5A) is required to be removed in order for 2-PIS-3-22AA, Reactor High Pressure A1 Channel to be placed in trip as directed.	**Critical Step** ______ SAT ______UNSAT ______N/A

Step 2 (continued):

2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11)

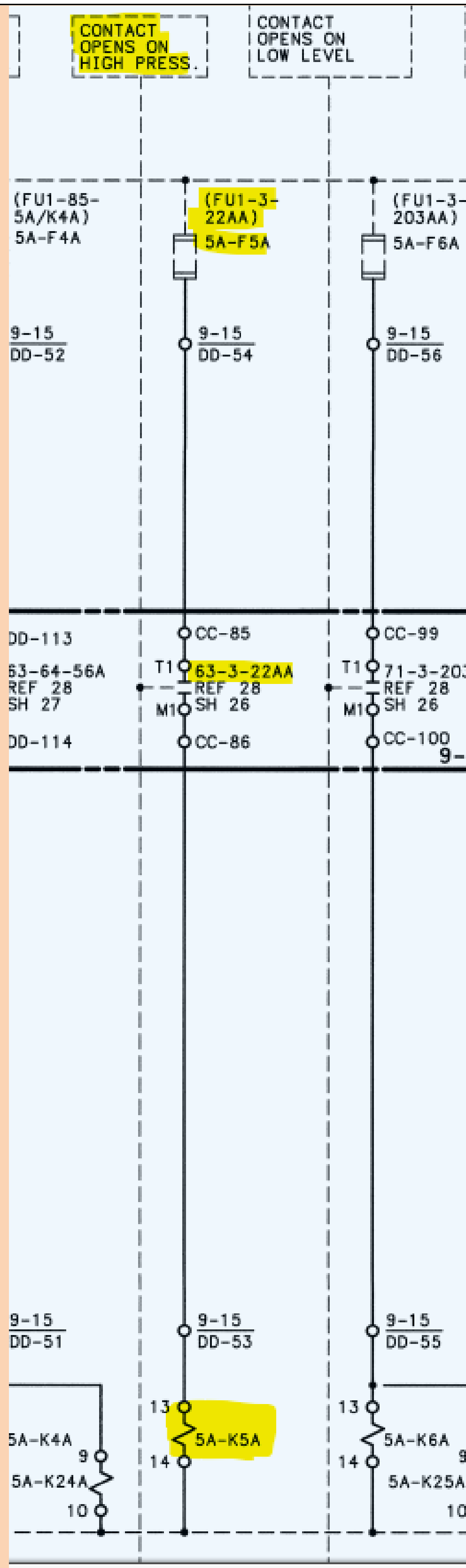
BFN Unit 2 | Reactor Protection System | 2-OI-99 Rev. 0093 Page 100 of 106

Attachment 3
(Page 5 of 11)

Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
2-PIS-3-22AA RX HIGH PRESS A1 CHANNEL Function: 3	2-FU1-3-22AA (5AF5A)	2-RLY-099-05AK05A	9-15	2-730E915-9 2-45E671-26	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN A CHANNEL
2-PIS-3-22BB RX HIGH PRESS B1 CHANNEL Function: 3	2-FU1-3-22BA (5AF5B)	2-RLY-099-05AK05B	9-17	2-730E915-10 2-45E671-38	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN B CHANNEL
2-PIS-3-22C RX HIGH PRESS A2 CHANNEL Function: 3	2-FU1-3-22CA (5AF5C)	2-RLY-099-05AK05C	9-15	2-730E915-9 2-45E671-32	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN A CHANNEL
2-PIS-3-22D RX HIGH PRESS B2 CHANNEL Function: 3	2-FU1-3-22DA (5AF5D)	2-RLY-099-05AK05D	9-17	2-730E915-10 2-45E671-44	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN B CHANNEL

NOTE:
Device Function corresponds to the TS Table 3.3.1.1 Functions.

Step 2 (continued): Print 2-730E915-9		**Critical Step** ______ SAT ______UNSAT ______N/A

EXAMINER CUE: Once the Operator identifies that 2-FU1-3-22AA has to be pulled in accordance with 2-OI-99, Reactor Protection System, Attachment 3 and/or Print 2-730E915-9 for the failed instrument (2-PIS-3-22AA, Reactor High Pressure A1 Channel) Inform the candidate "Another Operator will finish this procedure. This completes your task".
END OF TASK

STOP TIME:__________

Provide to Applicant

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS: 2-PIS-3-22AA, Reactor High Pressure A1 Channel was being tested by the Instrument Mechanics to support a surveillance. It failed to meet its required AC steps, resulting in it being declared INOPERABLE.

INITIATING CUES: As the Unit 2 Reactor Operator, the Nuclear Unit Senior Operator (NUSO) has directed you to perform the Tech Spec Required Action to place the 2-PIS-3-22AA, Reactor High Pressure A1 Channel in trip in accordance with plant procedures.

Determine **ALL** of the following:

- What plant procedure(s)/document(s) is(are) used to perform the Tech Spec Required Action?
- Identify how the Required Action is performed in accordance with the respective plant procedure(s)/document(s) from above?

Answer:

____
____
____
____
____
____
____
____
____
____

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Evaluate Recombiner Performance
JPM NUMBER:	510	REVISION:	4

TASK APPLICABILITY:	☐SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-066-NO-02 / Perform Recombiner Performance Evaluation			
K/A RATINGS:	RO 4.2			
K/A STATEMENT:	2.2.44: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.			
RELATED PRA INFORMATION:	None			
SAFETY FUNCTION:	EQUIPMENT CONTROL - ADMIN			

EVALUATION LOCATION:	☐In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☒ Other - List	Classroom		

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 10 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ____________________ __________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ____________________ __________
Validator *Date*

Approved by: ____________________ __________
Site Training Management *Date*

Approved by: ____________________ __________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: __510__

RO ____ SRO ____ DATE: _______

TASK STANDARD: The Examinee is expected to evaluate Off-Gas Recombiner Performance to determine if it meets Acceptance Criteria.

Operator Fundamental evaluated:
OF-1 Monitoring Plant Indications and Conditions Closely

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 3-OI-66

VERIFICATION TIME: __10 min_

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
2	08/16/17	ALL	Converted JPM to new format
3	11/30/20	ALL	Updated JPM

Procedure Revisions

Procedure	Revision
3-OI-66	80

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

You are a Unit 3 Operator with the following plant conditions:

- Reactor Power is 91%, nearing the end of a Reactor Startup following an outage
- Hydrogen Water Chemistry System is NOT in-service following being shut down in accordance with 3-OI-4, Hydrogen Water Chemistry System
- Off-Gas Preheater, Recombiner, and SJAEs are in operation in accordance with 3-OI-66, Off-Gas System, Section 5.0.
- The operating steam jet is operating properly

INITIATING CUE:

The Shift Manager has directed you to perform 3-OI-66, Off-Gas System, Section 6.1[1], Recombiner Performance Evaluation, and identify if any actions are required. Conditions are as follows:

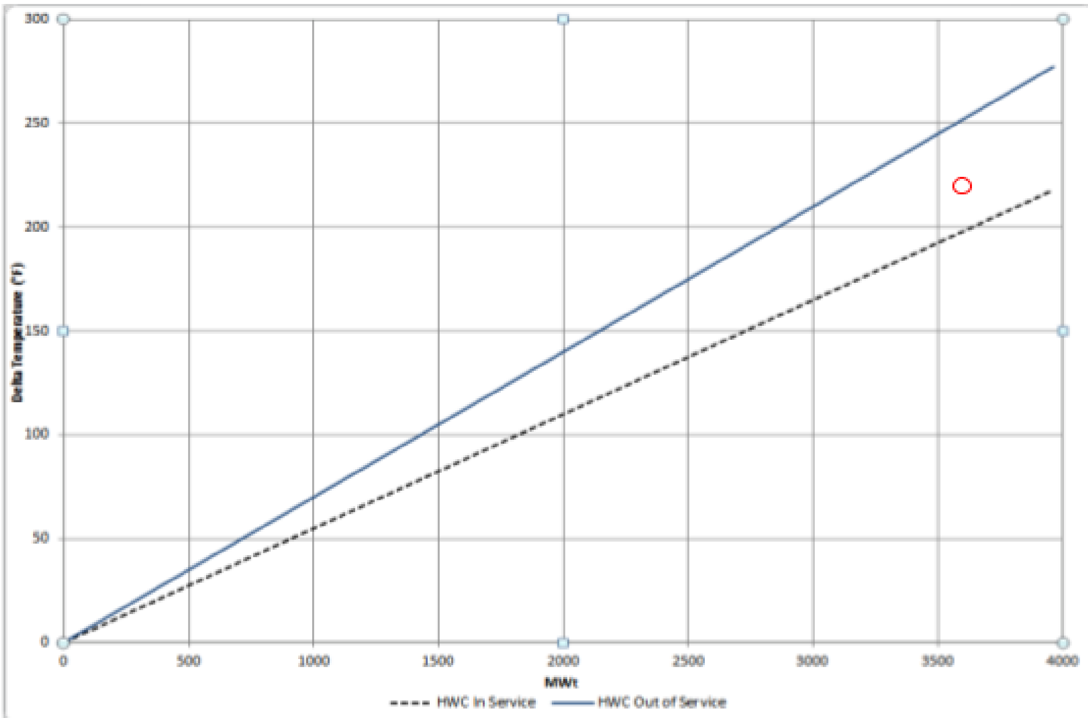
RECOMBINER 3A, INLET TEMP, 3-TI-66-75A	392 ºF
RECOMBINER 3B, INLET TEMP, 3-TI-66-75B	320 ºF
GLY/RECMB/OG MOIST SEP TEMPERATURE, 3-TRS-66-106	
RECOMBINER 3A CENTER, 3-TE-66-77AB	612 ºF
RECOMBINER 3B CENTER, 3-TE-66-77BB	380 ºF
Core Thermal Power (MWt)	3600 MWt
Percent Power (% RTP)	91%
ANALYZER 3A, 3-H2A-66-96A	OPERABLE - reading 0.26% H_2
ANALYZER 3B, 3-H2A-66-96B	OPERABLE - reading 0.26% H_2

KEY

BFN Unit 3	Off-Gas System	3-OI-66 Rev. 0080 Page 145 of 155

Attachment 1
(Page 1 of 1)

Recombine Performance Evaluation - ΔT to Reactor Power



Evaluation is satisfactory when intersection point of ΔT to Reactor Power is above the appropriate line.

<u>For 3952mwt</u>

HWC in service	ΔT ≥ 217°F
HWC out of service	ΔT ≥ 277°F

<u>CURVE FACTORS</u>

Normal Water Chemistry (NWC)	ΔT = 0.070°F per MWt
Hydrogen Water Chemistry (HWC)	ΔT = 0.055°F per MWt

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: **NOTES** 1) The production of hydrogen and oxygen in the Reactor is dependent upon Reactor Power level and upon the amount of hydrogen injected by the Hydrogen Water Chemistry System if in service. Since the recombination of hydrogen and oxygen is exothermic, the operating temperature of the recombiner is also dependent upon power level and the status of the HWC System. 2) Following startup, while still at low power, recombiner performance and hydrogen concentration should be closely monitored. [1] **PERFORM** a recombiner performance evaluation as follows: [1.1] **DETERMINE** in-service recombiner inlet temperature as indicated on applicable temperature indicator, Panel 3-9-53. • RECOMBINER 3A, INLET TEMP 3-TI-66-75A • RECOMBINER 3B, INLET TEMP 3-TI-66-75B Expected Action(s): Determines the in-service recombiner operating (inlet) temperature as indicated on RECOMBINER 3A, INLET TEMP, 3-TI-66-75A as 392 ºF on Panel 3-9-53 (from handout).	____ SAT _____UNSAT _____N/A
Step 2: [1.2] **DETERMINE** in-service recombiner operating (center) temperature as indicated on GLY/RECMB/OG MOIST SEP TEMPERATURE recorder, 3-TRS-66-106, Panel 3-9-53. Expected Action(s): Determines the in-service recombiner operating (center) temperature as indicated on RECOMBINER 3A CENTER, 3-TE-66-77AB as 612 ºF, on 3-TRS-66-77, Panel 3-9-53 (from handout).	____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [1.3] **CALCULATE** the temperature difference (ΔT) between the values obtained in Steps 6.1[1.1] and 6.1[1.2]. Expected Action(s): Calculates Recombiner 3A inlet/center ΔT (612 ºF - 392 ºF) and determines ΔT is 220 ºF.	**Critical Step** ____ SAT _____UNSAT _____N/A
Step 4: [1.4] **DETERMINE** the Reactor Thermal Power (MWt) from process computer. Expected Action(s): Determines Reactor Thermal Power is 3600 MWt from the handout.	____ SAT _____UNSAT _____N/A
Step 5: [1.5] **USING** Attachment 1, **PLOT** the corresponding point of Reactor Power in MWt and ΔT. Expected Action(s): Using Attachment 1, plots corresponding point of Reactor Power (3600 MWt) and ΔT (220 ºF). The candidate also may determine that the required minimum ΔT corresponding to 3600 MWt is 252 ºF. Calculation: ΔT = 0.070 ºF per MWt 0.070 X 3600 = 252 ºF	**Critical Step** ____ SAT _____UNSAT _____N/A
Examiner Note: Either method (calculation or plotting) is acceptable	

STEP / STANDARD	SAT / UNSAT
Step 6: [1.6] **ENSURE** point on Attachment 1 is above or equal to the appropriate line (HWC In Service or HWC Out of Service). Expected Action(s): Determines from Attachment 1 that calculated ∆T vs MWt plots **BELOW** the HWC Out of Service line. Candidate may also use calculated ∆T from curve factor to determine that actual ∆T (220 ºF) is well below the HWC Out of Service line on graph.	**Critical Step** ____ SAT _____UNSAT _____N/A
Step 7: [2] **IF** in-service recombiner performance is below the minimum allowable, **THEN:** Expected Action(s): Following the candidate notifying their SRO that the performance is UNSAT, the JPM task is complete. The candidate is not expected to proceed with [2].	____ SAT _____UNSAT _____N/A
END OF TASK	

STOP TIME:__________

Provide to Applicant

****
CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.
****

You are a Unit 3 Operator with the following plant conditions:

- Reactor Power is 91%, nearing the end of a Reactor Startup following an outage
- Hydrogen Water Chemistry System is NOT in-service following being shut down in accordance with 3-OI-4, Hydrogen Water Chemistry System
- Off-Gas Preheater, Recombiner, and SJAEs are in operation in accordance with 3-OI-66, Off-Gas System, Section 5.0.
- The operating steam jet is operating properly

INITIATING CUE:

The Shift Manager has directed you to perform 3-OI-66, Off-Gas System, Section 6.1[1], Recombiner Performance Evaluation, and identify if any actions are required. Conditions are as follows:

RECOMBINER 3A, INLET TEMP, 3-TI-66-75A	392 ºF
RECOMBINER 3B, INLET TEMP, 3-TI-66-75B	320 ºF
GLY/RECMB/OG MOIST SEP TEMPERATURE, 3-TRS-66-106	
RECOMBINER 3A CENTER, 3-TE-66-77AB	612 ºF
RECOMBINER 3B CENTER, 3-TE-66-77BB	380 ºF
Core Thermal Power (MWt)	3600 MWt
Percent Power (% RTP)	91%
ANALYZER 3A, 3-H2A-66-96A	OPERABLE - reading 0.26% H_2
ANALYZER 3B, 3-H2A-66-96B	OPERABLE - reading 0.26% H_2

Job Performance Measure (JPM)

SITE: BFN		JPM TITLE:	Review a Radiological Work Permit (RWP)
JPM NUMBER:	682	REVISION:	2

TASK APPLICABILITY:	☐ SRO	☐ STA	☒ UO	☐ NAUO
TASK NUMBER / TASK TITLE(S):	A-000-AD-35 / Use a Radiation Work Permit			
K/A RATINGS:	K/A RATING: RO 3.5			
K/A STATEMENT:	2.3.7 Ability to comply with Radiation Work Permit requirements during normal or abnormal conditions.			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	RADIATION CONTROL - ADMIN			

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☒ Other - List	Classroom		

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 10 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ____________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ____________
Validator *Date*

Approved by: ______________________ ____________
Site Training Management *Date*

Approved by: ______________________ ____________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 682

RO ____ SRO ____ DATE: ________

TASK STANDARD: The Examinee is expected to review an RWP to determine if a task can be completed without exceeding dose exposure limits.

PRA: NA

REFERENCES/PROCEDURES NEEDED: NPG-SPP-05.18

VALIDATION TIME: 10 minutes

PERFORMANCE TIME: ______

COMMENTS: ____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: ______________________________ DATE: ___________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
1	11/19/2020	ALL	JPM update
2	02/25/2021	ALL	RWP format revision

Procedure Revisions

Procedure	Revision
NPG-SPP-05.18	9

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

You are a Unit 3 AUO assigned to a task that will require you to manually close 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION and place a mechanical restraining device on the valve given the following:

- <u>10 minutes</u> to close the valve
- <u>15 minutes</u> to install the mechanical restraining device

The dose rate at 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION VALVE, is 300 mrem/hr.

Note: Assume **NO** dose for transit time.

Use the attached Radiological Work Permit **(**RWP) to accomplish your task

INITIATING CUE:

Given the conditions above, determine if this task **<u>CAN/CANNOT</u>** be performed in accordance with the attached Radiological Work Permit **(**RWP).

Note: Show all work to support your answer.

START TIME: __________

STEP / STANDARD	SAT / UNSAT
Step 1: Calculates expected dose to close 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION, and install a mechanical restraining device on the valve. Expected Action(s): 10 min to close valve + 15 min to install device = 25 min 25/60 = 0.417 hrs 0.417 hrs x 300 mRem/hr = **125 mrem** (close valve, install device) (Between 120.0 to 127.0 mrem is acceptable)	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: Determines if task **CAN/CANNOT** be accomplished in accordance with the attached RWP. Expected Action(s): The given RWP limit per entry is **100 mrem** (RWP pg. 2, step 3). Since **125 mrem** is greater than 100 mrem**,** determines that the task **CANNOT** be accomplished in accordance with the given initial conditions and attached RWP.	**Critical Step** _____ SAT _____UNSAT _____N/A

STOP TIME: __________

Provide to Applicant

****
CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.
****

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The dose rate at 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION VALVE, is 300 mrem/hr.

Note: Assume **NO** dose for transit time.

Use the attached Radiological Work Permit **(**RWP) to accomplish your task

INITIATING CUE:

Given the conditions above, determine if this task **CAN/CANNOT** be performed in accordance with the attached Radiological Work Permit **(**RWP).

Note: Show all work to support your answer.

Provide to Applicant

Radiological Work Permit

Num.21330002 **Rev.** 1 **Status** ACTIVE

BFN

General RWP Information

Description

Unit 3 Maintenance on RWCU (69) Systems, Work Area Description: Unit 3 Areas All Elevations

[RWP LIMITS: 100 mrem Dose Alarm and 500 mrem/hr Dose Rate Alarm]

Start Date: 01-JAN-This year	**Dose Alarm:** 100 mrem	**ALARA Plan:**
End Date: 01-Jan-Next year	**Dose Rate Alarm:** 500 mrem/hr	**RWP Type:** GENERAL
RP Coverage: INTERMITTENT	**Briefing Type:** INDIVIDUAL	

Stop Work Criteria

- STOP WORK in the event of Airborne Radioactivity > 10 DAC.
- HOLD POINT: If Airborne Radioactivity exceeds 0.3 DAC,stop work until a TEDE-ALARA evaluation can be performed IAW NPG-SPP-05.2.5 to evaluate respiratory protection and engineering control requirements.
- STOP WORK in the event Dose Rates at 30 cm are > 500 mrem/hr.
- STOP WORK in the event of a dose alarm or unanticipated dose rate alarm and notify RP immediately.
- STOP WORK in the event Alpha Contamination is greater than anticipated.
- STOP WORK in the event Beta-Gamma Contamination is greater than anticipated.

Expected Radiological Conditions

- **GA Dose Rates:** < 1 mrem/hr to 500 mrem/hr
- **Contact Dose Rates:** < 1 mrem/hr to 1,400 mrem/hr
- **Contamination Levels:** < 1,000 dpm/100cm2 to 10 mrad/hr/100cm2
- **Airborne Levels:** up to 10 DAC or up to 40 DAC-hrs in a single entry

Respiratory Instructions

The use of respiratory equipment is **CONDITIONAL** based on TEDE-ALARA evaluation results. The following respirators are allowed on this RWP:

Protective Clothing Requirements

Respiratory Instructions

The use of respiratory equipment is **CONDITIONAL** based on TEDE-ALARA evaluation results. The following respirators are allowed on this RWP:

- ULTRATWIN
- PAPR

Protective Clothing Requirements

- SURGEON'S CAP
- SHOE COVERS, ONE PAIR
- MODESTY CLOTHING
- GLOVES, RUBBER, ONE PAIR
- COVERALLS, ONE PAIR
- CLOTH INSERTS
- BOOTIES, ONE PAIR

*21110551

Provide to Applicant

Radiological Work Permit

Num.21330002 **Rev.** 1 **Status** ACTIVE

BFN

Dosimetry Instructions

Required Dosimetry

- TELEMETRY [WRM-2 OR SIMILAR DEVICE
- SELF READING DOSIMETER
- DOSIMETER OF LEGAL RECORD

Dosimetry Comments

- If dosimetry is lost or dropped, where it cannot be retrieved without leaving the immediate vicinity, the following steps shall be followed:
  A. Worker SHALL exit the IMMEDIATELY and report to RP
  B. A Qualified RP Technician shall attempt the recovery of the dosimetry
  C. If an Industrial Safety Hazard exists, a Qualified RP Technician may secure the job site while constantly monitoring the Radiological Conditions in the Area

Special Instructions

- A copy of the RWP is available for review on HIS-20 upon logging in.
- Special Dress Out requirements are permitted for laborers while performing trash and laundry activities.
- Use of respiratory protection equipment is conditional based on TEDE ALARA results.
- During radiological briefings, include discussions of specific anticipated dose rates and actions to be taken in the event of a dose rate alarm.
- RP is not to brief workers to Anticipated Alarms equal to or greater than 1,000 mrem/hr (Whole Body). No more than 3 Dose Rate Alarms are permitted per entry.
- Expected or anticipated SRD dose rate alarms should be planned, documented in eSOMS (include anticipated alarm, location, applicable WO#, and approving RP Supervisor name), and discussed with workers prior to entry into the area.
- Dose rate set points should not exceed the station's threshold for posting and controlling High Radiation Areas.
- Adjustments to SRD Set Points may be performed with approval of RP Supervision. The adjusted SRD Set Points SHALL be documented in eSOMS to include new Set Points, applicable WO#, and approving RP Supervisor name.
- Entry into a Locked High Radiation Area, Very High Radiation Area (LHRA, VHRA) is PROHIBITED on this RWP.

*21110551

Provide to Applicant

Radiological Work Permit

BFN

Num.21110551 **Rev.** 1 **Status** ACTIVE

General Work Instructions

- Radiological Protection (RP) Briefings shall be utilized prior to start of work and prior to moving into a new area to define scope of work and review the area radiological conditions.
- When Logging in to HIS-20 Perform Self-Checking to ensure the proper Work Order/Step and RWP Number is utilized.
- Monitor your SRD frequently. If dose exceeds 80% of SRD setpoint, then place systems and equipment in safe configuration and exit the area.
- During periods when HIS-20 is in the "local" mode, the default set points for the RWP are 50 mrem DOSE ALARM and 80 mrem/hr DOSE RATE ALARM.
- Dressout instructions: single dressout clothing requirements used for this RWP, unless otherwise directed by RP.
- Dressout requirements may be modified based on safety (e.g., rotating equipment, heat stress) and/or radiological conditions with RP Supervisor approval.
- A hood shall be required in a Contaminated Area if the worker is required to wear a Body Harness or utilize a phone.
- Dosimeter to be placed in the chest pocket of the PCs unless otherwise directed by RP.
- Notify RP of any activity requiring climbing 7 feet above the floor level, including temporary or permanent ladders and devices.
- Notify RP prior to any system breach, welding, grinding, or surface disturbing activities. RP shall be present for initial breaches of contaminated systems to ensure proper radiological controls are in place.
- Notify RP if Hoses or Cords Need to Cross the CA Boundary. Hoses and/or Cords must be secured to prevent Contamination outside the CA Boundaries.
- Based on radiological conditions and work activities, ensure the appropriate HEPA Unit/Vacuum is selected, issued and utilized per RP Instruction.
- Avoid Posted Hot Spots and/or Piping with Lead Shielding. Locate and utilize Low Dose Waiting Areas (LDWA). Practice ALARA.
- Upon exiting a CA, proceed to the NEAREST frisker; complete a hand & foot frisk . [If you receive an alarm - NOTIFY RP IMMEDIATELY].
- In the event a frisker is not available when exiting a CA, proceed directly to the NEAREST PCM.
- Upon performing a successful frisk and completion of task, proceed to the nearest PCM and perform a Whole Body Survey. [If the PCM alarms, survey again. If the PCM alarms a second time - NOTIFY RP IMMEDIATELY].
- Upon exiting the RCA, worker shall log out of the RWP in HIS-20.

Prepared by: FLATKINS **RPSS Approval:** JAELIAS

RPM Approval: JKSMITH **Final Approval:** JNSTYLES

*21110551

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Determine Crew Shift Staffing Requirements
JPM NUMBER:	678	REVISION:	3

TASK APPLICABILITY:	☒SRO	☐STA	☐UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	N/A			
K/A RATINGS:	SRO 3.9			
K/A STATEMENT:	2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.			
RELATED PRA INFORMATION:	None			
SAFETY FUNCTION:	Admin - Conduct of Operations			

EVALUATION LOCATION:	☐In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☒ Other - List	Classroom		

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 15 mins TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ____________________ __________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ____________________ __________
Validator *Date*

Approved by: ____________________ __________
Site Training Management *Date*

Approved by: ____________________ __________
Site Training Program Owner *Date*

OPERATOR: ______________________________ JPM Number: _678_

SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to review crew shift staffing and determine if all staffing requirements are met and if not, what positions must be filled and by what means.

Operator Fundamental evaluated:
OF-3 Operating the Plant with a Conservative Bias

PRA: N/A

REFERENCES/PROCEDURES NEEDED: OPDP-1, NPG-SPP-03.21, OSIL-25, Shift Manager's Staffing Sheet (attached)

VALIDATION TIME: 15 minutes

PERFORMANCE TIME: ___________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Rev No.	Effective Date	Pages Affected	Description
0	10/03/2018	ALL	New JPM
1	08/15/2019	ALL	Updated JPM
2	10/9/2019	ALL	Removed reference to Tech Specs.
3	09/17/2020	ALL	Updated JPM

Procedure Revisions

Procedure	Revision
OPDP-1	50
NPG-SPP-03.21	25
OSIL-25	12/18/17
Shift Manager's Staffing Sheet	DAYS

****
CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.
****

INITIAL CONDITIONS:

You are the Shift Manager (SM) on night shift, and you are assigned to verify that minimum shift staffing requirements are met for the upcoming **DAY** shift crew.

INITIATING CUES:

Review the attached Staffing Sheet and determine if any action is required in accordance with OPDP-1, Conduct of Operations or other staffing procedures as applicable.

Note**: NO** waivers will be used**.**

KEY Shift Manager Staffing

12/2/2020	Pager	Phone	DAYS		NIGHTS	
GROUP			3		5	
SM	17-073	7860/2173	Clark		Heugel	
SM-B						
STA		2168	Turner(*)		McCurdy	
Support						
LEAVE						
U1 NUSO		2175	Turner(*)		Cartwright	
U2 NUSO		2269	Shirley		McCurdy	
U3 NUSO		2373	Bennett		Rogenski	
OS NUSO			Grimme	IC	Welchans	IC
WCC			missing		Alsup	
Support						
LEAVE						
LEAVE			Spears		Fisher	
U1 BOARD RO		2192	Millsaps		Blakely	
U1 DESK RO		2191	Sockwell		Hargett	
U2 BOARD RO		2292	Wright		Holden	OT
U2 DESK RO		2291	missing		Miller	
U3 BOARD RO		2392	Cole		Metcalf	
U3 DESK RO		2391	missing		Grissom	
SST		7687				
Support						
Support					McCoig	E
Leave			Wheeler (SL)	SL	Sager	AL
Leave			Young	AL	McAbee (FSL)	FSL
RW UO	60-793	2372	KANEY (s)		Barnes	
Moving Resin/ULTREX						
ULTREX AUO		2404	REED		Jochum	ER2
U1 TB AUO	15-026	777-2821	JESS	ER1	Rogers	ER1
U1 RB AUO	13-604	777-2186	RICHARDSON	OT/ER2		
RW DEMINS	16-745		missing	ER3		
U2 TB AUO	14-932	777-1015	MCBAY	ER4	Young	ER4
U2 RB AUO	90-536	777-0673	MCCALPIN	ER5	Cleveland	ER5
OUTSIDE AUO	60-280	777-2873	SMITH	ER6	Donaldson	ER6
U3 TB AUO	30-618	777-2623	WOODFIN	ER7	Patrick	ER7
U3 RB AUO	96-024		BERRYMAN	ER8		
CONTROL BAY	13-146	777-2351	BREWER	ER9	Tomlinson	ER9
INTAKE AUO/Alt Leak	16-544	614-8530	KING (s)	OT	Wilhelm	
WCC						
WCC						
Break in/extra					McDow	ER3
Break in/extra					Smith	ER8
Break in/extra						
Break in/extra						
Fire Brigade						
Cooling Tower		729-3201				
Cooling Tower		729-3201				
LEAVE						
LEAVE						
LEAVE						AL
LEAVE			GRAHAM	SL		AL
LEAVE						

LI - LLRT Lvl I; LII - LLRT Lvl II; S-*OT scheduled OT		**% - no license duties(nld)**
Maintenance MGR	**729-7677 / 434-0824 / 16-057**	**(I) Incident Commander**
Work Week Manager	**729-7447**	**(s) Not emergency Responder Qual'd**
OPS Clerks	**729-2302/2190**	**(2) Not Fire Watch Qualified**
Cooling Towers	**729-3201 / 434-0830/ 729-7616**	**(d) Check Break in needed**
Chemistry	**729-2368 / 2913 15-912 / 20-564/19-164**	**(#) not Clearance writer qual'd (QE only)**
ER1-ER9 Assume Emergency Responder Positions		**(*) STA Qualified**
(TRN) Training (J) JITT (NLD) No License Duties (CAL) Cancel A/L		**& - No clearance quals**

After Shift Manager initials, forward a copy to the Operations clerks for retention

START TIME __________

STEP / STANDARD	SAT / UNSAT
Step 1: OPDP-1, Conduct of Operations.	

Attachment 1
(Page 2 of 2)

Shift Staffing

1.0 SHIFT STAFFING (continued)

Minimum Staffing	**BFN**
Shift Manager (SRO)	1
Nuclear Unit Senior Operator (SRO)	4
Unit Operator (UO)	6
Non Licensed (AUO)	9
STA**	1
Incident Commander*	1

*The Incident Commander will be a shift SRO not assigned to a unit or the STA role (PER 217578).

**The STA may fill the NUSO position provided that an additional SRO (not assigned to a unit or as IC) is available and can relieve the STA filling the NUSO position within 10 minutes. The individual relieving the STA must have knowledge of plant conditions in order to perform a turnover without delay. The STA function is still required upon entry into the Fire Safe Shutdown procedures (FSSs).

Expected Action(s):

Reviews OPDP-1, Conduct of Operations Attachment 1 for BFN Staffing requirements

______ SAT

______UNSAT

______N/A

EXAMINER NOTE: The Examinee may initially identify all of the missing operators in any order and/or state a call-in is required to meet minimum staffing in accordance with OPDP-1 by performing the following as applicable for the missing DAY shift operators:

1. Hold operators over from NIGHT shift for no more than 4 hours until Call-ins can be fulfilled for the minimum missing positions in accordance with NPG-SPP-03.21, Nuclear Fatigue Management Program, Section 3.2.7, 2.a.

NPG Standard Programs and Processes	Nuclear Fatigue Management Program	NPG-SPP-03.21 Rev. 0025 Page 29 of 82

3.2.7 Calculating Work Hours (continued)

2. Application

a. By example, if an individual who normally works a 12-hour shift schedule is requested to work additional hours from 0700 to 1900 on Friday, the following should be considered.

(1) Determine if more than 16 hours in a 24-hour period will be exceeded by reviewing hours worked during the 24-hour period prior to the stop time on Friday as reflected in the request to work additional hours.

2. Hold operators over OR arrange for replacement personnel to restore the shift compliment within 2 hours in accordance with OPDP-1, Conduct of Operations Section 2.0.B. Attachment 1.

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0050 Page 52 of 71

Attachment 1
(Page 2 of 2)

Shift Staffing

2.0 NOTIFICATION OF ABSENCES

A. Operations personnel unable to report for shift duty shall, before the scheduled time, inform the SM/NUSO of the situation. The SM or designee shall make necessary arrangements for obtaining a replacement.

B. In the case of illness or unexpected absence of the operations shift compliment personnel, the Shift Manager should hold a shift member over or arrange for replacement personnel to restore the shift complement within two hours.

3. Conduct Call-ins to meet the minimum staffing in accordance with OSIL-25, TVA BFN Operations Section Instruction Letter Overtime, Leave, and Relief Policy, Attachment 2

NOTE: The Examinee is NOT required to fill out the Call-in Request Form

TENNESSEE VALLEY AUTHORITY BROWNS FERRY NUCLEAR PLANT OPERATIONS SECTION INSTRUCTION LETTER OVERTIME, LEAVE, AND RELIEF POLICY	OSIL-25 PAGE 1 OF 3 12/18/17 Attachment 2

Instructions for filling out the Call-in Request Sheet

1) The Unit Operator and/or the Operations Clerk will assign the number of positions required to be filled for the shift in question. This will encompass the required positions and number required in each position including extra personnel required to support shift activities.

2) Shift Manager signs (signature) the call-in request sheet prior to initiating the call-in signifying he concurs with the positions and the number of persons required to fill the shift compliment. This can include any additional personnel required to support extra shift tasks. If the Ops Clerks are performing the OT call-in, the SM approval can be performed by telecom.

3) Columns will be filled out in "YES/NO" format using the following criteria;

- WORK, "Do you want to work the required shift?" This is to determine whether the individual wants to work the entire shift.
- WAIVER, "Will you require a waiver to work the entire shift?"
- FIT FOR DUTY, "Are you fit for duty?" (See Fitness For Duty Below)
- ALCOHOL, "Have you consumed alcohol in the past 5 hours?" (See Fitness For Duty Below)
- INITIALS, The Unit Operator or Ops Clerk (caller) initials in the row for the individual which has been called. The person entering the work hours into NFR and the Person performing the NFR entry IV will both initial the row for the individual that is coming in to work. They will also print their name at the bottom of the Call-in Request Form.

4) The SM and the Call Performer will print their name at the bottom of the Call-in Request Form. If the Ops Clerks are performing the OT call-in, the Clerk can print the SM's name on the form.

5) The completed Call-in Request Form shall be forwarded to the Ops NFR Administrator. The Ops NFR Administrator will file the Call-in Request Form in a fire-proof cabinet for the required retention period.

TENNESSEE VALLEY AUTHORITY BROWNS FERRY NUCLEAR PLANT OPERATIONS SECTION INSTRUCTION LETTER OVERTIME, LEAVE, AND RELIEF POLICY	OSIL-25 PAGE 3 OF 3 12/18/17 Attachment 2

Call-in Request Form

Shift/Group: Date: SM (Signature):

Number of Positions

US: UO: AUO: STA: SSS: 1st Responders:

List T&L for call-in by OT hours (list those requiring a waiver last)

Name	Phone #	Work? (Yes/No)	Waiver? (Yes/No)	Fit For Duty? (Yes/No)	Time Called	Time Needed to Report	Alcohol < 5 hrs? (Yes/No)	Call Performer (Initials)	NFR Entry 1st / IV (Initials/ Initials)
Min Shift Staffing position required or other need		Group # with opening		Reason for Min Staffing not met (SL, FSL, etc).					

Name	Phone #	Work? (Yes/No)	Waiver? (Yes/No)	Fit For Duty? (Yes/No)	Time Called	Time Needed to Report	Alcohol < 5 hrs? (Yes/No)	Call Performer (Initials)	NFR Entry 1st / IV (Initials/ Initials)
Min Shift Staffing position required or other need		Group # with opening		Reason for Min Staffing not met (SL, FSL, etc).					
				/					

Name	Phone #	Work? (Yes/No)	Waiver? (Yes/No)	Fit For Duty? (Yes/No)	Time Called	Time Needed to Report	Alcohol < 5 hrs? (Yes/No)	Call Performer (Initials)	NFR Entry 1st / IV (Initials/ Initials)
									/
Min Shift Staffing position required or other need		Group # with opening		Reason for Min Staffing not met (SL, FSL, etc).					

Call Performer (Print): SM Review (Print):

NFR Entry 1st (Print): NFR Entry IV (Print)

Retention Period: One (1) Year Page of Responsibility: Ops NFR Administrator

STEP / STANDARD	SAT / UNSAT
Step 2: Examinee reviews the NUSO and Work Control Center (WCC) positions on DAYS to determine if minimum staffing is met in accordance with OPDP-1, Attachment 1. Expected Action(s): Examinee notes that 4 NUSOs positions are filled (U1, U2, U3 and the Outside - OS) as required. However, Examinee notes that the following is required in accordance with OPDP-1, Attachment 1 which would be satisfied by the missing WCC position: • **The STA may fill the NUSO position provided that an additional SRO (not assigned to a unit or as IC) is available and can relieve the STA filling the NUSO position within 10 minutes. Given the above, in order to fill the Licensed NUSO-WCC position on DAYS, the Examinee may perform any of the following: • Hold a Licensed NUSO over from NIGHTS for up to 4 hours or • Hold a Licensed NUSO over or arrange for replacement personnel to restore the shift compliment within 2 hours or • Conduct Call-in for a Licensed NUSO	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 3: Examinee reviews the Unit Operator/Reactor Operator (RO) positions on DAYS to determine if 6 RO required minimum staffing is met in accordance with OPDP-1, Attachment 1. Expected Action(s): Given the above, in order to fill the U2 and U3 DESK RO missing positions on DAYS, the Examinee may perform any of the following: • Hold 2 Licensed ROs over from NIGHTS for up to 4 hours or • Hold 2 Licensed ROs over or arrange for replacement personnel to restore the shift compliment within 2 hours or • Conduct Call-ins for 2 Licensed ROs	**Critical Step** ______ SAT ______UNSAT ______N/A
STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: (For Step 3) RO Call-in: It is an acceptable practice of Operations to call to fill the SST slot. This is not required in accordance with OPDP-1, but calling enough to fill vacant positions and the SST position is acceptable.	

Step 4: Examinee reviews the 9 Non Licensed (AUO)/Emergency Responders (ER1-9) positions on DAYS to determine if minimum staffing is met in accordance with OPDP-1, Attachment 1. Expected Action(s): Examinee notes that the Emergency Responder **(ER-3)** position is not filled as assigned for the RW DEMINS AUO position. Given the above, in order to fill the missing **(ER-3)** position on DAYS, the Examinee will perform any of the following: • Assign Reed (ULTREX AUO) or • Hold AUO over from NIGHTS for up to 4 hours or • Hold AUO over or arrange for replacement personnel to restore the shift compliment within 2 hours or • Conduct Call-in for AUO	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER NOTE: The missing AUO ER-3 position cannot be filled using current on shift AUOs, since they are shown with an (s) beside their names, indicating they are not ER qualified.	

STOP TIME __________

Provide to Applicant

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

You are the Shift Manager (SM) on night shift, and you are assigned to verify that minimum shift staffing requirements are met for the upcoming **DAY** shift crew.

INITIATING CUES:

Review the attached Staffing Sheet and determine if any action is required in accordance with OPDP-1, Conduct of Operations or other staffing procedures as applicable.

Note**:** **NO** waivers will be used**.**

Provide to Applicant **Shift Manager Staffing**

12/2/2020	Pager	Phone	DAYS		NIGHTS	
GROUP			3		5	
SM	17-073	7860/2173	Clark		Heugel	
SM-B						
STA		2168	Turner(*)		McCurdy	
Support						
LEAVE						
U1 NUSO		2175	Turner(*)		Cartwright	
U2 NUSO		2269	Shirley		McCurdy	
U3 NUSO		2373	Bennett		Rogenski	
OS NUSO			Grimme	IC	Welchans	IC
WCC					Alsup	
Support						
LEAVE						
LEAVE			Spears		Fisher	
U1 BOARD RO		2192	Millsaps		Blakely	
U1 DESK RO		2191	Sockwell		Hargett	
U2 BOARD RO		2292	Wright		Holden	OT
U2 DESK RO		2291			Miller	
U3 BOARD RO		2392	Cole		Metcalf	
U3 DESK RO		2391			Grissom	
SST		7687				
Support						
Support					McCoig	E
Leave			Wheeler (SL)	SL	Sager	AL
Leave			Young	AL	McAbee (FSL)	FSL
RW UO	60-793	2372	KANEY (s)		Barnes	
Moving Resin/ULTREX						
ULTREX AUO		2404	REED		Jochum	ER2
U1 TB AUO	15-026	777-2821	JESS	ER1	Rogers	ER1
U1 RB AUO	13-604	777-2186	RICHARDSON	OT/ER2		
RW DEMINS	16-745			ER3		
U2 TB AUO	14-932	777-1015	MCBAY	ER4	Young	ER4
U2 RB AUO	90-536	777-0673	MCCALPIN	ER5	Cleveland	ER5
OUTSIDE AUO	60-280	777-2873	SMITH	ER6	Donaldson	ER6
U3 TB AUO	30-618	777-2623	WOODFIN	ER7	Patrick	ER7
U3 RB AUO	96-024		BERRYMAN	ER8		
CONTROL BAY	13-146	777-2351	BREWER	ER9	Tomlinson	ER9
INTAKE AUO/Alt Leak	16-544	614-8530	KING (s)	OT	Wilhelm	
WCC						
WCC						
Break in/extra					McDow	ER3
Break in/extra					Smith	ER8
Break in/extra						
Break in/extra						
Fire Brigade						
Cooling Tower		729-3201				
Cooling Tower		729-3201				
LEAVE						
LEAVE						
LEAVE						AL
LEAVE			GRAHAM	SL		AL
LEAVE						

LI - LLRT Lvl I; LII - LLRT Lvl II; S-*OT scheduled OT		% - no license duties(nld)
Maintenance MGR	729-7677 / 434-0824 / 16-057	(I) Incident Commander
Work Week Manager	729-7447	(s) Not emergency Responder Qual'd
OPS Clerks	729-2302/2190	(2) Not Fire Watch Qualified
Cooling Towers	729-3201 / 434-0830/ 729-7616	(d) Check Break in needed
Chemistry	729-2368 / 2913 15-912 / 20-564/19-164	(#) not Clearance writer qual'd (QE only)
ER1-ER9 Assume Emergency Responder Positions		(*) STA Qualified
(TRN) Training (J) JITT (NLD) No License Duties (CAL) Cancel A/L		& - No clearance quals

After Shift Manager initials, forward a copy to the Operations clerks for retention

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Place an RPS Channel in trip and determine REQUIRED ACTIONS in accordance with Technical Specifications
JPM NUMBER:	745-SRO	REVISION:	0

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-099-SU-02, Perform MSIV Closure – RPS Trip Functional Test			
K/A RATINGS:	SRO: 4.2			
K/A STATEMENT:	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.			
RELATED PRA INFORMATION:	Risk Significant RPS Scram Reduction			
SAFETY FUNCTION:	Admin - Conduct of Operations			

EVALUATION LOCATION:	☐In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☒ Other - List	Classroom		

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 15 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ______________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator *Date*

Approved by: ______________________ ______________
Site Training Management *Date*

Approved by: ______________________ ______________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: __745-SRO__

RO ____ SRO ____ DATE: _______

TASK STANDARD: For the failed RPS instrument 2-PIS-3-22AA, Reactor Vessel Steam Dome Pressure – High, the Examinee is expected to determine:

- The correct Technical Specification REQUIRED ACTION
- The procedure(s) and describe how to place the RPS instrument channel in trip

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 2-OI-99, Reactor Protection System
Unit 2 Tech Spec 3.3.1.1, RPS Instrumentation
Print 2-730E915-9

VERIFICATION TIME: __15 min__

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	9/24/20	All	Initial issue

Procedure Revisions

Procedure	Revision
2-OI-99	93
Unit 2 TS 3.3.1.1	Amend. 258
Print 2-730E915-9	29

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS: 2-PIS-3-22AA, Reactor High Pressure A1 Channel was being tested by the Instrument Mechanics to support a surveillance. It failed to meet its required Acceptance Criteria (AC) steps, resulting in it being declared INOPERABLE.

INITIATING CUES: As the Nuclear Unit Senior Operator (NUSO), you are required to determine **ALL** of the following:

- What is the Tech Spec and the Required Action?
- What plant procedure(s)/document(s) is(are) used to perform the Tech Spec Required Action?
- Identify how the Required Action is performed in accordance with the respective plant procedure(s)/document(s) from above?

Answer:

____

____

____

____

____

____

____

____

____

____

START TIME:__________

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>Step 1</u>:

Refers to Unit 2 Technical Specification 3.3.1.1, RPS Instrumentation

RPS Instrumentation
3.3.1.1

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

--------------------NOTE--------------------
Separate Condition entry is allowed for each channel.

<table>
<tr><th>CONDITION</th><th>REQUIRED ACTION</th><th>COMPLETION TIME</th></tr>
<tr><td>A. One or more required channels inoperable.</td><td>A.1 Place channel in trip.

<u>OR</u>

A.2 ------NOTE------
Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f.

Place associated trip system in trip.</td><td>12 hours

12 hours</td></tr>
</table>
(continued)

BFN-UNIT 2 3.3-1 Amendment No. 258
March 05, 1999
</td><td>
Critical Step

_____ SAT

_____UNSAT

_____N/A
</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT

Step 1 (cont):

RPS Instrumentation
3.3.1.1

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA(e)
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

Expected Action(s):

Determines Tech Spec 3.3.1.1 CONDITION A is NOT met and the REQUIRED ACTION is to place the A1 Channel in trip OR Place the associated trip system in TRIP in 12 hours.

EXAMINER NOTE: The Examinee may elect to first refer to Unit 2 Tech Spec 3.3.1.1, RPS Instrumentation and/or refer to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 13).

2-OI-99, Reactor Protection System, Attachment 3 list the respective failed instrument's (2-PIS-3-22AA, Reactor High Pressure A1 Channel) fuse, relay, prints and remarks/results.

STEP / STANDARD	SAT / UNSAT

Step 2:

Refers to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9 (next page) for 2-PIS-3-22AA, Reactor High Pressure A1 Channel.

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0093 Page 100 of 106

Attachment 3
(Page 5 of 11)

Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REM
2-PIS-3-22AA RX HIGH PRESS A1 CHANNEL Function: 3	2-FU1-3-22AA (5AF5A)	2-RLY-099-05AK05A	9-15	2-730E915-9 2-45E671-26	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRA
2-PIS-3-22BB RX HIGH PRESS B1 CHANNEL Function: 3	2-FU1-3-22BA (5AF5B)	2-RLY-099-05AK05B	9-17	2-730E915-10 2-45E671-38	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRA
2-PIS-3-22C RX HIGH PRESS A2 CHANNEL Function: 3	2-FU1-3-22CA (5AF5C)	2-RLY-099-05AK05C	9-15	2-730E915-9 2-45E671-32	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRA
2-PIS-3-22D RX HIGH PRESS B2 CHANNEL Function: 3	2-FU1-3-22DA (5AF5D)	2-RLY-099-05AK05D	9-17	2-730E915-10 2-45E671-44	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRA

NOTE:
Device Function corresponds to the TS Table 3.3.1.1 Functions.

Expected Action(s):

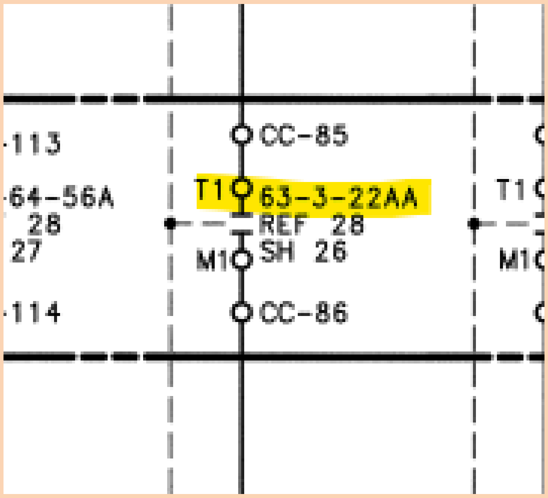
Examinee refers to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9 (next page) to reference the respective failed instrument 2-PIS-3-22AA, Reactor High Pressure A1 Channel.

Critical Step

______ SAT

______UNSAT

______N/A

STEP / STANDARD	SAT / UNSAT
<u>Step 2 continued:</u> Print 2-730E915-9 (2-PIS-3-22AA is located between A-3 and E-3 coordinates) 	

Step 3: Determine how the Required Action is performed to place 2-PIS-3-22AA, Reactor High Pressure A1 Channel in trip in accordance with 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9. Expected Action(s): Examinee determines that in accordance with 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9, fuse 2-FU1-3-22AA (5A-F5A) is required to be removed in order for 2-PIS-3-22AA, Reactor High Pressure A1 Channel to be placed in trip as directed.	**Critical Step** ______ SAT ______UNSAT ______N/A

Step 3 continued:

2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11)

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0093 Page 100 of 106

Attachment 3
(Page 5 of 11)

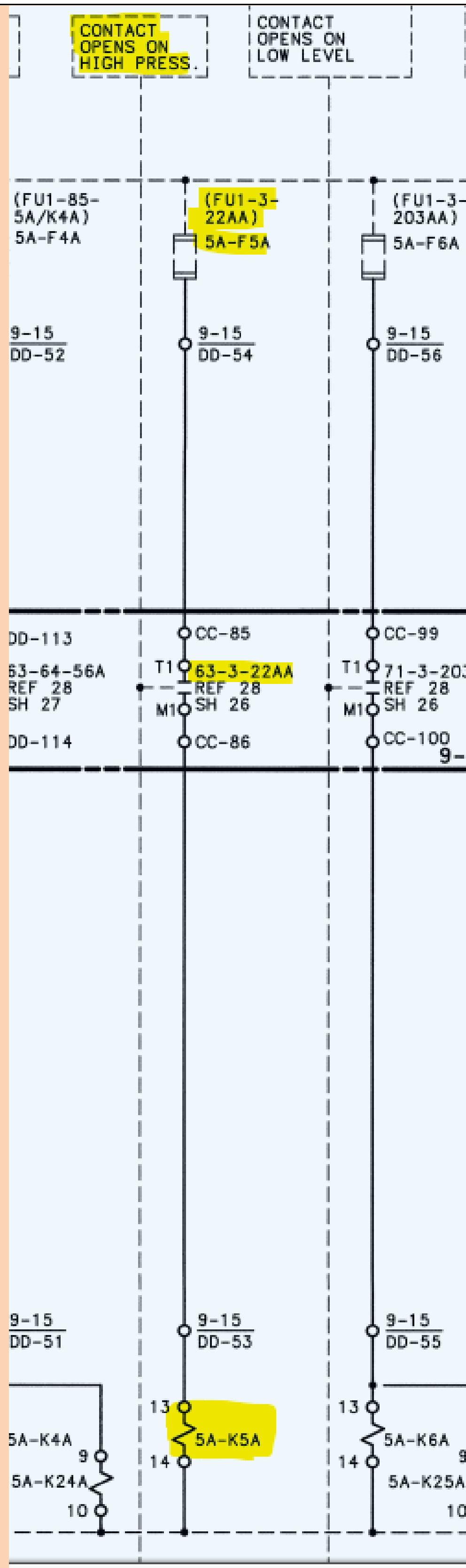
Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
2-PIS-3-22AA RX HIGH PRESS A1 CHANNEL Function: 3	2-FU1-3-22AA (5AF5A)	2-RLY-099-05AK05A	9-15	2-730E915-9 2-45E671-26	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN A CHANNEL
2-PIS-3-22BB RX HIGH PRESS B1 CHANNEL Function: 3	2-FU1-3-22BA (5AF5B)	2-RLY-099-05AK05B	9-17	2-730E915-10 2-45E671-38	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN B CHANNEL
2-PIS-3-22C RX HIGH PRESS A2 CHANNEL Function: 3	2-FU1-3-22CA (5AF5C)	2-RLY-099-05AK05C	9-15	2-730E915-9 2-45E671-32	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN A CHANNEL
2-PIS-3-22D RX HIGH PRESS B2 CHANNEL Function: 3	2-FU1-3-22DA (5AF5D)	2-RLY-099-05AK05D	9-17	2-730E915-10 2-45E671-44	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN B CHANNEL

NOTE:
Device Function corresponds to the TS Table 3.3.1.1 Functions.

Step 3 continued:

Print 2-730E915-9



EXAMINER CUE: Once the examinee identifies that: 1. Tech Spec 3.3.1.1 CONDITION A is entered with a COMPLETION TIME of 12 hours to place 2-PIS-3-22AA, Reactor High Pressure A1 Channel in trip 2. 2-OI-99, Reactor Protection System, Attachment 3 and/or by Print 2-730E915-9 must be referenced 3. 2-FU1-3-22AA has to be pulled in accordance with 2-OI-99, Reactor Protection System, Attachment 3 and/or by Print 2-730E915-9 for the failed instrument (2-PIS-3-22AA, Reactor High Pressure A1 Channel) Inform the candidate "Another Operator will finish this procedure. This completes your task".
END OF TASK

STOP TIME:__________

Provide to Applicant

****
CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.
****

INITIAL CONDITIONS: 2-PIS-3-22AA, Reactor High Pressure A1 Channel was being tested by the Instrument Mechanics to support a surveillance. It failed to meet its required Acceptance Criteria (AC) steps, resulting in it being declared INOPERABLE.

INITIATING CUES: As the Nuclear Unit Senior Operator (NUSO), you are required to determine **ALL** of the following:

- What is the Tech Spec and the Required Action?
- What plant procedure(s)/document(s) is(are) used to perform the Tech Spec Required Action?
- Identify how the Required Action is performed in accordance with the respective plant procedure(s)/document(s) from above?

Answer:

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Review a completed Surveillance
JPM NUMBER:		746-SRO	REVISION: 4

TASK APPLICABILITY:	☒SRO	☐STA	☐UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	S-000-AD-27, Assess LCO/TRM/ODCM Actions required for INOPERABLE equipment			
K/A RATINGS:	SRO 4.7			
K/A STATEMENT:	2.2.22 Knowledge of limiting conditions for operations and safety limits			
RELATED PRA INFORMATION:	None			
SAFETY FUNCTION:	Equipment Control - Admin			

EVALUATION LOCATION:	☐In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☒ Other - List	Classroom		

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 15 mins TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ______________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator *Date*

Approved by: ______________________ ______________
Site Training Management *Date*

Approved by: ______________________ ______________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 746-SRO

SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to determine if Acceptance Criteria (AC) has NOT been met during the performance of a Surveillance (SR). If AC is NOT met, the Examinee is expected to determine any required applicable Technical Specification Actions.

Operator Fundamental evaluated:
OF-1 Monitoring Plant Indications and Conditions Closely

PRA: N/A

REFERENCES/PROCEDURES NEEDED: (1) Completed 3-SR-3.8.7.1, Weekly Check of Power Availability to Required AC and DC Power Distribution Subsystems, but **NOT** identified by the Unit Operator.
(2) Partially completed TVA 40753, STS
(3) Unit 3 Tech Spec 3.8.7

VALIDATION TIME: 15 minutes

PERFORMANCE TIME: ________

COMMENTS: ____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: ____________________________ DATE: __________
EXAMINER

Rev No.	Effective Date	Pages Affected	Description
1	09/13/2018	ALL	Updated JPM
2	08/13/2019	ALL	Updated JPM
3	10/9/2019	ALL	Updated JPM
4	9/23/2020	ALL	Updated JPM

Procedure Revisions

Procedure	Revision
3-SR-3.8.7.1	15
TVA 40753, STS	3-SR-3.8.7.1
Unit 3 Tech Spec 3.8.7	212

****
CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.
****

INITIAL CONDITIONS:

Unit 3 is operating at 100% RTP. You are the Unit 3 Nuclear Unit Senior Operator (NUSO). The Balance of Plant Operator (BOP) has just completed 3-SR-3.8.7.1, Weekly Check of Power Availability to Required AC and DC Power Distribution Subsystems, and has given it to you review.

INITIATING CUES:

Conduct a review of 3-SR-3.8.7.1, Weekly Check of Power Availability to Required AC and DC Power Distribution Subsystems.

Determine OPERABILITY in accordance with Unit 3 Technical Specification 3.8.7, Distribution Systems - Operating.

START TIME ___________

STEP / STANDARD	SAT / UNSAT
Step 1: The Unit 3 Nuclear Unit Senior Operator (NUSO) ensures that the Balance of Plant Operator (BOP) has checked and initialed each step. Expected Action(s): NUSO notes that all initials are present.	______ SAT ______UNSAT ______N/A
EXAMINER NOTE: For JPM Steps 2-3 below, see next page for 3-SR-3.8.7.1 (page 13)	
Step 2: NUSO checks that the BOP has identified any anomalies. Expected Action(s): NUSO notes that the BOP recorded 432 Volts in 7.3[1.3.1].	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT

Step 3:

NUSO checks that the BOP has identified any anomalies.

Expected Action(s):

NUSO notes that Step 7.3[1.3.3] CHECK Voltage ≥ 440 volts, is NOT filled out correctly since Step 2 above recorded 7.3[1.3.1] as 432 volts.
Step 7.3[1.3.3] is an ACCEPTANCE CRITERIA (AC) step that was incorrectly signed off suggesting that the board voltage met the SR requirement.

BFN Unit 3	Weekly Check of Power Availability to Required AC and DC Power Distribution Subsystems	3-SR-3.8.7.1 Rev. 0015 Page 13 of 24

Date today

7.3 480 V Board Voltages (continued)

[1.3] 480V SD BD 3B VOLTAGE

[1.3.1] **RECORD** the Voltage below:
(N/A if unavailable)

432 VOLTS TJ

[1.3.2] **IF** Voltage is ≥ 500 VOLTS or Voltage Indication is unavailable, **THEN**

PERFORM the following: (Otherwise N/A)

A. **REQUEST** EM to obtain Voltages locally. N/A

B. **RECORD** the Highest Voltage obtained between AΦ to BΦ, BΦ to CΦ, and CΦ to AΦ voltages :

__________ VOLTS N/A

[1.3.3] **CHECK** Voltage ≥ 440 VOLTS TJ (AC)

[1.3.4] **CHECK** Voltage ≤ 508 VOLTS TJ

Critical Step

_____ SAT

_____UNSAT

_____N/A

EXAMINER CUE: For JPM Step 4 below, the SRO examinee may ask if the voltages have been verified by electrical maintenance as noted in P&L's of the SR.
If so, inform examinee that all voltages have been verified as indicated.

EXAMINER CUE:

The examinee should request Tech Spec 3.8.7, Distribution Systems – Operating, to determine 480 volt Board OPERABILITY. It is NOT required to know the Tech Spec section by number or name from memory.

Once requested, provide the examinee with Unit 2 Tech Spec 3.8.7, Distribution Systems – Operating.

Step 4:

NUSO determines that 480V Shutdown Board (SD BD) 3B is INOPERABLE in accordance with Tech Spec 3.8.7.

Expected Action(s):

The NUSO determines that all AC steps are NOT met therefore fails SR. The NUSO will enter Tech Spec 3.8.7 CONDITION B with REQUIRED ACTION B1 to Restore the Board to OPERABLE status in 8 hours.

Distribution Systems - Operating
3.8.7

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Unit 3 480 V Shutdown Board inoperable. OR 480 V RMOV Board 3A inoperable. OR 480 V RMOV Board 3B inoperable.	-------------------NOTE--------------- Enter Condition C when Condition B results in no power source to 480 volt RMOV board 3D or 3E. -- B.1 Restore Board to OPERABLE status.	8 hours AND 12 days from discovery of failure to meet LCO

Critical Step

______ SAT

______UNSAT

______N/A

STOP TIME __________

Provide to Applicant

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

Unit 3 is operating at 100% RTP. You are the Unit 3 Nuclear Unit Senior Operator (NUSO). The Balance of Plant Operator (BOP) has just completed 3-SR-3.8.7.1, Weekly Check of Power Availability to Required AC and DC Power Distribution Subsystems, and has given it to you review.

INITIATING CUES:

Conduct a review of 3-SR-3.8.7.1, Weekly Check of Power Availability to Required AC and DC Power Distribution Subsystems.

Determine OPERABILITY in accordance with Unit 3 Technical Specification 3.8.7, Distribution Systems - Operating.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Review a Radiological Work Permit
JPM NUMBER:	682-SRO	REVISION:	2

TASK APPLICABILITY:	☒ SRO	☐ STA	☐ UO	☐ NAUO
TASK NUMBER / TASK TITLE(S):	A-000-AD-35 / Use a Radiation Work Permit			
K/A RATINGS:	K/A RATING: SRO 3.6			
K/A STATEMENT:	2.3.7 Ability to comply with Radiation Work Permit requirements during normal or abnormal conditions.			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	RADIATION CONTROL - ADMIN			

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☒ Other - List	Classroom		

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 10 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Developed by: ______________________ ______________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator *Date*

Approved by: ______________________ ______________
Site Training Management *Date*

Approved by: ______________________ ______________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: 682-SRO

RO ____ SRO ____ DATE: ________

TASK STANDARD: The Examinee is expected to review an RWP to determine if a task can be completed without exceeding dose exposure limits.

PRA: NA

REFERENCES/PROCEDURES NEEDED: NPG-SPP-05.1, NPG-SPP-05.18

VALIDATION TIME: 10 minutes

PERFORMANCE TIME: ______

COMMENTS: ____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
1	11/19/2020	ALL	JPM update
2	01/25/2021	ALL	RWP format revision

Procedure Revisions

Procedure	Revision
NPG-SPP-05.1	12
NPG-SPP-05.18	9

****
CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.
****

INITIAL CONDITIONS:

You are the Unit 3 Nuclear Unit Senior Operator (NUSO) conducting a task pre-job brief with an AUO who currently has a cumulative yearly dose of 1890 mrem.

The task requires the AUO to manually close 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION and place a mechanical restraining device on the valve given the following:

- <u>10 minutes</u> to close the valve
- <u>15 minutes</u> to install the mechanical restraining device

The dose rate at 3-FCV-69-2 is 300 mrem/hr.

Note: Assume **NO** dose for transit time.

INITIATING CUE:

Given the conditions above:

1. Determine if the AUO task **<u>CAN/CANNOT</u>** be performed in accordance with the attached Radiological Work Permit **(**RWP)?

2. Determine if the AUO task **<u>CAN/CANNOT</u>** be performed **WITHOUT** requiring additional authorization in accordance with NPG-SPP-05.1, Radiological Controls?

Note: Show all work to support both answers.

____
____
____
____

START TIME: __________

STEP / STANDARD	SAT / UNSAT
<u>Step 1</u>: Calculates expected dose to close 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION, and install a mechanical restraining device on the valve. <u>Expected Action(s):</u> 10 min to close valve + 15 min to install device = 25 min 25/60 = 0.417 hrs 0.417 hrs x 300 mRem/hr = **125 mrem** (close valve, install device) (Between 120.0 to 127.0 mrem is acceptable)	**Critical Step** _____ SAT _____UNSAT _____N/A
<u>Step 2</u>: Determines if task <u>**CAN/CANNOT**</u> be accomplished in accordance with the attached RWP. <u>Expected Action(s):</u> The given RWP limit per entry is **100 mrem** (RWP pg. 2, step 3). Since **125 mrem** is greater than 100 mrem**,** determines that the task **CANNOT** be accomplished in accordance with the given initial conditions and attached RWP.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT

Step 3:

Determines if task **CAN/CANNOT** be performed **WITHOUT** requiring additional authorization in accordance with NPG-SPP-05.1, Radiological Controls, Table 1 - TVA Annual Administrative Dose Level Program.

Expected Action(s): Examinee determines that:

125 mrem (close valve and install device)
+
1890 mrem (cumulative yearly dose)

= 2015 mrem (total dose after task)
(2010 - 2017.0 mrem acceptable)

2015 mrem (total dose after task) is **more** than 2 TEDE (2 Rem / 2000 mrem) which **DOES** require additional authorization; therefore task **CANNOT** be performed without the additional authorization.

NPG Standard Programs and Processes	Radiological Controls	NPG-SPP-05.1 Rev. 0013 Page 15 of 54

3.2.4 Exposure Control (continued)

TABLE 1

ADMINISTRATIVE DOSE LEVEL PROGRAM

Dose Equivalent (Rem)	Requirement	Authorization to Exceed (signatures)
Up to **0.5 TEDE** (or 1.5 LDE, 5.0 SDE and 5.0 SDE,ME)	Statement of current year dose and previous years dose signed by individual	Not applicable
Up to **2.0 TEDE** (or 12 LDE, 40 SDE and 40 SDE ME) all sources	NRC FORM-4 or equivalent to document current year and previous years dose equivalent	Not applicable
To exceed **2.0 TEDE** all sources	Same as above	RPM/RSO
To exceed **3.0 TEDE** all sources	Same as above	RPM/RSO, and Plant Manager[1]
To exceed **4.0 TEDE** (or 12 LDE, 40 SDE and 40 SDE ME) all sources	Same as above	RPM/RSO, Plant Manager[1], and Site VP[2]
To exceed **5.0 TEDE[3] all sources**	Form-4 information must be verified and a Planned Special Exposure initiated in Accordance with RCTP-114	RPM/RSO, Plant Manager[1], and Site VP[2]

1 At non-nuclear plant sites, this will be the RSO's immediate supervisor.

2 At non-nuclear plant sites, this will be the applicable TVA VP.

3 Authorizations for a planned special exposure will only be considered in an exceptional situation when alternatives that might avoid the dose estimated to result from the planned special exposure are unavailable or impractical.

Critical Step

____ SAT

_____UNSAT

_____N/A

STOP TIME: __________

Provide to Applicant

****
CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.
****

INITIAL CONDITIONS:

You are the Unit 3 Nuclear Unit Senior Operator (NUSO) conducting a task pre-job brief with an AUO who currently has a cumulative yearly dose of 1890 mrem.

The task requires the AUO to manually close 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION and place a mechanical restraining device on the valve given the following:

- 10 minutes to close the valve
- 15 minutes to install the mechanical restraining device

The dose rate at 3-FCV-69-2 is 300 mrem/hr.

Note: Assume **NO** dose for transit time.

INITIATING CUE:

Given the conditions above:

3. Determine if the AUO task **CAN/CANNOT** be performed in accordance with the attached Radiological Work Permit **(**RWP)?

4. Determine if the AUO task **CAN/CANNOT** be performed **WITHOUT** requiring additional authorization in accordance with NPG-SPP-05.1, Radiological Controls?

Note: Show all work to support both answers.

Provide to Applicant

Radiological Work Permit

Num.21330002 **Rev.** 1 **Status** ACTIVE

BFN

General RWP Information

Description

Unit 3 Maintenance on RWCU (69) Systems, Work Area Description: Unit 3 Areas All Elevations

[RWP LIMITS: 100 mrem Dose Alarm and 500 mrem/hr Dose Rate Alarm]

Start Date: 01-JAN-This year	**Dose Alarm:** 100 mrem	**ALARA Plan:**
End Date: 01-Jan-Next year	**Dose Rate Alarm:** 500 mrem/hr	**RWP Type:** GENERAL
RP Coverage: INTERMITTENT	**Briefing Type:** INDIVIDUAL	

Stop Work Criteria

- STOP WORK in the event of Airborne Radioactivity > 10 DAC.
- HOLD POINT: If Airborne Radioactivity exceeds 0.3 DAC,stop work until a TEDE-ALARA evaluation can be performed IAW NPG-SPP-05.2.5 to evaluate respiratory protection and engineering control requirements.
- STOP WORK in the event Dose Rates at 30 cm are > 500 mrem/hr.
- STOP WORK in the event of a dose alarm or unanticipated dose rate alarm and notify RP immediately.
- STOP WORK in the event Alpha Contamination is greater than anticipated.
- STOP WORK in the event Beta-Gamma Contamination is greater than anticipated.

Expected Radiological Conditions

- **GA Dose Rates:** < 1 mrem/hr to 500 mrem/hr
- **Contact Dose Rates:** < 1 mrem/hr to 1,400 mrem/hr
- **Contamination Levels:** < 1,000 dpm/100cm2 to 10 mrad/hr/100cm2
- **Airborne Levels:** up to 10 DAC or up to 40 DAC-hrs in a single entry

Respiratory Instructions

The use of respiratory equipment is **CONDITIONAL** based on TEDE-ALARA evaluation results. The following respirators are allowed on this RWP:

Protective Clothing Requirements

Respiratory Instructions

The use of respiratory equipment is **CONDITIONAL** based on TEDE-ALARA evaluation results. The following respirators are allowed on this RWP:

- ULTRATWIN
- PAPR

Protective Clothing Requirements

- SURGEON'S CAP
- SHOE COVERS, ONE PAIR
- MODESTY CLOTHING
- GLOVES, RUBBER, ONE PAIR
- COVERALLS, ONE PAIR
- CLOTH INSERTS
- BOOTIES, ONE PAIR

*21110551

Provide to Applicant

Radiological Work Permit

BFN

Num.21330002 **Rev.** 1 **Status** ACTIVE

Dosimetry Instructions

Required Dosimetry

- TELEMETRY [WRM-2 OR SIMILAR DEVICE
- SELF READING DOSIMETER
- DOSIMETER OF LEGAL RECORD

Dosimetry Comments

- If dosimetry is lost or dropped, where it cannot be retrieved without leaving the immediate vicinity, the following steps shall be followed:
  A. Worker SHALL exit the IMMEDIATELY and report to RP
  B. A Qualified RP Technician shall attempt the recovery of the dosimetry
  C. If an Industrial Safety Hazard exists, a Qualified RP Technician may secure the job site while constantly monitoring the Radiological Conditions in the Area

Special Instructions

- A copy of the RWP is available for review on HIS-20 upon logging in.
- Special Dress Out requirements are permitted for laborers while performing trash and laundry activities.
- Use of respiratory protection equipment is conditional based on TEDE ALARA results.
- During radiological briefings, include discussions of specific anticipated dose rates and actions to be taken in the event of a dose rate alarm.
- RP is not to brief workers to Anticipated Alarms equal to or greater than 1,000 mrem/hr (Whole Body). No more than 3 Dose Rate Alarms are permitted per entry.
- Expected or anticipated SRD dose rate alarms should be planned, documented in eSOMS (include anticipated alarm, location, applicable WO#, and approving RP Supervisor name), and discussed with workers prior to entry into the area.
- Dose rate set points should not exceed the station's threshold for posting and controlling High Radiation Areas.
- Adjustments to SRD Set Points may be performed with approval of RP Supervision. The adjusted SRD Set Points SHALL be documented in eSOMS to include new Set Points, applicable WO#, and approving RP Supervisor name.
- Entry into a Locked High Radiation Area, Very High Radiation Area (LHRA, VHRA) is PROHIBITED on this RWP.

*21110551

Provide to Applicant

Radiological Work Permit

Num.21110551 **Rev.** 1 **Status** ACTIVE

BFN

General Work Instructions

- Radiological Protection (RP) Briefings shall be utilized prior to start of work and prior to moving into a new area to define scope of work and review the area radiological conditions.
- When Logging in to HIS-20 Perform Self-Checking to ensure the proper Work Order/Step and RWP Number is utilized.
- Monitor your SRD frequently. If dose exceeds 80% of SRD setpoint, then place systems and equipment in safe configuration and exit the area.
- During periods when HIS-20 is in the "local" mode, the default set points for the RWP are 50 mrem DOSE ALARM and 80 mrem/hr DOSE RATE ALARM.
- Dressout instructions: single dressout clothing requirements used for this RWP, unless otherwise directed by RP.
- Dressout requirements may be modified based on safety (e.g., rotating equipment, heat stress) and/or radiological conditions with RP Supervisor approval.
- A hood shall be required in a Contaminated Area if the worker is required to wear a Body Harness or utilize a phone.
- Dosimeter to be placed in the chest pocket of the PCs unless otherwise directed by RP.
- Notify RP of any activity requiring climbing 7 feet above the floor level, including temporary or permanent ladders and devices.
- Notify RP prior to any system breach, welding, grinding, or surface disturbing activities. RP shall be present for initial breaches of contaminated systems to ensure proper radiological controls are in place.
- Notify RP if Hoses or Cords Need to Cross the CA Boundary. Hoses and/or Cords must be secured to prevent Contamination outside the CA Boundaries.
- Based on radiological conditions and work activities, ensure the appropriate HEPA Unit/Vacuum is selected, issued and utilized per RP Instruction.
- Avoid Posted Hot Spots and/or Piping with Lead Shielding. Locate and utilize Low Dose Waiting Areas (LDWA). Practice ALARA.
- Upon exiting a CA, proceed to the NEAREST frisker; complete a hand & foot frisk . [If you receive an alarm - NOTIFY RP IMMEDIATELY].
- In the event a frisker is not available when exiting a CA, proceed directly to the NEAREST PCM.
- Upon performing a successful frisk and completion of task, proceed to the nearest PCM and perform a Whole Body Survey. [If the PCM alarms, survey again. If the PCM alarms a second time - NOTIFY RP IMMEDIATELY].
- Upon exiting the RCA, worker shall log out of the RWP in HIS-20.

Prepared by: FLATKINS **RPSS Approval:** JAELIAS

RPM Approval: JKSMITH **Final Approval:** JNSTYLES

*21110551

TVA	**Job Performance Measure (JPM)**

SITE:	BFN	JPM TITLE:	Emergency Action Level Classification
JPM NUMBER:	738-SRO	REVISION:	2

TASK APPLICABILITY:	☒SRO	☐STA	☐UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	S-000-EM-21 / Classify and Declare an Abnormal/Emergency Event			
K/A RATINGS:	SRO 4.6			
K/A STATEMENT:	2.4.41 Knowledge of the Emergency Action Level thresholds and classifications.			
RELATED PRA INFORMATION:	None			
SAFETY FUNCTION:	N/A			

EVALUATION LOCATION:	☐In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☒ Other - List	Classroom		

APPLICABLE METHOD OF TESTING: ☐ Discussion ☐ Simulate/Walkthrough ☒ Perform

TIME FOR COMPLETION: 30 min TIME CRITICAL (Y/N) Y ALTERNATE PATH (Y/N) N

Developed by: ______________________ ______________
Developer *Date*

(Ensure validator is briefed on exam security per NPG-SPP-17.8.1)
(See JPM Validation Checklist in NPG-SPP-17.8.2)

Validated by: ______________________ ______________
Validator *Date*

Approved by: ______________________ ______________
Site Training Management *Date*

Approved by: ______________________ ______________
Site Training Program Owner *Date*

Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 738-SRO

SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to classify an Event and complete the Initial Notification Form within the required time.

Operator Fundamental evaluated:
OF-1 Monitoring Plant Indications and Conditions Closely

PRA: N/A

REFERENCES/PROCEDURES NEEDED: EPIP-1, EPIP-2, EPIP-3, EPIP-4, EPIP-5

VALIDATION TIME: 30 minutes

PERFORMANCE TIME: ___________

COMMENTS: ____
____
____
____
____
____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: ___________________________ DATE: __________
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	12/11/2019	ALL	Initial issue
1	10/13/2020	ALL	JPM update
2	02/25/2021	ALL	Procedure update

Procedure Revisions

Procedure	Revision
EPIP-1	60
EPIP-2	40
EPIP-3	43
EPIP-4	42
EPIP-5	57

****
CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.
****

INITIAL CONDITIONS:

You are the Shift Manager. Unit 3 is operating at 100% Reactor Power with normal operating plant parameters, with the following plant conditions:

- BFN is currently conducting a Dry Cask Storage Campaign
- Fuel bundle movement is in progress in the Unit 3 Spent Fuel Pool is preparation for loading Dry Casks
- 15 minute average wind speed is 3 mph (at 91 meters)
- 15 minute average wind direction is from 90 degrees (at 91 meters)
- While moving an irradiated fuel bundle, the bundle disengages from the crane and the following conditions occur:
  - Refuel Floor personnel observe the release of gas bubbles from the dropped fuel bundle and evacuate the Refuel Floor
  - FUEL POOL FLOOR AREA RADIATION HIGH (3-RA-90-1A), Panel 3-9-3A, Window 1 alarms
  - REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH (3-RA-90-250A), Panel 3-9-3A, Window 4 alarms
  - REFUELING ZONE EXHAUST RADIATION HIGH, (3-RA-90-140A), Panel 3-9-3A, Window 34 alarms

Additionally, the following conditions exist:
- **NO** previous Events have been classified; therefore, Emergency Facilities (CECC, TSC. OSC) have **NOT** been staffed
- Emergency Director Judgement shall **NOT** be used as a basis for classification

INITIATING CUE:

Classify the Event **AND** complete the required Initial Notification Form.

This JPM is TIME CRITICAL

KEY

RA2:

RA2 - Significant lowering of water level above, or damage to, irradiated fuel.

(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.

OR

(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by alarm on **ANY** of the following radiation monitors:

- 1,2,3-RM-90-1A Fuel Pool Floor
- 1,2,3-RM-90-142A Reactor Zone Exhaust
- 1,2,3-RM-90-250A Reactor, Turbine, Refuel Exhaust
- 1,2,3-RM-90-140A Refueling Zone Exhaust

OR

(3) Lowering of spent fuel pool level to 650' 4".

KEY

KEY

BFN Unit 0	ALERT	EPIP-3 Rev. 0043 Page 12 of 29

Attachment 1
(Page 1 of 1)

Alert Initial Notification Form

1. ☐ This is a Drill ☐ This is an Actual Event-Repeat-This is an Actual Event

2. The Site Emergency Director at **Browns Ferry** has declared an **ALERT**.

3. **Initiating Condition (IC) Designator:** RA2 (USE ONLY ONE IC DESIGNATOR)

4. **Radiological Conditions:** (Check One under both Airborne and Liquid Column.)

Airborne Releases Offsite	**Liquid Releases Offsite**
☐ Minor releases within federally approved limits[1]	☐ Minor releases within federally approved limits[1]
☐Releases above federally approved limits[1]	☐Releases above federally approved limits[1]
☐ Release information not known	☐ Release information not known

5. **Event Declared:** Time: Enters Time Central Time Date: Enters Date

6. **Protective Action Recommendation:** ☒ None

[1]-Technical Specifications/Offsite Dose Calculation Manual

Completed By: ____________________

Peer Reviewed By: ____________________

EXAMINER NOTE: The YELLOW HIGHLIGHTED steps above are designated as Critical Steps in accordance with Licensed Operator Requalification Performance Indicator Standards

START TIME: _______________

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure copies of Attachment 1 from EPIP 2, 3, 4, 5 are available.	
EXAMINER NOTE: This JPM has two Time Critical sections. The candidates will have 15 minutes to classify the Event once they understand their task, AND then 15 minutes to complete any required paperwork for Notification after they complete the Classification.	
Step 1: Classifies the Event using EPIP-1. Expected Action(s): Refers to EPIP-1, and given the plant conditions declares an **Alert – RA2** (Significant lowering of Water Level above, or damage to, irradiated fuel) within 15 minutes based on the following: • Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by alarm on **ANY** of the following radiation monitors: – 3-RM-90-1A, Fuel Pool Floor (alarming) – 3-RM-90-142A, Reactor Zone Exhaust (**NOT** alarming) – 3-RM-90-250A, Reactor, Turbine, Refuel Floor Exhaust (alarming) – 3-RM-90-140A, Refueling Zone Exhaust (alarming) **TIME CLASSIFICATION COMPLETE:_______________**	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: When the candidate informs the Examiner that they have made an Event Classification, hand them the procedure they have chosen to start the next Time Critical portion of the JPM. Note the time that the candidate's 15 minute time starts in Step 2 below.	
Step 2: Implement EPIP-3, ALERT. **TIME START ______________________** Expected Action(s): Implements EPIP-3, ALERT.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: **3.0 EMERGENCY CLASSIFICATION ACTIONS** **3.0 EMERGENCY CLASSIFICATION ACTIONS** **NOTES** • Procedure steps can be performed concurrently. • All procedure steps must be completed. • All procedure attachments must be returned to the SED. • Section 3.1 (as soon as possible, within 15 Minutes from classification declaration) and Section 3.4 (as soon as possible, not to exceed 60 minutes from classification declaration) are time critical. • A Shift Technical Advisor or Senior Reactor Operator (SRO) should peer review Attachment 1 completion. **CAUTION** • Ongoing or anticipated security events or severe weather may present a danger to normal staffing and other Emergency Plan implementation processes. Observe all procedural steps carefully during severe weather and security related events. • Step 3.1[2] of the Main Body and Attachment 7, Steps 1.1[1] and 1.1[4] CANNOT be delegated. [1] **WHEN** the Technical Support Center (TSC) Site Emergency Director (SED) has assumed the responsibilities from the Shift Manager (SM)/SED, **THEN CONTINUE** in this procedure at Attachment 7. Otherwise continue in this procedure. Expected Action(s): Continues in EPIP-3, ALERT, as the TSC has not yet been staffed.	_____ SAT _____UNSAT _____N/A

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>Step 4:

3.1 State of Alabama Notification

NOTE
Notification of the State of Alabama is required to be completed as soon as possible not to exceed 15 minutes from the time of emergency classification declaration.

[1] PERFORM the following:
[1.1] RECORD the following:
• Time of ALERT Event Classification: _____

Expected Action(s):

Enters the time of Alert Event Classification.</td><td>_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td>Step 5:

[1.2] IF the Central Emergency Control Center (CECC) is NOT activated, THEN CONTINUE in this procedure at step 3.1[2]. Otherwise continue in this section.

Expected Action(s):

Proceeds to Step 3.1[2], as the CECC has not been activated.</td><td>_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td>Step 6:

[2] COMPLETE Attachment 1, “Alert Initial Notification Form.”

Expected Action(s):

Completes Attachment 1, and simulates notifying the State within 15 minutes by bringing completed Attachment 1 to the Examiner. The following are Critical items on Attachment 1:
• Initiating Condition (IC) Designator (Attachment 1, Step 3)
• Time and Date Event Declared (Attachment 1, Step 5)</td><td>Critical Step

_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: When the notification paperwork has been completed, inform the candidate “Your task is complete.”</td></tr>
</table>

STOP TIME:__________

TVA	**Job Performance Measure (JPM)**

Provide to Applicant

****
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INITIATING CUE:

Classify the Event **AND** complete the required Initial Notification Form.

This JPM is TIME CRITICAL

Provide to Applicant

BFN Unit 0	ALERT	EPIP-3 Rev. 0043 Page 12 of 29

Attachment 1
(Page 1 of 1)

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☐Releases above federally approved limits[1]	☐Releases above federally approved limits[1]
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5. **Event Declared:** Time: ___________ Central Time Date:__________

6. **Protective Action Recommendation:** ☒ None

[1]-Technical Specifications/Offsite Dose Calculation Manual

Completed By: ______________________________

Peer Reviewed By: ______________________________