Facility: Browns Ferry NPP Date of Exam: May 2022

<table>
<tr><th rowspan="2">Tier</th><th rowspan="2">Group</th><th colspan="12">RO K/A Category Points</th><th colspan="5">SRO-Only Points</th></tr>
<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>3</td><td colspan="3" rowspan="3">N/A</td><td>3</td><td>4</td><td colspan="2" rowspan="3">N/A</td><td>4</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>4</td><td>5</td><td>4</td><td>5</td><td>5</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>2</td><td>3</td><td>2</td><td>2</td><td>2</td><td>3</td><td>2</td><td>3</td><td>2</td><td>2</td><td>3</td><td>26</td><td colspan="2">2</td><td colspan="2">3</td><td>5</td></tr>
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<tr><td>Tier Totals</td><td>3</td><td>4</td><td>3</td><td>3</td><td>4</td><td>4</td><td>3</td><td>4</td><td>3</td><td>3</td><td>4</td><td>38</td><td colspan="2">4</td><td colspan="2">4</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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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		AA2.11: Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Forced Core Flow Circulation: Individual loop flow(s) AA2.09: Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Forced Core Flow Circulation: Reactor Pressure	3.6 3.4	xx xx
295003 (APE 3) Partial or Complete Loss of AC Power / 6						X	G.2.1.7: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation (Partial or Complete Loss of AC Power)	4.4	xx
295004 (APE 4) Partial or Total Loss of DC Power / 6	X						AK1.02: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Partial or Complete Loss of DC Power: Redundant DC power supplies	3.8	xx
295005 (APE 5) Main Turbine Generator Trip / 3		X					AK2.03: Knowledge of the relationship between Main Turbine Generator Trip and the following systems or components: Recirculation system	3.5	xx
295006 (APE 6) Scram / 1			X				AK3.01: Knowledge of the reasons for the following responses or actions as they apply to SCRAM: Reactor water level response	4.0	xx
295016 (APE 16) Control Room Abandonment / 7				X			AA1.01: Ability to operate and/or monitor the following as they apply to Control Room Abandonment: RPS	3.8	xx
295018 (APE 18) Partial or Complete Loss of CCW / 8					X		AA2.02: Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Component Cooling Water: Cooling water temperature	3.7	xx
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8						X	G2.4.50: Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure. (Partial or Complete Loss of Instrument Air)	4.2	xx
295021 (APE 21) Loss of Shutdown Cooling / 4	X					X	AK1.03: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of Shutdown Cooling: Adequate core cooling G2.4.41: Knowledge of the emergency action level thresholds and classifications (SRO Only) (Loss of Shutdown Cooling)	4.4 4.6	xx xx

295023 (APE 23) Refueling Accidents / 8		X					AK2.01: Knowledge of the relationship between Refueling Accidents and the following systems or components: Fuel handling equipment	3.5	xx
					X		AA2.03: Ability to determine and/or interpret the following as they apply to Refueling Accidents: Airborne contamination levels	3.2	xx
295024 High Drywell Pressure / 5			X				EK3.02: Knowledge of the reasons for the following responses or actions as they apply to High Drywell Pressure: Suppression pool spray	4.1	xx
						X	G2.4.2: Knowledge of system setpoints, interlocks and automatic actions associated with emergency and abnormal operating procedure entry conditions (High Drywell Pressure)	4.6	xx
295025 (EPE 2) High Reactor Pressure / 3				X			EA1.10: Ability to operate and/or monitor the following as they apply to High Reactor Pressure: Reactor water cleanup system	2.8	xx
295026 (EPE 3) Suppression Pool High Water Temperature / 5					X		EA2.03: Ability to determine and/or interpret the following as they apply to Suppression Pool High Water Temperature: Reactor Pressure	3.5	xx
295027 (EPE 4) High Containment Temperature (Mark III Containment Only) / 5									
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5						X	G2.2.44: 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 (High Drywell Temperature)	4.2	xx
					X		EA2.03: Ability to determine and/or interpret the following as they apply to High Drywell Temperature: Reactor Water Level	4.0	xx
295030 (EPE 7) Low Suppression Pool Water Level / 5	X						EK1.03: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Low Suppression Pool Water Level: Heat capacity	4.0	xx
295031 (EPE 8) Reactor Low Water Level / 2		X					EK2.06: Knowledge of the relationship between Reactor Low Water Level and the following systems or components: High-pressure coolant injection (HPCI)	4.1	xx
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1			X				EK3.02: Knowledge of the reasons for the following responses or actions as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Boron injection	4.2	xx
						X	G2.1.32: Ability to explain and apply system precautions, limitations, notes or cautions (Scram Condition Present and Reactor Power Above APRM Downscale or Unknown)	4.0	xx

295038 (EPE 15) High Offsite Radioactivity Release Rate / 9				X			EA1.08: Ability to operate and/or monitor the following as they apply to High Offsite Radioactivity Release Rate: MSIV leakage control	3.1	xx
600000 (APE 24) Plant Fire On Site / 8					X X		AA2.10: Ability to determine and/or interpret the following as they apply to Plant Fire on Site: Time limit of long-term-breathing air system for control room AA2.18: Assessment of control room habitability (SRO Only)	3.4 3.6	xx xx
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6						X	G2.1.20: Ability to interpret and execute procedure steps (Generator Voltage and Electric Grid Disturbances)	4.6	xx
K/A Category Totals:	3	3	3	3	4/4	4/3	Group Point Total:		20/7

ES-401 BWR Examination Outline Form ES-401-1
Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/SRO)

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				AK3.08: Knowledge of the reasons for the following responses or actions as they apply to Loss of Main Condenser Vacuum: Recirculation flow control system runbacks	3.5	xx
295007 (APE 7) High Reactor Pressure / 3				X			AA1.06: Ability to operate and/or monitor the following as they apply to High Reactor Pressure: Shutdown cooling system (RHR shutdown cooling mode)	3.6	xx
295008 (APE 8) High Reactor Water Level / 2					X		AA2.01: Ability to determine and/or interpret the following as they apply to High Reactor Water Level: Reactor water level	4.4	xx
295009 (APE 9) Low Reactor Water Level / 2						X	G2.4.18: Knowledge of the specific bases for emergency and abnormal operating procedures (Low Reactor Water Level)	4.0	xx
295010 (APE 10) High Drywell Pressure / 5									
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					X		AA2.01: Ability to determine and/or interpret the following as they apply to High Suppression Pool Water Temperature: Suppression pool temperature	4.3	xx
295014 (APE 14) Inadvertent Reactivity Addition / 1									
295015 (APE 15) Incomplete Scram / 1									
295017 (APE 17) Abnormal Offsite Release Rate / 9						X	G2.1.30: Ability to locate and operate components, including local controls (Abnormal Offsite Release Rate)	4.4	xx
295020 (APE 20) Inadvertent Containment Isolation / 5 & 7	X						AK1.05: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Inadvertent Containment Isolation: Loss of drywell/containment cooling	3.5	xx
295022 (APE 22) Loss of Control Rod Drive Pumps / 1									
295029 (EPE 6) High Suppression Pool Water Level / 5		X					EK2.07: Knowledge of the relationship between High Suppression Pool Water Level and the following systems or components: Drywell/containment water level	3.6	xx
295032 (EPE 9) High Secondary Containment Area Temperature / 5									

295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9			X				EK3.06: Knowledge of the reasons for the following responses or actions as they apply to High Secondary Containment Area Radiation Levels: Operating ventilation systems	3.6	xx
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9									
295035 (EPE 12) Secondary Containment High Differential Pressure / 5					X		EA2.01: Ability to determine and/or interpret the following as they apply to Secondary Containment High Differential Pressure: Secondary containment pressure	3.9	xx
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5									
500000 (EPE 16) High Containment Hydrogen Concentration / 5									
K/A Category Point Totals:	1	1	2	1	1/2	1/1	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											K1.16: Knowledge of the physical connections and/or cause and effect relationships between the RHR/LPCI: Injection Mode and the following systems: Component cooling water systems	2.7	xx
205000 (SF4 SCS) Shutdown Cooling		X										K2.01: Knowledge of electrical power supplies to the following: Pump motors	3.6	xx
		X										K2.02: Knowledge of electrical power supplies to the following: Motor-operated valves	3.3	xx
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection			X									K3.04: Knowledge of the effect that a loss or malfunction of the High-Pressure Coolant Injection System will have on the following systems or system parameters: Reactor power	3.6	xx
207000 (SF4 IC) Isolation (Emergency) Condenser														
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray				X								K4.05: Knowledge of Low-Pressure Core Spray System design features and/or interlocks that provide for the following: Pump minimum flow	3.4	xx
209002 (SF2, SF4 HPCS) High-Pressure Core Spray														
211000 (SF1 SLCS) Standby Liquid Control					X							K5.01: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Standby Liquid Control System: Effects of the moderator temperature coefficient of reactivity on boron	3.0	xx
						X						K6.06: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Standby Liquid Control System: Redundant reactivity control system	3.6	xx
212000 (SF7 RPS) Reactor Protection						X						K6.10: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Reactor Protection System: Reactor/turbine pressure regulating system	3.5	xx
											X	G2.1.20: Ability to interpret and execute procedure steps (Reactor Protection System)	4.6	xx
215003 (SF7 IRM) Intermediate-Range Monitor							X					A1.08: Ability to predict and/or monitor changes in parameters associated with operation of the Intermediate Range Monitor System, including: IRM back panel switches	3.1	xx

215004 (SF7 SRMS) Source-Range Monitor								X				A2.02: Ability to (a) predict the impacts of the following on the Source Range Monitor System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: SRMS inoperable condition	3.4	xx
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor									X			A3.03: Ability to monitor automatic operation of the Average Power Range Monitor/Local Power Range Monitor System, including: Meters and recorders	3.6	xx
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling								X		X		A4.12: Ability to manually operate and/or monitor in the control room: Turbine speed control A2.14: Ability to (a) predict the impacts of the following on the Reactor Core Isolation Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Rupture disc failure: exhaust-diaphragm	3.9 3.6	xx xx
218000 (SF3 ADS) Automatic Depressurization											X	G2.1.31: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup (Automatic Depressurization System)	4.6	xx
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff	X										X	K1.04: Knowledge of the physical connections and/or cause and effect relationships between the Primary Containment Isolation System/Nuclear Steam Supply Shutoff and the following systems: HPCI G2.4.31: Knowledge of annunciator alarms, indications, or response procedures (Primary Containment Isolation/Nuclear Steam Supply Shutoff)	4.2 4.1	xx xx
239002 (SF3 SRV) Safety Relief Valves		X										K2.01: Knowledge of electrical power supplies to the following: SRV solenoids	3.7	xx
259002 (SF2 RWLCS) Reactor Water Level Control			X					X				K3.04: Knowledge of the effect that a loss or malfunction of the Reactor Water Level Control System will have on the following systems or system parameters: Recirculation system A2.02: Ability to (a) predict the impacts of the following on the Reactor Water Level Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of any number of reactor feedwater flow inputs	3.3 3.8	xx xx
261000 (SF9 SGTS) Standby Gas Treatment				X								K4.02: Knowledge of Standby Gas Treatment System design features and/or interlocks that provide for the following: Charcoal bed decay heat removal	3.0	xx

262001 (SF6 AC) AC Electrical Distribution					X							K5.02: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AC Electrical Distribution: Breaker control power	3.5	xx
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)						X						K6.02: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Uninterruptible Power Supply (AC/DC): DC electrical distribution	3.4	xx
											X	G2.4.50: Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure (Uninterruptible Power Supply (AC/DC))	4.2	xx
263000 (SF6 DC) DC Electrical Distribution							X					A1.02: Ability to predict and/or monitor changes in parameters associated with operation of the DC Electrical Distribution, including: Lights and alarms	3.3	xx
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG								X				A2.08: Ability to (a) predict the impacts of the following on the Emergency Generators and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Initiation of emergency generator room fire protection system	3.1	xx
								X				A2.10: Ability to (a) predict the impacts of the following on the Emergency Generators and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: LOCA	4.4	xx
300000 (SF8 IA) Instrument Air									X			A3.04: Ability to monitor automatic operation of the Instrument Air System, including: Automatic isolation	3.4	xx
400000 (SF8 CCS) Component Cooling Water										X		A4.01: Ability to manually operate and/or monitor in the control room: CCW indications and control	3.8	xx
											X	G2.2.42: Ability to recognize system parameters that are entry-level conditions for technical specifications (Component Cooling Water System)	4.6	xx
510000 (SF4 SWS*) Service Water (Normal and Emergency)											X	G2.4.46: Ability to verify that the alarms are consistent with the plant conditions (Service Water (Normal and Emergency))	4.2	xx
K/A Category Point Totals:	2	3	2	2	2	3	2	3/2	2	2	3/3	Group Point Total:		26/5

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

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														
201003 (SF1 CRDM) Control Rod and Drive Mechanism														
201004 (SF7 RSCS) Rod Sequence Control														
201005 (SF1, SF7 RCIS) Rod Control and Information														
201006 (SF7 RWMS) Rod Worth Minimizer										X		A4.02: Ability to manually operate and/or monitor in the control room: Pushbutton indicating switches	3.2	xx
202001 (SF1, SF4 RS) Recirculation					X							K5.06: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Recirculation System: ATWS RPT	3.8	xx
202002 (SF1 RSCTL) Recirculation Flow Control														
204000 (SF2 RWCU) Reactor Water Cleanup								X				A2.05: Ability to (a) predict the impacts of the following on the Reactor Water Cleanup System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Abnormal valve position	3.0	xx
214000 (SF7 RPIS) Rod Position Information						X						K6.01: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Rod Position Information System: RPIS power supply	3.2	xx
215001 (SF7 TIP) Traversing In-Core Probe							X					A1.01: Ability to predict and/or monitor changes in parameters associated with operation of the Traversing In-Core Probe, including: Area radiation levels	3.1	xx
215002 (SF7 RBMS) Rod Block Monitor														
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			A3.04: Ability to monitor automatic operation of the Primary Containment System and Auxiliaries, including: Containment/drywell response during LOCA	4.2	xx
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode														

230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Spray Mode	X											K1.01: Knowledge of the physical connections and/or cause and effect relationships between the RHR/LPCI: Torus/Suppression Pool Spray Mode and the following systems: Primary containment	3.9	xx
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup								X				A2.09: Ability to (a) predict the impacts of the following on the Fuel Pool Cooling and Cleanup and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: AC electrical power failures	3.4	xx
234000 (SF8 FH) Fuel-Handling Equipment														
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	G2.4.30: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator (Reactor/Turbine Pressure Regulating)	4.1	xx
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary														
256000 (SF2 CDS) Condensate														
259001 (SF2 FWS) Feedwater				X								K4.12: Knowledge of Feedwater System design features and/or interlocks that provide for the following: RFP start permissives	3.1	xx
268000 (SF9 RW) Radwaste											X	G2.2.3: (Multi-unit license) Knowledge of the design, procedural, or operational differences between units	3.8	xx
271000 (SF9 OG) Offgas					X							K5.09: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Offgas System: Hydrogen and oxygen recombination	3.1	xx
272000 (SF7, SF9 RMS) Radiation Monitoring														
286000 (SF8 FPS) Fire Protection			X									K3.09: Knowledge of the effect that a loss or malfunction of the Fire Protection System will have on the following systems or system parameters: AC electrical distribution systems	2.8	xx

288000 (SF9 PVS) Plant Ventilation								X				A2.05: 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 operations: Extreme outside weather conditions	2.9	xx
290001 (SF5 SC) Secondary Containment														
290003 (SF9 CRV) Control Room Ventilation		X										K2.04: Knowledge of electrical power supplies to the following: Control room HVAC logic	3.1	xx
290002 (SF4 RVI) Reactor Vessel Internals														
51001 (SF8 CWS*) Circulating Water														
K/A Category Point Totals:	1	1	1	1	2	1	1	1/2	1	1	1/1	Group Point Total:		12/3

ES-401 **Generic Knowledge and Abilities Outline (Tier 3)** **Form ES-401-3**

Facility: Browns Ferry NPP   Date of Exam: May 2022

Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements	3.8	xx	-	-
	2.1.29	Knowledge of how to conduct system lineups, such as valves, breakers, or switches	4.1	xx	-	-
	2.1.36	Knowledge of procedures and limitations involved in core alterations	-	-	4.1	xx
	Subtotal			2		1
2. Equipment Control	2.2.22	Knowledge of limiting conditions for operation and safety limits	4.0	xx	-	-
	2.2.35	Ability to determine technical specification mode of operation	3.6	xx	-	-
	2.2.4	(Multi-unit license) Ability to explain the variations in control room layouts, systems, instrumentation, or procedural actions between units at a facility	-	-	3.6	xx
	2.2.19	Knowledge of maintenance work order requirements	-	-	3.4	xx
	Subtotal			2		2
3. Radiation Control	2.3.5	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms or personnel monitoring equipment	2.9	xx	-	-
	2.3.11	Ability to control radiation releases	3.8	xx	-	-
	2.3.12	Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters	3.2	xx	-	-
	2.3.6	Ability to approve liquid or gaseous release permits	-	-	3.8	xx
	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only)	-	-	3.8	xx
	Subtotal			3		2
4. Emergency Procedures/Plan	2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions	3.7	xx	-	-
	2.4.17	Knowledge of emergency and abnormal operating procedures terms and definitions	3.9	xx	-	-
	2.4.32	Knowledge of operator response to loss of annunciators	3.6	xx	-	-
	2.4.28	Knowledge of procedures relating to a security event (ensure that the test item includes no safeguards information)	-	-	4.1	xx
	2.4.38	Ability to take actions required by the facility emergency plan implementing procedures, including supporting or acting as emergency coordinator	-	-	4.4	xx
	Subtotal			3		2
Tier 3 Point Total				10		7

Form 4.1-BWR Boiling-Water Reactor Examination Outline

<table>
<tr><td colspan="8">Facility: Browns Ferry NPP</td><td colspan="11">Date of Exam: May 2022</td></tr>
<tr><td rowspan="2">Tier</td><td rowspan="2">Group</td><td colspan="12">RO K/A Category Points</td><td colspan="5">SRO-Only Points</td></tr>
<tr><td>K1</td><td>K2</td><td>K3</td><td>K4</td><td>K5</td><td>K6</td><td>A1</td><td>A2</td><td>A3</td><td>A4</td><td>G</td><td>Total</td><td colspan="2">A2</td><td colspan="2">G</td><td>Total</td></tr>
<tr><td rowspan="3">1.
Emergency and Abnormal Plant Evolutions</td><td>1</td><td>3</td><td>3</td><td>3</td><td colspan="3" rowspan="3">N/A</td><td>3</td><td>4</td><td colspan="2" rowspan="3">N/A</td><td>4</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>4</td><td>4</td><td>4</td><td>5</td><td>5</td><td>26</td><td colspan="2">6</td><td colspan="2">4</td><td>10</td></tr>
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Plant Systems</td><td>1</td><td>2</td><td>3</td><td>2</td><td>2</td><td>2</td><td>3</td><td>2</td><td>3</td><td>2</td><td>2</td><td>3</td><td>26</td><td colspan="2">2</td><td colspan="2">3</td><td>5</td></tr>
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<tr><td>Tier Totals</td><td>3</td><td>4</td><td>3</td><td>3</td><td>3</td><td>4</td><td>3</td><td>4</td><td>3</td><td>3</td><td>4</td><td>37</td><td colspan="2">4</td><td colspan="2">4</td><td>8</td></tr>
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Generic Knowledge and Abilities Categories</td><td colspan="2">CO</td><td colspan="5">EC</td><td colspan="2">RC</td><td colspan="3">EM</td><td rowspan="2">6</td><td>CO</td><td>EC</td><td>RC</td><td>EM</td><td rowspan="2">7</td></tr>
<tr><td colspan="2">2</td><td colspan="5">2</td><td colspan="2">1</td><td colspan="3">1</td><td>1</td><td>2</td><td>2</td><td>2</td></tr>
<tr><td rowspan="2">4. Theory</td><td colspan="4">Reactor Theory</td><td colspan="8">Thermodynamics</td><td rowspan="2">6</td><td colspan="5" rowspan="2"></td></tr>
<tr><td colspan="4">3</td><td colspan="8">3</td></tr>
</table>

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

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

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

Form 4.1-BWR

BWR Examination Outline

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		AA2.11: Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Forced Core Flow Circulation: Individual loop flow(s)	3.6	xx
					X		AA2.09: Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Forced Core Flow Circulation: Reactor Pressure	3.4	xx
295003 (APE 3) Partial or Complete Loss of AC Power / 6						X	G.2.1.7: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation (Partial or Complete Loss of AC Power)	4.4	xx
295004 (APE 4) Partial or Total Loss of DC Power / 6	X						AK1.02: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Partial or Complete Loss of DC Power: Redundant DC power supplies	3.8	xx
295005 (APE 5) Main Turbine Generator Trip / 3		X					AK2.03: Knowledge of the relationship between Main Turbine Generator Trip and the following systems or components: Recirculation system	3.5	xx
295006 (APE 6) Scram / 1			X				AK3.01: Knowledge of the reasons for the following responses or actions as they apply to SCRAM: Reactor water level response	4.0	xx
295016 (APE 16) Control Room Abandonment / 7				X			AA1.01: Ability to operate and/or monitor the following as they apply to Control Room Abandonment: RPS	3.8	xx
295018 (APE 18) Partial or Complete Loss of CCW / 8					X		AA2.02: Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Component Cooling Water: Cooling water temperature	3.7	xx
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8						X	G2.4.50: Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure. (Partial or Complete Loss of Instrument Air)	4.2	xx
295021 (APE 21) Loss of Shutdown Cooling / 4	X						AK1.03: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of Shutdown Cooling: Adequate core cooling	4.4	xx
						X	G2.4.41: Knowledge of the emergency action level thresholds and classifications (SRO Only) (Loss of Shutdown Cooling)	4.6	xx

295023 (APE 23) Refueling Accidents / 8		X			X		AK2.01: Knowledge of the relationship between Refueling Accidents and the following systems or components: Fuel handling equipment AA2.03: Ability to determine and/or interpret the following as they apply to Refueling Accidents: Airborne contamination levels	3.5 3.2	xx xx
295024 High Drywell Pressure / 5			X			X	EK3.02: Knowledge of the reasons for the following responses or actions as they apply to High Drywell Pressure: Suppression pool spray G2.4.2: Knowledge of system setpoints, interlocks and automatic actions associated with emergency and abnormal operating procedure entry conditions (High Drywell Pressure)	4.1 4.6	xx xx
295025 (EPE 2) High Reactor Pressure / 3				X			EA1.10: Ability to operate and/or monitor the following as they apply to High Reactor Pressure: Reactor water cleanup system	2.8	xx
295026 (EPE 3) Suppression Pool High Water Temperature / 5					X		EA2.03: Ability to determine and/or interpret the following as they apply to Suppression Pool High Water Temperature: Reactor Pressure	3.5	xx
295027 (EPE 4) High Containment Temperature (Mark III Containment Only) / 5									
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5					X	X	G2.2.44: 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 (High Drywell Temperature) EA2.03: Ability to determine and/or interpret the following as they apply to High Drywell Temperature: Reactor Water Level	4.2 4.0	xx xx
295030 (EPE 7) Low Suppression Pool Water Level / 5	X						EK1.03: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Low Suppression Pool Water Level: Heat capacity	4.0	xx
295031 (EPE 8) Reactor Low Water Level / 2		X					EK2.06: Knowledge of the relationship between Reactor Low Water Level and the following systems or components: High-pressure coolant injection (HPCI)	4.1	xx
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1			X			X	EK3.02: Knowledge of the reasons for the following responses or actions as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Boron injection G2.1.32: Ability to explain and apply system precautions, limitations, notes or cautions (Scram Condition Present and Reactor Power Above APRM Downscale or Unknown)	4.2 4.0	xx xx

295038 (EPE 15) High Offsite Radioactivity Release Rate / 9				X			EA1.08: Ability to operate and/or monitor the following as they apply to High Offsite Radioactivity Release Rate: MSIV leakage control	3.1	xx
600000 (APE 24) Plant Fire On Site / 8					X X		AA2.10: Ability to determine and/or interpret the following as they apply to Plant Fire on Site: Time limit of long-term-breathing air system for control room AA2.18: Assessment of control room habitability (SRO Only)	3.4 3.6	xx xx
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6						X	G2.1.20: Ability to interpret and execute procedure steps (Generator Voltage and Electric Grid Disturbances)	4.6	xx
K/A Category Totals:	3	3	3	3	4/4	4/3	Group Point Total:		20/7

Form 4.1-BWR
BWR Examination Outline
Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/SRO)
Page 3

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									
295007 (APE 7) High Reactor Pressure / 3				X			AA1.06: Ability to operate and/or monitor the following as they apply to High Reactor Pressure: Shutdown cooling system (RHR shutdown cooling mode)	3.6	xx
295008 (APE 8) High Reactor Water Level / 2					X		AA2.01: Ability to determine and/or interpret the following as they apply to High Reactor Water Level: Reactor water level	4.4	xx
295009 (APE 9) Low Reactor Water Level / 2						X	G2.4.18: Knowledge of the specific bases for emergency and abnormal operating procedures (Low Reactor Water Level)	4.0	xx
295010 (APE 10) High Drywell Pressure / 5									
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					X		AA2.01: Ability to determine and/or interpret the following as they apply to High Suppression Pool Water Temperature: Suppression pool temperature	4.3	xx
295014 (APE 14) Inadvertent Reactivity Addition / 1									
295015 (APE 15**) Incomplete Scram / 1									
295017 (APE 17) Abnormal Offsite Release Rate / 9						X	G2.1.30: Ability to locate and operate components, including local controls (Abnormal Offsite Release Rate)	4.4	xx
295020 (APE 20) Inadvertent Containment Isolation / 5 & 7	X						AK1.05: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Inadvertent Containment Isolation: Loss of drywell/containment cooling	3.5	xx
295022 (APE 22) Loss of Control Rod Drive Pumps / 1									
295029 (EPE 6) High Suppression Pool Water Level / 5		X					EK2.07: Knowledge of the relationship between High Suppression Pool Water Level and the following systems or components: Drywell/containment water level	3.6	xx
295032 (EPE 9) High Secondary Containment Area Temperature / 5									

295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9			X				EK3.06: Knowledge of the reasons for the following responses or actions as they apply to High Secondary Containment Area Radiation Levels: Operating ventilation systems	3.6	xx
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9									
295035 (EPE 12) Secondary Containment High Differential Pressure / 5					X		EA2.01: Ability to determine and/or interpret the following as they apply to Secondary Containment High Differential Pressure: Secondary containment pressure	3.9	xx
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5									
500000 (EPE 16) High Containment Hydrogen Concentration / 5									
K/A Category Point Totals:	1	1	1	1	1/2	1/1	Group Point Total:		6/3

Form 4.1-BWR

BWR Examination Outline

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											K1.16: Knowledge of the physical connections and/or cause and effect relationships between the RHR/LPCI: Injection Mode and the following systems: Component cooling water systems	2.7	xx
205000 (SF4 SCS) Shutdown Cooling		X										K2.01: Knowledge of electrical power supplies to the following: Pump motors	3.6	xx
		X										K2.02: Knowledge of electrical power supplies to the following: Motor-operated valves	3.3	xx
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection			X									K3.04: Knowledge of the effect that a loss or malfunction of the High-Pressure Coolant Injection System will have on the following systems or system parameters: Reactor power	3.6	xx
207000 (SF4 IC) Isolation (Emergency) Condenser														
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray				X								K4.05: Knowledge of Low-Pressure Core Spray System design features and/or interlocks that provide for the following: Pump minimum flow	3.4	xx
209002 (SF2, SF4 HPCS) High-Pressure Core Spray														
211000 (SF1 SLCS) Standby Liquid Control					X							K5.01: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Standby Liquid Control System: Effects of the moderator temperature coefficient of reactivity on boron	3.0	xx
						X						K6.06: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Standby Liquid Control System: Redundant reactivity control system	3.6	xx
212000 (SF7 RPS) Reactor Protection						X						K6.10: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Reactor Protection System: Reactor/turbine pressure regulating system	3.5	xx
											X	G2.1.20: Ability to interpret and execute procedure steps (Reactor Protection System)	4.6	xx
215003 (SF7 IRM) Intermediate-Range Monitor							X					A1.08: Ability to predict and/or monitor changes in parameters associated with operation of the Intermediate Range Monitor System, including: IRM back panel switches	3.1	xx

215004 (SF7 SRMS) Source-Range Monitor								X				A2.02: Ability to (a) predict the impacts of the following on the Source Range Monitor System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: SRMS inoperable condition	3.4	xx
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor									X			A3.03: Ability to monitor automatic operation of the Average Power Range Monitor/Local Power Range Monitor System, including: Meters and recorders	3.6	xx
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling								X		X		A4.12: Ability to manually operate and/or monitor in the control room: Turbine speed control A2.14: Ability to (a) predict the impacts of the following on the Reactor Core Isolation Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Rupture disc failure: exhaust-diaphragm	3.9 3.6	xx xx
218000 (SF3 ADS) Automatic Depressurization											X	G2.1.23: Ability to perform general or normal operating procedures during any plant condition (Automatic Depressurization System)	4.3	xx
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff	X										X	K1.04: Knowledge of the physical connections and/or cause and effect relationships between the Primary Containment Isolation System/Nuclear Steam Supply Shutoff and the following systems: HPCI G2.4.31: Knowledge of annunciator alarms, indications, or response procedures (Primary Containment Isolation/Nuclear Steam Supply Shutoff)	4.2 4.1	xx xx
239002 (SF3 SRV) Safety Relief Valves		X										K2.01: Knowledge of electrical power supplies to the following: SRV solenoids	3.7	xx
259002 (SF2 RWLCS) Reactor Water Level Control			X					X				K3.04: Knowledge of the effect that a loss or malfunction of the Reactor Water Level Control System will have on the following systems or system parameters: Recirculation system A2.02: Ability to (a) predict the impacts of the following on the Reactor Water Level Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of any number of reactor feedwater flow inputs	3.3 3.8	xx xx
261000 (SF9 SGTS) Standby Gas Treatment				X								K4.02: Knowledge of Standby Gas Treatment System design features and/or interlocks that provide for the following: Charcoal bed decay heat removal	3.0	xx

262001 (SF6 AC) AC Electrical Distribution					X							K5.02: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AC Electrical Distribution: Breaker control power	3.5	xx
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)						X						K6.02: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Uninterruptible Power Supply (AC/DC): DC electrical distribution	3.4	xx
											X	291008, K1.06: Interpreting one-line diagram of control circuitry (Uninterruptible Power Supply (AC/DC))	3.6	xx
263000 (SF6 DC) DC Electrical Distribution							X					A1.02: Ability to predict and/or monitor changes in parameters associated with operation of the DC Electrical Distribution, including: Lights and alarms	3.3	xx
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG								X				A2.08: Ability to (a) predict the impacts of the following on the Emergency Generators and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Initiation of emergency generator room fire protection system	3.1	xx
								X				A2.10: Ability to (a) predict the impacts of the following on the Emergency Generators and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: LOCA	4.4	xx
300000 (SF8 IA) Instrument Air									X			A3.04: Ability to monitor automatic operation of the Instrument Air System, including: Automatic isolation	3.4	xx
400000 (SF8 CCS) Component Cooling Water										X		A4.01: Ability to manually operate and/or monitor in the control room: CCW indications and control	3.8	xx
											X	G2.2.42: Ability to recognize system parameters that are entry-level conditions for technical specifications (Component Cooling Water System)	4.6	xx
510000 (SF4 SWS*) Service Water (Normal and Emergency)											X	G2.1.2: Knowledge of operator responsibilities during any mode of plant operation (Service Water (Normal and Emergency))	4.1	xx
K/A Category Point Totals:	2	3	2	2	2	3	2	3/2	2	2	3/3	Group Point Total:		26/5

Form 4.1-BWR

BWR Examination Outline
Plant Systems—Tier 2/Group 2 (RO/SRO)

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														
201003 (SF1 CRDM) Control Rod and Drive Mechanism														
201004 (SF7 RSCS) Rod Sequence Control														
201005 (SF1, SF7 RCIS) Rod Control and Information														
201006 (SF7 RWMS) Rod Worth Minimizer										X		A4.02: Ability to manually operate and/or monitor in the control room: Pushbutton indicating switches	3.2	xx
202001 (SF1, SF4 RS) Recirculation					X							K5.06: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Recirculation System: ATWS RPT	3.8	xx
202002 (SF1 RSCTL) Recirculation Flow Control														
204000 (SF2 RWCU) Reactor Water Cleanup								X				A2.05: Ability to (a) predict the impacts of the following on the Reactor Water Cleanup System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Abnormal valve position	3.0	xx
214000 (SF7 RPIS) Rod Position Information						X						K6.01: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Rod Position Information System: RPIS power supply	3.2	xx
215001 (SF7 TIP) Traversing In-Core Probe							X					A1.01: Ability to predict and/or monitor changes in parameters associated with operation of the Traversing In-Core Probe, including: Area radiation levels	3.1	xx
215002 (SF7 RBMS) Rod Block Monitor														
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			A3.04: Ability to monitor automatic operation of the Primary Containment System and Auxiliaries, including: Containment/drywell response during LOCA	4.2	xx
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode														

230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Spray Mode	X											K1.01: Knowledge of the physical connections and/or cause and effect relationships between the RHR/LPCI: Torus/Suppression Pool Spray Mode and the following systems: Primary containment	3.9	xx
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup								X				A2.09: Ability to (a) predict the impacts of the following on the Fuel Pool Cooling and Cleanup and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: AC electrical power failures	3.4	xx
234000 (SF8 FH) Fuel-Handling Equipment														
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	G2.4.30: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator (Reactor/Turbine Pressure Regulating)	4.1	xx
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary														
256000 (SF2 CDS) Condensate														
259001 (SF2 FWS) Feedwater				X								K4.12: Knowledge of Feedwater System design features and/or interlocks that provide for the following: RFP start permissives	3.1	xx
268000 (SF9 RW) Radwaste											X	291007, K1.07: Principles of demineralizer operation (Radwaste)	2.5	xx
271000 (SF9 OG) Offgas														
272000 (SF7, SF9 RMS) Radiation Monitoring														
286000 (SF8 FPS) Fire Protection			X									K3.09: Knowledge of the effect that a loss or malfunction of the Fire Protection System will have on the following systems or system parameters: AC electrical distribution systems	2.8	xx
288000 (SF9 PVS) Plant Ventilation								X				A2.05: 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 operations: Extreme outside weather conditions	2.9	xx
290001 (SF5 SC) Secondary Containment														

290003 (SF9 CRV) Control Room Ventilation		X										K2.04: Knowledge of electrical power supplies to the following: Control room HVAC logic	3.1	xx
290002 (SF4 RVI) Reactor Vessel Internals														
510001 (SF8 CWS*) Circulating Water														
K/A Category Point Totals:	1	1	1	1	1	1	1	1/2	1	1	1/1	Group Point Total:		11/3

Form 4.1-BWR

BWR Examination Outline
Generic Knowledge and Abilities—Tier 3 (RO/SRO)

Page 6

Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements	3.8	xx	-	-
	2.1.29	Knowledge of how to conduct system lineups, such as valves, breakers, or switches	4.1	xx	-	-
	2.1.36	Knowledge of procedures and limitations involved in core alterations	-	-	4.1	xx
	Subtotal			2		1
2. Equipment Control	2.2.22	Knowledge of limiting conditions for operation and safety limits	4.0	xx	-	-
	2.2.35	Ability to determine technical specification mode of operation	3.6	xx	-	-
	2.2.4	(Multi-unit license) Ability to explain the variations in control room layouts, systems, instrumentation, or procedural actions between units at a facility	-	-	3.6	xx
	2.2.19	Knowledge of maintenance work order requirements	-	-	3.4	xx
	Subtotal			2		2
3. Radiation Control	2.3.12	Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters	3.2	xx	-	-
	2.3.6	Ability to approve liquid or gaseous release permits	-	-	3.8	xx
	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only)	-	-	3.8	xx
	Subtotal			1		2
4. Emergency Procedures/Plan	2.4.32	Knowledge of operator response to loss of annunciators	3.6	xx	-	-
	2.4.28	Knowledge of procedures relating to a security event (ensure that the test item includes no safeguards information)	-	-	4.1	xx
	2.4.38	Ability to take actions required by the facility emergency plan implementing procedures, including supporting or acting as emergency coordinator	-	-	4.4	xx
	Subtotal			1		2
Tier 3 Point Total				6		7

Form 4.1-BWR

BWR Examination Outline
Theory—Tier 4 (RO)

Page 7

Category	K/A #	Topic	RO	
			IR	#
1. Reactor Theory	292001 K1.02	Define prompt and delayed neutrons	3.1	xx
	292005 K1.12	Describe effects of deep and shallow control rods on axial and radial flux distribution	2.9	xx
	292007 K1.03	Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points.	2.7	xx
	Subtotal			3
2. Thermodynamics	293003 K1.22	Explain the usefulness of steam tables to the control room operator	3.2	xx
	293005 K1.06	Describe how changes in system parameters affect thermodynamic efficiency	2.6	xx
	293007 K1.03	Heat Transfer, Explain the manner in which fluid films affects heat transfer	2.8	xx
	Subtotal			3
Tier 4 Point Total				6

Facility Browns Ferry NPP Date of Examination: 5/16/22

Examination Level: RO ☒ SRO ☐ Operating Test Number: 22-04

Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)		Type Code* (Step 3)
Conduct of Operations	JPM 556	Drywell Leakage Calculation	R, M
	K/A 2.1.7 (RO 4.4)	Ability to evaluate plant performance and make operational judgments based on operating characteristics, Reactor behavior, and instrument interpretation.	
Conduct of Operations	JPM 661	Determine Adequate Performance of License Reactivation	R, D
	K/A 2.1.4 (RO 3.3)	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10 CFR Part 55	
Equipment Control	JPM 680	Perform 2-SR-3.4.2.1, Jet Pump Mismatch and Operability	R, M
	K/A 2.2.12 (RO 3.7)	Knowledge of surveillance procedures	
Radiation Control	JPM 682	Review a Radiological Work Permit (RWP)	R, P
	K/A 2.3.12 (RO 3.2)	Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters.	
Emergency Plan	N/A		N/A

Instructions for completing Form 3.2-1, “Administrative Topics Outline”

1. For each license level, determine the number of administrative job performance measures (JPMs) and topic areas as follows:

Topic	**Number of JPMs**	
	RO*	**SRO and RO Retakes**
Conduct of Operations	1 (or 2)	2
Equipment Control	1 (or 0)	1
Radiation Control	1 (or 0)	1
Emergency Plan	1 (or 0)	1
Total	**4**	**5**

* Reactor operator (RO) applicants do not need to be evaluated on every topic (i.e., “Equipment Control,” “Radiation Control,” or “Emergency Plan” can be omitted by doubling up on “Conduct of Operations”), unless the applicant is taking only the administrative topics portion of the operating test (with a waiver or excusal of the other portions).

2. Enter the associated knowledge and abilities (K/A) statement and summarize the administrative activities for each JPM.

3. For each JPM, specify the type codes for location and source as follows:

Location:

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

Source and Source Criteria:

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

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

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

Reactor Operator

1. **Conduct of Ops** – Drywell Leakage Calculation

   Using 2-SR-2, Instrument Checks and Observations, and given Floor and Equipment Drain readings calculates leak rates and determines if leak rates are acceptable in accordance with Technical Specifications.

   K/A 2.1.7: Ability to evaluate plant performance and make operational judgments based on operating characteristics, Reactor behavior, and instrument interpretation. (RO 4.4)

2. **Conduct of Ops** – Determine Adequate Performance of License Reactivation

   Using OPDP-10, License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions and a provided table determines which Unit Operator has performed the necessary steps to reactivate their license.

   K/A 2.1.4: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10 CFR Part 55. (RO 3.3)

3. **Equipment Control** – Perform 2-SR-3.4.2.1, Jet Pump Mismatch and Operability

   Given plant conditions, determine if Jet Pump flow mismatch meets Technical Specification requirements in accordance with 2-SR-3.4.2.1, Jet Pump Mismatch and Operability.

   K/A 2.2.12 Knowledge of surveillance procedures. (RO 3.7)

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.

   K/A 2.3.12: Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters. (RO 3.2)

5. **Emergency Plan** – N/A

Facility Browns Ferry NPP Date of Examination: 5/16/22

Examination Level: RO ☐ SRO ☒ Operating Test Number: 22-04

Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)		Type Code* (Step 3)
Conduct of Operations	JPM 556	Drywell Leakage Calculation	R, M
	K/A 2.1.7 (SRO 4.7)	Ability to evaluate plant performance and make operational judgments based on operating characteristics, Reactor behavior, and instrument interpretation.	
Conduct of Operations	JPM 753	Determine Protected Equipment Requirements	R, N
	K/A 2.1.39 (SRO 4.3)	Knowledge of conservative decision-making practices	
Equipment Control	JPM 746	Review Power Availability Surveillance	R, M
	K/A 2.2.37 (SRO 4.6)	Ability to determine operability or availability of safety-related equipment	
Radiation Control	JPM 749	Determine ACTIONS required to allow releases in accordance with 0-ODCM-001, OFFSITE DOSE CALCULATION MANUAL	R, P
	K/A 2.3.11 (SRO 4.3)	Ability to control radiation releases	
Emergency Plan	JPM 752	Emergency Action Level Classification	R, N
	K/A 2.4.41 (SRO 4.6)	Knowledge of the Emergency Action Level thresholds and Classifications	

Instructions for completing Form 3.2-1, “Administrative Topics Outline”

1. For each license level, determine the number of administrative job performance measures (JPMs) and topic areas as follows:

Topic	**Number of JPMs**	
	RO*	**SRO and RO Retakes**
Conduct of Operations	1 (or 2)	2
Equipment Control	1 (or 0)	1
Radiation Control	1 (or 0)	1
Emergency Plan	1 (or 0)	1
Total	**4**	**5**

* Reactor operator (RO) applicants do not need to be evaluated on every topic (i.e., “Equipment Control,” “Radiation Control,” or “Emergency Plan” can be omitted by doubling up on “Conduct of Operations”), unless the applicant is taking only the administrative topics portion of the operating test (with a waiver or excusal of the other portions).

2. Enter the associated knowledge and abilities (K/A) statement and summarize the administrative activities for each JPM.

3. For each JPM, specify the type codes for location and source as follows:

Location:

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

Source and Source Criteria:

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

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

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

Senior Reactor Operator

1. **Conduct of Ops** – Drywell Leakage Calculation

   Using 2-SR-2, Instrument Checks and Observations, and given Floor and Equipment Drain readings calculates leak rates and determines if leak rates are acceptable and if any Technical Specification action(s) must be taken.

   K/A 2.1.7: Ability to evaluate plant performance and make operational judgments based on operating characteristics, Reactor behavior, and instrument interpretation. (SRO 4.7)

2. **Conduct of Ops** – Determine Protected Equipment Requirements

   Given an abnormal plant equipment configuration, uses ODM-4.18, Protected Equipment and NPG-SPP-07.3.4, Protected Equipment to determine the proper administrative barrier(s) to guard against inadvertently rendering equipment important to plant safety inoperable.

   K/A 2.1.39: Knowledge of conservative decision-making practices. (SRO 4.3)

3. **Equipment Control** – Review Power Availability Surveillance

   Given a completed Surveillance Requirement, 3-SR-3.8.7.1 Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems, determines that an Acceptance Criteria (AC) step was not met and required Technical Specification actions.

   K/A 2.2.37 Ability to determine operability or availability of safety-related equipment (SRO 4.6)

4. **Radiation Control** – Determine ACTIONS required to allow releases in accordance with 0-ODCM-001, OFFSITE DOSE CALCULATION MANUAL

   Given that an exhaust radiation monitor is taken out of service for maintenance, determine the governing procedure and determine what ACTIONS must be taken to allow continued releases via the affected pathway.

   K/A 2.3.11: Ability to control radiation releases (SRO 4.3)

5. **Emergency Plan** – Emergency Action Level Classification (3.1-G)

   Given plant conditions uses EPIP-1, Emergency Classification Procedure classifies an event and fills out the required notification form.

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

Facility: Browns Ferry NPP
Date of Examination: 5/16/22
Exam Level: ☒ RO ☐ SRO-I ☐ SRO-U
Operating Test No.: 22-04

Control Room Systems			
System / JPM Title		**Type Code**	**Safety Function**
a. JPM 708A	Inject Boron during an ATWS per EOI-Appendix-3A, SLC Injection	A, D, EN, S	1
b. JPM 751A	Align RCIC Suction to the Suppression Pool per EOI-Appendix-5C, Injection System Lineup RCIC	A, L, N, S	2
c. JPM 627A	Place HPCI in Reactor Pressure Control per EOI-Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode	A, D, EN, L, S	4
d. JPM 750	Vent the Drywell per OI-64, Primary Containment System	EN, N, S	5
e. JPM 725	Transfer ‘C’ 4KV Unit Board from the Start Bus to The USST Per 0-OI-57A, Switchyard And 4160V AC Electrical System	D, P, S	6
f. JPM 290A	Perform SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test	A, D, EN, P, S	7
g. JPM 39	Align Containment Atmosphere Dilution to Drywell Control Air per EOI-Appendix-8G, Crosstie CAD to Drywell Control Air	D, L, S	8
h. JPM 55A	Emergency Vent Primary Containment per EOI Appendix-13, Emergency Venting Primary Containment	A, D, EN, L, S	9
In-Plant Systems			
i. JPM 754	Align Components per AOI-100-2, Control Room Abandonment, Attachment 3, Part B	E, L, N, R	6
j. JPM 314	Defeat ARI Logic Trips per EOI Appendix-2, Defeating ARI Logic Trips	D, E, L	7
k. JPM 755	Bypass HPCI High Temperature Isolation per EOI Appendix-16L, Bypassing HPCI High Temperature Isolation	E, L, N	2

1. Determine the number of control room system and in-plant system job performance measures (JPMs) to develop using the following table:

License Level	Control Room	In-Plant	Total
Reactor Operator (RO)	8	3	11
Senior Reactor Operator-Instant (SRO-I)	7	3	10
Senior Reactor Operator-Upgrade (SRO-U)	2 or 3	3 or 2	5

2. Select safety functions and systems for each JPM as follows:

   Refer to Section 1.9 of the applicable knowledge and abilities (K/A) catalog for the plant systems organized by safety function. For pressurized-water reactor operating tests, the primary and secondary systems listed under Safety Function 4, "Heat Removal from Reactor Core," in Section 1.9 of the applicable K/A catalog, may be treated as separate safety functions (i.e., two systems, one primary and one secondary, may be selected from Safety Function 4). From the safety function groupings identified in the K/A catalog, select the appropriate number of plant systems by safety functions to be evaluated based on the applicant's license level (see the table in step 1).

   For RO/SRO-I applicants: Each of the control room system JPMs and, separately, each of the in-plant system JPMs must evaluate a different safety function, and the same system or evolution cannot be used to evaluate more than one safety function in each location. One of the control room system JPMs must be an engineered safety feature.

   For SRO-U applicants: Evaluate SRO-U applicants on five different safety functions. One of the control room system JPMs must be an engineered safety feature, and the same system or evolution cannot be used to evaluate more than one safety function.

3. Select a task for each JPM that supports, either directly or indirectly and in a meaningful way, the successful fulfillment of the associated safety function. Select the task from the applicable K/A catalog (K/As for plant systems or emergency and abnormal plant evolutions) or the facility licensee's site-specific task list. If this task has an associated K/A, the K/A should have an importance rating of at least 2.5 in the RO column. K/As that have importance ratings of less than 2.5 may be used if justified based on plant priorities; inform the NRC chief examiner if selecting K/As with an importance rating less than 2.5.
   The selected tasks must be different from the events and evolutions conducted during the simulator operating test and tasks tested on the written examination. A task that is similar to a simulator scenario event may be acceptable if the actions required to complete the task are significantly different from those required in response to the scenario event.

   Apply the following specific task selection criteria:

   - At least one of the tasks shall be related to a shutdown or low-power condition.
   - Four to six of the tasks for RO and SRO-I applicants shall require execution of alternative paths within the facility licensee's operating procedures. Two to three of the tasks for SRO-U applicants shall require execution of alternative paths within the facility licensee's operating procedures. At least one alternate path JPM must be new or modified from the bank.
   - At least one of the tasks conducted in the plant shall evaluate the applicant's ability to implement actions required during an emergency or abnormal condition.
   - At least one of the tasks conducted in the plant shall require the applicant to enter the radiologically controlled area. This provides an excellent opportunity for the applicant to discuss or demonstrate radiation control administrative subjects.

   If it is not possible to develop or locate a suitable task for a selected system, return to step 2 and select a different system.

4. For each JPM, specify the codes for type, source, and location:

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

Reactor Operator

Job Performance Measures

a. JPM 708A **Title:** Inject Boron during an ATWS per EOI Appendix-3A, SLC Injection

Description: The candidate will inject Standby Liquid Control (SLC) in accordance with EOI Appendix 3A. When an SLC Pump is started, Reactor Water Cleanup (RWCU) will fail to isolate and the RWCU Pumps will fail to trip. The candidate will manually isolate RWCU and verify the RWCU Pumps are tripped.

K/A: 211000 Standby Liquid Control System A1.08; Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: RWCU system lineup (3.9)

b. JPM 751A **Title:** Align RCIC Suction to the Suppression Pool per EOI EOI-Appendix-5C, Injection System Lineup RCIC

Description: The candidate will switch RCIC suction sources in accordance with EOI-Appendix-5C. The RCIC Condensate Storage Tank (CST) Suction Valve (FCV-71-19) will not automatically close, requiring candidate action to close the valve.

K/A 217000 RCIC Reactor Core Isolation Cooling A4.03; Ability to manually operate and/or monitor in the Control Room: System Valves (3.8)

c. JPM 627A **Title:** Place HPCI in Reactor Pressure Control per EOI-Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode

Description: The candidate will place HPCI in Pressure Control in accordance EOI-Appendix-11C. When HPCI is in Reactor Pressure Control mode, a steam leak will develop and HPCI Steam Valves will fail to isolate. The candidate will manually close the HPCI Steam Valves.

K/A 206000 HPCI High-Pressure Coolant Injection System (BWR 2, 3, 4) A2.10; Ability to (a) predict the impacts of the following on the High-Pressure Coolant Injection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System isolation (4.7)

d.	JPM 750	**Title:** Vent the Drywell per OI-64, Primary Containment Systems
		Description: The candidate will vent the Drywell in accordance with OI-64, Primary Containment Systems in order to reduce Drywell Pressure.
	K/A	223001 PCS Primary Containment System and Auxiliaries A2.07; Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High Drywell Pressure (4.4)
e.	JPM 725	**Title:** Transfer ‘C’ 4KV Unit Board from the Start Bus to The USST In Accordance With 0-OI-57A, Switchyard And 4160V AC Electrical System
		Description: The candidate will transfer ‘C’ 4KV Unit Board from the Start Bus to the USST in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System.
	K/A	262001 A.C. Electrical Distribution A4.01: Ability to manually operate and/or monitor in the Control Room: Breakers and disconnects (3.7)
f.	JPM 290A	**Title:** Perform SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test
		Description: The candidate will perform SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test. When Reactor Protection System (RPS) B is tested, two (2) Control Rods will SCRAM, requiring the candidate to insert a manual Reactor SCRAM.
	K/A	212000 Reactor Protection System A1.04: Ability to predict and/or monitor changes in parameters associated with operation of the Reactor Protection System, including: RPS Bus Status (3.7)
g.	JPM 39	**Title:** Align CAD to Drywell Control Air per EOI Appendix-8G, Crosstie CAD to Drywell Control Air
		Description: The candidate will cross tie CAD to Drywell Control Air in accordance with EOI Appendix 8G.
	K/A	295019 Partial or Complete Loss of Instrument Air AA1.01: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Backup air supply (3.4)

h.	JPM 55A	**Title:** Emergency Vent Primary Containment per EOI-Appendix-13, Emergency Venting Primary Containment **Description:** The candidate will emergency vent Primary Containment in accordance with EOI-Appendix-13. The Suppression Chamber vent lineup will fail, and an alternate vent method must be used.
	K/A	288000 PVS 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 operations: High Drywell Pressure (3.5)
i.	JPM 754	**Title:** Align Components per AOI-100-2, Control Room Abandonment, Attachment 3, Part B **Description:** The candidate will perform field operations necessary to align breakers on switchboards as required during Control Room Abandonment in accordance with AOI-100-2, Control Room Abandonment, Attachment 3, Part B.
	K/A	295016 Control Room Abandonment AA1.04; Ability to operate and/or monitor the following as they apply to Control Room Abandonment: AC Electrical Distribution. (3.6)
j.	JPM 314	**Title:** Perform EOI-Appendix-2, Defeating ARI Logic Trips **Description:** The candidate will perform field operations necessary to defeat Alternate Rod Insertion (ARI) trips in accordance with EOI-Appendix-2, Defeating ARI Logic Trips.
	K/A	295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA1.01; Ability to operate and/or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Reactor protection system. (4.2)

k. JPM 755 **Title:** Perform 1-EOI-APPENDIX-16L, Bypassing HPCI High Temperature Isolation

Description: The candidate will perform field operations necessary to bypass High-Pressure Coolant Injection (HPCI) high temperature isolation signals in accordance with EOI-Appendix-16L, Bypassing HPCI High Temperature Isolation.

K/A 206000 HPCI High-Pressure Coolant Injection System A2.10; Ability to (a) predict the impacts of the following on the High-Pressure Coolant Injection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System isolation.

Facility: Browns Ferry NPP
Date of Examination: 5/16/22
Exam Level: ☐ RO ☒ SRO-I ☐ SRO-U
Operating Test No.: 22-04

Control Room Systems			
System / JPM Title		**Type Code***	**Safety Function**
a. JPM 708A	Inject Boron during an ATWS per EOI-Appendix-3A, SLC Injection	A, D, EN, S	1
b. JPM 751A	Align RCIC Suction to the Suppression Pool per EOI-Appendix-5C, Injection System Lineup RCIC	A, L, N, S	2
c. JPM 627A	Place HPCI in Reactor Pressure Control per EOI-Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode	A, D, EN, L, S	4
d. JPM 750	Vent the Drywell per OI-64, Primary Containment System	EN, N, S	5
e. JPM 725	Transfer 'C' 4KV Unit Board from the Start Bus to The USST Per 0-OI-57A, Switchyard And 4160V AC Electrical System	D, P, S	6
f. JPM 290A	Perform SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test	A, D, EN, P, S	7
g. N/A			
h. JPM 55A	Emergency Vent Primary Containment per EOI Appendix-13, Emergency Venting Primary Containment	A, D, EN, L, S	9
In-Plant Systems			
i. JPM 754	Align Components per AOI-100-2, Control Room Abandonment, Attachment 3, Part B	E, L, N, R	6
j. JPM 314	Defeat ARI Logic Trips per EOI Appendix-2, Defeating ARI Logic Trips	D, E, L	7
k. JPM 755	Bypass HPCI High Temperature Isolation per EOI Appendix-16L, Bypassing HPCI High Temperature Isolation	E, L, N	2

1. Determine the number of control room system and in-plant system job performance measures (JPMs) to develop using the following table:

License Level	Control Room	In-Plant	Total
Reactor Operator (RO)	8	3	11
Senior Reactor Operator-Instant (SRO-I)	7	3	10
Senior Reactor Operator-Upgrade (SRO-U)	2 or 3	3 or 2	5

2. Select safety functions and systems for each JPM as follows:

Refer to Section 1.9 of the applicable knowledge and abilities (K/A) catalog for the plant systems organized by safety function. For pressurized-water reactor operating tests, the primary and secondary systems listed under Safety Function 4, “Heat Removal from Reactor Core,” in Section 1.9 of the applicable K/A catalog, may be treated as separate safety functions (i.e., two systems, one primary and one secondary, may be selected from Safety Function 4). From the safety function groupings identified in the K/A catalog, select the appropriate number of plant systems by safety functions to be evaluated based on the applicant’s license level (see the table in step 1).
For RO/SRO-I applicants: Each of the control room system JPMs and, separately, each of the in-plant system JPMs must evaluate a different safety function, and the same system or evolution cannot be used to evaluate more than one safety function in each location. One of the control room system JPMs must be an engineered safety feature.
For SRO-U applicants: Evaluate SRO-U applicants on five different safety functions. One of the control room system JPMs must be an engineered safety feature, and the same system or evolution cannot be used to evaluate more than one safety function.

3. Select a task for each JPM that supports, either directly or indirectly and in a meaningful way, the successful fulfillment of the associated safety function. Select the task from the applicable K/A catalog (K/As for plant systems or emergency and abnormal plant evolutions) or the facility licensee’s site-specific task list. If this task has an associated K/A, the K/A should have an importance rating of at least 2.5 in the RO column. K/As that have importance ratings of less than 2.5 may be used if justified based on plant priorities; inform the NRC chief examiner if selecting K/As with an importance rating less than 2.5.

The selected tasks must be different from the events and evolutions conducted during the simulator operating test and tasks tested on the written examination. A task that is similar to a simulator scenario event may be acceptable if the actions required to complete the task are significantly different from those required in response to the scenario event.

Apply the following specific task selection criteria:

- At least one of the tasks shall be related to a shutdown or low-power condition.
- Four to six of the tasks for RO and SRO-I applicants shall require execution of alternative paths within the facility licensee’s operating procedures. Two to three of the tasks for SRO-U applicants shall require execution of alternative paths within the facility licensee’s operating procedures. At least one alternate path JPM must be new or modified from the bank.
- At least one of the tasks conducted in the plant shall evaluate the applicant’s ability to implement actions required during an emergency or abnormal condition.
- At least one of the tasks conducted in the plant shall require the applicant to enter the radiologically controlled area. This provides an excellent opportunity for the applicant to discuss or demonstrate radiation control administrative subjects.

If it is not possible to develop or locate a suitable task for a selected system, return to step 2 and select a different system.

4. For each JPM, specify the codes for type, source, and location:

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

Senior Reactor Operator (Instant)

Job Performance Measures

a. JPM 708A **Title:** Inject Boron during an ATWS per EOI Appendix-3A, SLC Injection

Description: The candidate will inject Standby Liquid Control (SLC) in accordance with EOI Appendix 3A. When an SLC Pump is started, Reactor Water Cleanup (RWCU) will fail to isolate and the RWCU Pumps will fail to trip. The candidate will manually isolate RWCU and verify the RWCU Pumps are tripped.

K/A: 211000 Standby Liquid Control System A1.08; Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: RWCU system lineup (3.9)

b. JPM 751 **Title:** Align RCIC Suction to the Suppression Pool per EOI EOI-Appendix-5C, Injection System Lineup RCIC

Description: The candidate will switch RCIC suction sources in accordance with EOI-Appendix-5C. The RCIC Condensate Storage Tank (CST) Suction Valve (FCV-71-19) will not automatically close, requiring candidate action to close the valve.

K/A 217000 RCIC Reactor Core Isolation Cooling A4.03; Ability to manually operate and/or monitor in the Control Room: System Valves (3.8)

c. JPM 627A **Title:** Place HPCI in Reactor Pressure Control per EOI-Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode

Description: The candidate will place HPCI in pressure Control in accordance EOI-Appendix-11C. When HPCI is in Reactor Pressure Control mode, a steam leak will develop and HPCI Steam Valves will fail to isolate. The candidate will manually close the HPCI Steam Valves.

K/A 206000 HPCI High-Pressure Coolant Injection System (BWR 2, 3, 4) A2.10; Ability to (a) predict the impacts of the following on the High-Pressure Coolant Injection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System isolation (4.0)

d. JPM 750 **Title:** Vent the Drywell per OI-64, Primary Containment Systems

Description: The candidate will vent the Drywell in accordance with OI-64, Primary Containment Systems in order to reduce Drywell Pressure.

K/A 223001 PCS Primary Containment System and Auxiliaries A2.07; Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High Drywell Pressure (4.3)

e. JPM 725 **Title:** Transfer ‘C’ 4KV Unit Board from the Start Bus to The USST In Accordance With 0-OI-57A, Switchyard And 4160V AC Electrical System

Description: The candidate will transfer ‘C’ 4KV Unit Board from the Start Bus to the USST in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System.

K/A 262001 A.C. Electrical Distribution A4.01: Ability to manually operate and/or monitor in the Control Room: Breakers and disconnects (3.7)

f. JPM 290A **Title:** Perform SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test

Description: The candidate will perform SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test. When Reactor Protection System (RPS) B is tested, two (2) Control Rods will SCRAM, requiring the candidate to insert a manual Reactor SCRAM.

K/A 212000 Reactor Protection System A1.04: Ability to predict and/or monitor changes in parameters associated with operation of the Reactor Protection System, including: RPS Bus Status (3.7)

g. N/A

h. JPM 55A **Title:** Emergency Vent Primary Containment per EOI-Appendix-13, Emergency Venting Primary Containment

Description: The candidate will emergency vent Primary Containment in accordance with EOI-Appendix-13. The Suppression Chamber vent lineup will fail, and an alternate vent method must be used.

K/A 288000 PVS 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 operations: High Drywell Pressure (3.6)

i. JPM 754 **Title:** Align Components per AOI-100-2, Control Room Abandonment, Attachment 3, Part B

Description: The candidate will perform field operations necessary to align breakers on switchboards as required during Control Room Abandonment in accordance with AOI-100-2, Control Room Abandonment, Attachment 3, Part B.

K/A 295016 Control Room Abandonment AA1.04; Ability to operate and/or monitor the following as they apply to Control Room Abandonment: AC Electrical Distribution. (3.6)

j. JPM 314 **Title:** Perform EOI-Appendix-2, Defeating ARI Logic Trips

Description: The candidate will perform field operations necessary to defeat Alternate Rod Insertion (ARI) trips in accordance with EOI-Appendix-2, Defeating ARI Logic Trips.

K/A 295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA1.01; Ability to operate and/or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Reactor protection system. (4.2)

k.	JPM 755	**Title:** Perform 1-EOI-APPENDIX-16L, Bypassing HPCI High Temperature Isolation
		Description: The candidate will perform field operations necessary to bypass High-Pressure Coolant Injection (HPCI) high temperature isolation signals in accordance with EOI-Appendix-16L, Bypassing HPCI High Temperature Isolation.
	K/A	206000 HPCI High-Pressure Coolant Injection System A2.10; Ability to (a) predict the impacts of the following on the High-Pressure Coolant Injection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System isolation. (4.0)

Facility: Browns Ferry NPP

Date of Examination: 5/16/22

Exam Level: ☐ RO ☐ SRO-I ☒ SRO-U

Operating Test No.: 22-04

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 708A	Inject Boron during an ATWS per EOI-Appendix-3A, SLC Injection	A, D, EN, S	1
b. JPM 751A	Align RCIC Suction to the Suppression Pool per EOI-Appendix-5C, Injection System Lineup RCIC	A, L, N, S	2
c. JPM 627A	Place HPCI in Reactor Pressure Control per EOI-Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode	A, D, EN, L, S	4
d. N/A			
e. N/A			
f. N/A			
g. N/A			
h. N/A			
In-Plant Systems: @ **3 for RO, 3 for SRO-I, 3 or 2 for SRO-U**			
i. JPM 754	Align Components per AOI-100-2, Control Room Abandonment, Attachment 3, Part B	E, L, N, R	6
j. JPM 314	Defeat ARI Logic Trips per EOI Appendix-2, Defeating ARI Logic Trips	D, E, L	7
k. N/A			

1. Determine the number of control room system and in-plant system job performance measures (JPMs) to develop using the following table:

License Level	**Control Room**	**In-Plant**	**Total**
Reactor Operator (RO)	8	3	11
Senior Reactor Operator-Instant (SRO-I)	7	3	10
Senior Reactor Operator-Upgrade (SRO-U)	2 or 3	3 or 2	5

2. Select safety functions and systems for each JPM as follows:

   Refer to Section 1.9 of the applicable knowledge and abilities (K/A) catalog for the plant systems organized by safety function. For pressurized-water reactor operating tests, the primary and secondary systems listed under Safety Function 4, “Heat Removal from Reactor Core,” in Section 1.9 of the applicable K/A catalog, may be treated as separate safety functions (i.e., two systems, one primary and one secondary, may be selected from Safety Function 4). From the safety function groupings identified in the K/A catalog, select the appropriate number of plant systems by safety functions to be evaluated based on the applicant’s license level (see the table in step 1).

   For RO/SRO-I applicants: Each of the control room system JPMs and, separately, each of the in-plant system JPMs must evaluate a different safety function, and the same system or evolution cannot be used to evaluate more than one safety function in each location. One of the control room system JPMs must be an engineered safety feature.

   For SRO-U applicants: Evaluate SRO-U applicants on five different safety functions. One of the control room system JPMs must be an engineered safety feature, and the same system or evolution cannot be used to evaluate more than one safety function.

3. Select a task for each JPM that supports, either directly or indirectly and in a meaningful way, the successful fulfillment of the associated safety function. Select the task from the applicable K/A catalog (K/As for plant systems or emergency and abnormal plant evolutions) or the facility licensee’s site-specific task list. If this task has an associated K/A, the K/A should have an importance rating of at least 2.5 in the RO column. K/As that have importance ratings of less than 2.5 may be used if justified based on plant priorities; inform the NRC chief examiner if selecting K/As with an importance rating less than 2.5.
   The selected tasks must be different from the events and evolutions conducted during the simulator operating test and tasks tested on the written examination. A task that is similar to a simulator scenario event may be acceptable if the actions required to complete the task are significantly different from those required in response to the scenario event.

   Apply the following specific task selection criteria:

   - At least one of the tasks shall be related to a shutdown or low-power condition.
   - Four to six of the tasks for RO and SRO-I applicants shall require execution of alternative paths within the facility licensee’s operating procedures. Two to three of the tasks for SRO-U applicants shall require execution of alternative paths within the facility licensee’s operating procedures. At least one alternate path JPM must be new or modified from the bank.
   - At least one of the tasks conducted in the plant shall evaluate the applicant’s ability to implement actions required during an emergency or abnormal condition.
   - At least one of the tasks conducted in the plant shall require the applicant to enter the radiologically controlled area. This provides an excellent opportunity for the applicant to discuss or demonstrate radiation control administrative subjects.

   If it is not possible to develop or locate a suitable task for a selected system, return to step 2 and select a different system.

4. For each JPM, specify the codes for type, source, and location:

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

Senior Reactor Operator (Upgrade)

Job Performance Measures

a. JPM 708A **Title:** Inject Boron during an ATWS per EOI Appendix-3A, SLC Injection

Description: The candidate will inject Standby Liquid Control (SLC) in accordance with EOI Appendix 3A. When an SLC Pump is started, Reactor Water Cleanup (RWCU) will fail to isolate and the RWCU Pumps will fail to trip. The candidate will manually isolate RWCU and verify the RWCU Pumps are tripped.

K/A: 211000 Standby Liquid Control System A1.08; Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: RWCU system lineup (3.9)

b. JPM 751A **Title:** Align RCIC Suction to the Suppression Pool per EOI EOI-Appendix-5C, Injection System Lineup RCIC

Description: The candidate will switch RCIC suction sources in accordance with EOI-Appendix-5C. The RCIC Condensate Storage Tank (CST) Suction Valve (FCV-71-19) will not automatically close, requiring candidate action to close the valve.

K/A 217000 RCIC Reactor Core Isolation Cooling A4.03: Ability to manually operate and/or monitor in the Control Room: System Valves (3.8)

c. JPM 627A **Title:** Place HPCI in Reactor Pressure Control per EOI-Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode

Description: The candidate will place HPCI in pressure Control in accordance EOI-Appendix-11C. When HPCI is in Reactor Pressure Control mode, a steam leak will develop and HPCI Steam Valves will fail to isolate. The candidate will manually close the HPCI Steam Valves.

K/A 206000 HPCI High-Pressure Coolant Injection System (BWR 2, 3, 4) A2.10: Ability to (a) predict the impacts of the following on the High-Pressure Coolant Injection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System isolation (4.0)

d. N/A

e. N/A

f. N/A

g.	N/A	
h.	N/A	
i.	JPM 754	**Title:** Align Components per AOI-100-2, Control Room Abandonment, Attachment 3, Part B
		Description: The candidate will perform field operations necessary to align breakers on switchboards as required during Control Room Abandonment in accordance with AOI-100-2, Control Room Abandonment, Attachment 3, Part B.
	K/A	295016 Control Room Abandonment AA1.04; Ability to operate and/or monitor the following as they apply to Control Room Abandonment: AC Electrical Distribution. (3.6)
j.	JPM 314	**Title:** Perform EOI-Appendix-2, Defeating ARI Logic Trips
		Description: The candidate will perform field operations necessary to defeat Alternate Rod Insertion (ARI) trips in accordance with EOI-Appendix-2, Defeating ARI Logic Trips.
	K/A	295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA1.01; Ability to operate and/or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Reactor protection system. (4.2)
k.	N/A	

Examination Outline Cross-reference:	Level	RO	SRO
286000 (SF8 FPS) Fire Protection	Tier #	2	------
K2.02 (10CFR 55.41.7)	Group #	2	------
Knowledge of the electrical power supplies to the following:	K/A #	286000K2.02	
- Fire Pumps	Importance Rating	3.2	------

Proposed Question: **# 1**

4 KV Shutdown Board ___________ is the **NORMAL** power supply to the 'A' Fire Pump.

A. A

B. B

C. C

D. D

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with system drawing 0-45E724-1, 4160V SHUTDOWN BOARD A, and 0-OI-26, High Pressure Fire Protection System, Attachment 3 (Electrical Lineup Checklist) Fire Pump 'A' is powered by 4KV Shutdown Board 'A'.

B INCORRECT: Incorrect but plausible in that the Electrical Distribution System at Browns Ferry is complex and confusing. There are four (4) 4KV Shutdown Boards listed as potential power supplies, and any board could power 'A' Fire Pump. There are numerous instances of cross-connected power supplies at Browns Ferry which adds to the complexity of the question.

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

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

RO Level Justification: Tests the candidate's knowledge of the electrical power supply to High Pressure Fire Pumps. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): 2-OI-26, ATT 3, Rev. 95 (Attach if not previously provided)
0-45E724-1, Rev. 44

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.049 Obj. 9d_ (As available)

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

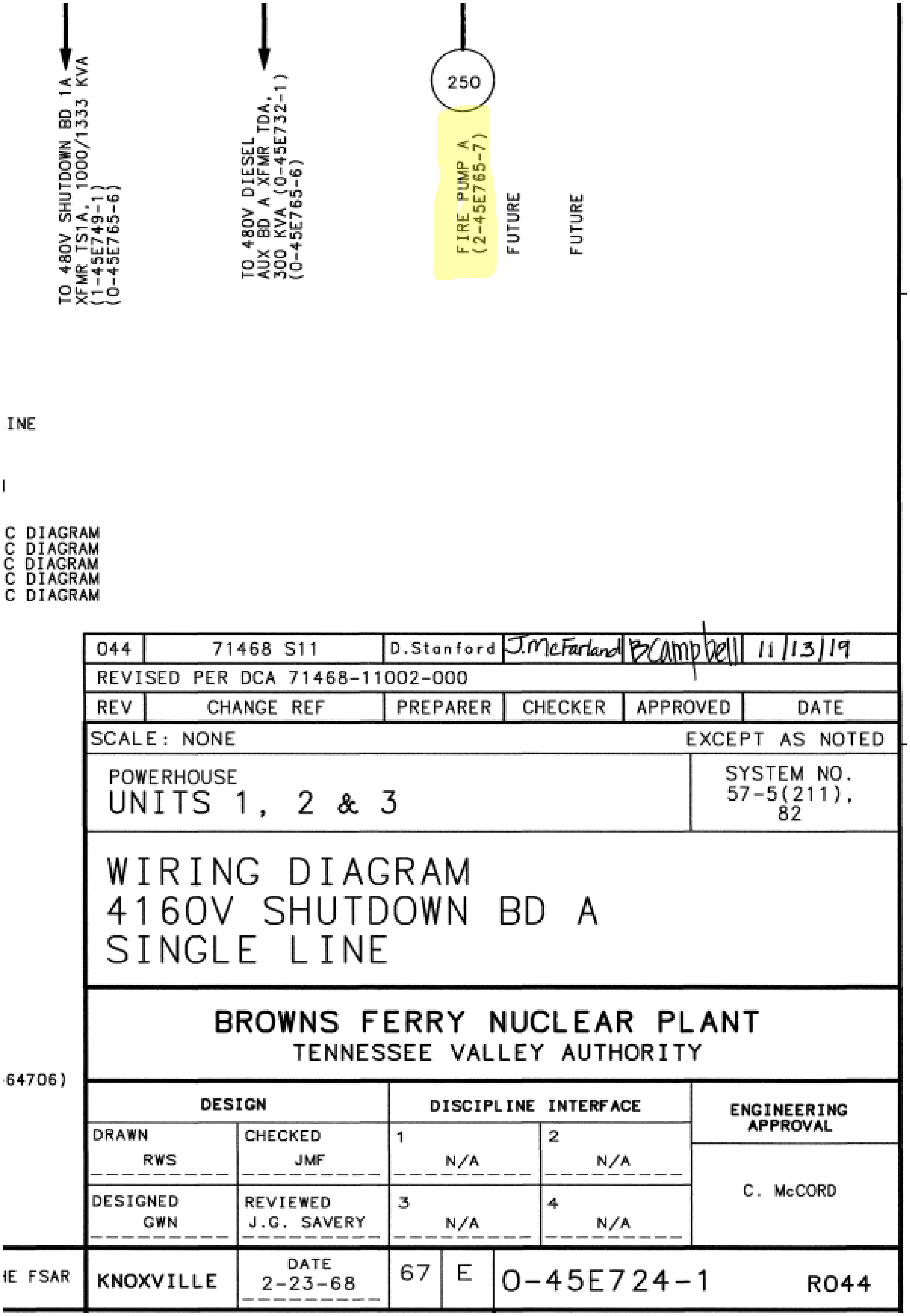
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Excerpt from Drawing 0-45E724-1:



REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE
044	71468 S11	D.Stanford	J.McFarland	BCampbell	11/13/19

REVISED PER DCA 71468-11002-000

SCALE: NONE EXCEPT AS NOTED

POWERHOUSE
UNITS 1, 2 & 3

SYSTEM NO.
57-5(211),
82

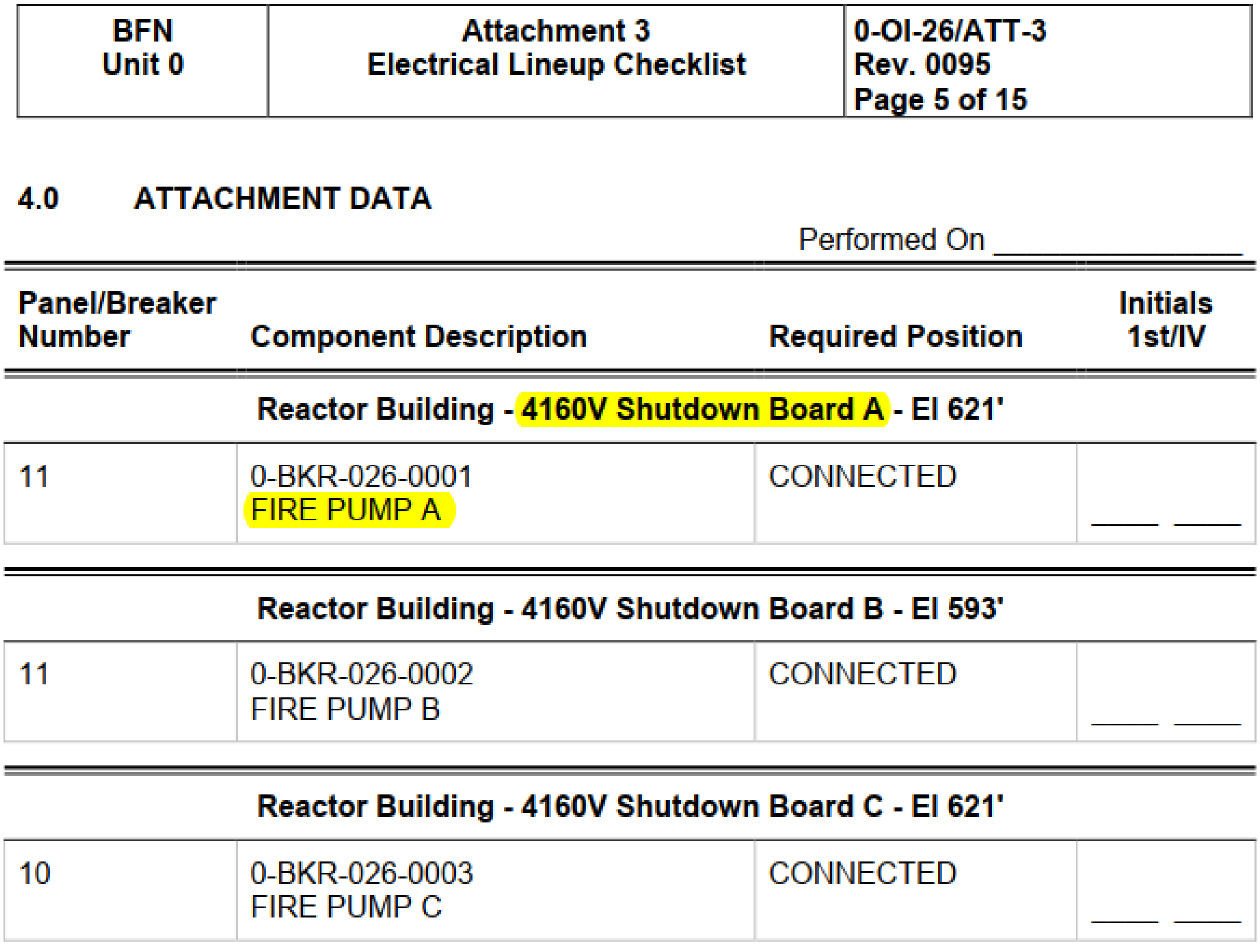
WIRING DIAGRAM
4160V SHUTDOWN BD A
SINGLE LINE

BROWNS FERRY NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY

DESIGN		DISCIPLINE INTERFACE		ENGINEERING APPROVAL
DRAWN RWS	CHECKED JMF	1 N/A	2 N/A	C. McCORD
DESIGNED GWN	REVIEWED J.G. SAVERY	3 N/A	4 N/A	

KNOXVILLE | DATE 2-23-68 | 67 | E | 0-45E724-1 | R044

Excerpt from 0-OI-26, Attachment 3:



BFN **Unit 0**	**Attachment 3** **Electrical Lineup Checklist**	**0-OI-26/ATT-3** **Rev. 0095** **Page 5 of 15**

4.0 ATTACHMENT DATA

Performed On ________________

Panel/Breaker Number	**Component Description**	**Required Position**	**Initials 1st/IV**
	Reactor Building - 4160V Shutdown Board A - El 621'		
11	0-BKR-026-0001 FIRE PUMP A	CONNECTED	____ ____
	Reactor Building - 4160V Shutdown Board B - El 593'		
11	0-BKR-026-0002 FIRE PUMP B	CONNECTED	____ ____
	Reactor Building - 4160V Shutdown Board C - El 621'		
10	0-BKR-026-0003 FIRE PUMP C	CONNECTED	____ ____

Examination Outline Cross-reference:

295016 (APE 16) Control Room Abandonment /7

A1.01 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT:

- RPS

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295016 AA1.01	
Importance Rating	3.8	------

Proposed Question: **# 2**

Unit 3 is operating at 100% RTP when the following plant conditions occur:

- Control Room abandonment has been directed due to toxic gas in the Control Room

In accordance with 3-AOI-100-2, Control Room Abandonment, the **IMMEDIATE ACTION** to insert a Reactor SCRAM will be performed from Panel ________.

A. 3-9-5

B. 3-9-15

C. 3-9-17

D. 3-25-32

Proposed Answer: **A**

Explanation (Optional):		
	A	**CORRECT**: (*See attached*) In accordance with 3-AOI-100-2, Control Room Abandonment, the IMMEDIATE ACTION to DEPRESS REACTOR SCRAM A and B pushbuttons from Panel 3-9-5 is required to be completed prior to evacuating the Control Room.
	B	INCORRECT: Incorrect but plausible in that 3-AOI-100-2, Attachment 9 – Removal and Replacement of RPS SCRAM Solenoid Fuses, states RPS Bus A fuses are removed from Panel 3-9-15 located in Unit 3 Aux Instrument Room.
	C	INCORRECT: Incorrect but plausible in that 3-AOI-100-2, Attachment 9 – Removal and Replacement of RPS SCRAM Solenoid Fuses, states RPS Bus B fuses are removed from Panel 3-9-17 located in Unit 3 Aux Instrument Room.
	D	INCORRECT: Incorrect but plausible in that 3-AOI-100-2, lists the last IMMEDIATE ACTION to proceed to Backup Control Panel 3-25-32 which is where numerous subsequent actions will occur.

RO Level Justification: Tests the candidate's knowledge and ability to operate the Reactor Protection System (RPS) during a SCRAM as it relates to Control Room Abandonment. This question is rated as memory due to strictly recalling facts related to Abnormal Operating Instructions.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

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

Technical Reference(s): 3-AOI-100-2, Rev. 27 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.074 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 **X**

Comprehension or Analysis

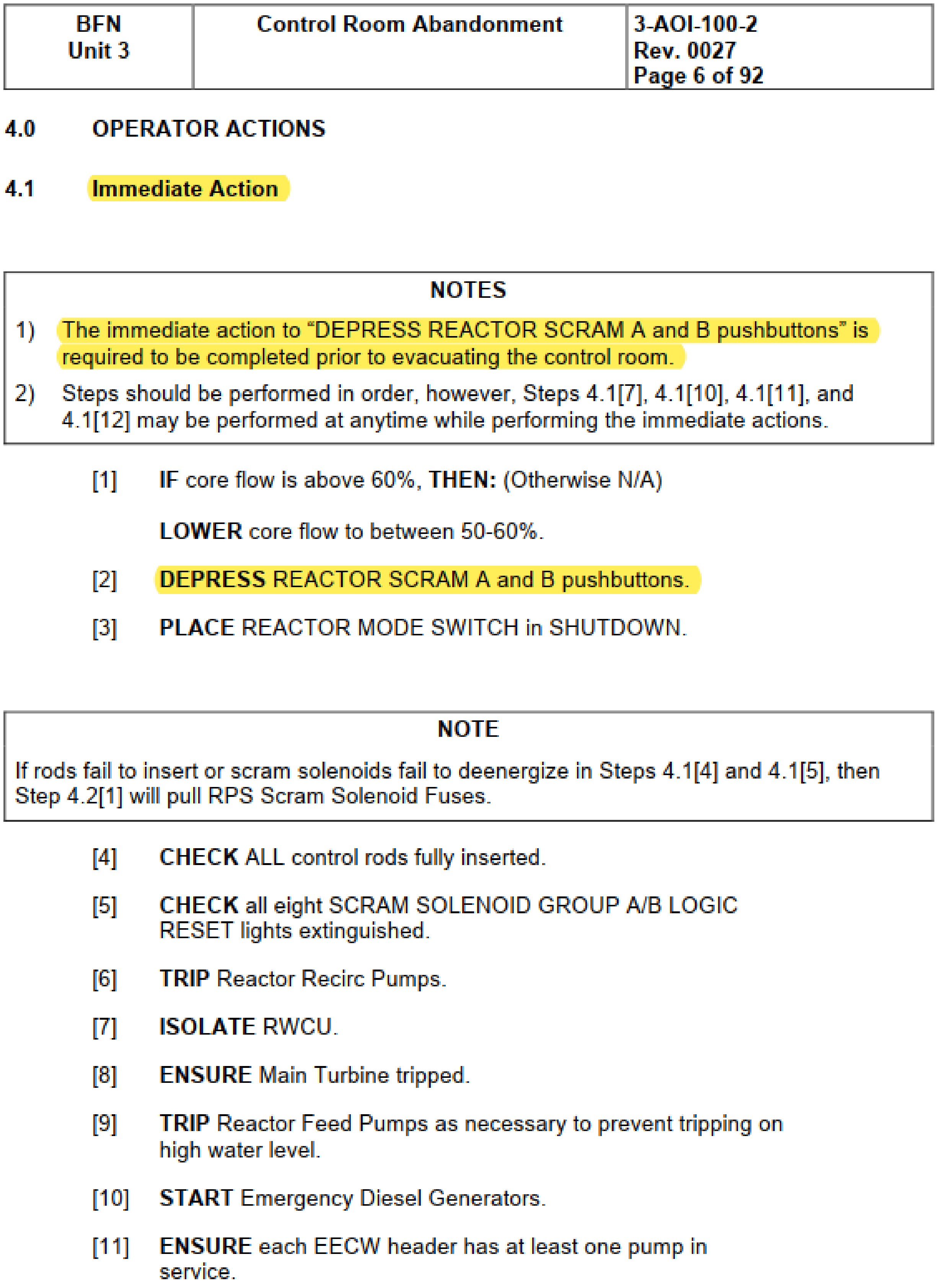
10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Excerpts from 3-AOI-100-2:



BFN **Unit 3**	**Control Room Abandonment**	**3-AOI-100-2** **Rev. 0027** **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.

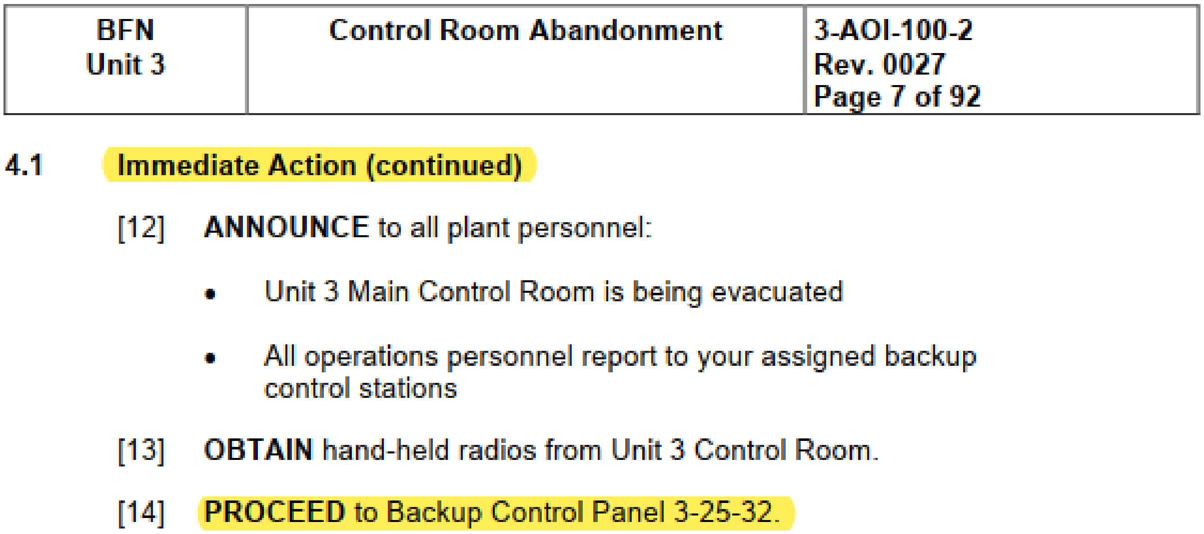
[8] **ENSURE** Main Turbine tripped.

[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.

Supports Distractor D:



BFN Unit 3	Control Room Abandonment	3-AOI-100-2 Rev. 0027 Page 7 of 92

4.1 Immediate Action (continued)

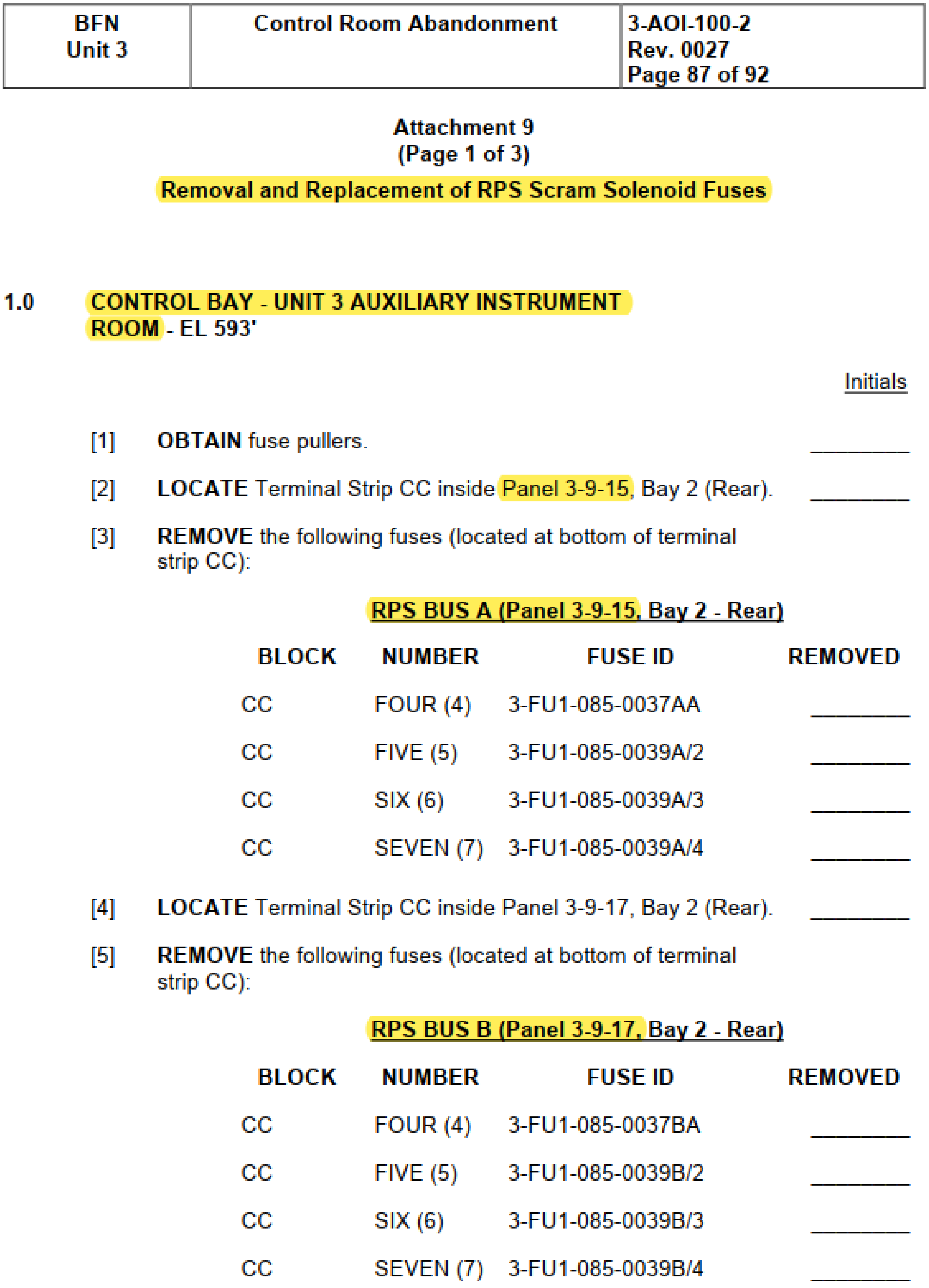
[12] **ANNOUNCE** to all plant personnel:

- Unit 3 Main Control Room is being evacuated
- All operations personnel report to your assigned backup control stations

[13] **OBTAIN** hand-held radios from Unit 3 Control Room.

[14] **PROCEED** to Backup Control Panel 3-25-32.

Supports Distractors B and C:



BFN **Unit 3**	**Control Room Abandonment**	**3-AOI-100-2** **Rev. 0027** **Page 87 of 92**

Attachment 9
(Page 1 of 3)
Removal and Replacement of RPS Scram Solenoid Fuses

1.0 CONTROL BAY - UNIT 3 AUXILIARY INSTRUMENT ROOM - EL 593'

Initials

[1] **OBTAIN** fuse pullers. _________

[2] **LOCATE** Terminal Strip CC inside Panel 3-9-15, Bay 2 (Rear). _________

[3] **REMOVE** the following fuses (located at bottom of terminal strip CC):

RPS BUS A (Panel 3-9-15, Bay 2 - Rear)

BLOCK	NUMBER	FUSE ID	REMOVED
CC	FOUR (4)	3-FU1-085-0037AA	_________
CC	FIVE (5)	3-FU1-085-0039A/2	_________
CC	SIX (6)	3-FU1-085-0039A/3	_________
CC	SEVEN (7)	3-FU1-085-0039A/4	_________

[4] **LOCATE** Terminal Strip CC inside Panel 3-9-17, Bay 2 (Rear). _________

[5] **REMOVE** the following fuses (located at bottom of terminal strip CC):

RPS BUS B (Panel 3-9-17, Bay 2 - Rear)

BLOCK	NUMBER	FUSE ID	REMOVED
CC	FOUR (4)	3-FU1-085-0037BA	_________
CC	FIVE (5)	3-FU1-085-0039B/2	_________
CC	SIX (6)	3-FU1-085-0039B/3	_________
CC	SEVEN (7)	3-FU1-085-0039B/4	_________

Examination Outline Cross-reference:

259006 (APE 6) SCRAM /1

AK3.01 (10CFR 55.41.5)

Knowledge of the reasons for the following responses or actions as they apply to SCRAM:

- Reactor Water Level Response

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	259006AK3.01	
Importance Rating	4.0	------

Proposed Question: **# 3**

Unit 2 was operating at 95% RTP with the following plant conditions:

- Reactor Feedwater Pump 2C has just been secured for maintenance

An event then occurs which results in an automatic Reactor SCRAM.

- Reactor Water Level reached (-) 15 inches before recovering to the normal band

Given the conditions above and in accordance with 2-OI-3, Reactor Feedwater System, Reactor Feedwater Pump (RFP), ____(1)____ will control Reactor Water Level in automatic while the other RFP Turbine will runback to ____(2)____ rpm.

A. **(1)** 2A
(2) 480

B. **(1)** 2A
(2) 600

C. **(1)** 2B
(2) 480

D. **(1)** 2B
(2) 600

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible if the candidate confuses the RFWCS SCRAM Response polling sequence to select RFP 2A. The second part is incorrect but plausible in that in accordance with 2-OI-68, Reactor Recirculation System, Recirc Pumps will automatically runback to 480 rpm based on a limiter setpoint of 28%.

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-3, Reactor Feedwater System, Attachment 5 – RFWCS SCRAM Response, given the conditions are met for the RFP control logic to initiate, the polling sequence is RFP 2C, 2B, 2A. Once one RFP in AUTO is found (normally RFP 2C), but in this case it is 2B RFP, the polled RFPT will have a low speed clamp applied to control Reactor Water Level in automatic to prevent overfill of the Reactor Vessel following a SCRAM. For second part, the remaining RFPT, 2A, is automatically transferred to MANUAL and a speed runback occurs to 600 rpm.

RO Level Justification: Tests the candidate's knowledge of the reasons for the Reactor Water Level response as it applies to a Reactor SCRAM. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s):	2-OI-3, Rev. 162	(Attach if not previously provided)
	2-OI-68, Rev. 164	

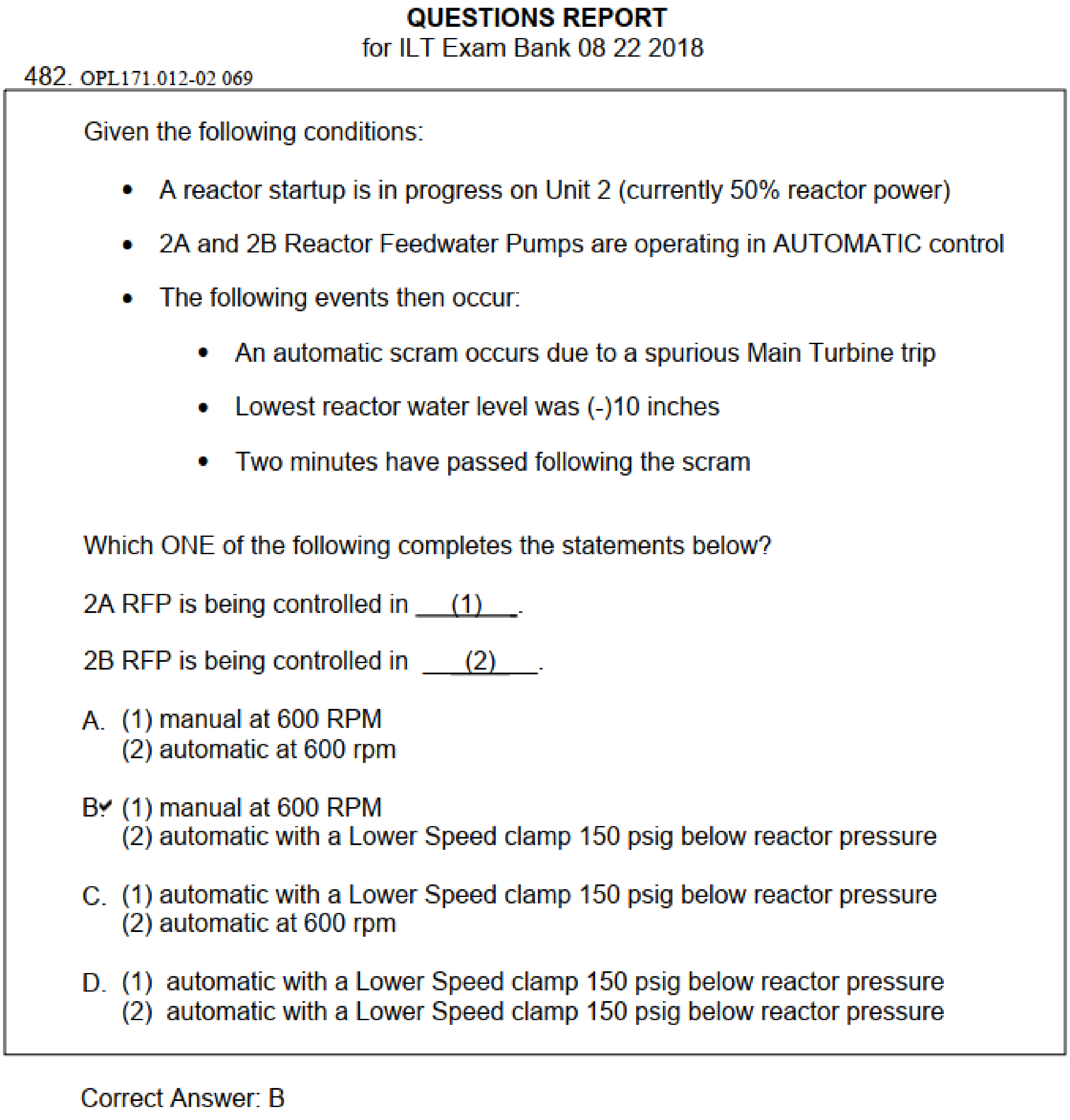
Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.012 Obj. 5d (As available)

Question Source:	Bank #		
	Modified Bank #	OPL171.012-02 069 #482	(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

482. OPL171.012-02 069

Given the following conditions:

- A reactor startup is in progress on Unit 2 (currently 50% reactor power)
- 2A and 2B Reactor Feedwater Pumps are operating in AUTOMATIC control
- The following events then occur:
  - An automatic scram occurs due to a spurious Main Turbine trip
  - Lowest reactor water level was (-)10 inches
  - Two minutes have passed following the scram

Which ONE of the following completes the statements below?

2A RFP is being controlled in __(1)__.

2B RFP is being controlled in __(2)__.

A. (1) manual at 600 RPM
(2) automatic at 600 rpm

B✓ (1) manual at 600 RPM
(2) automatic with a Lower Speed clamp 150 psig below reactor pressure

C. (1) automatic with a Lower Speed clamp 150 psig below reactor pressure
(2) automatic at 600 rpm

D. (1) automatic with a Lower Speed clamp 150 psig below reactor pressure
(2) automatic with a Lower Speed clamp 150 psig below reactor pressure

Correct Answer: B

Excerpt from 2-OI-3:

BFN **Unit 2**	**Reactor Feedwater System**	**2-OI-3** **Rev. 0162** **Page 275 of 303**

Attachment 5
(Page 2 of 3)
RFWCS Scram Response

DESCRIPTION:

The RFW Control System Scram Response logic is a software feature that prevents overfilling the Reactor vessel following a scram.

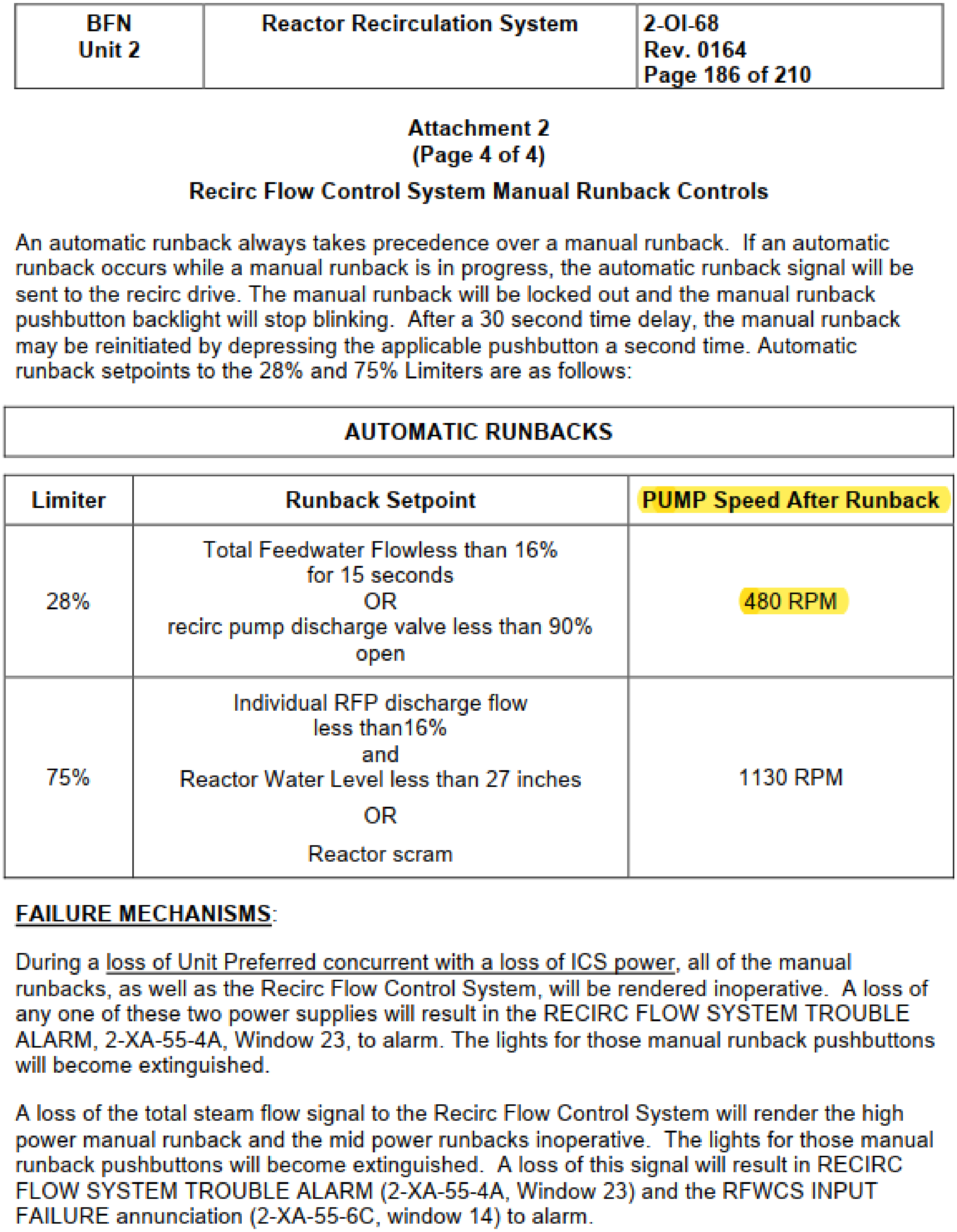
For the Scram Response logic to initiate, <u>all</u> of the following conditions must be met:

- Scram Response logic is not inhibited (yellow indicating light at inhibit handswitch extinguished, **REFER TO** Figure 5).
- Reactor Water Level Control PDS, 2-LIC-46-5 on Panel 2-9-5, is in AUTO <u>AND</u> at least one individual RFPT Speed Control PDS in AUTO.
- Either RPS A or B Backup Scram channel activates.
- Reactor water level (narrow range) falls below 0 inches.

When either backup scram relay energizes, after a 3 second time delay, RFW level control automatically transfers to single element. With Scram Response logic picked up, red indicating light at inhibit pushbutton (**REFER TO** Figure 5) will illuminate. Logic then polls RFP controls for availability of RFP in AUTO. The polling sequence is RFP 2C, 2B, and 2A.

Once one RFP in AUTO is found (normally RFP 2C), the polled RFPT will have a low speed clamp applied to it to prevent pump from backing down to 600 rpm. This clamp will track Reactor pressure and maintain the RFP discharge pressure approximately 150 psig below reactor pressure. As Reactor pressure changes, the speed of RFP changes to maintain this pressure difference. An upper speed clamp of 4100 rpm is also applied to this pump. The remaining pumps (normally RFP 2A and 2B) are transferred to MANUAL and RFPT speeds set at approximately 600 rpm. These pumps are still available for MANUAL or AUTOMATIC control to upper speed limit of 5800 rpm.

Excerpt from 2-OI-68:



BFN Unit 2	**Reactor Recirculation System**	**2-OI-68 Rev. 0164 Page 186 of 210**

Attachment 2
(Page 4 of 4)

Recirc Flow Control System Manual Runback Controls

An automatic runback always takes precedence over a manual runback. If an automatic runback occurs while a manual runback is in progress, the automatic runback signal will be sent to the recirc drive. The manual runback will be locked out and the manual runback pushbutton backlight will stop blinking. After a 30 second time delay, the manual runback may be reinitiated by depressing the applicable pushbutton a second time. Automatic runback setpoints to the 28% and 75% Limiters are as follows:

AUTOMATIC RUNBACKS

Limiter	**Runback Setpoint**	**PUMP Speed After Runback**
28%	Total Feedwater Flowless than 16% for 15 seconds OR recirc pump discharge valve less than 90% open	480 RPM
75%	Individual RFP discharge flow less than16% and Reactor Water Level less than 27 inches OR Reactor scram	1130 RPM

FAILURE MECHANISMS:

During a loss of Unit Preferred concurrent with a loss of ICS power, all of the manual runbacks, as well as the Recirc Flow Control System, will be rendered inoperative. A loss of any one of these two power supplies will result in the RECIRC FLOW SYSTEM TROUBLE ALARM, 2-XA-55-4A, Window 23, to alarm. The lights for those manual runback pushbuttons will become extinguished.

A loss of the total steam flow signal to the Recirc Flow Control System will render the high power manual runback and the mid power runbacks inoperative. The lights for those manual runback pushbuttons will become extinguished. A loss of this signal will result in RECIRC FLOW SYSTEM TROUBLE ALARM (2-XA-55-4A, Window 23) and the RFWCS INPUT FAILURE annunciation (2-XA-55-6C, window 14) to alarm.

Examination Outline Cross-reference:

215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor

A3.03 (10CFR 55.41.7)

Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including:

- Meters and recorders

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	215005A3.03	
Importance Rating	3.6	------

Proposed Question: **# 4**

Unit 2 is operating at 100% RTP.

Which **ONE** of the following plant transients will cause the APRM Operator Display Assemblies (ODAs) to **AUTOMATICALLY** display the Oscillation Power Range Monitor (OPRM) Bargraph Screens?

A. Recirc Pump 2A trip

B. Inadvertent HPCI injection

C. Control Rod drifting into the core

D. Feedwater Heater 2A1 Extraction Steam isolation

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with 2-OI-92B, Average Power Range Monitoring, there are a total of 4 Operator Display Assemblies (ODAs, 2 for the APRM (1 and 3, 2 and 4)/OPRMs and 2 for the Rod Block Monitor (RBM A and B) System on Panel 2-9-5. Each APRM ODA is shared by 2 APRM/OPRMs. Also, there are additional operator displays on Panel 2-9-14 divided into 3 sections for each respective channel of ARPMs/OPRMs and RBMs. Upon a Recirc Pump trip, the ODA OPRM bargraph screens will be automatically displayed on the ODA for both APRMs when either APRM enters the Power-to-Flow map region (0-TI-248) where instability can occur as defined by the OPRM trip enabled setpoint. The candidate must know what causes the APRM ODA to shift to the OPRM bargraph screens especially since 7 different parameters or signal/setpoints are transmitted to each ODA from the associated APRM instruments.

B INCORRECT: Incorrect but plausible in that a cold water injection from an inadvertent HPCI injection could damage fuel, but does not cause instabilities. This will cause Reactor Power to rise and illustrated as moving up on the Power-to-Flow map.

C INCORRECT: Incorrect but plausible in that a Control Rod insertion moves Reactor Power straight down the Power-to-Flow map and not toward the OPRM enabled region.

D INCORRECT: Incorrect but plausible in that this will cause Reactor Power to rise and illustrated as moving up on the Power-to-Flow map.

RO Justification: Tests the candidate's ability to monitor automatic operations of Average Power Range Monitor System as it relates to indications in the Main Control Room. This question is rated as Memory due to the requirement to strictly recall facts related to operation of the APRMs.

Technical Reference(s): 2-OI-92B, Rev. 43 (Attach if not previously provided)
0-TI-248, Rev. 122
OPL171.148, Rev. 18

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.148, Obj. 13 (As available)

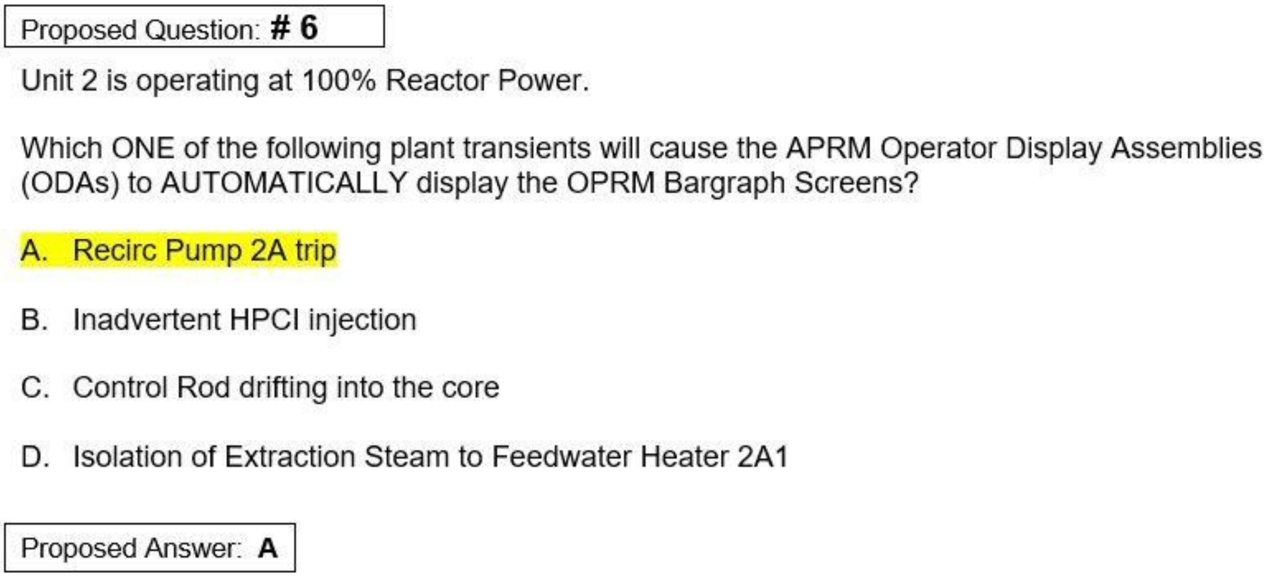
Question Source:	Bank #	BFN 0801 AUDIT #6
	Modified Bank #	
	New	
Question History:	Last NRC Exam	

(Note changes or attach parent)

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

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

Copy of Bank Question:



Proposed Question: **# 6**

Unit 2 is operating at 100% Reactor Power.

Which ONE of the following plant transients will cause the APRM Operator Display Assemblies (ODAs) to AUTOMATICALLY display the OPRM Bargraph Screens?

A. Recirc Pump 2A trip

B. Inadvertent HPCI injection

C. Control Rod drifting into the core

D. Isolation of Extraction Steam to Feedwater Heater 2A1

Proposed Answer: **A**

Excerpts from OPL171.148 Lesson Plan:

OPL171.148, POWER RANGE NUCLEAR MONITORING SYSTEM, REV.# 18

i. APRM Instrument (Panel 9-14)

   1) A description of the operation of the displays associated with APRMs is located in Appendix D.

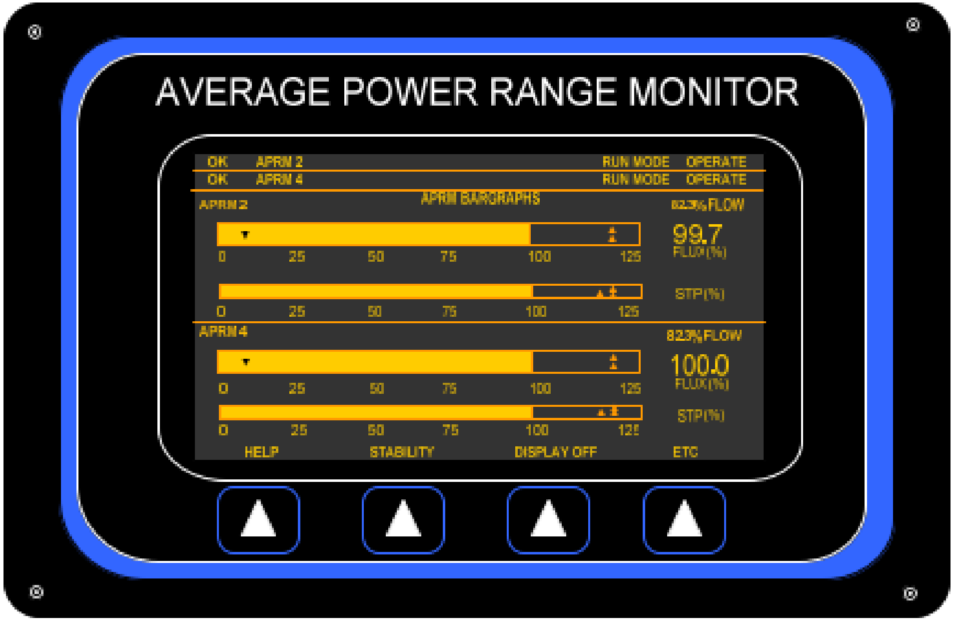


Figure 10 - Operator Display Assembly

4. Operator Display Assembly (ODA) indication

   a. The Operator Display Assembly (ODA) on Main Control Room Panel 9-5 provides APRM, LPRM, OPRM, RBM, and Recirculation flow indications for the operators.

Right Unit/Train/Component

   b. There a total of 4 ODAs, 2 for the APRMs and 1 each per RBM channel.

   c. Each APRM ODA is shared by 2 APRMs.

      1) APRMs 1 and 3 input to one ODA, APRMs 2 and 4 input to the other ODA.

   d. The APRM ODA provides indication only.

OPL171.148, POWER RANGE NUCLEAR MONITORING SYSTEM, REV.# 18

e. The following information is transmitted to the ODA from associated APRM instruments:

1) Average neutron flux

2) Simulated thermal power

3) Total flow signal

4) LPRM detector signals

5) Upscale and downscale trips and set points

6) Instrument status (i.e., Instrument ID and Mode, Self-Test, Bypass)

7) Oscillation Power Range Monitor displays

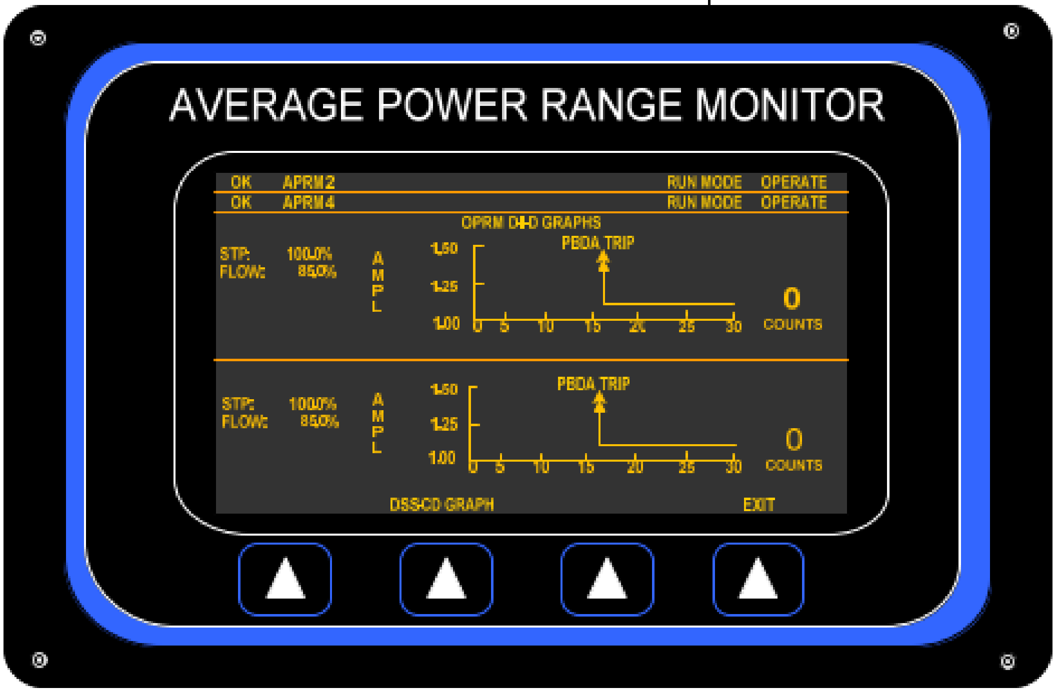


Figure 11 - Operator Display Assembly - OPRM D-I-D Graphs

OPL171.148, POWER RANGE NUCLEAR MONITORING SYSTEM, REV.# 18

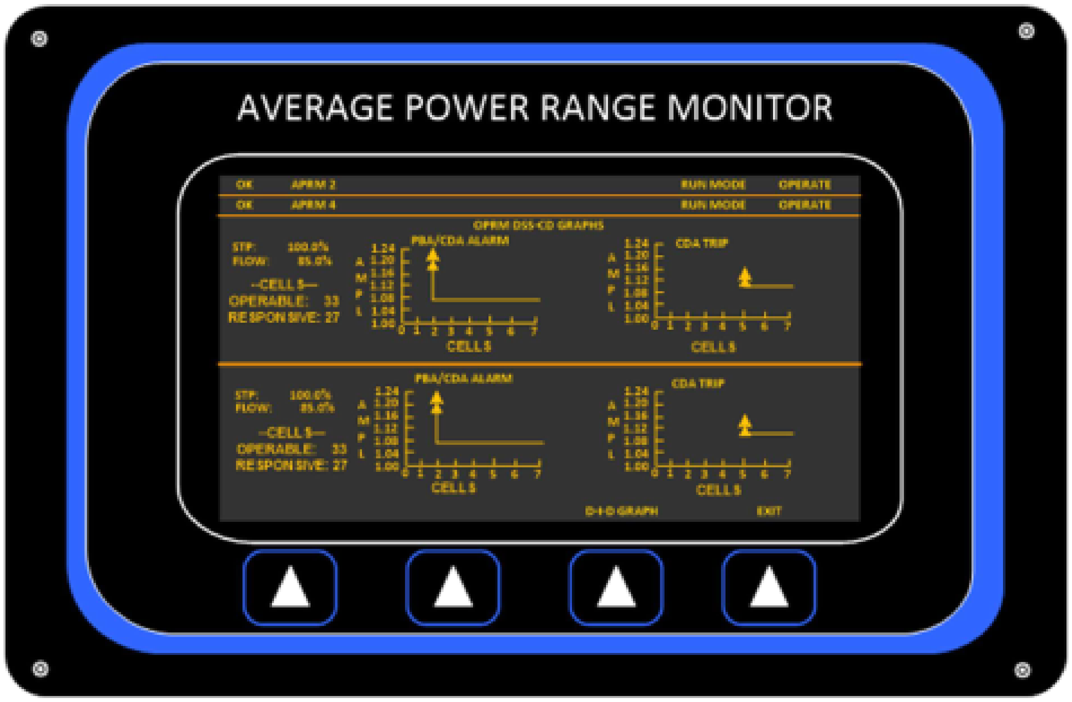


Figure 12 – DSS-CD Graphs

a) OPRM displays on the ODA

  (1) “OPRM TRIP ENABLED”

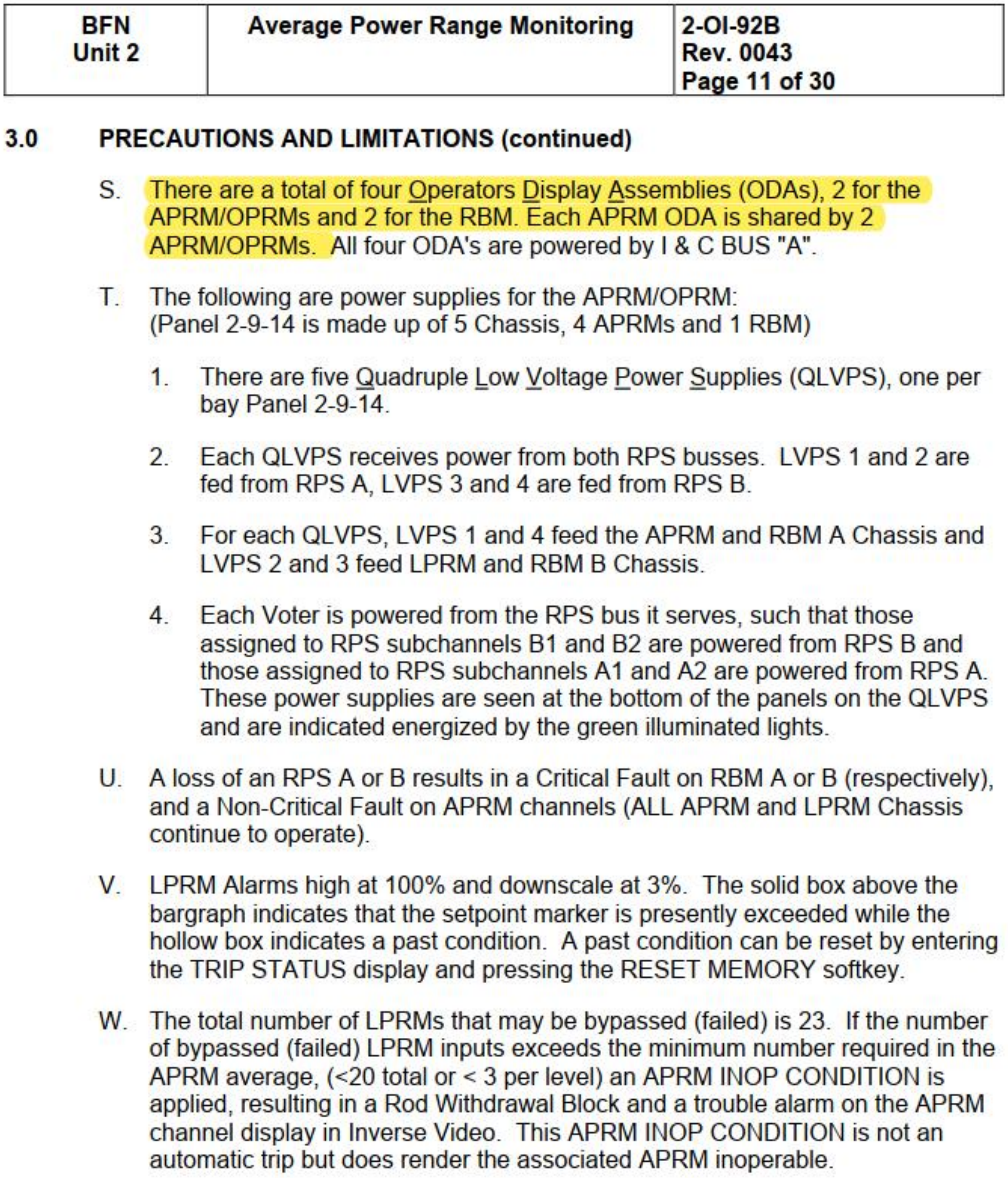
    (a) Displayed for each APRM when entering the power/flow region where instability can occur. The OPRM bar graphs screen is automatically displayed for both APRM ODAs when either APRM is within the region.

    (b) The message will be replaced with **“ANTICIPATED INSTABILITY”** whenever a Pre Trip (alarm) set point has been reached by any of the OPRM algorithms.

    If an oscillation trip exists, as defined by the OPRM trip set points, the message will be replaced with **“INSTABILITY DETECTED”** whenever a Trip set point has been reached by any one of the OPRM algorithms.

<75%(<80% for U1) Recirc drive (pump) flow with >23% Simulated Thermal Power.
Reference T-Mod BFN-1-2020-068-003 for U1 Difference.

Excerpts from 2-OI-92B:



BFN **Unit 2**	**Average Power Range Monitoring**	**2-OI-92B** **Rev. 0043** **Page 11 of 30**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

S. There are a total of four Operators Display Assemblies (ODAs), 2 for the APRM/OPRMs and 2 for the RBM. Each APRM ODA is shared by 2 APRM/OPRMs. All four ODA's are powered by I & C BUS "A".

T. The following are power supplies for the APRM/OPRM:
(Panel 2-9-14 is made up of 5 Chassis, 4 APRMs and 1 RBM)

1. There are five Quadruple Low Voltage Power Supplies (QLVPS), one per bay Panel 2-9-14.

2. Each QLVPS receives power from both RPS busses. LVPS 1 and 2 are fed from RPS A, LVPS 3 and 4 are fed from RPS B.

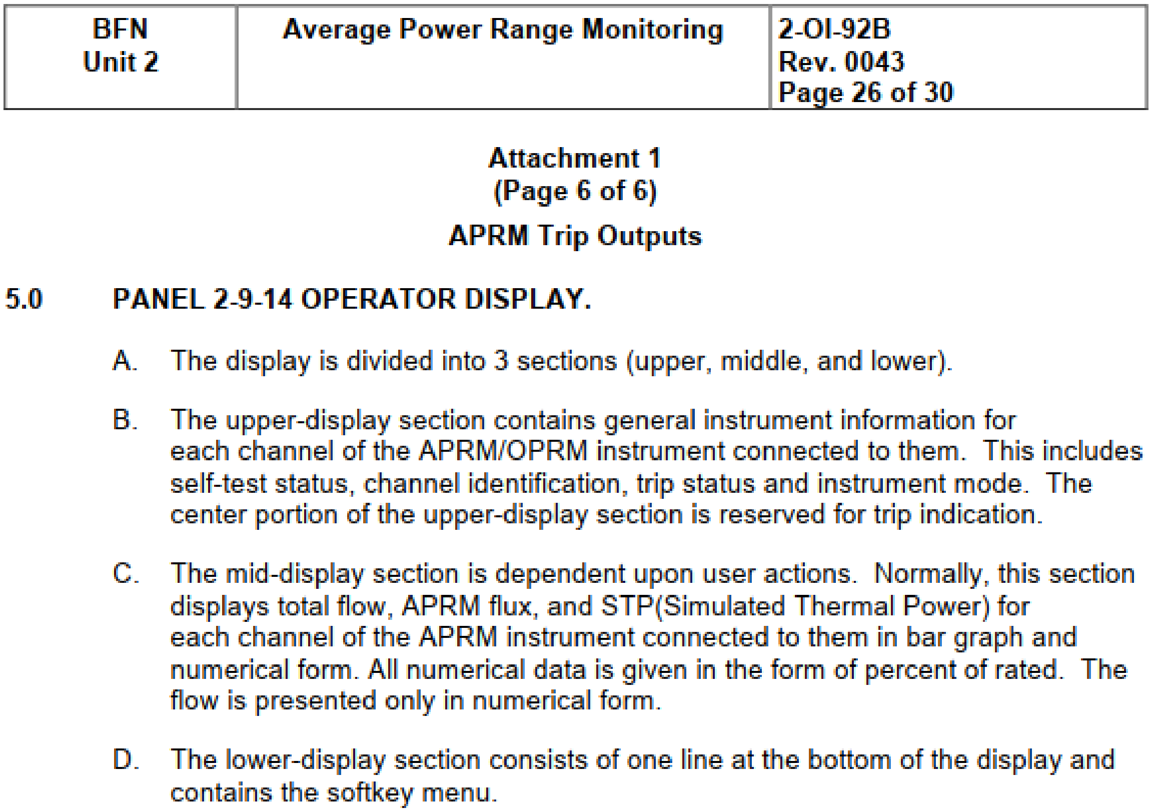
3. For each QLVPS, LVPS 1 and 4 feed the APRM and RBM A Chassis and LVPS 2 and 3 feed LPRM and RBM B Chassis.

4. Each Voter is powered from the RPS bus it serves, such that those assigned to RPS subchannels B1 and B2 are powered from RPS B and those assigned to RPS subchannels A1 and A2 are powered from RPS A. These power supplies are seen at the bottom of the panels on the QLVPS and are indicated energized by the green illuminated lights.

U. A loss of an RPS A or B results in a Critical Fault on RBM A or B (respectively), and a Non-Critical Fault on APRM channels (ALL APRM and LPRM Chassis continue to operate).

V. LPRM Alarms high at 100% and downscale at 3%. The solid box above the bargraph indicates that the setpoint marker is presently exceeded while the hollow box indicates a past condition. A past condition can be reset by entering the TRIP STATUS display and pressing the RESET MEMORY softkey.

W. The total number of LPRMs that may be bypassed (failed) is 23. If the number of bypassed (failed) LPRM inputs exceeds the minimum number required in the APRM average, (<20 total or < 3 per level) an APRM INOP CONDITION is applied, resulting in a Rod Withdrawal Block and a trouble alarm on the APRM channel display in Inverse Video. This APRM INOP CONDITION is not an automatic trip but does render the associated APRM inoperable.



BFN Unit 2	**Average Power Range Monitoring**	**2-OI-92B Rev. 0043 Page 26 of 30**

Attachment 1
(Page 6 of 6)

APRM Trip Outputs

5.0 PANEL 2-9-14 OPERATOR DISPLAY.

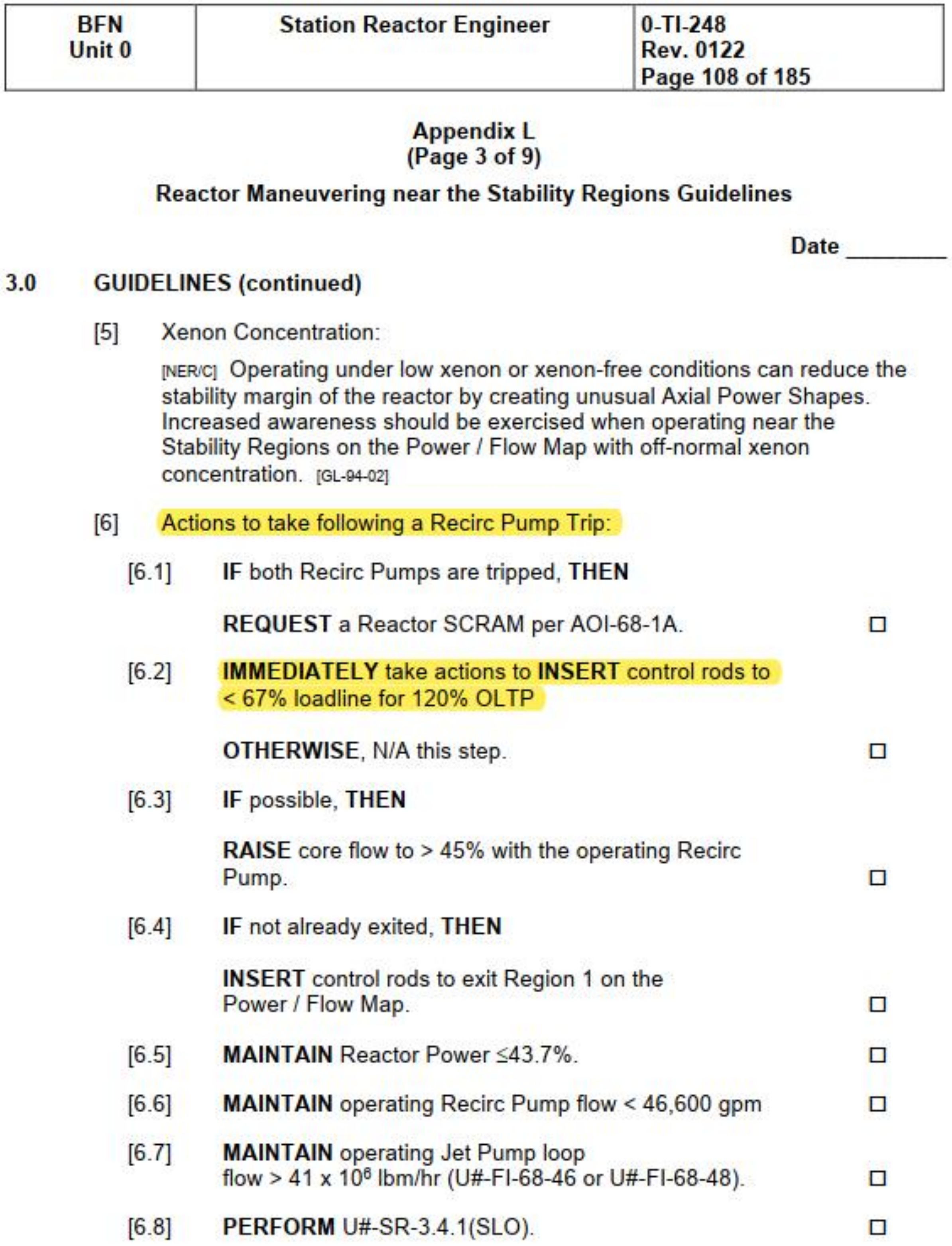
A. The display is divided into 3 sections (upper, middle, and lower).

B. The upper-display section contains general instrument information for each channel of the APRM/OPRM instrument connected to them. This includes self-test status, channel identification, trip status and instrument mode. The center portion of the upper-display section is reserved for trip indication.

C. The mid-display section is dependent upon user actions. Normally, this section displays total flow, APRM flux, and STP(Simulated Thermal Power) for each channel of the APRM instrument connected to them in bar graph and numerical form. All numerical data is given in the form of percent of rated. The flow is presented only in numerical form.

D. The lower-display section consists of one line at the bottom of the display and contains the softkey menu.

Excerpts from 2-TI-248:



BFN **Unit 0**	**Station Reactor Engineer**	**0-TI-248** **Rev. 0122** **Page 108 of 185**

Appendix L
(Page 3 of 9)

Reactor Maneuvering near the Stability Regions Guidelines

Date ________

3.0 GUIDELINES (continued)

[5] Xenon Concentration:

[NER/C] Operating under low xenon or xenon-free conditions can reduce the stability margin of the reactor by creating unusual Axial Power Shapes. Increased awareness should be exercised when operating near the Stability Regions on the Power / Flow Map with off-normal xenon concentration. [GL-94-02]

[6] Actions to take following a Recirc Pump Trip:

[6.1] **IF** both Recirc Pumps are tripped, **THEN**

REQUEST a Reactor SCRAM per AOI-68-1A. ☐

[6.2] **IMMEDIATELY** take actions to **INSERT** control rods to < 67% loadline for 120% OLTP

OTHERWISE, N/A this step. ☐

[6.3] **IF** possible, **THEN**

RAISE core flow to > 45% with the operating Recirc Pump. ☐

[6.4] **IF** not already exited, **THEN**

INSERT control rods to exit Region 1 on the Power / Flow Map. ☐

[6.5] **MAINTAIN** Reactor Power ≤43.7%. ☐

[6.6] **MAINTAIN** operating Recirc Pump flow < 46,600 gpm ☐

[6.7] **MAINTAIN** operating Jet Pump loop flow > 41 x 10^6 lbm/hr (U#-FI-68-46 or U#-FI-68-48). ☐

[6.8] **PERFORM** U#-SR-3.4.1(SLO). ☐

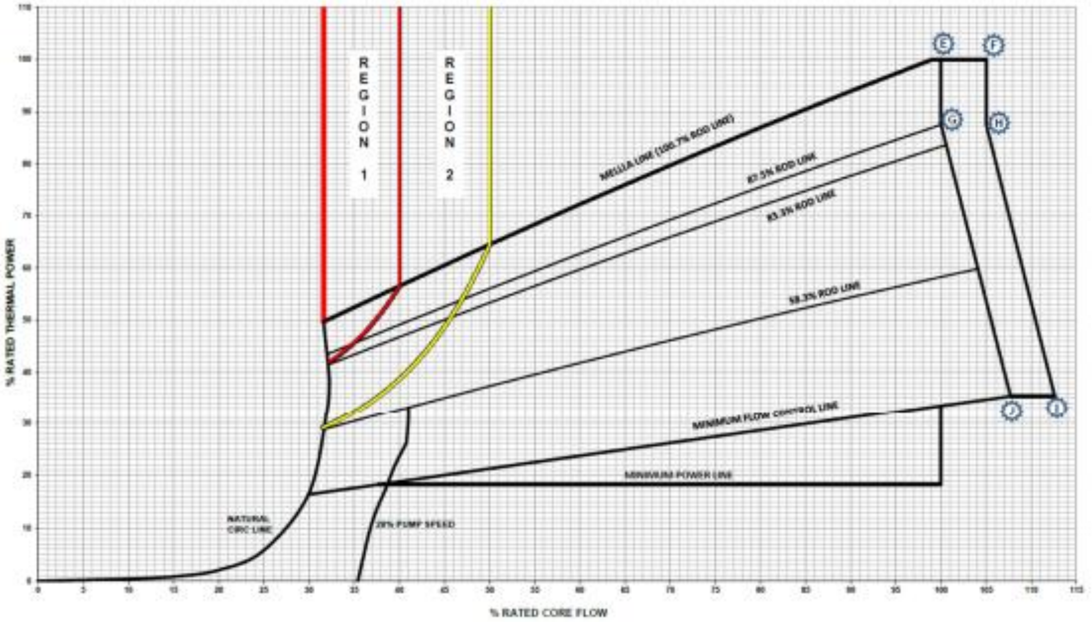
BFN **Unit 0**	**Station Reactor Engineer**	**0-TI-248** **Rev. 0122** **Page 110 of 185**

Appendix L
(Page 5 of 9)

Reactor Maneuvering near the Stability Regions Guidelines

3.0 GUIDELINES (continued)

Recirculation System Operating Power / Flow Map for 120% OLTP/MELLLA



[7] Power Flow Map for 120% OLTP EPU/MELLLA:

[7.1] Natural Circulation Line:

Represents reactor power response if control rods were withdrawn in the absence of Reactor Recirc Pump operation. This line does not represent an operating boundary. Core Flow measurement can be inaccurate at conditions along this line.

[7.2] 28% Dual Pump Speed Line:

Expected reactor power level would follow this line whenever the control rods were withdrawn with both Reactor Recirc Pumps at 28% speed (~480 RPM).

Examination Outline Cross-reference:

259002 (SF2 RWLCS) Reactor Water Level Control

K3.04 (10CFR 55.41.7)

Knowledge of the effect that a loss or malfunction of the Reactor Water Level Control System will have on the following systems or system parameters:

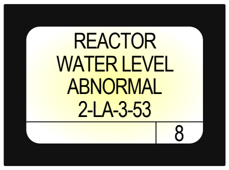
- Recirculation system

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	259002K3.04	
Importance Rating	3.3	------

Proposed Question: **# 5**

Unit 2 is operating at 100% RTP when the following conditions occur:

- RFPT ‘2A’ trips
- REACTOR WATER LEVEL ABNORMAL (2-9-5A, Window 8) alarms



Given the above conditions and in accordance with 2-OI-68, Reactor Recirculation System, the Recirc Pump _________ limiter will be enforcing.

A. 28%

B. 58%

C. 75%

D. 90%

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that this is the Recirc Pump Speed Limit following an automatic runback with TOTAL Feedwater flow lost (< 16%) versus an INDIVIDUAL Feed flow being lost (< 16%). There are multiple automatic and manual runbacks, so they are easily confused by candidates.

B INCORRECT: Incorrect but plausible in that this is the expected Core Flow Value following a manual Core Flow Runback. There are multiple automatic and manual runbacks, so they are easily confused by candidates.

C **CORRECT**: (*See attached*) In accordance with the given alarm 2-9-5A, Window 8, and given that RFPT ‘2A’ has tripped, the individual RFP ‘2A’ flow is less than 16% rated flow AND Reactor Water Level reached ≤ 27 inches in triggering the Low Reactor Water Level alarm point. Therefore, this makes up the 75% Limiter Runback and Recirc Pump speeds will automatically runback to 75% (1130 RPM).

D INCORRECT: Incorrect but plausible in that this is the expected Core Flow Value following a manual High Power (Upper Power) Runback. There are multiple automatic and manual runbacks, so they are easily confused by candidates.

RO Level Justification: Tests the candidate's knowledge of the effect of Reactor Feedwater Pump Trip on the Recirc Flow Control System by requiring the candidate to use specific plant conditions to determine the subsequent effect on the Recirc Flow Control 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. The candidate must know the alarm setpoint, integrate that with the loss of a Reactor Feedwater Pump and apply it to limits in a completely different system.

Technical Reference(s):	2-OI-68, Rev. 165	(Attach if not previously provided)
	2-AOI-3-1, Rev. 24	
	2-ARP-9-4A, Rev. 50	
	2-ARP-9-5A, Rev. 61	

Proposed references to be provided to applicants during examination: **REACTOR WATER LEVEL ABNORMAL (2-9-5A, Window 8)**

Learning Objective: OPL171.007 Obj. 14 (As available)

Question Source:	Bank #	BFN 1804 NRC #56	
	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:

Proposed Question: **# 56**

Unit 2 is operating at 100% RTP when RFPT ‘A’ trips. As a result, Reactor Water Level lowers until the following alarm is received:

- REACTOR WATER LEVEL ABNORMAL (2-9-5A, Window 8)

Given the above, which ONE of the following identifies the resulting **approximate** Recirc Pump Speed Limit one minute later in accordance with 2-OI-68, Reactor Recirculation System?

A. 28%

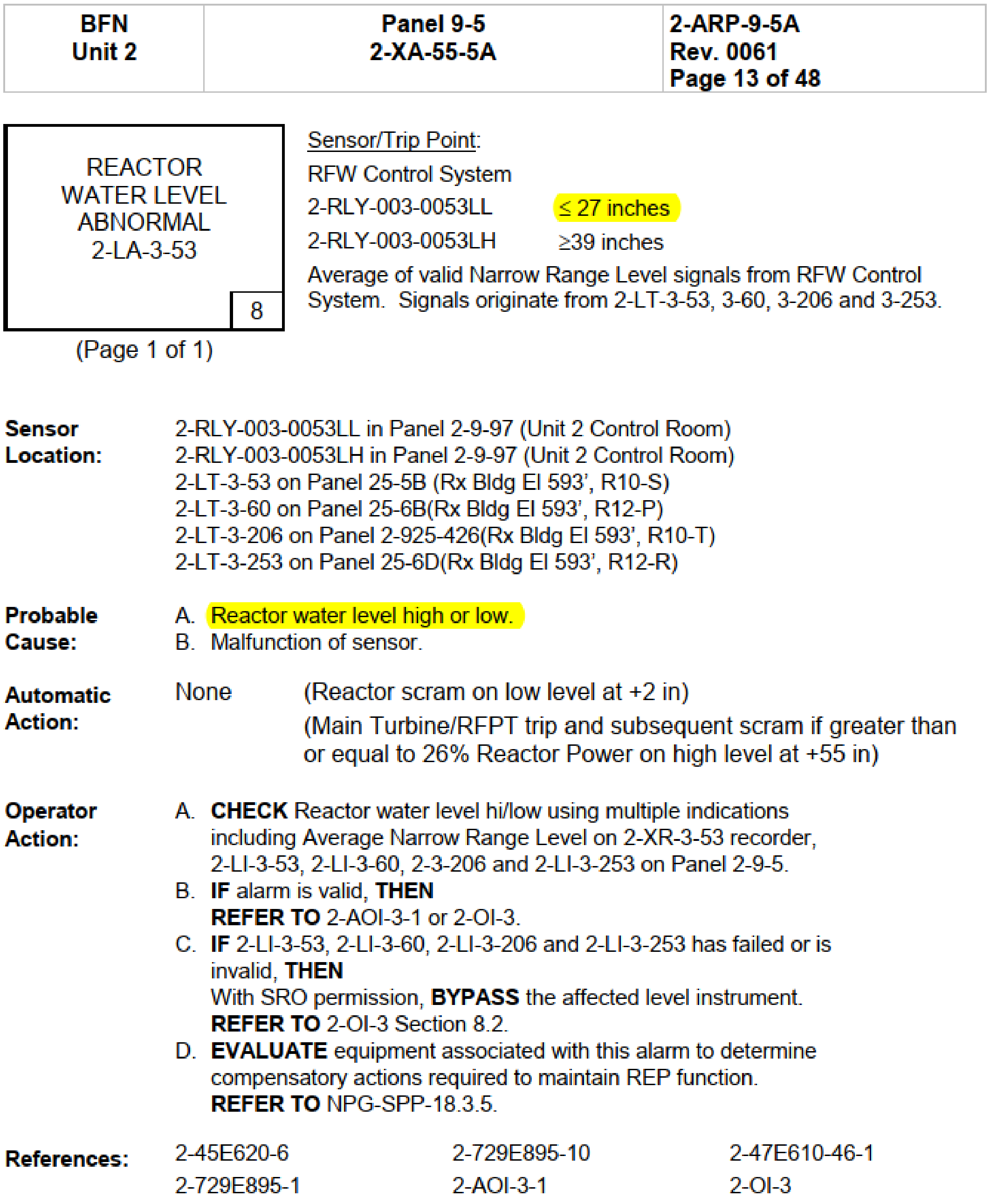
B. 60%

C. 75%

D. 90%

Proposed Answer: **C**

Excerpt from 2-ARP-9-5A:



BFN **Unit 2**	**Panel 9-5** **2-XA-55-5A**	**2-ARP-9-5A** **Rev. 0061** **Page 13 of 48**

REACTOR
WATER LEVEL
ABNORMAL
2-LA-3-53

8

(Page 1 of 1)

Sensor/Trip Point:

RFW Control System

2-RLY-003-0053LL	≤ 27 inches
2-RLY-003-0053LH	≥39 inches

Average of valid Narrow Range Level signals from RFW Control System. Signals originate from 2-LT-3-53, 3-60, 3-206 and 3-253.

Sensor Location:
2-RLY-003-0053LL in Panel 2-9-97 (Unit 2 Control Room)
2-RLY-003-0053LH in Panel 2-9-97 (Unit 2 Control Room)
2-LT-3-53 on Panel 25-5B (Rx Bldg El 593’, R10-S)
2-LT-3-60 on Panel 25-6B(Rx Bldg El 593’, R12-P)
2-LT-3-206 on Panel 2-925-426(Rx Bldg El 593’, R10-T)
2-LT-3-253 on Panel 25-6D(Rx Bldg El 593’, R12-R)

Probable Cause:
A. Reactor water level high or low.
B. Malfunction of sensor.

Automatic Action:
None (Reactor scram on low level at +2 in)
(Main Turbine/RFPT trip and subsequent scram if greater than or equal to 26% Reactor Power on high level at +55 in)

Operator Action:

A. **CHECK** Reactor water level hi/low using multiple indications including Average Narrow Range Level on 2-XR-3-53 recorder, 2-LI-3-53, 2-LI-3-60, 2-3-206 and 2-LI-3-253 on Panel 2-9-5.

B. **IF** alarm is valid, **THEN**
REFER TO 2-AOI-3-1 or 2-OI-3.

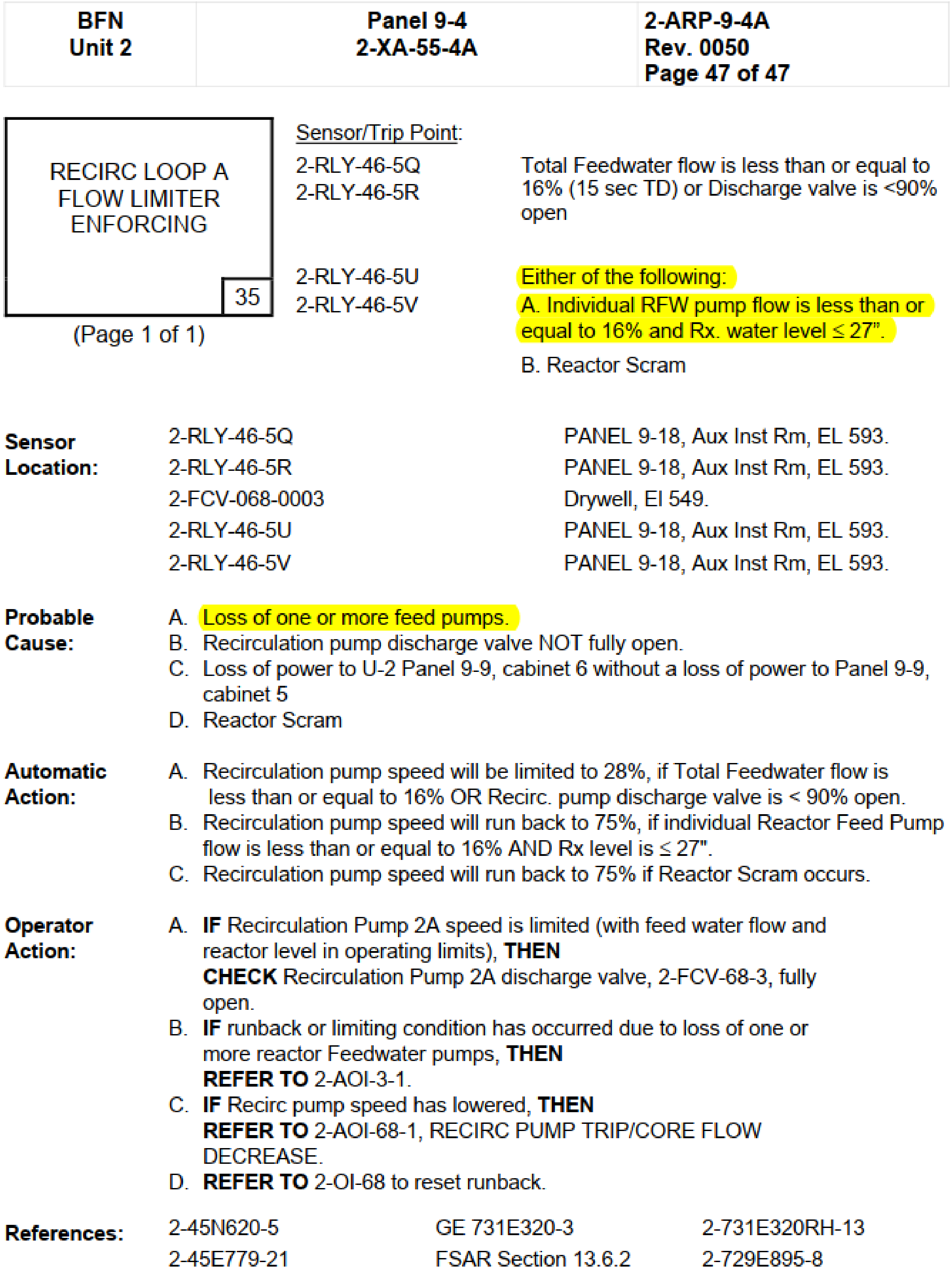
C. **IF** 2-LI-3-53, 2-LI-3-60, 2-LI-3-206 and 2-LI-3-253 has failed or is invalid, **THEN**
With SRO permission, **BYPASS** the affected level instrument.
REFER TO 2-OI-3 Section 8.2.

D. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function.
REFER TO NPG-SPP-18.3.5.

References:

2-45E620-6	2-729E895-10	2-47E610-46-1
2-729E895-1	2-AOI-3-1	2-OI-3

Excerpt from 2-ARP-9-4A:



BFN **Unit 2**	**Panel 9-4** **2-XA-55-4A**	**2-ARP-9-4A** **Rev. 0050** **Page 47 of 47**

RECIRC LOOP A
FLOW LIMITER
ENFORCING
35

(Page 1 of 1)

Sensor/Trip Point:

2-RLY-46-5Q 2-RLY-46-5R	Total Feedwater flow is less than or equal to 16% (15 sec TD) or Discharge valve is <90% open
2-RLY-46-5U 2-RLY-46-5V	Either of the following: A. Individual RFW pump flow is less than or equal to 16% and Rx. water level ≤ 27”. B. Reactor Scram

Sensor Location:

2-RLY-46-5Q	PANEL 9-18, Aux Inst Rm, EL 593.
2-RLY-46-5R	PANEL 9-18, Aux Inst Rm, EL 593.
2-FCV-068-0003	Drywell, El 549.
2-RLY-46-5U	PANEL 9-18, Aux Inst Rm, EL 593.
2-RLY-46-5V	PANEL 9-18, Aux Inst Rm, EL 593.

Probable Cause:

A. Loss of one or more feed pumps.
B. Recirculation pump discharge valve NOT fully open.
C. Loss of power to U-2 Panel 9-9, cabinet 6 without a loss of power to Panel 9-9, cabinet 5
D. Reactor Scram

Automatic Action:

A. Recirculation pump speed will be limited to 28%, if Total Feedwater flow is less than or equal to 16% OR Recirc. pump discharge valve is < 90% open.
B. Recirculation pump speed will run back to 75%, if individual Reactor Feed Pump flow is less than or equal to 16% AND Rx level is ≤ 27".
C. Recirculation pump speed will run back to 75% if Reactor Scram occurs.

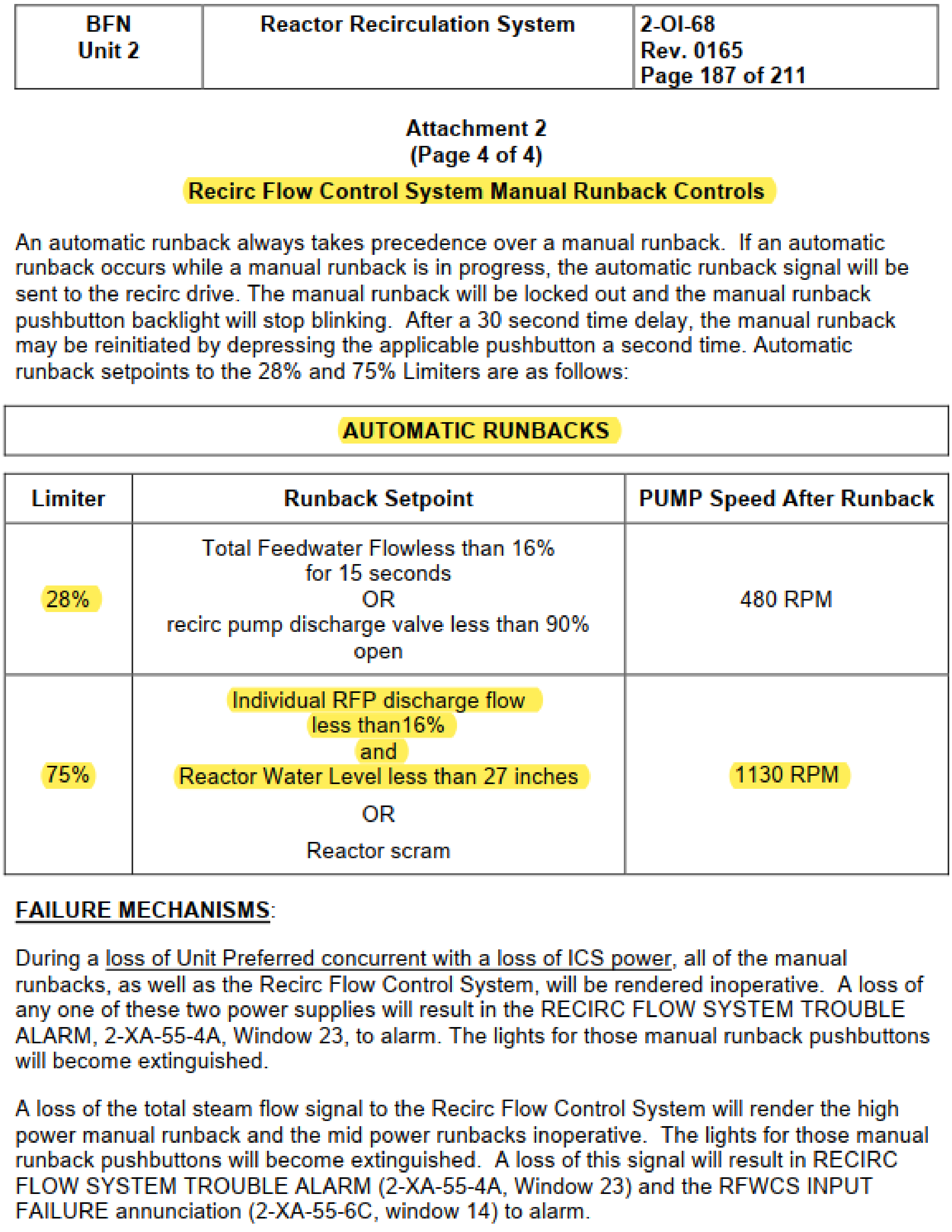
Operator Action:

A. **IF** Recirculation Pump 2A speed is limited (with feed water flow and reactor level in operating limits), **THEN**
CHECK Recirculation Pump 2A discharge valve, 2-FCV-68-3, fully open.
B. **IF** runback or limiting condition has occurred due to loss of one or more reactor Feedwater pumps, **THEN**
REFER TO 2-AOI-3-1.
C. **IF** Recirc pump speed has lowered, **THEN**
REFER TO 2-AOI-68-1, RECIRC PUMP TRIP/CORE FLOW DECREASE.
D. **REFER TO** 2-OI-68 to reset runback.

References:

2-45N620-5	GE 731E320-3	2-731E320RH-13
2-45E779-21	FSAR Section 13.6.2	2-729E895-8

Excerpts from 2-OI-68:



BFN Unit 2	**Reactor Recirculation System**	**2-OI-68 Rev. 0165 Page 187 of 211**

Attachment 2
(Page 4 of 4)

Recirc Flow Control System Manual Runback Controls

An automatic runback always takes precedence over a manual runback. If an automatic runback occurs while a manual runback is in progress, the automatic runback signal will be sent to the recirc drive. The manual runback will be locked out and the manual runback pushbutton backlight will stop blinking. After a 30 second time delay, the manual runback may be reinitiated by depressing the applicable pushbutton a second time. Automatic runback setpoints to the 28% and 75% Limiters are as follows:

AUTOMATIC RUNBACKS

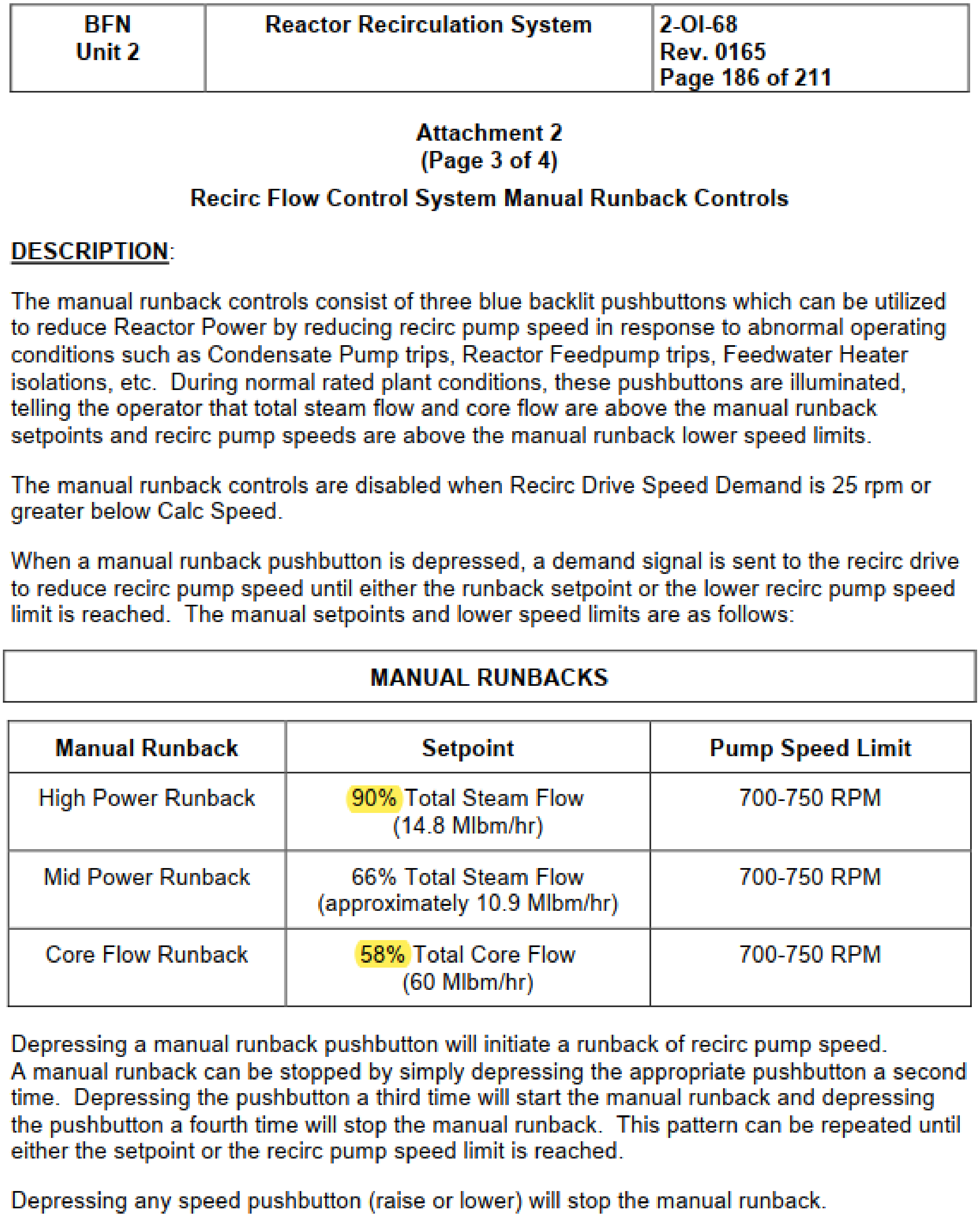
Limiter	**Runback Setpoint**	**PUMP Speed After Runback**
28%	Total Feedwater Flowless than 16% for 15 seconds OR recirc pump discharge valve less than 90% open	480 RPM
75%	Individual RFP discharge flow less than16% and Reactor Water Level less than 27 inches OR Reactor scram	1130 RPM

FAILURE MECHANISMS:

During a loss of Unit Preferred concurrent with a loss of ICS power, all of the manual runbacks, as well as the Recirc Flow Control System, will be rendered inoperative. A loss of any one of these two power supplies will result in the RECIRC FLOW SYSTEM TROUBLE ALARM, 2-XA-55-4A, Window 23, to alarm. The lights for those manual runback pushbuttons will become extinguished.

A loss of the total steam flow signal to the Recirc Flow Control System will render the high power manual runback and the mid power runbacks inoperative. The lights for those manual runback pushbuttons will become extinguished. A loss of this signal will result in RECIRC FLOW SYSTEM TROUBLE ALARM (2-XA-55-4A, Window 23) and the RFWCS INPUT FAILURE annunciation (2-XA-55-6C, window 14) to alarm.

Supports Distractors B, D



BFN Unit 2	Reactor Recirculation System	2-OI-68 Rev. 0165 Page 186 of 211

Attachment 2
(Page 3 of 4)

Recirc Flow Control System Manual Runback Controls

DESCRIPTION:

The manual runback controls consist of three blue backlit pushbuttons which can be utilized to reduce Reactor Power by reducing recirc pump speed in response to abnormal operating conditions such as Condensate Pump trips, Reactor Feedpump trips, Feedwater Heater isolations, etc. During normal rated plant conditions, these pushbuttons are illuminated, telling the operator that total steam flow and core flow are above the manual runback setpoints and recirc pump speeds are above the manual runback lower speed limits.

The manual runback controls are disabled when Recirc Drive Speed Demand is 25 rpm or greater below Calc Speed.

When a manual runback pushbutton is depressed, a demand signal is sent to the recirc drive to reduce recirc pump speed until either the runback setpoint or the lower recirc pump speed limit is reached. The manual setpoints and lower speed limits are as follows:

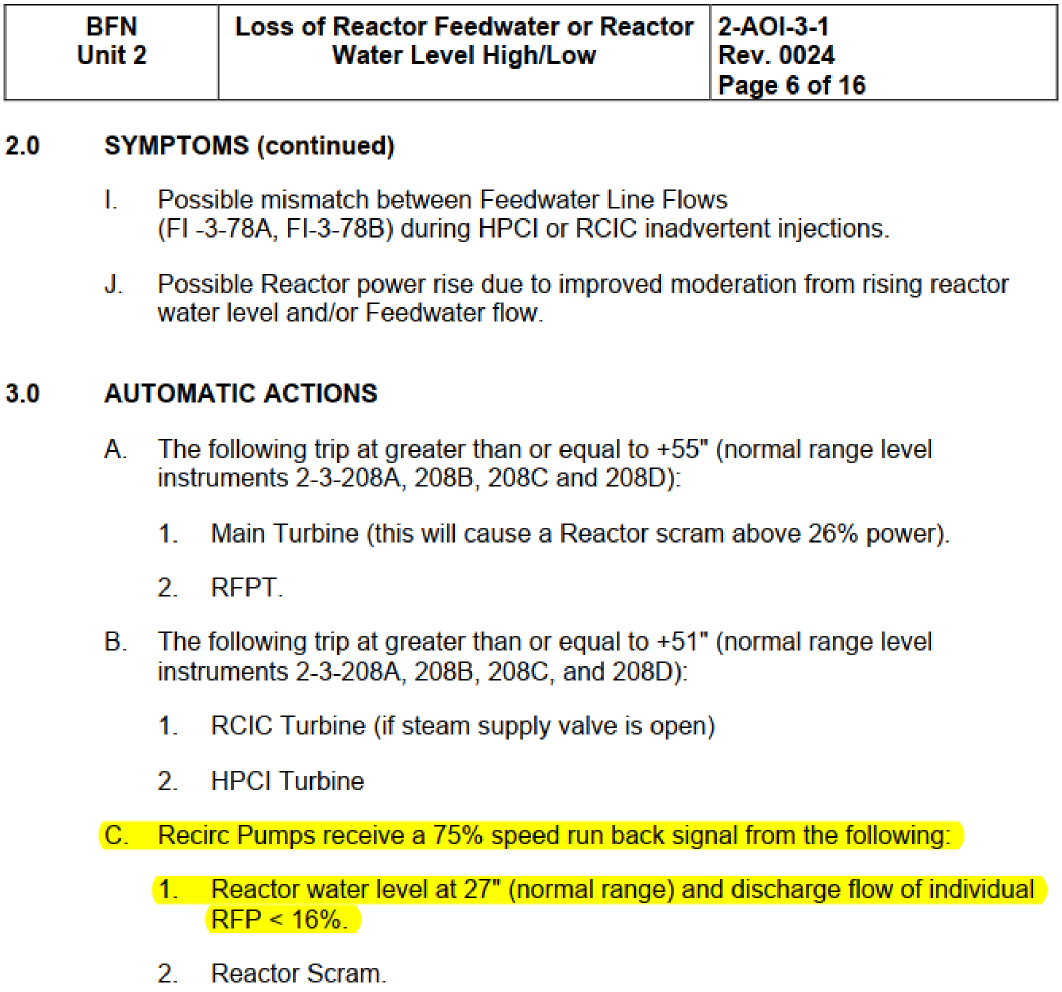
MANUAL RUNBACKS

Manual Runback	Setpoint	Pump Speed Limit
High Power Runback	90% Total Steam Flow (14.8 Mlbm/hr)	700-750 RPM
Mid Power Runback	66% Total Steam Flow (approximately 10.9 Mlbm/hr)	700-750 RPM
Core Flow Runback	58% Total Core Flow (60 Mlbm/hr)	700-750 RPM

Depressing a manual runback pushbutton will initiate a runback of recirc pump speed. A manual runback can be stopped by simply depressing the appropriate pushbutton a second time. Depressing the pushbutton a third time will start the manual runback and depressing the pushbutton a fourth time will stop the manual runback. This pattern can be repeated until either the setpoint or the recirc pump speed limit is reached.

Depressing any speed pushbutton (raise or lower) will stop the manual runback.

Excerpt from 2-AOI-3-1:



BFN **Unit 2**	**Loss of Reactor Feedwater or Reactor Water Level High/Low**	**2-AOI-3-1** **Rev. 0024** **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

C. Recirc Pumps receive a 75% speed run back signal from the following:

1. Reactor water level at 27" (normal range) and discharge flow of individual RFP < 16%.
2. Reactor Scram.

Examination Outline Cross-reference:

211000 (SF1 SLCS) Standby Liquid Control

K6.06 (10CFR 55.41.7)

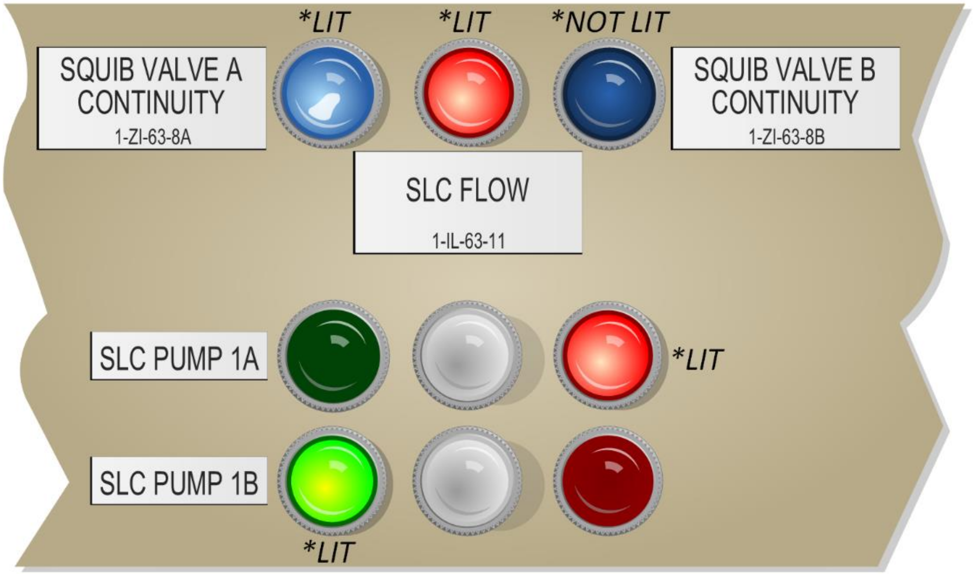
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Standby Liquid Control System:

- Redundant reactivity control system

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	211000K6.06	
Importance Rating	3.6	------

Proposed Question: **# 6**

Unit 1 has suffered an Anticipated Transient Without SCRAM (ATWS) with the following indications:



Given the indications above, SLC Squib Valve **(1)** is open and the time to reach Hot Shutdown Boron Weight is **(2)** compared to the time with both squib valves open.

A. (1) A
(2) longer

B. (1) A
(2) the same

C. (1) B
(2) longer

D. (1) B
(2) the same

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate confuses Squib Valve firing indications in that Squib Valve ‘A’ blue light is indicated as LIT. Second part is incorrect but plausible in that a candidate thinks that if two valves open vs one will allow for a higher system flow.

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 1-EOI Appendix-3A, SLC INJECTION, Squib Valve B has fired. This is indicated by SQUIB VALVE ‘B’ CONTINUITY blue light being **extinguished** as shown in the provided drawing. For second part, although with both Squib Valves open provides two initial paths, the paths come back together and inject through a single common pipe as one flow path. Therefore, regardless of whether one or two Squib Valves have opened, the flow rate to inject to the vessel is the same. The reference to Hot Shutdown Boron Weight is when SLC has injected long enough to so SLC tank level has lowered by 12%.

RO Level Justification: Tests the candidate’s knowledge of the redundant reactivity control design of the Standby Liquid Control (SLC) System operation as it relates to Anticipated Transient Without SCRAM (ATWS) 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): OPL 171.039, Rev. 24 (Attach if not previously provided)
1-EOI-Appendix-3A, Rev. 0

Proposed references to be provided to applicants during examination: **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**

Learning Objective: OPL171.039 Obj. 4d (As available)

Question Source:	Bank #	BFN 1306 NRC #33	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2013	

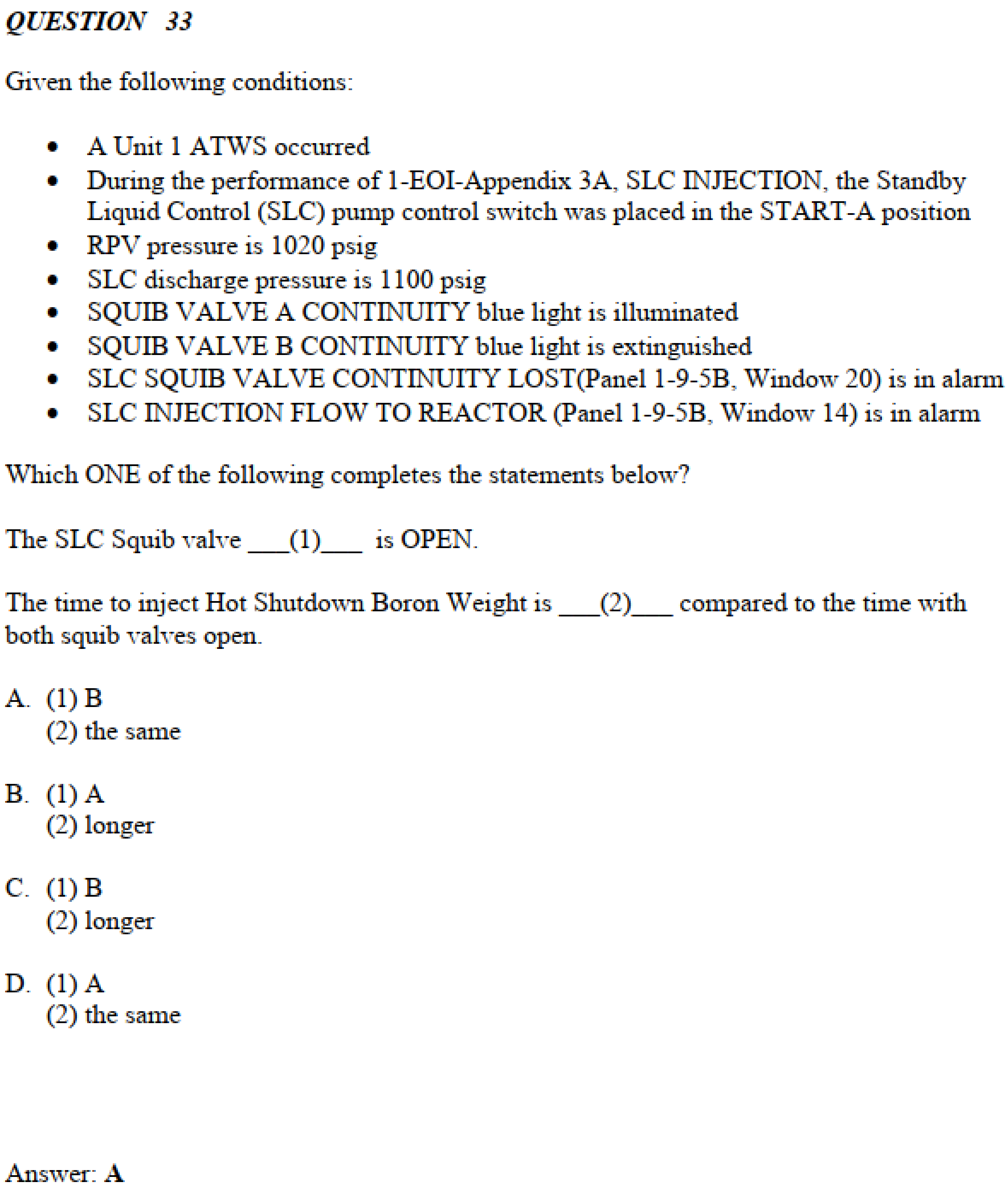
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

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

55.43

Comments:

Copy of Bank Question:



QUESTION 33

Given the following conditions:

- A Unit 1 ATWS occurred
- During the performance of 1-EOI-Appendix 3A, SLC INJECTION, the Standby Liquid Control (SLC) pump control switch was placed in the START-A position
- RPV pressure is 1020 psig
- SLC discharge pressure is 1100 psig
- SQUIB VALVE A CONTINUITY blue light is illuminated
- SQUIB VALVE B CONTINUITY blue light is extinguished
- SLC SQUIB VALVE CONTINUITY LOST(Panel 1-9-5B, Window 20) is in alarm
- SLC INJECTION FLOW TO REACTOR (Panel 1-9-5B, Window 14) is in alarm

Which ONE of the following completes the statements below?

The SLC Squib valve ___(1)___ is OPEN.

The time to inject Hot Shutdown Boron Weight is ___(2)___ compared to the time with both squib valves open.

A. (1) B
(2) the same

B. (1) A
(2) longer

C. (1) B
(2) longer

D. (1) A
(2) the same

Answer: **A**

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. _____

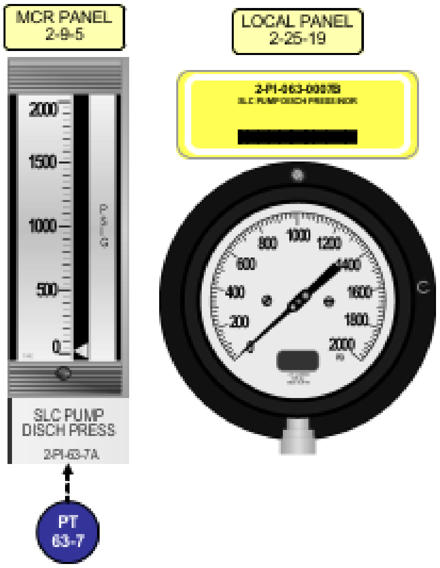
Excerpts from OPL171.039 Lesson Plan:

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

e. To prevent bypass flow from one pump in case of relief valve failure in the line from the other pump, a check valve is installed in the pump discharge pipe downstream of each relief valve line.

NLO Obj. 4i,
NLOR Obj. 3f, 7c / e
ILT Obj. 4j, 12c / e
LOR Obj. 12c / e

The discharge valve for each SLC Pump is a locked-open, manual SLC PUMP DISCHARGE SHUTOFF VALVE: SHV-63-515 and SHV-63-517 (SLC Pump A and B respectively). The discharge piping, downstream of the discharge valves, combines to form a common discharge header that contains pressure and flow instrumentation for system monitoring. SLC PUMP DISCHARGE PRESSURE TRANSMITTER (PT-63-7), which is powered by 120V AC I&C B, and associated SLC PUMP DISCHARGE PRESSURE indicator (PI-63-7A), provide discharge header pressure indication (0-2000 psig) on Control Room Panel 9-5. Local discharge pressure indication is also available at Panel 25-19 SLC PUMP DISCHARGE PRESSURE INDICATOR (PI-63-7B).



Downstream of the pumps, the discharge piping splits into two separate headers; each containing an explosive A(B) SLC SQUIB VALVE (FCV-63-8A and 8B).

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

D. SLC System Operation / Flowpath

1. SLC Pump discharge flow passes through the Squib Valves and combines into a single injection line, where the flow signal is developed.

NLO Obj. 2, 4g

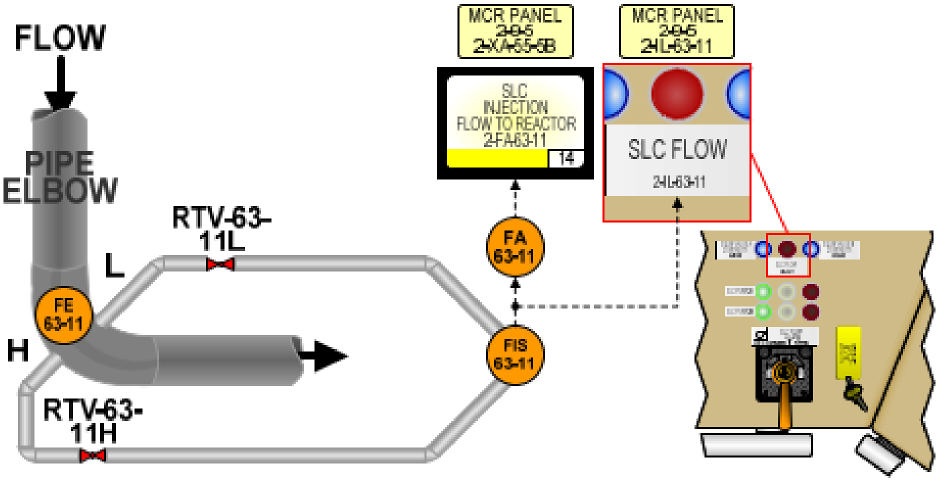
NLOR Obj. 1, 3e, 7a / b / c / e / f / g

ILT Obj. 2, 4h, 9, 12a / b / c / e / f / g

LOR Obj. 9, 12a / b / c / e / f / g

SLC System flow indication is provided by an elbow meter SLC FLOW ELEMENT (FE-63-11). The elbow meter is another flow measuring device based on differential pressure. As the fluid changes direction in a pipe bend, a low pressure area is created on the inner pipe radius. A high pressure area is created on the outer pipe radius. This pressure difference is proportional to the square of the volumetric flow rate. Elbow flow meters indicate small pressure differences. Because of this, the instrument is noted for its repeatability and for its accuracy at high flowrates.

The flow element provides input to a SLC FLOW TO REACTOR INDICATOR SWITCH (FIS-63-11), which then feeds an alarm output and a red “SLC Flow” indicating light (both on Control Room Panel 9-5). The power supply to the red flow-indicating light is 120V AC I&C A.



Annunciator SLC INJECTION FLOW TO REACTOR, (9-5B Window 14) is provided to inform operators that SLC has been initiated or warn operators of a sensor malfunction. The flow alarm activates at a sensed flow of 40 gpm and rising.

Examination Outline Cross-reference:

Conduct of Operations

G2.1.1 (10CFR 55.41.10)

Knowledge of conduct of operations requirements

Level	RO	SRO
Tier #	3	------
Group #	-------	------
K/A #	G2.1.1	
Importance Rating	3.8	------

Proposed Question: **# 7**

In accordance with OPDP-1, Conduct of Operations, the BFN Reactor Controls Area is defined as Panels 9-3 through **(1)** and when the Reactor is operating at 100% RTP, the OATC **(2)** **REQUIRED** to notify the NUSO prior to leaving the Reactor Controls Area.

A. (1) 9-8
(2) is

B. (1) 9-8
(2) is NOT

C. (1) 9-54
(2) is

D. (1) 9-54
(2) is NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with OPDP-1, Conduct of Operations, the Reactor Control Area is defined as the area encompassed by Panels 9-3 through 9-8 for BFN. For second part, in accordance with OPDP-1, the Operator at the Controls (OATC) is required to make an announcement when they are leaving the area such that the Nuclear Unit Senior Operator is aware of their status.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible in that the Balance of Plant Operator (BOP) nor the NUSO are required to announce that they are leaving the Reactor Controls Area, therefore it is plausible that the OATC would not be required to announce that they are leaving the Reactor Controls Area.

C INCORRECT**:** First part is incorrect but plausible in that there are numerous panels in the Control Room including Panel 9-54, and any area could be considered the Reactor Controls Area, as there is equipment on all panels directly related to safe Reactor operation. 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 knowledge of the requirements of OPDP-1, Conduct of Operations, requirements with regard to the Operator at the Controls and the definition of the Controls Area. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): OPDP-1, Rev. 53 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.071, Obj. 3c (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**

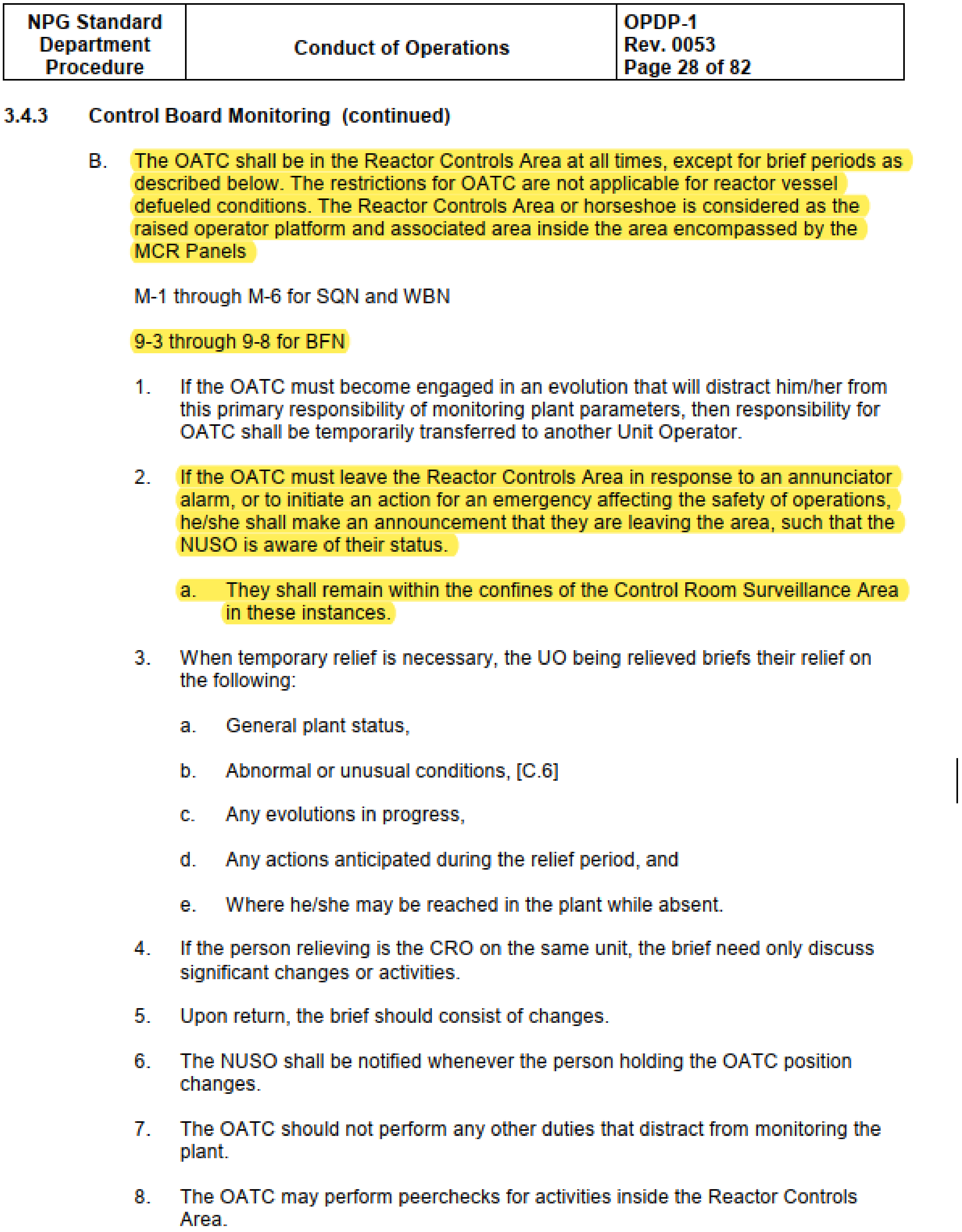
Comprehension or Analysis

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. 0053** **Page 28 of 82**

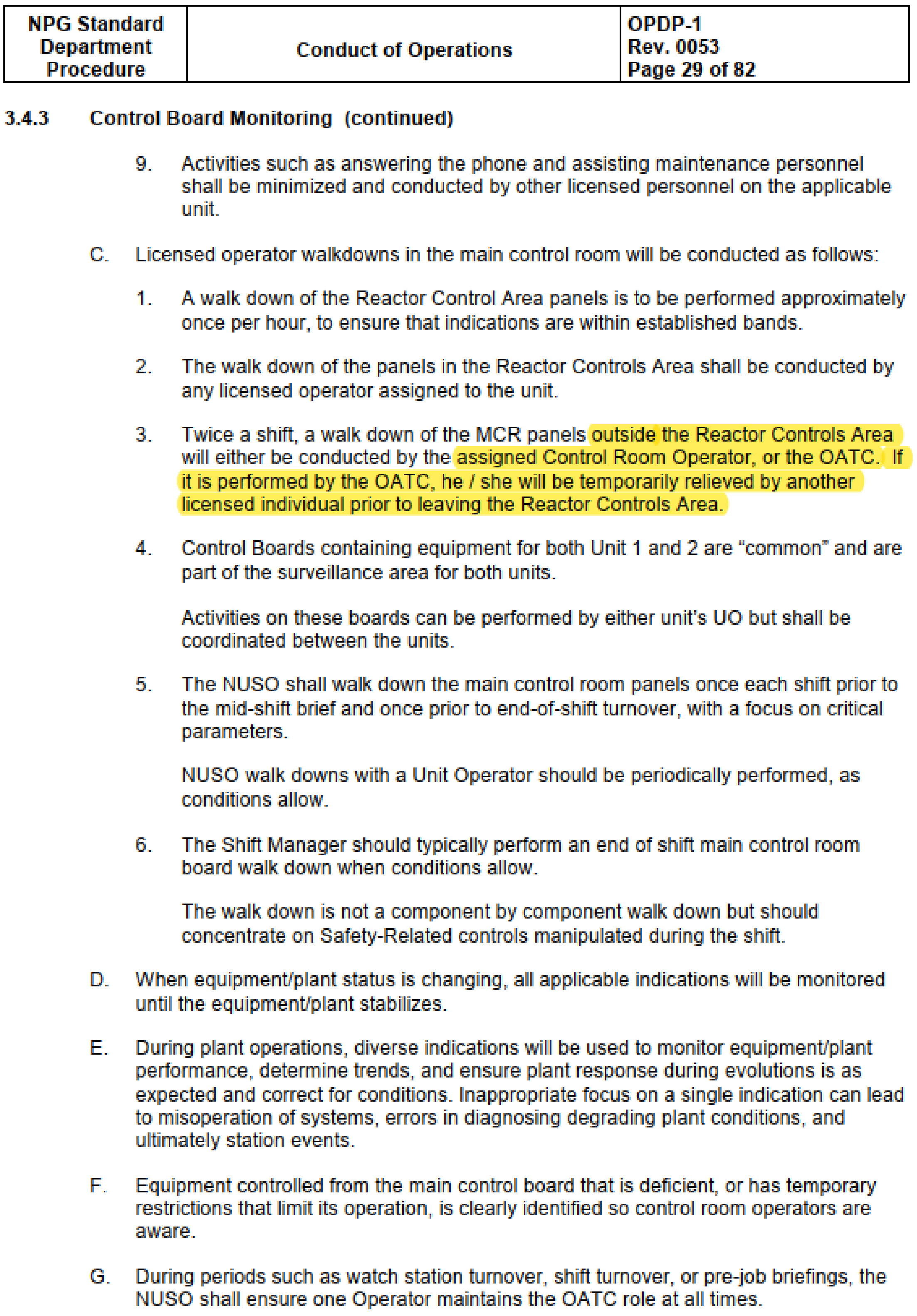
3.4.3 Control Board Monitoring (continued)

B. The OATC shall be in the Reactor Controls Area at all times, except for brief periods as described below. The restrictions for OATC are not applicable for reactor vessel defueled conditions. The Reactor Controls Area or horseshoe is considered as the raised operator platform and associated area inside the area encompassed by the MCR Panels

M-1 through M-6 for SQN and WBN

9-3 through 9-8 for BFN

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 their relief on the following:
   a. General plant status,
   b. Abnormal or unusual conditions, [C.6]
   c. Any evolutions in progress,
   d. Any actions anticipated during the relief period, and
   e. Where he/she may be reached in the plant while absent.
4. If the person relieving is the CRO on the same unit, the brief need only discuss significant changes or activities.
5. Upon return, the brief should consist of changes.
6. The NUSO shall be notified whenever the person holding the OATC position changes.
7. The OATC should not perform any other duties that distract from monitoring the plant.
8. The OATC may perform peerchecks for activities inside the Reactor Controls Area.



NPG Standard Department Procedure	**Conduct of Operations**	**OPDP-1** **Rev. 0053** **Page 29 of 82**

3.4.3 Control Board Monitoring (continued)

9. 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.

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.

4. Control Boards containing equipment for both Unit 1 and 2 are "common" and are part of the surveillance area for both units.

   Activities on these boards can be performed by either unit's UO but shall be coordinated between the units.

5. The NUSO shall walk down the main control room panels once each shift prior to the mid-shift brief and once prior to end-of-shift turnover, with a focus on critical parameters.

   NUSO walk downs with a Unit Operator should be periodically performed, as conditions allow.

6. The Shift Manager should typically perform an end of shift main control room board walk down when conditions allow.

   The walk down is not a component by component walk down but should concentrate on Safety-Related controls manipulated during the shift.

D. When equipment/plant status is changing, all applicable indications will be monitored until the equipment/plant stabilizes.

E. During plant operations, diverse indications will be used to monitor equipment/plant performance, determine trends, and ensure plant response during evolutions is as expected and correct for conditions. Inappropriate focus on a single indication can lead to misoperation of systems, errors in diagnosing degrading plant conditions, and ultimately station events.

F. Equipment controlled from the main control board that is deficient, or has temporary restrictions that limit its operation, is clearly identified so control room operators are aware.

G. During periods such as watch station turnover, shift turnover, or pre-job briefings, the NUSO shall ensure one Operator maintains the OATC role at all times.

Examination Outline Cross-reference:	Level	RO	SRO
205000 (SF4 SCS) Shutdown Cooling	Tier #	2	------
K2.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of the electrical power supplies to the following: • Pump motors	K/A #	205000K2.01	
	Importance Rating	3.6	------

Proposed Question: **# 8**

Unit 3 is operating at 100% RTP when the following conditions occur:

- Loss of Offsite Power
- ‘3B’ EDG mechanically fails
- Reactor Water Level lowers to (-) 130 inches

Given the conditions above, 3B RHR Pump ___**(1)**___ automatically start and 3C RHR Pump ___**(2)**___ automatically start.

A. (1) will NOT
(2) will

B. (1) will NOT
(2) will NOT

C. (1) will
(2) will

D. (1) will
(2) will NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate confuses 3B RHR Pump’s power supply as ‘3EB’ 4KV Shutdown Board or ‘3B’ EDG. Second part is incorrect plausible if the candidate confuses 3C RHR Pump’s power supply as ‘3EC’ 4KV Shutdown Board or ‘3C’ EDG.

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 3-OI-74, RHR System, Low Pressure Coolant Injection (LPCI) will initiate when Reactor Water Level reaches (-) 122 inches. Depending if normal power available (NVA) or Diesel Generator Volts available (DGVA), RHR Pumps will automatically sequence on. Given a Loss of Offsite Power has occurred, 3B RHR Pump will automatically start since it is powered by '3C' EDG. For second part, given that '3B' EDG has failed, '3C' RHR Pump will not automatically start. It not only lost normal power from 3EC 4KV Shutdown Board due to the Loss of Offsite Power, but remains de-energized since it does not have alternate power.

RO Level Justification: Tests the candidate's knowledge of the electrical power supplies to RHR Pumps in a Loss of Offsite Power, Accident signal condition that support Shutdown Cooling. 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)

3-OI-74, Rev. 132

3-OI-82, Rev. 153

OPL 171.044, Rev. 22

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044 Obj. 12c (As available)

Question Source:	Bank #	BFN 2104 #12	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

Question Cognitive Level:

Memory or Fundamental Knowledge

Comprehension or Analysis **X**

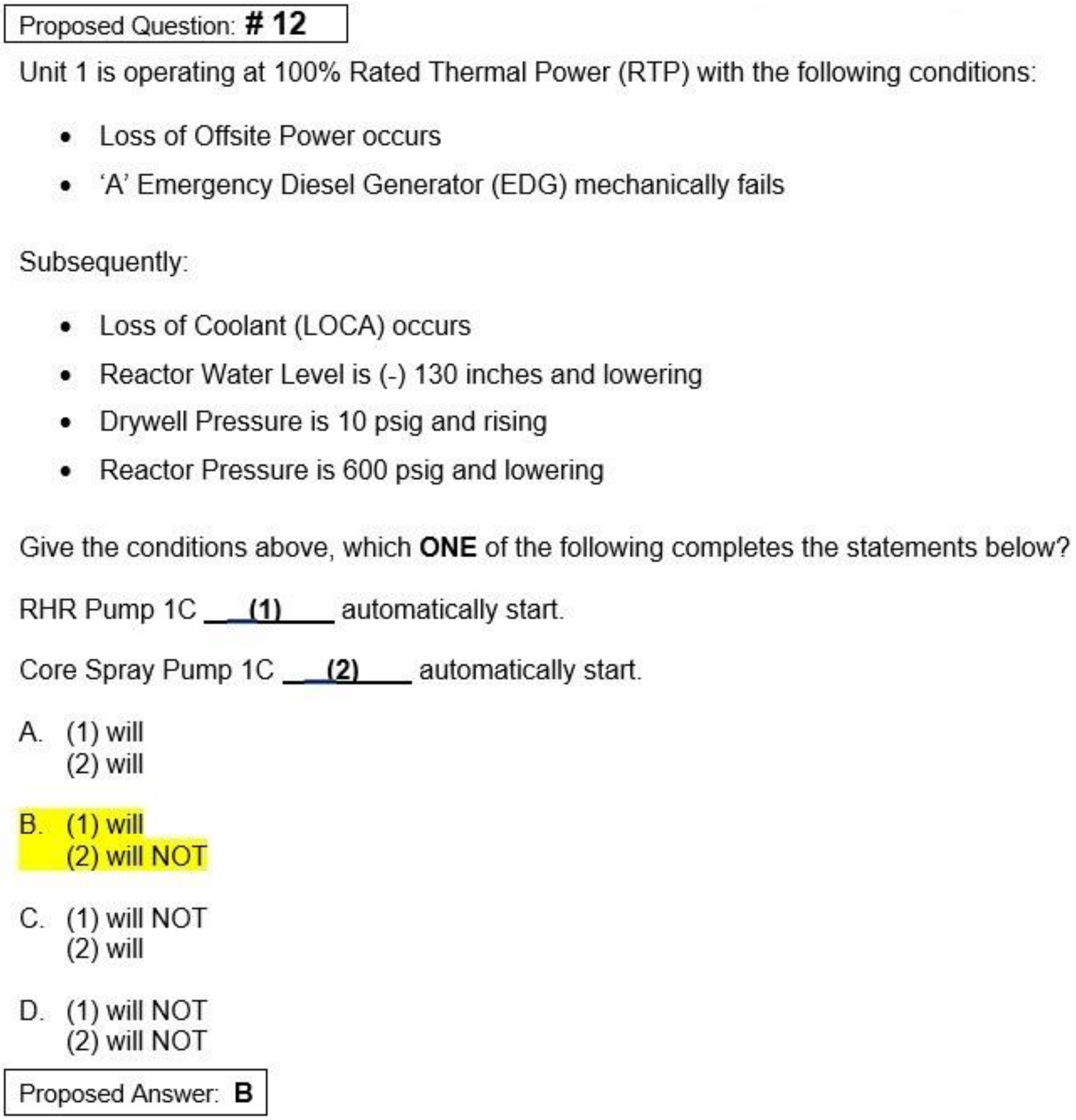
10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Copy of Bank Question:



Proposed Question: **# 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

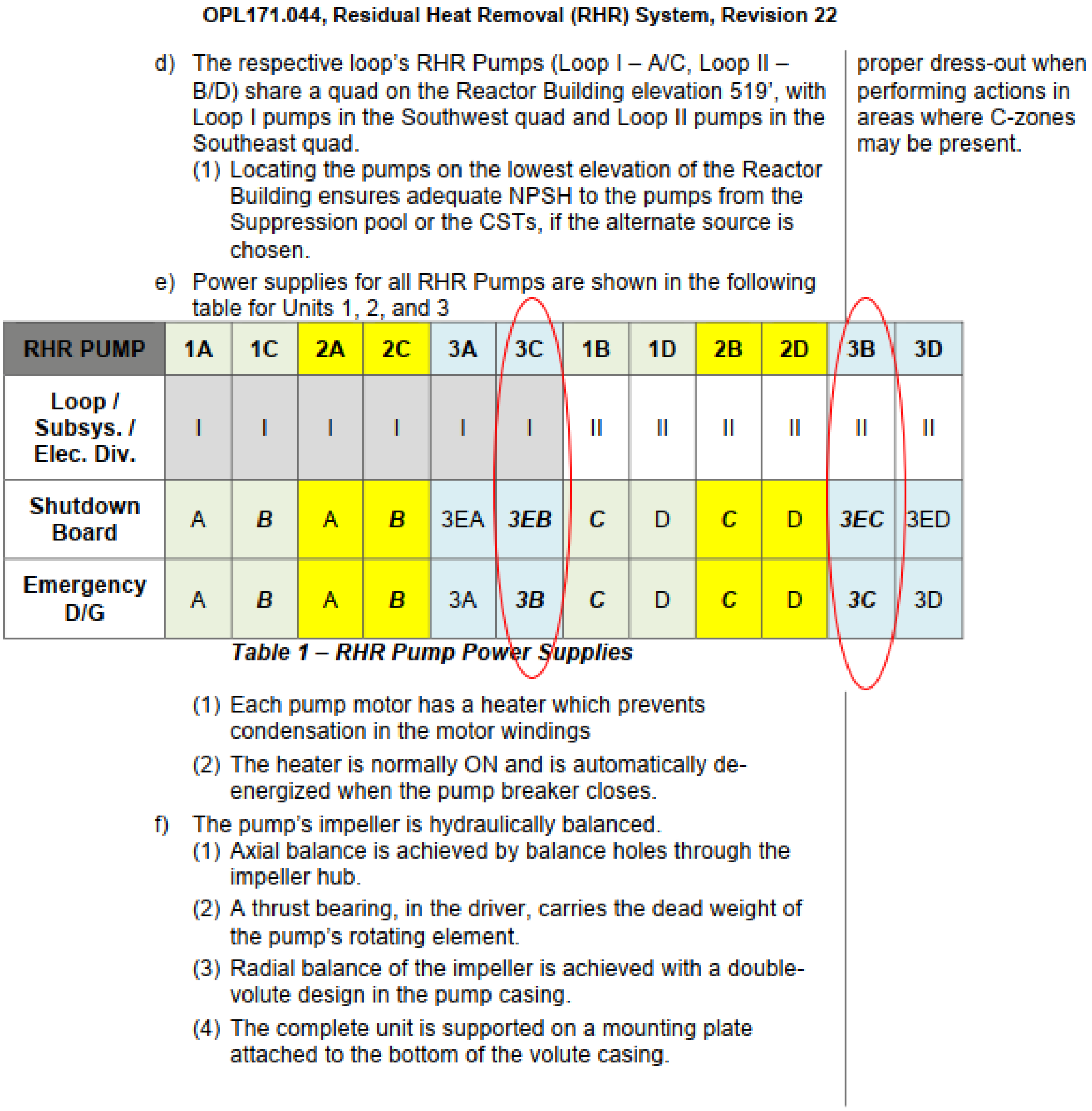
B. (1) will
(2) will NOT

C. (1) will NOT
(2) will

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

Proposed Answer: **B**

Excerpt from OPL171.044 Lesson Plan:



OPL171.044, Residual Heat Removal (RHR) System, Revision 22

d) The respective loop's RHR Pumps (Loop I – A/C, Loop II – B/D) share a quad on the Reactor Building elevation 519', with Loop I pumps in the Southwest quad and Loop II pumps in the Southeast quad.
   (1) Locating the pumps on the lowest elevation of the Reactor Building ensures adequate NPSH to the pumps from the Suppression pool or the CSTs, if the alternate source is chosen.

proper dress-out when performing actions in areas where C-zones may be present.

e) Power supplies for all RHR Pumps are shown in the following table for Units 1, 2, and 3

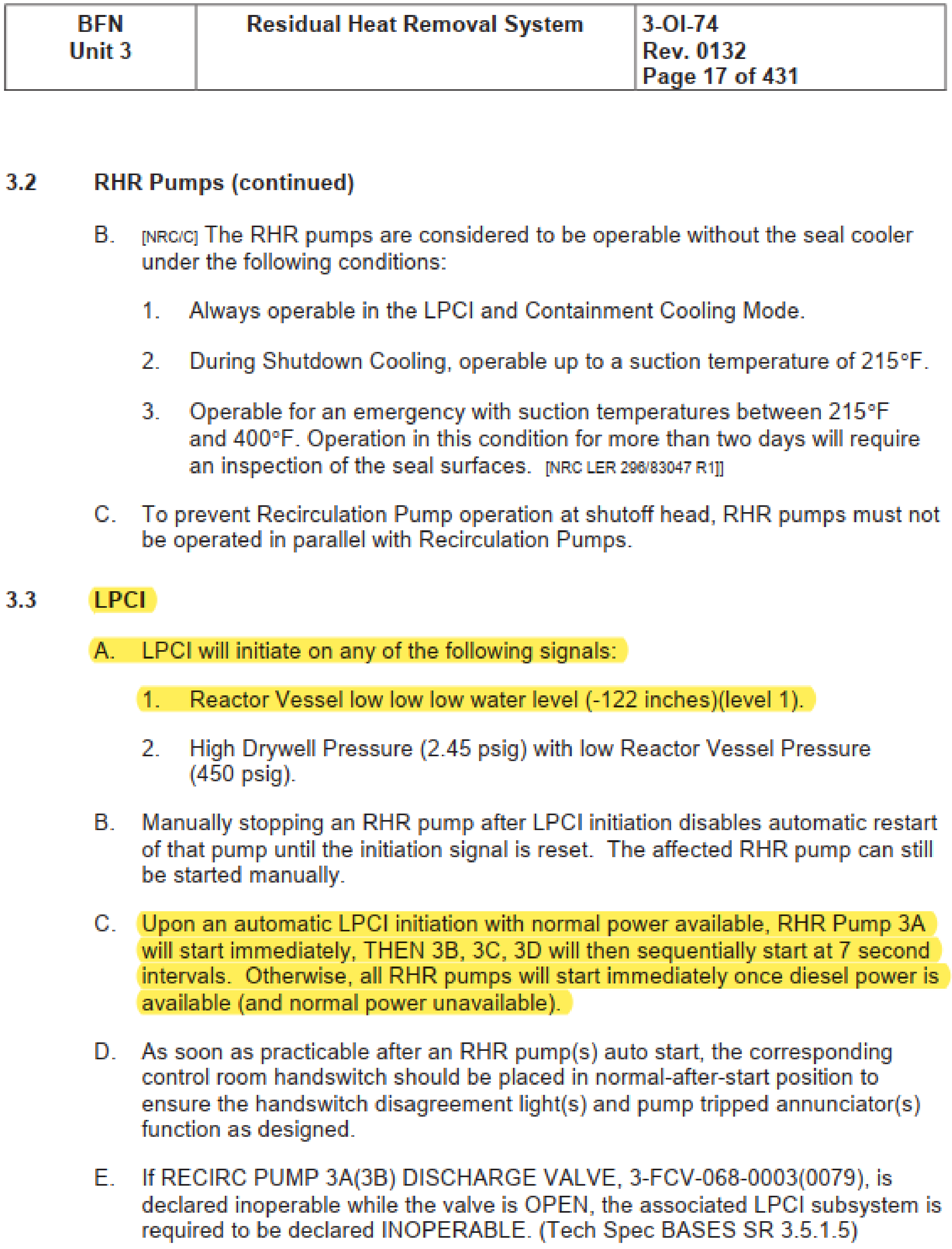
RHR PUMP	**1A**	**1C**	**2A**	**2C**	**3A**	**3C**	**1B**	**1D**	**2B**	**2D**	**3B**	**3D**
Loop / Subsys. / Elec. Div.	I	I	I	I	I	I	II	II	II	II	II	II
Shutdown Board	A	*B*	A	*B*	3EA	*3EB*	*C*	D	*C*	D	*3EC*	3ED
Emergency D/G	A	*B*	A	*B*	3A	*3B*	*C*	D	*C*	D	*3C*	3D

Table 1 – RHR Pump Power Supplies

   (1) Each pump motor has a heater which prevents condensation in the motor windings
   (2) The heater is normally ON and is automatically de-energized when the pump breaker closes.

f) The pump's impeller is hydraulically balanced.
   (1) Axial balance is achieved by balance holes through the impeller hub.
   (2) A thrust bearing, in the driver, carries the dead weight of the pump's rotating element.
   (3) Radial balance of the impeller is achieved with a double-volute design in the pump casing.
   (4) The complete unit is supported on a mounting plate attached to the bottom of the volute casing.

Excerpt from 3-OI-74:



BFN Unit 3	**Residual Heat Removal System**	**3-OI-74 Rev. 0132 Page 17 of 431**

3.2 RHR Pumps (continued)

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 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:

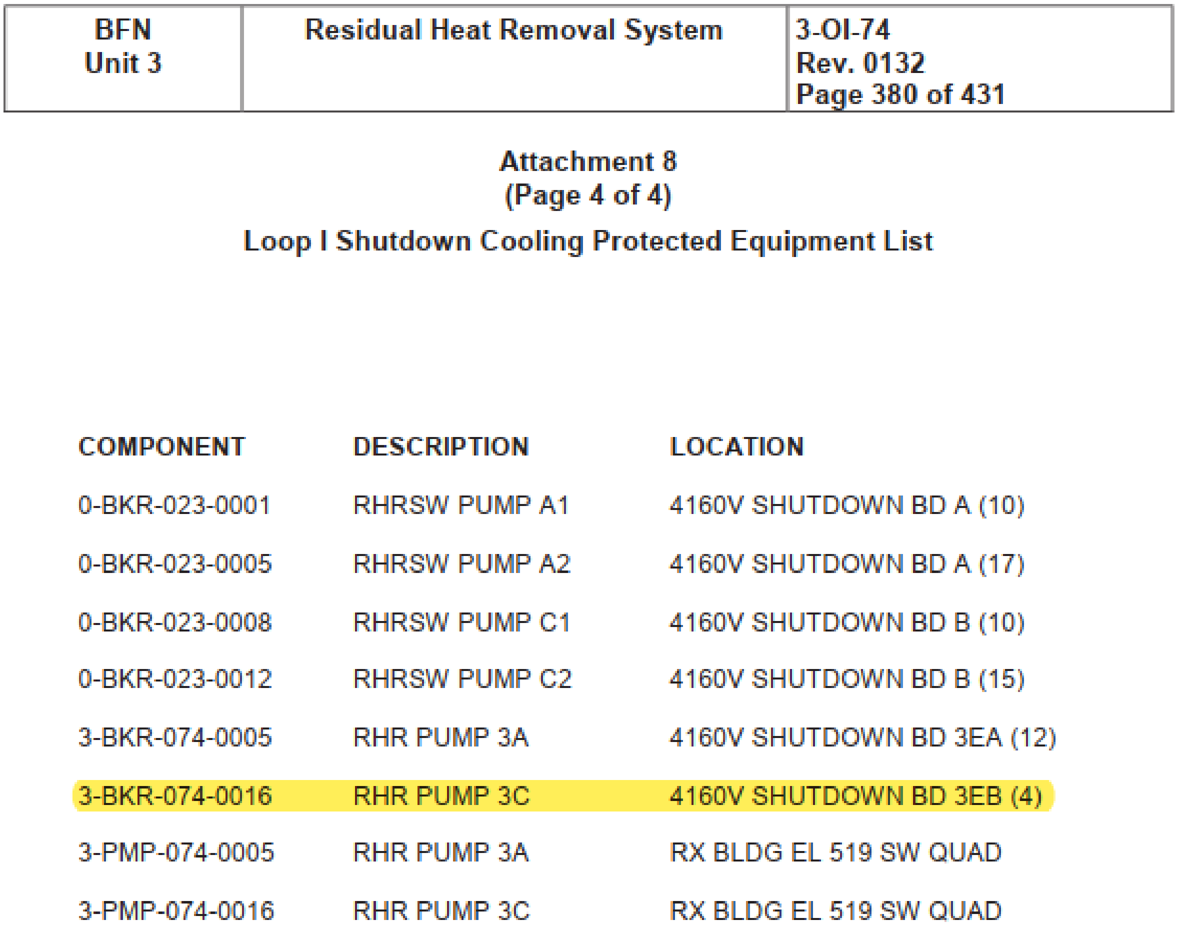
1. Reactor Vessel low low low water level (-122 inches)(level 1).
2. High Drywell Pressure (2.45 psig) with low Reactor Vessel Pressure (450 psig).

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 3A will start immediately, THEN 3B, 3C, 3D will then sequentially start at 7 second intervals. Otherwise, all RHR pumps will 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 handswitch should be placed in normal-after-start position to ensure the handswitch disagreement light(s) and pump tripped annunciator(s) function as designed.

E. If RECIRC PUMP 3A(3B) DISCHARGE VALVE, 3-FCV-068-0003(0079), is declared inoperable while the valve is OPEN, the associated LPCI subsystem is required to be declared INOPERABLE. (Tech Spec BASES SR 3.5.1.5)

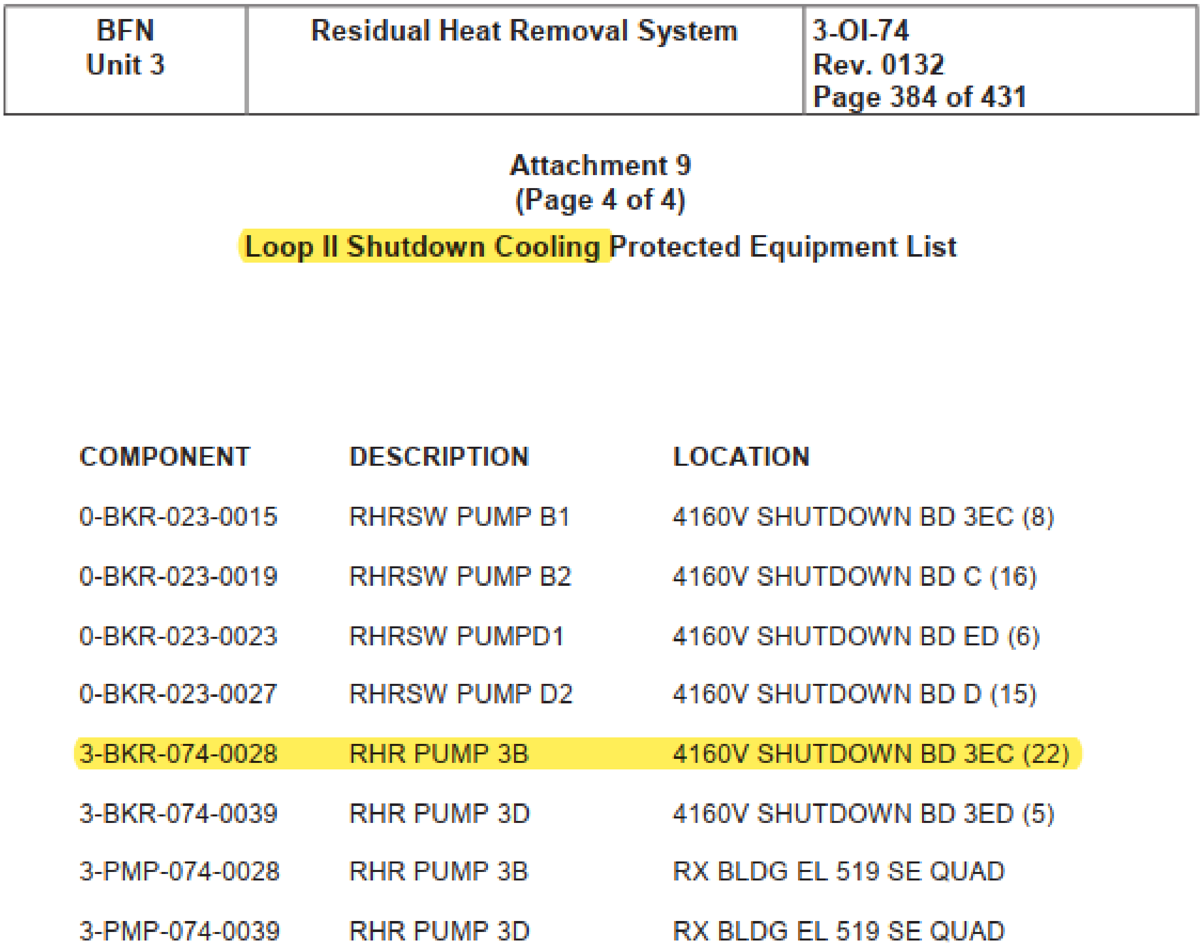


BFN Unit 3	Residual Heat Removal System	3-OI-74 Rev. 0132 Page 380 of 431

Attachment 8
(Page 4 of 4)

Loop I Shutdown Cooling Protected Equipment List

COMPONENT	DESCRIPTION	LOCATION
0-BKR-023-0001	RHRSW PUMP A1	4160V SHUTDOWN BD A (10)
0-BKR-023-0005	RHRSW PUMP A2	4160V SHUTDOWN BD A (17)
0-BKR-023-0008	RHRSW PUMP C1	4160V SHUTDOWN BD B (10)
0-BKR-023-0012	RHRSW PUMP C2	4160V SHUTDOWN BD B (15)
3-BKR-074-0005	RHR PUMP 3A	4160V SHUTDOWN BD 3EA (12)
3-BKR-074-0016	RHR PUMP 3C	4160V SHUTDOWN BD 3EB (4)
3-PMP-074-0005	RHR PUMP 3A	RX BLDG EL 519 SW QUAD
3-PMP-074-0016	RHR PUMP 3C	RX BLDG EL 519 SW QUAD

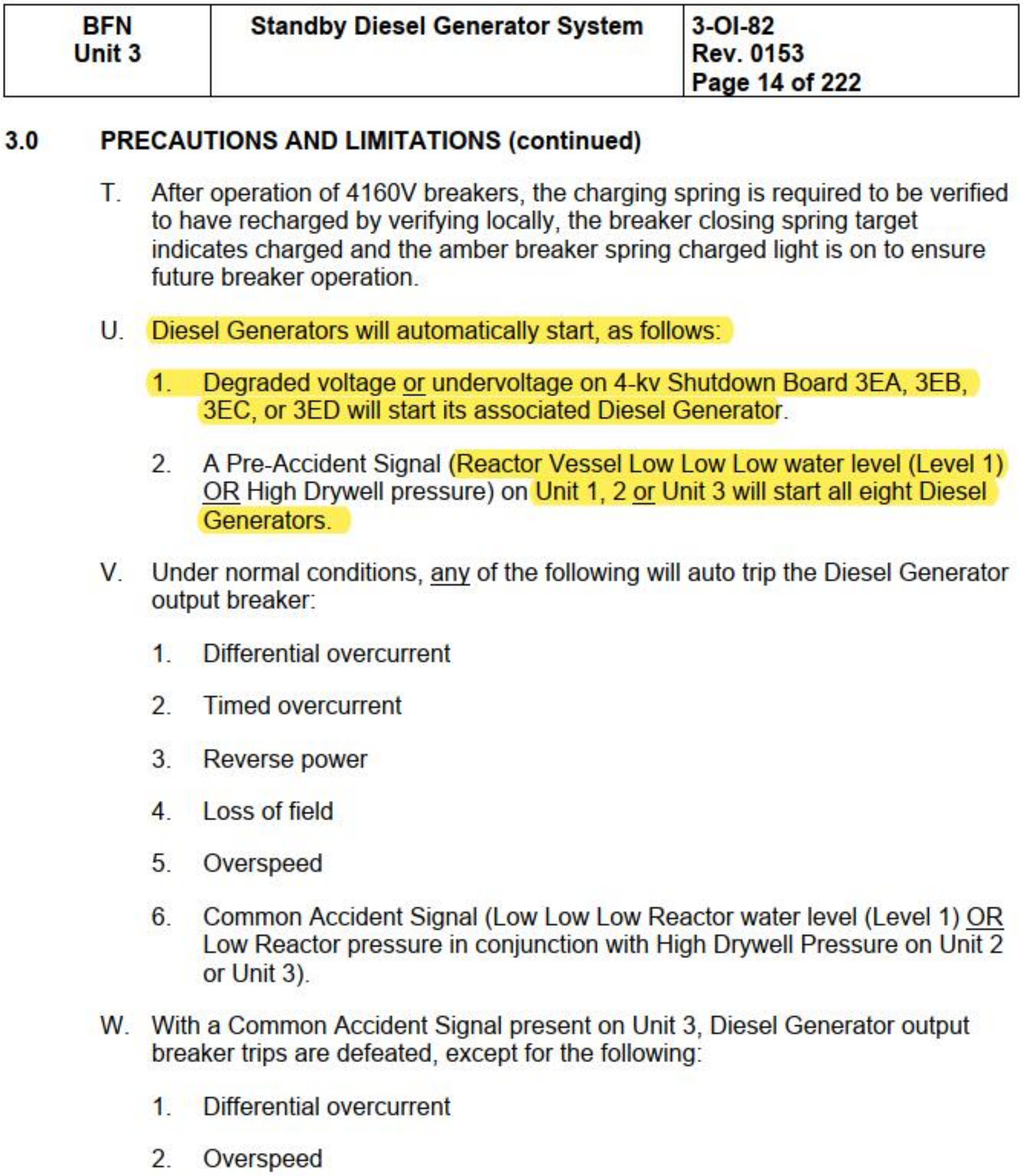


BFN Unit 3	Residual Heat Removal System	3-OI-74 Rev. 0132 Page 384 of 431

Attachment 9
(Page 4 of 4)
Loop II Shutdown Cooling Protected Equipment List

COMPONENT	DESCRIPTION	LOCATION
0-BKR-023-0015	RHRSW PUMP B1	4160V SHUTDOWN BD 3EC (8)
0-BKR-023-0019	RHRSW PUMP B2	4160V SHUTDOWN BD C (16)
0-BKR-023-0023	RHRSW PUMPD1	4160V SHUTDOWN BD ED (6)
0-BKR-023-0027	RHRSW PUMP D2	4160V SHUTDOWN BD D (15)
3-BKR-074-0028	RHR PUMP 3B	4160V SHUTDOWN BD 3EC (22)
3-BKR-074-0039	RHR PUMP 3D	4160V SHUTDOWN BD 3ED (5)
3-PMP-074-0028	RHR PUMP 3B	RX BLDG EL 519 SE QUAD
3-PMP-074-0039	RHR PUMP 3D	RX BLDG EL 519 SE QUAD

Excerpt from 3-OI-82:



BFN Unit 3	**Standby Diesel Generator System**	**3-OI-82 Rev. 0153 Page 14 of 222**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

T. 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.

U. Diesel Generators will automatically start, as follows:

1. Degraded voltage or undervoltage on 4-kv Shutdown Board 3EA, 3EB, 3EC, or 3ED will start its associated Diesel Generator.
2. A Pre-Accident Signal (Reactor Vessel Low Low Low water level (Level 1) OR High Drywell pressure) on Unit 1, 2 or Unit 3 will start all eight Diesel Generators.

V. 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 (Level 1) OR Low Reactor pressure in conjunction with High Drywell Pressure on Unit 2 or Unit 3).

W. With a Common Accident Signal present on Unit 3, Diesel Generator output breaker trips are defeated, except for the following:

1. Differential overcurrent
2. Overspeed

Examination Outline Cross-reference:

295004 (APE 4) Partial or Complete Loss of D.C. Power / 6

AK1.02 (10CFR 55.41.10)

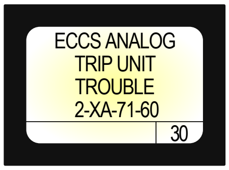
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Partial or Complete Loss of DC Power:

- Redundant DC power supplies

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295004AK1.02	
Importance Rating	3.8	------

Proposed Question: **# 9**

Unit 2 is operating at 100% RTP with the following plant conditions:



- ECCS ANALOG TRIP UNIT TROUBLE (2-9-3E, Window 30) alarms
- A malfunction causes a failure of Division 1 ECCS ATU Inverter

Which **ONE** of the following accurately describes the effects of this loss on the HPCI **AND** RCIC Systems?

HPCI is ___**(1)**___ and RCIC is___**(2)**___.

A. (1) OPERABLE
(2) OPERABLE

B. (1) OPERABLE
(2) INOPERABLE

C. (1) INOPERABLE
(2) OPERABLE

D. (1) INOPERABLE
(2) INOPERABLE

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B).* Second part is incorrect but plausible in that RCIC (HPCI) is powered by the Division I(II) ECCS ATU which is powered by 250V DC RMOV Board 2A(2B), and the given conditions will result in a loss of power to the RCIC flow controller, rendering RCIC INOPERABLE. HPCI remains OPERABLE. The Divisional Power scheme at Browns Ferry is confusing and the power supplies to ECCS loads are often mixed up.

B **CORRECT**: (*See attached*) In accordance with the given alarm 2-9-3E, Window 30, 2-AOI-57-11 is referenced for a loss of power to the ECCS ATU Panel. In accordance with 2-AOI-57-11, Loss of Power to an ECCS ATU Panel/ECCS Inverter, Division I ECCS ATU Inverter supplies power specifically to RCIC, not HPCI. HPCI is powered by Division II, and is therefore still OPERABLE. 250V DC RMOV Board 2B breaker 1B1 is still closed and supplying power to the Aux Instrument Room Panel 9-81 ATUs, therefore isolation logic is still available and OPERABLE. For second part, in accordance with 2-AOI-57-11, since RCIC is powered by the Division I ECCS ATU Inverter, a loss of power will render the RCIC flow controller INOPERABLE. Additionally, RCIC parameter indications are lost and along with the loss of the RCIC flow controller, RCIC is INOPERABLE.

C INCORRECT: First part is incorrect but plausible in that HPCI is powered by the Division II ECCS ATU Inverter, not Division I like RCIC which is INOPERABLE. Therefore, given the conditions in the stem, HPCI is OPERABLE. 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 effect of a loss of the Division I ECCS ATU Inverter will have on the OPERABILITY of HPCI and 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 specific knowledge and its meaning to predict the correct outcome. The divisional power scheme at BFN is complex, and candidate must use the information provided concerning the loss of one division of ECCS ATU Power Supplies and determine the effect on the operability of HPCI and RCIC.

Technical Reference(s):	2-AOI-57-11, Rev.17	(Attach if not previously provided)
	2-ARP-9-3E, Rev. 32	
	OPL171.102, Rev.11	

Proposed references to be provided to applicants during examination: **ECCS ANALOG TRIP UNIT TROUBLE (2-9-3E, Window 30)**

Learning Objective: OPL171.102 Obj. 6b (As available)

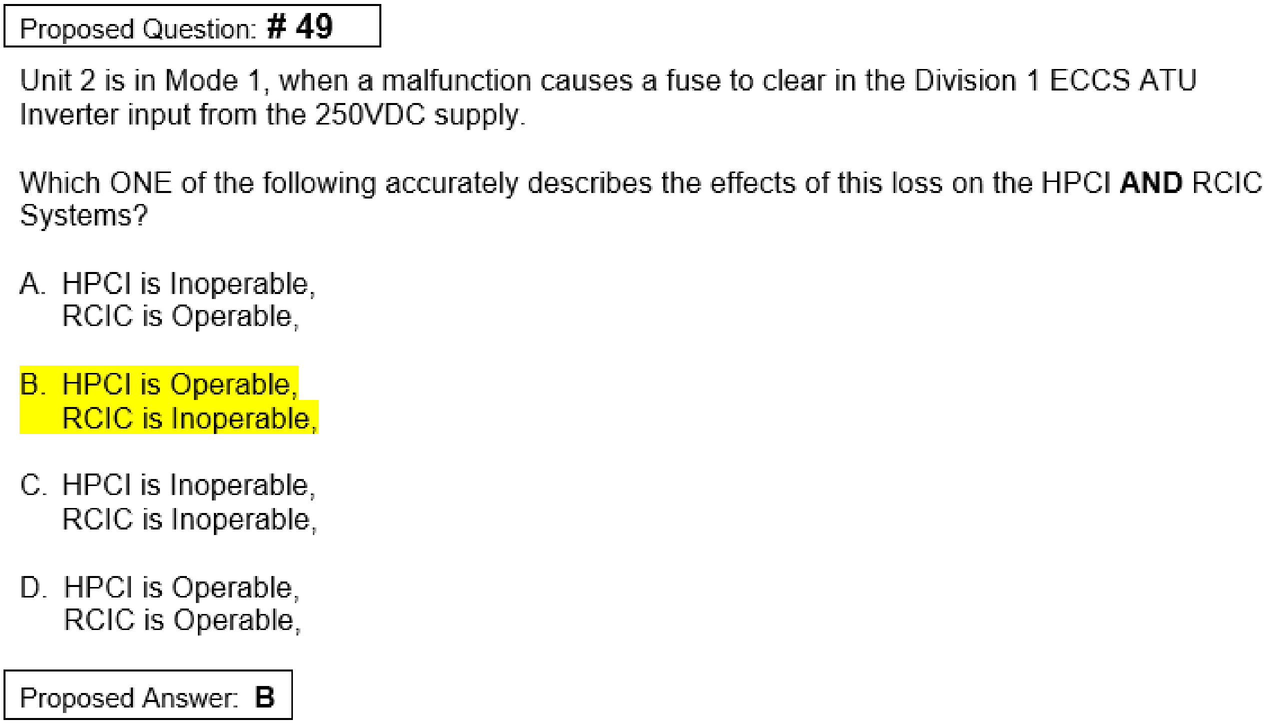
Question Source:	Bank #	**BFN 1006 AUDIT #49**	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

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

Comments:

Copy of Bank Question:



Proposed Question: **# 49**

Unit 2 is in Mode 1, when a malfunction causes a fuse to clear in the Division 1 ECCS ATU Inverter input from the 250VDC supply.

Which ONE of the following accurately describes the effects of this loss on the HPCI **AND** RCIC Systems?

A. HPCI is Inoperable,
RCIC is Operable,

B. HPCI is Operable,
RCIC is Inoperable,

C. HPCI is Inoperable,
RCIC is Inoperable,

D. HPCI is Operable,
RCIC is Operable,

Proposed Answer: **B**

Excerpts from 2-ARP-9-3E:

BFN **Unit 2**	**Panel 9-3** **2-XA-55-3E**	**2-ARP-9-3E** **Rev. 0032** **Page 34 of 41**

ECCS ANALOG
TRIP UNIT
TROUBLE
2-XA-71-60

30

(Page 1 of 2)

Sensor/Trip Point:

Relay 74-71-60-1X1	Relay 74-71-60-2X1
Relay 74-71-60-1X2	Relay 74-71-60-2X2
Relay 74-71-60-1AX1	Relay 74-71-60-2AX1
Relay 74-71-60-1AX2	Relay 74-71-60-2AX2
Relay 74-71-601	Relay 74-71-602

Contact ALC from Inverter

Sensor Location:	Trip UNIT: Panel 9-81 and 9-82 Unit 2 Aux. Inst.Room	INVERTER: Shutdown Board Rooms C and D

Probable Cause:

A. Calibration of master trip unit(s) in progress.
B. Card out of card file.
C. Failure of a master or slave trip unit.
D. Power supply failure.
E. Inverter trouble.
F. Loss of 2A or 2B 250VDC RMOV BD.
G. Valid alarm.
H. Loss of power to cabinet fan.
I. Purging the Drywell.

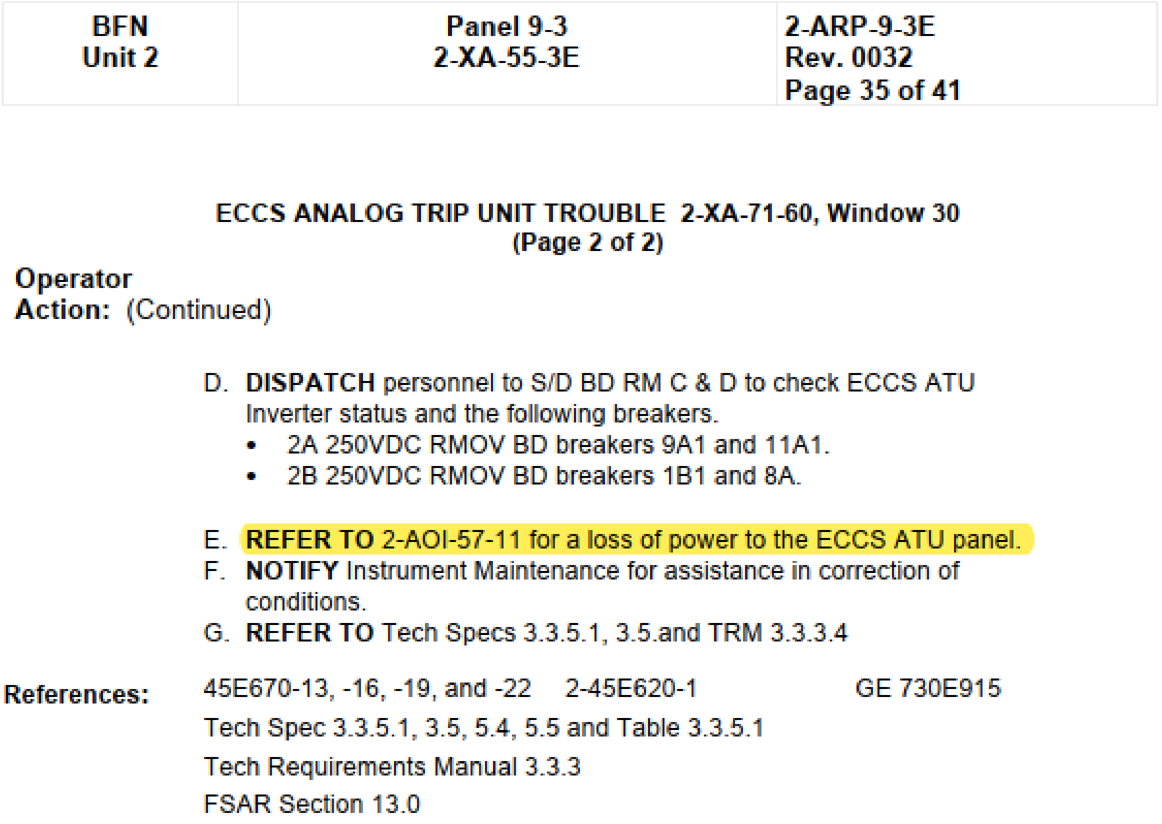
Operator Action:

A. This alarm in conjunction with HPCI or RCIC logic power failure alarm may indicate a loss of the 250VDC RMOV BD.

B. **DISPATCH** personnel with key to Aux. Inst. Room, Panels 9-81 and 9-82 to check:
  - Trip unit being calibrated or calibration unit power switch in ON position.
  - Fuse 2-FU2-071-0060/1A, DDD block, and 2-FU1-71-60/1 BBA block, in Panel 9-81,.
  - Fuse 2-FU2-071-0060/2A, DDD block, and 2-FU1-71-60/2 BBA block, in Panel 9-82,.
  - Fuse 2-FU1-071-0601A (cabinet fan) in Panel 9-81.
  - Fuse 2-FU1-071-0602A (cabinet fan) in Panel 9-82.

C. **IF** fuse 2-FU1-256-1F has cleared, **THEN** **REQUEST** EM to verify capacitors and SCRs are good prior to replacing the fuse.

Continued on Next Page



BFN **Unit 2**	**Panel 9-3** **2-XA-55-3E**	**2-ARP-9-3E** **Rev. 0032** **Page 35 of 41**

ECCS ANALOG TRIP UNIT TROUBLE 2-XA-71-60, Window 30
(Page 2 of 2)

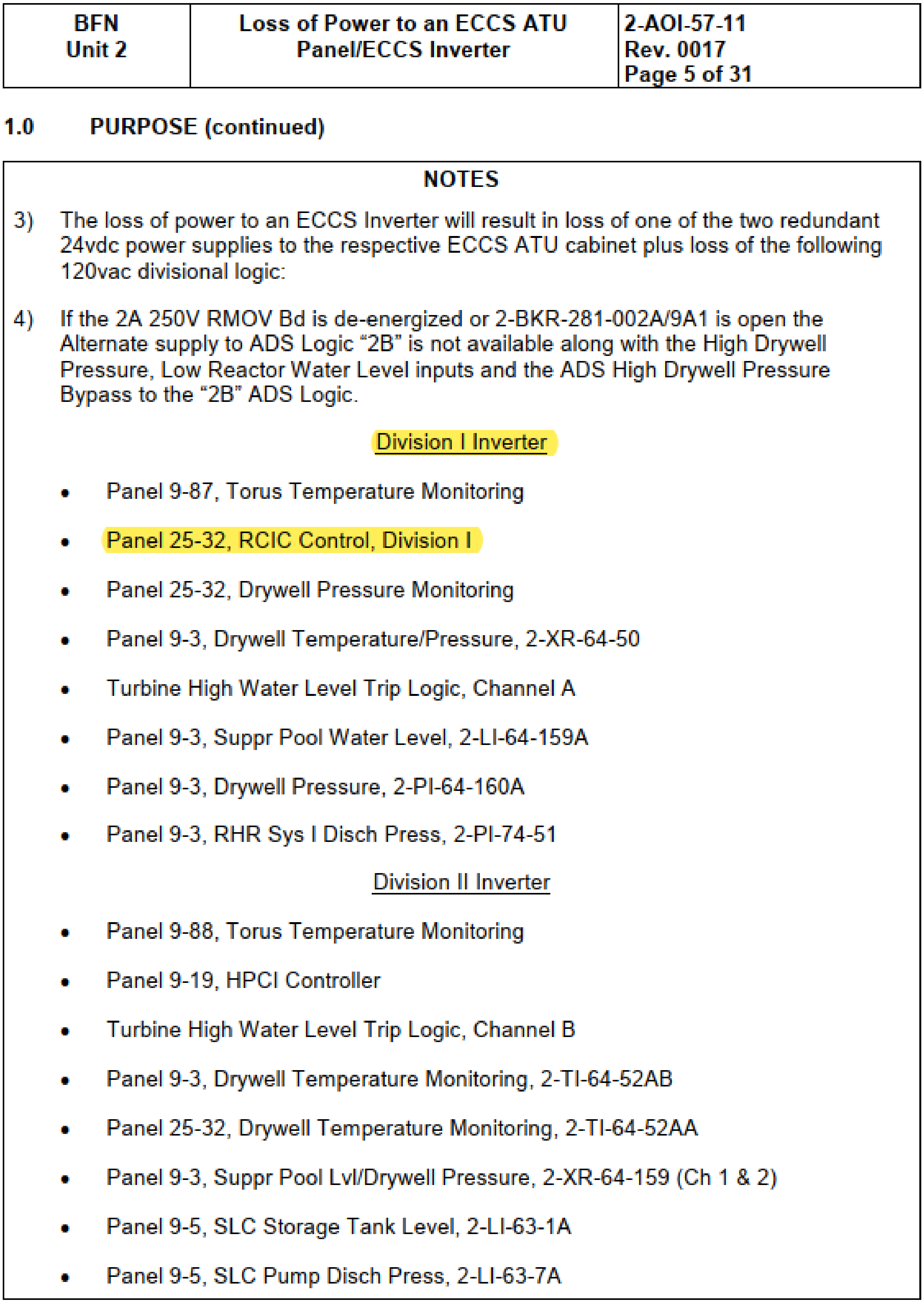
Operator Action: (Continued)

D. **DISPATCH** personnel to S/D BD RM C & D to check ECCS ATU Inverter status and the following breakers.
- 2A 250VDC RMOV BD breakers 9A1 and 11A1.
- 2B 250VDC RMOV BD breakers 1B1 and 8A.

E. **REFER TO** 2-AOI-57-11 for a loss of power to the ECCS ATU panel.
F. **NOTIFY** Instrument Maintenance for assistance in correction of conditions.
G. **REFER TO** Tech Specs 3.3.5.1, 3.5.and TRM 3.3.3.4

References: 45E670-13, -16, -19, and -22 2-45E620-1 GE 730E915
Tech Spec 3.3.5.1, 3.5, 5.4, 5.5 and Table 3.3.5.1
Tech Requirements Manual 3.3.3
FSAR Section 13.0

Excerpts from 2-AOI-57-11:



BFN **Unit 2**	**Loss of Power to an ECCS ATU Panel/ECCS Inverter**	**2-AOI-57-11** **Rev. 0017** **Page 5 of 31**

1.0 PURPOSE (continued)

NOTES

3) The loss of power to an ECCS Inverter will result in loss of one of the two redundant 24vdc power supplies to the respective ECCS ATU cabinet plus loss of the following 120vac divisional logic:

4) If the 2A 250V RMOV Bd is de-energized or 2-BKR-281-002A/9A1 is open the Alternate supply to ADS Logic “2B” is not available along with the High Drywell Pressure, Low Reactor Water Level inputs and the ADS High Drywell Pressure Bypass to the “2B” ADS Logic.

Division I Inverter

- Panel 9-87, Torus Temperature Monitoring
- Panel 25-32, RCIC Control, Division I
- Panel 25-32, Drywell Pressure Monitoring
- Panel 9-3, Drywell Temperature/Pressure, 2-XR-64-50
- Turbine High Water Level Trip Logic, Channel A
- Panel 9-3, Suppr Pool Water Level, 2-LI-64-159A
- Panel 9-3, Drywell Pressure, 2-PI-64-160A
- Panel 9-3, RHR Sys I Disch Press, 2-PI-74-51

Division II Inverter

- Panel 9-88, Torus Temperature Monitoring
- Panel 9-19, HPCI Controller
- Turbine High Water Level Trip Logic, Channel B
- Panel 9-3, Drywell Temperature Monitoring, 2-TI-64-52AB
- Panel 25-32, Drywell Temperature Monitoring, 2-TI-64-52AA
- Panel 9-3, Suppr Pool Lvl/Drywell Pressure, 2-XR-64-159 (Ch 1 & 2)
- Panel 9-5, SLC Storage Tank Level, 2-LI-63-1A
- Panel 9-5, SLC Pump Disch Press, 2-LI-63-7A

BFN **Unit 2**	**Loss of Power to an ECCS ATU Panel/ECCS Inverter**	**2-AOI-57-11** **Rev. 0017** **Page 12 of 31**

4.2 Subsequent Action (continued)

NOTES

1) Refer to the FPR/FPRM for entry into applicable LCOs due to loss of instrumentation and/or electrical breakers.

2) This procedure is based on no safety related systems out of service. Additional Tech Spec LCOs may be imposed if an active Tech Spec LCO was in effect prior to initiation of this AOI.

3) Upon a concurrent loss of the Division II Inverter and the ATU Panel 9-82, REFER TO Technical Specification 3.5.1 condition D for additional LCOs that will be applicable.

[3] **IF** Division I(II) ECCS ATU Panel 9-81(9-82) is lost AND Core Spray Loop I(II) Injection is required, **THEN**

ENSURE Reactor Pressure is Less Than 450 psig AND **DEFEAT** Rx Low Pressure Interlocks. REFER TO 2-OI-75.

[4] **IF** Division I ECCS ATU Panel 9-81 is lost REFER TO appropriate ARPs for Suppression Chamber Vacuum relief valves.

NOTE

Loss of the Div I Inverter will cause the RCIC Flow Controller to be INOP,but the alarm will not annunciate. Performance of the following step will restore RCIC controller function but will not allow RCIC to be declared operable.

[5] **IF** ECCS Div I inverter is lost AND it is desired to keep RCIC functional, **THEN**

PLACE 2-XS-256-1 ATU BUS MANUAL TRANSFER on Panel 2-25-32 in ALT.

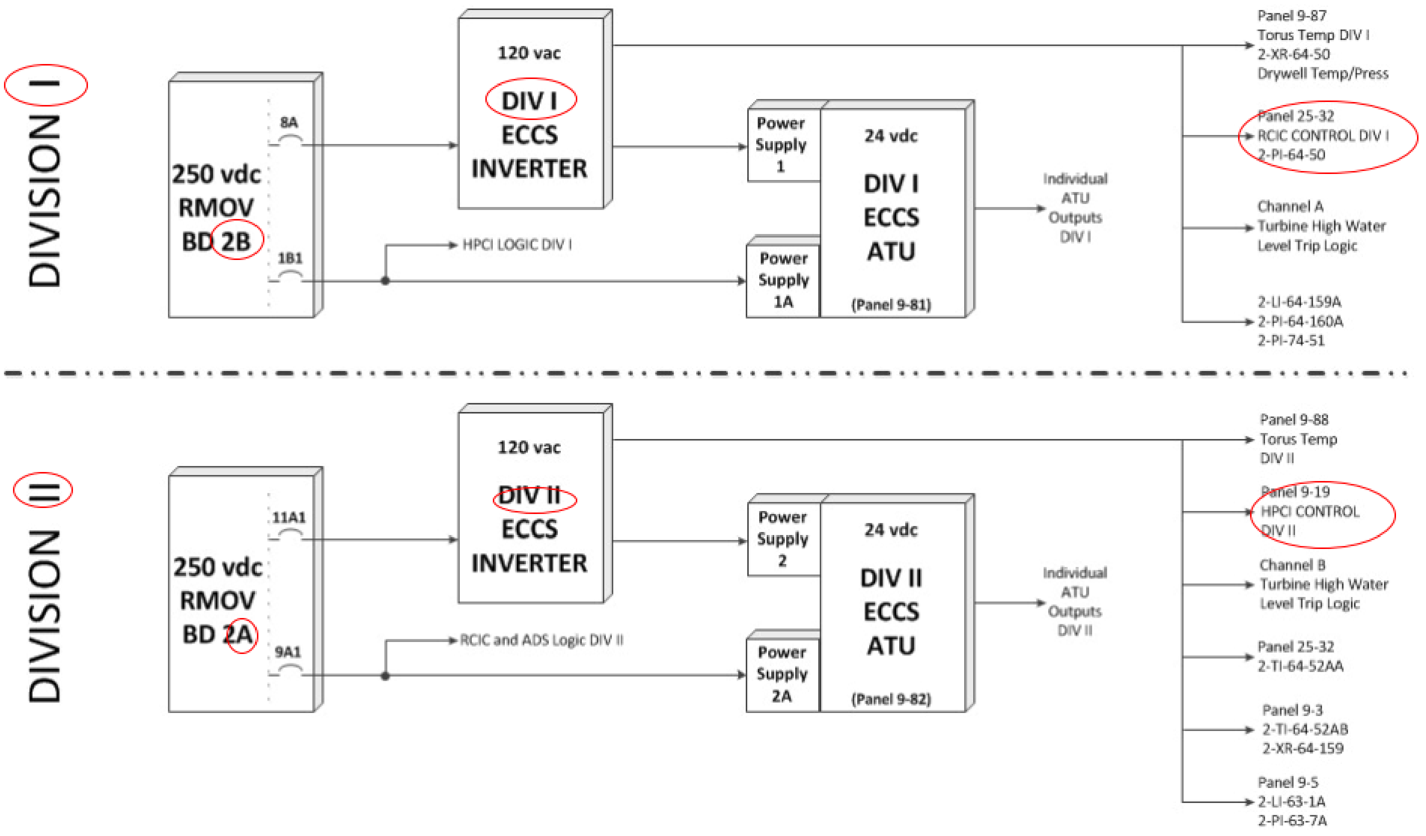
[6] **PERFORM** an inspection of the following for abnormalities:

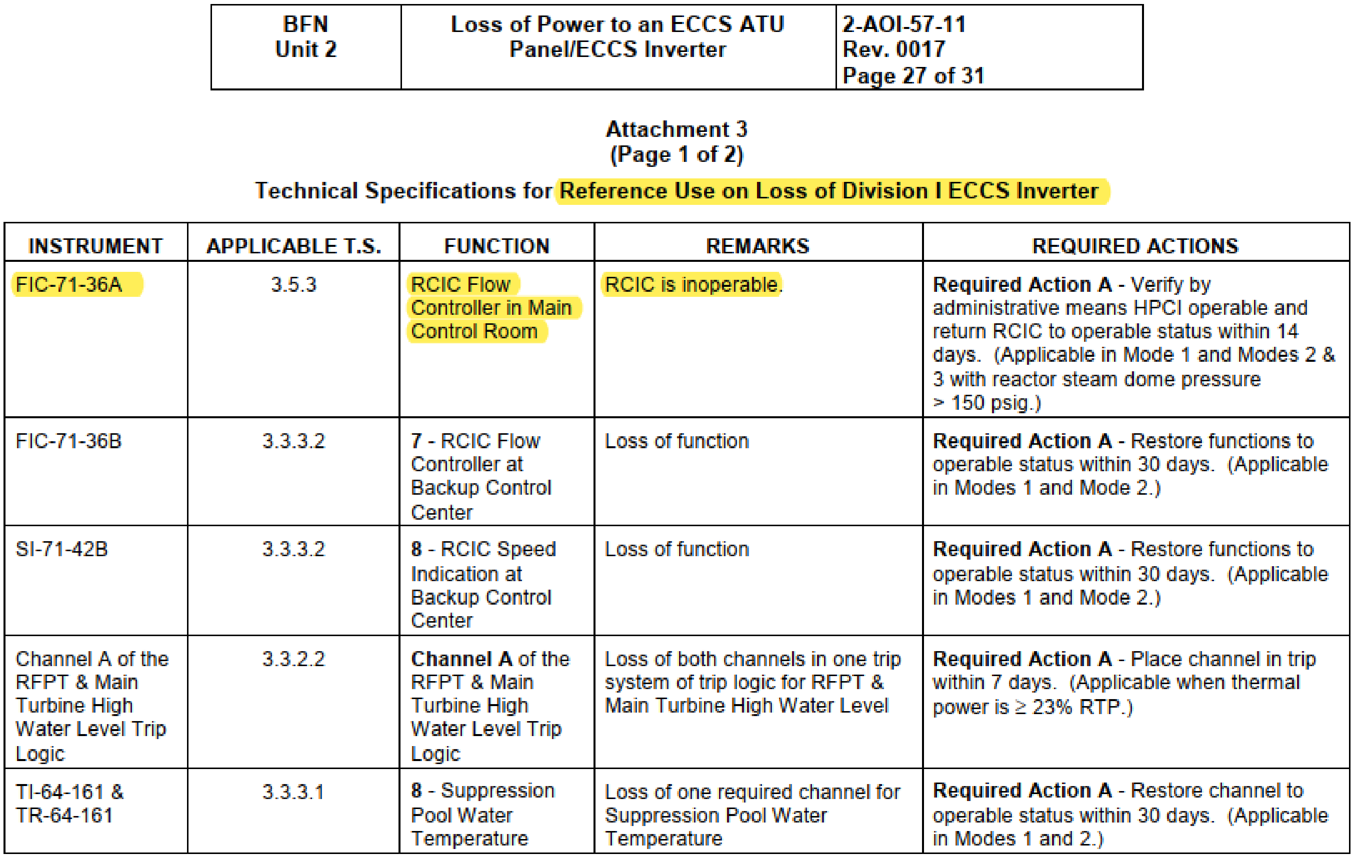
- ECCS ATU Panel 9-81(82) in the Aux Instrument Room.
- ECCS Div I(II) inverter behind 250V RMOV board 2B(2A).
- 250V RMOV board 2B(2A) compartments 1B1 & 8A(9A1 & 11A1).
- Fuse 2-FU2-071-0060/1A, and 2-FU1-71-60/1, in Panel 9-81.
- Fuse 2-FU2-071-0060/2A, and 2-FU1-71-60/2, in Panel 9-82.

BFN **Unit 2**	**Loss of Power to an ECCS ATU Panel/ECCS Inverter**	**2-AOI-57-11** **Rev. 0017** **Page 31 of 31**

Attachment 5
(Page 1 of 1)

Redundant Power Supplies





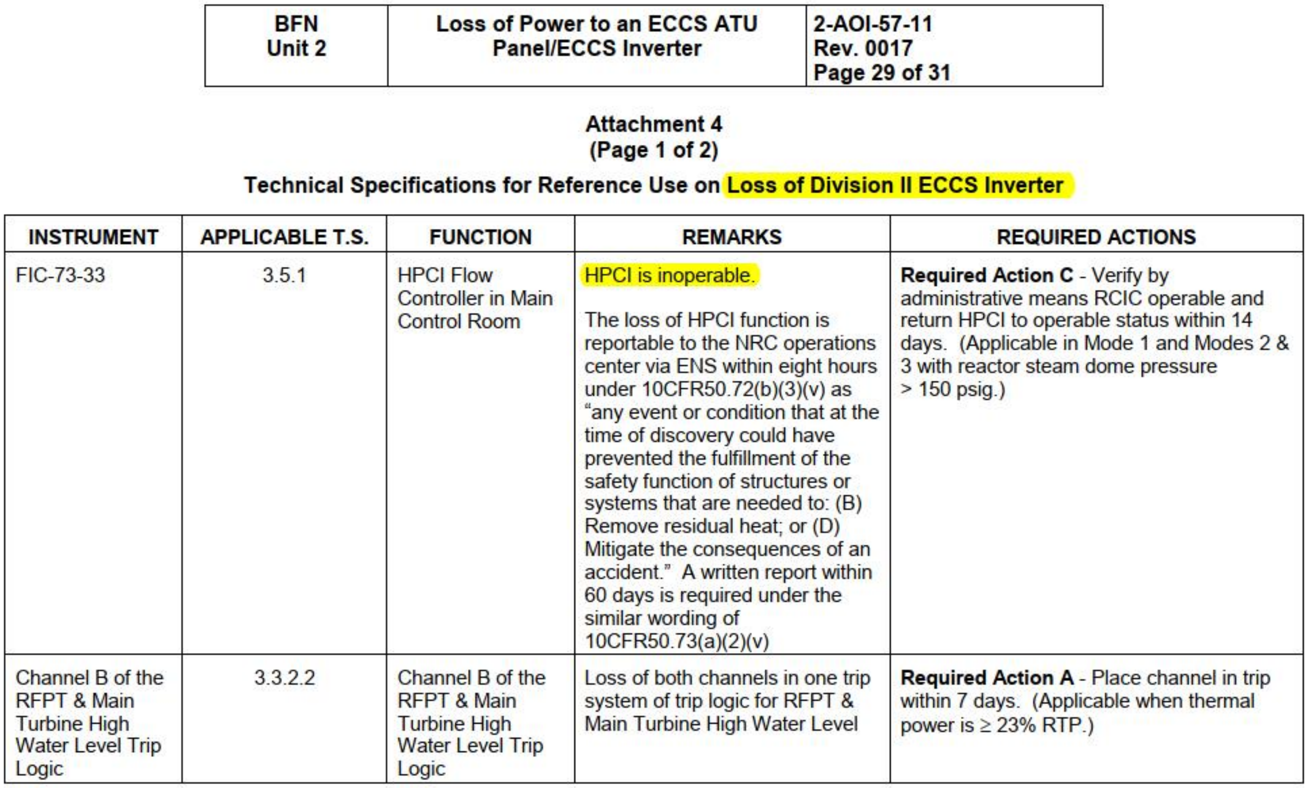
BFN Unit 2	**Loss of Power to an ECCS ATU Panel/ECCS Inverter**	**2-AOI-57-11 Rev. 0017 Page 27 of 31**

Attachment 3
(Page 1 of 2)

Technical Specifications for Reference Use on Loss of Division I ECCS Inverter

INSTRUMENT	APPLICABLE T.S.	FUNCTION	REMARKS	REQUIRED ACTIONS
FIC-71-36A	3.5.3	RCIC Flow Controller in Main Control Room	RCIC is inoperable.	**Required Action A** - Verify by administrative means HPCI operable and return RCIC to operable status within 14 days. (Applicable in Mode 1 and Modes 2 & 3 with reactor steam dome pressure > 150 psig.)
FIC-71-36B	3.3.3.2	**7** - RCIC Flow Controller at Backup Control Center	Loss of function	**Required Action A** - Restore functions to operable status within 30 days. (Applicable in Modes 1 and Mode 2.)
SI-71-42B	3.3.3.2	**8** - RCIC Speed Indication at Backup Control Center	Loss of function	**Required Action A** - Restore functions to operable status within 30 days. (Applicable in Modes 1 and Mode 2.)
Channel A of the RFPT & Main Turbine High Water Level Trip Logic	3.3.2.2	**Channel A** of the RFPT & Main Turbine High Water Level Trip Logic	Loss of both channels in one trip system of trip logic for RFPT & Main Turbine High Water Level	**Required Action A** - Place channel in trip within 7 days. (Applicable when thermal power is ≥ 23% RTP.)
TI-64-161 & TR-64-161	3.3.3.1	**8** - Suppression Pool Water Temperature	Loss of one required channel for Suppression Pool Water Temperature	**Required Action A** - Restore channel to operable status within 30 days. (Applicable in Modes 1 and 2.)

Supports Distractors C(1), D(1):



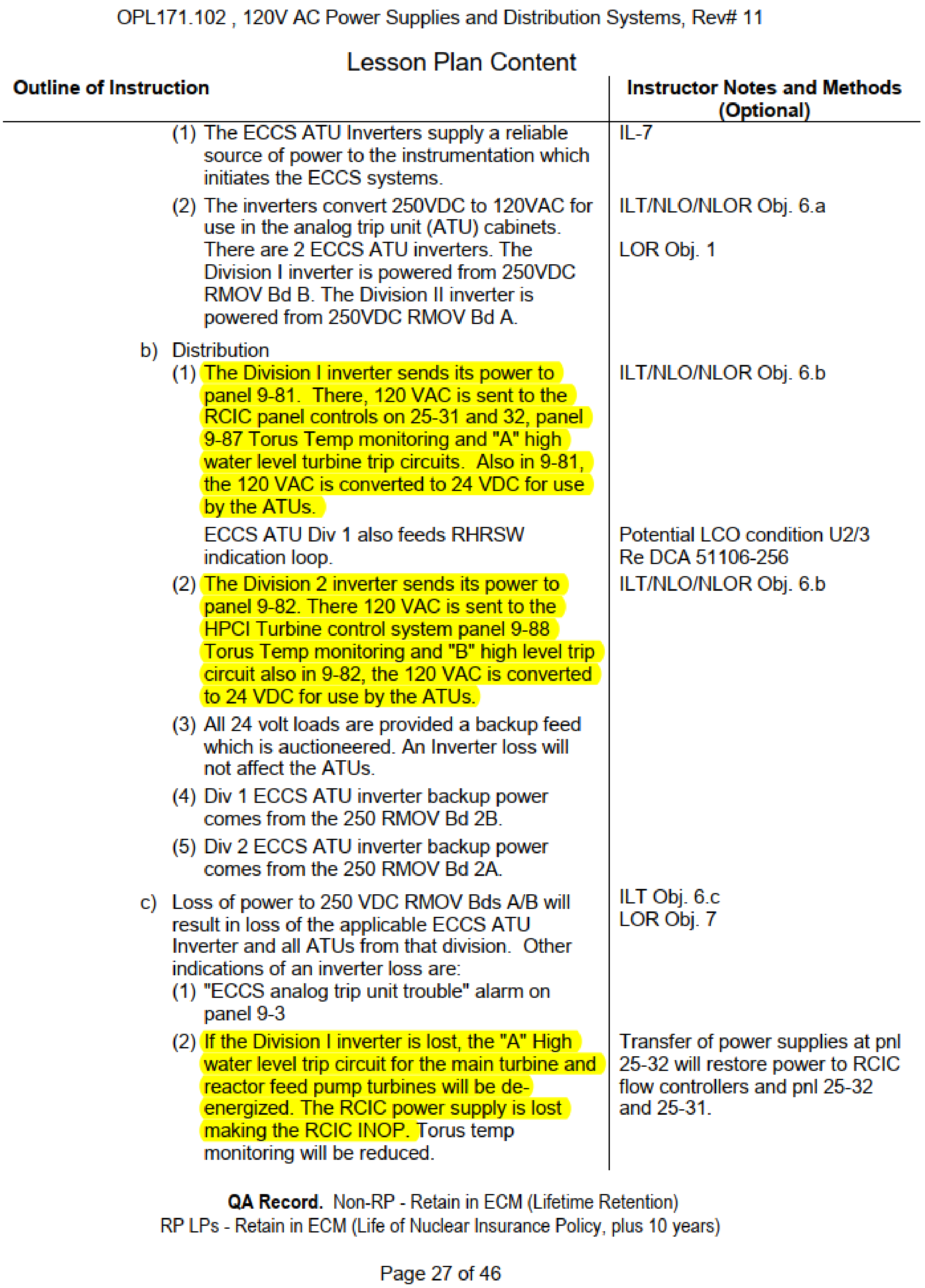
BFN Unit 2	**Loss of Power to an ECCS ATU Panel/ECCS Inverter**	**2-AOI-57-11 Rev. 0017 Page 29 of 31**

Attachment 4
(Page 1 of 2)

Technical Specifications for Reference Use on Loss of Division II ECCS Inverter

INSTRUMENT	APPLICABLE T.S.	FUNCTION	REMARKS	REQUIRED ACTIONS
FIC-73-33	3.5.1	HPCI Flow Controller in Main Control Room	HPCI is inoperable. The loss of HPCI function is reportable to the NRC operations center via ENS within eight hours under 10CFR50.72(b)(3)(v) as “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: (B) Remove residual heat; or (D) Mitigate the consequences of an accident.” A written report within 60 days is required under the similar wording of 10CFR50.73(a)(2)(v)	**Required Action C** - Verify by administrative means RCIC operable and return HPCI to operable status within 14 days. (Applicable in Mode 1 and Modes 2 & 3 with reactor steam dome pressure > 150 psig.)
Channel B of the RFPT & Main Turbine High Water Level Trip Logic	3.3.2.2	Channel B of the RFPT & Main Turbine High Water Level Trip Logic	Loss of both channels in one trip system of trip logic for RFPT & Main Turbine High Water Level	**Required Action A** - Place channel in trip within 7 days. (Applicable when thermal power is ≥ 23% RTP.)

Excerpts from OPL171.102 Lesson Plan:



OPL171.102 , 120V AC Power Supplies and Distribution Systems, Rev# 11

Lesson Plan Content

Outline of Instruction	**Instructor Notes and Methods (Optional)**
(1) The ECCS ATU Inverters supply a reliable source of power to the instrumentation which initiates the ECCS systems.	IL-7
(2) The inverters convert 250VDC to 120VAC for use in the analog trip unit (ATU) cabinets. There are 2 ECCS ATU inverters. The Division I inverter is powered from 250VDC RMOV Bd B. The Division II inverter is powered from 250VDC RMOV Bd A.	ILT/NLO/NLOR Obj. 6.a LOR Obj. 1
b) Distribution	
(1) The Division I inverter sends its power to panel 9-81. There, 120 VAC is sent to the RCIC panel controls on 25-31 and 32, panel 9-87 Torus Temp monitoring and "A" high water level turbine trip circuits. Also in 9-81, the 120 VAC is converted to 24 VDC for use by the ATUs.	ILT/NLO/NLOR Obj. 6.b
ECCS ATU Div 1 also feeds RHRSW indication loop.	Potential LCO condition U2/3 Re DCA 51106-256
(2) The Division 2 inverter sends its power to panel 9-82. There 120 VAC is sent to the HPCI Turbine control system panel 9-88 Torus Temp monitoring and "B" high level trip circuit also in 9-82, the 120 VAC is converted to 24 VDC for use by the ATUs.	ILT/NLO/NLOR Obj. 6.b
(3) All 24 volt loads are provided a backup feed which is auctioneered. An Inverter loss will not affect the ATUs.	
(4) Div 1 ECCS ATU inverter backup power comes from the 250 RMOV Bd 2B.	
(5) Div 2 ECCS ATU inverter backup power comes from the 250 RMOV Bd 2A.	
c) Loss of power to 250 VDC RMOV Bds A/B will result in loss of the applicable ECCS ATU Inverter and all ATUs from that division. Other indications of an inverter loss are:	ILT Obj. 6.c LOR Obj. 7
(1) "ECCS analog trip unit trouble" alarm on panel 9-3	
(2) If the Division I inverter is lost, the "A" High water level trip circuit for the main turbine and reactor feed pump turbines will be de-energized. The RCIC power supply is lost making the RCIC INOP. Torus temp monitoring will be reduced.	Transfer of power supplies at pnl 25-32 will restore power to RCIC flow controllers and pnl 25-32 and 25-31.

QA Record. Non-RP - Retain in ECM (Lifetime Retention)
RP LPs - Retain in ECM (Life of Nuclear Insurance Policy, plus 10 years)

Lesson Plan Content

Outline of Instruction	**Instructor Notes and Methods (Optional)**
(3) If the Division 2 inverter is lost, the HPCI turbine controller will not function. This is also indicated by the alarm "HPCI 120 VAC Power Failure" on panel 9-3. Torus Temp monitoring will be reduced. "B" High water level trip circuit for main turbine and RFPT.	
7. Overview of 120VAC Distribution	IL-10
a) This shows an overview for the following 120VAC distribution. (1) Unit Preferred (2) Unit Non-Preferred (3) Plant Preferred (4) Plant Non-Preferred	
C. 120 Volt AC Procedure Coverage Guideline	
1. OI	
a) 0-OI-57C (120VAC OI) Discuss the following: (1) Precautions and Limitations - Section 3 (2) Unit Preferred MG set startup-section 5.5 (3) Maintenance bypass feed for I&C buses-section 8.10 review notes, caution, and table.	
2. AOI	
a) 0-AOI-57-3 (Loss of Plant Preferred) Discuss the following: (1) 1.0 Purpose, note 1 (2) Auto Actions, Notes, Section 3A thru 3E (3) Subsequent actions 4.2.2; 4.2.7; 4.2.9; 4.2.10; 4.2.12	
b) AOI-57-4 (Loss of Unit Preferred) Discuss the following: (1) Important symptoms, Section 2 (2) Automatic Actions, Section 3.0 (3) RPIS, Section 4.2.3 (4) Transfer cabinets, Section 4.2.5 & 4.2.6 (5) Caution about RPS on transformer	ILT Obj. 2.c, 2.e LOR Obj. 3.b, 3.c
c) AOI-57-5A (Loss of I&C Bus A) Discuss the following: (1) Section 1.0, Notes 1, 2 & 3 (2) Automatic Actions; Cautions & Notes	NLO/NLOR Obj. 1.e LOR Obj. 2.c

Examination Outline Cross-reference:

295025 (EPE 2) High Reactor Pressure / 3

EA1.10 (10CFR 55.41.7)

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

- Reactor water cleanup system

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295025EA1.10	
Importance Rating	2.8	------

Proposed Question: **# 10**

Unit 1 is operating at 100% RTP when the following conditions occur:

- Manual SCRAM inserted
- Lowest Reactor Water Level was (-) 38 inches
- Drywell Pressure is 3.2 psig
- Main Condenser Vacuum is currently 3 inches mercury (Hg)
- Current Reactor Water Level is (+) 10 inches

Given the conditions above, Reactor Pressure can be controlled using __________.

Note: 1-EOI Appendix-8B, Reopening MSIVs/Bypass Valve Operation
1-EOI Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode
1-EOI Appendix-11F, Alternate RPV Pressure Control Systems, RFPT on Minimum Flow
1-EOI Appendix-11E, Alternate RPV Pressure Control Systems, RWCU System Recirc/Blowdown Mode

A. 1-EOI Appendix-11C

B. 1-EOI Appendix-11F

C. 1-EOI Appendix-11E

D. 1-EOI Appendix-8B

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that EOI Appendix-11C can be used for Reactor Pressure Control if HPCI is not needed for Reactor Water Level control. However, given that Reactor Water Level was (-) 38 inches, may elect to use HPCI for injection. Additionally, when Drywell Pressure reached (+) 2.45 psig, HPCI automatically initiated. EOI Appendix-11C requires that any automatic initiation signals (Drywell Pressure or Reactor Water Level (-) 45 inches) are cleared to use HPCI in Reactor Pressure Control mode.

B INCORRECT: Incorrect but plausible in that EOI Appendix-11F can be used for Reactor Pressure Control if RFPTs are available and not needed for Reactor Water Level control. However, in accordance with 1-AOI-47-3, Loss of Condenser Vacuum, RFPTs trip when Condenser Vacuum reaches (-) 7 inches Hg. Additionally, EOI Appendix-11F, states to verify Hotwell Pressure is at or below (-) 7 inches Hg.

C **CORRECT:** *(See attached)* In accordance with 1-AOI-64-2a, Group 3 RWCU Isolation, a Low Reactor Water Level of (+) 2 inches results in a PCIS isolation for RWCU. Given Reactor Water Level is currently (+) 10 inches, RWCU is now available for use once it's PCIS isolation has been reset. In accordance with 1-EOI Appendix-11E, Alternate RPV Pressure Control Systems, RWCU System Recirc/Blowdown Mode, the first step to place RWCU in Recirc mode is to reset PCIS.

D INCORRECT: Incorrect but plausible in that EOI Appendix-8B is the preferred method used for Reactor Pressure Control if the Main Turbine Bypass Valves are open. However, in accordance with 1-AOI-47-3, Loss of Condenser Vacuum, Main Turbine Bypass Valves closure occurs when Condenser Vacuum reaches (-) 7 inches Hg. Additionally, EOI Appendix-8B, states to check Condenser Vacuum is greater than 7 inches Hg.

RO Level Justification: Tests the candidate's ability to monitor the Reactor Water Cleanup System as it relates Reactor Pressure Control. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): (Attach if not previously provided)
1-EOI Appendix-8B, Rev. 5
1-EOI Appendix-11C, Rev. 3
1-EOI Appendix-11E, Rev. 2
1-EOI Appendix-11F, Rev. 0
1-AOI-64-2a, Rev. 3

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.013 Obj. 9b (As available)

Question Source:

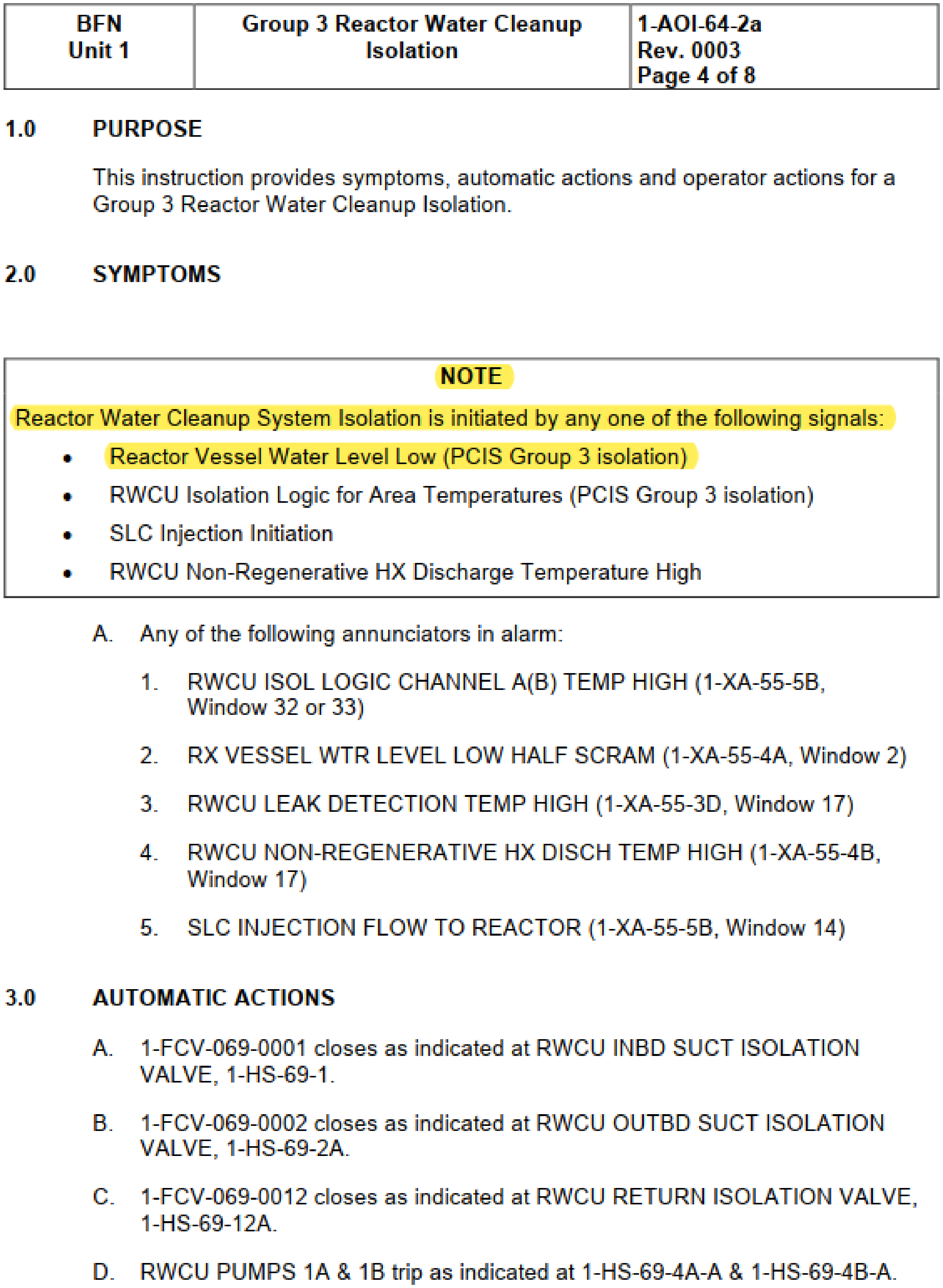
Bank #		
Modified Bank #		(Note changes or attach parent)
New	**X**	

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-AOI-64-2a:



BFN Unit 1	**Group 3 Reactor Water Cleanup Isolation**	**1-AOI-64-2a Rev. 0003 Page 4 of 8**

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

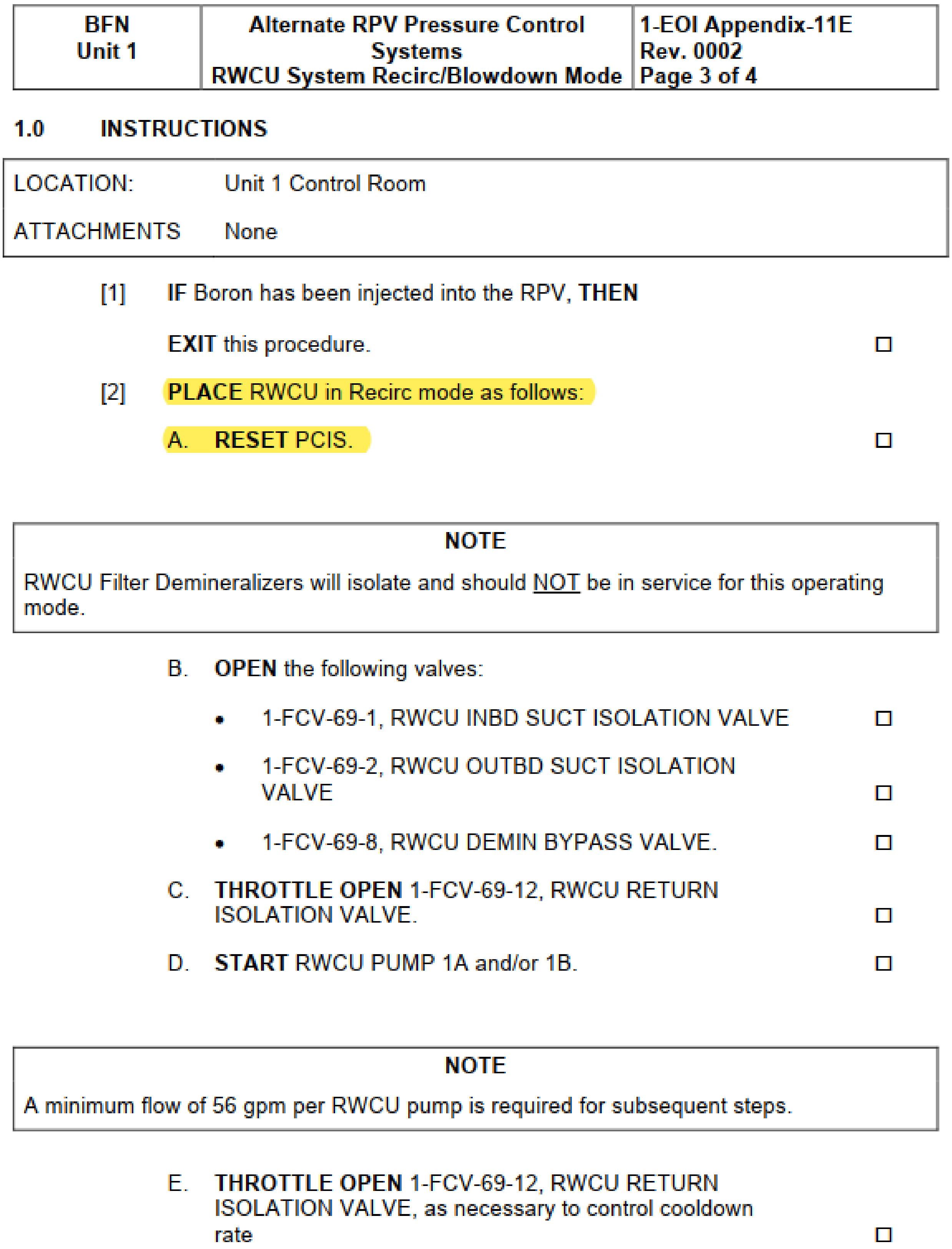
A. 1-FCV-069-0001 closes as indicated at RWCU INBD SUCT ISOLATION VALVE, 1-HS-69-1.

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-EOI Appendix-11E:



BFN **Unit 1**	**Alternate RPV Pressure Control Systems** **RWCU System Recirc/Blowdown Mode**	**1-EOI Appendix-11E** **Rev. 0002** **Page 3 of 4**

1.0 INSTRUCTIONS

LOCATION:	Unit 1 Control Room
ATTACHMENTS	None

[1] **IF** Boron has been injected into the RPV, **THEN**

EXIT this procedure. ☐

[2] **PLACE** RWCU in Recirc mode as follows:

A. **RESET** PCIS. ☐

NOTE

RWCU Filter Demineralizers will isolate and should NOT be in service for this operating mode.

B. **OPEN** the following valves:

- 1-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE ☐
- 1-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE ☐
- 1-FCV-69-8, RWCU DEMIN BYPASS VALVE. ☐

C. **THROTTLE OPEN** 1-FCV-69-12, RWCU RETURN ISOLATION VALVE. ☐

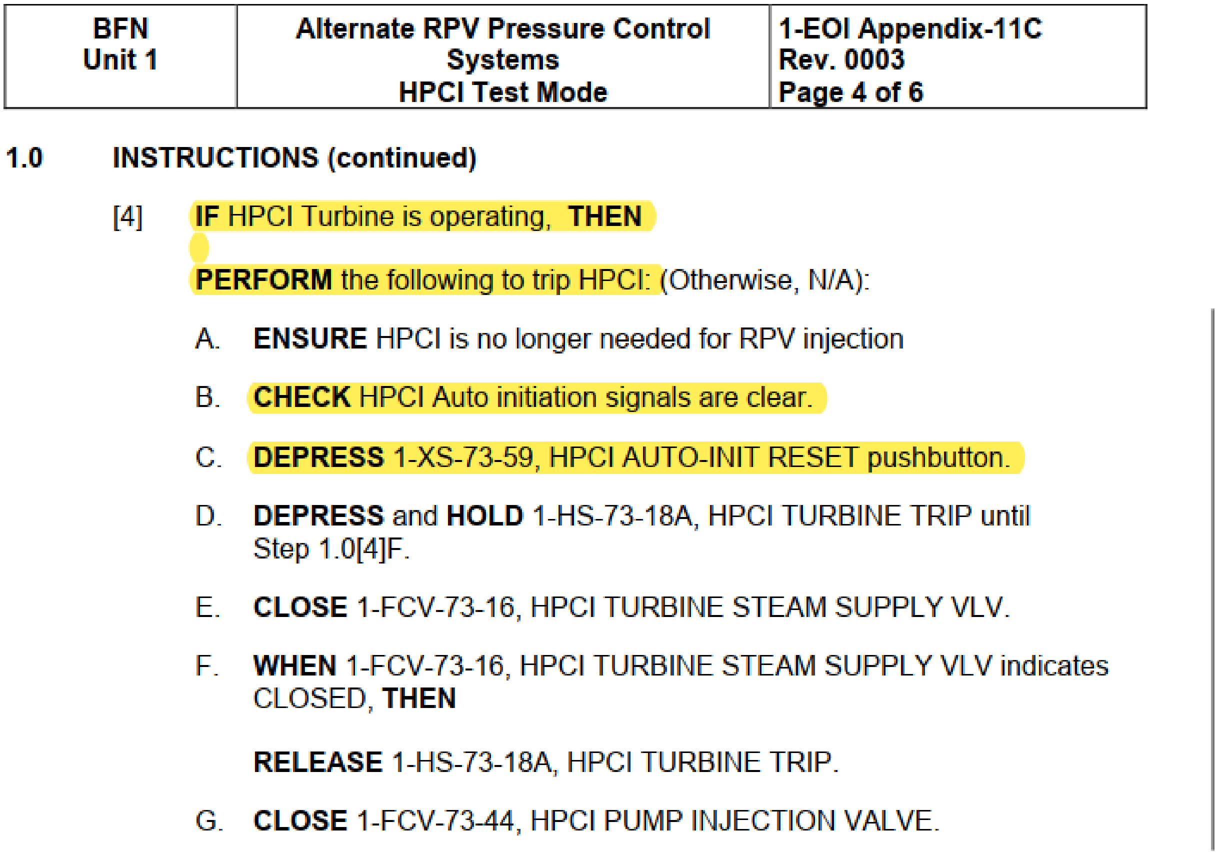
D. **START** RWCU PUMP 1A and/or 1B. ☐

NOTE

A minimum flow of 56 gpm per RWCU pump is required for subsequent steps.

E. **THROTTLE OPEN** 1-FCV-69-12, RWCU RETURN ISOLATION VALVE, as necessary to control cooldown rate ☐

Excerpt from 1-EOI Appendix-11C: Supports Distractor ‘A’



BFN Unit 1	**Alternate RPV Pressure Control Systems HPCI Test Mode**	**1-EOI Appendix-11C Rev. 0003 Page 4 of 6**

1.0 INSTRUCTIONS (continued)

[4] **IF** HPCI Turbine is operating, **THEN**

PERFORM the following to trip HPCI: (Otherwise, N/A):

A. **ENSURE** HPCI is no longer needed for RPV injection

B. **CHECK** HPCI Auto initiation signals are clear.

C. **DEPRESS** 1-XS-73-59, HPCI AUTO-INIT RESET pushbutton.

D. **DEPRESS** and **HOLD** 1-HS-73-18A, HPCI TURBINE TRIP until Step 1.0[4]F.

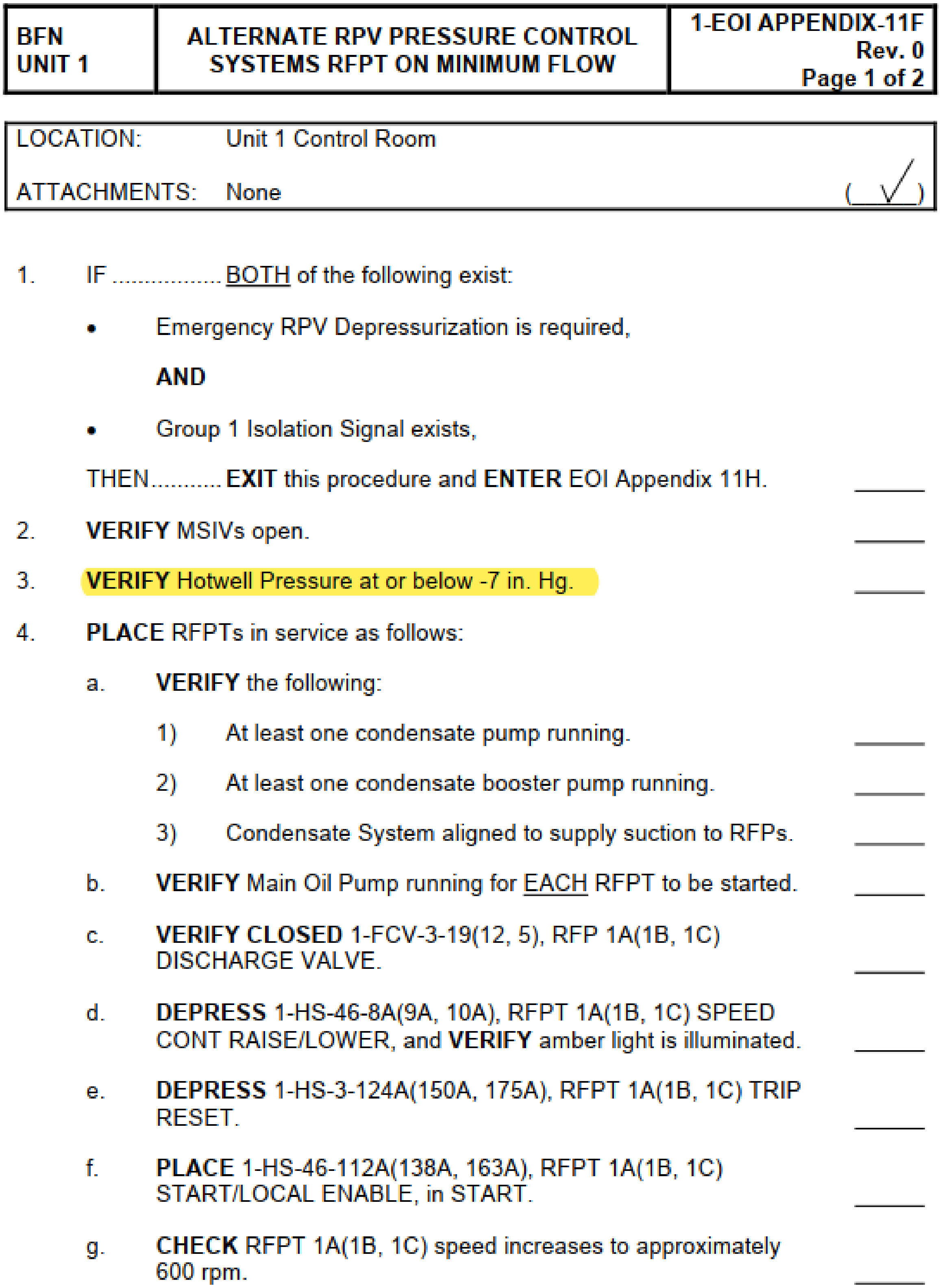
E. **CLOSE** 1-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV.

F. **WHEN** 1-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV indicates CLOSED, **THEN**

RELEASE 1-HS-73-18A, HPCI TURBINE TRIP.

G. **CLOSE** 1-FCV-73-44, HPCI PUMP INJECTION VALVE.

Excerpt from 1-EOI Appendix-11F: Supports Distractor ‘B’



BFN **UNIT 1**	**ALTERNATE RPV PRESSURE CONTROL SYSTEMS RFPT ON MINIMUM FLOW**	**1-EOI APPENDIX-11F** **Rev. 0** **Page 1 of 2**

LOCATION: Unit 1 Control Room

ATTACHMENTS: None (__√__)

1. IF .... BOTH of the following exist:

   - Emergency RPV Depressurization is required,

     AND

   - Group 1 Isolation Signal exists,

   THEN.......... **EXIT** this procedure and **ENTER** EOI Appendix 11H.

2. **VERIFY** MSIVs open.

3. **VERIFY** Hotwell Pressure at or below -7 in. Hg.

4. **PLACE** RFPTs in service as follows:

   a. **VERIFY** the following:

      1) At least one condensate pump running.

      2) At least one condensate booster pump running.

      3) Condensate System aligned to supply suction to RFPs.

   b. **VERIFY** Main Oil Pump running for EACH RFPT to be started.

   c. **VERIFY CLOSED** 1-FCV-3-19(12, 5), RFP 1A(1B, 1C) DISCHARGE VALVE.

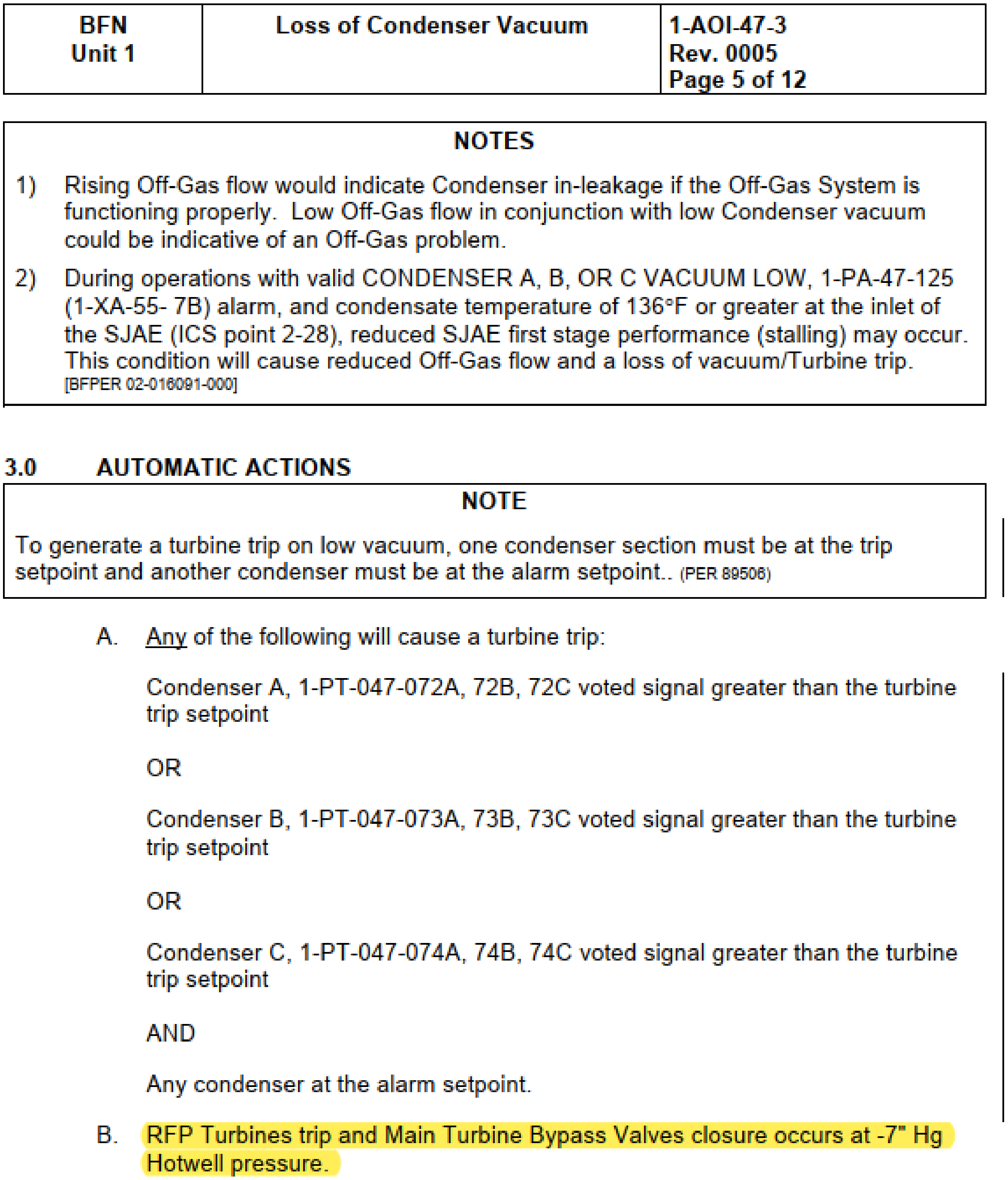
   d. **DEPRESS** 1-HS-46-8A(9A, 10A), RFPT 1A(1B, 1C) SPEED CONT RAISE/LOWER, and **VERIFY** amber light is illuminated.

   e. **DEPRESS** 1-HS-3-124A(150A, 175A), RFPT 1A(1B, 1C) TRIP RESET.

   f. **PLACE** 1-HS-46-112A(138A, 163A), RFPT 1A(1B, 1C) START/LOCAL ENABLE, in START.

   g. **CHECK** RFPT 1A(1B, 1C) speed increases to approximately 600 rpm.

Excerpt from 1-AOI-47-3: Supports Distractors ‘B’ and ‘D’



BFN Unit 1	**Loss of Condenser Vacuum**	**1-AOI-47-3 Rev. 0005 Page 5 of 12**

NOTES

1) Rising Off-Gas flow would indicate Condenser in-leakage 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, 1-PA-47-125 (1-XA-55- 7B) 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]

3.0 AUTOMATIC ACTIONS

NOTE

To generate a turbine trip on low vacuum, one condenser section must be at the trip setpoint and another condenser must be at the alarm setpoint.. (PER 89506)

A. Any of the following will cause a turbine trip:

Condenser A, 1-PT-047-072A, 72B, 72C voted signal greater than the turbine trip setpoint

OR

Condenser B, 1-PT-047-073A, 73B, 73C voted signal greater than the turbine trip setpoint

OR

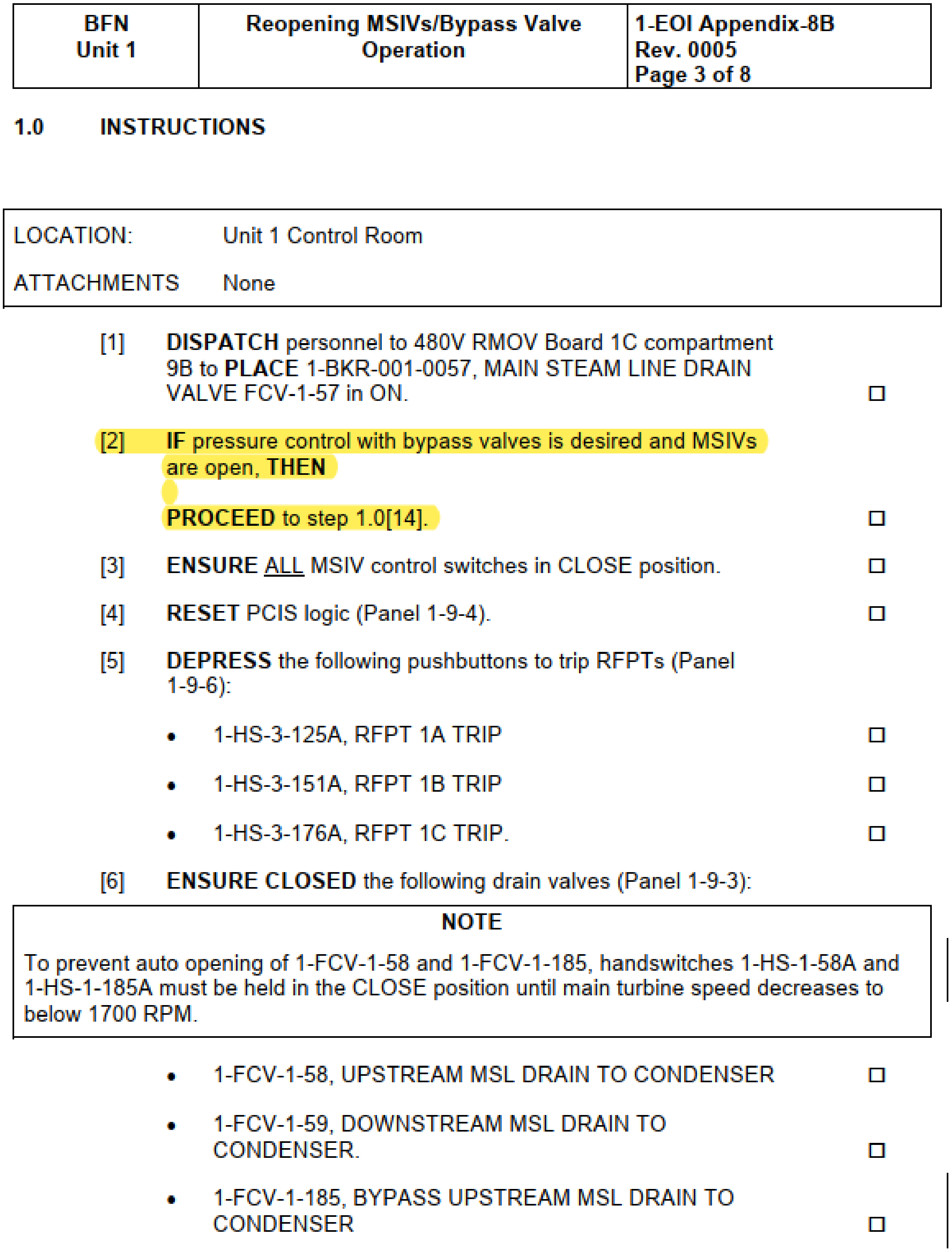
Condenser C, 1-PT-047-074A, 74B, 74C voted signal greater than the turbine trip setpoint

AND

Any condenser at the alarm setpoint.

B. RFP Turbines trip and Main Turbine Bypass Valves closure occurs at -7" Hg Hotwell pressure.

Excerpts from 1-EOI Appendix-8B: Supports Distractor ‘D’



BFN Unit 1	**Reopening MSIVs/Bypass Valve Operation**	**1-EOI Appendix-8B Rev. 0005 Page 3 of 8**

1.0 INSTRUCTIONS

LOCATION:	Unit 1 Control Room
ATTACHMENTS	None

[1] **DISPATCH** personnel to 480V RMOV Board 1C compartment 9B to **PLACE** 1-BKR-001-0057, MAIN STEAM LINE DRAIN VALVE FCV-1-57 in ON. ☐

[2] **IF** pressure control with bypass valves is desired and MSIVs are open, **THEN**

PROCEED to step 1.0[14]. ☐

[3] **ENSURE** ALL MSIV control switches in CLOSE position. ☐

[4] **RESET** PCIS logic (Panel 1-9-4). ☐

[5] **DEPRESS** the following pushbuttons to trip RFPTs (Panel 1-9-6):

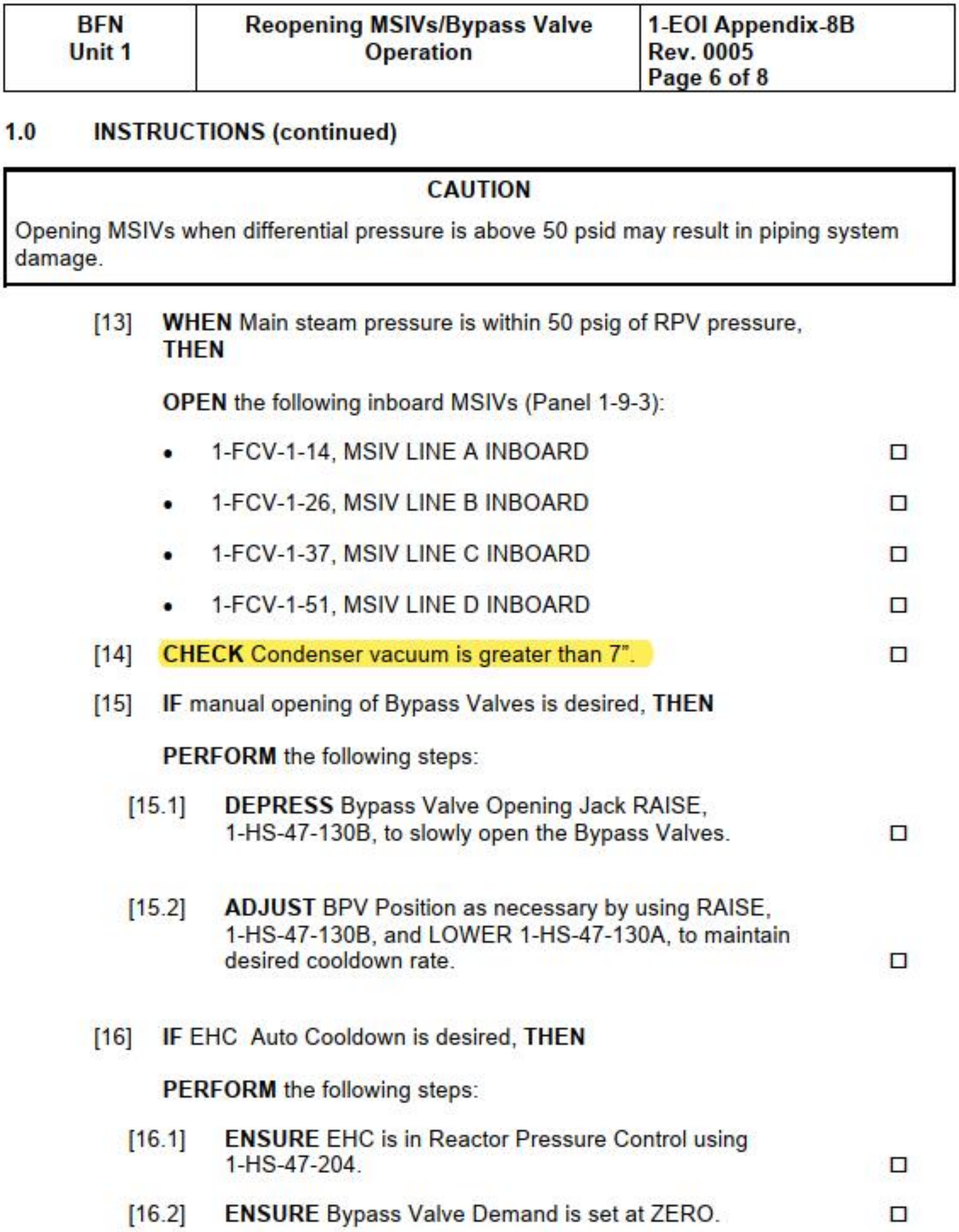
- 1-HS-3-125A, RFPT 1A TRIP ☐
- 1-HS-3-151A, RFPT 1B TRIP ☐
- 1-HS-3-176A, RFPT 1C TRIP. ☐

[6] **ENSURE CLOSED** the following drain valves (Panel 1-9-3):

NOTE

To prevent auto opening of 1-FCV-1-58 and 1-FCV-1-185, handswitches 1-HS-1-58A and 1-HS-1-185A must be held in the CLOSE position until main turbine speed decreases to below 1700 RPM.

- 1-FCV-1-58, UPSTREAM MSL DRAIN TO CONDENSER ☐
- 1-FCV-1-59, DOWNSTREAM MSL DRAIN TO CONDENSER. ☐
- 1-FCV-1-185, BYPASS UPSTREAM MSL DRAIN TO CONDENSER ☐



BFN **Unit 1**	**Reopening MSIVs/Bypass Valve Operation**	**1-EOI Appendix-8B** **Rev. 0005** **Page 6 of 8**

1.0 INSTRUCTIONS (continued)

CAUTION

Opening MSIVs when differential pressure is above 50 psid may result in piping system damage.

[13] **WHEN** Main steam pressure is within 50 psig of RPV pressure, **THEN**

OPEN the following inboard MSIVs (Panel 1-9-3):

- 1-FCV-1-14, MSIV LINE A INBOARD ☐
- 1-FCV-1-26, MSIV LINE B INBOARD ☐
- 1-FCV-1-37, MSIV LINE C INBOARD ☐
- 1-FCV-1-51, MSIV LINE D INBOARD ☐

[14] **CHECK** Condenser vacuum is greater than 7". ☐

[15] **IF** manual opening of Bypass Valves is desired, **THEN**

PERFORM the following steps:

[15.1] **DEPRESS** Bypass Valve Opening Jack RAISE, 1-HS-47-130B, to slowly open the Bypass Valves. ☐

[15.2] **ADJUST** BPV Position as necessary by using RAISE, 1-HS-47-130B, and LOWER 1-HS-47-130A, to maintain desired cooldown rate. ☐

[16] **IF** EHC Auto Cooldown is desired, **THEN**

PERFORM the following steps:

[16.1] **ENSURE** EHC is in Reactor Pressure Control using 1-HS-47-204. ☐

[16.2] **ENSURE** Bypass Valve Demand is set at ZERO. ☐

Examination Outline Cross-reference:	Level	RO	SRO
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling	Tier #	2	------
A4.12 (10CFR 55.41.7)	Group #	1	------
Ability to manually operate and/or monitor in the control room: • Turbine speed control	K/A #	217000A4.12	
	Importance Rating	3.9	------

Proposed Question: **# 11**

In accordance with 1-OI-71, Reactor Core Isolation Cooling System, the RCIC **MINIMUM** speed is **(1)** and maintaining turbine speed above this limit is accomplished **(2)** .

A. **(1)** 2100 rpm
(2) MANUALLY by the Operator

B. **(1)** 2100 rpm
(2) AUTOMATICALLY by an electrical speed clamp

C. **(1)** 2400 rpm
(2) MANUALLY by the Operator

D. **(1)** 2400 rpm
(2) AUTOMATICALLY by an electrical speed clamp

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 1-OI-71, Reactor Core Isolation Cooling System, Attachment 4 – RCIC Injection System Lineup Hard Card, the Operator is required to check for proper operation by ensuring that RCIC Turbine speed accelerates above 2100 rpm. For second part, the Operator is required to manually maintain RCIC Turbine speed between 2100 and 4750 rpm on 1-SI-71-42A, RCIC TURBINE SPEED during normal operation.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that the Reactor Feed Pumps have a low speed clamp to prevent it backing down to 600 rpm and upper speed clamp of 4100 rpm.

C INCORRECT: First part is incorrect in that the HPCI Injection System Lineup Hard Card states that the Operator is required to maintain HPCI Turbine speed accelerates above 2400 rpm. 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 and monitor RCIC Turbine speed control in the Main Control Room. This question is rated as Memory due to the requirement to strictly recall facts related to operation of the RCIC System.

Technical Reference(s): 1-OI-71, Rev. 28 (Attach if not previously provided)

1-OI-73, Rev. 32

1-OI-3, Rev. 55

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.040 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	**X**
	Comprehension or Analysis	

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

Excerpts from 1-OI-71:

BFN **Unit 1**	**Reactor Core Isolation Cooling System**	**1-OI-71** **Rev. 0028** **Page 10 of 68**

3.1 General Precautions (continued)

E. RCIC PMP MIN FLOW VLV, 1-FCV-071-0034, 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 TURB TRIP/THROT VLV, 1-FCV-071-0009, is required to be manually reset at the turbine. [INPO SOER 82-008] If there is NO positive indication in Control Room that 1-FCV-071-0009 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 SUPP CHBR TURB EXH VAC RELIEF VLV, 1-FCV-071-0059, 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 HPCI and RCIC are shut down 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 SRO.

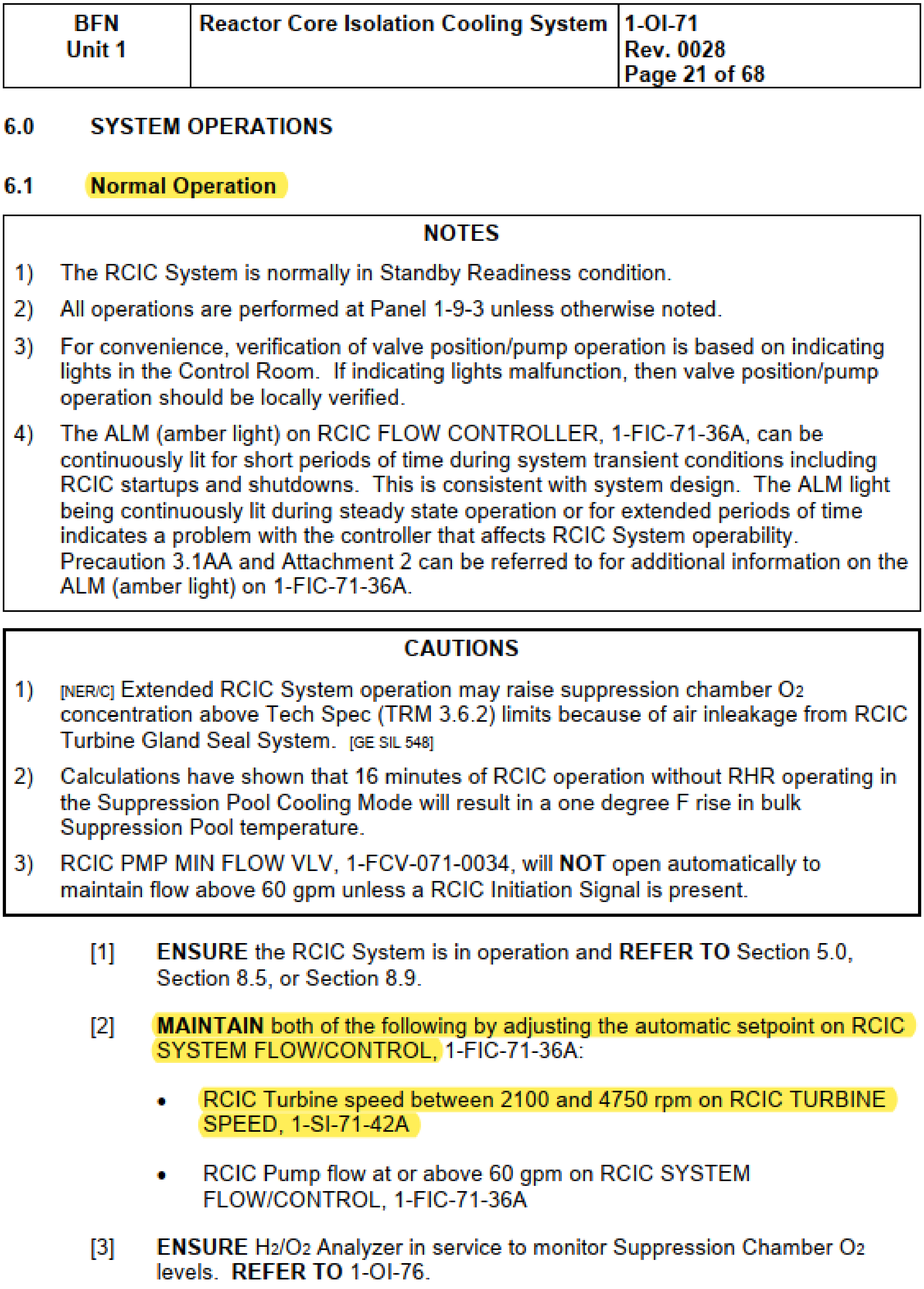
K. Injection of Suppression Pool water into the RPV should be avoided whenever possible to prevent degradation of primary system water quality.

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, 1-HS-71-9A. Section 8.4 is used to restore the turbine to operation if required.

M. When operable, RCIC SYSTEM FLOW/CONTROL, 1-FIC-71-36A, should be in AUTO in order to provide more stable system operation.

N. When RCIC SYSTEM FLOW/CONTROL, 1-FIC-71-36A, is operated in MANUAL, turbine speed should be raised as rapidly as possible to prevent turbine exhaust check valve chatter.

O. When RCIC STEAM LINE INBD ISOLATION VLV, 1-FCV-071-0002 or RCIC TURB STM SUP OUTBD ISOL VLV, 1-FCV-071-0003, are closed, MN STM LINE DRAIN INBD ISOL VLV, 1-FCV-001-0055 and MAIN STEAM LINE OUTBD ISOL VLV, 1-FCV-001-0056, should be open to drain RCIC Stm Line.



BFN Unit 1	Reactor Core Isolation Cooling System	1-OI-71 Rev. 0028 Page 21 of 68

6.0 SYSTEM OPERATIONS

6.1 Normal Operation

NOTES

1) The RCIC System is normally in Standby Readiness condition.
2) All operations are performed at Panel 1-9-3 unless otherwise noted.
3) For convenience, verification of valve position/pump operation is based on indicating lights in the Control Room. If indicating lights malfunction, then valve position/pump operation should be locally verified.
4) The ALM (amber light) on RCIC FLOW CONTROLLER, 1-FIC-71-36A, can be continuously lit for short periods of time during system transient conditions including RCIC startups and shutdowns. This is consistent with system design. The ALM light being continuously lit during steady state operation or for extended periods of time indicates a problem with the controller that affects RCIC System operability. Precaution 3.1AA and Attachment 2 can be referred to for additional information on the ALM (amber light) on 1-FIC-71-36A.

CAUTIONS

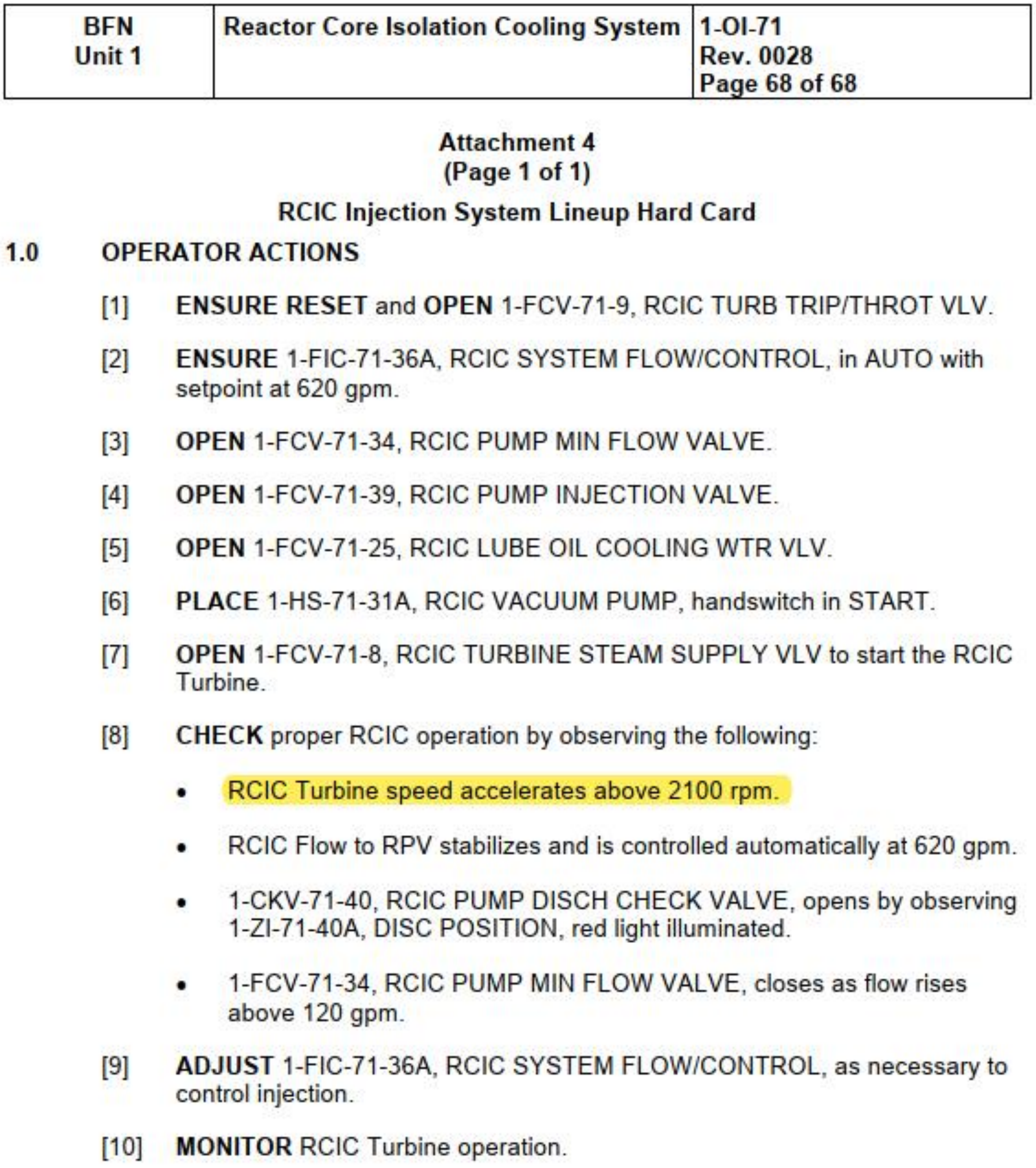
1) [NER/C] Extended RCIC System operation may raise suppression chamber O_2 concentration above Tech Spec (TRM 3.6.2) limits because of air inleakage from RCIC Turbine Gland Seal System. [GE SIL 548]
2) Calculations have shown that 16 minutes of RCIC operation without RHR operating in the Suppression Pool Cooling Mode will result in a one degree F rise in bulk Suppression Pool temperature.
3) RCIC PMP MIN FLOW VLV, 1-FCV-071-0034, will **NOT** open automatically to maintain flow above 60 gpm unless a RCIC Initiation Signal is present.

[1] **ENSURE** the RCIC System is in operation and **REFER TO** Section 5.0, Section 8.5, or Section 8.9.

[2] **MAINTAIN** both of the following by adjusting the automatic setpoint on RCIC SYSTEM FLOW/CONTROL, 1-FIC-71-36A:

- RCIC Turbine speed between 2100 and 4750 rpm on RCIC TURBINE SPEED, 1-SI-71-42A
- RCIC Pump flow at or above 60 gpm on RCIC SYSTEM FLOW/CONTROL, 1-FIC-71-36A

[3] **ENSURE** H_2/O_2 Analyzer in service to monitor Suppression Chamber O_2 levels. **REFER TO** 1-OI-76.



BFN **Unit 1**	**Reactor Core Isolation Cooling System**	**1-OI-71** **Rev. 0028** **Page 68 of 68**

Attachment 4
(Page 1 of 1)

RCIC Injection System Lineup Hard Card

1.0 OPERATOR ACTIONS

[1] **ENSURE RESET** and **OPEN** 1-FCV-71-9, RCIC TURB TRIP/THROT VLV.

[2] **ENSURE** 1-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with setpoint at 620 gpm.

[3] **OPEN** 1-FCV-71-34, RCIC PUMP MIN FLOW VALVE.

[4] **OPEN** 1-FCV-71-39, RCIC PUMP INJECTION VALVE.

[5] **OPEN** 1-FCV-71-25, RCIC LUBE OIL COOLING WTR VLV.

[6] **PLACE** 1-HS-71-31A, RCIC VACUUM PUMP, handswitch in START.

[7] **OPEN** 1-FCV-71-8, RCIC TURBINE STEAM SUPPLY VLV to start the RCIC Turbine.

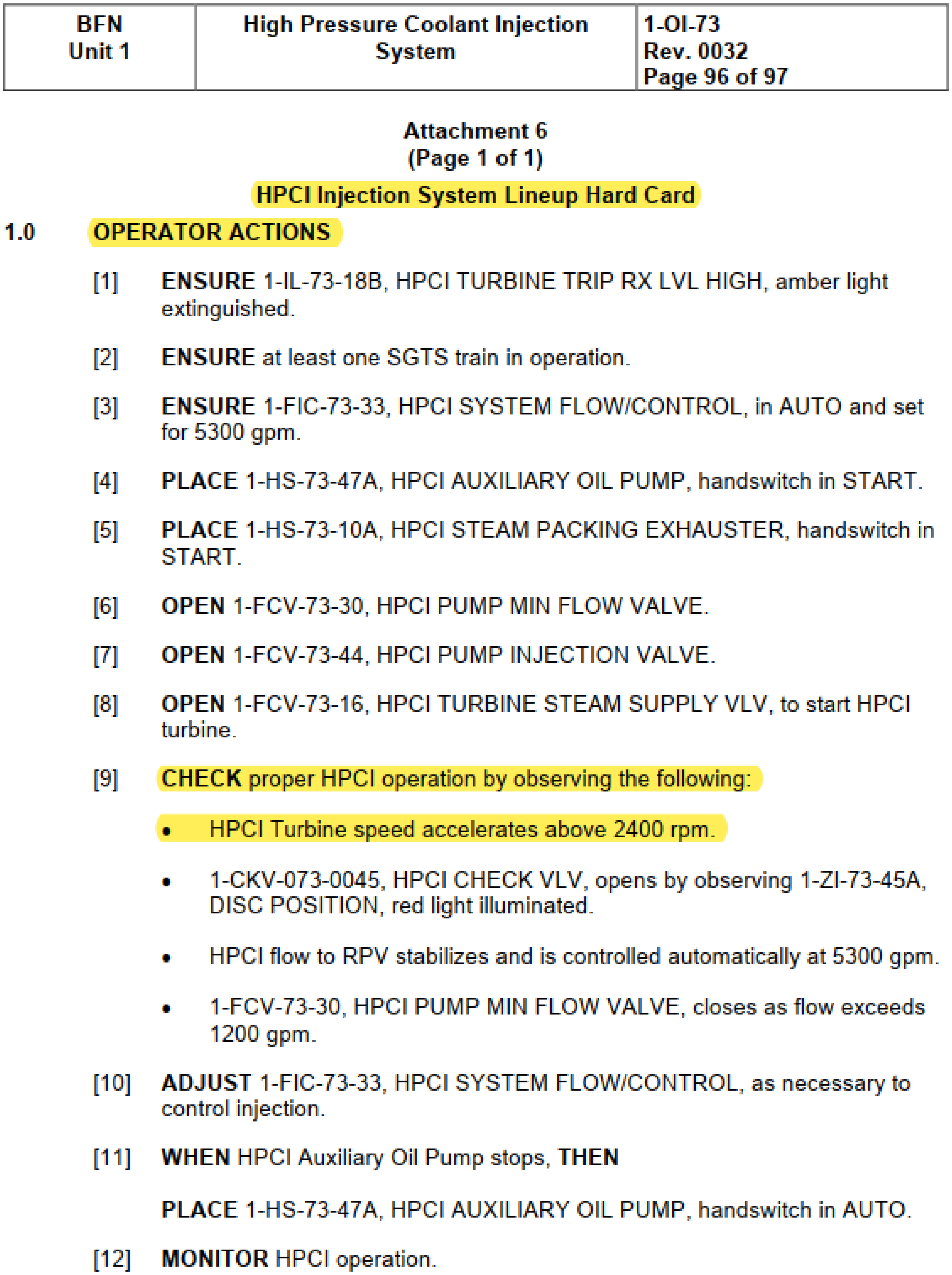
[8] **CHECK** proper RCIC operation by observing the following:

- RCIC Turbine speed accelerates above 2100 rpm.
- RCIC Flow to RPV stabilizes and is controlled automatically at 620 gpm.
- 1-CKV-71-40, RCIC PUMP DISCH CHECK VALVE, opens by observing 1-ZI-71-40A, DISC POSITION, red light illuminated.
- 1-FCV-71-34, RCIC PUMP MIN FLOW VALVE, closes as flow rises above 120 gpm.

[9] **ADJUST** 1-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection.

[10] **MONITOR** RCIC Turbine operation.

Excerpt from 1-OI-73: Supports Distractors C(1), D(1)



BFN Unit 1	High Pressure Coolant Injection System	1-OI-73 Rev. 0032 Page 96 of 97

Attachment 6
(Page 1 of 1)

HPCI Injection System Lineup Hard Card

1.0 OPERATOR ACTIONS

[1] **ENSURE** 1-IL-73-18B, HPCI TURBINE TRIP RX LVL HIGH, amber light extinguished.

[2] **ENSURE** at least one SGTS train in operation.

[3] **ENSURE** 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, in AUTO and set for 5300 gpm.

[4] **PLACE** 1-HS-73-47A, HPCI AUXILIARY OIL PUMP, handswitch in START.

[5] **PLACE** 1-HS-73-10A, HPCI STEAM PACKING EXHAUSTER, handswitch in START.

[6] **OPEN** 1-FCV-73-30, HPCI PUMP MIN FLOW VALVE.

[7] **OPEN** 1-FCV-73-44, HPCI PUMP INJECTION VALVE.

[8] **OPEN** 1-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, to start HPCI turbine.

[9] **CHECK** proper HPCI operation by observing the following:

- HPCI Turbine speed accelerates above 2400 rpm.
- 1-CKV-073-0045, HPCI CHECK VLV, opens by observing 1-ZI-73-45A, DISC POSITION, red light illuminated.
- HPCI flow to RPV stabilizes and is controlled automatically at 5300 gpm.
- 1-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds 1200 gpm.

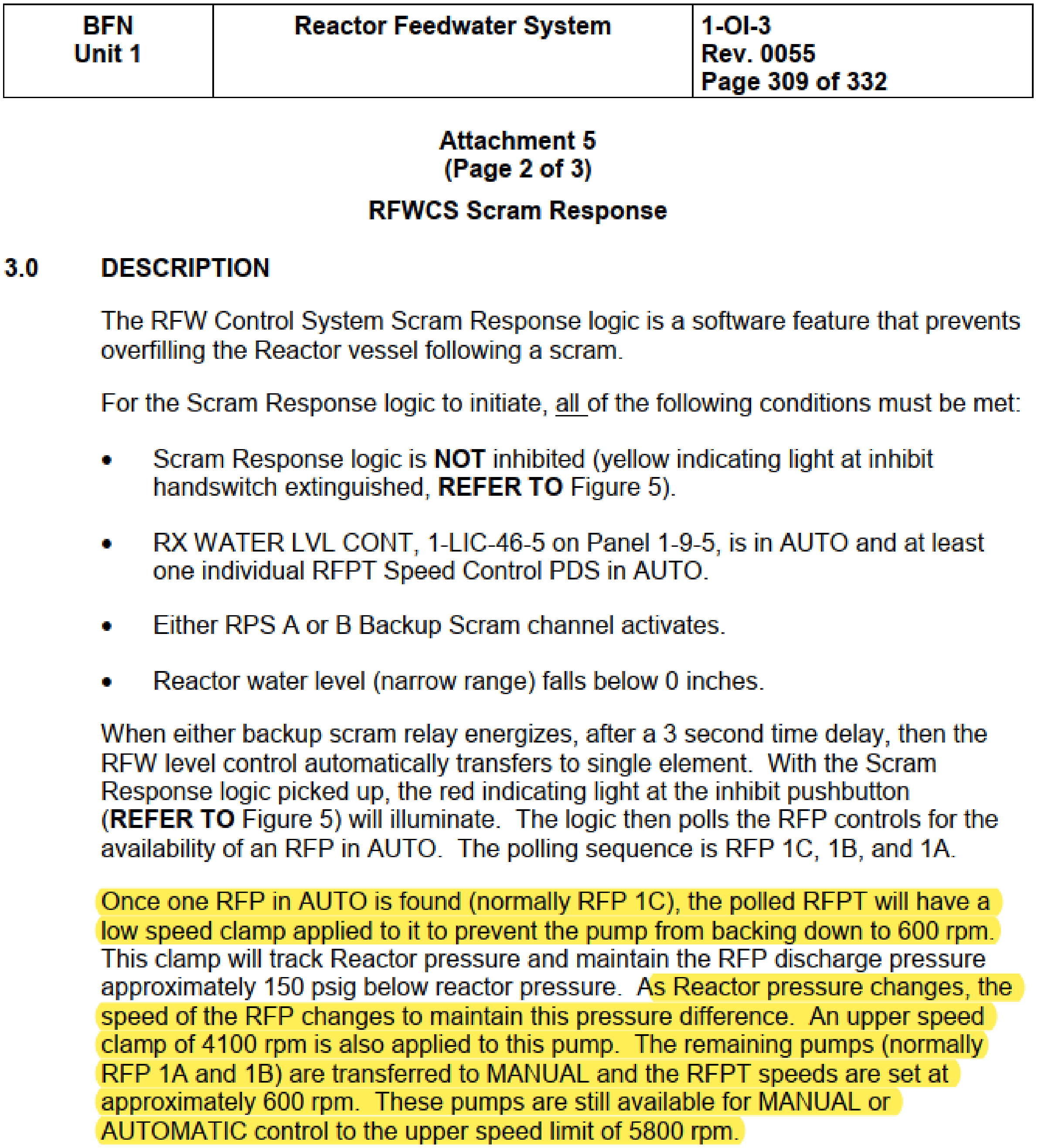
[10] **ADJUST** 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, as necessary to control injection.

[11] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN**

PLACE 1-HS-73-47A, HPCI AUXILIARY OIL PUMP, handswitch in AUTO.

[12] **MONITOR** HPCI operation.

Excerpt from 1-OI-3: Supports Distractors B(2), D(2)



BFN Unit 1	**Reactor Feedwater System**	**1-OI-3 Rev. 0055 Page 309 of 332**

Attachment 5
(Page 2 of 3)

RFWCS Scram Response

3.0 DESCRIPTION

The RFW Control System Scram Response logic is a software feature that prevents overfilling the Reactor vessel following a scram.

For the Scram Response logic to initiate, all of the following conditions must be met:

- Scram Response logic is **NOT** inhibited (yellow indicating light at inhibit handswitch extinguished, **REFER TO** Figure 5).
- RX WATER LVL CONT, 1-LIC-46-5 on Panel 1-9-5, is in AUTO and at least one individual RFPT Speed Control PDS in AUTO.
- Either RPS A or B Backup Scram channel activates.
- Reactor water level (narrow range) falls below 0 inches.

When either backup scram relay energizes, after a 3 second time delay, then the RFW level control automatically transfers to single element. With the Scram Response logic picked up, the red indicating light at the inhibit pushbutton (**REFER TO** Figure 5) will illuminate. The logic then polls the RFP controls for the availability of an RFP in AUTO. The polling sequence is RFP 1C, 1B, and 1A.

Once one RFP in AUTO is found (normally RFP 1C), the polled RFPT will have a low speed clamp applied to it to prevent the pump from backing down to 600 rpm. This clamp will track Reactor pressure and maintain the RFP discharge pressure approximately 150 psig below reactor pressure. As Reactor pressure changes, the speed of the RFP changes to maintain this pressure difference. An upper speed clamp of 4100 rpm is also applied to this pump. The remaining pumps (normally RFP 1A and 1B) are transferred to MANUAL and the RFPT speeds are set at approximately 600 rpm. These pumps are still available for MANUAL or AUTOMATIC control to the upper speed limit of 5800 rpm.

Examination Outline Cross-reference:	Level	RO	SRO
215003 (SF7 IRM) Intermediate-Range Monitor	Tier #	2	------
A1.08 (10CFR 55.41.5)	Group #	1	------
Ability to predict and/or monitor changes in parameters associated with operation of the Intermediate Range Monitor system, including: • IRM back panel switches	K/A #	215003A1.08	
	Importance Rating	3.1	------

Proposed Question: **# 12**

Unit 2 is in MODE 2 with the following conditions:

- A Reactor startup is in progress
- An Instrument Mechanic (IM) places 2-HS-92-7-41A, IRM ‘A’ Channel A INOP/INHIBIT switch in the STANDBY position

Given the conditions above, the IM’s actions will result in a **(1)** SCRAM.

In accordance with 2-OI-92A, Intermediate Range Monitors, IRM ‘A’ bypass operations are performed on Panel **(2)** .

A. (1) full
(2) 9-5 **AND** 9-12

B. (1) full
(2) 9-5

C. (1) half
(2) 9-5 **AND** 9-12

D. (1) half
(2) 9-5

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the candidate could confuse the IRM channel and trip logic associated with a full Reactor Protection System SCRAM. Second part is incorrect but plausible in that the IRM drawers located on Panel 2-9-12 can bypass the IRM switch position out-of-operate trip only during testing of the IRM channels.

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-9-5A, WINDOW 33 and 2-OI-92A, Intermediate Range Monitors, if one sensor actuates as is the case based on the given IM's actions, a half SCRAM will occur. For second part, all IRM bypass operations are performed on Panel 2-9-5 unless specifically stated otherwise.

RO Level Justification: Tests the candidate's ability to predict and monitor changes associated with Intermediate Range Monitor System panel 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 specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s):	2-OI-92A, Rev. 29	(Attach if not previously provided)
	2-ARP-9-5A, Rev. 61	
	OPL171.020, Rev. 13	

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.020 Obj. 6 (As available)

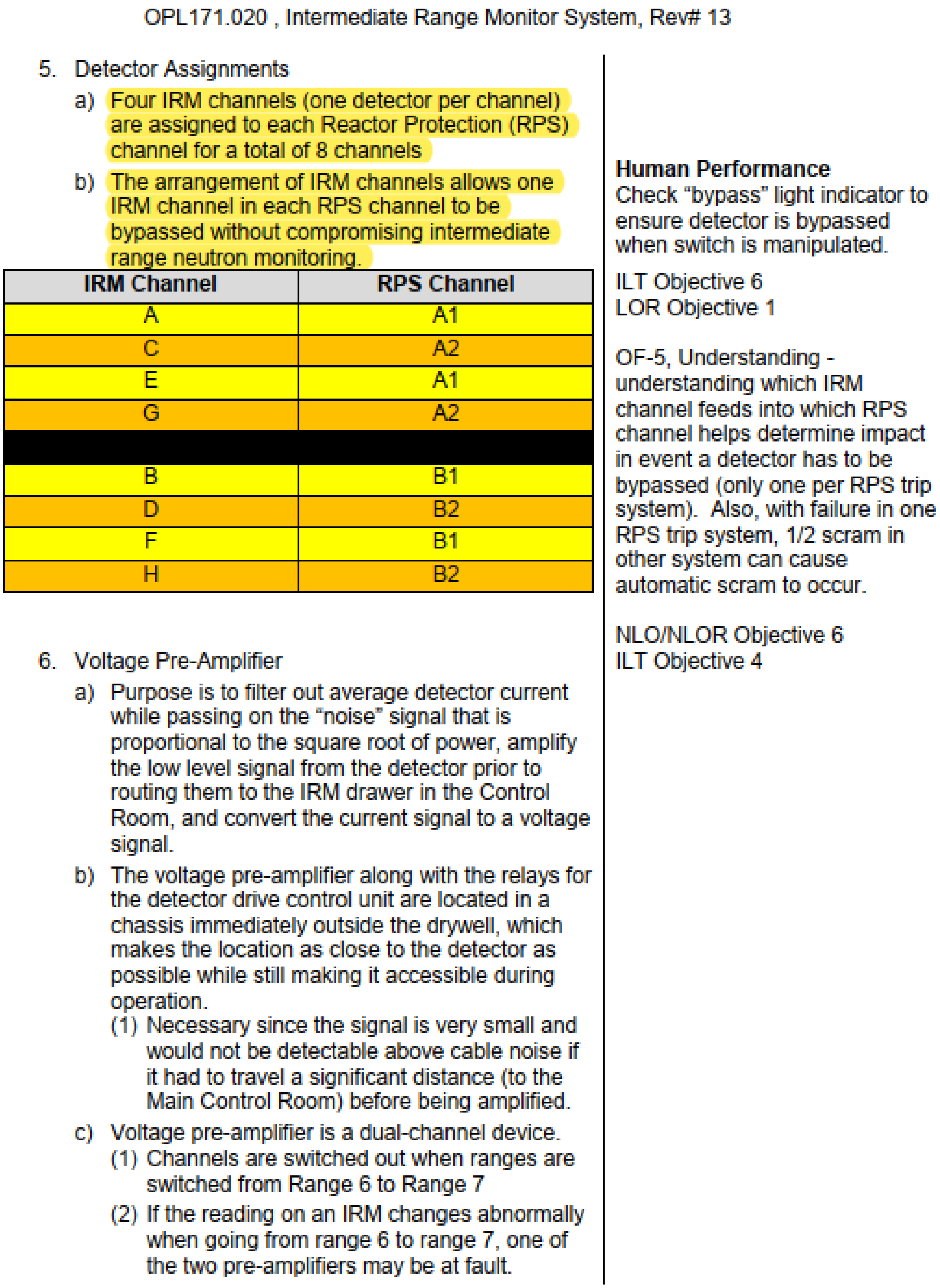
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

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

Comments:

Excerpt from OPL171.020 Lesson Plan:



OPL171.020 , Intermediate Range Monitor System, Rev# 13

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

Human Performance
Check "bypass" light indicator to ensure detector is bypassed when switch is manipulated.

ILT Objective 6
LOR Objective 1

OF-5, Understanding - understanding which IRM channel feeds into which RPS channel helps determine impact in event a detector has to be bypassed (only one per RPS trip system). Also, with failure in one RPS trip system, 1/2 scram in other system can cause automatic scram to occur.

6. Voltage Pre-Amplifier

NLO/NLOR Objective 6
ILT Objective 4

   a) Purpose is to filter out average detector current while passing on the "noise" signal that is proportional to the square root of power, amplify the low level signal from the detector prior to routing them to the IRM drawer in the Control Room, and convert the current signal to a voltage signal.
   b) The voltage pre-amplifier along with the relays for the detector drive control unit are located in a chassis immediately outside the drywell, which makes the location as close to the detector as possible while still making it accessible during operation.
      (1) Necessary since the signal is very small and would not be detectable above cable noise if it had to travel a significant distance (to the Main Control Room) before being amplified.
   c) Voltage pre-amplifier is a dual-channel device.
      (1) Channels are switched out when ranges are switched from Range 6 to Range 7
      (2) If the reading on an IRM changes abnormally when going from range 6 to range 7, one of the two pre-amplifiers may be at fault.

Excerpt from 2-ARP-9-5A:

BFN **Unit 2**	**Panel 9-5** **2-XA-55-5A**	**2-ARP-9-5A** **Rev. 0061** **Page 44 of 48**

IRM
CH A, C, E, G
HI-HI/INOP

33

(Page 1 of 2)

Sensor/Trip Point:

Relay K-16

A. HI-HI ≥ 116.4 on 125 scale
B. INOP.
  1. Hi voltage low.
  2. Module unplugged.
  3. Function switch NOT in OPERATE.
  4. Loss of ± 24 VDC to monitor.

Sensor Location: Control Room Panel 2-9-12.

Probable Cause:

A. Flux level at or above setpoint.
B. One or more inoperable conditions exist.
C. Testing in progress.
D. Malfunction of sensor.
E. Control rod drop accident.

Automatic Action:

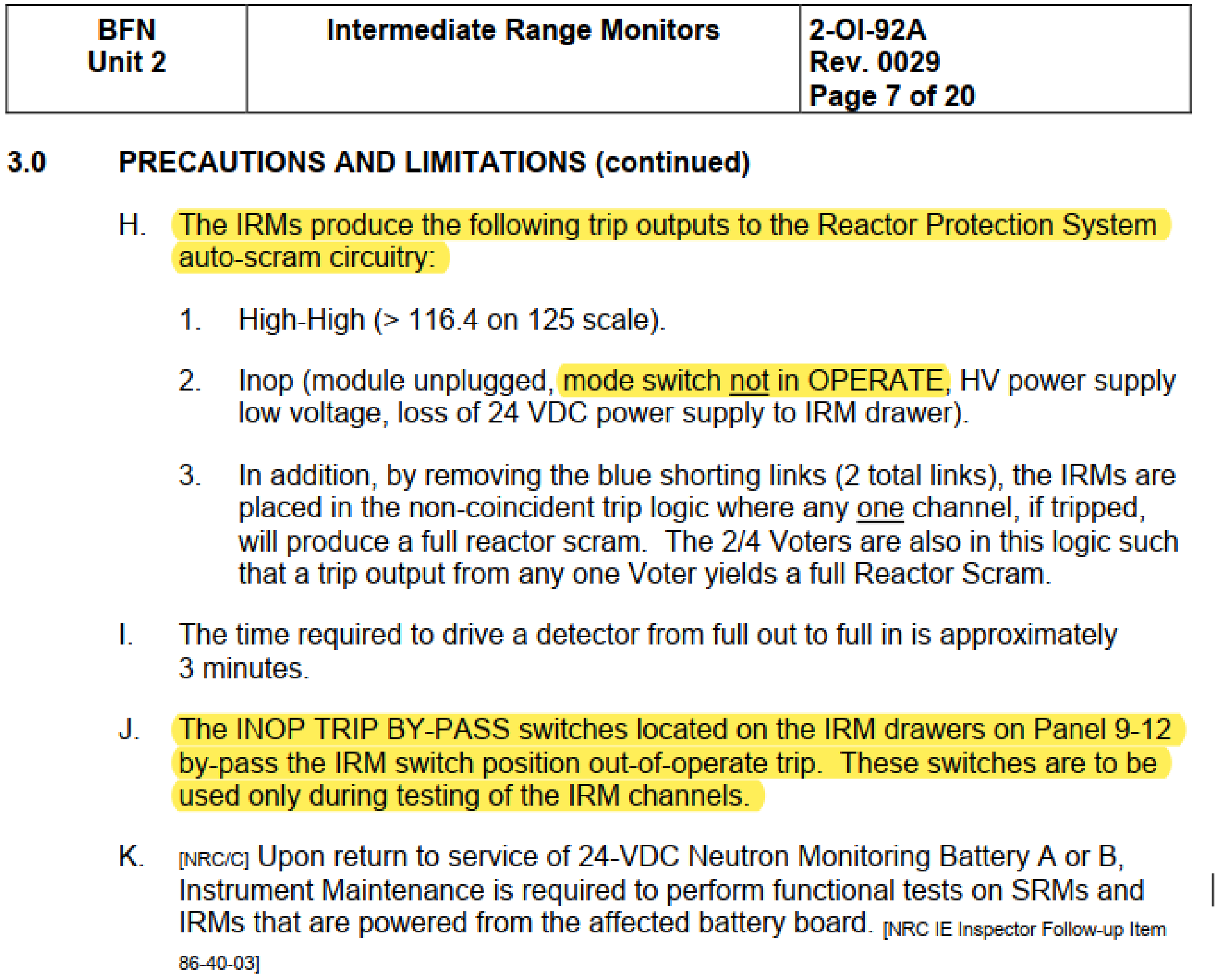
A. Half-scram if one sensor actuates (except with Rx Mode Switch in RUN).
B. Reactor scram if one sensor per channel actuates, (except with Rx Mode Switch in RUN).

Operator Action:

A. **STOP** any reactivity changes.
B. **CHECK** alarm by multiple indications.
C. **PLACE** range switch in a position to clear alarm or **BYPASS** initiating channel to reset half-scram. **REFER TO** 2-OI-92A.
D. With SRO permission, **RESET** Half Scram. **REFER TO** 2-OI-99.
E. **IF** alarm is from a control rod drop, **THEN** **REFER TO** 2-AOI-85-1.
F. [NRC/C] **IF** one or more IRM recorder reading is downscale, **THEN** **CHECK** for loss of ± 24 VDC power.
G. **NOTIFY** Instrument Maintenance that functional tests of any monitors indicating an INOP condition, including a downscale reading, are required before the instrument can be considered operable. [NRC IE item 86-40-03]
H. **NOTIFY** Reactor Engineer.

Continued on Next Page

Excerpts from 2-OI-92A: (Also supports Distractor A(2) and C(2))



BFN **Unit 2**	**Intermediate Range Monitors**	**2-OI-92A** **Rev. 0029** **Page 7 of 20**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

H. The IRMs produce the following trip outputs to the Reactor Protection System auto-scram circuitry:

1. High-High (> 116.4 on 125 scale).

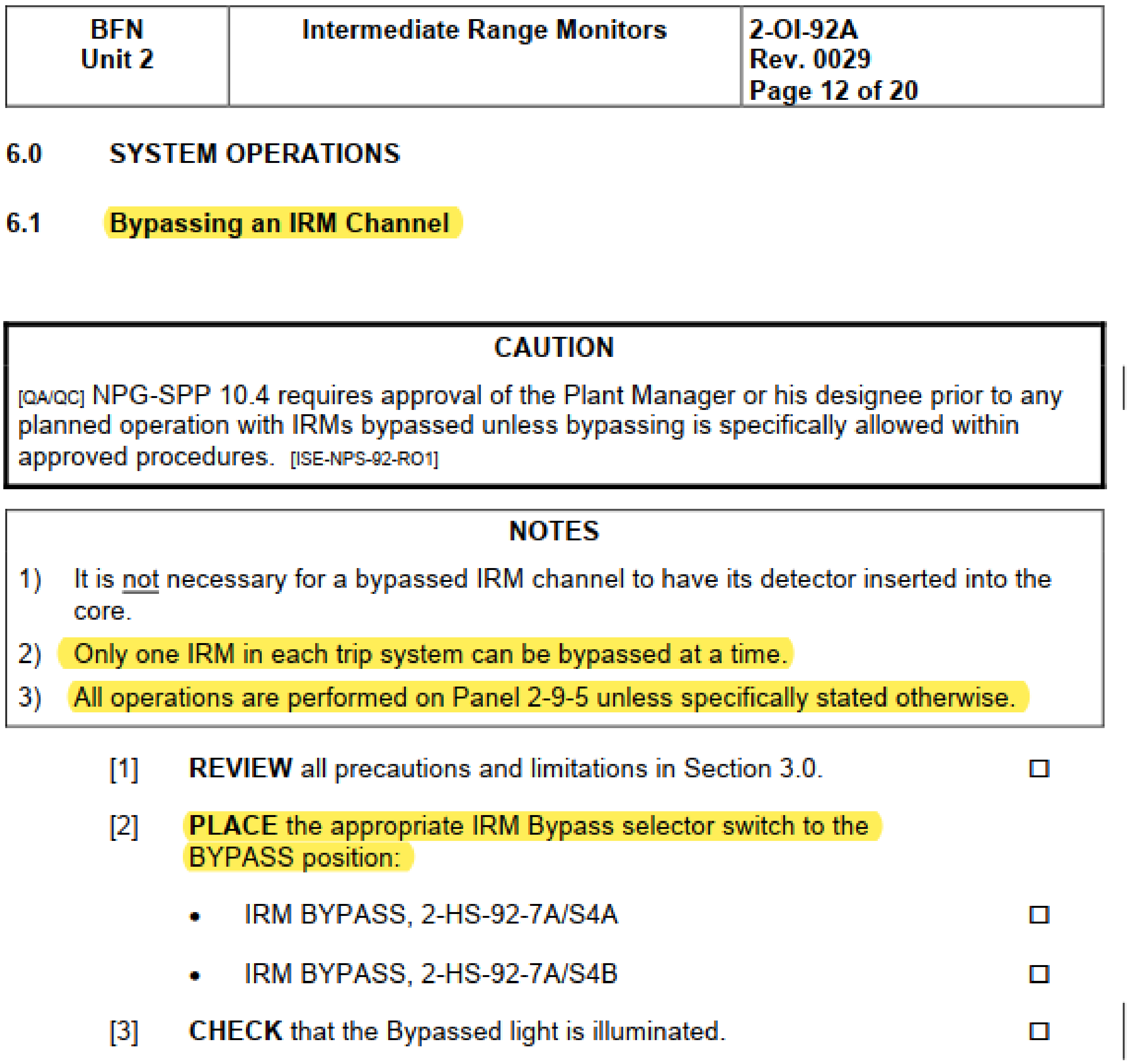
2. Inop (module unplugged, mode switch not in OPERATE, HV power supply low voltage, loss of 24 VDC power supply to IRM drawer).

3. In addition, by removing the blue shorting links (2 total links), the IRMs are placed in the non-coincident trip logic where any one channel, if tripped, will produce a full reactor scram. The 2/4 Voters are also in this logic such that a trip output from any one Voter yields a full Reactor Scram.

I. The time required to drive a detector from full out to full in is approximately 3 minutes.

J. The INOP TRIP BY-PASS switches located on the IRM drawers on Panel 9-12 by-pass the IRM switch position out-of-operate trip. These switches are to be used only during testing of the IRM channels.

K. [NRC/C] Upon return to service of 24-VDC Neutron Monitoring Battery A or B, Instrument Maintenance is required to perform functional tests on SRMs and IRMs that are powered from the affected battery board. [NRC IE Inspector Follow-up Item 86-40-03]



BFN Unit 2	Intermediate Range Monitors	2-OI-92A Rev. 0029 Page 12 of 20

6.0 SYSTEM OPERATIONS

6.1 Bypassing an IRM Channel

CAUTION

[QA/QC] 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. [ISE-NPS-92-RO1]

NOTES

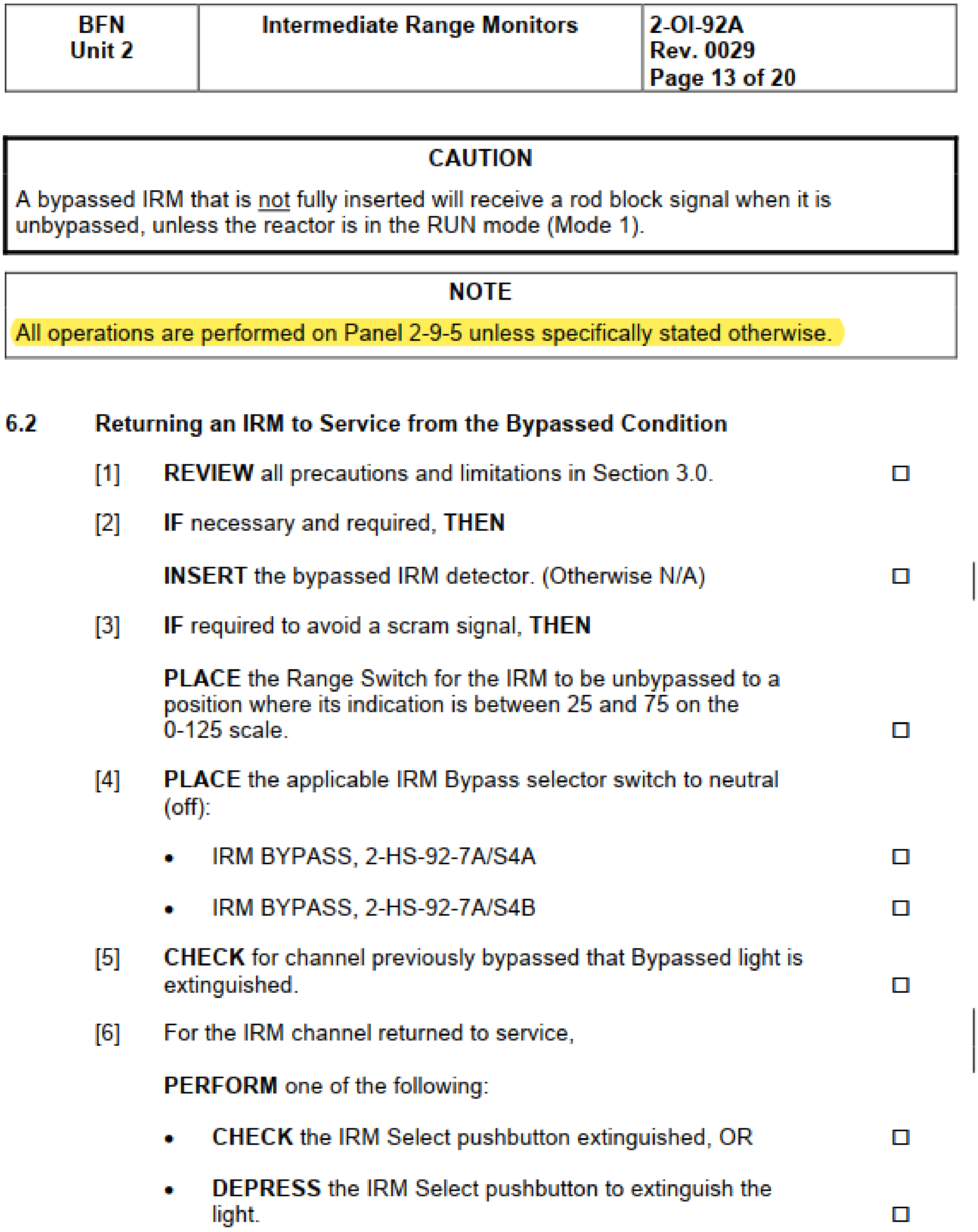
1) It is not necessary for a bypassed IRM channel to have its detector inserted into the core.
2) Only one IRM in each trip system can be bypassed at a time.
3) All operations are performed on Panel 2-9-5 unless specifically stated otherwise.

[1] **REVIEW** all precautions and limitations in Section 3.0. ☐

[2] **PLACE** the appropriate IRM Bypass selector switch to the BYPASS position:

- IRM BYPASS, 2-HS-92-7A/S4A ☐
- IRM BYPASS, 2-HS-92-7A/S4B ☐

[3] **CHECK** that the Bypassed light is illuminated. ☐



BFN Unit 2	**Intermediate Range Monitors**	**2-OI-92A Rev. 0029 Page 13 of 20**

CAUTION

A bypassed IRM that is not fully inserted will receive a rod block signal when it is unbypassed, unless the reactor is in the RUN mode (Mode 1).

NOTE

All operations are performed on Panel 2-9-5 unless specifically stated otherwise.

6.2 Returning an IRM to Service from the Bypassed Condition

[1] **REVIEW** all precautions and limitations in Section 3.0. ☐

[2] **IF** necessary and required, **THEN**

INSERT the bypassed IRM detector. (Otherwise N/A) ☐

[3] **IF** required to avoid a scram signal, **THEN**

PLACE the Range Switch for the IRM to be unbypassed to a position where its indication is between 25 and 75 on the 0-125 scale. ☐

[4] **PLACE** the applicable IRM Bypass selector switch to neutral (off):

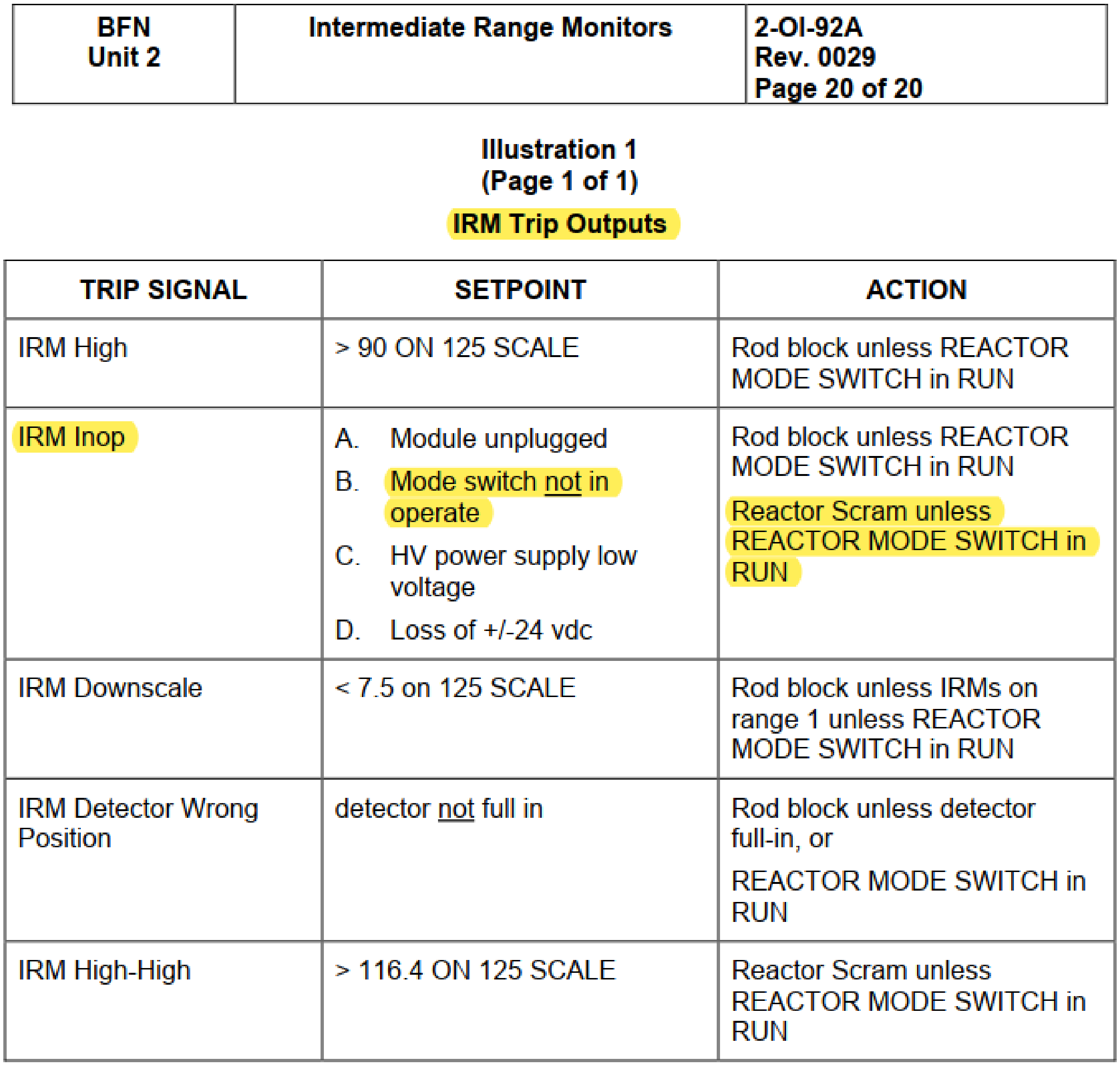
- IRM BYPASS, 2-HS-92-7A/S4A ☐
- IRM BYPASS, 2-HS-92-7A/S4B ☐

[5] **CHECK** for channel previously bypassed that Bypassed light is extinguished. ☐

[6] For the IRM channel returned to service,

PERFORM one of the following:

- **CHECK** the IRM Select pushbutton extinguished, OR ☐
- **DEPRESS** the IRM Select pushbutton to extinguish the light. ☐



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

Examination Outline Cross-reference:

295020 (APE 20) Inadvertent Containment Isolation / 5 & 7

AK1.05 (10CFR 55.41.9)

Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Inadvertent Containment Isolation:

- Loss of drywell/containment cooling system

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295020AK1.05	
Importance Rating	3.5	------

Proposed Question: **# 13**

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, the Drywell Blowers ____**(1)**____ running and 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: First part is correct (*See B*). 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)* In accordance with 1-AOI-64-2D, Group 6 Ventilation System Isolation, a multitude of isolations and closures are listed which occur, however Drywell Blowers do not isolate/trip. For second part, in accordance with 1-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell, the normal Drywell vent path is not available, as bypassing the isolation signals is not allowed in the AOI. Only 1-EOI Appendix-8E, 12, or 13, if necessary, can be used to bypass certain Group 6 Isolations.

C INCORRECT: 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 (Reactor and Refuel Zones) system components. 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 effect of a Primary Containment Isolation signal on Drywell Cooling. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to containment isolations.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s):
OPL171.017, Rev. 22 (Attach if not previously provided)
1-AOI-64-1, Rev. 3
1-AOI-64-2D, Rev. 21
1-EOI-Appendix-8E, Rev. 0
1-EOI-Appendix-12, Rev. 2
1-EOI-Appendix-13, Rev. 4

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.017 Obj.3c (As available)

Question Source:

Bank #	BFN 2104 #27	
Modified Bank #		(Note changes or attach parent)
New		

Question History:

Last NRC Exam	2021

Question Cognitive Level:

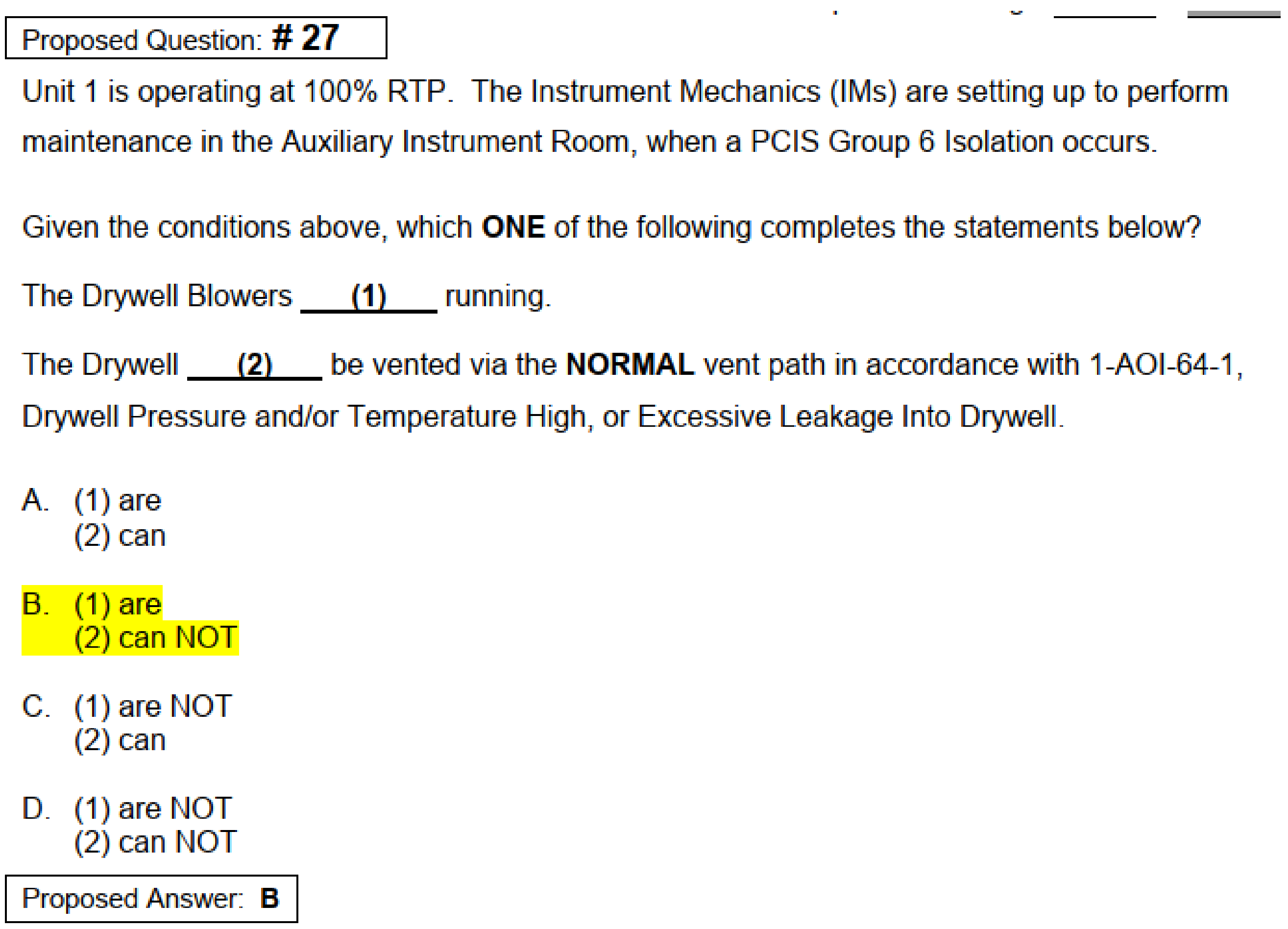
Memory or Fundamental Knowledge	**X**
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Copy of Bank Question:



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

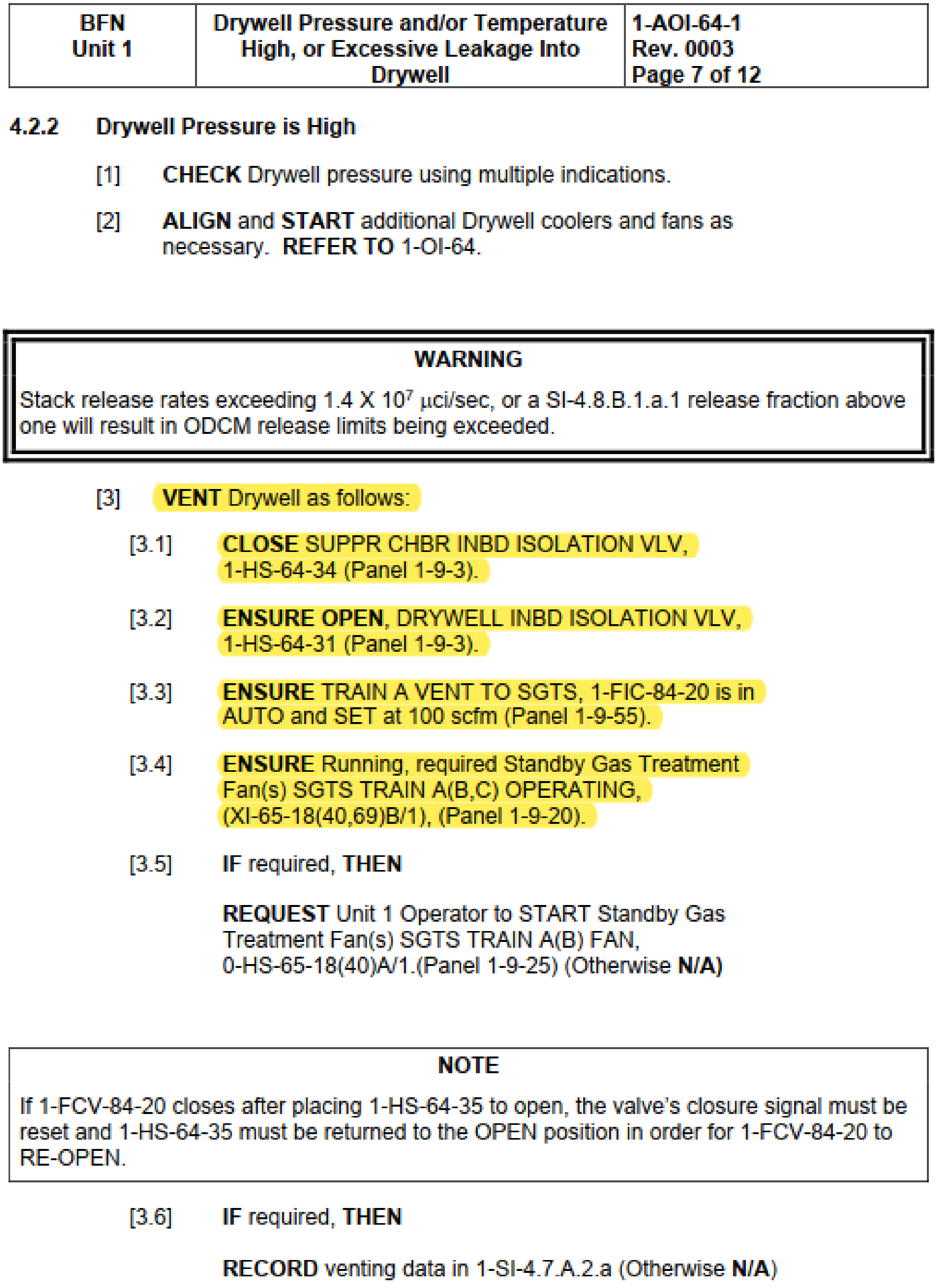
Proposed Answer: **B**

Excerpt from OPL171.017 Lesson Plan:

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 22

(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 H2/O2 ANALYZER ISOLATION BYPASS Keylock switch (HA-76-69) on Panel 9-54 is used to bypass all Group 6 Isolations as necessary to allow placing the H2/O2 Analyzers in service per EOI-2, Step PC/H-2, Appendix 19.	ILT- 3c LOR- 3c
(d) Isolation logic defeating is prescribed in OIs, POIs, etc. for special operating conditions to prevent inadvertent Isolations.	
(e) The Post Accident Sample System (PASS) Isolation valves (PASS Liquid & Gas Return Isolation Valves (HS-43-40/42) and the PASS RHR Supply Isolation Valves (HS-43-50/56) Isolation signals are bypassed by placing the PASS PCIS LOGIC BYPASS Keylock switches (HS-43-39/41) in BYPASS on Panels 9-54/55.	ILT- 3c LOR- 3c
(f) Two keylock switches (64-24, 64-25) on Panel 9-3 bypass the interlock between the reactor mode switch and the drywell and torus vent and purge dampers. These dampers are normally interlocked shut when the mode switch is in RUN. These switches do not bypass a group 6 Isolation.	ILT- 3c LOR- 3c

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. 0003 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] **ENSURE OPEN**, DRYWELL INBD ISOLATION VLV, 1-HS-64-31 (Panel 1-9-3).

[3.3] **ENSURE** TRAIN A VENT TO SGTS, 1-FIC-84-20 is in AUTO and SET at 100 scfm (Panel 1-9-55).

[3.4] **ENSURE** 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**)

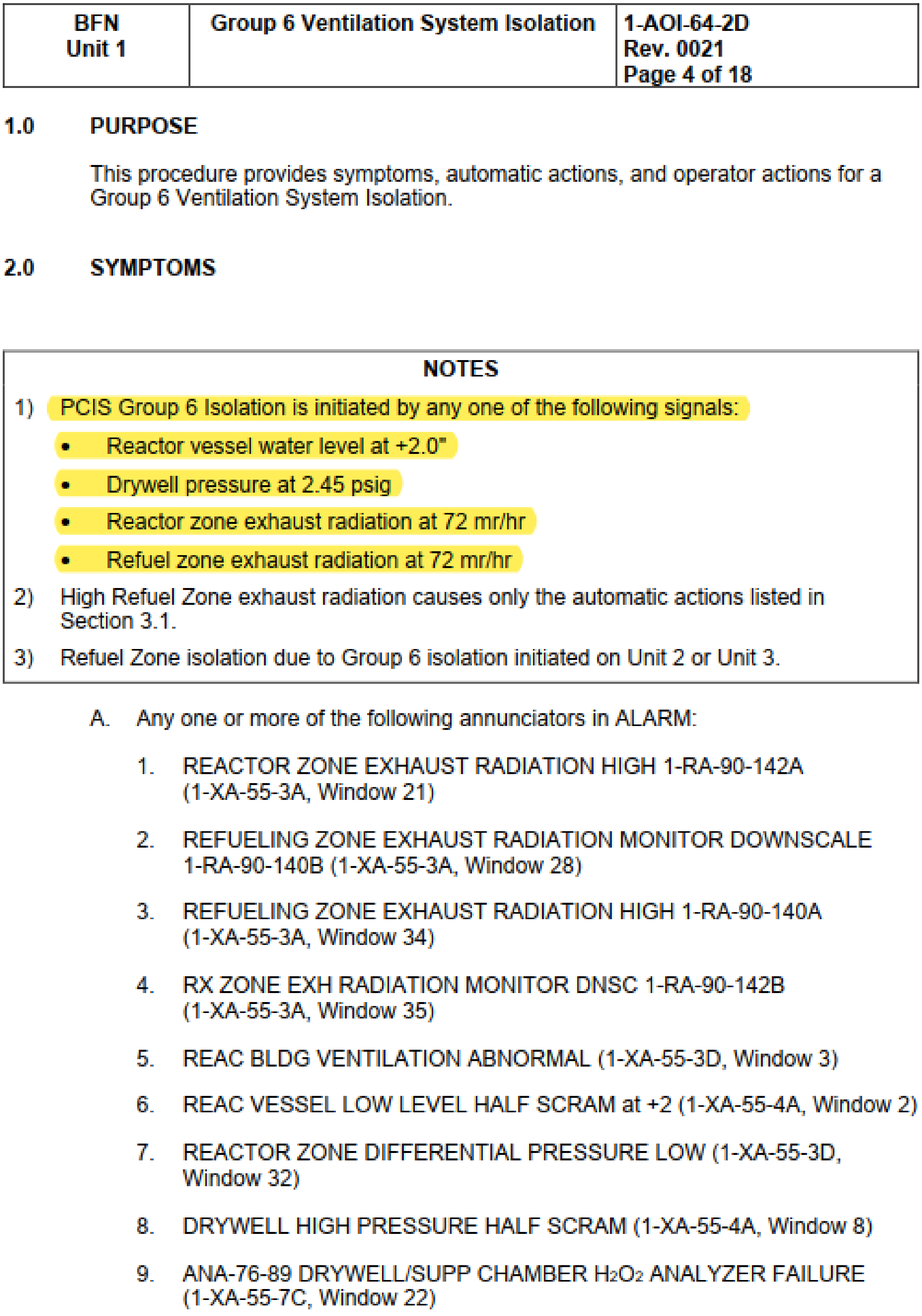
NOTE

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. 0021 Page 4 of 18**

1.0 PURPOSE

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

2.0 SYMPTOMS

NOTES

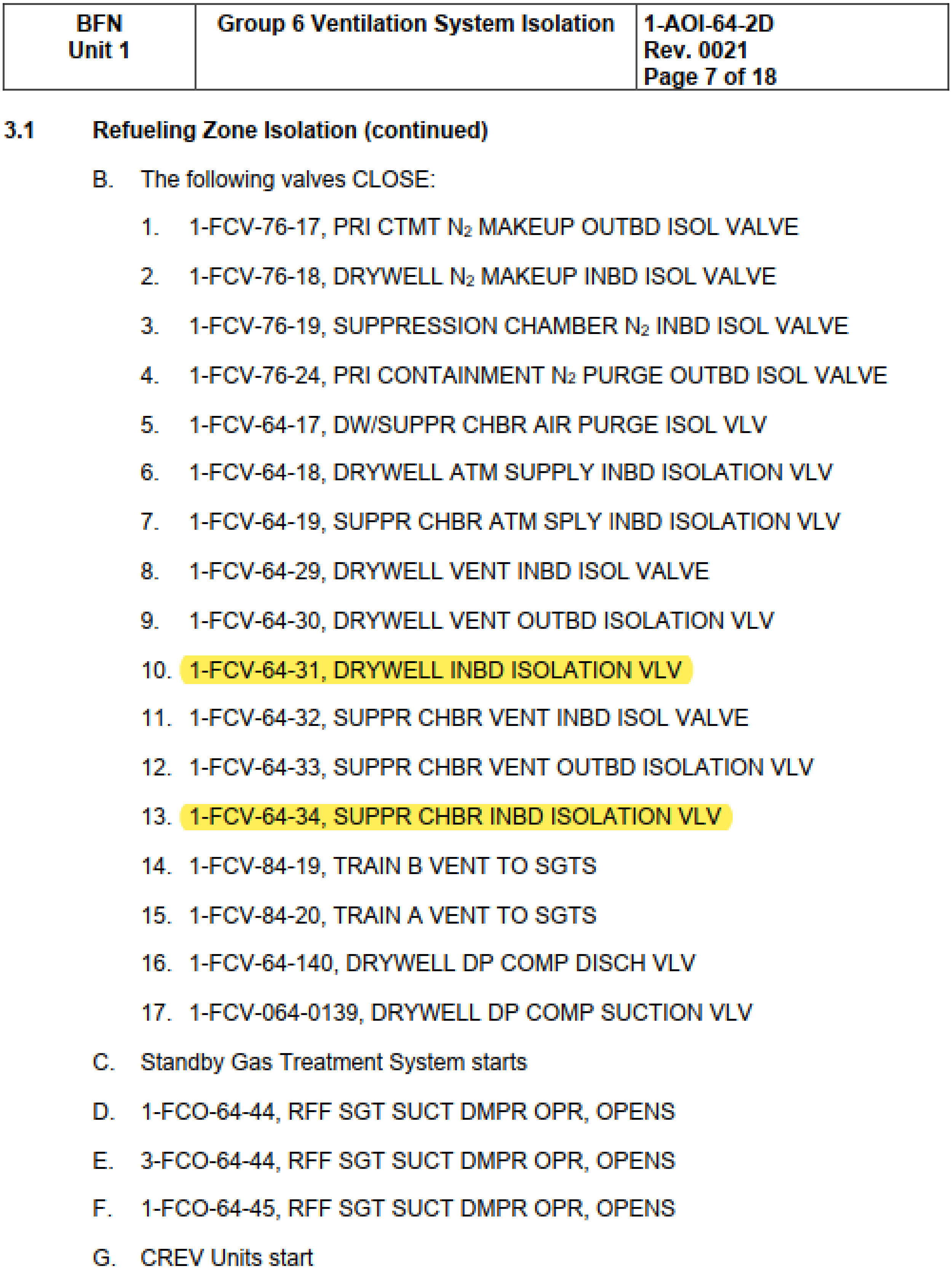
1) PCIS Group 6 Isolation is initiated by any one of the following signals:
   - Reactor vessel water level at +2.0"
   - Drywell pressure at 2.45 psig
   - Reactor zone exhaust radiation at 72 mr/hr
   - Refuel zone exhaust radiation at 72 mr/hr

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. 0021** **Page 7 of 18**

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

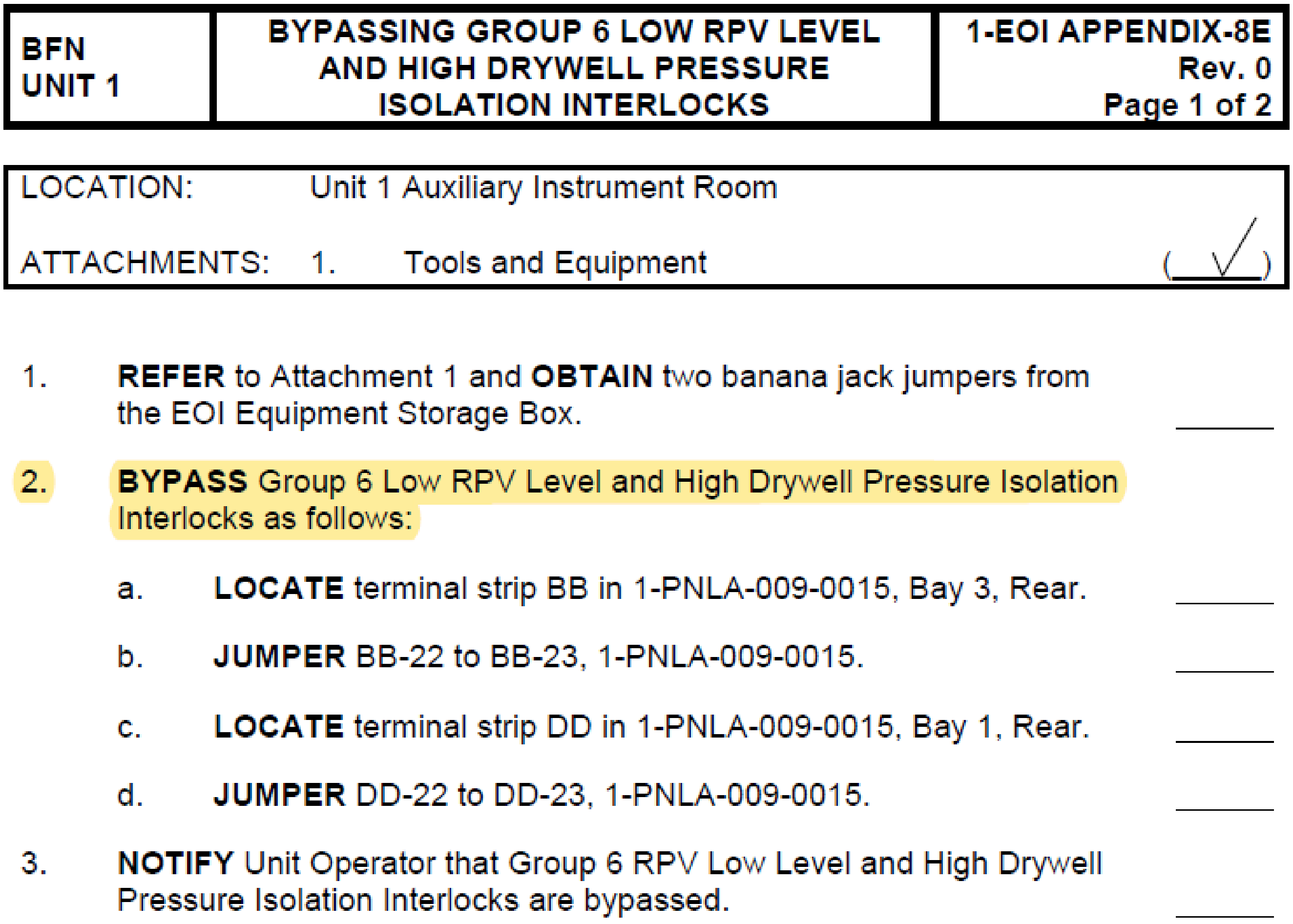
D. 1-FCO-64-44, RFF SGT SUCT DMPR OPR, OPENS

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

Excerpt from 1-EOI-Appendix-8E: Supports Distractors A(2), C(2)



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. _______

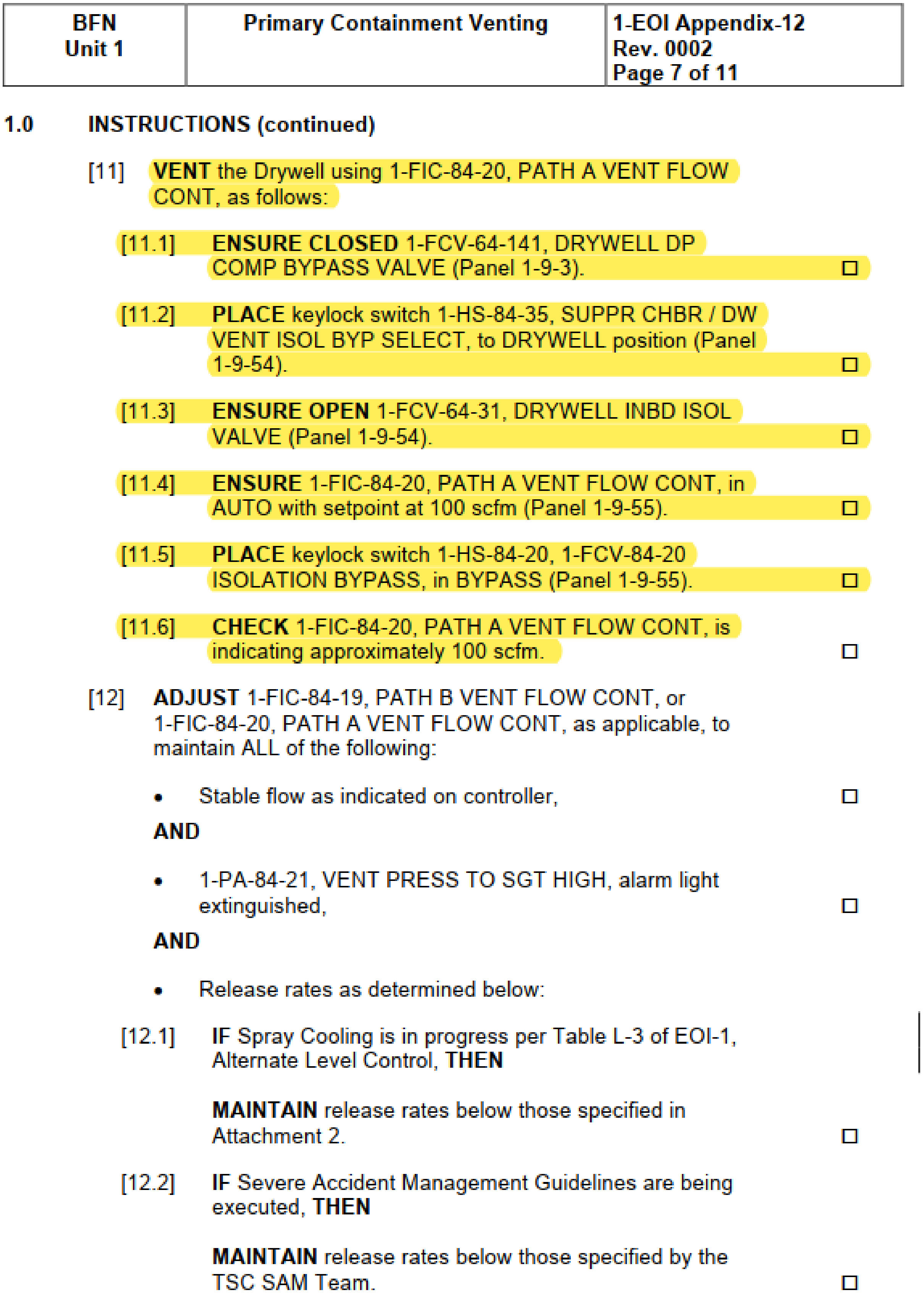
   b. **JUMPER** BB-22 to BB-23, 1-PNLA-009-0015. _______

   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-EOI Appendix-12: Supports Distractors A(2), C(2)



BFN **Unit 1**	**Primary Containment Venting**	**1-EOI Appendix-12** **Rev. 0002** **Page 7 of 11**

1.0 INSTRUCTIONS (continued)

[11] **VENT** the Drywell using 1-FIC-84-20, PATH A VENT FLOW CONT, as follows:

- [11.1] **ENSURE CLOSED** 1-FCV-64-141, DRYWELL DP COMP BYPASS VALVE (Panel 1-9-3). ☐
- [11.2] **PLACE** keylock switch 1-HS-84-35, SUPPR CHBR / DW VENT ISOL BYP SELECT, to DRYWELL position (Panel 1-9-54). ☐
- [11.3] **ENSURE OPEN** 1-FCV-64-31, DRYWELL INBD ISOL VALVE (Panel 1-9-54). ☐
- [11.4] **ENSURE** 1-FIC-84-20, PATH A VENT FLOW CONT, in AUTO with setpoint at 100 scfm (Panel 1-9-55). ☐
- [11.5] **PLACE** keylock switch 1-HS-84-20, 1-FCV-84-20 ISOLATION BYPASS, in BYPASS (Panel 1-9-55). ☐
- [11.6] **CHECK** 1-FIC-84-20, PATH A VENT FLOW CONT, is indicating approximately 100 scfm. ☐

[12] **ADJUST** 1-FIC-84-19, PATH B VENT FLOW CONT, or 1-FIC-84-20, PATH A VENT FLOW CONT, as applicable, to maintain ALL of the following:

- Stable flow as indicated on controller, ☐

AND

- 1-PA-84-21, VENT PRESS TO SGT HIGH, alarm light extinguished, ☐

AND

- Release rates as determined below:

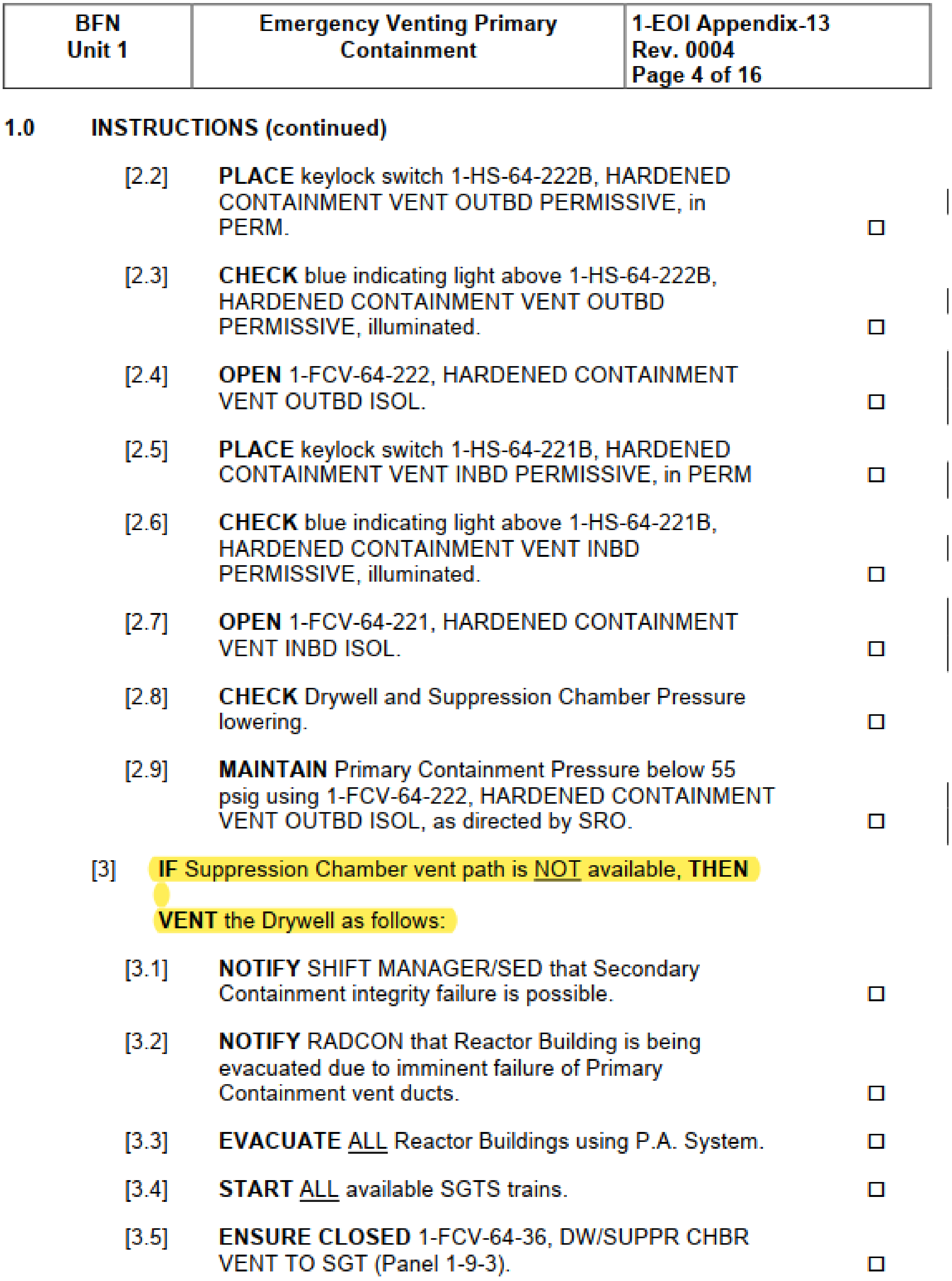
[12.1] **IF** Spray Cooling is in progress per Table L-3 of EOI-1, Alternate Level Control, **THEN**

MAINTAIN release rates below those specified in Attachment 2. ☐

[12.2] **IF** Severe Accident Management Guidelines are being executed, **THEN**

MAINTAIN release rates below those specified by the TSC SAM Team. ☐

Excerpt from 1-EOI Appendix-13: Supports Distractors A(2), C(2)



BFN **Unit 1**	**Emergency Venting Primary Containment**	**1-EOI Appendix-13** **Rev. 0004** **Page 4 of 16**

1.0 INSTRUCTIONS (continued)

[2.2] **PLACE** keylock switch 1-HS-64-222B, HARDENED CONTAINMENT VENT OUTBD PERMISSIVE, in PERM. ☐

[2.3] **CHECK** blue indicating light above 1-HS-64-222B, HARDENED CONTAINMENT VENT OUTBD PERMISSIVE, illuminated. ☐

[2.4] **OPEN** 1-FCV-64-222, HARDENED CONTAINMENT VENT OUTBD ISOL. ☐

[2.5] **PLACE** keylock switch 1-HS-64-221B, HARDENED CONTAINMENT VENT INBD PERMISSIVE, in PERM ☐

[2.6] **CHECK** blue indicating light above 1-HS-64-221B, HARDENED CONTAINMENT VENT INBD PERMISSIVE, illuminated. ☐

[2.7] **OPEN** 1-FCV-64-221, HARDENED CONTAINMENT VENT INBD ISOL. ☐

[2.8] **CHECK** Drywell and Suppression Chamber Pressure lowering. ☐

[2.9] **MAINTAIN** Primary Containment Pressure below 55 psig using 1-FCV-64-222, HARDENED CONTAINMENT VENT OUTBD ISOL, 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** 1-FCV-64-36, DW/SUPPR CHBR VENT TO SGT (Panel 1-9-3). ☐

Examination Outline Cross-reference:

263000 (SF6 DC) DC Electrical Distribution
A1.02 (10CFR 55.41.5)
Ability to predict and/or monitor changes in parameters associated with operation of the DC Electrical Distribution, including:

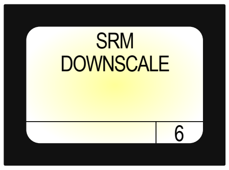
- Lights and alarms

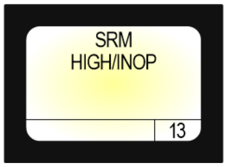
Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	263000A1.02	
Importance Rating	3.3	------

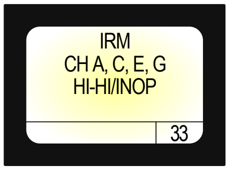
Proposed Question: **# 14**

A Unit 3 Reactor Startup is in progress, when the Operator observes the following:

- SRM DOWNSCALE, (3-9-5A, Window 6), alarms
- SRM HIGH/INOP, (3-9-5A, Window 13) alarms
- IRM CH A, C, E, G HI-HI/INOP, (3-9-5A, Window 33), alarms







Given the conditions above, which **ONE** of the following power sources, if lost, would cause these failures?

A. 24VDC Power Distribution Panel

B. 125VDC Power Distribution Panel

C. 120VAC RPS Power Supply Distribution Panel

D. 120VAC Instrument and Control Power Distribution Panel

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with the given alarm response procedures for the alarms SRM HIGH/NOP (IRM CH A, C, E, G HI-HI/INOP), if one or more SRM (IRM) recorder reading is downscale, then check for a loss of +/- 24VDC. SRM and IRM power supplies are from unregulated +/- 24 VDC power from the Neutron Monitoring batteries.

B INCORRECT: Incorrect but plausible in that all neutron monitoring detectors receive various voltage supplies from 0-350 Volts. Any combination of those voltages could be thought to be correct by the candidate. Although supplied voltages can be varied, the actual 125V distribution panel does not provide power to the neutron monitoring system.

C INCORRECT: Incorrect but plausible in that since RPS provides power to the IRM trip units since they are in RPS logic. Stated alarms are not indicative of a loss of RPS.

D INCORRECT: Incorrect but plausible in that since I&C provides power to various IRM components including drives. The stated alarms would not be indicative of a loss of I&C.

RO Level Justification: Tests the candidate's ability to predict and monitor changes in parameters associated with the DC Electrical Distribution as it relates to lights and alarms. 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-ARP-9-5A, Rev. 54	(Attach if not previously provided)
	3-OI-92, Rev. 17	
	3-OI-92A, Rev. 18	
	OPL171.019, Rev. 15	
	OPL171.020, Rev. 13	

Proposed references to be provided to applicants during examination: **SRM DOWNSCALE, (3-9-5A, Window 6), SRM HIGH/INOP, (3-9-5A, Window 13), IRM CH A, C, E, G HI-HI/INOP, (3-9-5A, Window 33)**

Learning Objective: OPL171.019 Obj 6b (As available)

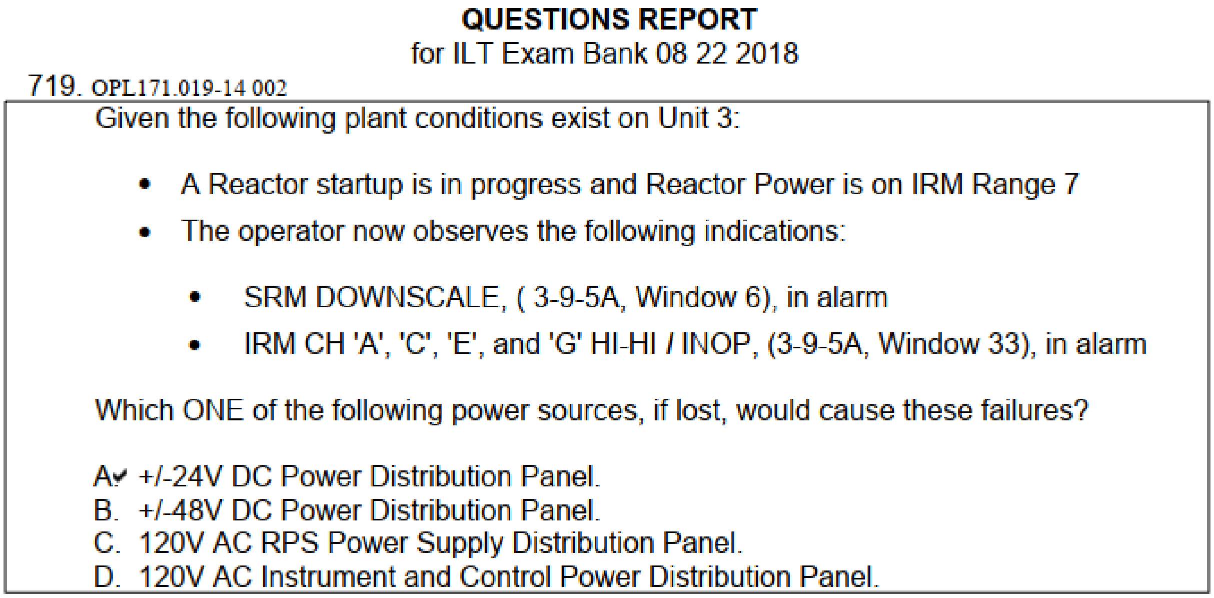
Question Source:	Bank #	OPL171.019-14 002 #719	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Copy of Bank Question:



QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

719. OPL171.019-14 002

Given the following plant conditions exist on Unit 3:

- A Reactor startup is in progress and Reactor Power is on IRM Range 7
- The operator now observes the following indications:
  - SRM DOWNSCALE, (3-9-5A, Window 6), in alarm
  - IRM CH 'A', 'C', 'E', and 'G' HI-HI / INOP, (3-9-5A, Window 33), in alarm

Which ONE of the following power sources, if lost, would cause these failures?

A.✓ +/-24V DC Power Distribution Panel.
B. +/-48V DC Power Distribution Panel.
C. 120V AC RPS Power Supply Distribution Panel.
D. 120V AC Instrument and Control Power Distribution Panel.

Excerpts from 3-ARP-9-5A:

BFN **Unit 3**	**Panel 9-5** **3-XA-55-5A**	**3-ARP-9-5A** **Rev. 0054** **Page 10 of 48**

SRM
DOWNSCALE

6

(Page 1 of 1)

Sensor/Trip Point:

Relay K-19 Count rate ≤ 5 cps.

Sensor Location: Control Room Panel 3-9-12.

Probable Cause:

A. Unbypassed SRM channel with count rate at or below 3 counts per second.
B. SI or SR in progress.
C. Malfunction of sensor.

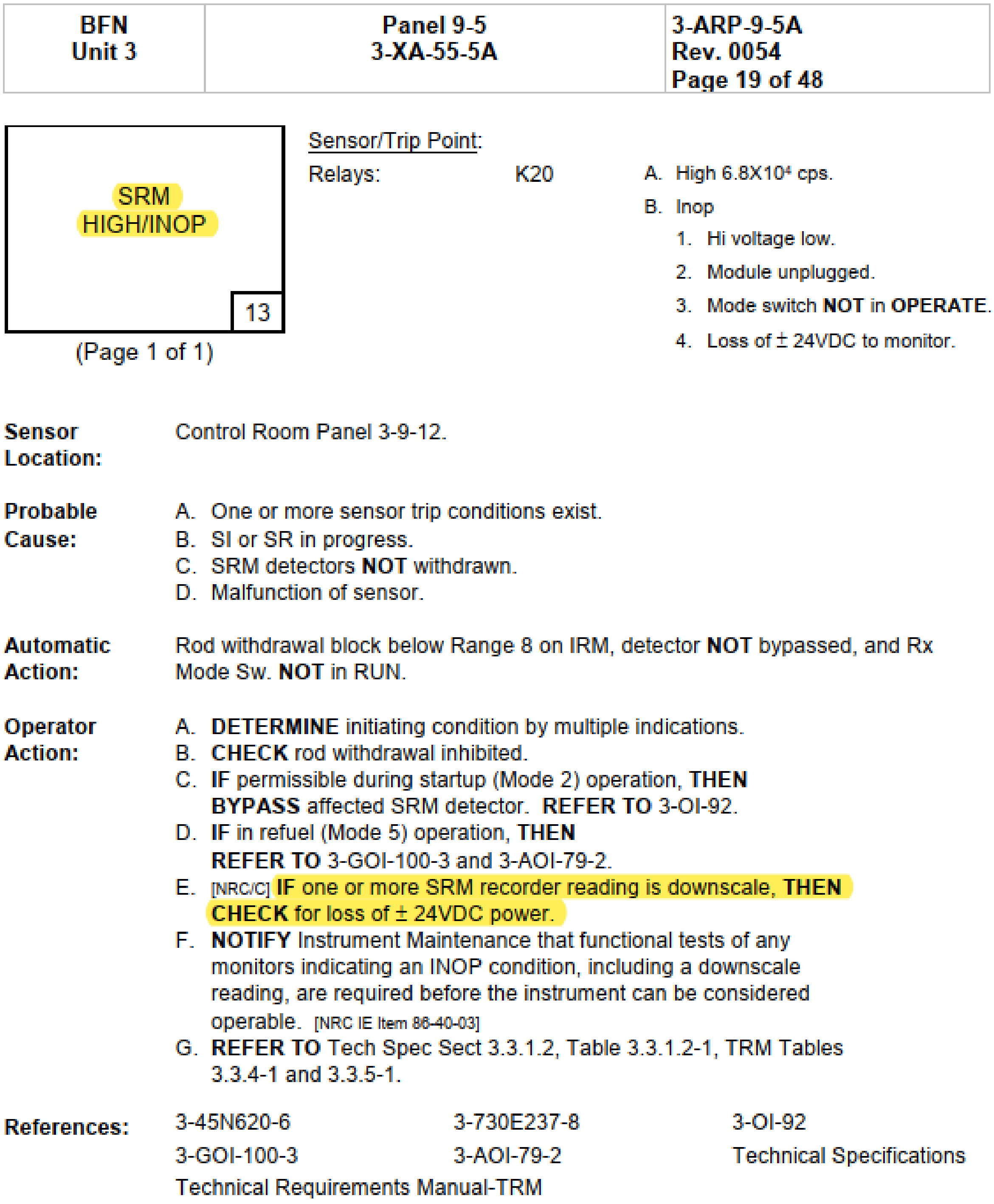
Automatic Action: Rod block below IRM range 3 and Rx Mode Sw. **NOT** in RUN.

Operator Action:

A. **CHECK** SRM downscale.
B. **IF** alarm is valid, **THEN** **REFER TO** 3-OI-92 during startup Mode 2 operation or 0-GOI-100-3C during refuel Mode 5 operation.
C. **REFER TO** Tech Spec Sect 3.3.1.2, Table 3.3.1.2-1, TRM Tables 3.3.4-1 and 3.3.5-1.

References:

3-45E620-6	3-730E237-8	3-OI-92
0-GOI-100-3C	Technical Specifications	Technical Requirements Manual-TRM



BFN **Unit 3**	**Panel 9-5** **3-XA-55-5A**	**3-ARP-9-5A** **Rev. 0054** **Page 19 of 48**

SRM
HIGH/INOP
13

(Page 1 of 1)

Sensor/Trip Point:

Relays: K20

A. High $6.8X10^4$ cps.

B. Inop
  1. Hi voltage low.
  2. Module unplugged.
  3. Mode switch **NOT** in **OPERATE**.
  4. Loss of ± 24VDC to monitor.

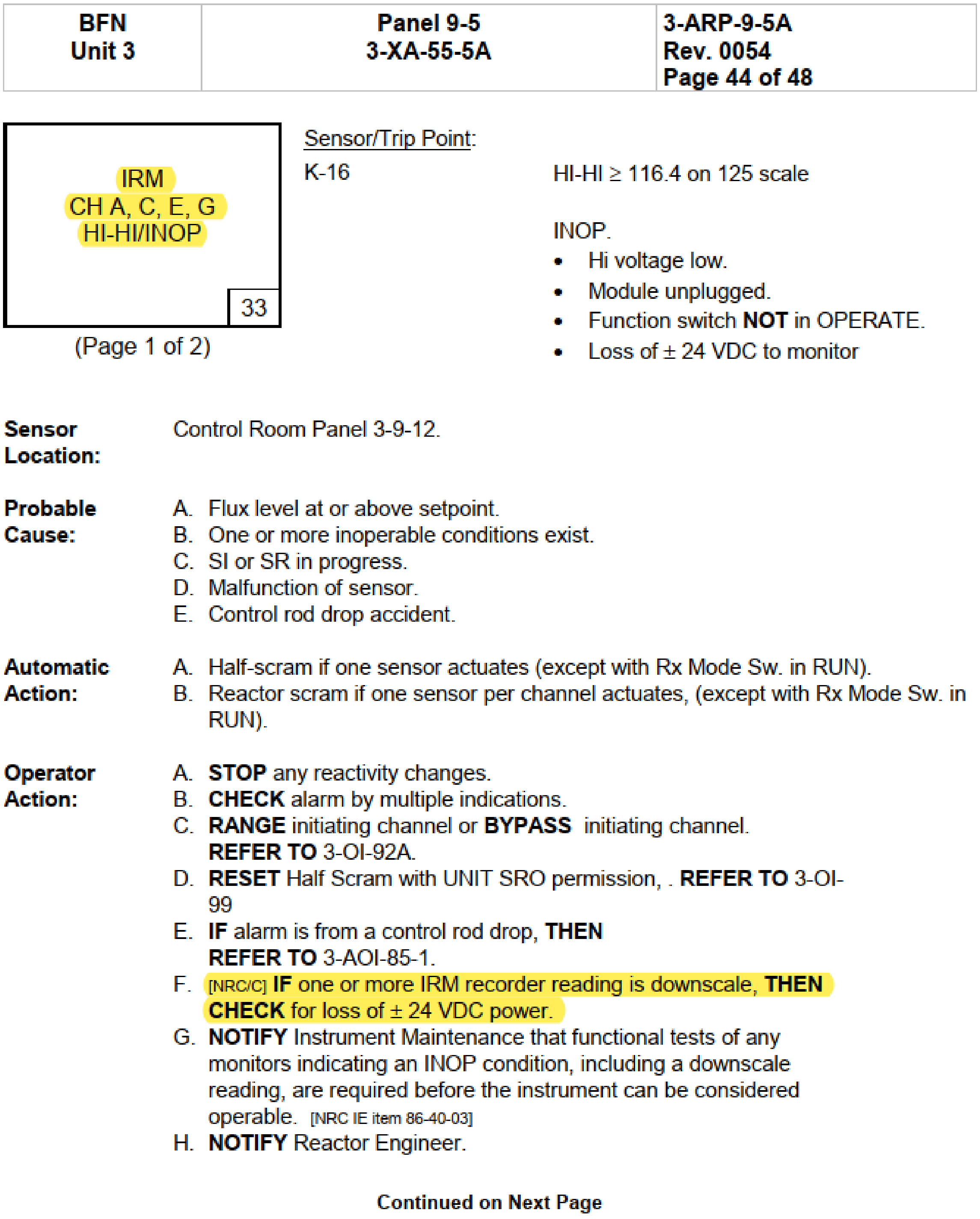
Sensor Location: Control Room Panel 3-9-12.

Probable Cause:
A. One or more sensor trip conditions exist.
B. SI or SR in progress.
C. SRM detectors **NOT** withdrawn.
D. Malfunction of sensor.

Automatic Action: Rod withdrawal block below Range 8 on IRM, detector **NOT** bypassed, and Rx Mode Sw. **NOT** in RUN.

Operator Action:
A. **DETERMINE** initiating condition by multiple indications.
B. **CHECK** rod withdrawal inhibited.
C. **IF** permissible during startup (Mode 2) operation, **THEN** **BYPASS** affected SRM detector. **REFER TO** 3-OI-92.
D. **IF** in refuel (Mode 5) operation, **THEN** **REFER TO** 3-GOI-100-3 and 3-AOI-79-2.
E. [NRC/C] **IF** one or more SRM recorder reading is downscale, **THEN** **CHECK** for loss of ± 24VDC power.
F. **NOTIFY** Instrument Maintenance that functional tests of any monitors indicating an INOP condition, including a downscale reading, are required before the instrument can be considered operable. [NRC IE Item 86-40-03]
G. **REFER TO** Tech Spec Sect 3.3.1.2, Table 3.3.1.2-1, TRM Tables 3.3.4-1 and 3.3.5-1.

References:
3-45N620-6 3-730E237-8 3-OI-92
3-GOI-100-3 3-AOI-79-2 Technical Specifications
Technical Requirements Manual-TRM



BFN Unit 3	**Panel 9-5 3-XA-55-5A**	**3-ARP-9-5A Rev. 0054 Page 44 of 48**

IRM
CH A, C, E, G
HI-HI/INOP
33

(Page 1 of 2)

Sensor/Trip Point:

K-16 HI-HI ≥ 116.4 on 125 scale

INOP.
- Hi voltage low.
- Module unplugged.
- Function switch **NOT** in OPERATE.
- Loss of ± 24 VDC to monitor

Sensor Location: Control Room Panel 3-9-12.

Probable Cause:

A. Flux level at or above setpoint.
B. One or more inoperable conditions exist.
C. SI or SR in progress.
D. Malfunction of sensor.
E. Control rod drop accident.

Automatic Action:

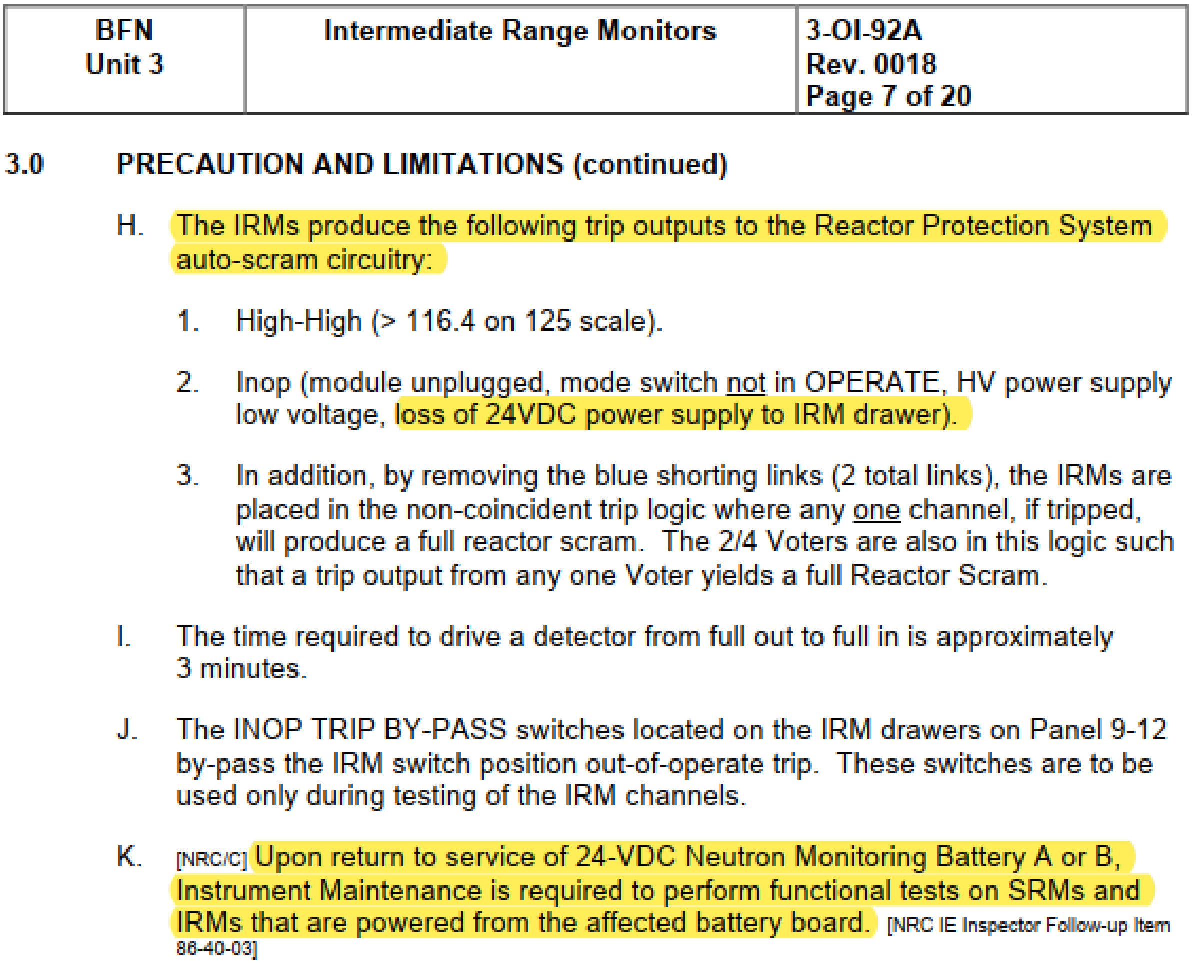
A. Half-scram if one sensor actuates (except with Rx Mode Sw. in RUN).
B. Reactor scram if one sensor per channel actuates, (except with Rx Mode Sw. in RUN).

Operator Action:

A. **STOP** any reactivity changes.
B. **CHECK** alarm by multiple indications.
C. **RANGE** initiating channel or **BYPASS** initiating channel. **REFER TO** 3-OI-92A.
D. **RESET** Half Scram with UNIT SRO permission, . **REFER TO** 3-OI-99
E. **IF** alarm is from a control rod drop, **THEN** **REFER TO** 3-AOI-85-1.
F. [NRC/C] **IF** one or more IRM recorder reading is downscale, **THEN** **CHECK** for loss of ± 24 VDC power.
G. **NOTIFY** Instrument Maintenance that functional tests of any monitors indicating an INOP condition, including a downscale reading, are required before the instrument can be considered operable. [NRC IE item 86-40-03]
H. **NOTIFY** Reactor Engineer.

Continued on Next Page

Excerpts from 3-OI-92A:



BFN Unit 3	**Intermediate Range Monitors**	**3-OI-92A Rev. 0018 Page 7 of 20**

3.0 PRECAUTION AND LIMITATIONS (continued)

H. The IRMs produce the following trip outputs to the Reactor Protection System auto-scram circuitry:

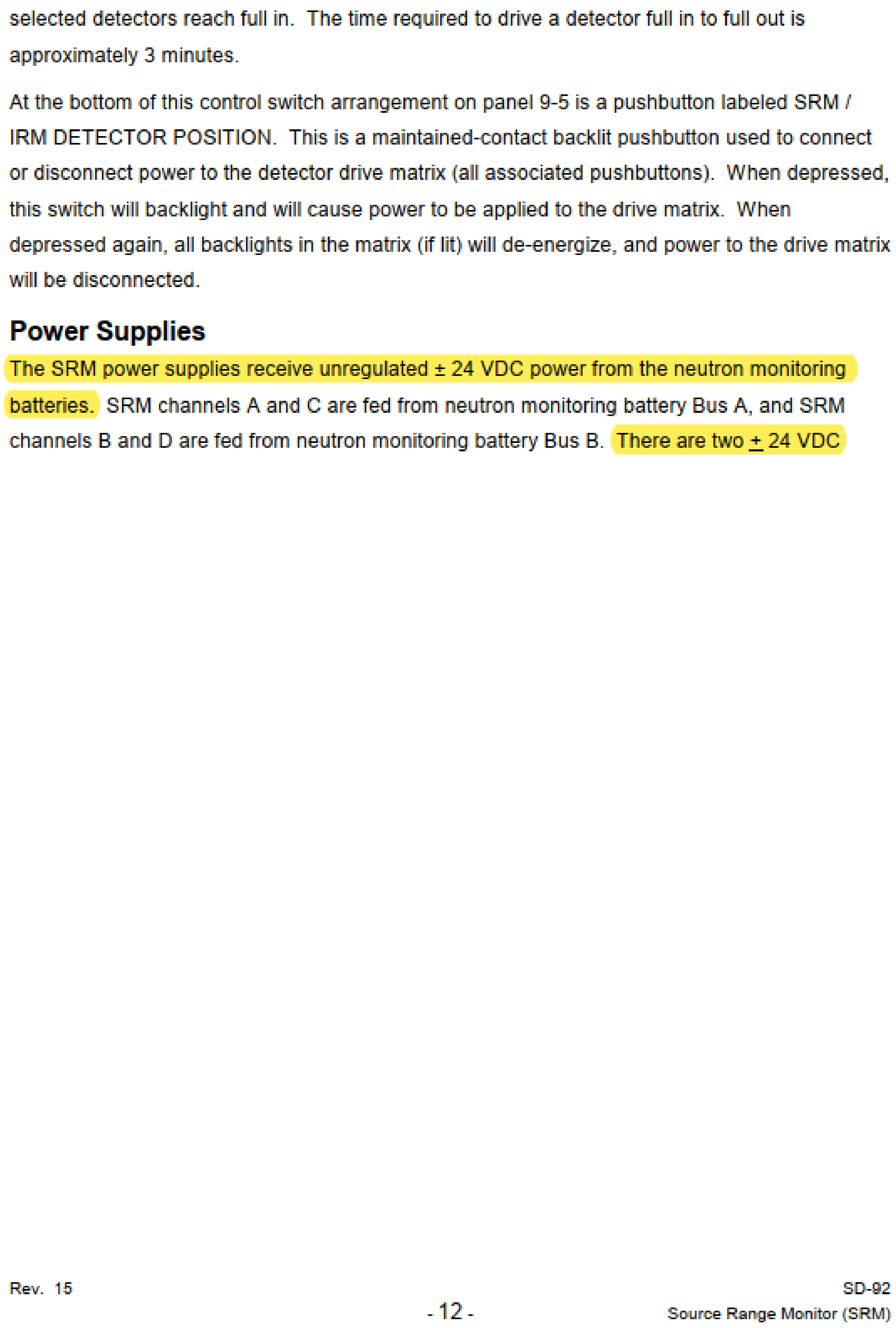
1. High-High (> 116.4 on 125 scale).
2. Inop (module unplugged, mode switch <u>not</u> in OPERATE, HV power supply low voltage, loss of 24VDC power supply to IRM drawer).
3. In addition, by removing the blue shorting links (2 total links), the IRMs are placed in the non-coincident trip logic where any <u>one</u> channel, if tripped, will produce a full reactor scram. The 2/4 Voters are also in this logic such that a trip output from any one Voter yields a full Reactor Scram.

I. The time required to drive a detector from full out to full in is approximately 3 minutes.

J. The INOP TRIP BY-PASS switches located on the IRM drawers on Panel 9-12 by-pass the IRM switch position out-of-operate trip. These switches are to be used only during testing of the IRM channels.

K. [NRC/C] Upon return to service of 24-VDC Neutron Monitoring Battery A or B, Instrument Maintenance is required to perform functional tests on SRMs and IRMs that are powered from the affected battery board. [NRC IE Inspector Follow-up Item 86-40-03]

Excerpts from OPL171.019:



selected detectors reach full in. The time required to drive a detector full in to full out is approximately 3 minutes.

At the bottom of this control switch arrangement on panel 9-5 is a pushbutton labeled SRM / IRM DETECTOR POSITION. This is a maintained-contact backlit pushbutton used to connect or disconnect power to the detector drive matrix (all associated pushbuttons). When depressed, this switch will backlight and will cause power to be applied to the drive matrix. When depressed again, all backlights in the matrix (if lit) will de-energize, and power to the drive matrix will be disconnected.

Power Supplies

The SRM power supplies receive unregulated ± 24 VDC power from the neutron monitoring batteries. SRM channels A and C are fed from neutron monitoring battery Bus A, and SRM channels B and D are fed from neutron monitoring battery Bus B. There are two ± 24 VDC

Rev. 15

- 12 -

SD-92
Source Range Monitor (SRM)

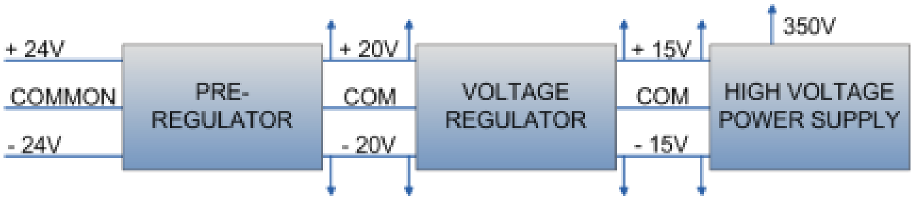
battery chargers for each battery. The two battery chargers are connected in parallel with each 24V section of the center-tapped 48V neutron monitoring batteries.

±24V Neutron Monitoring Battery Chargers			
	Unit 1	Unit 2	Unit 3
Bus A Battery Chargers	A1-1 and A2-1	A1-2 and A2-2	A1-3 and A2-3
Bus B Battery Chargers	B1-1 and B2-1	B1-2 and B2-2	B1-3 and B2-3

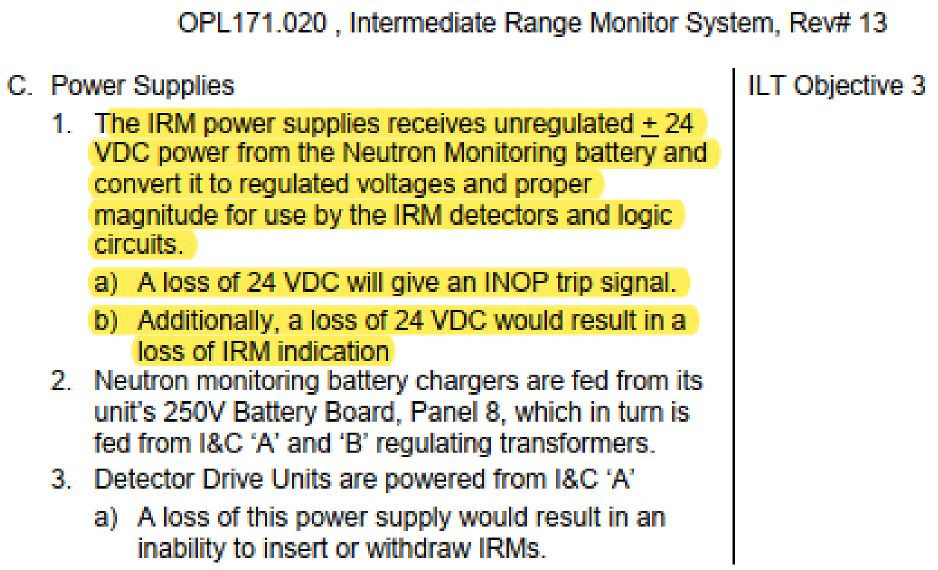
The neutron monitoring system battery chargers are powered by 120 VAC Instrument and Control Buses. The ± 24 VDC is converted to regulated voltages of proper magnitude within the SRM circuitry for use by the SRM detectors and logic circuits.

There are three levels of voltage regulation:

- The pre-regulator takes the ± 24 VDC input and reduces it to a relatively constant ± 20 VDC for use by the voltage regulator.
- The voltage regulator takes the ± 20 VDC input from the pre-regulator and reduces it to a well regulated ± 15 VDC for use by the logic and high voltage power supply.
- The high voltage power supply takes the ± 15 VDC input from the voltage regulator and produces an adjustable voltage (100-350 VDC) for use as the operating bias on the detector. The high voltage is set at 350 VDC.



Excerpts from OPL171.020:



OPL171.020 , Intermediate Range Monitor System, Rev# 13

C. Power Supplies

ILT Objective 3

1. The IRM power supplies receives unregulated ± 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.

Examination Outline Cross-reference:	Level	RO	SRO
295003 (APE 3) Partial or Complete Loss of A.C. Power / 6	Tier #	1	------
G2.1.7 (10CFR 55.41.5)	Group #	1	------
Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.	K/A #	295003G2.1.7	
	Importance Rating	4.4	------

Proposed Question: **# 15**

All three Units are operating at 100% RTP when the following conditions occur:

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

Given the conditions above, in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV)/ Station Blackout, the minimum action(s) required to re-energize the affected 480V Distribution Board(s) is to transfer 480V Shutdown Board ___**(1)**___ to alternate.

The respective board ___**(2)**___ be manually transferred from Main Control Room (MCR) Panel 9-8.

A. (1) 1B
(2) can

B. (1) 1B
(2) can NOT

C. (1) 2A
(2) can

D. (1) 2A
(2) can NOT

Proposed Answer: **C**

Explanation (Optional):		
	A	INCORRECT: The first part is incorrect but plausible in that 480V Shutdown Board 1B’s normal power supply is 4KV Shutdown Board C (‘C’ EDG) which was not impacted in the stem of the question. 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 if the candidate confuses 480V Shutdown Board transfer capabilities from Panel 9-8 with 480V RMOV Boards Only 480V RMOV Boards 1/2/3 C’s can be transferred to alternate from Panel 9-8.

C **CORRECT:** *(See attached)* In accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV) Station Blackout, Restoration of Electrical Busses is performed using Attachment 1 and Table 1. Given a Loss of Offsite occurs and 'B' EDG fails to start, the normal and Diesel Generator power supply were lost therefore the minimum actions is to transfer 480V Shutdown Board 2A to alternate. For second part, 0-AOI-57-1A Attachment 1, Restoration of Electrical Busses, states that 480V Shutdown Board 1A, 1B, 2A, 2B, 3A, 3B can be transferred from the respective Unit's Panel 9-8.

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 to make operational judgements as it relates to a Loss of Offsite Power with a loss of an Emergency Diesel Generator. 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 related to the complex BFN's A.C. Electrical Distribution System.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 0-AOI-57-1A, Rev. 114 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

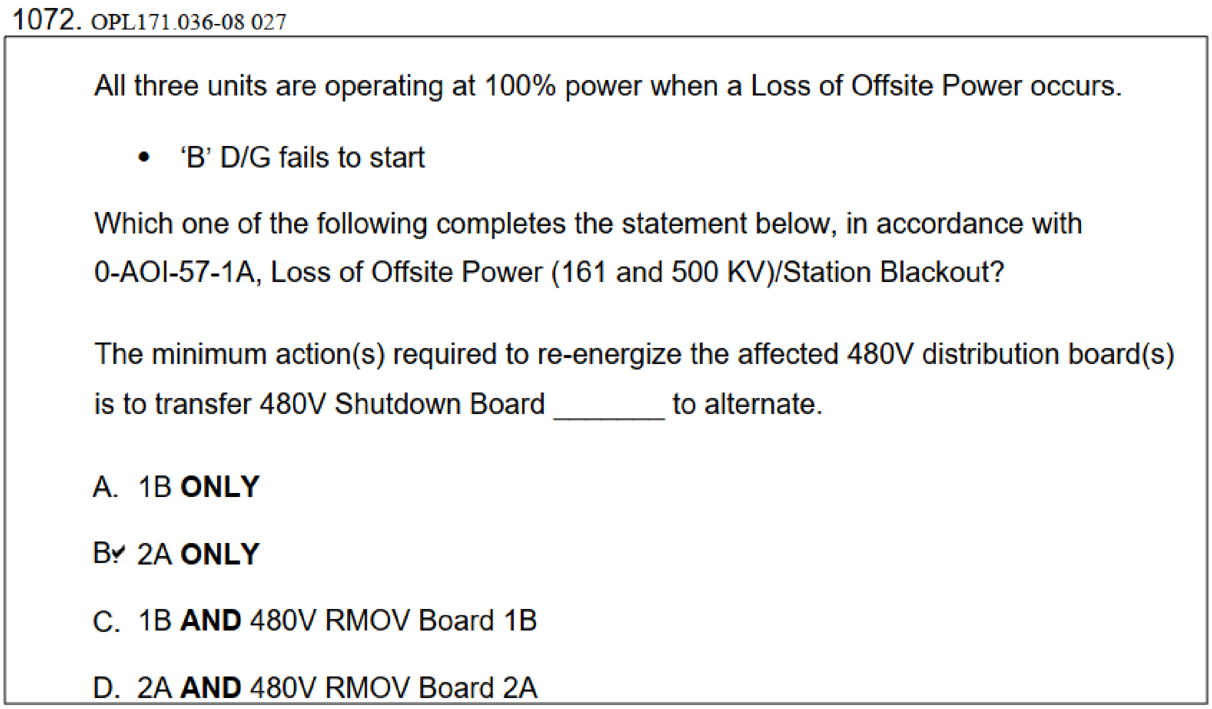
Question Source:	Bank #		
	Modified Bank #	OPL171.036-08 027 #1072	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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:



1072. OPL171.036-08 027

All three units are operating at 100% power when a Loss of Offsite Power occurs.

- ‘B’ D/G fails to start

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

The minimum action(s) required to re-energize the affected 480V distribution board(s) is to transfer 480V Shutdown Board ________ to alternate.

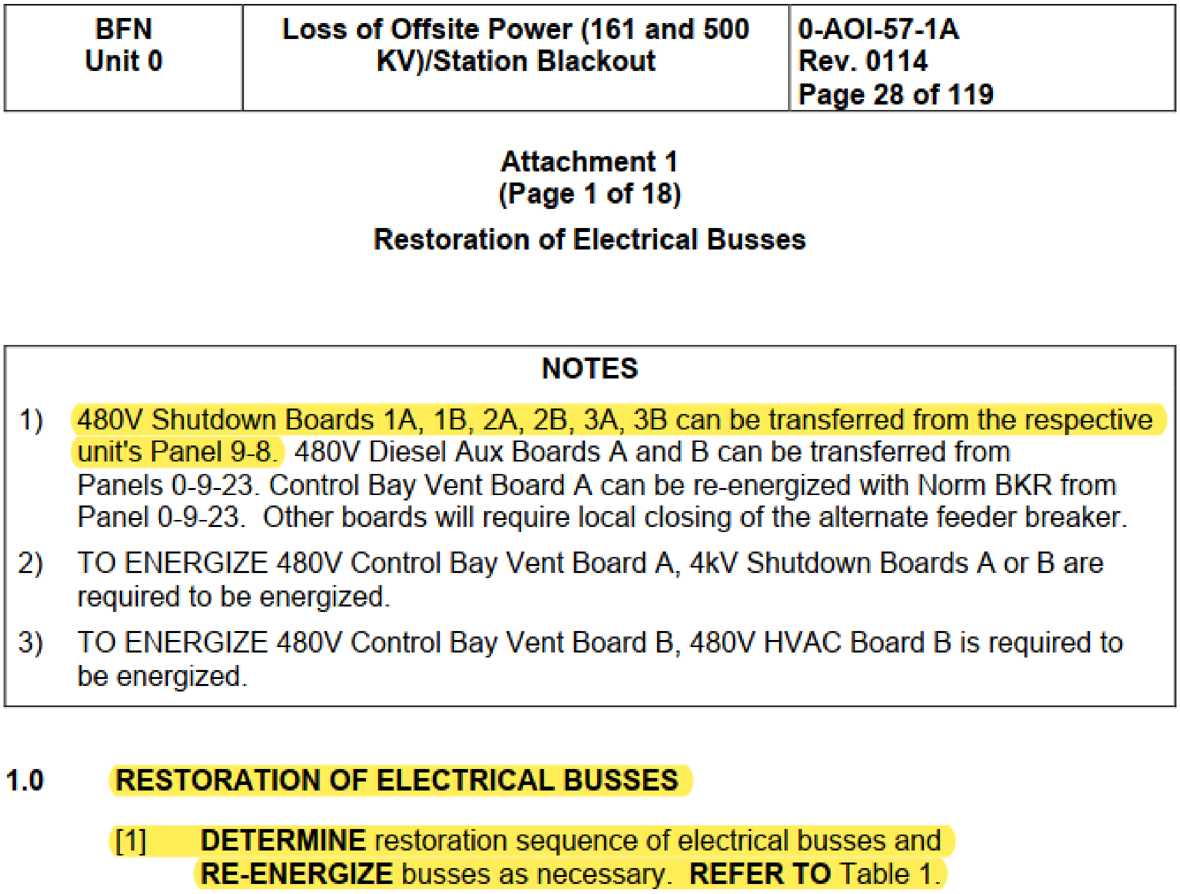
A. 1B **ONLY**

B✓ 2A **ONLY**

C. 1B **AND** 480V RMOV Board 1B

D. 2A **AND** 480V RMOV Board 2A

Excerpts from 0-AOI-57-1A:



BFN **Unit 0**	**Loss of Offsite Power (161 and 500 KV)/Station Blackout**	**0-AOI-57-1A** **Rev. 0114** **Page 28 of 119**

Attachment 1
(Page 1 of 18)

Restoration of Electrical Busses

NOTES

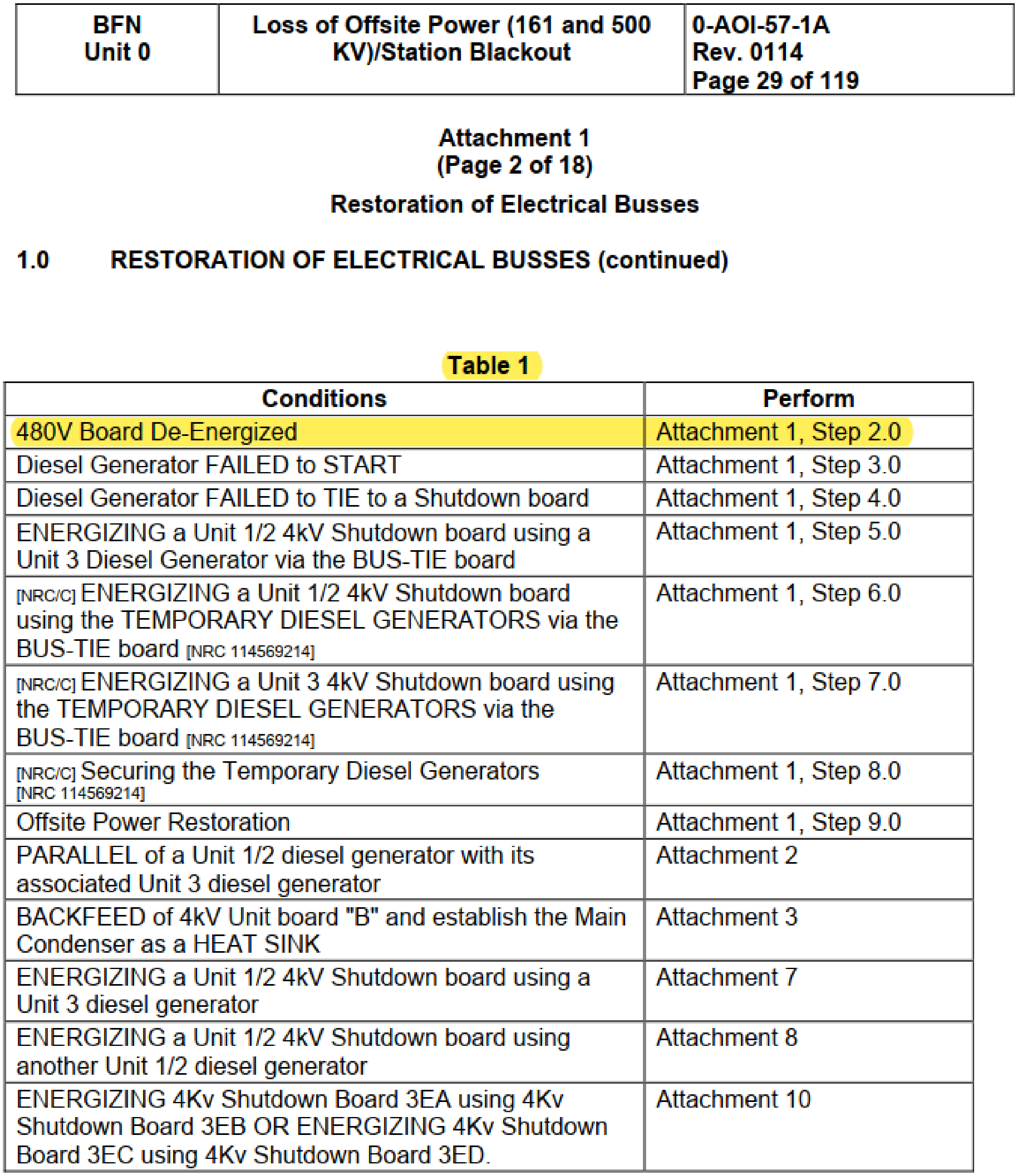
1) 480V Shutdown Boards 1A, 1B, 2A, 2B, 3A, 3B can be transferred from the respective unit's Panel 9-8. 480V Diesel Aux Boards A and B can be transferred from Panels 0-9-23. Control Bay Vent Board A can be re-energized with Norm BKR from Panel 0-9-23. Other boards will require local closing of the alternate feeder breaker.

2) TO ENERGIZE 480V Control Bay Vent Board A, 4kV Shutdown Boards A or B are required to be energized.

3) TO ENERGIZE 480V Control Bay Vent Board B, 480V HVAC Board B is required to be energized.

1.0 RESTORATION OF ELECTRICAL BUSSES

[1] **DETERMINE** restoration sequence of electrical busses and **RE-ENERGIZE** busses as necessary. **REFER TO** Table 1.



BFN Unit 0	Loss of Offsite Power (161 and 500 KV)/Station Blackout	0-AOI-57-1A Rev. 0114 Page 29 of 119

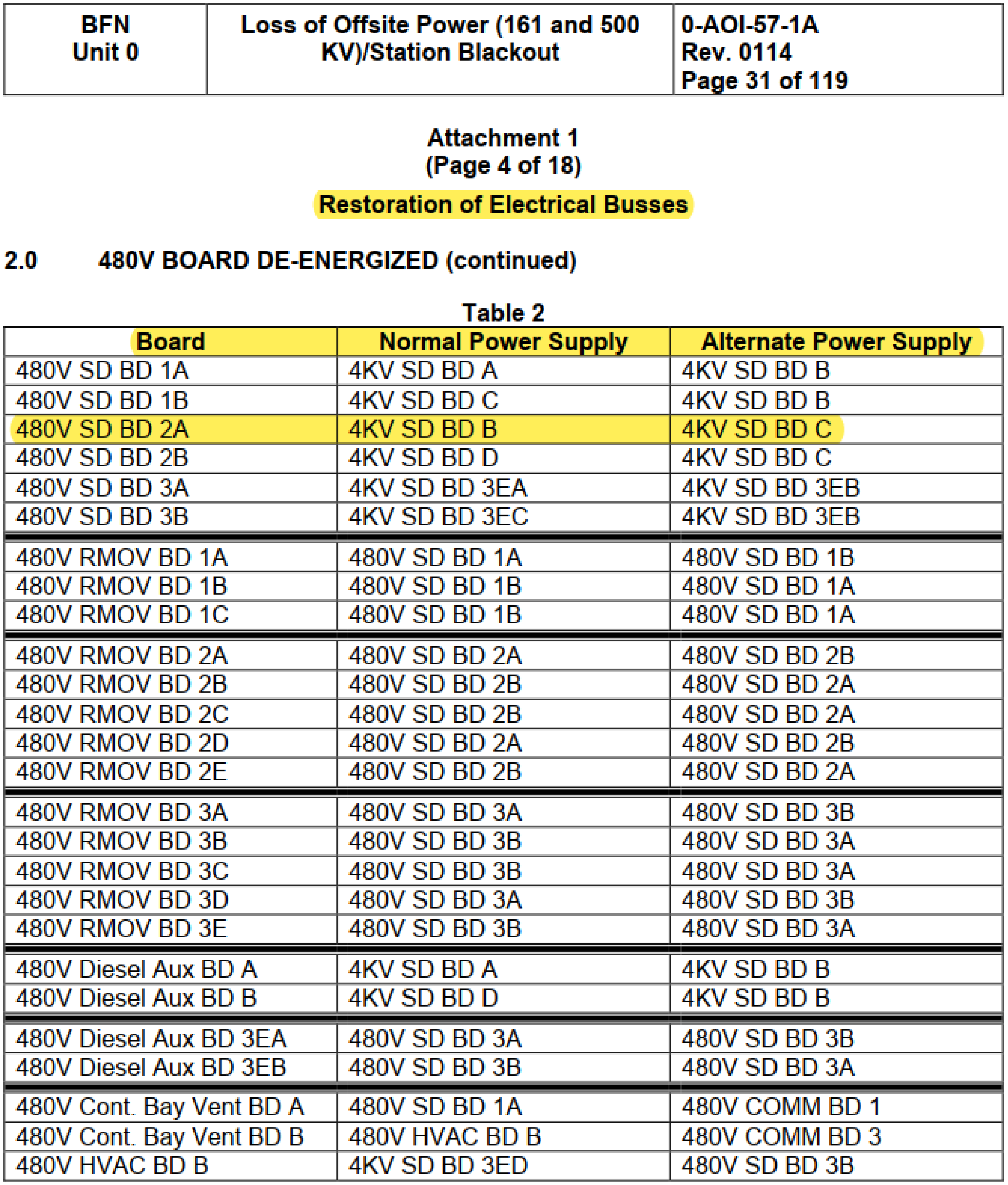
Attachment 1
(Page 2 of 18)

Restoration of Electrical Busses

1.0 RESTORATION OF ELECTRICAL BUSSES (continued)

Table 1

Conditions	Perform
480V Board De-Energized	Attachment 1, Step 2.0
Diesel Generator FAILED to START	Attachment 1, Step 3.0
Diesel Generator FAILED to TIE to a Shutdown board	Attachment 1, Step 4.0
ENERGIZING a Unit 1/2 4kV Shutdown board using a Unit 3 Diesel Generator via the BUS-TIE board	Attachment 1, Step 5.0
[NRC/C] ENERGIZING a Unit 1/2 4kV Shutdown board using the TEMPORARY DIESEL GENERATORS via the BUS-TIE board [NRC 114569214]	Attachment 1, Step 6.0
[NRC/C] ENERGIZING a Unit 3 4kV Shutdown board using the TEMPORARY DIESEL GENERATORS via the BUS-TIE board [NRC 114569214]	Attachment 1, Step 7.0
[NRC/C] Securing the Temporary Diesel Generators [NRC 114569214]	Attachment 1, Step 8.0
Offsite Power Restoration	Attachment 1, Step 9.0
PARALLEL of a Unit 1/2 diesel generator with its associated Unit 3 diesel generator	Attachment 2
BACKFEED of 4kV Unit board "B" and establish the Main Condenser as a HEAT SINK	Attachment 3
ENERGIZING a Unit 1/2 4kV Shutdown board using a Unit 3 diesel generator	Attachment 7
ENERGIZING a Unit 1/2 4kV Shutdown board using another Unit 1/2 diesel generator	Attachment 8
ENERGIZING 4Kv Shutdown Board 3EA using 4Kv Shutdown Board 3EB OR ENERGIZING 4Kv Shutdown Board 3EC using 4Kv Shutdown Board 3ED.	Attachment 10



BFN Unit 0	**Loss of Offsite Power (161 and 500 KV)/Station Blackout**	**0-AOI-57-1A Rev. 0114 Page 31 of 119**

Attachment 1
(Page 4 of 18)

Restoration of Electrical Busses

2.0 480V BOARD DE-ENERGIZED (continued)

Table 2

Board	Normal Power Supply	Alternate Power Supply
480V SD BD 1A	4KV SD BD A	4KV SD BD B
480V SD BD 1B	4KV SD BD C	4KV SD BD B
480V SD BD 2A	4KV SD BD B	4KV SD BD C
480V SD BD 2B	4KV SD BD D	4KV SD BD C
480V SD BD 3A	4KV SD BD 3EA	4KV SD BD 3EB
480V SD BD 3B	4KV SD BD 3EC	4KV SD BD 3EB
480V RMOV BD 1A	480V SD BD 1A	480V SD BD 1B
480V RMOV BD 1B	480V SD BD 1B	480V SD BD 1A
480V RMOV BD 1C	480V SD BD 1B	480V SD BD 1A
480V RMOV BD 2A	480V SD BD 2A	480V SD BD 2B
480V RMOV BD 2B	480V SD BD 2B	480V SD BD 2A
480V RMOV BD 2C	480V SD BD 2B	480V SD BD 2A
480V RMOV BD 2D	480V SD BD 2A	480V SD BD 2B
480V RMOV BD 2E	480V SD BD 2B	480V SD BD 2A
480V RMOV BD 3A	480V SD BD 3A	480V SD BD 3B
480V RMOV BD 3B	480V SD BD 3B	480V SD BD 3A
480V RMOV BD 3C	480V SD BD 3B	480V SD BD 3A
480V RMOV BD 3D	480V SD BD 3A	480V SD BD 3B
480V RMOV BD 3E	480V SD BD 3B	480V SD BD 3A
480V Diesel Aux BD A	4KV SD BD A	4KV SD BD B
480V Diesel Aux BD B	4KV SD BD D	4KV SD BD B
480V Diesel Aux BD 3EA	480V SD BD 3A	480V SD BD 3B
480V Diesel Aux BD 3EB	480V SD BD 3B	480V SD BD 3A
480V Cont. Bay Vent BD A	480V SD BD 1A	480V COMM BD 1
480V Cont. Bay Vent BD B	480V HVAC BD B	480V COMM BD 3
480V HVAC BD B	4KV SD BD 3ED	480V SD BD 3B

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-reference:

295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5
G2.2.44 (10CFR 55.41.5)
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.

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295028G2.2.44	
Importance Rating	4.2	------

Proposed Question: **# 16**

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

- Drywell Temperature is 255 ºF and rising
- Suppression Pool Water Level is 15 feet
- 480V RMOV BD 2B is de-energized due to a fault

Given the conditions above, Drywell Sprays will be initiated on RHR Loop **(1)**.

Drywell Temperature indication is monitored on Panel(s) **(2)**.

A. (1) I
(2) 9-3 **OR** 9-4

B. (1) I
(2) 9-3 **ONLY**

C. (1) II
(2) 9-3 **OR** 9-4

D. (1) II
(2) 9-3 **ONLY**

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that Panel 9-4 does display alarms related to Drywell parameters. Additionally, Panel 9-4 does have indications for Drywell Equipment Drain Sump Temperature, Recirc Pumps discharge temperature, multiple RWCU heat exchanger temperatures, RBCCW Pump suction header temperature. However, as far as panel indications in the Main Control Room, Drywell Temperature indication can only be monitored on Panel 9-3.

B **CORRECT:** *(See attached)* In accordance with 2-EOI-2, Primary Containment, entry is required at Drywell Temperature of 160 ºF thereby requiring the use of 2-EOI Appendix-17B, RHR System Operation Drywell Sprays. Given that 480V RMOV Board 2B is de-energized, both 2-FCV-74-75(74), RHR System Loop II Drywell Spray Inboard (Outboard) Valves have lost power. Where 480V RMOV Board 2A provides power to both 2-FCV-74-61(60), RHR System Loop I Drywell Spray Inboard (Outboard) Valves. Therefore, Drywell Sprays will be initiated on RHR Loop I. For second part, in accordance with 2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell, Drywell Temperature is monitored on Main Control Panel 9-3 using both analog and digital indications. Additionally, a number of alarm windows are displayed on Panel 9-3 related to high Drywell Temperature.

C INCORRECT: First part is incorrect but plausible in that 480V RMOV Board 2A provides power to both 2-FCV-74-61(60), RHR System Loop I Drywell Spray Inboard (Outboard) Valves. A number of ECCS component's power supplies are not conventional and often confused by candidates. Second part is incorrect *(See A).*

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

RO Level Justification: Test the candidate's ability to interpret Main Control Room indications as it relates to the Primary Containment parameter for 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (3) The progression of an event.

Technical Reference(s): 2-ARP-9-3B, Rev. 39 (Attach if not previously provided)
2-OI-74/ATT-3, Rev. 144
2-AOI-64-1, Rev. 27
OPL171.044, Rev. 22

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.016 Obj. 25U1 (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**

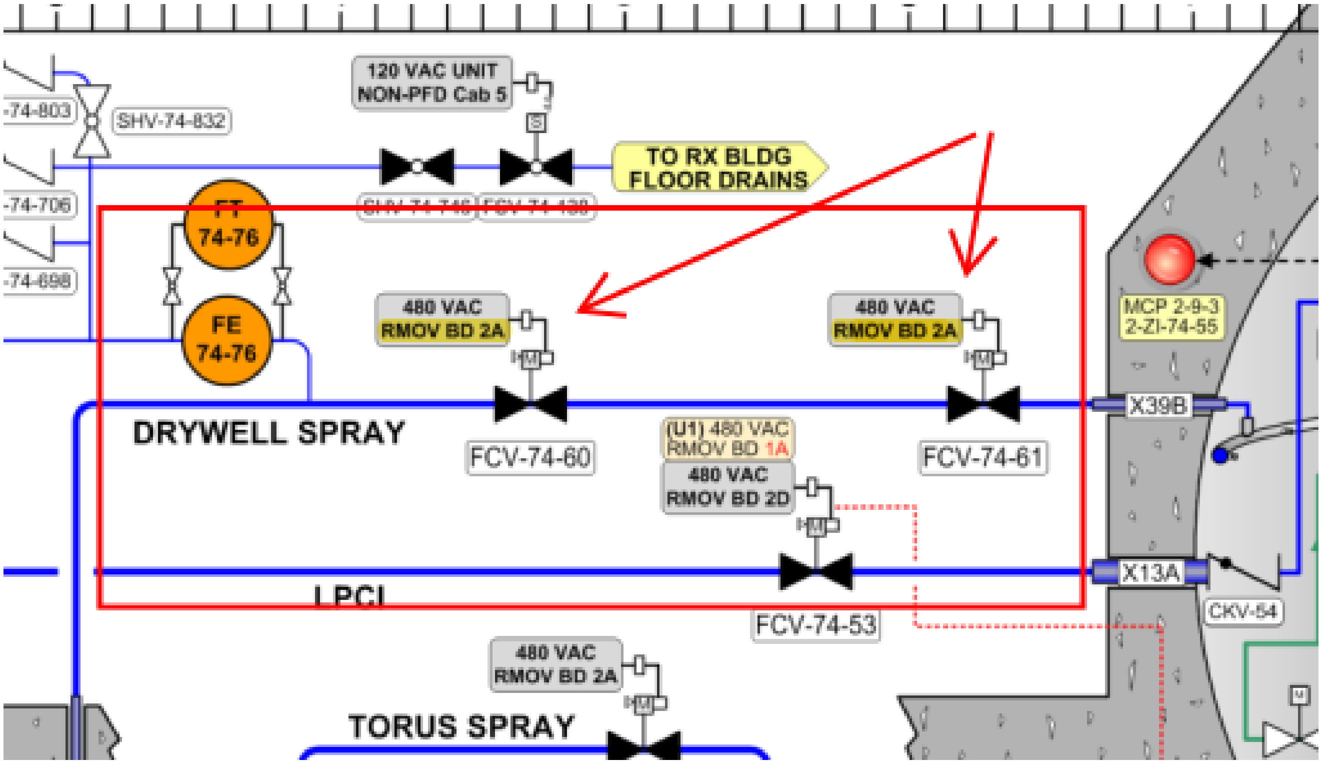
Form 4.2-1 **Written Examination Question Worksheet**

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

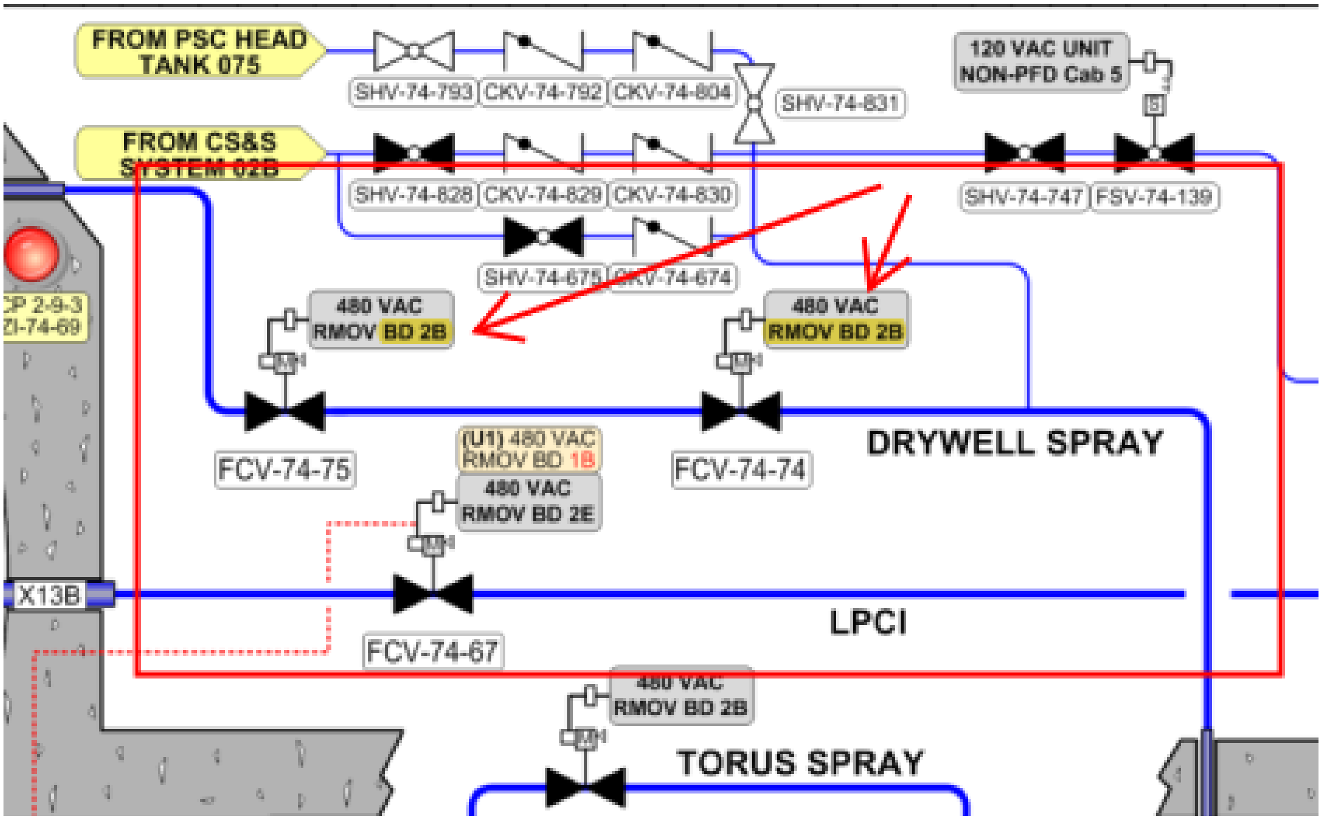
Comments:

Excerpt from OPL171.044 Lesson Plan:

- RHR Loop I Drywell Spray Valves/power supplies

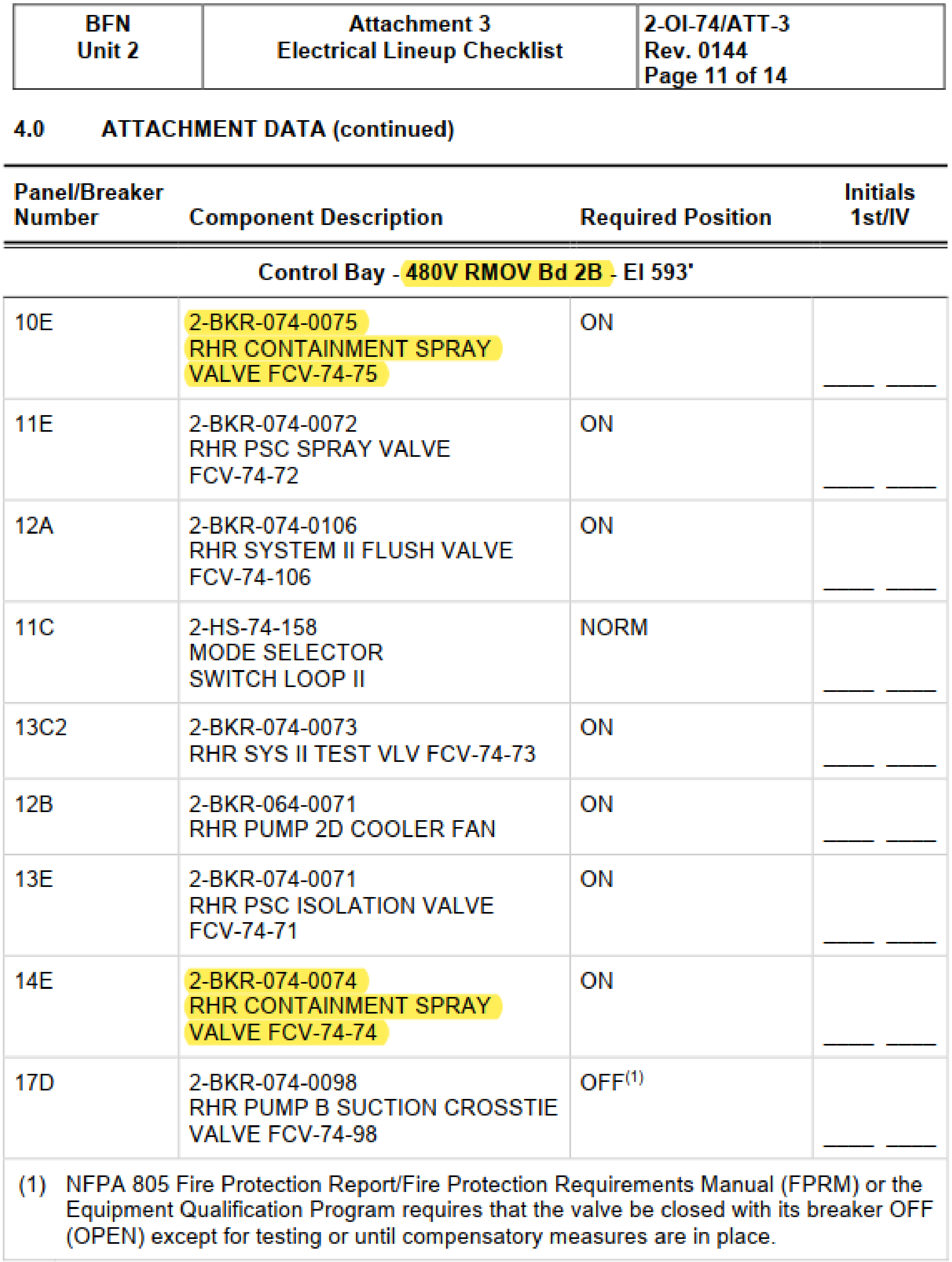


- RHR Loop II Drywell Spray Valves/power supplies



Excerpts from 2-OI-74/ATT-3:

- RHR Loop II Drywell Spray Valves/power supplies



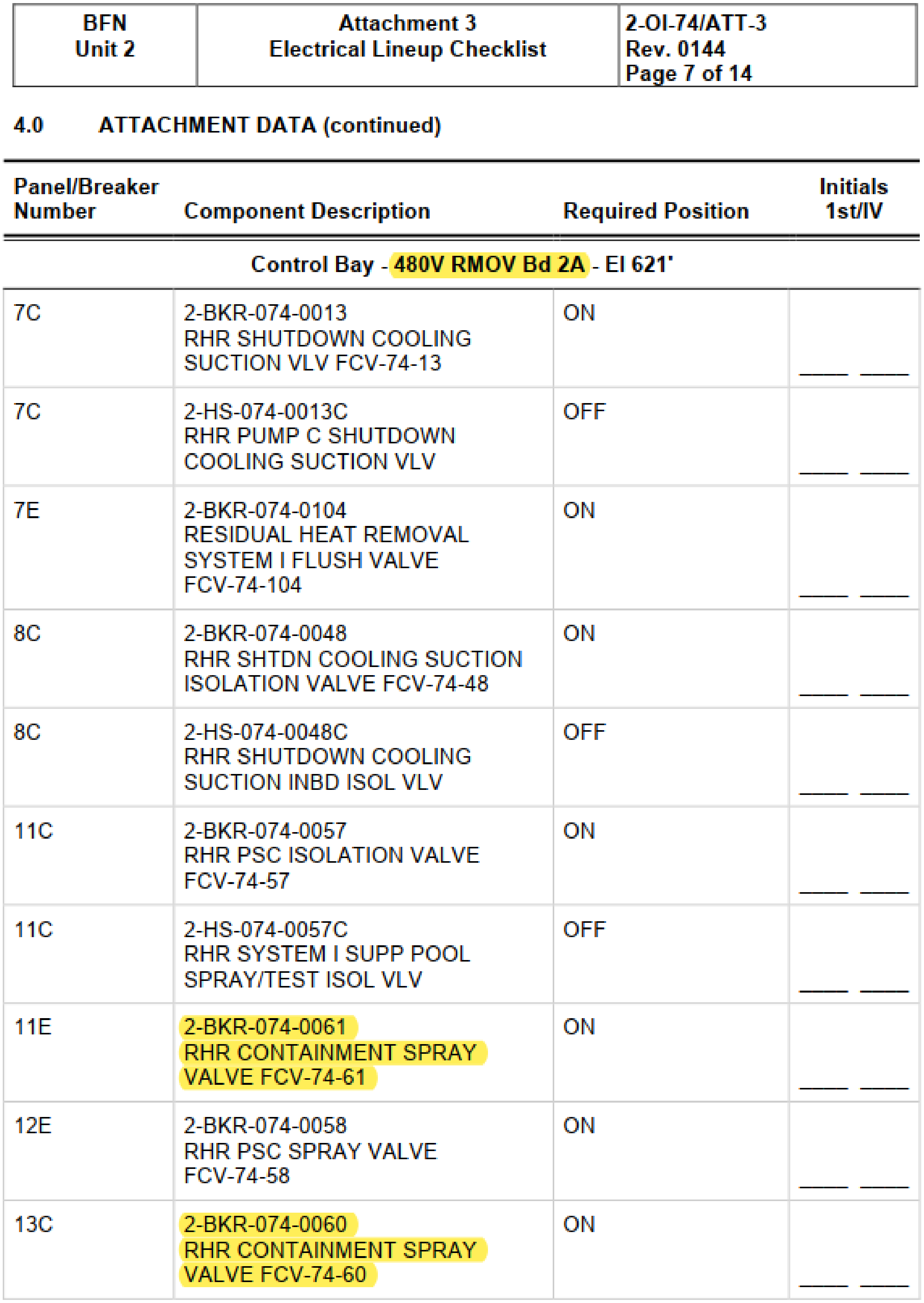
BFN **Unit 2**	**Attachment 3** **Electrical Lineup Checklist**	**2-OI-74/ATT-3** **Rev. 0144** **Page 11 of 14**

4.0 ATTACHMENT DATA (continued)

Panel/Breaker Number	**Component Description**	**Required Position**	**Initials 1st/IV**
	Control Bay - 480V RMOV Bd 2B - El 593'		
10E	2-BKR-074-0075 RHR CONTAINMENT SPRAY VALVE FCV-74-75	ON	____ ____
11E	2-BKR-074-0072 RHR PSC SPRAY VALVE FCV-74-72	ON	____ ____
12A	2-BKR-074-0106 RHR SYSTEM II FLUSH VALVE FCV-74-106	ON	____ ____
11C	2-HS-74-158 MODE SELECTOR SWITCH LOOP II	NORM	____ ____
13C2	2-BKR-074-0073 RHR SYS II TEST VLV FCV-74-73	ON	____ ____
12B	2-BKR-064-0071 RHR PUMP 2D COOLER FAN	ON	____ ____
13E	2-BKR-074-0071 RHR PSC ISOLATION VALVE FCV-74-71	ON	____ ____
14E	2-BKR-074-0074 RHR CONTAINMENT SPRAY VALVE FCV-74-74	ON	____ ____
17D	2-BKR-074-0098 RHR PUMP B SUCTION CROSSTIE VALVE FCV-74-98	OFF(1)	____ ____

(1) NFPA 805 Fire Protection Report/Fire Protection Requirements Manual (FPRM) or the Equipment Qualification Program requires that the valve be closed with its breaker OFF (OPEN) except for testing or until compensatory measures are in place.

- RHR Loop II Drywell Spray Valves/power supplies

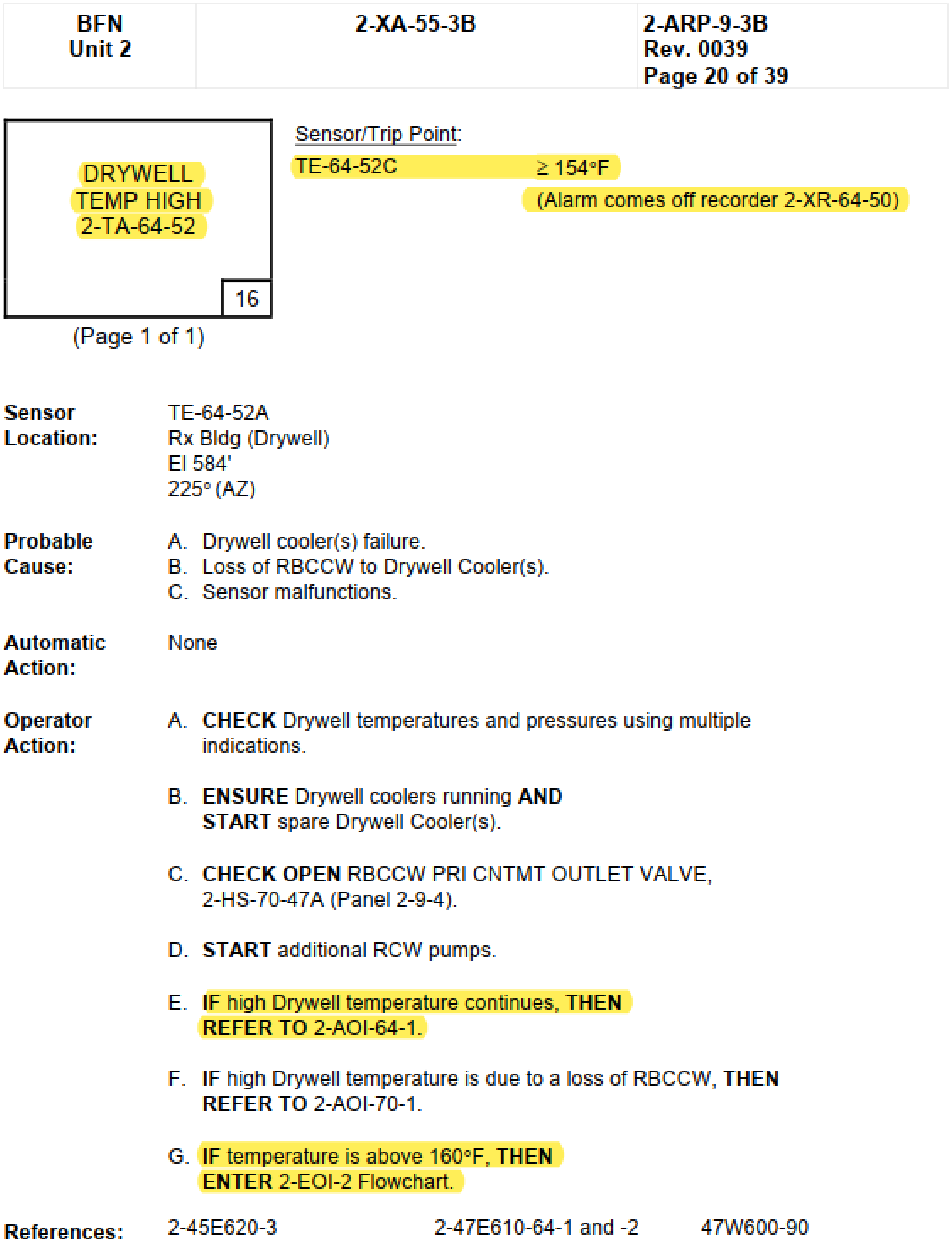


BFN Unit 2	Attachment 3 Electrical Lineup Checklist	2-OI-74/ATT-3 Rev. 0144 Page 7 of 14

4.0 ATTACHMENT DATA (continued)

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
	Control Bay - 480V RMOV Bd 2A - El 621'		
7C	2-BKR-074-0013 RHR SHUTDOWN COOLING SUCTION VLV FCV-74-13	ON	____ ____
7C	2-HS-074-0013C RHR PUMP C SHUTDOWN COOLING SUCTION VLV	OFF	____ ____
7E	2-BKR-074-0104 RESIDUAL HEAT REMOVAL SYSTEM I FLUSH VALVE FCV-74-104	ON	____ ____
8C	2-BKR-074-0048 RHR SHTDN COOLING SUCTION ISOLATION VALVE FCV-74-48	ON	____ ____
8C	2-HS-074-0048C RHR SHUTDOWN COOLING SUCTION INBD ISOL VLV	OFF	____ ____
11C	2-BKR-074-0057 RHR PSC ISOLATION VALVE FCV-74-57	ON	____ ____
11C	2-HS-074-0057C RHR SYSTEM I SUPP POOL SPRAY/TEST ISOL VLV	OFF	____ ____
11E	2-BKR-074-0061 RHR CONTAINMENT SPRAY VALVE FCV-74-61	ON	____ ____
12E	2-BKR-074-0058 RHR PSC SPRAY VALVE FCV-74-58	ON	____ ____
13C	2-BKR-074-0060 RHR CONTAINMENT SPRAY VALVE FCV-74-60	ON	____ ____

Excerpt from 2-ARP-9-3B:



BFN **Unit 2**	**2-XA-55-3B**	**2-ARP-9-3B** **Rev. 0039** **Page 20 of 39**

DRYWELL
TEMP HIGH
2-TA-64-52
16

(Page 1 of 1)

Sensor/Trip Point:

TE-64-52C ≥ 154°F

(Alarm comes off recorder 2-XR-64-50)

Sensor Location: TE-64-52A
Rx Bldg (Drywell)
El 584'
225° (AZ)

Probable Cause:
A. Drywell cooler(s) failure.
B. Loss of RBCCW to Drywell Cooler(s).
C. Sensor malfunctions.

Automatic Action: None

Operator Action:

A. **CHECK** Drywell temperatures and pressures using multiple indications.

B. **ENSURE** Drywell coolers running **AND** **START** spare Drywell Cooler(s).

C. **CHECK OPEN** RBCCW PRI CNTMT OUTLET VALVE, 2-HS-70-47A (Panel 2-9-4).

D. **START** additional RCW pumps.

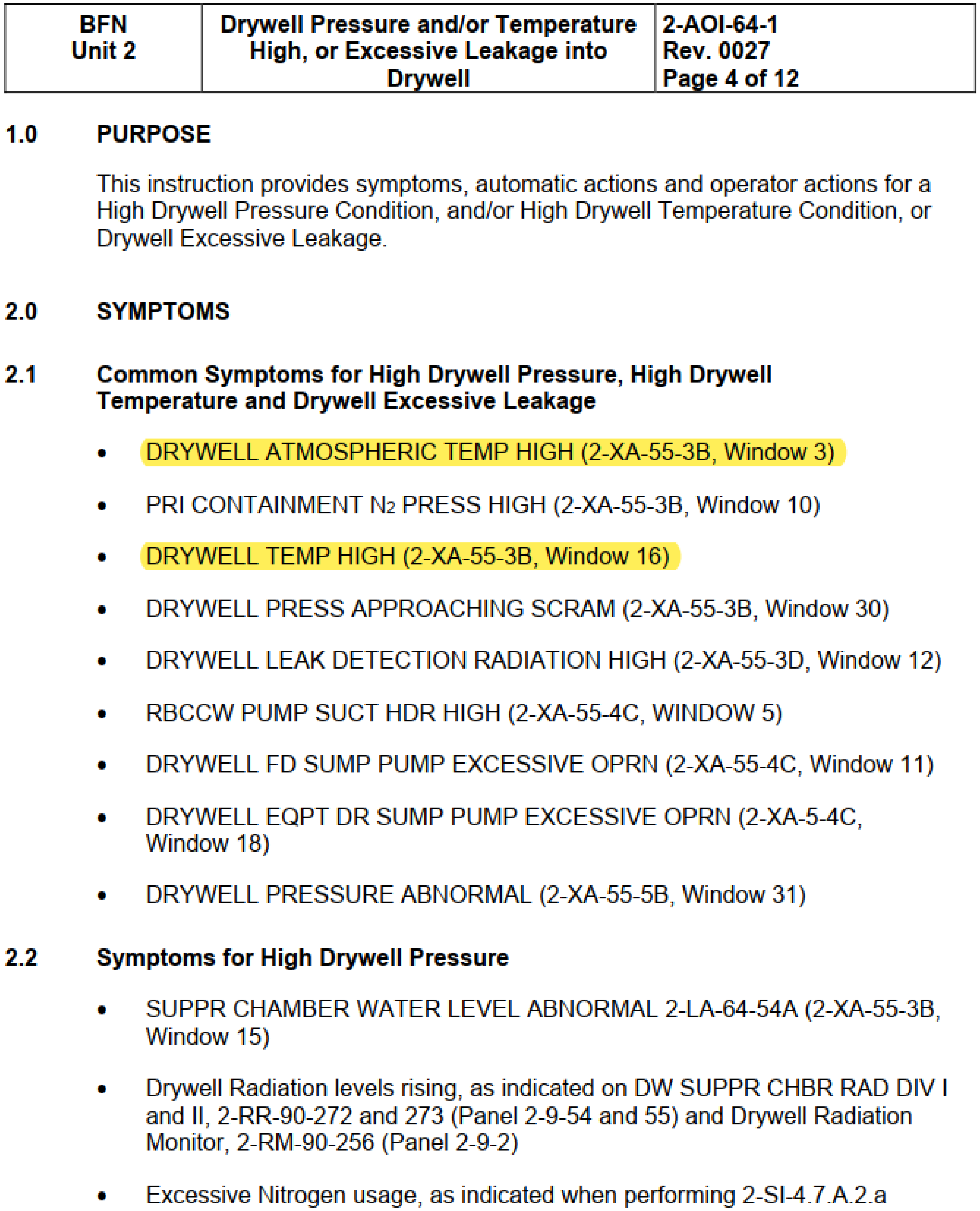
E. **IF** high Drywell temperature continues, **THEN** **REFER TO** 2-AOI-64-1.

F. **IF** high Drywell temperature is due to a loss of RBCCW, **THEN** **REFER TO** 2-AOI-70-1.

G. **IF** temperature is above 160°F, **THEN** **ENTER** 2-EOI-2 Flowchart.

References: 2-45E620-3 2-47E610-64-1 and -2 47W600-90

Excerpts from 2-AOI-64-1:



BFN Unit 2	**Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell**	**2-AOI-64-1 Rev. 0027 Page 4 of 12**

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a High Drywell Pressure Condition, and/or High Drywell Temperature Condition, or Drywell Excessive Leakage.

2.0 SYMPTOMS

2.1 Common Symptoms for High Drywell Pressure, High Drywell Temperature and Drywell Excessive Leakage

- DRYWELL ATMOSPHERIC TEMP HIGH (2-XA-55-3B, Window 3)
- PRI CONTAINMENT N_2 PRESS HIGH (2-XA-55-3B, Window 10)
- DRYWELL TEMP HIGH (2-XA-55-3B, Window 16)
- DRYWELL PRESS APPROACHING SCRAM (2-XA-55-3B, Window 30)
- DRYWELL LEAK DETECTION RADIATION HIGH (2-XA-55-3D, Window 12)
- RBCCW PUMP SUCT HDR HIGH (2-XA-55-4C, WINDOW 5)
- DRYWELL FD SUMP PUMP EXCESSIVE OPRN (2-XA-55-4C, Window 11)
- DRYWELL EQPT DR SUMP PUMP EXCESSIVE OPRN (2-XA-5-4C, Window 18)
- DRYWELL PRESSURE ABNORMAL (2-XA-55-5B, Window 31)

2.2 Symptoms for High Drywell Pressure

- SUPPR CHAMBER WATER LEVEL ABNORMAL 2-LA-64-54A (2-XA-55-3B, Window 15)
- Drywell Radiation levels rising, as indicated on DW SUPPR CHBR RAD DIV I and II, 2-RR-90-272 and 273 (Panel 2-9-54 and 55) and Drywell Radiation Monitor, 2-RM-90-256 (Panel 2-9-2)
- Excessive Nitrogen usage, as indicated when performing 2-SI-4.7.A.2.a

BFN Unit 2	**Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell**	**2-AOI-64-1 Rev. 0027 Page 5 of 12**

2.3 Symptoms for High Drywell Temperature

- DRYWELL NORM OPERATING PRESS HIGH (2-XA-55-3B, Window 19)
- Drywell temperature rising, as indicated on DRYWELL TEMPERATURE/PRESSURE, 2-XR-064-050 (Panel 2-9-3)
- Drywell pressure rising, as indicated on DRYWELL TEMPERATURE/PRESSURE, 2-XR-064-050 (Panel 2-9-3)

2.4 Symptoms for Drywell Excessive Leakage

- DRYWELL NORM OPERATING PRESS HIGH (2-XA-55-3B, Window 19)
- DRYWELL FD SUMP LEVEL ABN (2-XA-55-4C, Window 2)
- DRYWELL EQPT DR SUMP LEVEL ABN (2-XA-55-4C, Window 9)
- RBCCW SURGE TANK LEVEL LOW (2-XA-55-4C, Window 13)
- DRYWELL EQPT DR SUMP TEMP HIGH (2-XA-55-4C, Window 16)
- REACTOR WATER LEVEL ABNORMAL (2-XA-55-5A, Window 8)
- RECIRC PUMP A NO. 2 SEAL LEAKAGE HIGH 2-FA-68-55 (2-XA-55-4A, Window 18)
- RECIRC PUMP A NO. 1 SEAL LEAKAGE ABN 2-PA-68-63 (2-XA-55-4A, Window 25)
- RECIRC PUMP B NO. 2 SEAL LEAKAGE HIGH 2-FA-68-68 (2-XA-55-4B, Window 18)
- RECIRC PUMP B NO. 1 SEAL LEAKAGE ABN 2-PA-68-75 (2-RA-55-4B, Window 25)

3.0 AUTOMATIC ACTIONS

None

Examination Outline Cross-reference:	Level	RO	SRO
600000 (APE 24) Plant Fire On Site / 8	Tier #	1	------
AA2.05 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to Plant Fire on Site: • Ventilation alignment necessary to secure affected area	K/A #	600000AA2.05	
	Importance Rating	3.2	------

Proposed Question: **# 17**

Unit 1 was operating at 100% RTP when events occurred resulting in the following conditions:

- Drywell Pressure reached 2.5 psig
- A fire has been reported in the Unit 1 Reactor Building
- 0-AOI-26-1, Fire Response, has been entered

In accordance with the AOI, the Assistant Unit Operators (AUOs) will **INITIALLY** report to **(1)** and smoke is currently being removed by **(2)**.

A. (1) their assigned Control Room
(2) Reactor Zone Ventilation

B. (1) their assigned Control Room
(2) Standby Gas Treatment Fans

C. (1) the Incident Commander
(2) Reactor Zone Ventilation

D. (1) the Incident Commander
(2) Standby Gas Treatment Fans

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). The second part is incorrect but plausible in that in accordance with both 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. The candidate may not correlate that high Drywell Pressure will automatically isolate Reactor Zone Ventilation.

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. For second part, in accordance with 0-AOI-26-1, Fire Response, (continuous use procedure) states to place ventilation systems in service to remove smoke from affected area with Reactor Building ventilation being placed in service first, next Refuel Zone and then followed by Standby Gas Treatment only as necessary. However, given that Drywell Pressure is greater than 2.45 psig, a Group 6 PCIS Isolation has occurred. That means that both Reactor and Refuel Zone Supply and Exhaust Fans automatically isolated. Additionally, the Standby Gas Treatment System (SGT) will automatically initiate when Drywell Pressure rises to 2.45 psig, therefore smoke is currently being removed by SGT.

C INCORRECT: 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. 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 determine ventilation alignment as it relates to a plant fire on site. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 0-OI-65, Rev. 55 (Attach if not previously provided)
0-AOI-26-1, Rev. 22
1-AOI-64-2D, Rev. 21

Proposed references to be provided to applicants during examination: **NONE**

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

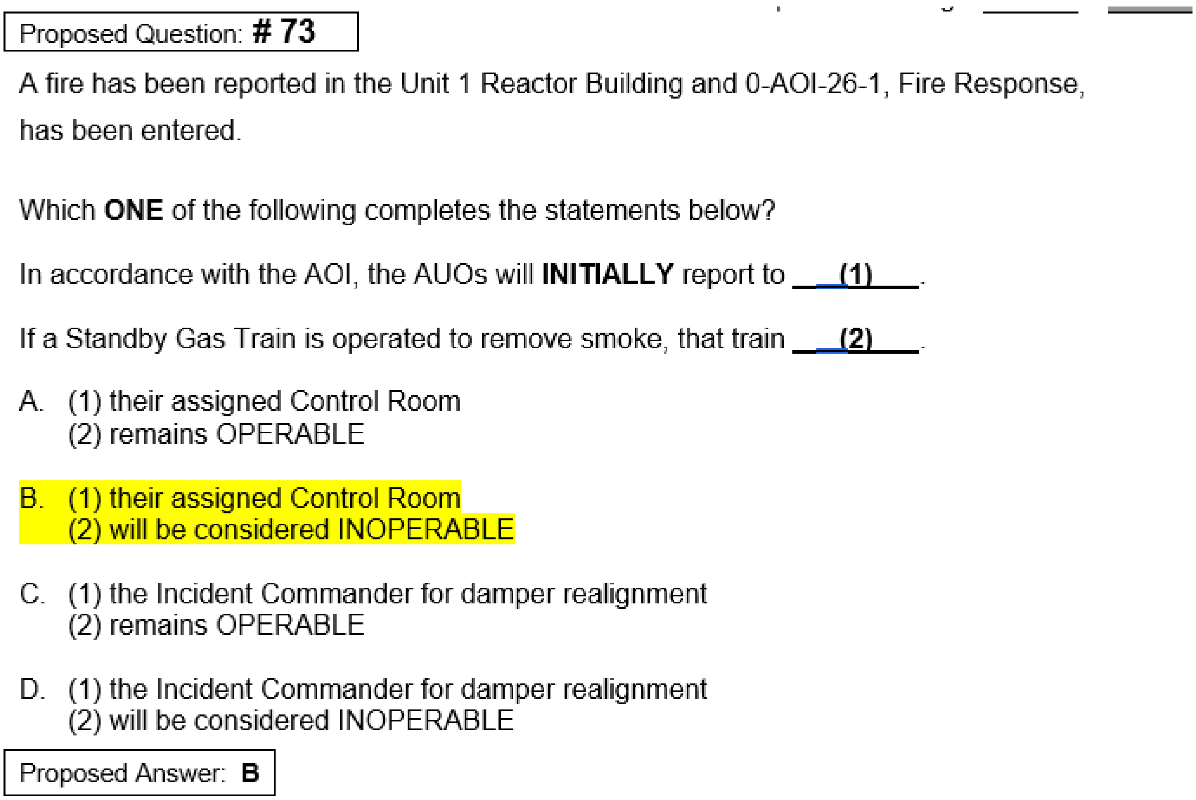
Question Source:	Bank #		
	Modified Bank #	BFN 21-04 #73	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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: **# 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

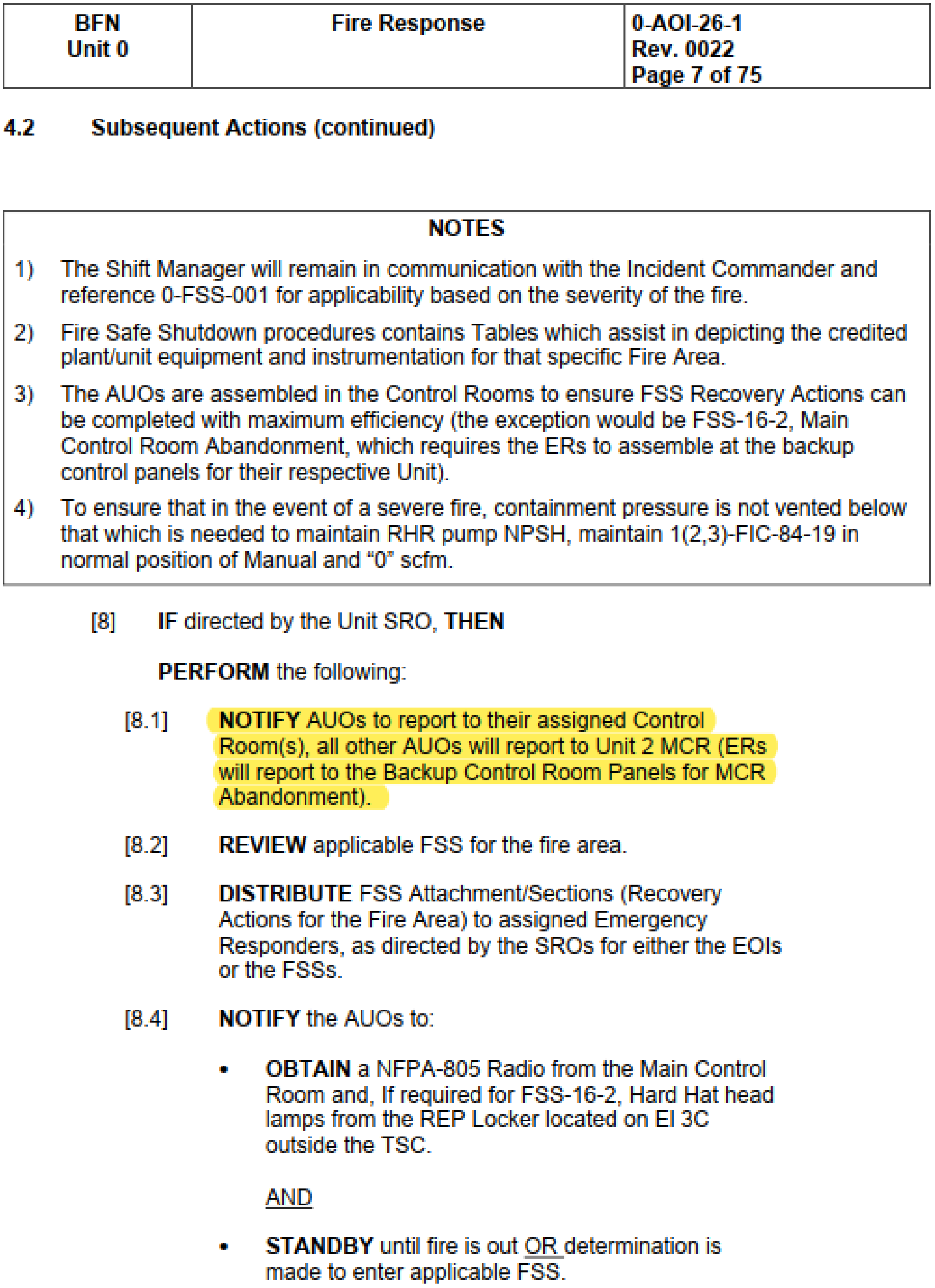
B. (1) their assigned Control Room
(2) will be considered INOPERABLE

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**

Excerpts from 0-AOI-26-1:



BFN **Unit 0**	**Fire Response**	**0-AOI-26-1** **Rev. 0022** **Page 7 of 75**

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.

[8] **IF** directed by the Unit SRO, **THEN**

PERFORM the following:

[8.1] **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).

[8.2] **REVIEW** applicable FSS for the fire area.

[8.3] **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.

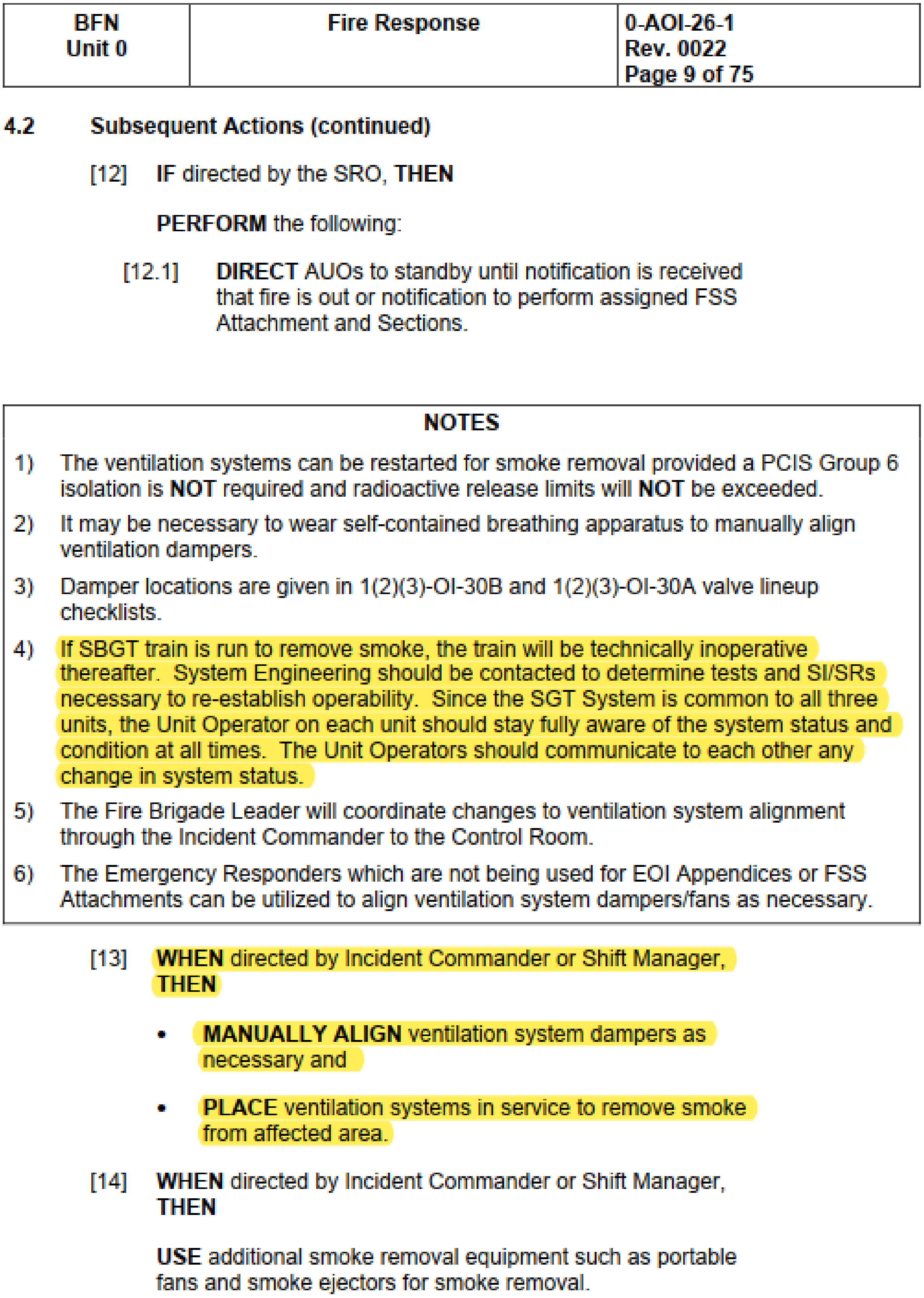
[8.4] **NOTIFY** the AUOs to:

- **OBTAIN** a NFPA-805 Radio from the Main Control Room and, If required for FSS-16-2, Hard Hat head lamps from the REP Locker located on El 3C outside the TSC.

  AND

- **STANDBY** until fire is out OR determination is made to enter applicable FSS.

Supports Distractors C(1), D(1)



BFN **Unit 0**	**Fire Response**	**0-AOI-26-1** **Rev. 0022** **Page 9 of 75**

4.2 Subsequent Actions (continued)

[12] IF directed by the SRO, **THEN**

PERFORM the following:

[12.1] **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.

[13] **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.

[14] **WHEN** directed by Incident Commander or Shift Manager, **THEN**

USE additional smoke removal equipment such as portable fans and smoke ejectors for smoke removal.

BFN **Unit 0**	**Fire Response**	**0-AOI-26-1** **Rev. 0022** **Page 10 of 75**

4.2 Subsequent Actions (continued)

[15] IF Unit 1 Reactor Bldg needs smoke removal, **THEN**

PERFORM the following:

[15.1] **START** Unit 1 Reactor Zone Supply and Exhaust Fans as necessary. **REFER TO** 1-OI-30B.

[15.2] **START** Unit 1 Refuel Zone Supply and Exhaust Fans as necessary. **REFER TO** 1-OI-30A.

[15.3] **START** SGT as necessary. **REFER TO** 0-OI-65.

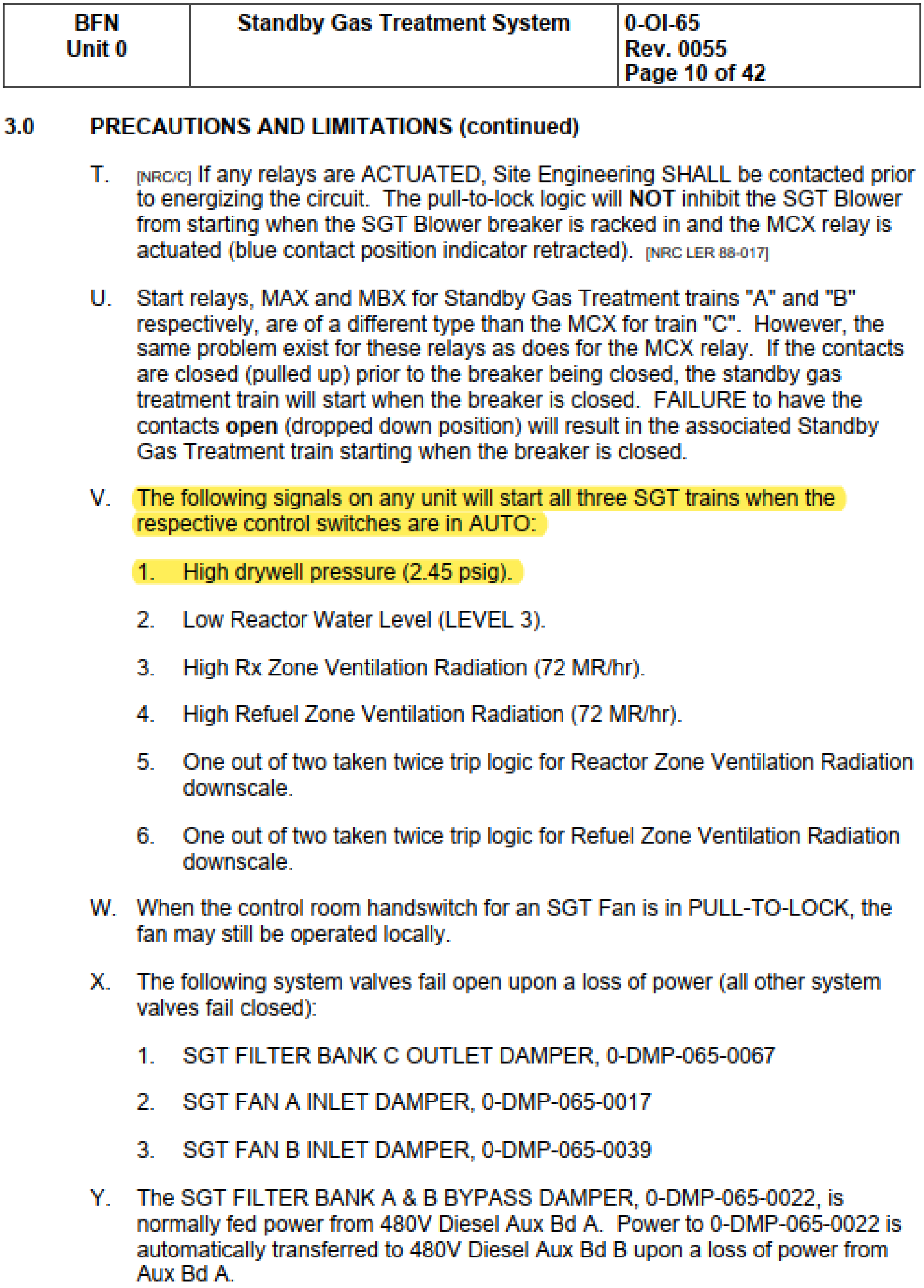
[15.4] **WHEN** smoke removal has been completed, **THEN**

PERFORM the following as applicable:

- **RESTORE** Unit 1 reactor zone ventilation to normal. **REFER TO** 1-OI-30B.
- **RESTORE** Unit 1 refuel zone ventilation to normal. **REFER TO** 1-OI-30A.
- **IF** SGT System is operating for smoke removal, **THEN**

  RESTORE it to standby readiness. **REFER TO** 0-OI-65.

Excerpts from 0-OI-65:



BFN **Unit 0**	**Standby Gas Treatment System**	**0-OI-65** **Rev. 0055** **Page 10 of 42**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

T. [NRC/C] If any relays are ACTUATED, Site Engineering SHALL be contacted prior to energizing the circuit. The pull-to-lock logic will **NOT** inhibit the SGT Blower from starting when the SGT Blower breaker is racked in and the MCX relay is actuated (blue contact position indicator retracted). [NRC LER 88-017]

U. Start relays, MAX and MBX for Standby Gas Treatment trains "A" and "B" respectively, are of a different type than the MCX for train "C". However, the same problem exist for these relays as does for the MCX relay. If the contacts are closed (pulled up) prior to the breaker being closed, the standby gas treatment train will start when the breaker is closed. FAILURE to have the contacts **open** (dropped down position) will result in the associated Standby Gas Treatment train starting when the breaker is closed.

V. The following signals on any unit will start all three SGT trains when the respective control switches are in AUTO:

1. High drywell pressure (2.45 psig).
2. Low Reactor Water Level (LEVEL 3).
3. High Rx Zone Ventilation Radiation (72 MR/hr).
4. High Refuel Zone Ventilation Radiation (72 MR/hr).
5. One out of two taken twice trip logic for Reactor Zone Ventilation Radiation downscale.
6. One out of two taken twice trip logic for Refuel Zone Ventilation Radiation downscale.

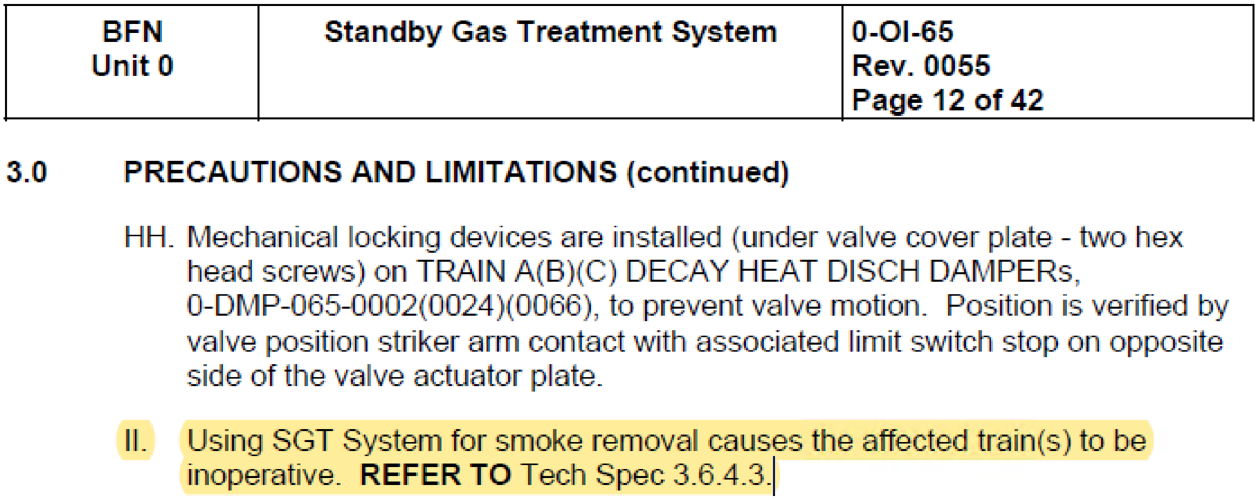
W. When the control room handswitch for an SGT Fan is in PULL-TO-LOCK, the fan may still be operated locally.

X. The following system valves fail open upon a loss of power (all other system valves fail closed):

1. SGT FILTER BANK C OUTLET DAMPER, 0-DMP-065-0067
2. SGT FAN A INLET DAMPER, 0-DMP-065-0017
3. SGT FAN B INLET DAMPER, 0-DMP-065-0039

Y. The SGT FILTER BANK A & B BYPASS DAMPER, 0-DMP-065-0022, is normally fed power from 480V Diesel Aux Bd A. Power to 0-DMP-065-0022 is automatically transferred to 480V Diesel Aux Bd B upon a loss of power from Aux Bd A.

Supports Distractors A(2), C(2)



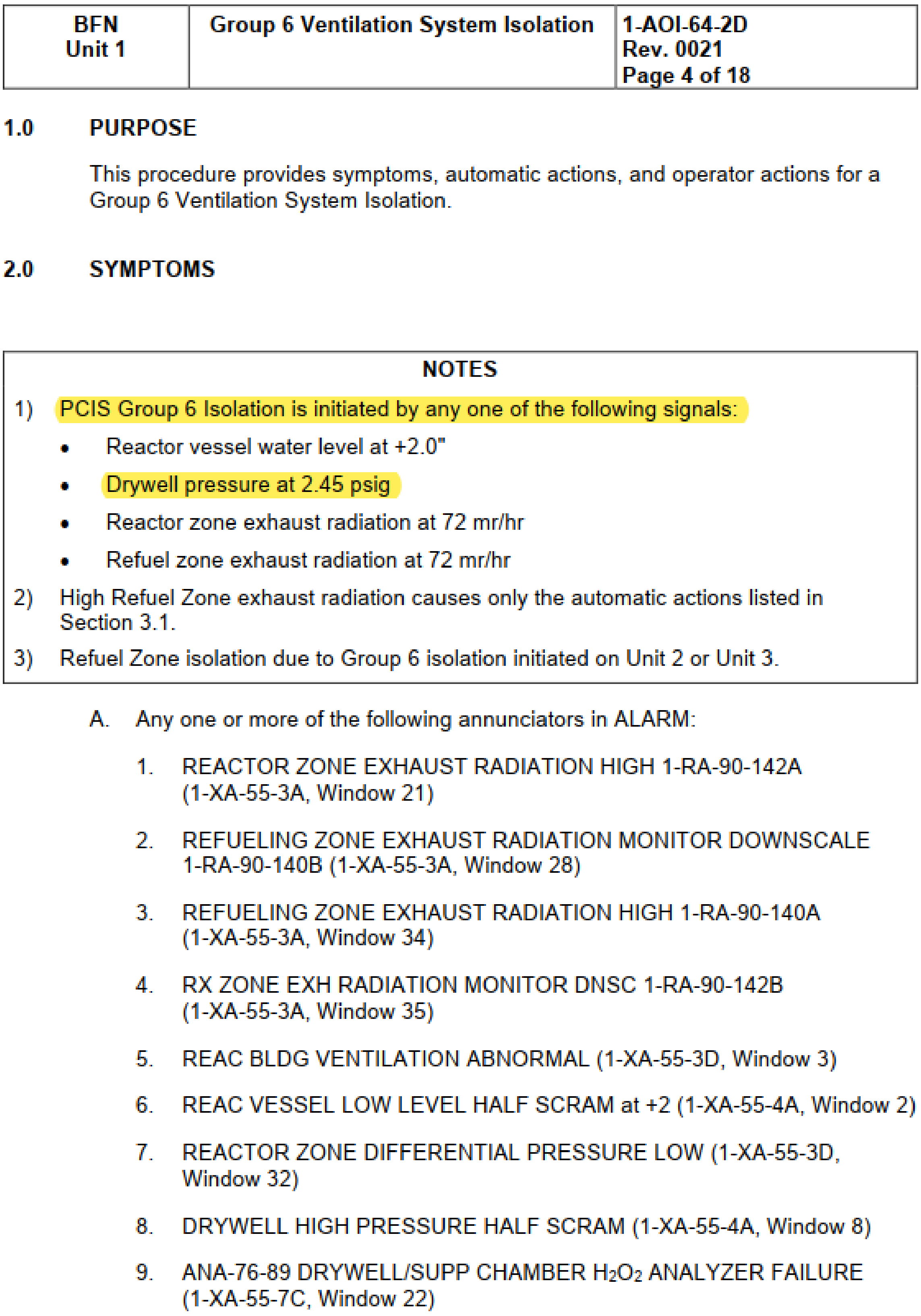
BFN **Unit 0**	**Standby Gas Treatment System**	**0-OI-65** **Rev. 0055** **Page 12 of 42**

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.

Excerpts from 1-AOI-64-2D:



BFN **Unit 1**	**Group 6 Ventilation System Isolation**	**1-AOI-64-2D** **Rev. 0021** **Page 4 of 18**

1.0 PURPOSE

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

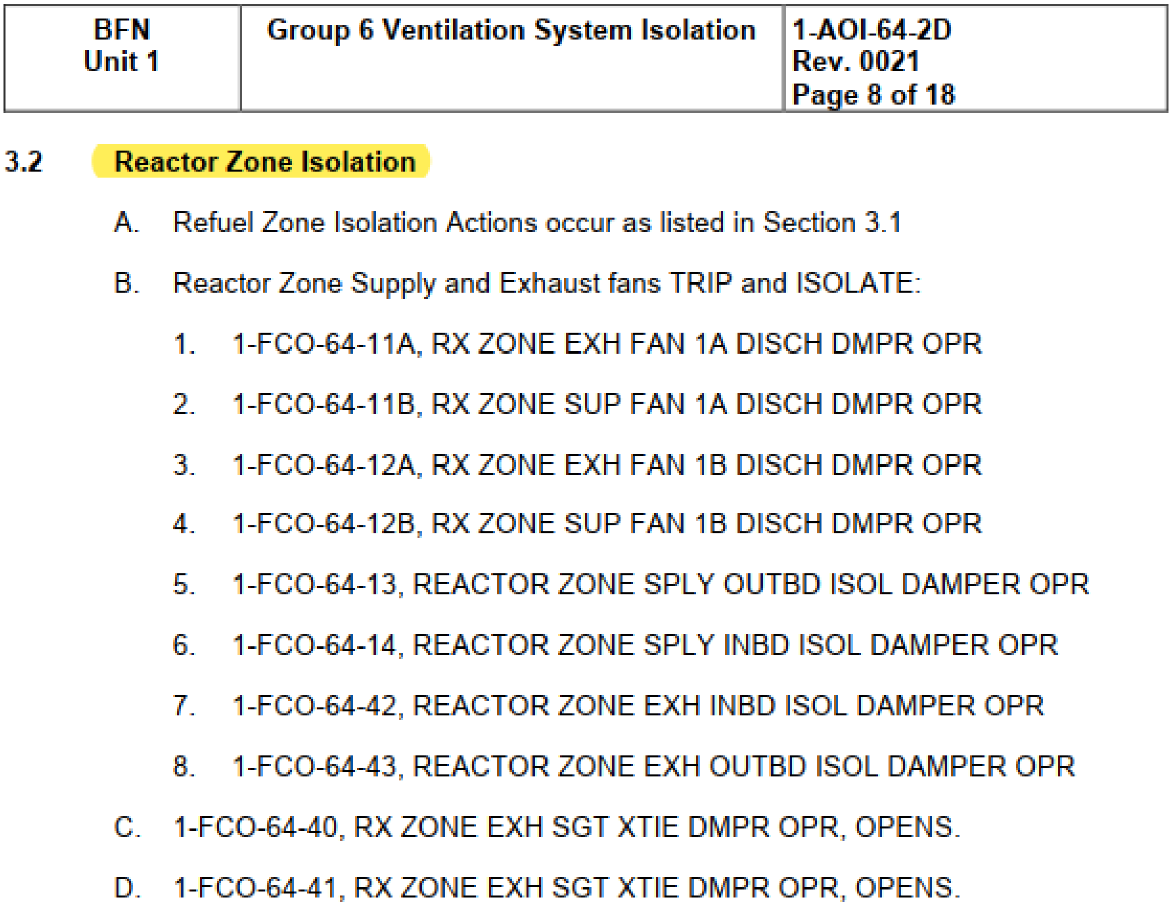
2.0 SYMPTOMS

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. 0021 Page 8 of 18**

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. 1-FCO-64-11A, RX ZONE EXH FAN 1A DISCH DMPR OPR
2. 1-FCO-64-11B, RX ZONE SUP FAN 1A DISCH DMPR OPR
3. 1-FCO-64-12A, RX ZONE EXH FAN 1B DISCH DMPR OPR
4. 1-FCO-64-12B, RX ZONE SUP FAN 1B DISCH DMPR OPR
5. 1-FCO-64-13, REACTOR ZONE SPLY OUTBD ISOL DAMPER OPR
6. 1-FCO-64-14, REACTOR ZONE SPLY INBD ISOL DAMPER OPR
7. 1-FCO-64-42, REACTOR ZONE EXH INBD ISOL DAMPER OPR
8. 1-FCO-64-43, REACTOR ZONE EXH OUTBD ISOL DAMPER OPR

C. 1-FCO-64-40, RX ZONE EXH SGT XTIE DMPR OPR, OPENS.

D. 1-FCO-64-41, RX ZONE EXH SGT XTIE DMPR OPR, OPENS.

Examination Outline Cross-reference:

261000 (SF9 SGTS) Standby Gas Treatment

K4.02 (10CFR 55.41.7)

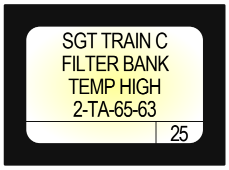
Knowledge of Standby Gas Treatment System design features and/or interlock that provide for the following:

- Charcoal bed decay heat removal

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	261000K4.02	
Importance Rating	3.0	------

Proposed Question: **# 18**

An event has occurred on Unit 2, requiring the use of Standby Gas Treatment (SGT) System, with the following conditions:



- SGT TRAIN C FILTER BANK TEMP HIGH (2-9-3B, Window 25) alarms
- The Outside AUO reports that SGT ‘C’ charcoal filter temperature is 150 ºF

Given the conditions above, the SGT decay heat removal mode **(1)** initiated and the carbon bed filters **(2)** designed to remove iodine.

A. (1) is automatically
(2) are

B. (1) is automatically
(2) are NOT

C. (1) must be manually
(2) are

D. (1) must be manually
(2) are NOT

Proposed Answer: **C**

Explanation (Optional):	A	INCORRECT: First part is incorrect but plausible in that Standby Gas Trains automatically start when required by plant conditions. However, in accordance with 0-OI-65, Standby Gas Treatment System when filter bank temperature reaches 150 ºF, decay heat removal must be manually initiated. 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 Standby Gas Treatment System contains numerous filters that do not remove iodine, but important to proper system operation such as the roughing filter, HEPA pre-filter, and HEPA after-filter.

C **CORRECT**: *(See attached)* In accordance with SGT TRAIN C FILTER BANK TEMP HIGH (2-9-3B, Window 25), the Operator is to initiate SGT in decay heat removal, Section 8.0, which has steps that are all manual. For second part, the charcoal beds are designed to remove 99.9% of elemental iodine with conditions of 70% relative humidity at 190%.

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 Standby Gas Treatment System design features as it relates to charcoal bed decay heat removal. 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.

Technical Reference(s): 0-OI-65, Rev. 55 (Attach if not previously provided)

OPL171.018, Rev. 12U1

Proposed references to be provided to applicants during examination: **SGT TRAIN C FILTER BANK TEMP HIGH (2-9-3B, Window 25)**

Learning Objective: OPL171.018, Obj 6e (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

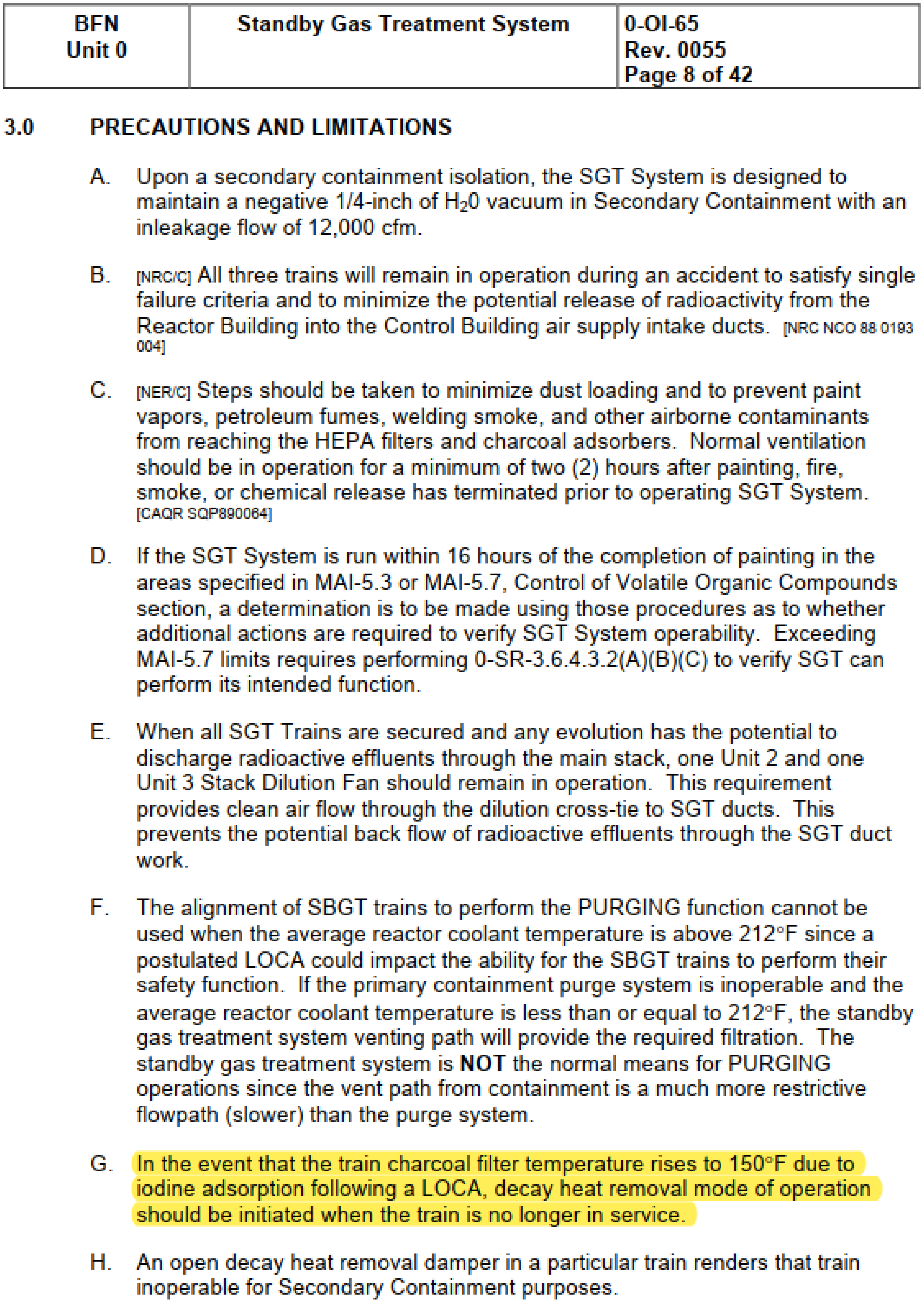
Comprehension or Analysis **X**

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

55.43

Comments:

Excerpt from 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.

Excerpt from OPL171.018:

OPL171.018, Standby Gas Treatment (SGT) System, Rev.12U1

   c) Moisture drains by gravity to SGT sump and is then pumped to Radwaste

LOR/NLOR Objective 4a

NLO Objective 2

4. Relative Humidity Heater
   a) The relative humidity heater reduces relative humidity to <70%.
   b) 40kW heaters for relative humidity control. SGT A and B powered from A and B 480V DG Aux Boards respectively. SGT C from the 480V SGT Board.

ILT Objective 6b

LOR/NLOR Objective 4b

NLO Objective 2

   c) 15kW charcoal bed heaters formerly maintained a 125°F charcoal bed temperature when SGT was out of service. Heater control switches were spring-return-to-neutral and required resetting after SGT operation. Due to the Technical Specification requirement of 10 hours monthly operation with the relative humidity heaters in service, the charcoal bed heaters are no longer needed.
   d) The relative humidity heater is energized automatically on startup by the fan breaker closure and is de-energized on shutdown by fan breaker opening.
   e) The heater will also trip if ambient temperature reaches 180°F.
   f) If 480V Load Shed logic is initiated, the Train A and B relative humidity heaters will automatically trip. They will restart after 40 seconds. Train C is not affected by the 480V load shed logic.
   g) Relative humidity heater control switches (12, 34, 60) in ON or OFF cause annunciation.
5. Roughing Filter (Prefilter)

   Used to remove large particles (dust, dirt, lint) and to protect HEPA filter.

ILT Objective 6c

LOR/NLOR Objective 4c

NLO Objective 2

6. HEPA filter

   Removes 99.9% of 0.3 micron particles

ILT Objective 6d

LOR/NLOR Objective 4d

NLO Objective 2

7. Charcoal Bed (Adsorber Type)
   a) Designed to remove at least 95% of iodine in the form of methyl iodine (CH_3I) and 99.9% of elemental iodine upon entering conditions of 70% relative humidity at 190°F.
   b) Made up of individual rectangular canisters of activated charcoal.
   c) Vertical air flow

ILT Objective 6e

LOR/NLOR Objective 4e

NLO Objective 2

8. HEPA afterfilter
   a) Identical to first HEPA filter

ILT Objective 6f

LOR/NLOR Objective 4f

Examination Outline Cross-reference:

215001 (SF7 TIP) Traversing In-Core Probe

A1.01 (10CFR 55.41.7)

Ability to predict and/or monitor changes in parameters associated with operation of the Traversing In-Core Probe, including

- Area radiation levels

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	215001A1.01	
Importance Rating	3.1	------

Proposed Question: **# 19**

Unit 2 is operating at 100% RTP with the following conditions:

- Traversing Incore Probe (TIP) operations are in progress
- Area Radiation Monitor 2-RI-90-22A, TIP ROOM EL 565 REACTOR BLDG alarms

Given the conditions above, a PCIS Group 8 Isolation ___**(1)**___ occur and EOI entry ___**(2)**___ required.

A. (1) will
(2) is NOT

B. (1) will
(2) is

C. (1) will NOT
(2) is NOT

D. (1) will NOT
(2) is

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First 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. Second part is incorrect but plausible in that an alarm on several of the radiation monitors located throughout the Reactor and Turbine Buildings that alarm on Panel 2-9-11 does not require EOI entry. Therefore, it is plausible to assume that an alarm on this particular radiation monitor would not be an EOI entry condition.

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)* A PCIS Group 8 Isolation is caused by (+) 2 inches Reactor Water Level and (+) 2.45 psig Drywell Pressure, but not by 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. For second part, in accordance with 2-EOI-3, Secondary Containment Control, Table SC-2, an alarm on 2-RI-90-22A is an EOI-3 entry condition.

RO Level Justification: Tests the candidate’s ability to predict and monitor changes in parameters associated with the operation of Transversing In-Core Probe (TIP) and Area Radiation Monitors as it relates to area radiation levels. This question is rated as memory due to strictly recalling facts related to the TIP Area Radiation Monitor concerning PCIS isolations and EOI Entry Conditions.

Technical Reference(s): 2-EOI-3, Rev. 18 (Attach if not previously provided)
2-AOI-64-2E, Rev. 17
2-ARP-9-3A, Rev. 57

Proposed references to be provided to applicants during examination: **NONE**

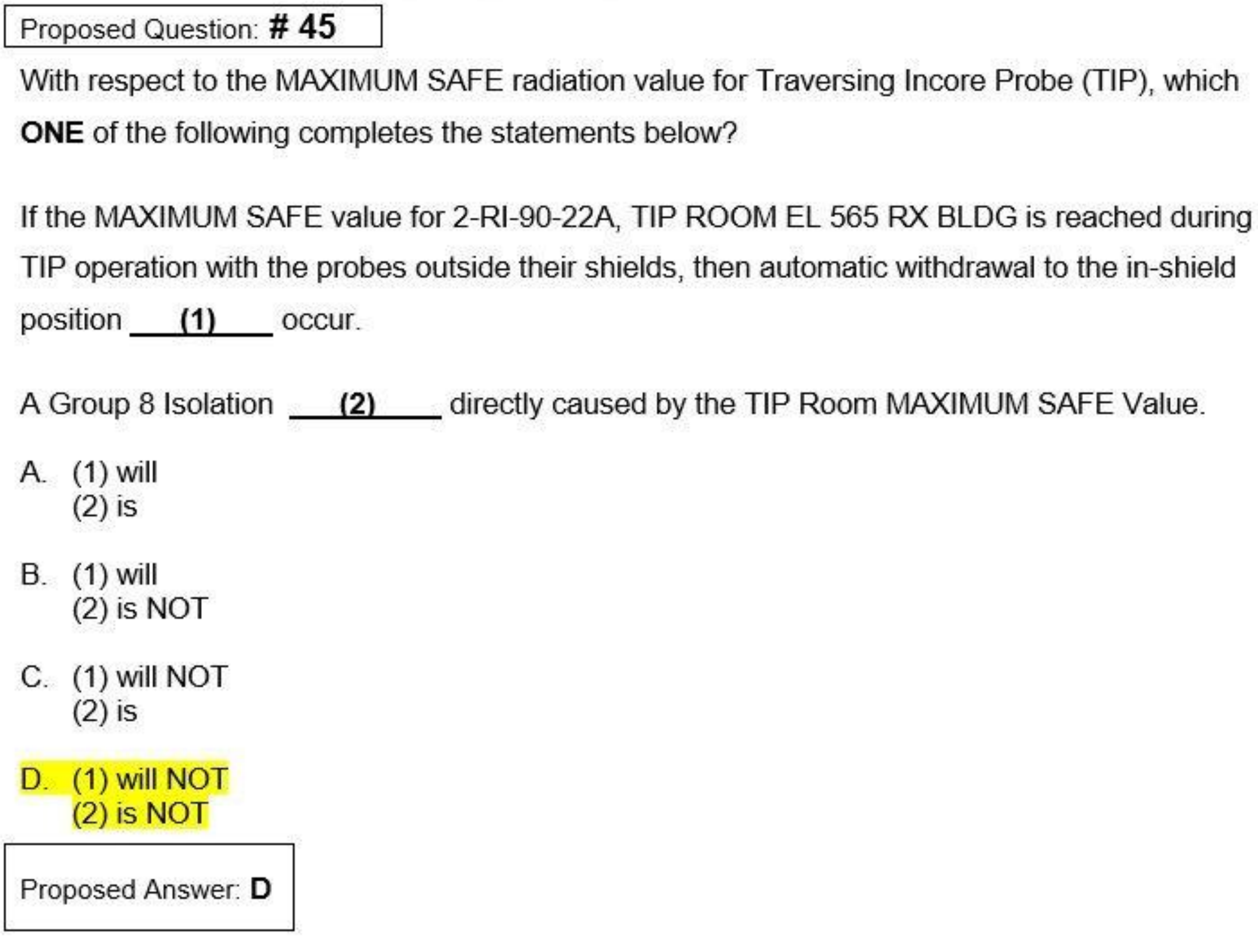
Learning Objective: OPL171.023, Obj. 5 (As available)

Question Source:	Bank #		(Note changes or attach parent)
	Modified Bank #	BFN 21-04 #45	
	New		
Question History:	Last NRC Exam	2021	

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

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

Copy of Bank Question:



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

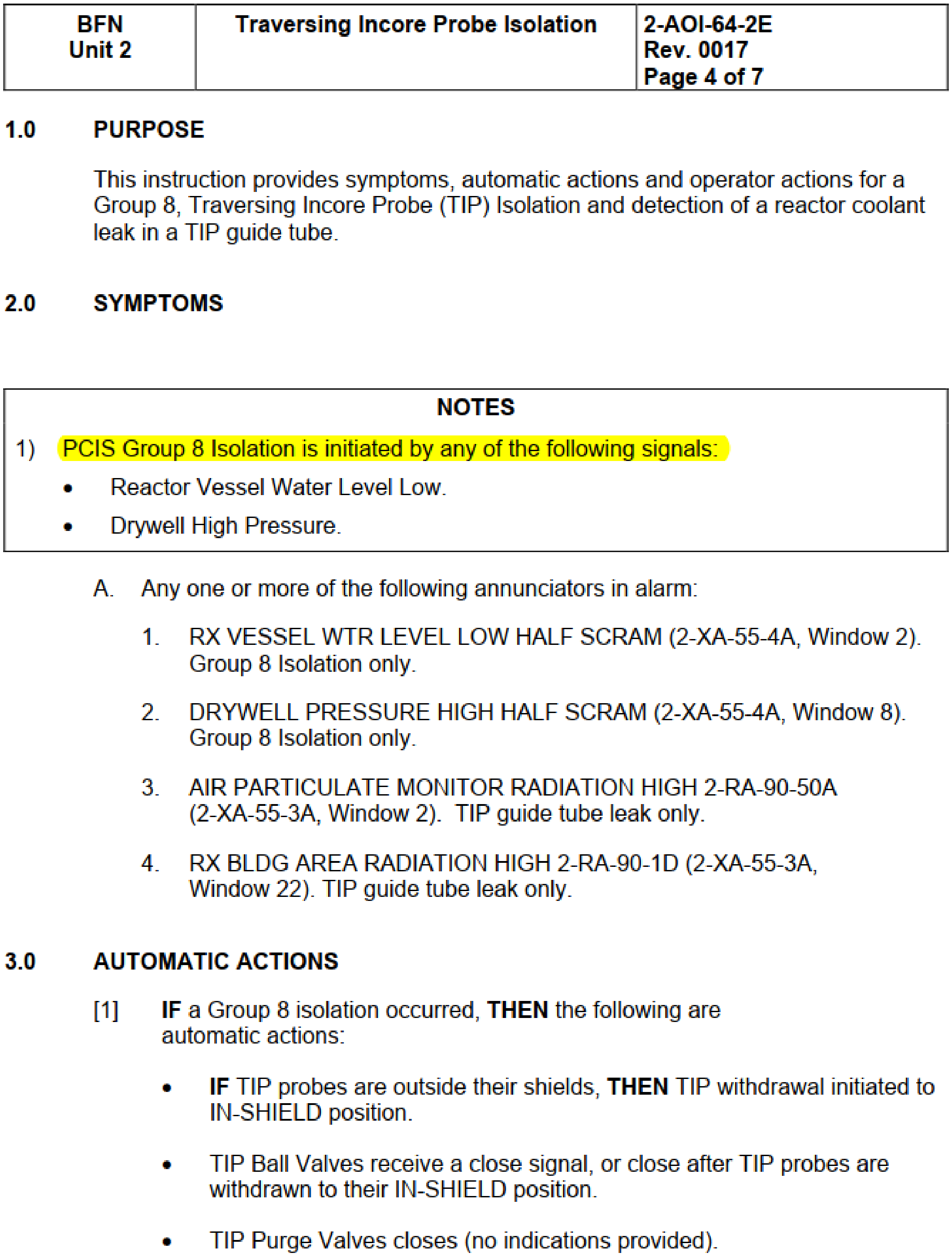
B. (1) will
(2) is NOT

C. (1) will NOT
(2) is

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

Proposed Answer: **D**

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

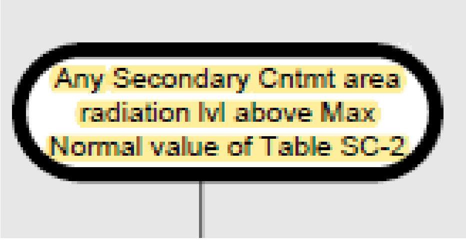
[1] **IF** a Group 8 isolation occurred, **THEN** the following are automatic actions:

- **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-EOI-3:



Any Secondary Cntmt area radiation lvl above Max Normal value of Table SC-2

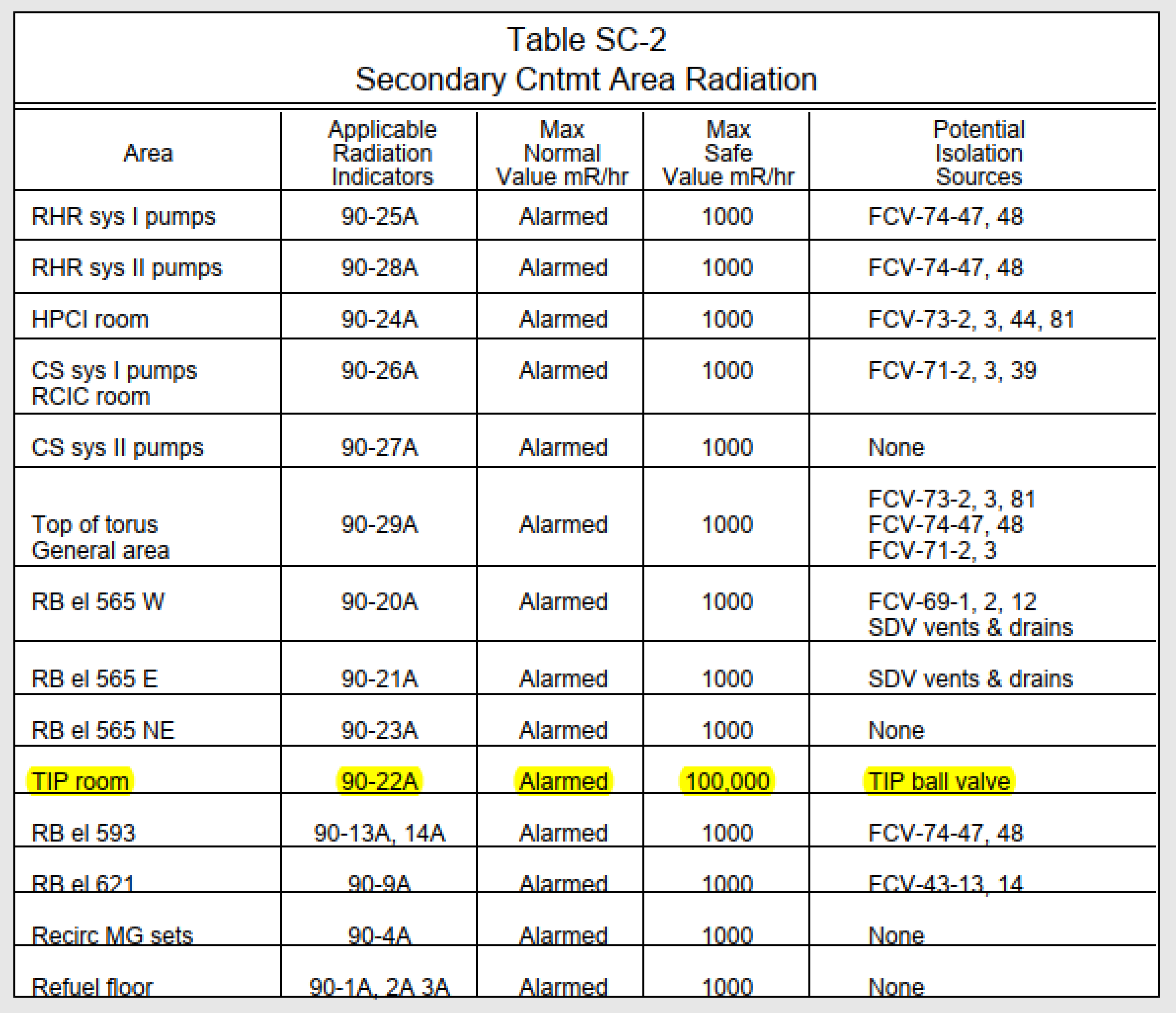
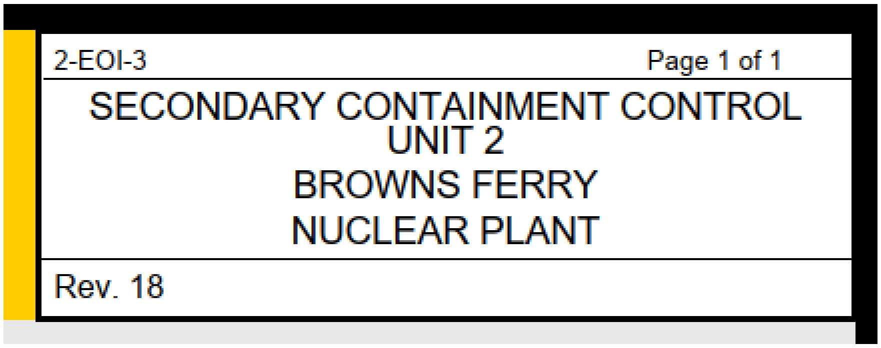


Table SC-2
Secondary Cntmt Area Radiation

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. 18

Excerpts from 2-ARP-9-3A: Supports Distractors A(2), C(2)

BFN **Unit 2**	**Panel 9-3** **2-XA-55-3A**	**2-ARP-9-3A** **Rev. 0057** **Page 51 of 60**

TURBINE BLDG
AREA RADIATION
HIGH

2-RA-90-1E

29

(Page 1 of 2)

Sensor/Trip Point:

2-RI-90-5A
2-RI-90-6A
2-RI-90-7A
2-RI-90-10A
2-RI-90-16A
2-RI-90-17A
2-RI-90-19A
2-RI-90-31A

For setpoints **REFER TO** 2-SIMI-90B.

Sensor Location:

RE-90-5	Generator Operating Floor TB El 617'	T-8 D-LINE
RE-90-6	RFP Operating Floor TB El 617'	T-6 E-LINE
RE-90-7	Turbine Operating Floor TB El 617'	T-9 K-LINE
RE-90-10	FW Heater Area TB El 586'	T-6 J-LINE
RE-90-16	Hotwell Pump Area TB El 557'	T-8 C-LINE
RE-90-17	Condenser Room Area TB El 557'	T-8 F-LINE
RE-90-19	Outside Steam Tunnel TB El 565'	T-9 J-LINE
RE-90-31	RCW Pump Area TB El 557'	T-6 C-LINE

Probable Cause:

A. Radiation levels have risen above alarm setpoint.
B. Sensor malfunction.

Automatic Action: None

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** the TSC is **NOT** manned, **THEN**
USE public address system to evacuate area where high airborne conditions exist.

C. **IF** the TSC is manned, **THEN**
NOTIFY the TSC to evacuate non-essential personnel from affected areas.

D. **NOTIFY** RAD PRO.

Continued on Next Page

BFN **Unit 2**	**Panel 9-3** **2-XA-55-3A**	**2-ARP-9-3A** **Rev. 0057** **Page 52 of 60**

TURBINE BLDG AREA RADIATION HIGH 2-RA-90-1E, Window 29
(Page 2 of 2)

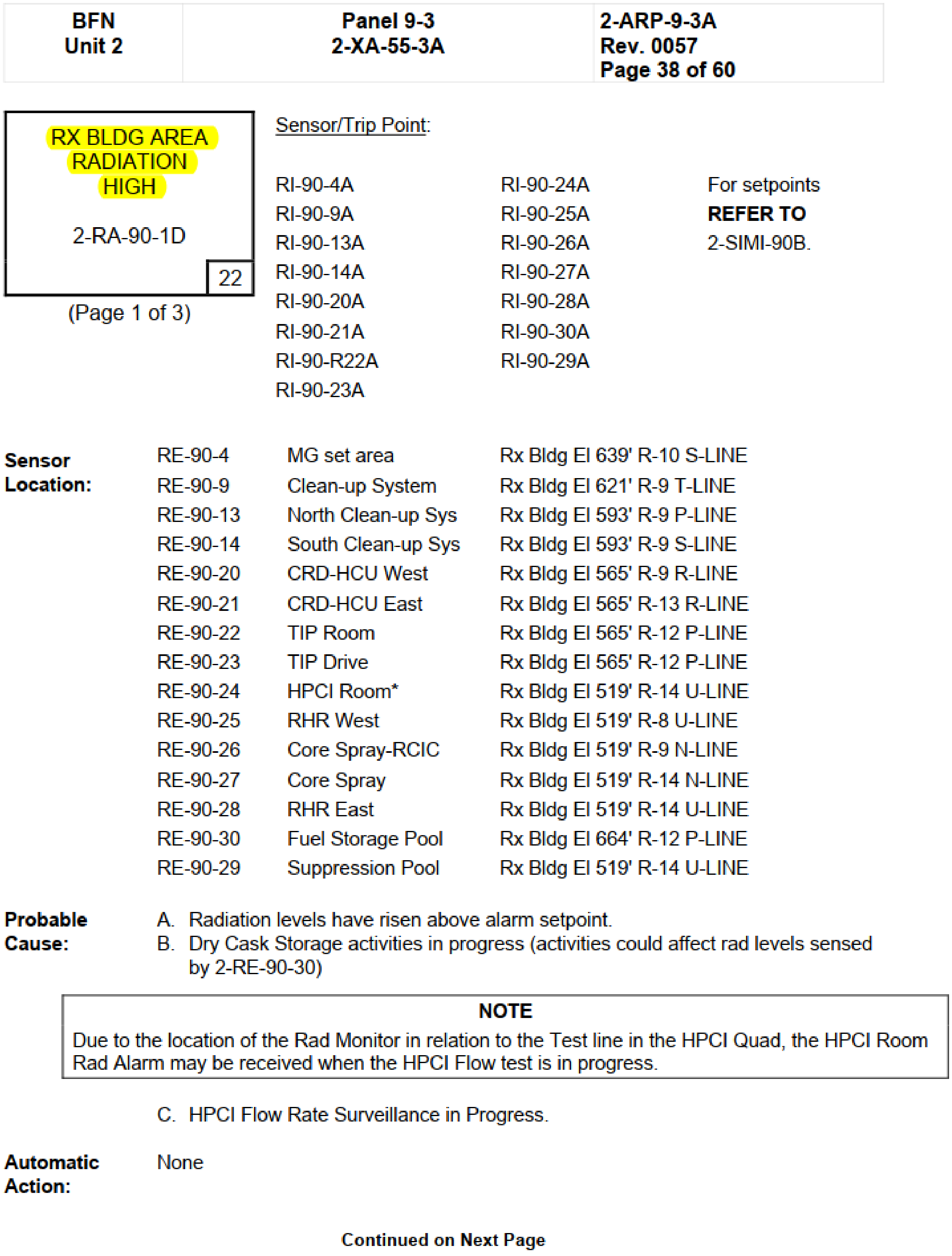
Operator Action: (Continued)

E. **MONITOR** other parameters providing input to this annunciator frequently as these parameters will be masked from alarming while this alarm is sealed in.

F. **IF** alarm is due to sensor malfunction, **THEN** **REFER TO** 0-OI-55.

G. **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	2-47E610-90-1	GE 0-730E356-1



BFN **Unit 2**	**Panel 9-3** **2-XA-55-3A**	**2-ARP-9-3A** **Rev. 0057** **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-90-9A
RI-90-13A
RI-90-14A
RI-90-20A
RI-90-21A
RI-90-R22A
RI-90-23A

RI-90-24A
RI-90-25A
RI-90-26A
RI-90-27A
RI-90-28A
RI-90-30A
RI-90-29A

For setpoints **REFER TO** 2-SIMI-90B.

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)

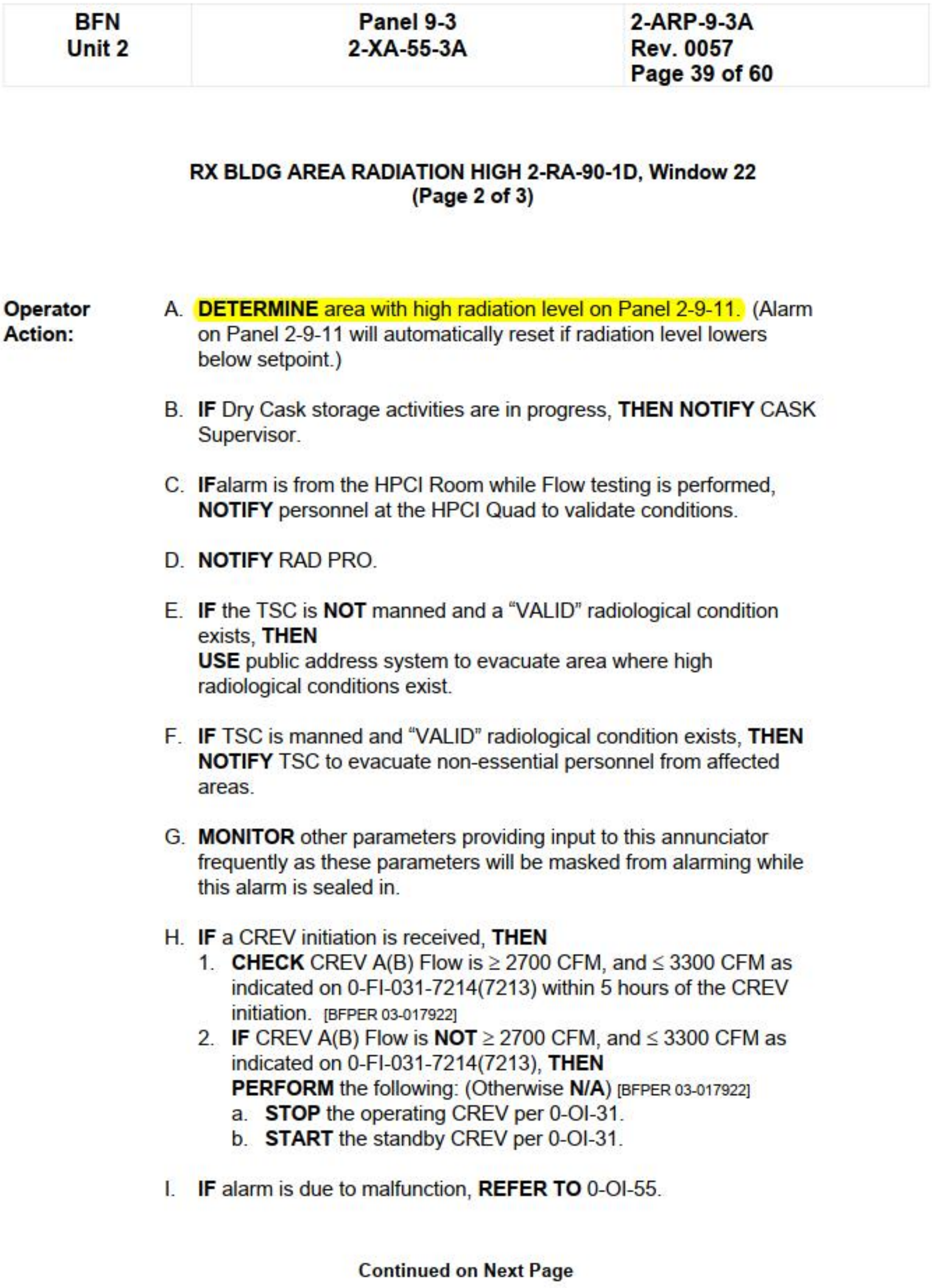
NOTE

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. 0057** **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. **IF**alarm is from the HPCI Room while Flow testing is performed, **NOTIFY** personnel at the HPCI Quad to validate conditions.

D. **NOTIFY** RAD PRO.

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.

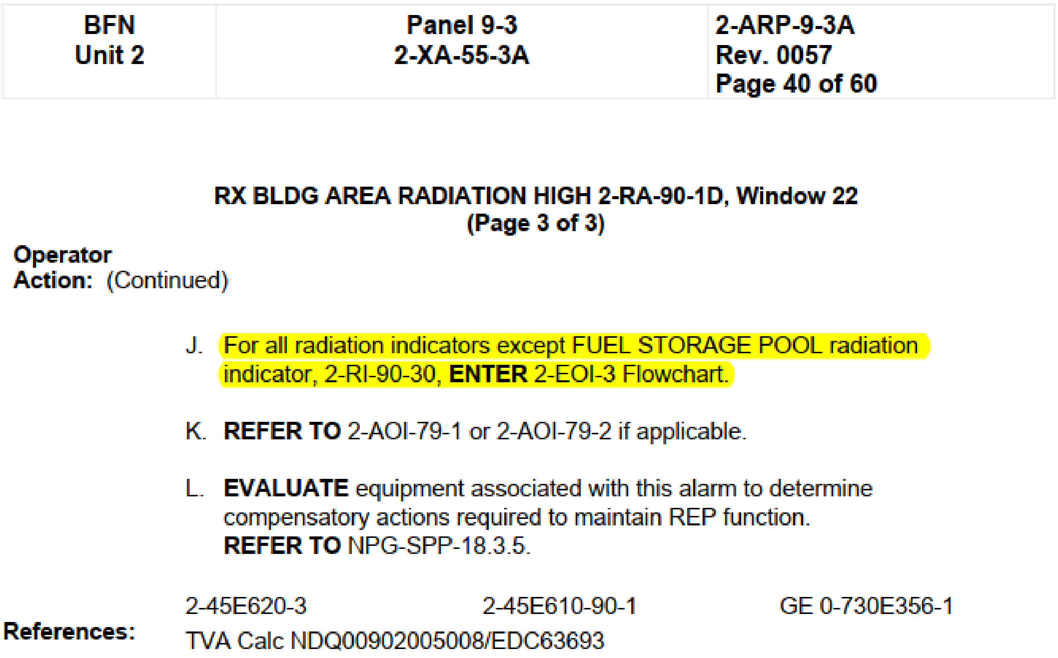
F. **IF** TSC is manned and “VALID” radiological condition exists, **THEN NOTIFY** TSC to evacuate non-essential personnel from affected areas.

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. [BFPER 03-017922]
   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**) [BFPER 03-017922]
      a. **STOP** the operating CREV per 0-OI-31.
      b. **START** the standby CREV per 0-OI-31.

I. **IF** alarm is due to malfunction, **REFER TO** 0-OI-55.

Continued on Next Page



BFN **Unit 2**	**Panel 9-3** **2-XA-55-3A**	**2-ARP-9-3A** **Rev. 0057** **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

Examination Outline Cross-reference:

202001 (SF1, SF4 RS) Recirculation

K5.06 (10CFR 55.41.5)

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

♦ ATWS RPT

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	202001K5.06	
Importance Rating	3.8	------

Proposed Question: **# 20**

The Limiting Condition for Operations (LCO) statement for Unit 2 Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS - RPT) Instrumentation, LCO 3.3.4.2, requires OPERABILITY of the function for Reactor Vessel Water Level at ___**(1)**___ and Reactor Steam Dome Pressure at ___**(2)**___.

A. **(1)** (-) 45 inches
(2) 1073 psig

B. **(1)** (-) 45 inches
(2) 1162 psig

C. **(1)** (-) 122 inches
(2) 1073 psig

D. **(1)** (-) 122 inches
(2) 1162 psig

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B)*. Second part is incorrect but plausible in that RPS Instrumentation LCO 3.3.1.1 contains Reactor Vessel Steam Dome Pressure - High, however it is specifically related to the High Reactor Pressure SCRAM signal at 1073 psig not for ATWS - RPT at 1162 psig.

B **CORRECT**: *(See attached)* In accordance with Tech Spec 3.3.4.2, the LCO states Two channels per trip system for each ATWS - RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level - Low Low, Level 2 (Level 2 at BFN is (-) 45 inches)

b. Reactor Steam Dome Pressure - High (which was recently changed from 1148 psig to 1162 psig due to Power Uprate). These signals could be reached when conditions are such that RPS might NOT have successfully SCRAMMED the Reactor.

C INCORRECT: First part is incorrect but plausible in that Primary Containment Isolation (PCIS) Instrumentation LCO 3.3.6.1 contains Reactor Vessel Water Level - Low Low Low, Level 1 (Level 1 at BFN is (-) 122 inches), however it is specifically related to the PCIS Group 1 Isolation signal for MSIV closure. Additionally, (-) 122 inches is part of ECCS RHRSW and EDG initiation logic. 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 above the line information for the Tech Spec LCO related to Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS - RPT) Instrumentation. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): (Attach if not previously provided)

OPL171.003, Rev. 26
2-OI-99, Rev. 96
2-ARP-9-4A, Rev. 51
U2 Tech Specs 3.3.1.1, Amend. 258
U2 Tech Specs 3.3.4.2, Amend. 253
U2 Tech Specs 3.3.6.1, Amend. 296

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.003, Obj. 11 (As available)

Question Source:	Bank #	BFN NRC 1804 #17	
	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:

Proposed Question: **# 17**

Which ONE of the following completes the statements below?

The Limiting Condition for Operations (LCO) statement for Unit 2 Anticipated Transient Without SCRAM Recirculation Pump Trip (**ATWS - RPT**) Instrumentation, LCO 3.3.4.2, requires operability of the function for the Reactor Vessel Water Level at ___**(1)**___ and for the Reactor Steam Dome Pressure at ___**(2)**___ ?

A. **(1)** (-) 45 inches
(2) 1073 psig

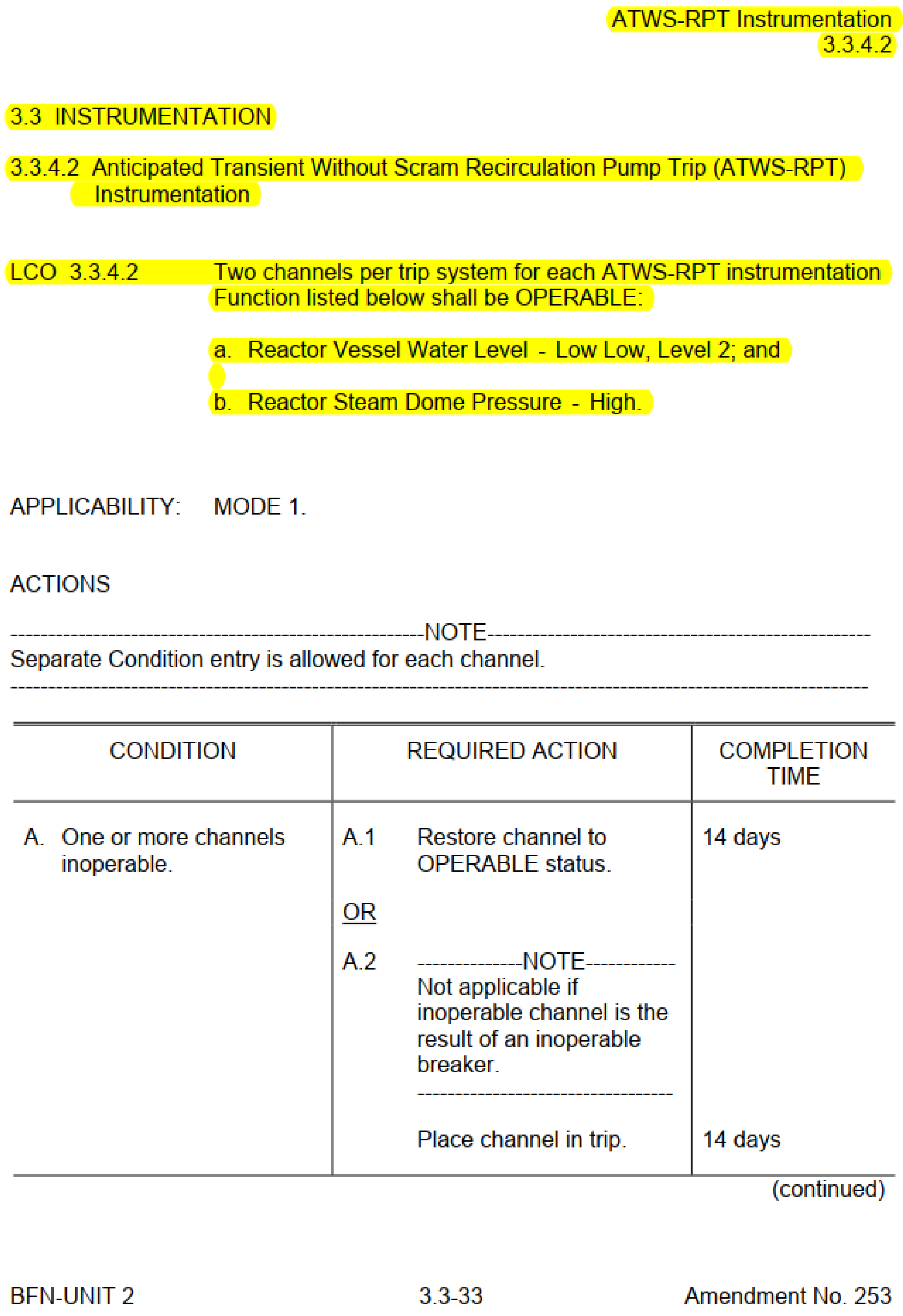
B. **(1)** (-) 45 inches
(2) 1148 psig

C. **(1)** (-) 122 inches
(2) 1073 psig

D. **(1)** (-) 122 inches
(2) 1148 psig

Proposed Answer: **B**

Excerpt from applicable U2 Tech Spec 3.3.4.2:



ATWS-RPT Instrumentation
3.3.4.2

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level - Low Low, Level 2; and

b. Reactor Steam Dome Pressure - High.

APPLICABILITY: MODE 1.

ACTIONS

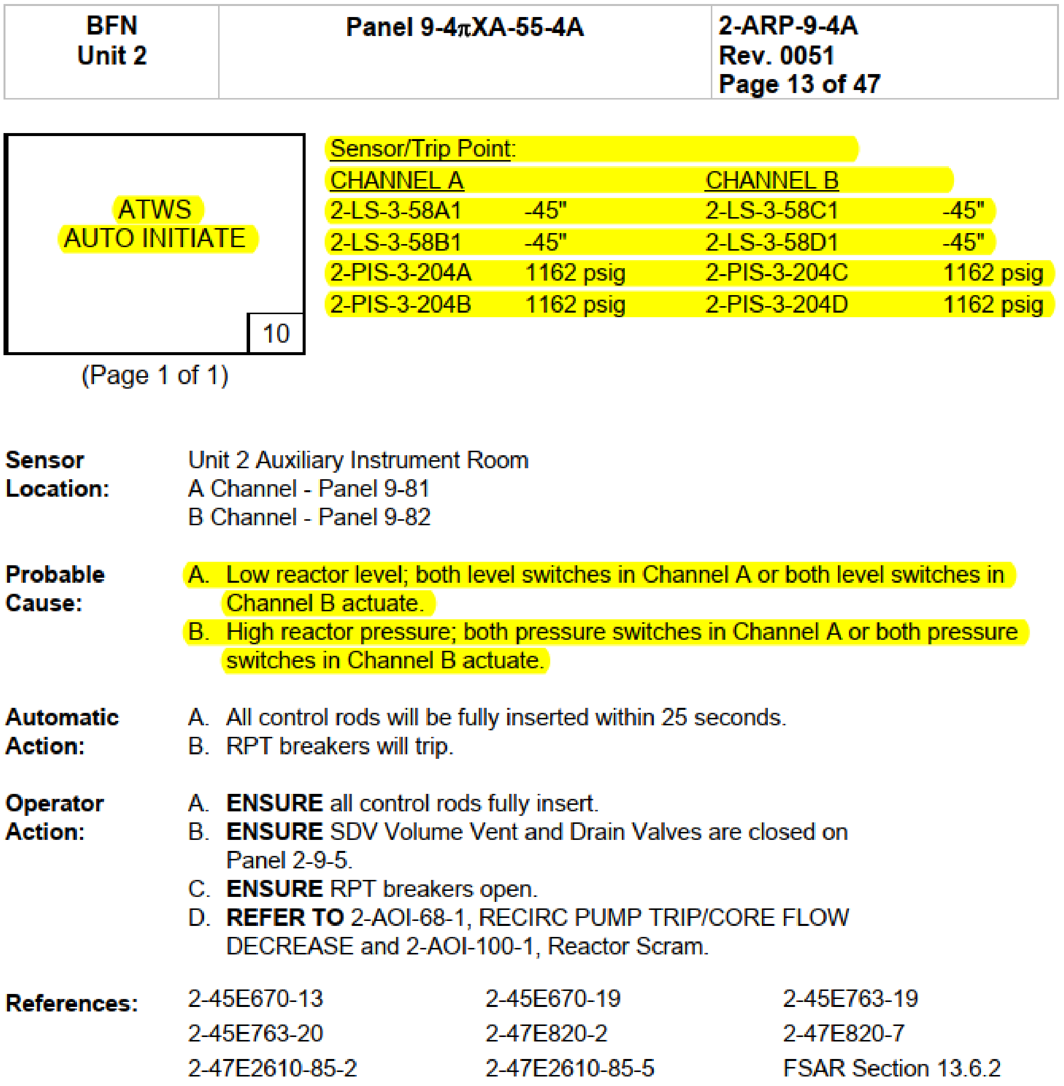
---NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	14 days
	<u>OR</u>	
	A.2 --------------NOTE------------ Not applicable if inoperable channel is the result of an inoperable breaker. ---------------------------------- Place channel in trip.	14 days

(continued)

BFN-UNIT 2 3.3-33 Amendment No. 253

Excerpt from 2-ARP-9-4A: Lists all related ATWS - RPT instrumentation/setpoints



BFN **Unit 2**	**Panel 9-4πXA-55-4A**	**2-ARP-9-4A** **Rev. 0051** **Page 13 of 47**

ATWS
AUTO INITIATE
10

(Page 1 of 1)

Sensor/Trip Point:

CHANNEL A		CHANNEL B	
2-LS-3-58A1	-45"	2-LS-3-58C1	-45"
2-LS-3-58B1	-45"	2-LS-3-58D1	-45"
2-PIS-3-204A	1162 psig	2-PIS-3-204C	1162 psig
2-PIS-3-204B	1162 psig	2-PIS-3-204D	1162 psig

Sensor Location:
Unit 2 Auxiliary Instrument Room
A Channel - Panel 9-81
B Channel - Panel 9-82

Probable Cause:
A. Low reactor level; both level switches in Channel A or both level switches in Channel B actuate.
B. High reactor pressure; both pressure switches in Channel A or both pressure switches in Channel B actuate.

Automatic Action:
A. All control rods will be fully inserted within 25 seconds.
B. RPT breakers will trip.

Operator Action:
A. **ENSURE** all control rods fully insert.
B. **ENSURE** SDV Volume Vent and Drain Valves are closed on Panel 2-9-5.
C. **ENSURE** RPT breakers open.
D. **REFER TO** 2-AOI-68-1, RECIRC PUMP TRIP/CORE FLOW DECREASE and 2-AOI-100-1, Reactor Scram.

References:

2-45E670-13	2-45E670-19	2-45E763-19
2-45E763-20	2-47E820-2	2-47E820-7
2-47E2610-85-2	2-47E2610-85-5	FSAR Section 13.6.2

Excerpts from U2 Tech Spec 3.3.1.1: Supports Distractors A(2), C(2)

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.

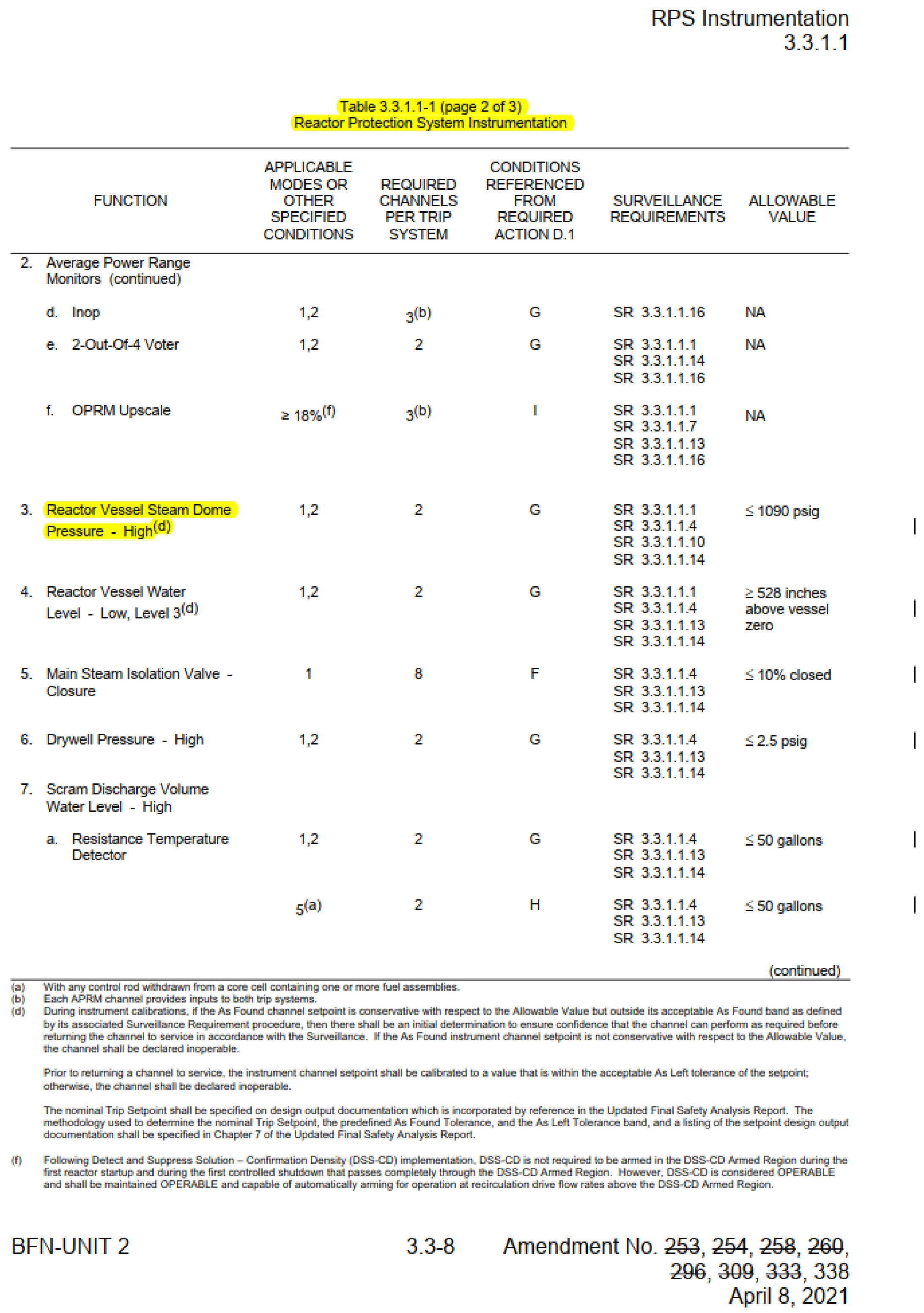
APPLICABILITY: According to Table 3.3.1.1-1.

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)



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	≥ 18%(f)	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	NA
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 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 3.3.1.1.4 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 3.3.1.1.4 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.4 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 3.3.1.1.4 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.4 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 that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

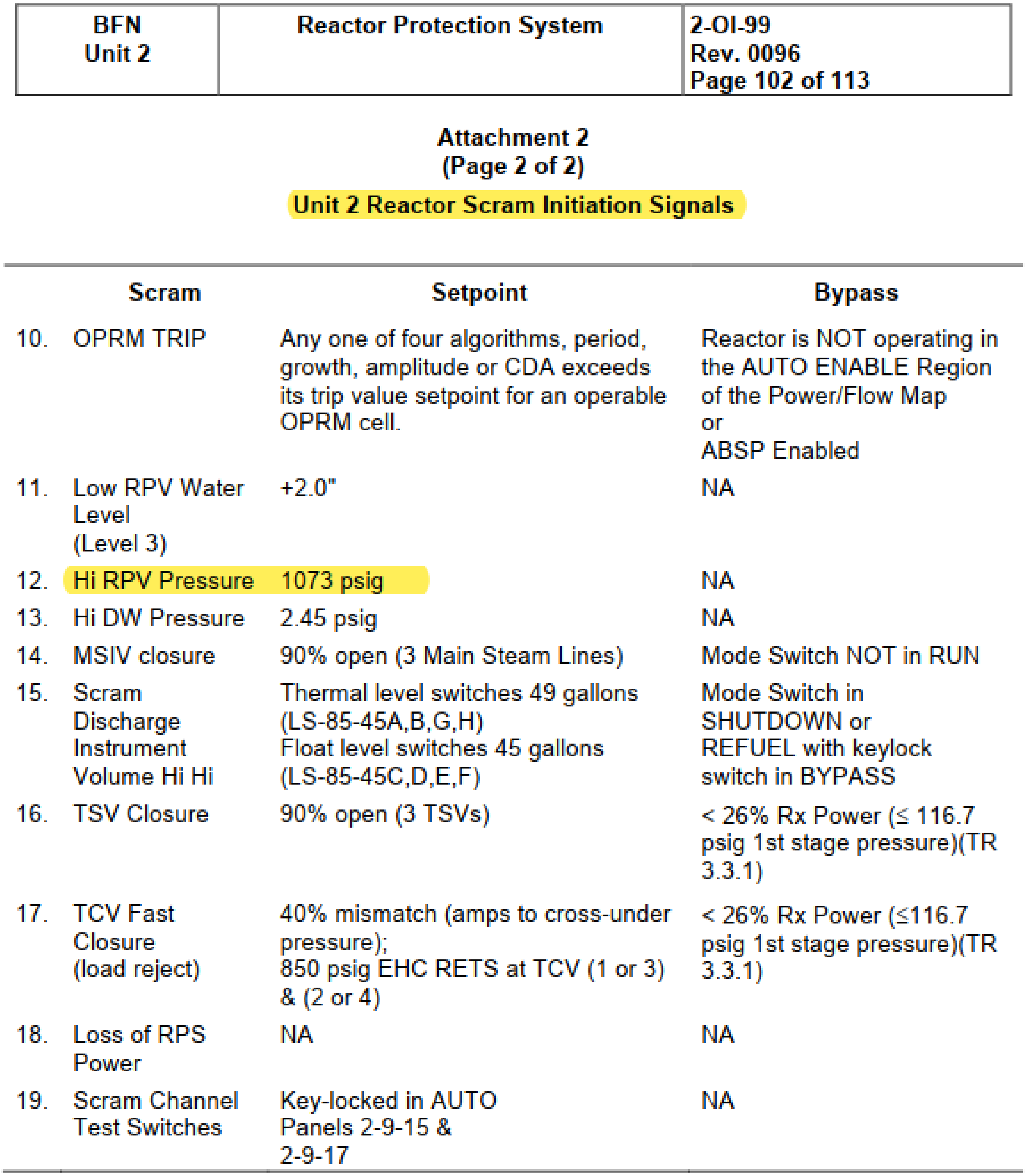
Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

(f) Following Detect and Suppress Solution – Confirmation Density (DSS-CD) implementation, DSS-CD is not required to be armed in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD is considered OPERABLE and shall be maintained OPERABLE and capable of automatically arming for operation at recirculation drive flow rates above the DSS-CD Armed Region.

BFN-UNIT 2 3.3-8 Amendment No. ~~253~~, ~~254~~, ~~258~~, ~~260~~, ~~296~~, ~~309~~, ~~333~~, 338
April 8, 2021

Excerpt from 2-OI-99: Supports Distractors A(2), C(2)



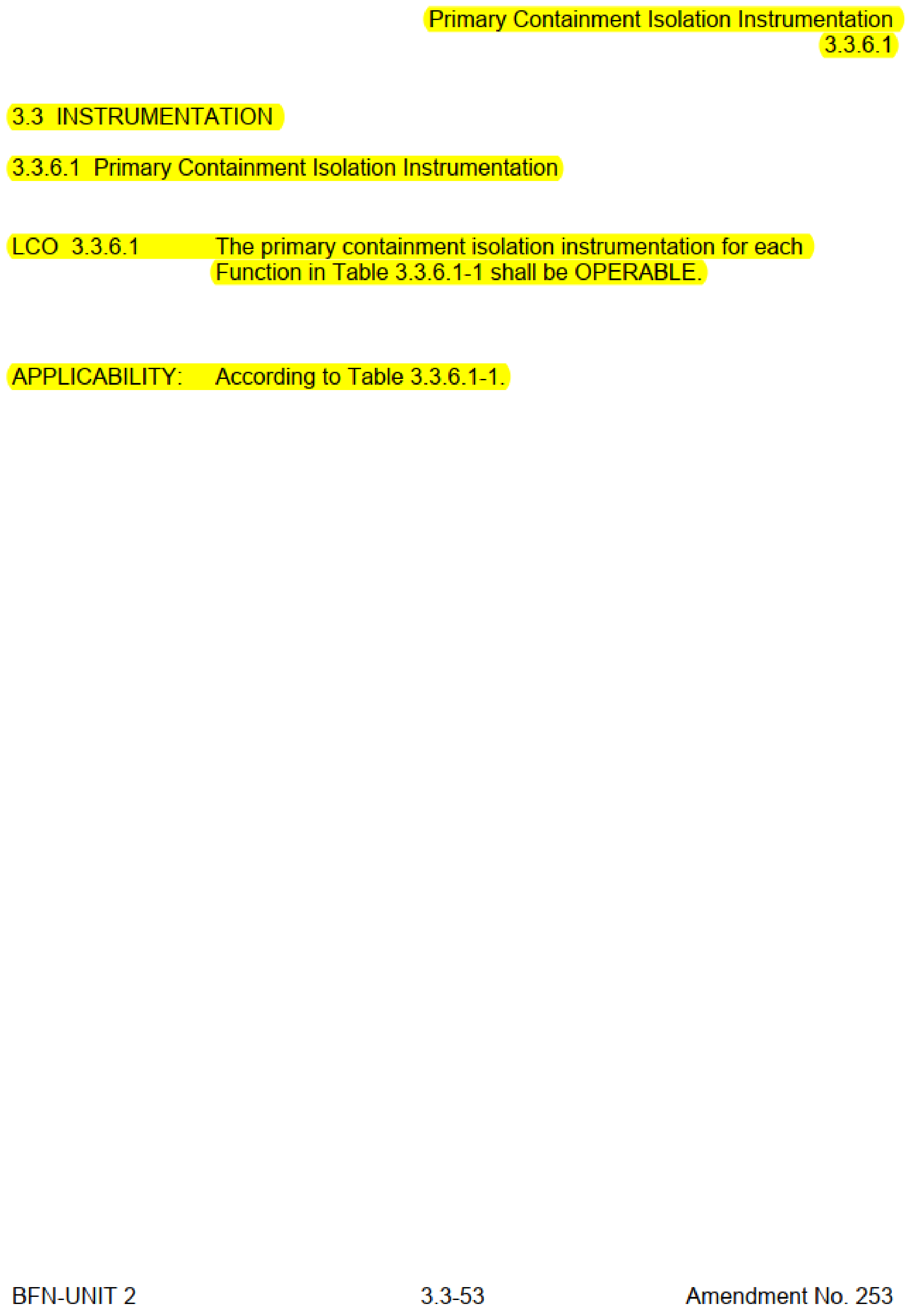
BFN **Unit 2**	**Reactor Protection System**	**2-OI-99** **Rev. 0096** **Page 102 of 113**

Attachment 2
(Page 2 of 2)

Unit 2 Reactor Scram Initiation Signals

Scram	**Setpoint**	**Bypass**
10. OPRM TRIP	Any one of four algorithms, period, growth, amplitude or CDA exceeds its trip value setpoint for an operable OPRM cell.	Reactor is NOT operating in the AUTO ENABLE Region of the Power/Flow Map or ABSP Enabled
11. Low RPV Water Level (Level 3)	+2.0"	NA
12. Hi RPV Pressure	1073 psig	NA
13. Hi DW Pressure	2.45 psig	NA
14. MSIV closure	90% open (3 Main Steam Lines)	Mode Switch NOT in RUN
15. Scram Discharge Instrument Volume Hi Hi	Thermal level switches 49 gallons (LS-85-45A,B,G,H) Float level switches 45 gallons (LS-85-45C,D,E,F)	Mode Switch in SHUTDOWN or REFUEL with keylock switch in BYPASS
16. TSV Closure	90% open (3 TSVs)	< 26% Rx Power (≤ 116.7 psig 1st stage pressure)(TR 3.3.1)
17. TCV Fast Closure (load reject)	40% mismatch (amps to cross-under pressure); 850 psig EHC RETS at TCV (1 or 3) & (2 or 4)	< 26% Rx Power (≤116.7 psig 1st stage pressure)(TR 3.3.1)
18. Loss of RPS Power	NA	NA
19. Scram Channel Test Switches	Key-locked in AUTO Panels 2-9-15 & 2-9-17	NA

Excerpts from U2 Tech Spec 3.3.6.1: Supports Distractors C(1), D(1)

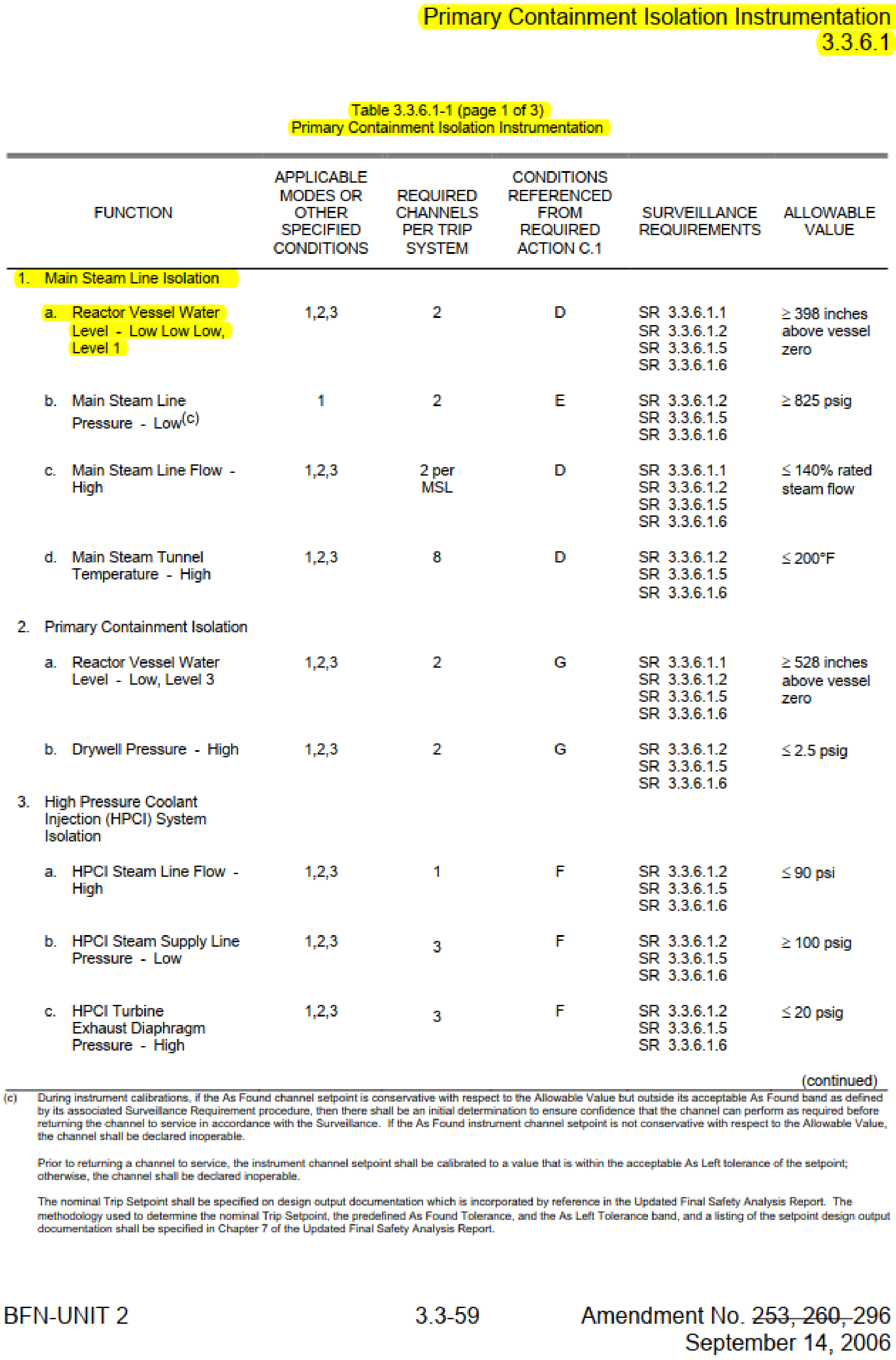


3.3 INSTRUMENTATION

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.



Primary Containment Isolation Instrumentation
3.3.6.1

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 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure - Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

(continued)

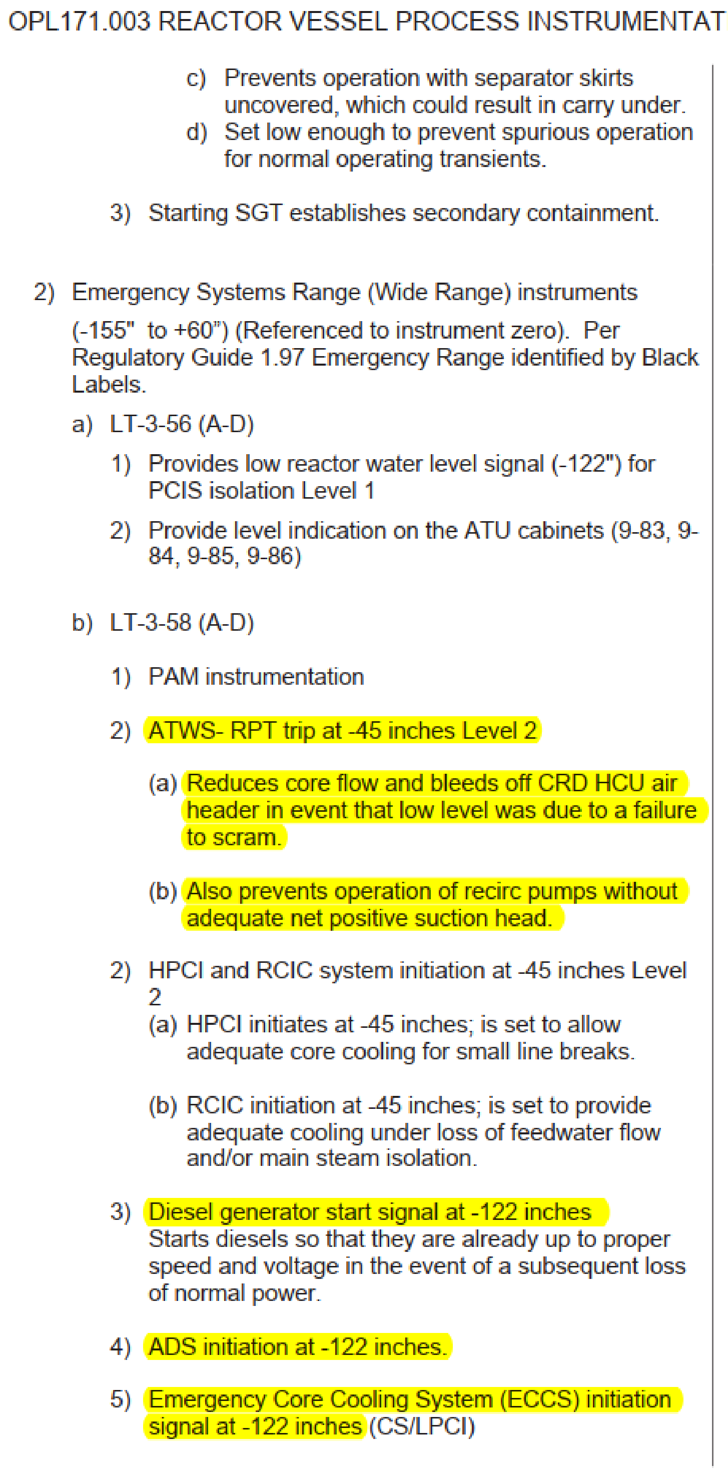
(c) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

BFN-UNIT 2 3.3-59 Amendment No. ~~253, 260,~~ 296
September 14, 2006

Excerpts from OPL171.003 Lesson Plan:



OPL171.003 REACTOR VESSEL PROCESS INSTRUMENTATION REV 26

        c) Prevents operation with separator skirts uncovered, which could result in carry under.
        d) Set low enough to prevent spurious operation for normal operating transients.

    3) Starting SGT establishes secondary containment.

2) Emergency Systems Range (Wide Range) instruments (-155" to +60") (Referenced to instrument zero). Per Regulatory Guide 1.97 Emergency Range identified by Black Labels.

    a) LT-3-56 (A-D)
        1) Provides low reactor water level signal (-122") for PCIS isolation Level 1
        2) Provide level indication on the ATU cabinets (9-83, 9-84, 9-85, 9-86)

    b) LT-3-58 (A-D)
        1) PAM instrumentation
        2) ATWS- RPT trip at -45 inches Level 2
            (a) Reduces core flow and bleeds off CRD HCU air header in event that low level was due to a failure to scram.
            (b) Also prevents operation of recirc pumps without adequate net positive suction head.
        2) HPCI and RCIC system initiation at -45 inches Level 2
            (a) HPCI initiates at -45 inches; is set to allow adequate core cooling for small line breaks.
            (b) RCIC initiation at -45 inches; is set to provide adequate cooling under loss of feedwater flow and/or main steam isolation.
        3) Diesel generator start signal at -122 inches Starts diesels so that they are already up to proper speed and voltage in the event of a subsequent loss of normal power.
        4) ADS initiation at -122 inches.
        5) Emergency Core Cooling System (ECCS) initiation signal at -122 inches (CS/LPCI)

OPL171.003 REACTOR VESSEL PROCESS INSTRUMENTATION REV 26

L-CHK-85-764A, B chattering after CRD pressure dropped below reactor pressure.

2. Reactor Vessel Pressure Instrumentation (Utilizes same piping as vessel water level instrumentation)
   a) Pressure Instrument Functions
      1) PT-3-204 (A-D)
         a) Provides the high reactor vessel pressure signal at 1162 psig to initiate an ATWS - RPT.
         b) Provides input for opening logic for all SRVs. Uses slave relays (4 per SRV) set at 1135, 1145, or 1155 psig in a 2 of 2 once logic to open SRV.
         c) Provides input to EHC for Reactor Pressure Control
         d) Provides pressure indication on the ATU cabinets (9-83, 9-84, 9-85, 9-86).
         e) Pressure input is from Steam space
      2) PT-3-22-AA, -BB, -C, -D
         a) Provide the reactor vessel high pressure (1073 psig) signal to RPS for reactor scram.
         b) Provide reactor pressure indication on the ATU cabinets (9-83, 9-84, 9-85, 9-86).
         c) Pressure input is from the Steam space
      3) PIS-3-22A and B:
         a) Trip mechanical vacuum pumps if reactor pressure is >600 psig and condenser vacuum is >22 inches Hg.
         b) Pressure input is from Steam space.
      4) PT-3-54, -61, -207
         a) Provide pressure input to the FWLCS
         b) Provides reactor pressure indication on recorder PR-3-53 (Panel 9-5) over a range of 0-1500 psig (average pressure). Reactor high pressure alarm is actuated at 1058 psig. PT-3-54, -61, -207 provide reactor pressure indication on Panel 9-5.
         c) Pressure input is from the Steam space

Obj. ILT-14, NLO-8, NLOR-4

Obj. ILT-14, NLO-8, NLOR-4

Obj. ILT-14, NLO-8, NLOR-4

Obj. ILT-14, NLO-8, NLOR-4

Examination Outline Cross-reference:

295019 (APE 19) Partial or Complete Loss of Instrument Air / 8

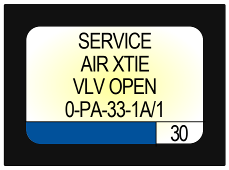
G2.4.50 (10CFR 55.41.10)

Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure.

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295019G2.4.50	
Importance Rating	4.2	------

Proposed Question: **# 21**

Unit 1 is operating at 100% RTP when a Control Air leak develops, resulting in the following conditions:



- SERVICE AIR CROSSTIE VALVE OPEN (1-9-20B, Window 30) alarms
- Control Air Pressure is 69 psig and slowly lowering

Given the conditions above and in accordance with the appropriate Alarm Response Procedure, 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE opened at ___**(1)**___ Control Air Pressure.

A manual Reactor SCRAM is required if Control Air Pressure continues to lower to less than ___**(2)**___ as indicated by 1-PI-32-20 on Panel 1-9-20 in accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors.

A. (1) 70 psig
(2) 55 psig

B. (1) 70 psig
(2) 66 psig

C. (1) 85 psig
(2) 55 psig

D. (1) 85 psig
(2) 66 psig

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that 0-OI-32, Control Air System, Attachment 1 – Control Air System Pressure Spectrum lists CONTROL AIR PRESSURE LOW alarms at 70 psig Control Air Pressure. 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 (*See D*).

C **CORRECT:** *(See attached)* In accordance with the given SERVICE AIR CROSSTIE VALVE OPEN (1-9-20B, Window 30), 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE, opens at 85 psig Control Air Pressure. For second part, in accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors, a manual Reactor SCRAM is required if Control Air Pressure continues to lower to less than 55 psig.

D INCORRECT: The first part is correct (*See C*). The second part is incorrect but plausible in that 0-OI-32, Control Air System, Attachment 1 – Control Air System Pressure Spectrum lists SCRAM PILOT AIR HEADER PRESSURE LOW alarms at 66 psig Control Air Pressure.

RO Level Justification: Tests the candidate's ability to verify Control Air System setpoints and the controls identified in Alarm Response Procedures as it relates to partial or complete loss of instrument air. This question is rated as Memory due to the requirement to strictly recall facts related to operation of the Control Air System.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s):	1-ARP-9-20B, Rev. 38	(Attach if not previously provided)
	0-AOI-32-1, Rev. 56	
	1-ARP-9-5B, Rev. 24	
	0-OI-32, Rev. 142	

Proposed references to be provided to applicants during examination: **SERVICE AIR CROSSTIE VALVE OPEN (1-9-20B, Window 30)**

Learning Objective: OPL171.054, Obj.7 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 2104 #60	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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:

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

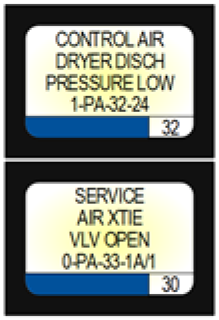
Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	300000A2.01	

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

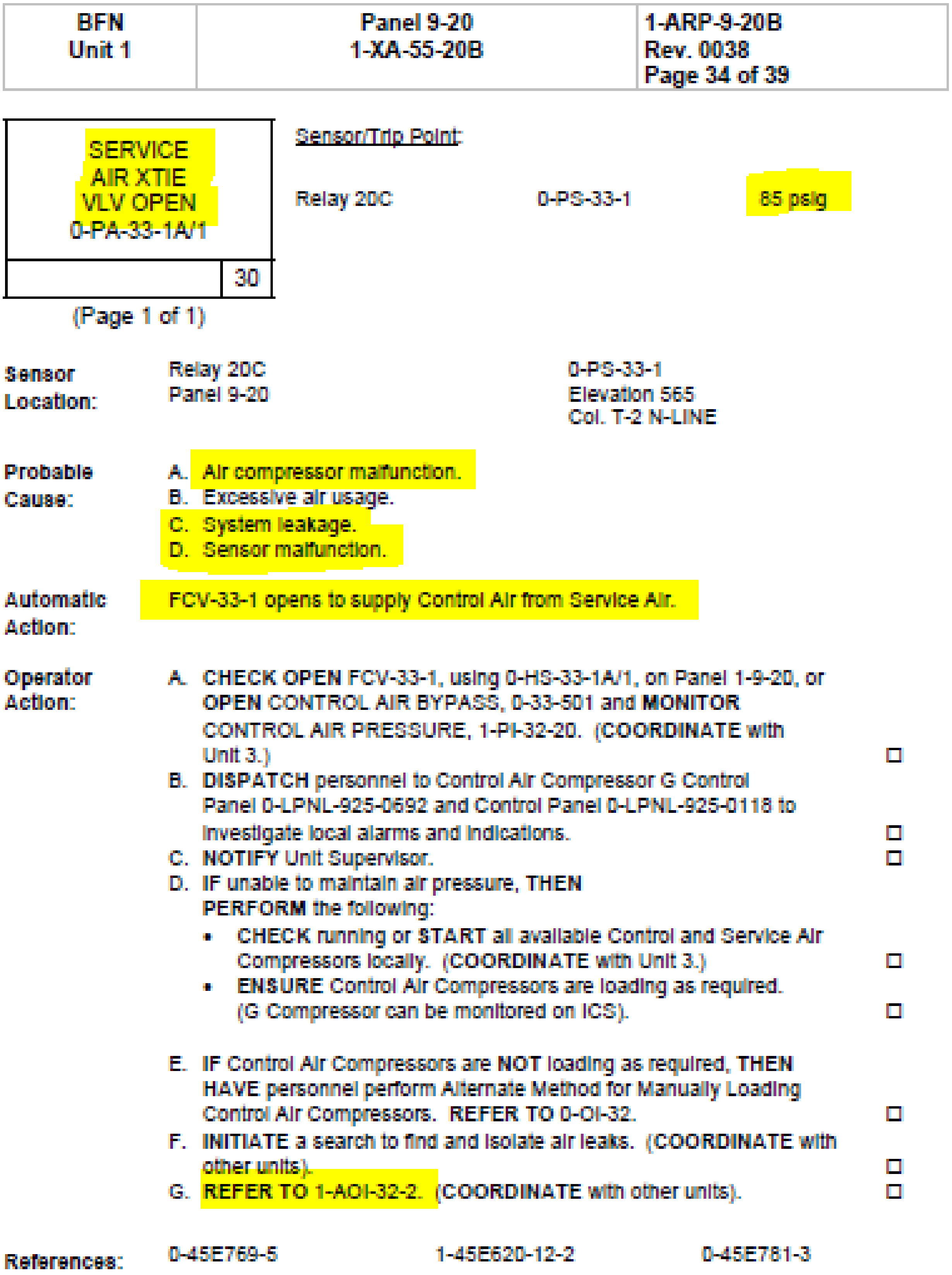
B. (1) 70 psig
(2) is NOT

C. (1) 85 psig
(2) is

D. (1) 85 psig
(2) is NOT

Proposed Answer: **D**

Excerpt from 1-ARP-9-20B:



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:

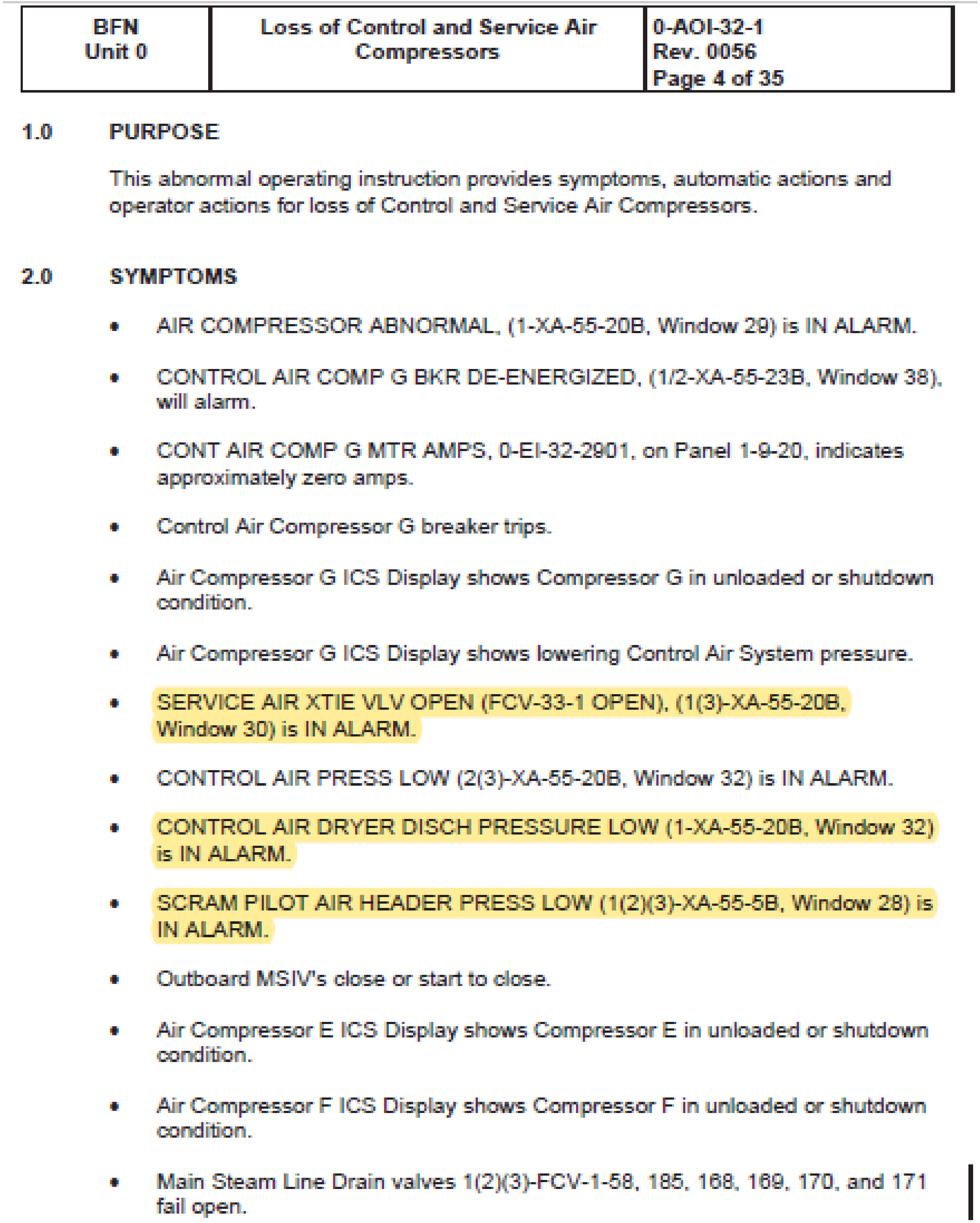
A. **CHECK OPEN** FCV-33-1, using 0-HS-33-1A/1, on Panel 1-9-20, or **OPEN** CONTROL AIR BYPASS, 0-33-501 and **MONITOR** CONTROL AIR PRESSURE, 1-PI-32-20. (**COORDINATE** with Unit 3.) ☐
B. **DISPATCH** personnel to Control Air Compressor G Control Panel 0-LPNL-925-0692 and Control Panel 0-LPNL-925-0118 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:



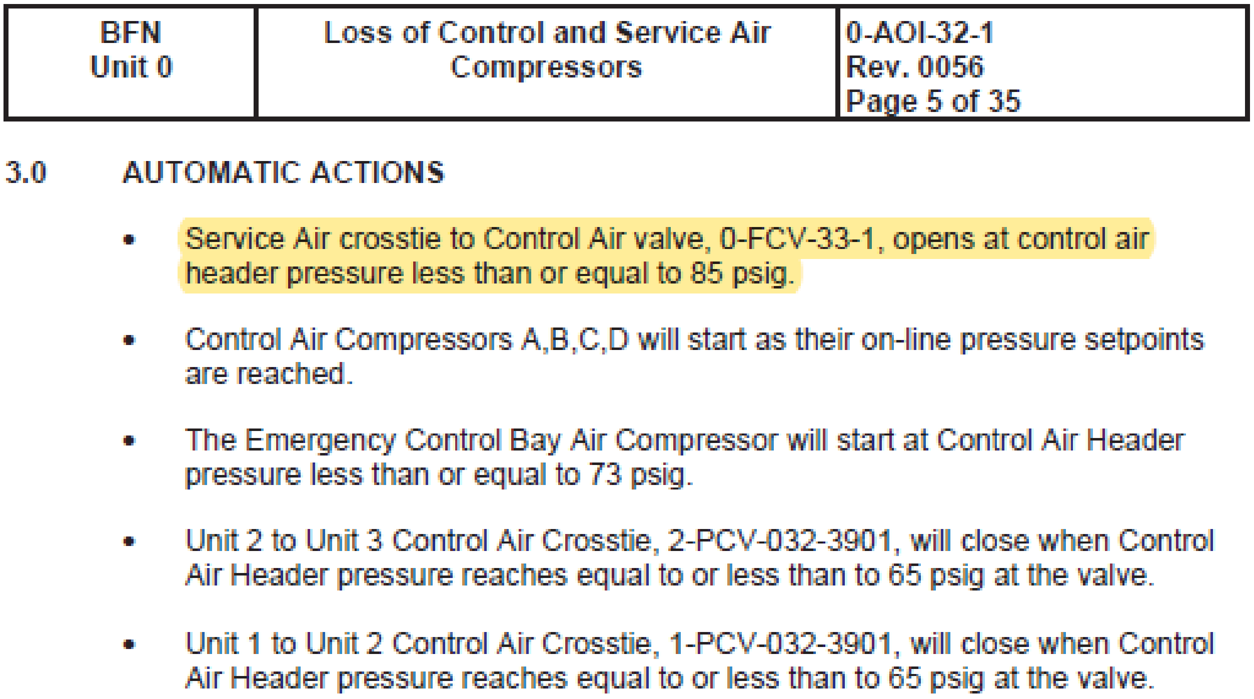
BFN **Unit 0**	**Loss of Control and Service Air Compressors**	**0-AOI-32-1** **Rev. 0056** **Page 4 of 35**

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

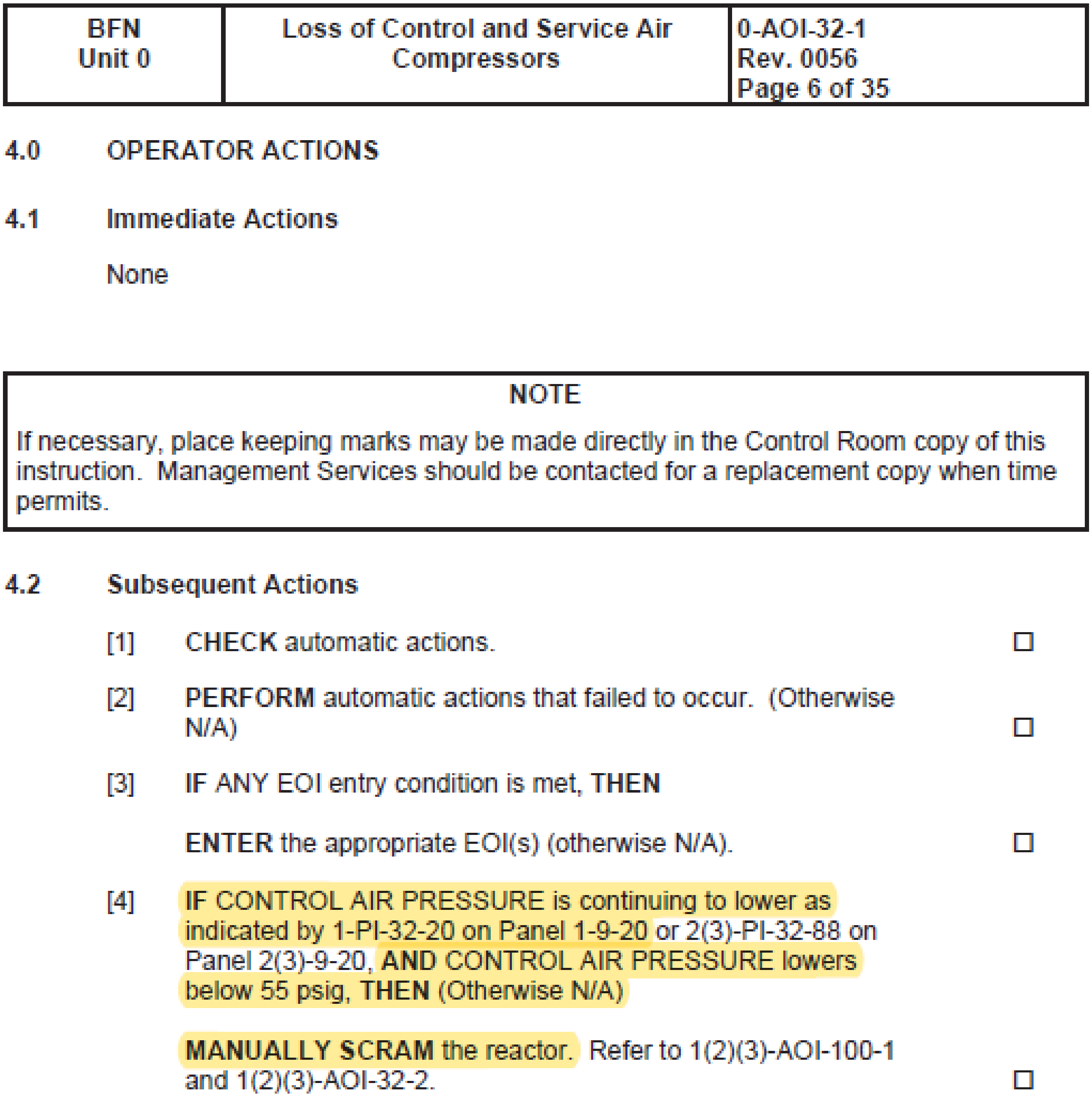
- AIR COMPRESSOR ABNORMAL, (1-XA-55-20B, Window 29) is IN ALARM.
- CONTROL AIR COMP G BKR DE-ENERGIZED, (1/2-XA-55-23B, Window 38), will alarm.
- CONT AIR COMP G MTR AMPS, 0-EI-32-2901, on Panel 1-9-20, indicates approximately zero amps.
- Control Air Compressor G breaker trips.
- Air Compressor G ICS 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)

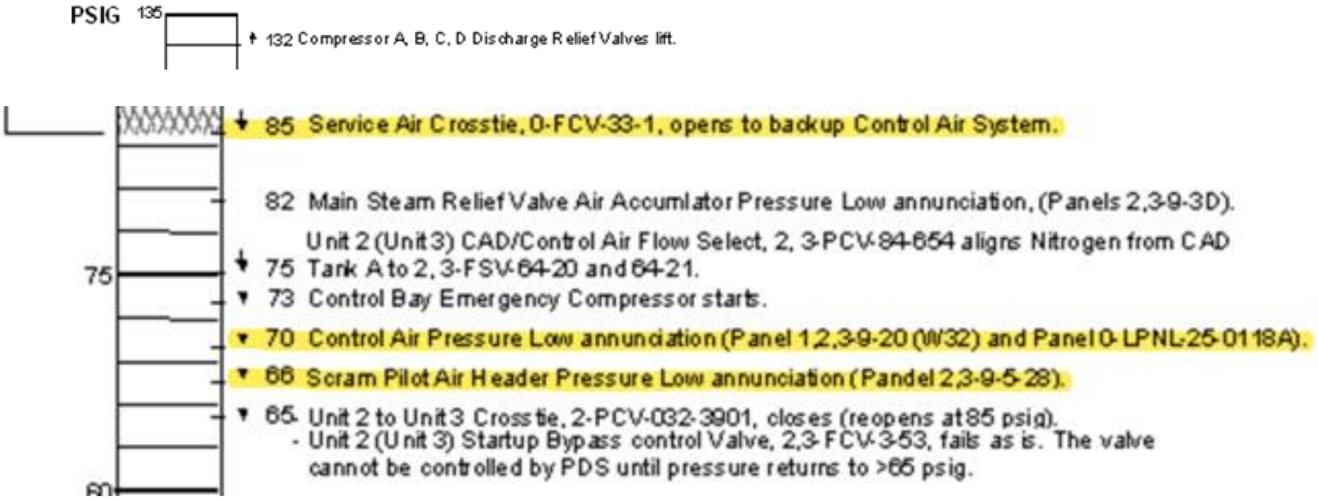
MANUALLY SCRAM the reactor. Refer to 1(2)(3)-AOI-100-1 and 1(2)(3)-AOI-32-2. ☐

Excerpt from 0-OI-32: Illustrates related Control Air System Pressure spectrum

BFN **Unit 0**	**Control Air System**	**0-OI-32** **Rev. 0142** **Page 73 of 117**

Attachment 1
(Page 1 of 1)

Control Air System Pressure Spectrum



132 Compressor A, B, C, D Discharge Relief Valves lift.

85 Service Air Crosstie, 0-FCV-33-1, opens to backup Control Air System.

82 Main Steam Relief Valve Air Accumlator Pressure Low annunciation, (Panels 2,3-9-3D).

75 Unit 2 (Unit 3) CAD/Control Air Flow Select, 2, 3-PCV-84-654 aligns Nitrogen from CAD Tank A to 2, 3-FSV-64-20 and 64-21.

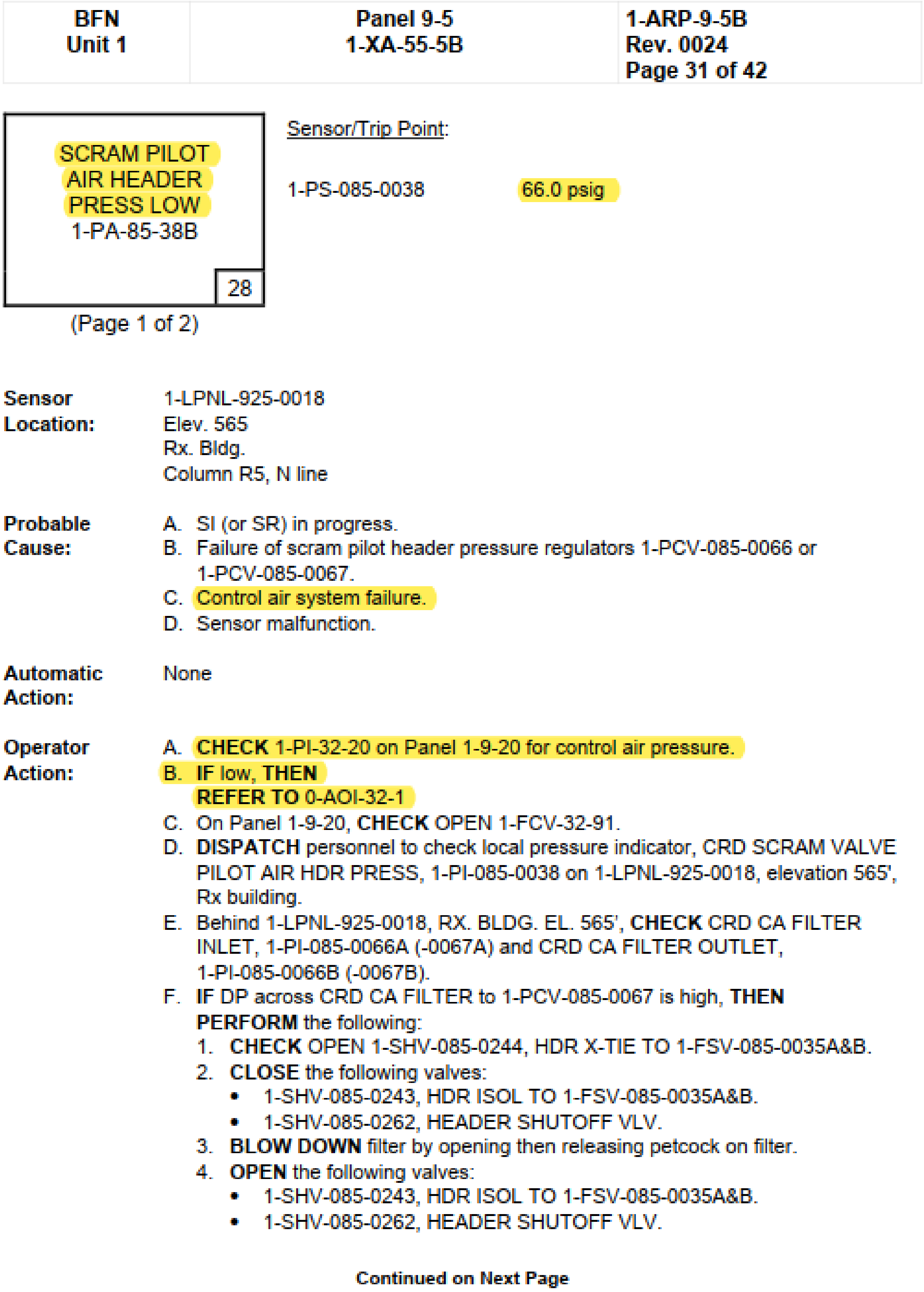
73 Control Bay Emergency Compressor starts.

70 Control Air Pressure Low annunciation (Panel 1,2,3-9-20 (W32) and Panel 0-LPNL-25-0118A).

66 Scram Pilot Air Header Pressure Low annunciation (Pandel 2,3-9-5-28).

65 Unit 2 to Unit 3 Crosstie, 2-PCV-032-3901, closes (reopens at 85 psig).
- Unit 2 (Unit 3) Startup Bypass control Valve, 2,3-FCV-3-53, fails as is. The valve cannot be controlled by PDS until pressure returns to >65 psig.

Excerpt from 1-ARP-9-5B: Supports Distractors B(2), D(2)



BFN **Unit 1**	**Panel 9-5** **1-XA-55-5B**	**1-ARP-9-5B** **Rev. 0024** **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 66.0 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.

Continued on Next Page

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-reference:

205000 (SF4 SCS) Shutdown Cooling
K2.02 (10CFR 55.41.7)
Knowledge of electrical power supplies to the following:

- Motor-Operated valve

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	205000K2.02	
Importance Rating	3.3	------

Proposed Question: **# 22**

Unit 2 is in MODE 4 when the following conditions occur:

- RHR Loop II is in Shutdown Cooling
- 2B and 2D RHR Pumps are in service
- 480V Shutdown Board 2B trips

Given the conditions above, 2-FCV-74-67, RHR SYSTEM II **INBOARD** INJECTION VALVE, ____**(1)**____ **AUTOMATICALL**Y close and the running RHR Pumps are ____**(2)**____ tripped.

A. (1) will
(2) manually

B. (1) will
(2) automatically

C. (1) will **NOT**
(2) manually

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

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the candidate could easily confuse the equipment impacted when 480V Shutdown Board 2B trips. 480V RMOV Board 2B, 2C, or 2E loads are impacted from the board loss. The candidate could believe that 2-FCV-74-67, RHR SYS II INBOARD INJECTION VALVE will automatically close. Second part is incorrect but plausible in that the candidate may not realize that the suction path is lost since 2-FCV-74-47, RHR SHUTDOWN COOLING OUTBOARD SUCTION ISOLATION VALVE closes. Additionally, Core Spray Pumps do not have suction valve interlocks while the RHR Pumps do.

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-OI-74, RHR System, 2-FCV-74-67, RHR SYS II INBOARD INJECTION VALVE, is powered from 480V RMOV Board 2E which is powered from 480V Shutdown Board 2B. The valve motor has lost power. For second part, 2-FCV-74-47, RHR SHUTDOWN COOLING OUTBOARD SUCTION ISOLATION VALVE closes on a loss of RPS 2B. RPS 2B is powered from the 480V RMOV Board 2B, which is powered from 480V Shutdown Board 2B that was given as lost. In turn, the running RHR Loop II Pumps are automatically tripped. 2-FCV-74-47, RHR SHUTDOWN COOLING SUCTION OUTBOARD ISOLATION VALVE is powered from the 250V RMOV Board 2A.

RO Level Justification: Tests the candidate's knowledge of the electrical power supplies to RHR motor operated valves during a power supply loss that support Shutdown Cooling. 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): 2-OI-74, Rev. 189 (Attach if not previously provided)
2-OI-99, Rev. 96
OPL171.044, Rev. 22

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.044 Obj. 3.f (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1510 #29	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2015	

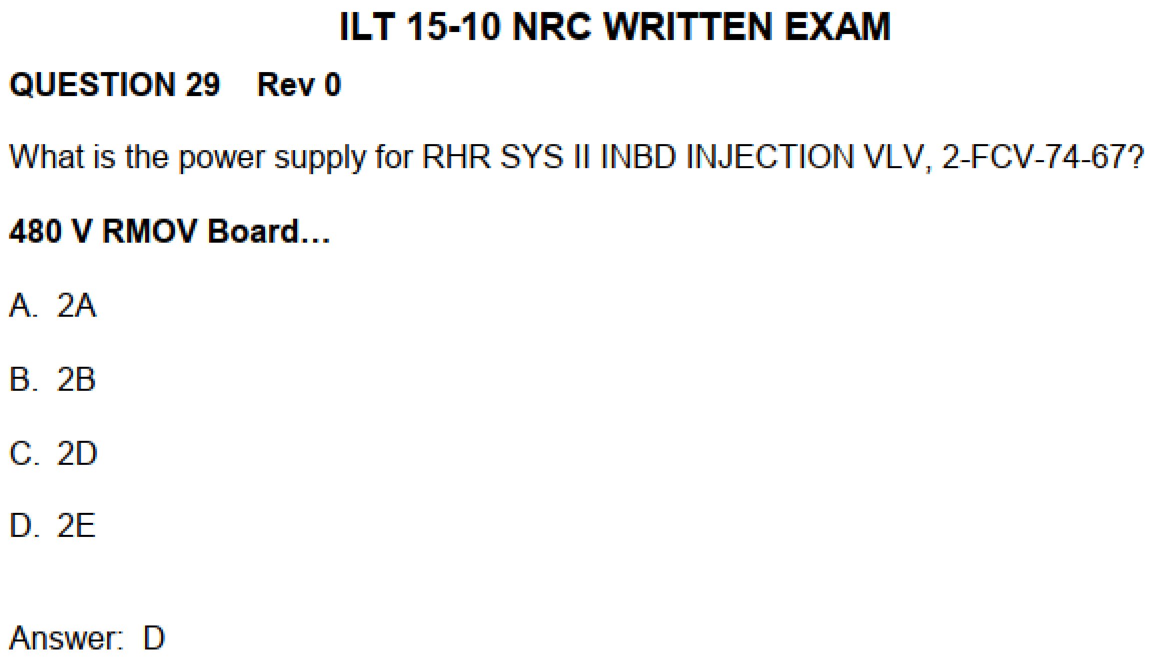
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	**X**

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

Comments:

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

Copy of Bank Question:



ILT 15-10 NRC WRITTEN EXAM

QUESTION 29 Rev 0

What is the power supply for RHR SYS II INBD INJECTION VLV, 2-FCV-74-67?

480 V RMOV Board…

A. 2A

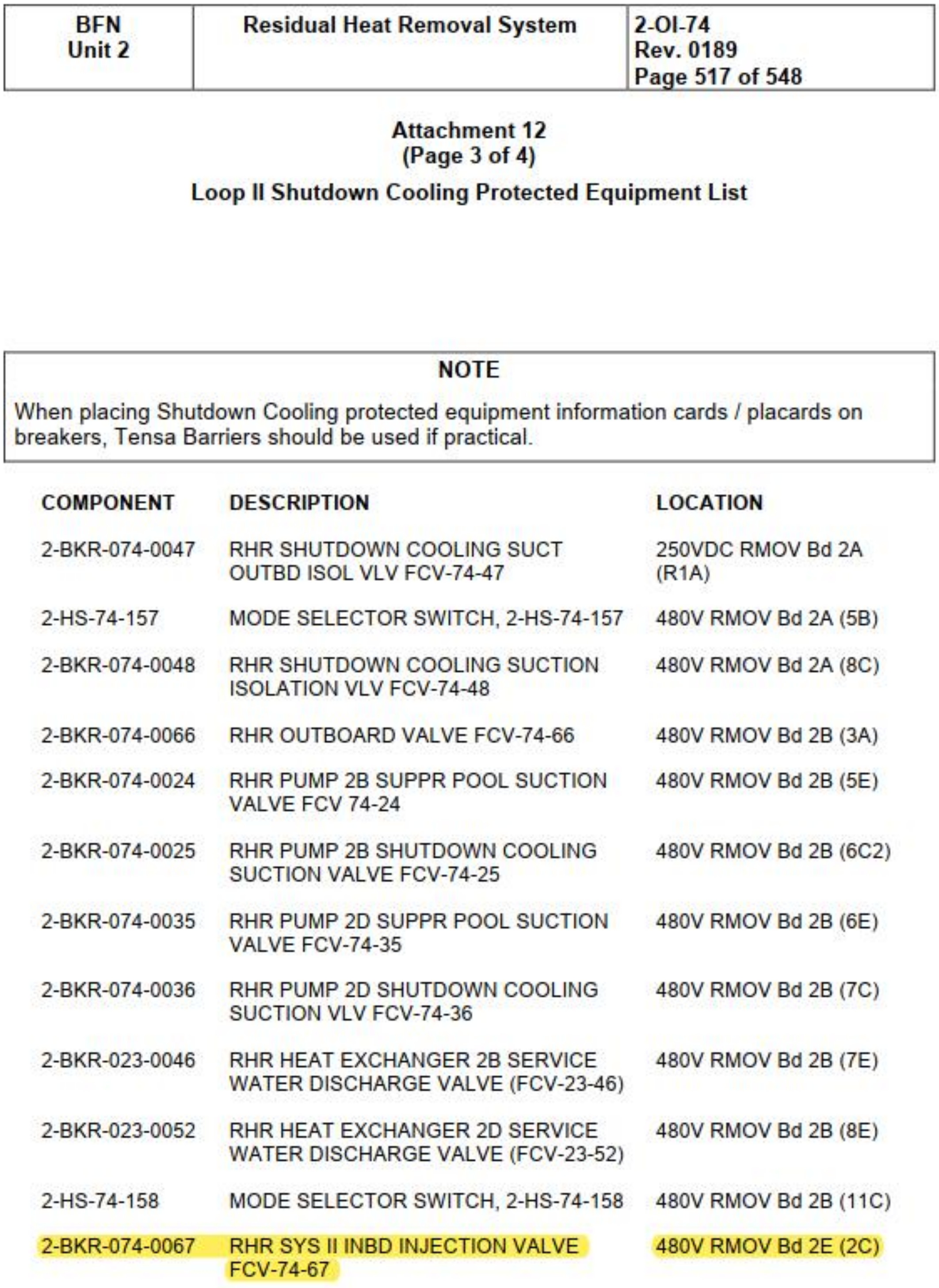
B. 2B

C. 2D

D. 2E

Answer: D

Excerpt from 2-OI-74:



BFN **Unit 2**	**Residual Heat Removal System**	**2-OI-74** **Rev. 0189** **Page 517 of 548**

Attachment 12
(Page 3 of 4)

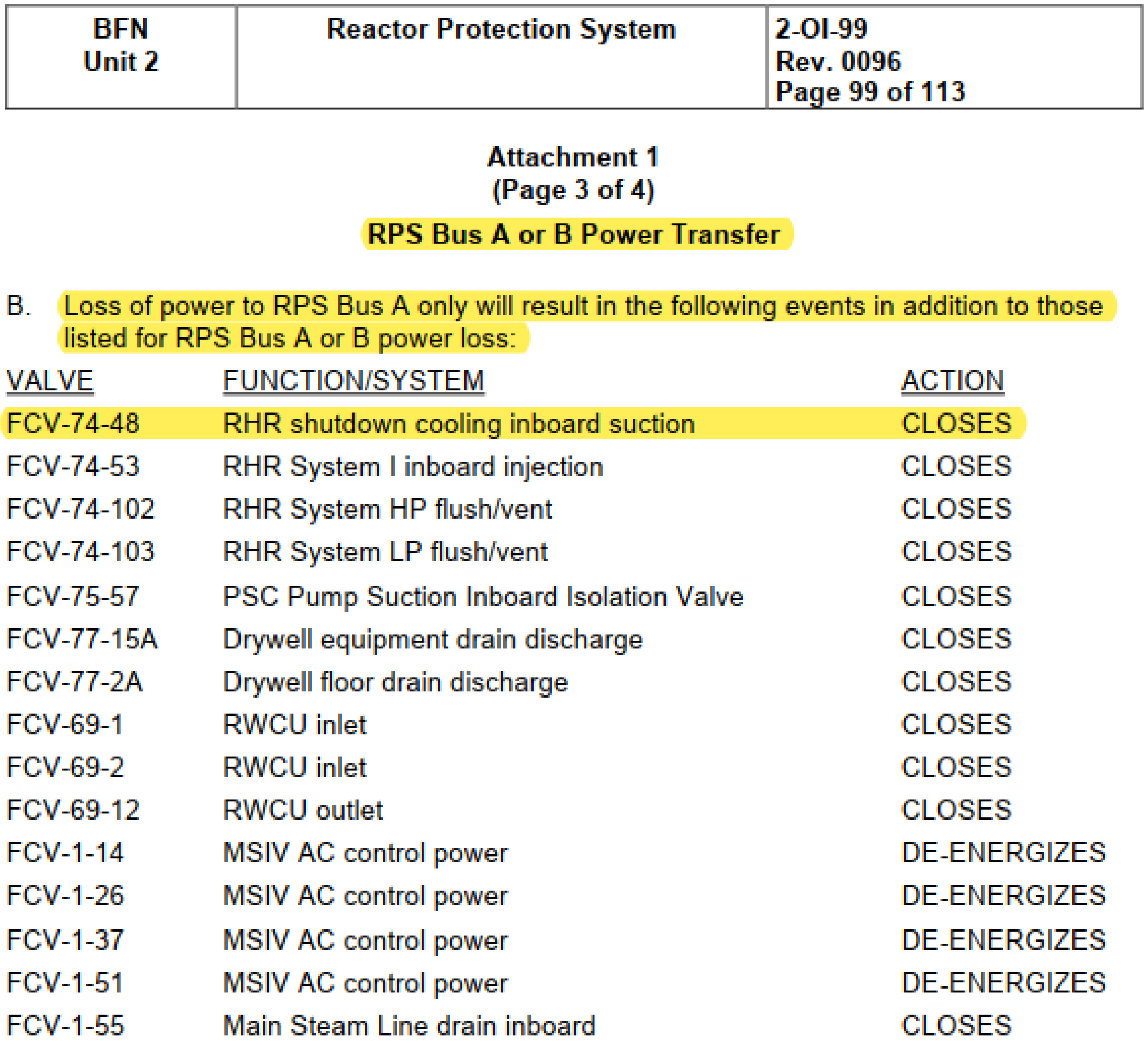
Loop II Shutdown Cooling Protected Equipment List

NOTE

When placing Shutdown Cooling protected equipment information cards / placards on breakers, Tensa Barriers should be used if practical.

COMPONENT	DESCRIPTION	LOCATION
2-BKR-074-0047	RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV FCV-74-47	250VDC RMOV Bd 2A (R1A)
2-HS-74-157	MODE SELECTOR SWITCH, 2-HS-74-157	480V RMOV Bd 2A (5B)
2-BKR-074-0048	RHR SHUTDOWN COOLING SUCTION ISOLATION VLV FCV-74-48	480V RMOV Bd 2A (8C)
2-BKR-074-0066	RHR OUTBOARD VALVE FCV-74-66	480V RMOV Bd 2B (3A)
2-BKR-074-0024	RHR PUMP 2B SUPPR POOL SUCTION VALVE FCV 74-24	480V RMOV Bd 2B (5E)
2-BKR-074-0025	RHR PUMP 2B SHUTDOWN COOLING SUCTION VALVE FCV-74-25	480V RMOV Bd 2B (6C2)
2-BKR-074-0035	RHR PUMP 2D SUPPR POOL SUCTION VALVE FCV-74-35	480V RMOV Bd 2B (6E)
2-BKR-074-0036	RHR PUMP 2D SHUTDOWN COOLING SUCTION VLV FCV-74-36	480V RMOV Bd 2B (7C)
2-BKR-023-0046	RHR HEAT EXCHANGER 2B SERVICE WATER DISCHARGE VALVE (FCV-23-46)	480V RMOV Bd 2B (7E)
2-BKR-023-0052	RHR HEAT EXCHANGER 2D SERVICE WATER DISCHARGE VALVE (FCV-23-52)	480V RMOV Bd 2B (8E)
2-HS-74-158	MODE SELECTOR SWITCH, 2-HS-74-158	480V RMOV Bd 2B (11C)
2-BKR-074-0067	RHR SYS II INBD INJECTION VALVE FCV-74-67	480V RMOV Bd 2E (2C)

Excerpts from 2-OI-99:



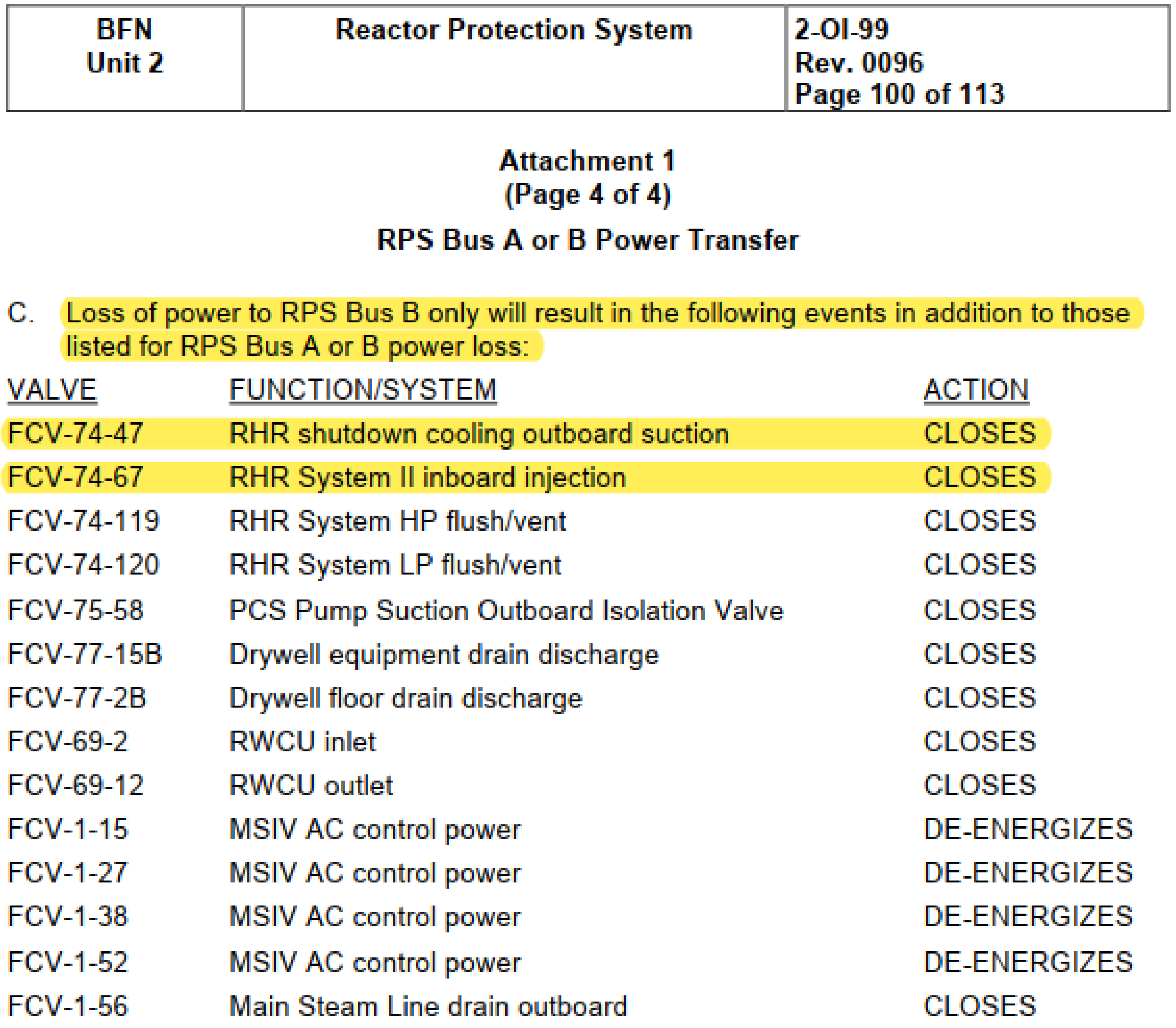
BFN **Unit 2**	**Reactor Protection System**	**2-OI-99** **Rev. 0096** **Page 99 of 113**

Attachment 1
(Page 3 of 4)

RPS Bus A or B Power Transfer

B. Loss of power to RPS Bus A only will result in the following events in addition to those listed for RPS Bus A or B power loss:

VALVE	FUNCTION/SYSTEM	ACTION
FCV-74-48	RHR shutdown cooling inboard suction	CLOSES
FCV-74-53	RHR System I inboard injection	CLOSES
FCV-74-102	RHR System HP flush/vent	CLOSES
FCV-74-103	RHR System LP flush/vent	CLOSES
FCV-75-57	PSC Pump Suction Inboard Isolation Valve	CLOSES
FCV-77-15A	Drywell equipment drain discharge	CLOSES
FCV-77-2A	Drywell floor drain discharge	CLOSES
FCV-69-1	RWCU inlet	CLOSES
FCV-69-2	RWCU inlet	CLOSES
FCV-69-12	RWCU outlet	CLOSES
FCV-1-14	MSIV AC control power	DE-ENERGIZES
FCV-1-26	MSIV AC control power	DE-ENERGIZES
FCV-1-37	MSIV AC control power	DE-ENERGIZES
FCV-1-51	MSIV AC control power	DE-ENERGIZES
FCV-1-55	Main Steam Line drain inboard	CLOSES



BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0096 Page 100 of 113

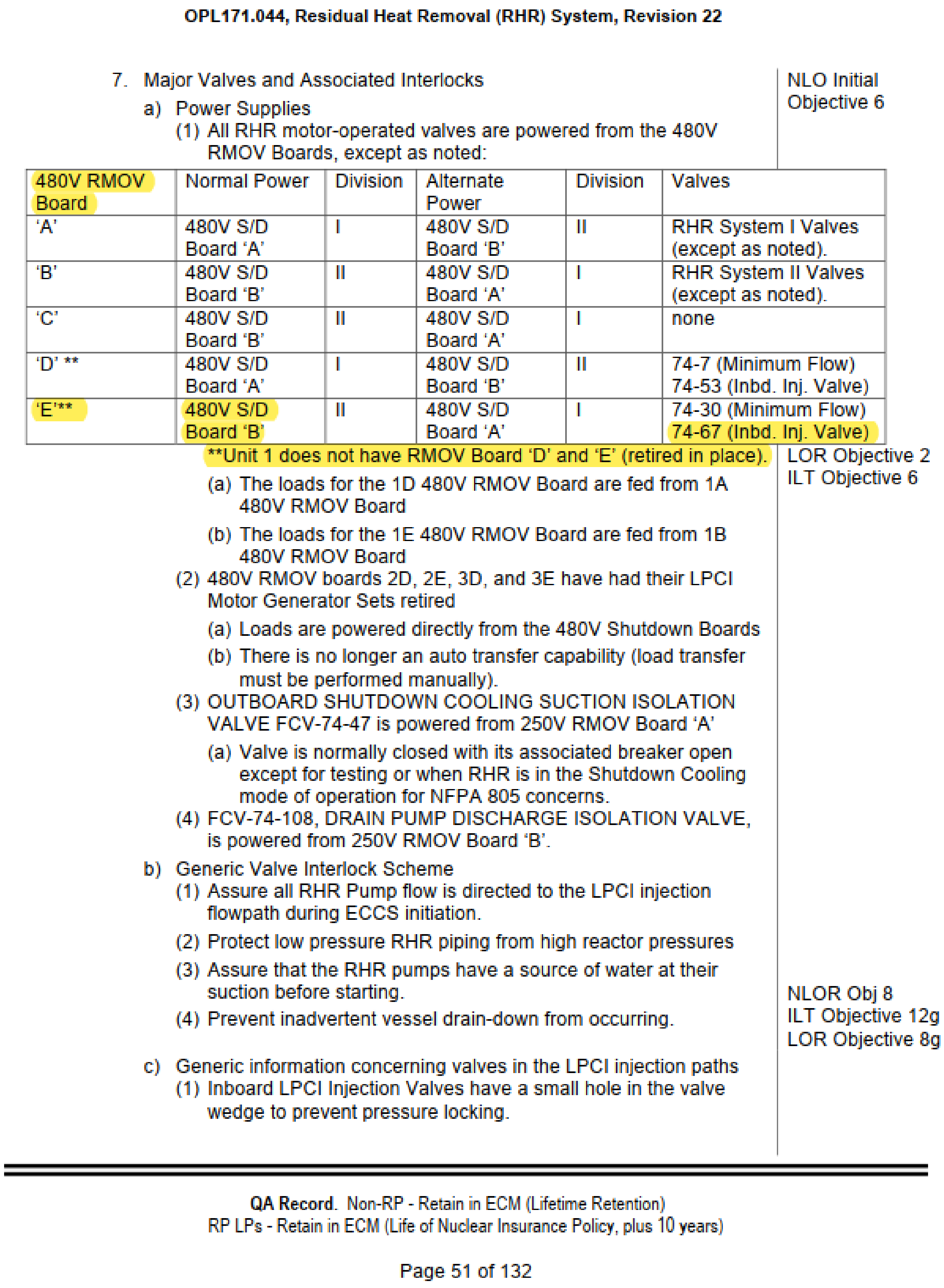
Attachment 1
(Page 4 of 4)

RPS Bus A or B Power Transfer

C. Loss of power to RPS Bus B only will result in the following events in addition to those listed for RPS Bus A or B power loss:

VALVE	FUNCTION/SYSTEM	ACTION
FCV-74-47	RHR shutdown cooling outboard suction	CLOSES
FCV-74-67	RHR System II inboard injection	CLOSES
FCV-74-119	RHR System HP flush/vent	CLOSES
FCV-74-120	RHR System LP flush/vent	CLOSES
FCV-75-58	PCS Pump Suction Outboard Isolation Valve	CLOSES
FCV-77-15B	Drywell equipment drain discharge	CLOSES
FCV-77-2B	Drywell floor drain discharge	CLOSES
FCV-69-2	RWCU inlet	CLOSES
FCV-69-12	RWCU outlet	CLOSES
FCV-1-15	MSIV AC control power	DE-ENERGIZES
FCV-1-27	MSIV AC control power	DE-ENERGIZES
FCV-1-38	MSIV AC control power	DE-ENERGIZES
FCV-1-52	MSIV AC control power	DE-ENERGIZES
FCV-1-56	Main Steam Line drain outboard	CLOSES

Excerpt from OPL171.044: Supports Distractors A(1), B(1)



OPL171.044, Residual Heat Removal (RHR) System, Revision 22

NLO Initial Objective 6

7. Major Valves and Associated Interlocks
   a) Power Supplies
      (1) All RHR motor-operated valves are powered from the 480V RMOV Boards, except as noted:

480V RMOV Board	Normal Power	Division	Alternate Power	Division	Valves
'A'	480V S/D Board 'A'	I	480V S/D Board 'B'	II	RHR System I Valves (except as noted).
'B'	480V S/D Board 'B'	II	480V S/D Board 'A'	I	RHR System II Valves (except as noted).
'C'	480V S/D Board 'B'	II	480V S/D Board 'A'	I	none
'D' **	480V S/D Board 'A'	I	480V S/D Board 'B'	II	74-7 (Minimum Flow) 74-53 (Inbd. Inj. Valve)
'E'**	480V S/D Board 'B'	II	480V S/D Board 'A'	I	74-30 (Minimum Flow) 74-67 (Inbd. Inj. Valve)

**Unit 1 does not have RMOV Board 'D' and 'E' (retired in place).

LOR Objective 2
ILT Objective 6

(a) The loads for the 1D 480V RMOV Board are fed from 1A 480V RMOV Board

(b) The loads for the 1E 480V RMOV Board are fed from 1B 480V RMOV Board

(2) 480V RMOV boards 2D, 2E, 3D, and 3E have had their LPCI Motor Generator Sets retired

(a) Loads are powered directly from the 480V Shutdown Boards

(b) There is no longer an auto transfer capability (load transfer must be performed manually).

(3) OUTBOARD SHUTDOWN COOLING SUCTION ISOLATION VALVE FCV-74-47 is powered from 250V RMOV Board 'A'

(a) Valve is normally closed with its associated breaker open except for testing or when RHR is in the Shutdown Cooling mode of operation for NFPA 805 concerns.

(4) FCV-74-108, DRAIN PUMP DISCHARGE ISOLATION VALVE, is powered from 250V RMOV Board 'B'.

b) Generic Valve Interlock Scheme
   (1) Assure all RHR Pump flow is directed to the LPCI injection flowpath during ECCS initiation.
   (2) Protect low pressure RHR piping from high reactor pressures
   (3) Assure that the RHR pumps have a source of water at their suction before starting.
   (4) Prevent inadvertent vessel drain-down from occurring.

NLOR Obj 8
ILT Objective 12g
LOR Objective 8g

c) Generic information concerning valves in the LPCI injection paths
   (1) Inboard LPCI Injection Valves have a small hole in the valve wedge to prevent pressure locking.

QA Record. Non-RP - Retain in ECM (Lifetime Retention)
RP LPs - Retain in ECM (Life of Nuclear Insurance Policy, plus 10 years)

Examination Outline Cross-reference:	Level	RO	SRO
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation/1 & 4 **AA2.11** (10CFR 55.41.10)	Tier #	1	------
Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Forced Core Flow Circulation:	Group #	1	------
• Individual loop flow(s)	K/A #	295001AA2.11	
	Importance Rating	3.6	------

Proposed Question: **# 23**

Unit 2 was operating at 100% Rated Thermal Power (RTP) when the following conditions occurred:

- Reactor Recirculation Pump 2A tripped
- 2-FI-68-48, JET PUMP FLOW for RECIRC LOOP B indicates less than 40×10^6 lbm/hr

Given the conditions above, in accordance with 2-AOI-68-1, Recirc Pump Trip / Core Flow Decrease, Core Flow must be ___**(1)**___ to ensure accurate indication is provided and the Operator is required to insert Control Rods to less than ___**(2)**___ loadline.

A. (1) lowered
(2) 67%

B. (1) lowered
(2) 74%

C. (1) raised
(2) 67%

D. (1) raised
(2) 74%

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that Recirc Pump operation with one Recirc Pump out of service (2A) and the in-service Jet Pump Flow less than or equal to 41 $\times 10^6$ 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. 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 previous revisions of AOI-68-1A (consolidated into AOI-68-1, current revision) required inserting Control Rods to less than 74% loadline.

C **CORRECT**: (*See attached*) In accordance with 2-AOI-68-1, Recirc Pump Trip / Core Flow Decrease, with one Recirc Pump out of service, the operating Jet Pump Loop (in-service loop 2B) flow must be maintained greater than 41 x 10^6 lbm/hr while 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, the operating Jet Pump Loop 2B is required to be maintained greater than 41 x 10^6 lbm/hr as indicated using 2-FI-68-48, JET PUMP FLOW for RECIRC LOOP **B.**

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 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 required to adjust Core Flow to ensure accurate indication is maintained.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-AOI-68-1, Rev. 38 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.007 Obj. 2 (As available)

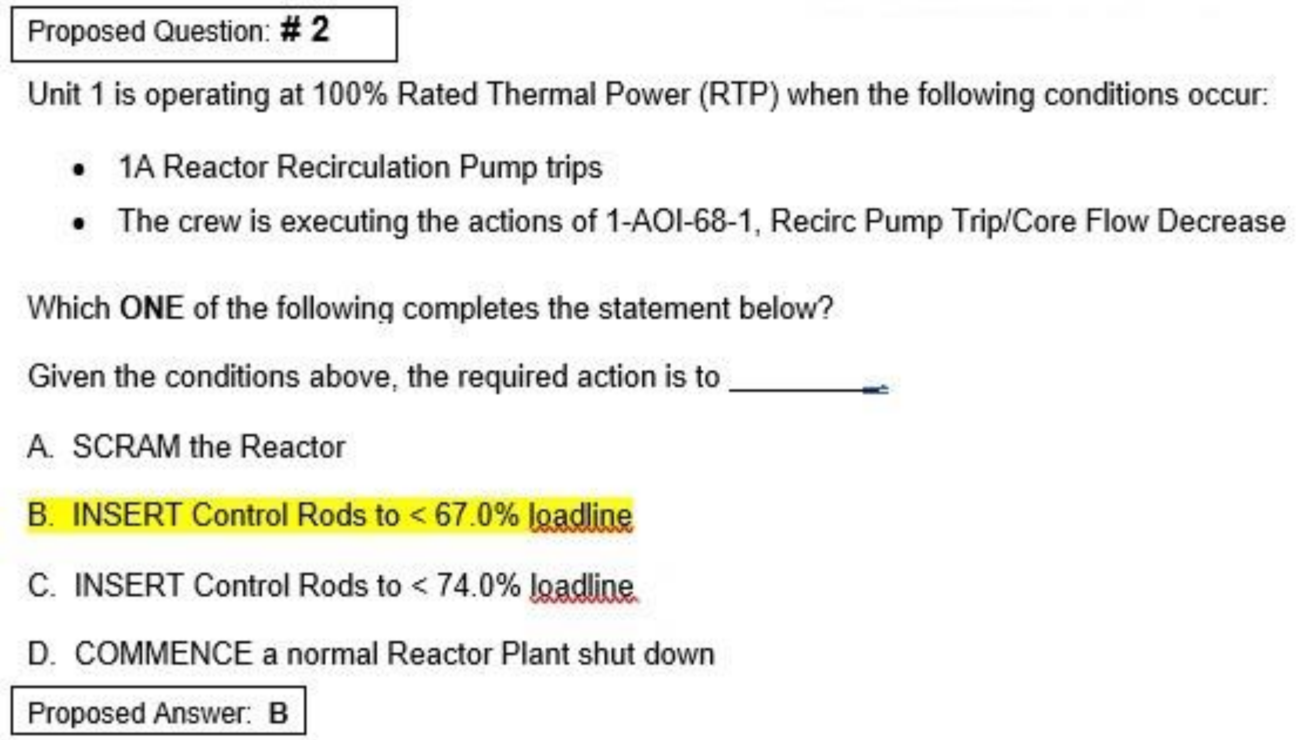
Question Source:	Bank #		
	Modified Bank #	BFN 21-04 #2	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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: **# 2**

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

- 1A Reactor Recirculation Pump trips
- 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 conditions above, the required action is to ________.

A. SCRAM the Reactor

B. INSERT Control Rods to < 67.0% loadline

C. INSERT Control Rods to < 74.0% loadline

D. COMMENCE a normal Reactor Plant shut down

Proposed Answer: **B**

Excerpt from 2-AOI-68-1:

BFN Unit 2	**Recirc Pump Trip/Core Flow Decrease**	**2-AOI-68-1 Rev. 0038 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.

BFN **Unit 2**	**Recirc Pump Trip/Core Flow Decrease**	**2-AOI-68-1** **Rev. 0038** **Page 9 of 14**

4.2 Subsequent Actions (continued)

[2.2] **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 per 0-TI-464, Reactivity Control Plan and 2-OI-85.

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

PERFORM 2-SR-3.3.1.1.I, Core Thermal Hydraulic Stability.

[2.4] [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 2A(2B) discharge valve 2-FCV-068-0003(0079) as necessary to maintain Recirc Loop in thermal equilibrium.

[2.5] **RAISE** core flow to greater than 45%. **REFERENCE** 2-OI-68.

[2.6] **MAINTAIN** operating Recirc pump flow less than 46,600 gpm. **REFERENCE** 2-OI-68.

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

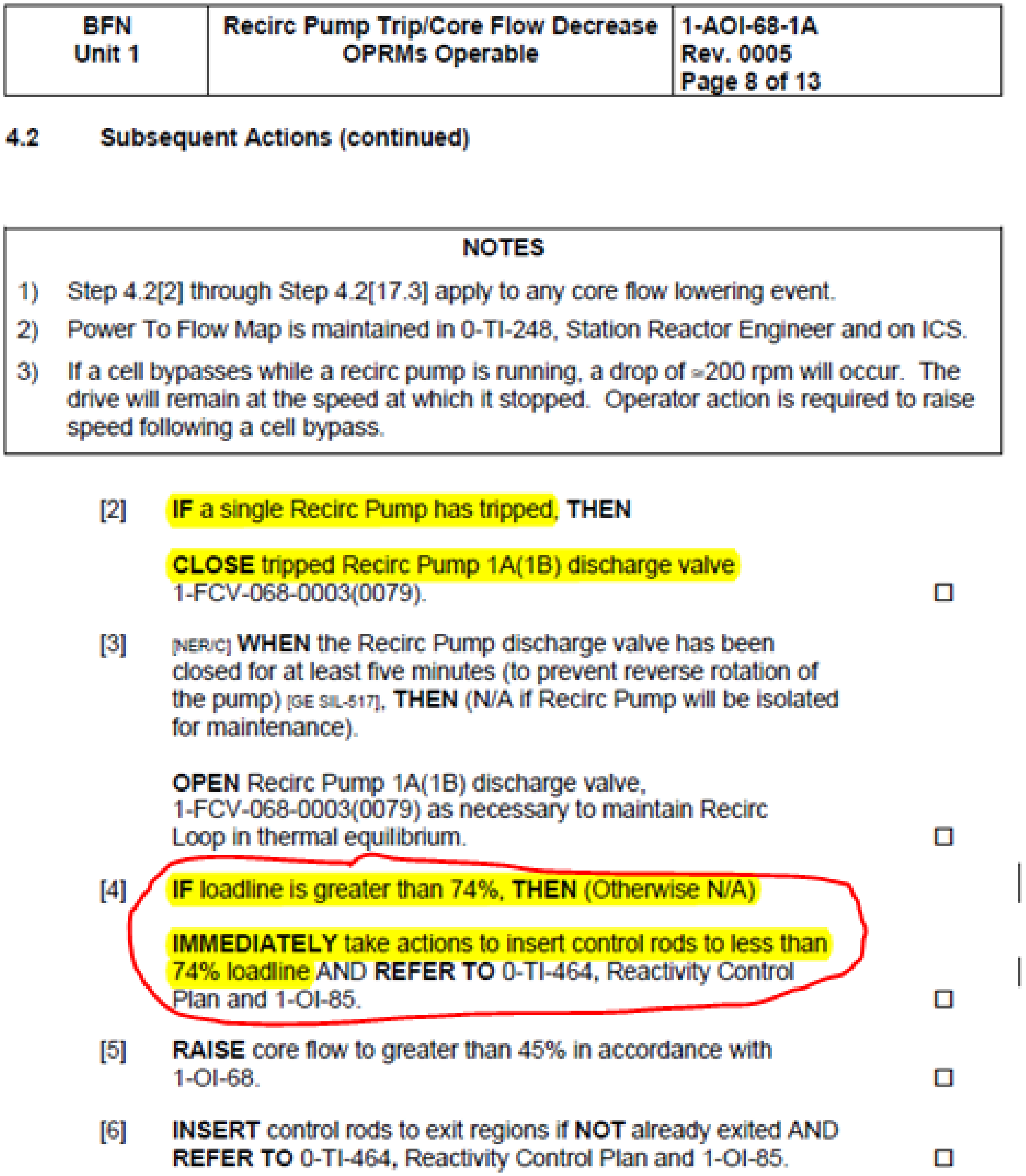
MAINTAIN operating jet pump loop flow greater than 41 x 10^6 lbm/hr (2-FI-68-46 or 2-FI-68-48). [GE SIL 517]

[2.8] **NOTIFY** Reactor Engineer to **PERFORM** the following:

  [2.8.1] 2-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation

  [2.8.2] 0-TI-248, Core Flow Determination in Single Loop Operation

Excerpt from 1-AOI-68-1**A** (previous revision consolidated into current revision of 1-AOI-68-1): supports Distractor B(2), D(2)



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:

295030 (EPE 7) Low Suppression Pool Water Level / 5
EK1.03 (10CFR 55.41.8)

Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Low Suppression Pool Water Level:

- Heat capacity

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295030EK1.03	
Importance Rating	4.0	------

Proposed Question: **# 24**

Unit 1 was operating at 100% RTP when an event occurred resulting in the following conditions:

- Suppression Pool Water Temperature is 150 °F and rising
- Suppression Chamber Pressure is 3 psig and rising
- SRVs are being cycled for Reactor Pressure Control
- Suppression Pool Water Level is 13 feet and lowering

Given the conditions above, the ____**(1)**____will be uncovered **FIRST**.

____**(2)**____ will be more effective at mitigating the **CURRENT** Heat Capacity limitations of Primary Containment.

A. **(1)** HPCI Turbine exhaust
(2) Suppression Chamber Sprays

B. **(1)** HPCI Turbine exhaust
(2) Suppression Pool Cooling

C. **(1)** Downcomer opening
(2) Suppression Chamber Sprays

D. **(1)** Downcomer opening
(2) Suppression Pool Cooling

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). The second part is incorrect but plausible if the candidate misinterprets the given parameters to believe Spraying the Suppression Chamber would be a more effective way to mitigate this condition. Suppression Chamber Sprays is required to be perform prior to reaching 12 psig in the Suppression Chamber.
Under normal LOCA circumstances, Spraying the Suppression Chamber has no benefit.

B **CORRECT:** *(See attached)* In accordance with EOIPM Section 0-V(G) EOI-2, Primary Containment Control Bases, for override SP/L-15, HPCI is required to be locked out by 12.75 feet Suppression Pool Water Level unless it's needed for Adequate Core Cooling. HPCI Turbine exhaust normally discharges under the Suppression Pool Water Level, where the suppression function condenses the steam and prevents pressurizing the Suppression Chamber itself. For second part, given that Suppression Pool Water Temperature is well above the 95 °F EOI-2 entry condition, action must be taken to mitigate the current Heat Capacity limitations. The effects of a lowering Suppression Pool Water Level on heat capacity are mitigated by the use of Suppression Pool Cooling which will be more effective at lowering Suppression Pool Water Temperature to ensure Curve 3, Heat Capacity Temperature Limit is not violated.

C INCORRECT: First part is incorrect but plausible if the candidate misapplies the key Suppression Pool Water Level at 11.5 feet which corresponds to the opening of the Downcomer. The openings of the 96 Downcomer pipes extend from the vent ring header down into the Suppression Pool. Suppression Pool Water Level must be maintained above the elevation of the Downcomer vent openings to ensure steam discharged from the Drywell in the Suppression Pool following a Primary System break will be adequately condensed. 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 operational implications on Primary Containment heat capacity as it relates to the most effective mitigation strategy for given degrading parameters. 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. The candidate must read the multiple indications correctly, integrate them together, and recall the specific facts associated with each parameter to reach a conclusion.

Technical Reference(s): 1-EOI-2, Rev. 8 (Attach if not previously provided)
EOIPM 0-V(G), Rev. 1
0-TI-394, Att. 7, Rev. 13

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.203 Obj. 13 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN NRC 2104 #22	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

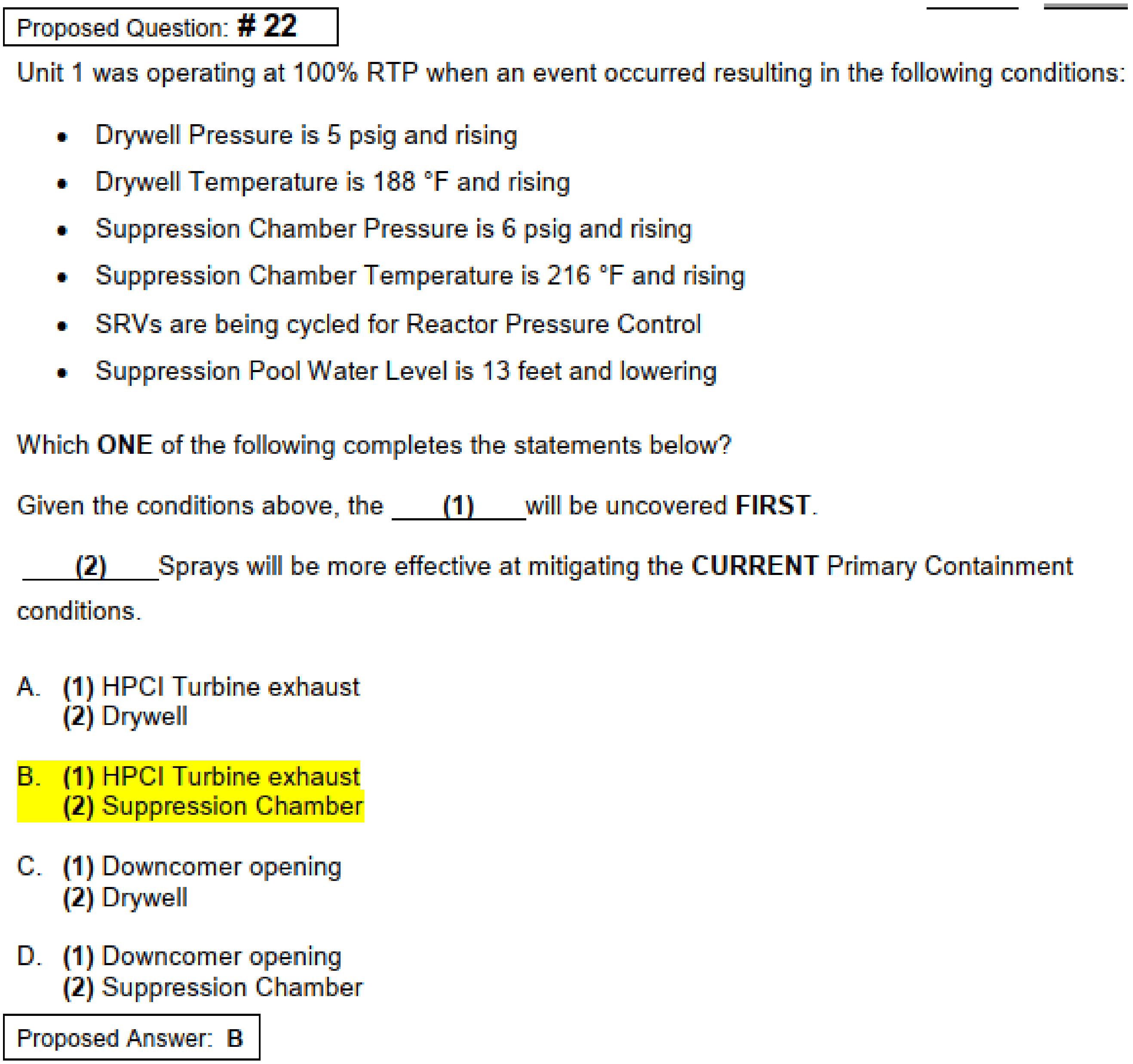
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

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

55.43

Comments:

Copy of Bank Question:



Proposed Question: **# 22**

Unit 1 was operating at 100% RTP when an event occurred resulting in the following conditions:

- Drywell Pressure is 5 psig and rising
- Drywell Temperature is 188 °F and rising
- Suppression Chamber Pressure is 6 psig and rising
- Suppression Chamber Temperature is 216 °F and rising
- SRVs are being cycled for Reactor Pressure Control
- Suppression Pool Water Level is 13 feet and lowering

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

Given the conditions above, the ____**(1)**____ will be uncovered **FIRST**.

____**(2)**____ Sprays will be more effective at mitigating the **CURRENT** Primary Containment conditions.

A. **(1)** HPCI Turbine exhaust
(2) Drywell

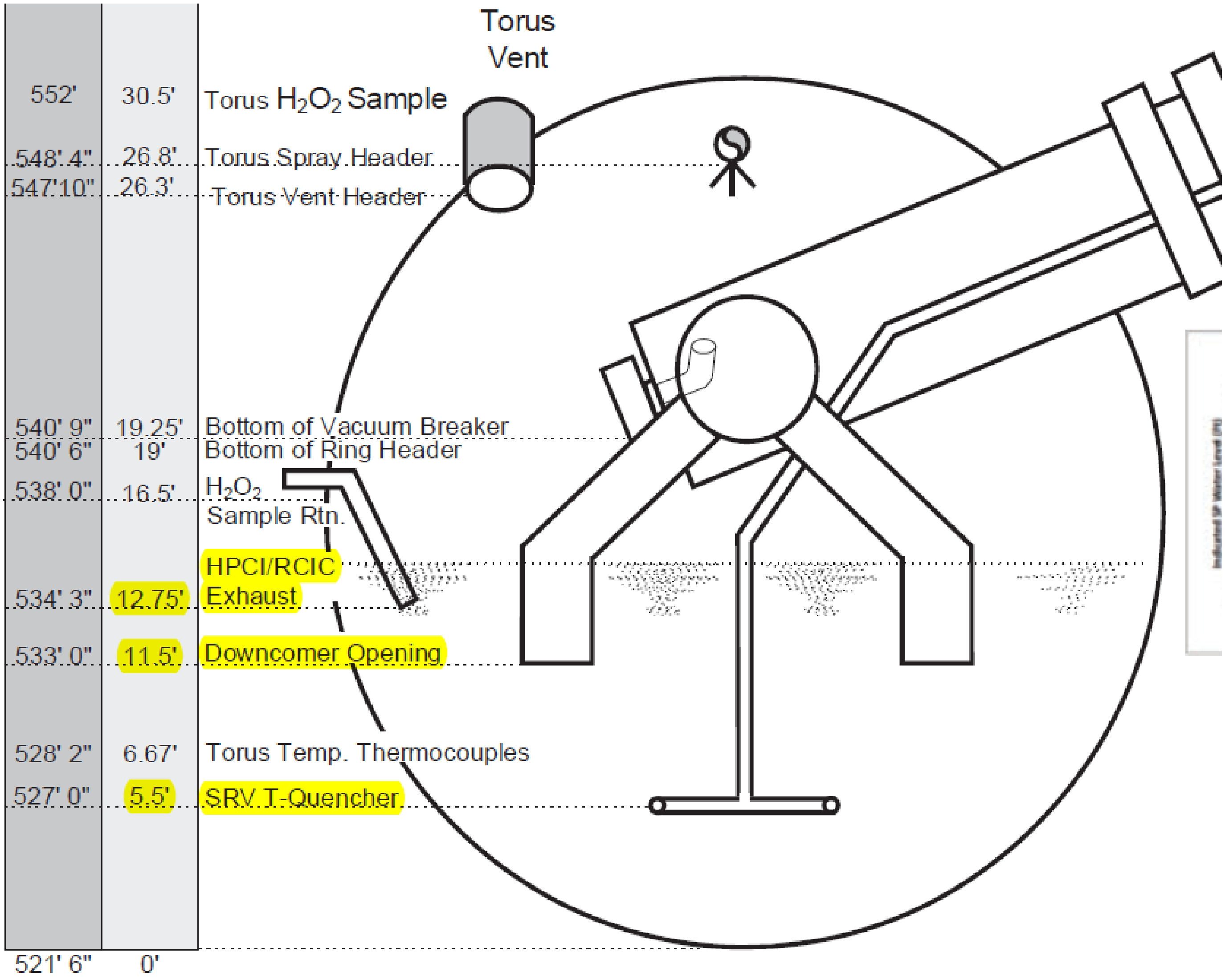
B. **(1)** HPCI Turbine exhaust
(2) Suppression Chamber

C. **(1)** Downcomer opening
(2) Drywell

D. **(1)** Downcomer opening
(2) Suppression Chamber

Proposed Answer: **B**

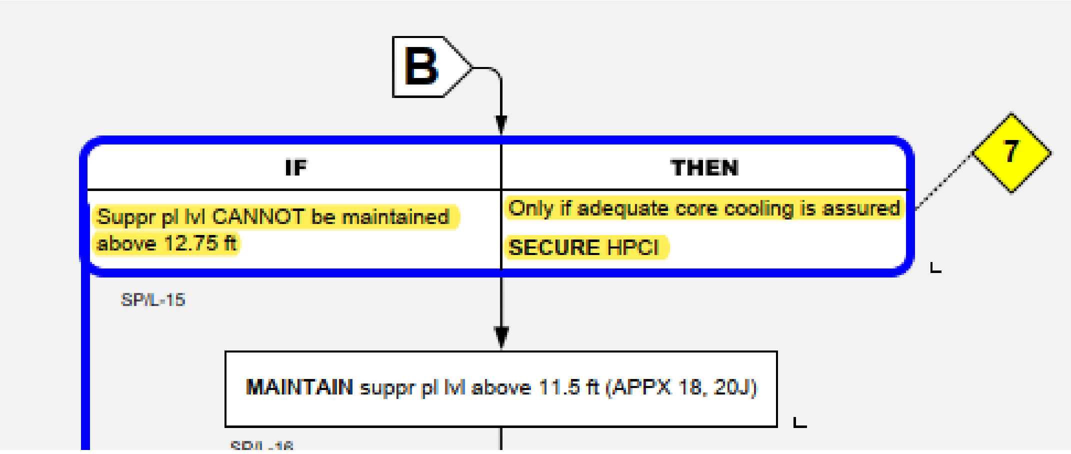
Excerpt from 0-TI-394, Att. 7: (referencing related Suppression Pool Levels)



Severe Accident Management Guidelines	
Technical Support Guidelines	
BROWNS FERRY NUCLEAR PLANT	
0-TI-394 Attachment 7	Rev 13

Excerpts from EOIPM Section 0-V(G):

- Supports A(1), B(1)



BFN **Unit 0**	**EOI-2, Primary Containment Control Bases**	**EOIPM Section 0-V(G)** **Rev. 0001** **Page 123 of 129**

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: SP/L-15

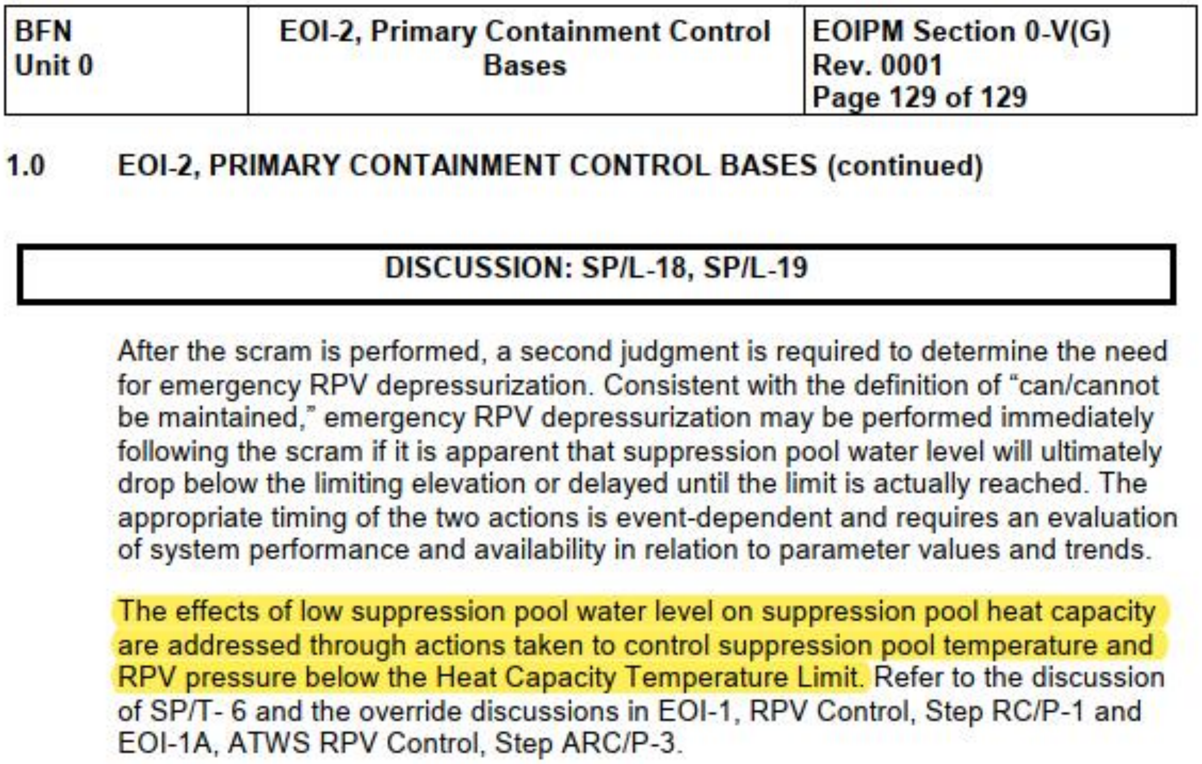
This retainment override step applies to the remaining steps of this flowpath.

Subsequent steps direct control of suppression pool water level relative to the elevation of the downcomer openings (**A.58**). While doing so, suppression pool water level could drop below the elevation of the HPCI turbine exhaust.

Operation of the HPCI turbine with its exhaust unsubmerged will tend to directly pressurize the suppression chamber. If suppression pool water level cannot be maintained above the elevation of the top of the HPCI exhaust, HPCI is therefore secured if not needed for core cooling.

If HPCI injection *is* needed for adequate core cooling, continued operation is permitted even with the turbine exhaust uncovered since:

- Supports B(2), D(2)



BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V(G) Rev. 0001 Page 129 of 129

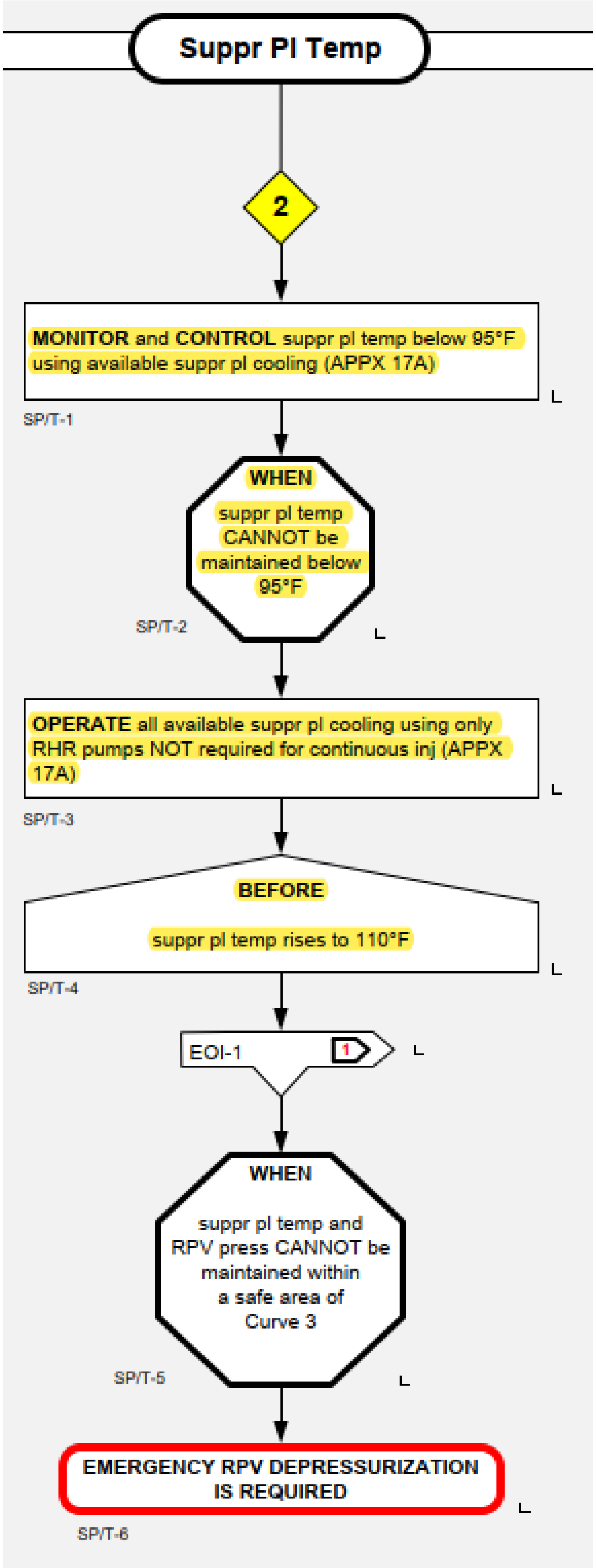
1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: SP/L-18, SP/L-19

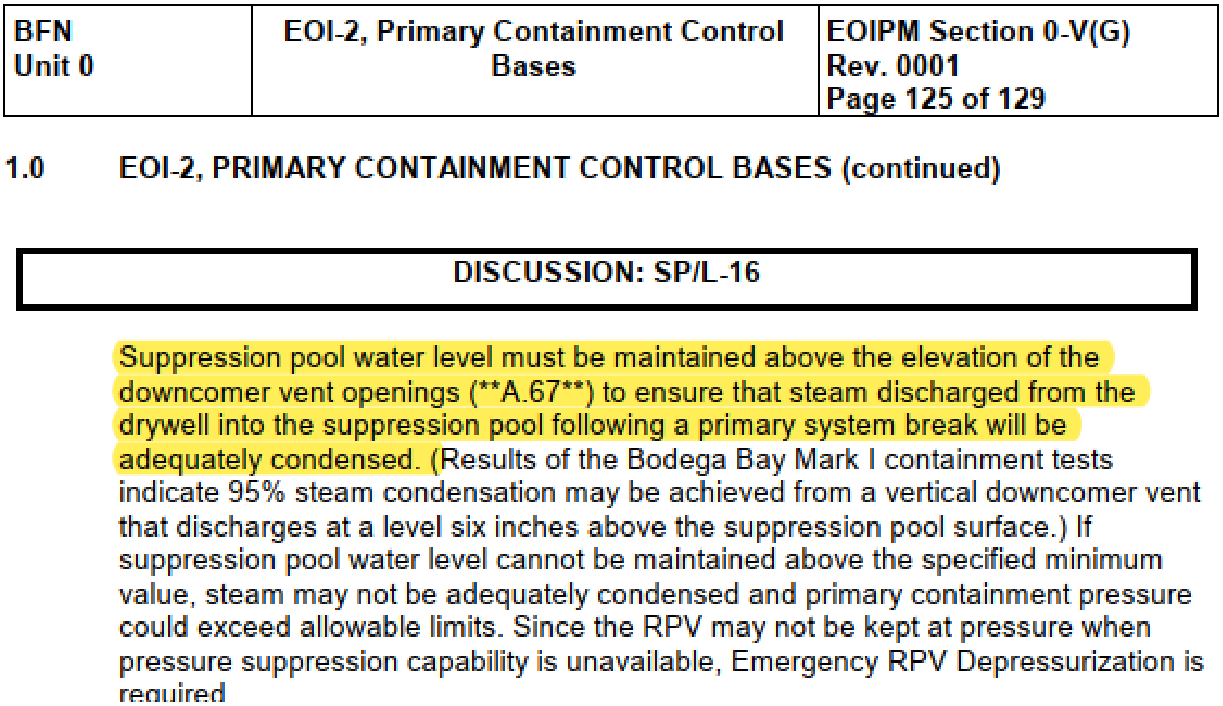
After the scram is performed, a second judgment is required to determine the need for emergency RPV depressurization. Consistent with the definition of "can/cannot be maintained," emergency RPV depressurization may be performed immediately following the scram if it is apparent that suppression pool water level will ultimately drop below the limiting elevation or delayed until the limit is actually reached. The appropriate timing of the two actions is event-dependent and requires an evaluation of system performance and availability in relation to parameter values and trends.

The effects of low suppression pool water level on suppression pool heat capacity are addressed through actions taken to control suppression pool temperature and RPV pressure below the Heat Capacity Temperature Limit. Refer to the discussion of SP/T- 6 and the override discussions in EOI-1, RPV Control, Step RC/P-1 and EOI-1A, ATWS RPV Control, Step ARC/P-3.

Excerpt from EOI-2: Supports B(2), D(2)



- Supports Distractors C(1), D(1) for Suppression Pool Level at 11.5 feet (downcomer openings)



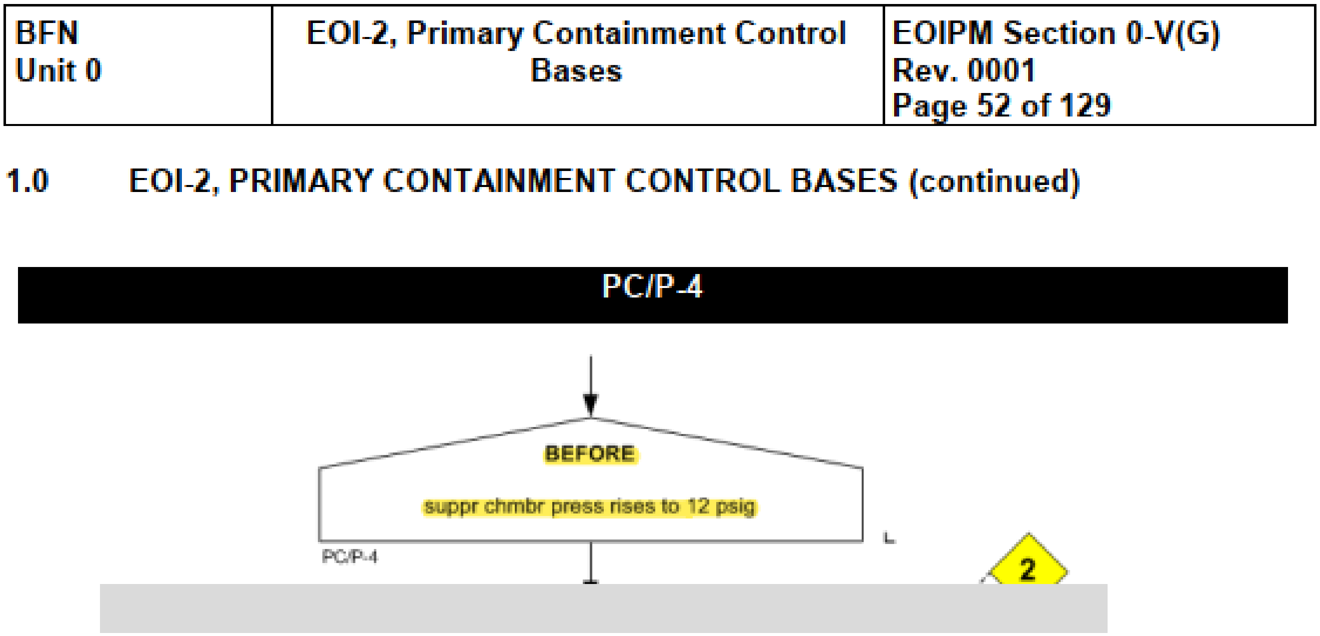
BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V(G) Rev. 0001 Page 125 of 129

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: SP/L-16

Suppression pool water level must be maintained above the elevation of the downcomer vent openings (**A.67**) to ensure that steam discharged from the drywell into the suppression pool following a primary system break will be adequately condensed. (Results of the Bodega Bay Mark I containment tests indicate 95% steam condensation may be achieved from a vertical downcomer vent that discharges at a level six inches above the suppression pool surface.) If suppression pool water level cannot be maintained above the specified minimum value, steam may not be adequately condensed and primary containment pressure could exceed allowable limits. Since the RPV may not be kept at pressure when pressure suppression capability is unavailable, Emergency RPV Depressurization is required

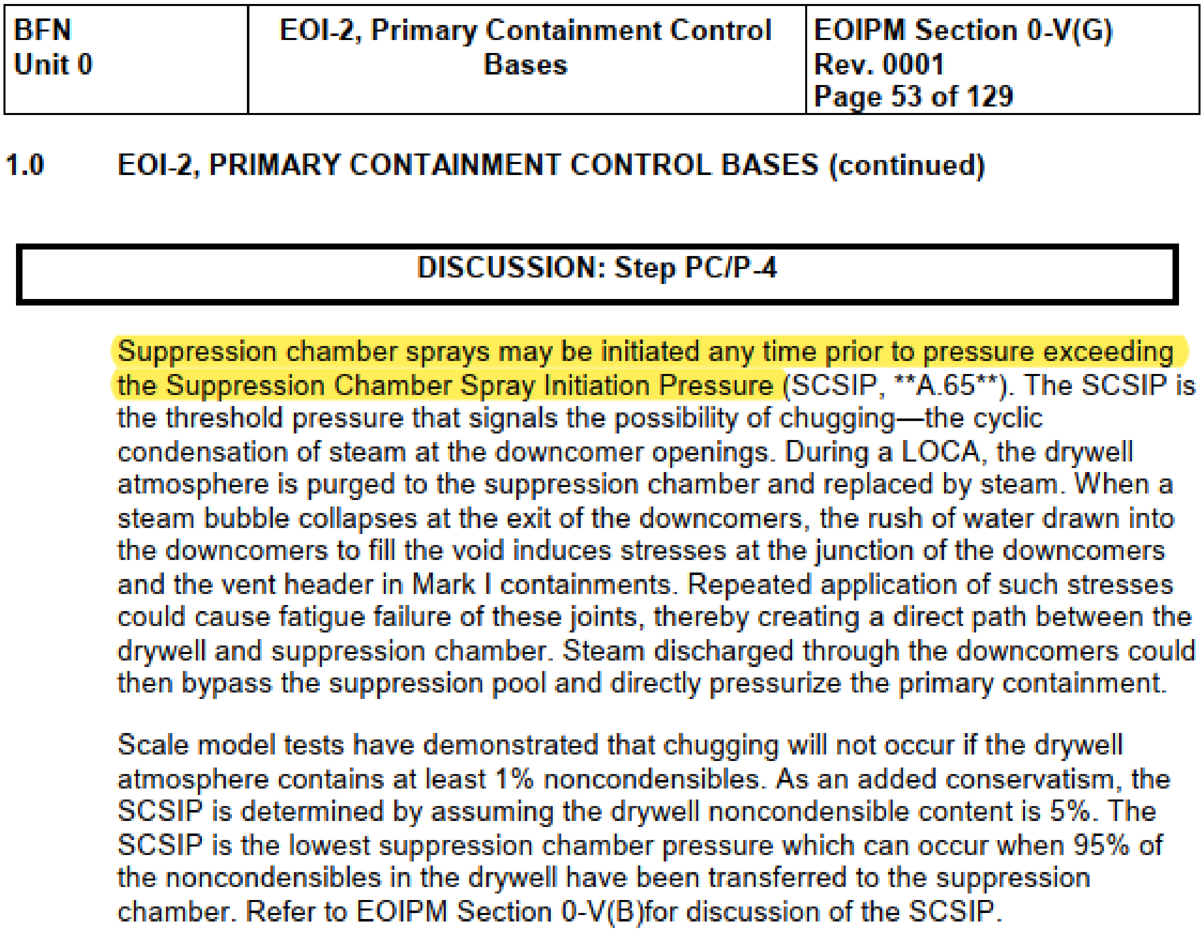
- Supports Distractors A(2), C(2) for Suppression Chamber Sprays



BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V(G) Rev. 0001 Page 52 of 129

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

PC/P-4



BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V(G) Rev. 0001 Page 53 of 129

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: Step PC/P-4

Suppression chamber sprays may be initiated any time prior to pressure exceeding the Suppression Chamber Spray Initiation Pressure (SCSIP, **A.65**). The SCSIP is the threshold pressure that signals the possibility of chugging—the cyclic condensation of steam at the downcomer openings. During a LOCA, the drywell atmosphere is purged to the suppression chamber and replaced by steam. When a steam bubble collapses at the exit of the downcomers, the rush of water drawn into the downcomers to fill the void induces stresses at the junction of the downcomers and the vent header in Mark I containments. Repeated application of such stresses could cause fatigue failure of these joints, thereby creating a direct path between the drywell and suppression chamber. Steam discharged through the downcomers could then bypass the suppression pool and directly pressurize the primary containment.

Scale model tests have demonstrated that chugging will not occur if the drywell atmosphere contains at least 1% noncondensibles. As an added conservatism, the SCSIP is determined by assuming the drywell noncondensible content is 5%. The SCSIP is the lowest suppression chamber pressure which can occur when 95% of the noncondensibles in the drywell have been transferred to the suppression chamber. Refer to EOIPM Section 0-V(B)for discussion of the SCSIP.

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

Examination Outline Cross-reference:

203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode

K1.16 (10CFR 55.41.7)

Knowledge of the physical connections and/or cause and effect relationships between the RHR/LPCI Injection Mode and the following systems:

- Component cooling water systems

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	203000 K1.16	
Importance Rating	2.7	------

Proposed Question: **# 25**

To place Suppression Pool Cooling in service, which **ONE** of the following correctly identifies which RHRSW Pumps are available to provide cooling for RHR Heat Exchangers?

RHRSW Pumps ___**(1)**___ supply cooling water to RHR Loop ___**(2)**___ heat exchangers.

A. (1) B1 and B2 **ONLY**
(2) II **ONLY**

B. (1) B1 and B2 **ONLY**
(2) I and II

C. (1) B1, B2 and D1, D2
(2) II **ONLY**

D. (1) B1, B2 and D1, D2
(2) I and II

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that RHRSW Pumps B1 and B2 do supply cooling to RHR loop II, however the only portion is incorrect. The EECW/RHRSW Pump alignments are easily confused especially since they serve all 3 Units. Twelve total RHRSW Pumps exist, with A3, B3, C3, D3 designated as primary EECW Pumps while A1, B1, C1, D1 serve as EECW backup. That leaves A1, A2, B1, B2, C1, C2, D1, D2 as RHRSW Pumps dedicated for RHR services. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that B or D RHRSW Pumps can supply RHR Loop I and II during Standby Coolant mode. The Standby Coolant mode and Suppression Pool Cooling mode of RHRSW and RHR Pumps are often confused between the Units. The Standby Coolant mode of RHR is Unit and Loop specific. D1 and D2 RHRSW Pumps serve Unit 1 RHR Loop II, Unit 2 Loop I D1 and D2 where Loop II is B1 and B2 RHRSW Pumps. Unit 3 is RHR Loop I supplied by B1, B2 RHRSW Pumps.

C **CORRECT:** *(See attached)* In accordance with 2-OI-74, RHR System, RHRSW Pumps B1, B2, D1, and D2 all can provide Suppression Pool Cooling to Loop II RHR heat exchangers for all 3 Units. For second part, B1, B2, D1, and D2 RHRSW Pumps only provide cooling to RHR Loop II heat exchangers.

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 component cooling water alignments for RHRSW as it relates to RHR heat exchangers. This question is rated as Memory due to the requirement to strictly recall operational facts supporting RHR /LPCI injection mode cooling.

Technical Reference(s): 2-OI-74, Rev. 189 (Attach if not previously provided)
0-OI-23, Rev. 109
OPL171.044, Rev. 22
OPL171.046, Rev. 19U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.044 Obj.2d (As available)

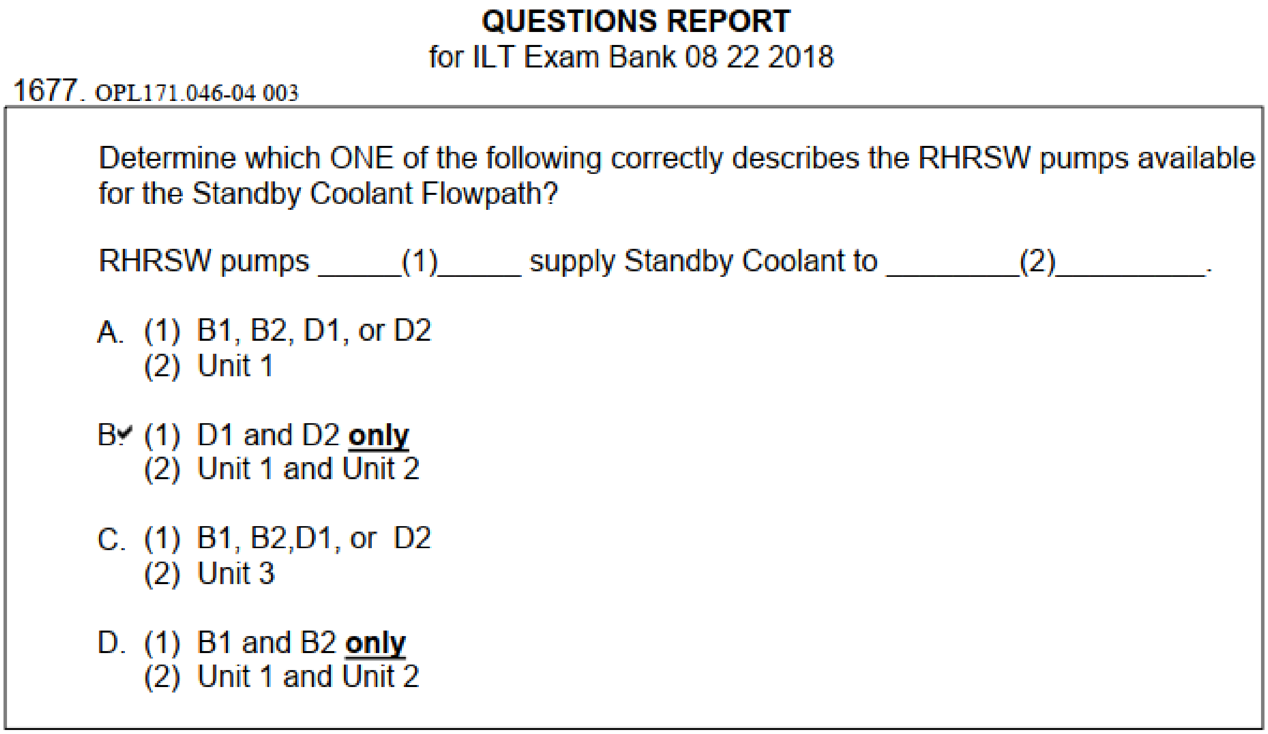
Question Source:	Bank #		
	Modified Bank #	OPL171.046-04 002 #1677	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

1677. OPL171.046-04 003

Determine which ONE of the following correctly describes the RHRSW pumps available for the Standby Coolant Flowpath?

RHRSW pumps ______(1)______ supply Standby Coolant to _________(2)__________.

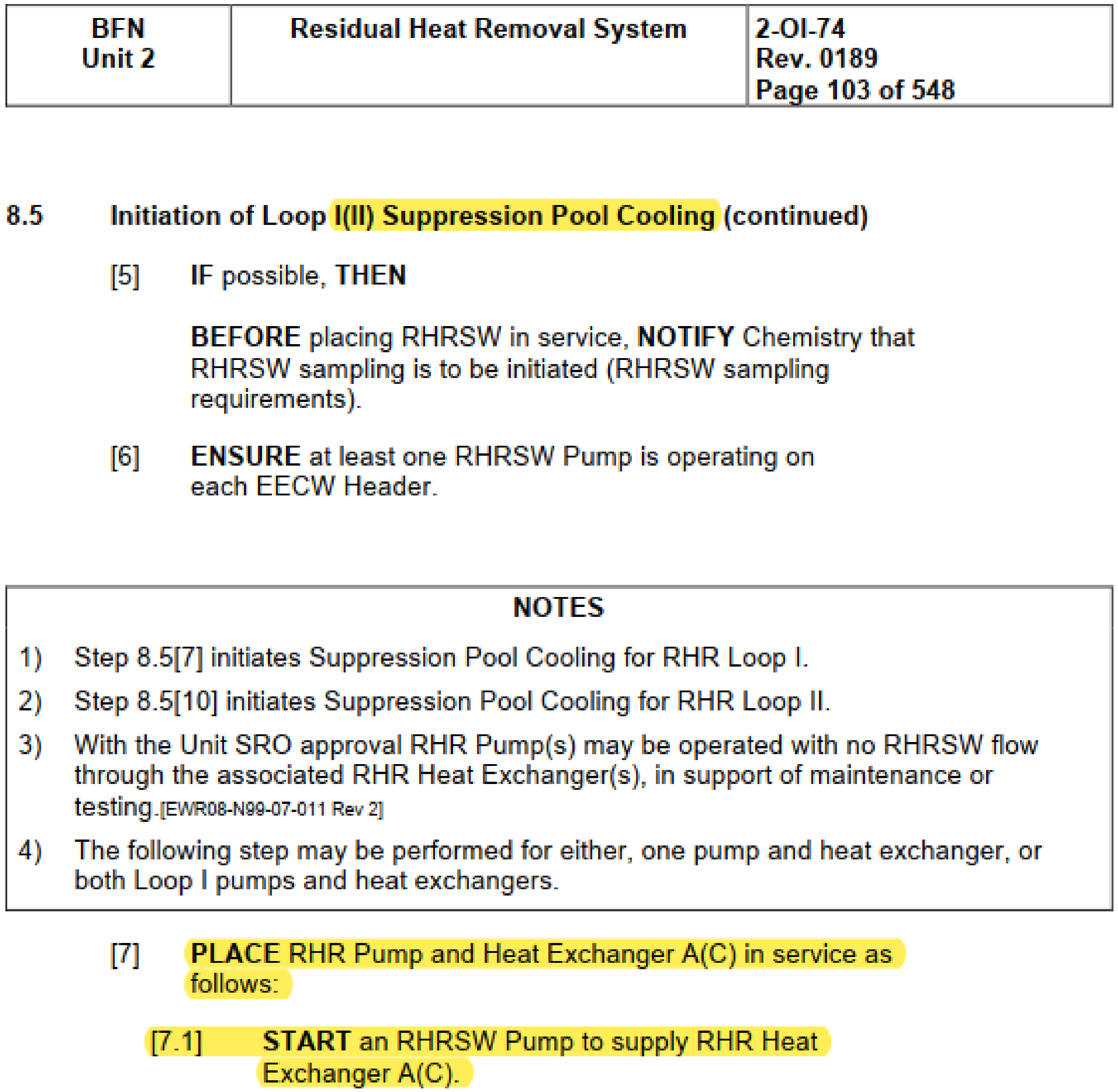
A. (1) B1, B2, D1, or D2
(2) Unit 1

B✓ (1) D1 and D2 **only**
(2) Unit 1 and Unit 2

C. (1) B1, B2,D1, or D2
(2) Unit 3

D. (1) B1 and B2 **only**
(2) Unit 1 and Unit 2

Excerpts from 2-OI-74:



BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0189 Page 103 of 548

8.5 Initiation of Loop I(II) Suppression Pool Cooling (continued)

[5] **IF** possible, **THEN**

BEFORE placing RHRSW in service, **NOTIFY** Chemistry that RHRSW sampling is to be initiated (RHRSW sampling requirements).

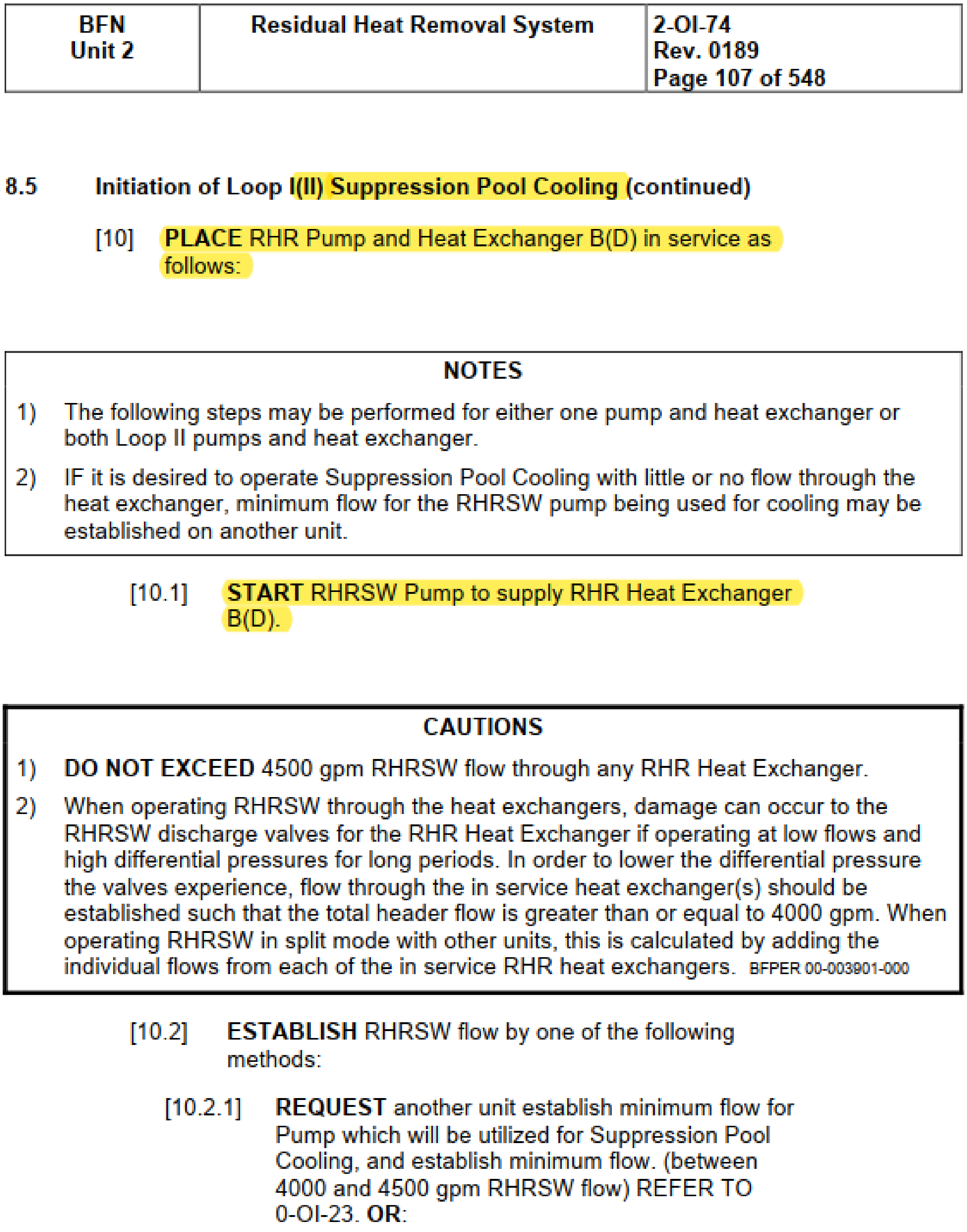
[6] **ENSURE** at least one RHRSW Pump is operating on each EECW Header.

NOTES

1) Step 8.5[7] initiates Suppression Pool Cooling for RHR Loop I.
2) Step 8.5[10] initiates Suppression Pool Cooling for RHR Loop II.
3) With the Unit SRO approval RHR Pump(s) may be operated with no RHRSW flow through the associated RHR Heat Exchanger(s), in support of maintenance or testing.[EWR08-N99-07-011 Rev 2]
4) The following step may be performed for either, one pump and heat exchanger, or both Loop I pumps and heat exchangers.

[7] **PLACE** RHR Pump and Heat Exchanger A(C) in service as follows:

[7.1] **START** an RHRSW Pump to supply RHR Heat Exchanger A(C).



BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0189 Page 107 of 548

8.5 Initiation of Loop I(II) Suppression Pool Cooling (continued)

[10] **PLACE** RHR Pump and Heat Exchanger B(D) in service as follows:

NOTES

1) The following steps may be performed for either one pump and heat exchanger or both Loop II pumps and heat exchanger.

2) IF it is desired to operate Suppression Pool Cooling with little or no flow through the heat exchanger, minimum flow for the RHRSW pump being used for cooling may be established on another unit.

[10.1] **START** RHRSW Pump to supply RHR Heat Exchanger B(D).

CAUTIONS

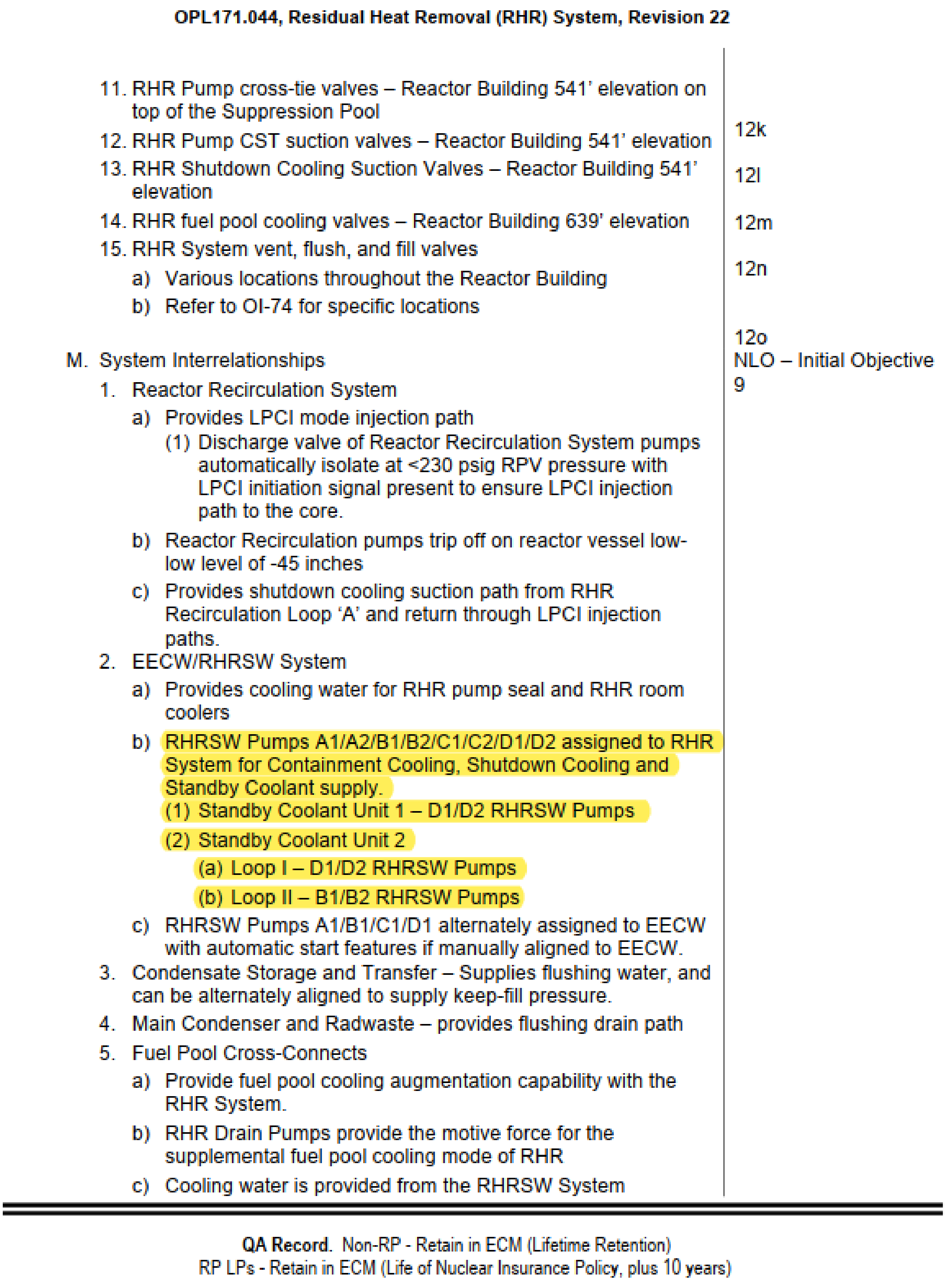
1) **DO NOT EXCEED** 4500 gpm RHRSW flow through any RHR Heat Exchanger.

2) 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

[10.2] **ESTABLISH** RHRSW flow by one of the following methods:

[10.2.1] **REQUEST** another unit establish minimum flow for Pump which will be utilized for Suppression Pool Cooling, and establish minimum flow. (between 4000 and 4500 gpm RHRSW flow) REFER TO 0-OI-23. **OR**:

Excerpt from OPL171.044 Lesson Plan:



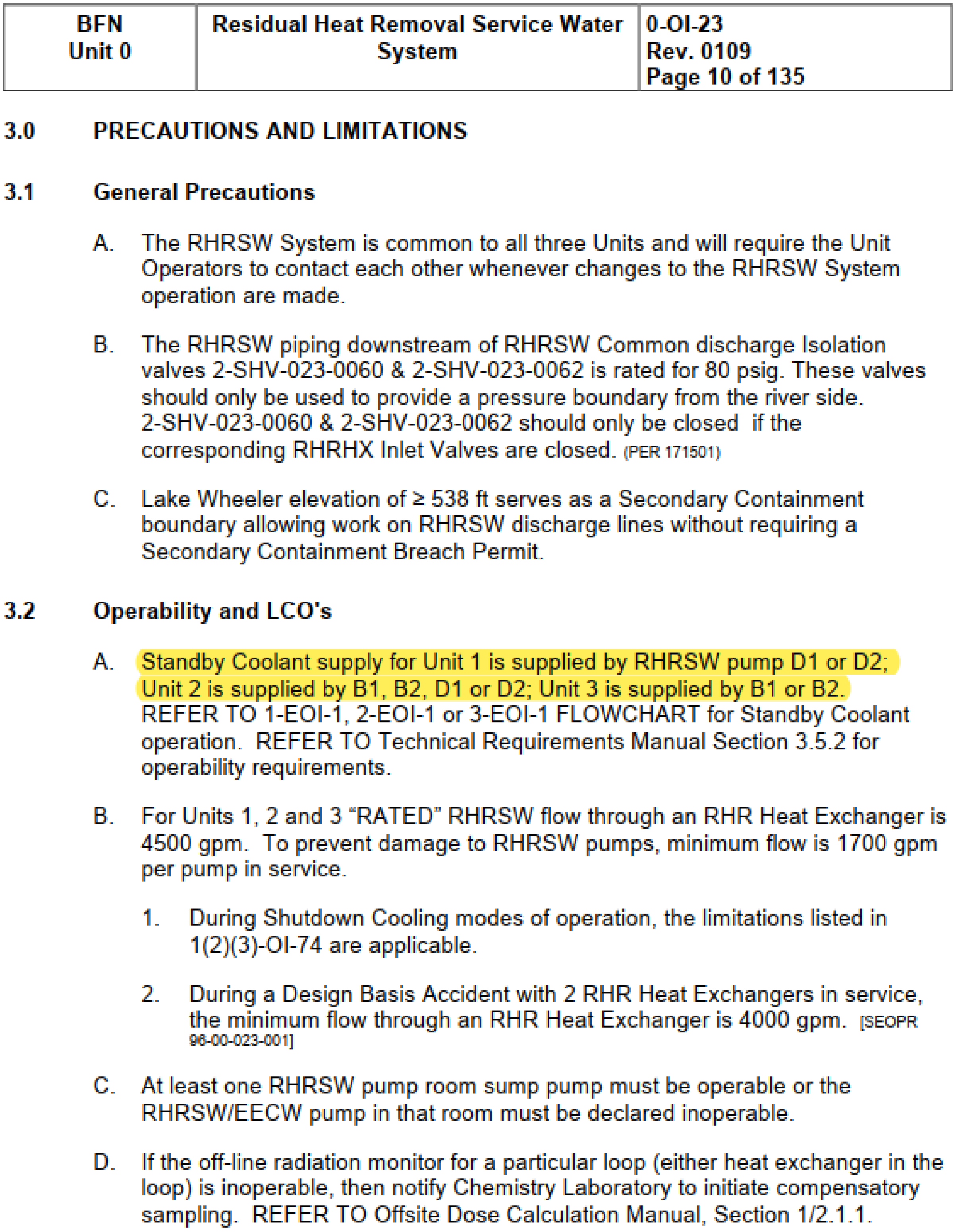
OPL171.044, Residual Heat Removal (RHR) System, Revision 22

11. RHR Pump cross-tie valves – Reactor Building 541' elevation on top of the Suppression Pool
12. RHR Pump CST suction valves – Reactor Building 541' elevation 12k
13. RHR Shutdown Cooling Suction Valves – Reactor Building 541' elevation 12l
14. RHR fuel pool cooling valves – Reactor Building 639' elevation 12m
15. RHR System vent, flush, and fill valves 12n
    a) Various locations throughout the Reactor Building
    b) Refer to OI-74 for specific locations

M. System Interrelationships 12o NLO – Initial Objective 9
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
    c) Cooling water is provided from the RHRSW System

QA Record. Non-RP - Retain in ECM (Lifetime Retention)
RP LPs - Retain in ECM (Life of Nuclear Insurance Policy, plus 10 years)

Excerpt from 0-OI-23:



BFN Unit 0	**Residual Heat Removal Service Water System**	**0-OI-23 Rev. 0109 Page 10 of 135**

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions

A. The RHRSW System is common to all three Units and will require the Unit Operators to contact each other whenever changes to the RHRSW System operation are made.

B. The RHRSW piping downstream of RHRSW Common discharge Isolation valves 2-SHV-023-0060 & 2-SHV-023-0062 is rated for 80 psig. These valves should only be used to provide a pressure boundary from the river side. 2-SHV-023-0060 & 2-SHV-023-0062 should only be closed if the corresponding RHRHX Inlet Valves are closed. (PER 171501)

C. Lake Wheeler elevation of ≥ 538 ft serves as a Secondary Containment boundary allowing work on RHRSW discharge lines without requiring a Secondary Containment Breach Permit.

3.2 Operability and LCO's

A. Standby Coolant supply for Unit 1 is supplied by RHRSW pump D1 or D2; Unit 2 is supplied by B1, B2, D1 or D2; Unit 3 is supplied by B1 or B2. REFER TO 1-EOI-1, 2-EOI-1 or 3-EOI-1 FLOWCHART for Standby Coolant operation. REFER TO Technical Requirements Manual Section 3.5.2 for operability requirements.

B. For Units 1, 2 and 3 “RATED” RHRSW flow through an RHR Heat Exchanger is 4500 gpm. To prevent damage to RHRSW pumps, minimum flow is 1700 gpm per pump in service.

1. During Shutdown Cooling modes of operation, the limitations listed in 1(2)(3)-OI-74 are applicable.
2. During a Design Basis Accident with 2 RHR Heat Exchangers in service, the minimum flow through an RHR Heat Exchanger is 4000 gpm. [SEOPR 96-00-023-001]

C. At least one RHRSW pump room sump pump must be operable or the RHRSW/EECW pump in that room must be declared inoperable.

D. If the off-line radiation monitor for a particular loop (either heat exchanger in the loop) is inoperable, then notify Chemistry Laboratory to initiate compensatory sampling. REFER TO Offsite Dose Calculation Manual, Section 1/2.1.1.

Excerpt from OPL171.046: Also supports Distractors A(1), B(1)

OPL171.046, Residual Heat Removal Service Water (RHRSW), Rev.#19U1

(3) For all temperatures above 150 °F, flow through the companion RHR Heat Exchanger shall be greater than or equal to 1500 gpm.

C. Alternate Flow Paths

1. Standby Cooling Mode provides a flowpath, utilizing the RHRSW and RHR system piping and valves, to pump river water from Wheeler Reservoir for any of the following modes of RHR operation:

Obj. 4, 7.b ILT
Obj. 3, 6.b LOR
Obj. 5 NLOR
Obj. 6 NLO
TP-4

a) LPCI (EOI App 7D)

b) Containment Sprays (EOI App 17B,17C)

c) Torus water level control (EOI App 18)

HU Tools: Questioning Attitude, Procedure Use & Adherence

d) Containment Flooding (SAMGs)

e) Fire Protection System injection (EOI App 7K)

Question: how does App 7K work?

2. EOI Appendices provide direction for placing standby coolant in service in the various modes.

1-FCV-23-57
2-FCV-23-57
TRM 3.5.2

RHR cross-tie motor operated valves connecting to the RHRSW piping used in conjunction with the respective standby coolant supply valve (FCV-23-57) permit associated RHRSW pumps to supply water to any unit's reactor.

a) Standby coolant for Unit 1 utilizes D1 or D2 RHRSW pumps.

b) Standby coolant for Unit 2 utilizes D1, D2, B1 or B2 RHRSW pumps.

c) Standby coolant for Unit 3 utilizes B1 or B2 RHRSW pumps.

3. Standby Coolant Flowpath

TP-4

a) Unit 1 Loop 2 - RHRSW pump D1 and D2 to 1D RHR Hx (all “D” Hx outlet valves are closed), 1-FCV-23-57, 1-FCV-74-101, then the appropriate RHR valves.

b) Unit 2 Loop 1 - RHRSW pump D1 and D2 to 1D RHR Hx (all “D” Hx outlet valves are closed), 1-FCV-23-57, 2-FCV-74-100, then the appropriate RHR valves.

c) Unit 2 Loop 2 - RHRSW pump B1 and B2 to 2B RHR Hx (all “B” Hx outlet valves are closed), 2-FCV-23-57, 2-FCV-74-101, then the appropriate RHR valves.

Ensures configuration control

Examination Outline Cross-reference:

239002 (SF3 SRV) Safety Relief Valves

K2.01 (10CFR 55.41.7)

Knowledge of electrical power supplies to the following:

- SRV solenoids

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	239002K2.01	
Importance Rating	3.7	------

Proposed Question: **# 26**

When Unit 3 Safety Relief Valve (SRV) solenoids experience an undervoltage condition on the normal power supply, the transfer to the alternate power supply **(1)**.

ALTERNATE electrical power for Unit 3 SRV Solenoids, where available, is supplied from 250 VDC **(2)**.

A. (1) occurs automatically
(2) RMOV Boards ONLY

B. (1) occurs automatically
(2) RMOV Boards and Battery Boards

C. (1) must be performed manually
(2) RMOV Boards ONLY

D. (1) must be performed manually
(2) RMOV Boards and Battery Boards

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that all Safety Relief Valves (SRVs) are primarily powered by the 250V DC RMOV Boards and the backup power supply for some SRVs is 250V DC RMOV Boards. Additionally, the normal and alternate power supplies to ADS Logic is the 250V DC RMOV boards.

B **CORRECT**: (*See attached*) In accordance with 3-AOI-1-1, Relief Valve Stuck Open, the ADS valves that have more than one power supply will automatically transfer on a loss of power when an undervoltage relays senses a loss of normal power supply voltage. For second part, the alternate power supply for SRVs is from both RMOV Boards and Battery Boards.

C INCORRECT: First part is incorrect but plausible in that during Control Room abandonment, manual transfers associated with power supplies are necessary. 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 whether the candidate has knowledge of power supplies to SRV solenoids and how those power supplies interrelate. Adding to the level of difficulty, these valves with alternate power supplies are utilized at the Remote Shutdown Panel; where a MANUAL transfer of power supplies to EMERGENCY is initiated. Additionally, normal power supplies are distributed among the three 250 VDC boards such that candidate may assume there is adequate time to manually transfer power, as the SRV LCO is predicated upon the Safety Mode only. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): 3-AOI-1-1, Rev. 14 (Attach if not previously provided)
OPL171.009, Rev. 17U1
OPL171.043, Rev. 16

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.009, Obj 4 (As available)

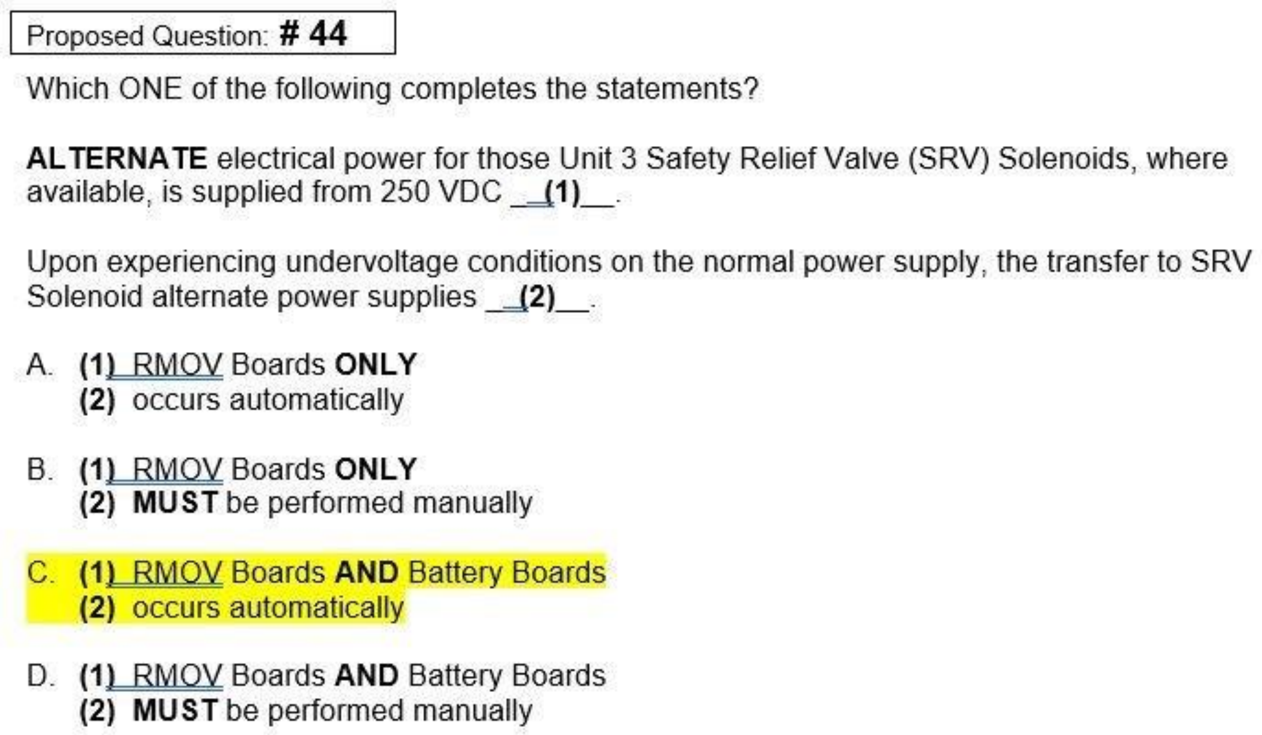
Question Source:	Bank #	BFN 1006 NRC #44	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2010	

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:



Proposed Question: **# 44**

Which ONE of the following completes the statements?

ALTERNATE electrical power for those Unit 3 Safety Relief Valve (SRV) Solenoids, where available, is supplied from 250 VDC __**(1)**__.

Upon experiencing undervoltage conditions on the normal power supply, the transfer to SRV Solenoid alternate power supplies __**(2)**__.

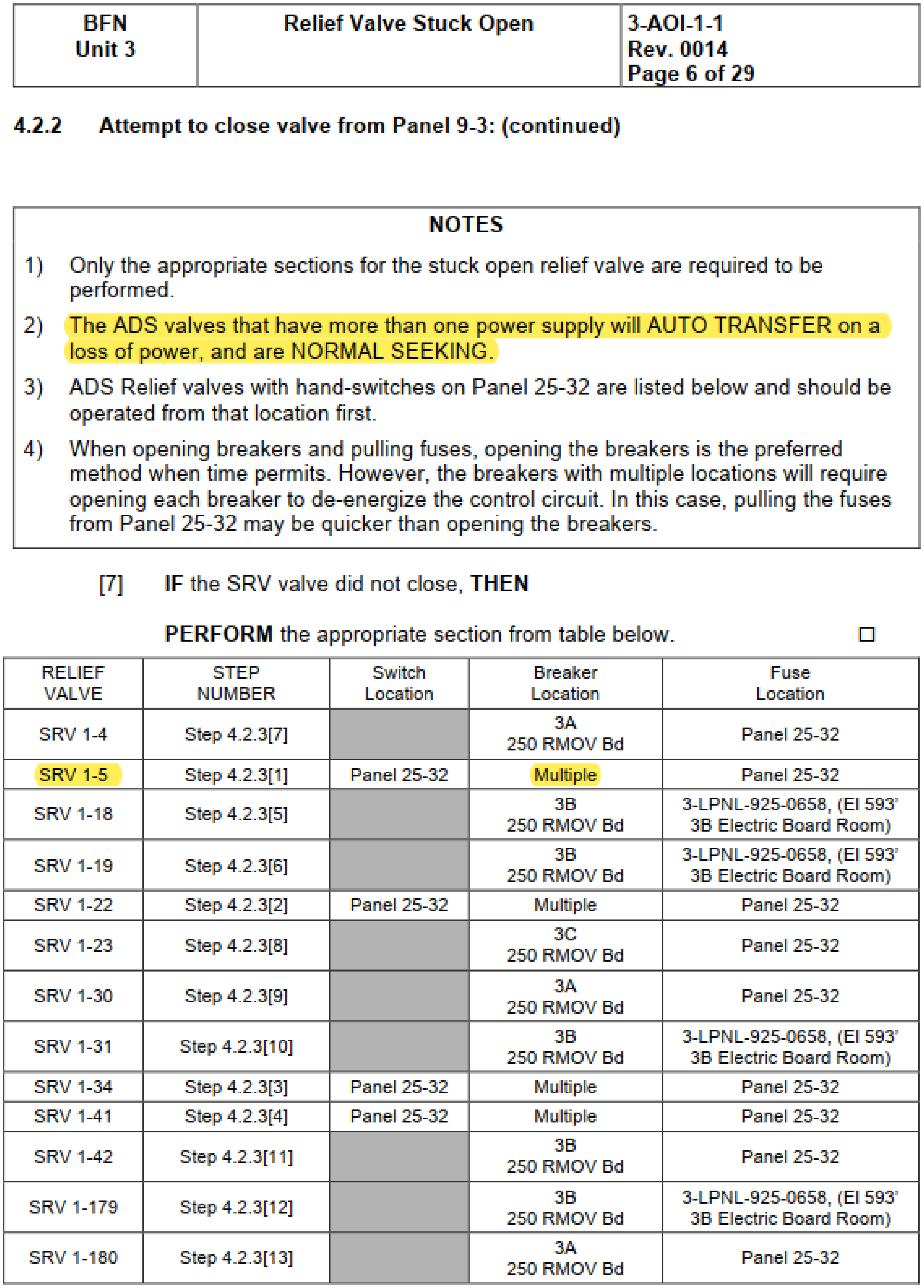
A. **(1)** RMOV Boards **ONLY**
(2) occurs automatically

B. **(1)** RMOV Boards **ONLY**
(2) **MUST** be performed manually

C. **(1)** RMOV Boards **AND** Battery Boards
(2) occurs automatically

D. **(1)** RMOV Boards **AND** Battery Boards
(2) **MUST** be performed manually

Excerpts from 3-AOI-1-1: (using SRV 1-5 as an example)



BFN Unit 3	**Relief Valve Stuck Open**	**3-AOI-1-1 Rev. 0014 Page 6 of 29**

4.2.2 Attempt to close valve from Panel 9-3: (continued)

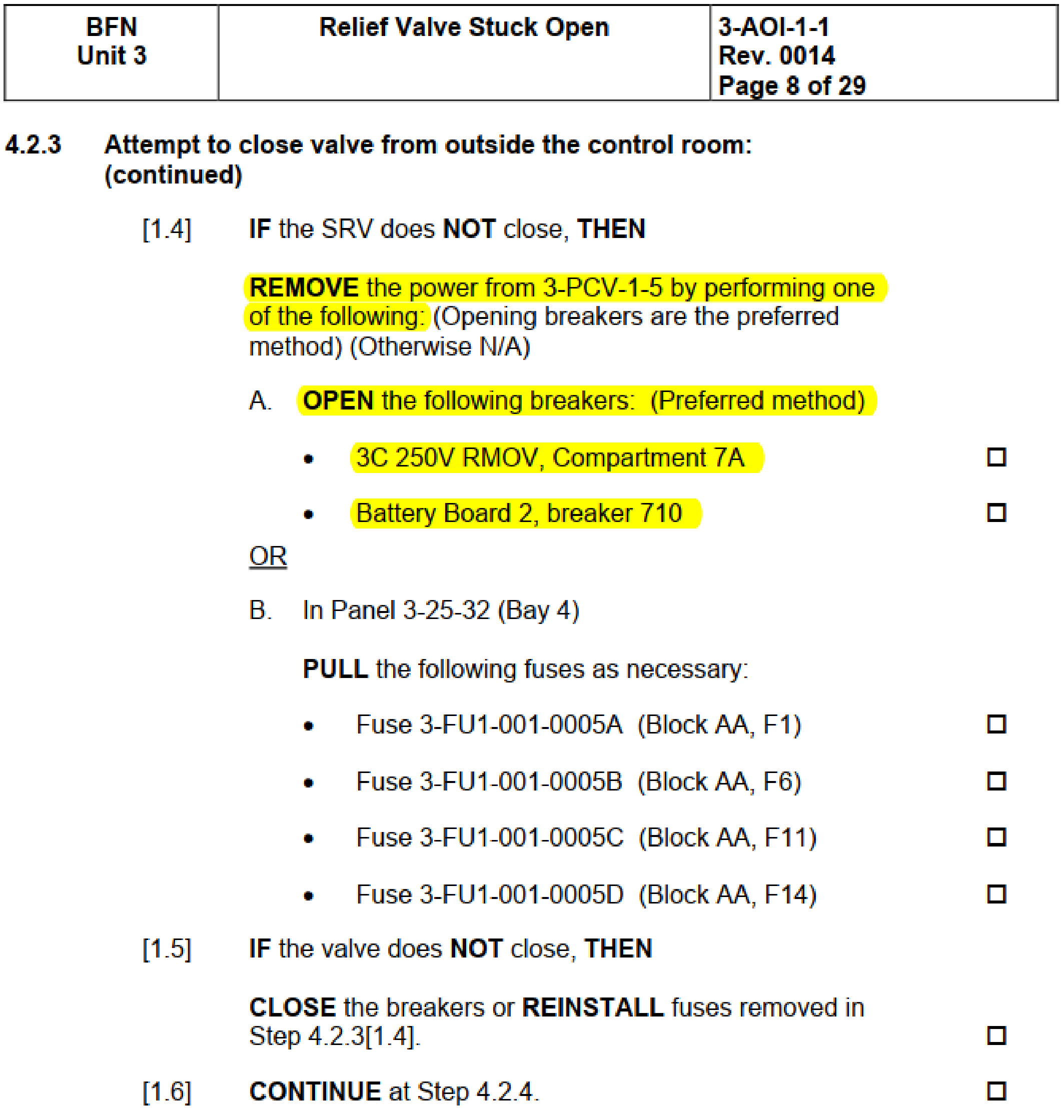
NOTES

1) Only the appropriate sections for the stuck open relief valve are required to be performed.
2) The ADS valves that have more than one power supply will AUTO TRANSFER on a loss of power, and are NORMAL SEEKING.
3) ADS Relief valves with hand-switches on Panel 25-32 are listed below and should be operated from that location first.
4) When opening breakers and pulling fuses, opening the breakers is the preferred method when time permits. However, the breakers with multiple locations will require opening each breaker to de-energize the control circuit. In this case, pulling the fuses from Panel 25-32 may be quicker than opening the breakers.

[7] **IF** the SRV valve did not close, **THEN**

PERFORM the appropriate section from table below. ☐

RELIEF VALVE	STEP NUMBER	Switch Location	Breaker Location	Fuse Location
SRV 1-4	Step 4.2.3[7]		3A 250 RMOV Bd	Panel 25-32
SRV 1-5	Step 4.2.3[1]	Panel 25-32	Multiple	Panel 25-32
SRV 1-18	Step 4.2.3[5]		3B 250 RMOV Bd	3-LPNL-925-0658, (El 593' 3B Electric Board Room)
SRV 1-19	Step 4.2.3[6]		3B 250 RMOV Bd	3-LPNL-925-0658, (El 593' 3B Electric Board Room)
SRV 1-22	Step 4.2.3[2]	Panel 25-32	Multiple	Panel 25-32
SRV 1-23	Step 4.2.3[8]		3C 250 RMOV Bd	Panel 25-32
SRV 1-30	Step 4.2.3[9]		3A 250 RMOV Bd	Panel 25-32
SRV 1-31	Step 4.2.3[10]		3B 250 RMOV Bd	3-LPNL-925-0658, (El 593' 3B Electric Board Room)
SRV 1-34	Step 4.2.3[3]	Panel 25-32	Multiple	Panel 25-32
SRV 1-41	Step 4.2.3[4]	Panel 25-32	Multiple	Panel 25-32
SRV 1-42	Step 4.2.3[11]		3B 250 RMOV Bd	Panel 25-32
SRV 1-179	Step 4.2.3[12]		3B 250 RMOV Bd	3-LPNL-925-0658, (El 593' 3B Electric Board Room)
SRV 1-180	Step 4.2.3[13]		3A 250 RMOV Bd	Panel 25-32



BFN **Unit 3**	**Relief Valve Stuck Open**	**3-AOI-1-1** **Rev. 0014** **Page 8 of 29**

4.2.3 Attempt to close valve from outside the control room: (continued)

[1.4] **IF** the SRV does **NOT** close, **THEN**

REMOVE the power from 3-PCV-1-5 by performing one of the following: (Opening breakers are the preferred method) (Otherwise N/A)

A. **OPEN** the following breakers: (Preferred method)

- 3C 250V RMOV, Compartment 7A ☐
- Battery Board 2, breaker 710 ☐

<u>OR</u>

B. In Panel 3-25-32 (Bay 4)

PULL the following fuses as necessary:

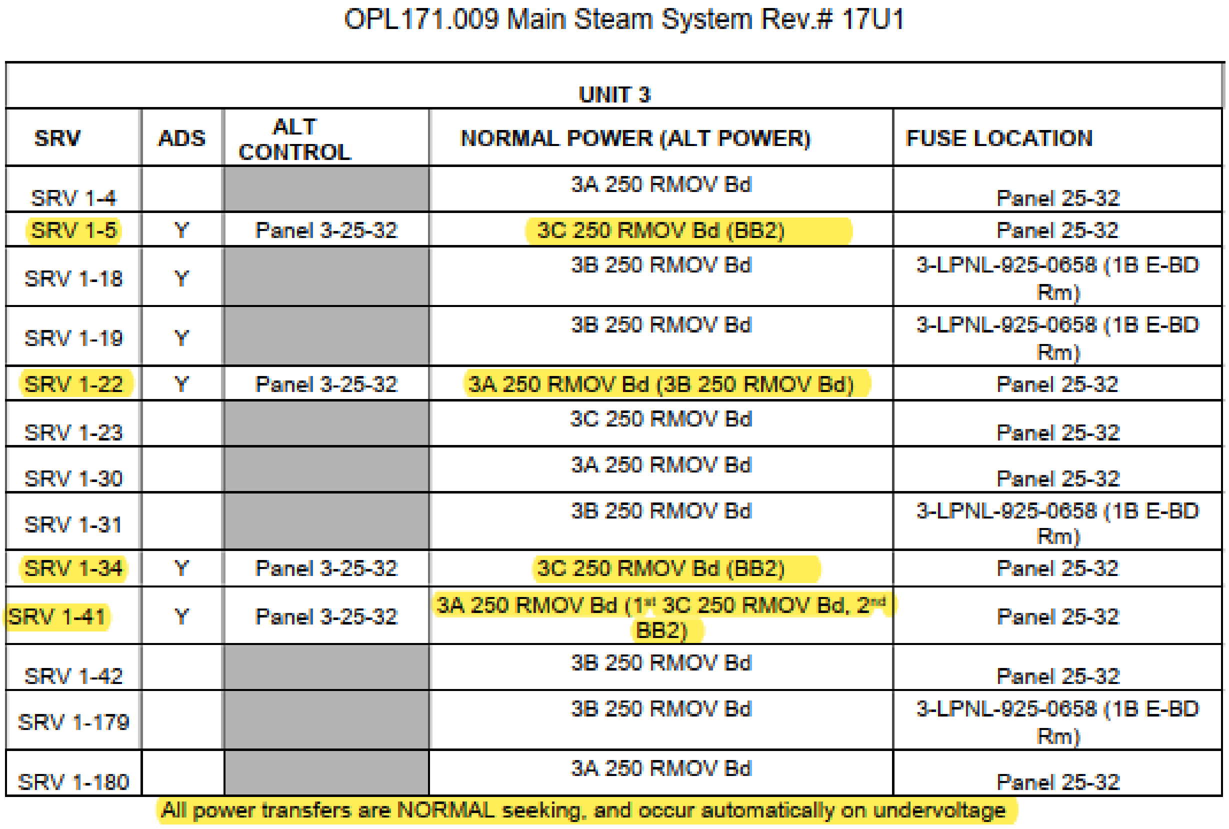
- Fuse 3-FU1-001-0005A (Block AA, F1) ☐
- Fuse 3-FU1-001-0005B (Block AA, F6) ☐
- Fuse 3-FU1-001-0005C (Block AA, F11) ☐
- Fuse 3-FU1-001-0005D (Block AA, F14) ☐

[1.5] **IF** the valve does **NOT** close, **THEN**

CLOSE the breakers or **REINSTALL** fuses removed in Step 4.2.3[1.4]. ☐

[1.6] **CONTINUE** at Step 4.2.4. ☐

Excerpt from OPL171.009 Lesson Plan:



OPL171.009 Main Steam System Rev.# 17U1

UNIT 3				
SRV	**ADS**	**ALT CONTROL**	**NORMAL POWER (ALT POWER)**	**FUSE LOCATION**
SRV 1-4			3A 250 RMOV Bd	Panel 25-32
SRV 1-5	Y	Panel 3-25-32	3C 250 RMOV Bd (BB2)	Panel 25-32
SRV 1-18	Y		3B 250 RMOV Bd	3-LPNL-925-0658 (1B E-BD Rm)
SRV 1-19	Y		3B 250 RMOV Bd	3-LPNL-925-0658 (1B E-BD Rm)
SRV 1-22	Y	Panel 3-25-32	3A 250 RMOV Bd (3B 250 RMOV Bd)	Panel 25-32
SRV 1-23			3C 250 RMOV Bd	Panel 25-32
SRV 1-30			3A 250 RMOV Bd	Panel 25-32
SRV 1-31			3B 250 RMOV Bd	3-LPNL-925-0658 (1B E-BD Rm)
SRV 1-34	Y	Panel 3-25-32	3C 250 RMOV Bd (BB2)	Panel 25-32
SRV 1-41	Y	Panel 3-25-32	3A 250 RMOV Bd (1st 3C 250 RMOV Bd, 2nd BB2)	Panel 25-32
SRV 1-42			3B 250 RMOV Bd	Panel 25-32
SRV 1-179			3B 250 RMOV Bd	3-LPNL-925-0658 (1B E-BD Rm)
SRV 1-180			3A 250 RMOV Bd	Panel 25-32

All power transfers are NORMAL seeking, and occur automatically on undervoltage

Excerpt from OPL171.043 Lesson Plan:

OPL171.043 , Automatic Depressurization System (ADS), Rev# 16

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
(d) The solenoid is actuated by: *(i) Manual demand (hand switch).* *(ii) Automatic blowdown demand (ADS) for 6 valves which are controlled by ADS.* *(iii) RPV high pressure switches.*	IL-3
(e) The operating air is supplied from the drywell control air system (nitrogen).	
(f) The SRV solenoids are powered from 250 VDC RMOV Boards or Battery Boards. Some SRV power supplies have relays in the bottom of panel 25-32 that allow them to swap to an alternate supply(s) when the normal supply voltage is lost	
(i) On Unit 1, SRV's 1-5, 1-22, 1-30, and 1-34 have auto transfer capabilities	
(ii) On Unit 2, SRV's 1-5, 1-22, 1-30, and 1-34 have auto transfer capabilities	Unit Differences
(iii) On Unit 3, SRV's 1-5, 1-22, 1-34, and 1-41 have auto transfer capabilities	
(g) Loss of air or power to an SRV would inhibit the relief function but not the safety function. Per TS 3.4.3 MSRV, operability is based on the safety function (spring action) and not the 'relief' function	ILT Obj. 2 LOR Obj. 1 NLOR Obj. 1 NLO Obj. 2
2. Vacuum breaker	
a) Two spring loaded vacuum breaker check valves are provided in each SRV discharge line to prevent drawing water up into the line due to steam condensation following termination of valve operation	
b) Without the vacuum breakers, water in the discharge lines above suppression pool water level could cause excessive hydraulic stresses to the T-quenchers and other torus structural components	
3. Accumulator and check valve arrangement	IL-2
a) ADS valves are provided with accumulator arrangements.	
b) Accumulators are provided to assure that the ADS valves can be held open for 30 minutes following a failure of the pneumatic supply to the accumulators.	
c) Accumulators are sized to contain sufficient air for that minimum of five valve operations following a loss of Drywell Control Air. Extended loss of pneumatic supply and system leakage would result in failure of the 'relief mode' of the MSRVs.	ILT Obj. 2 LOR Obj. 1 NLOR Obj. 1 NLO Obj. 2
d) EOI Appendix 8G crossties CAD to DWCA.	'A' CAD supplies 3 ADS valves 'B' CAD supplies 3 ADS valves HU Tool - Procedure Use

Examination Outline Cross-reference:	Level	RO	SRO
295029 (EPE 6) High Suppression Pool Water Level / 5	Tier #	1	------
EK2.07 (10CFR 55.41.7)	Group #	2	------
Knowledge of the relationship between High Suppression Pool Water Level and the following systems or components: • Drywell/containment water level	K/A #	295029EK2.07	
	Importance Rating	3.6	------

Proposed Question: **# 27**

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

At 0400:

- Suppression Pool Water Level is (-) 3 inches and rising at 1 inch per hour

Given the conditions above, the **EARLIEST** time that Technical Specification (Tech Spec) 3.6.2.2, Suppression Pool Water Level limit will be reached is **(1)** and 2-EOI-2, Primary Containment Control entry **(2)** be required for Suppression Pool Water Level.

A. (1) 0515
(2) will NOT

B. (1) 0515
(2) will

C. (1) 0615
(2) will NOT

D. (1) 0615
(2) will

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 0515 would result in a Suppression Pool Water Level of (-) 1.75 inches which would cause Suppression Chamber Level Abnormal (2-9-3B, Window 15) to annunciate. Second part is incorrect but plausible in that the candidate could miscalculate the rise of the Suppression Pool Water Level and/or confuse the result as it applies to 2-EOI-2 entry conditions or Tech Spec LCO requirements or 2-OI-64, Primary Containment normal operations for Suppression Pool Water Level.

B INCORRECT: First part is incorrect *(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)* Tech Spec 3.6.2.2 states that Suppression Pool Water Level shall be less (-) 6.25 inches with and (-) 7.25 inches without differential pressure control and ≤ (-) 1.0 inches. Given the rising rate of 1 inch per hour began at 0400 with a starting level of (-) 3.0 inches then at 0615, Suppression Pool Water Level would be (-) .75 inches. LCO 3.6.2.2 will no longer be met since (-) .75 inches is higher than (-) 1.0 inches. For second part, in accordance with 1-EOI-2, Primary Containment Control, entry is required when Suppression Pool Water Level > (-) 1.0 inch which is the case for (-) .75 inches.

RO Level Justification: Tests the candidate's knowledge of system parameters such as High Suppression Pool Water Level in containment as it relates to Technical Specifications and entry-level conditions of Emergency Operating Instructions. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (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)
2-OI-64, Rev. 129
2-ARP-9-3B, Rev. 38
2-EOI-2, Rev. 17

Proposed references to be provided to applicants during examination: **NONE**

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

Question Source:	Bank #		
	Modified Bank #	BFN 2104 #85	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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: **# 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

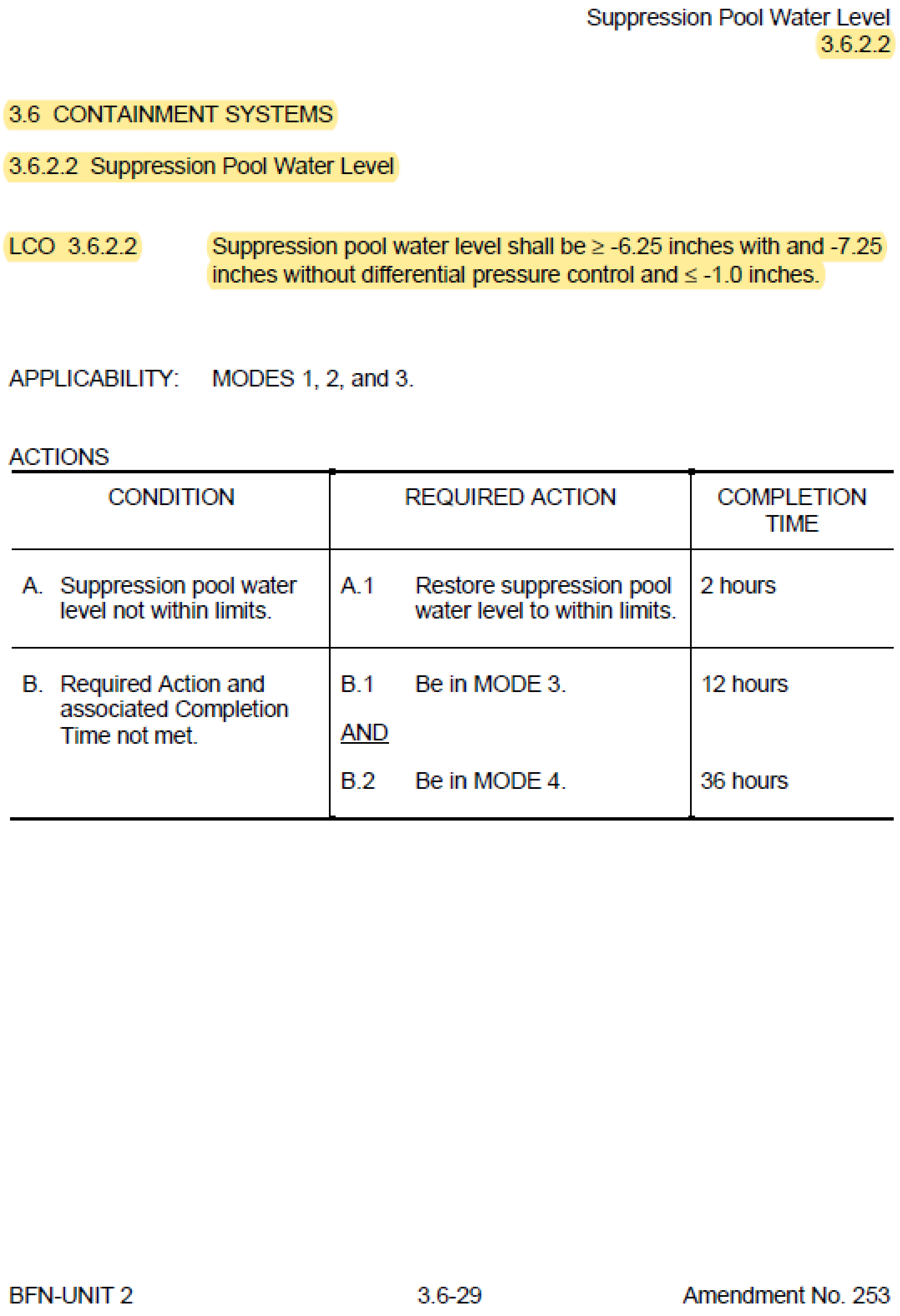
B. (1) 0315
(2) excessive clearing loads from MSRV discharges and excessive pool swell loads

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**

Excerpt from Unit 2 Tech Spec 3.6.2.2:



3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

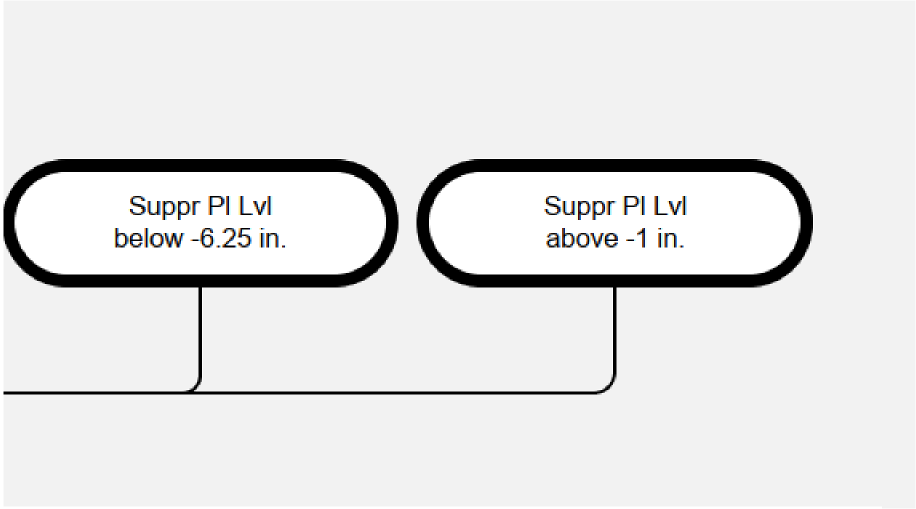
LCO 3.6.2.2 Suppression pool water level shall be ≥ -6.25 inches with and -7.25 inches without differential pressure control and ≤ -1.0 inches.

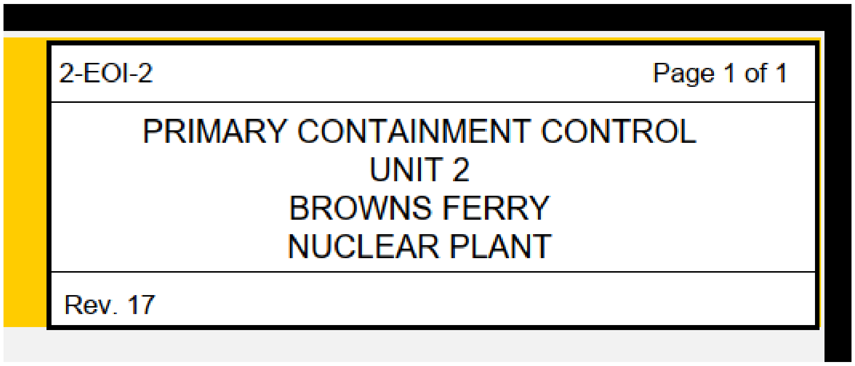
APPLICABILITY: MODES 1, 2, and 3.

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.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

Excerpt from 2-EOI-2:

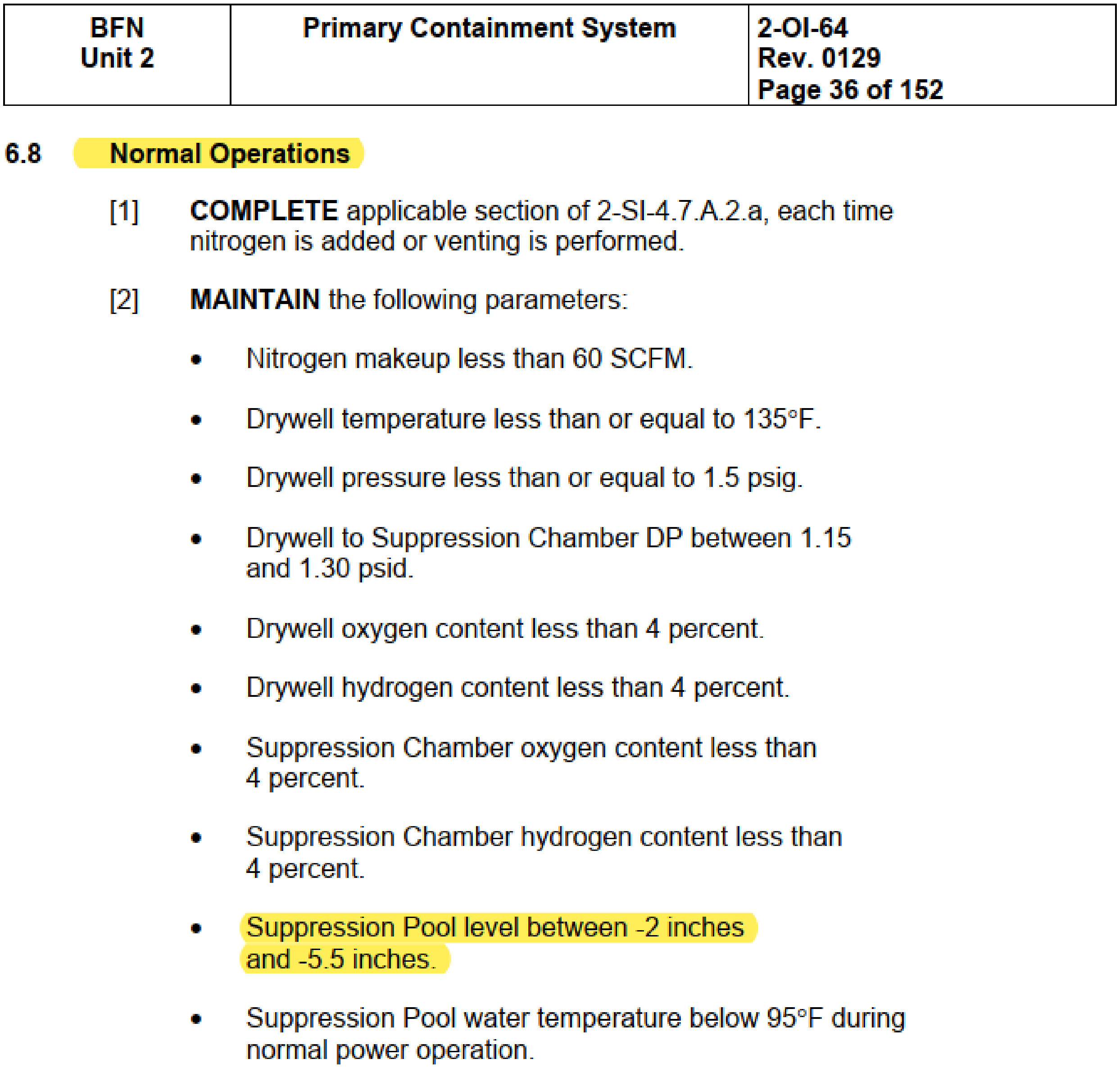



2-EOI-2 Page 1 of 1

PRIMARY CONTAINMENT CONTROL
UNIT 2
BROWNS FERRY
NUCLEAR PLANT

Rev. 17

Excerpts from 2-OI-64: Supports Distractors A(2), C(2)



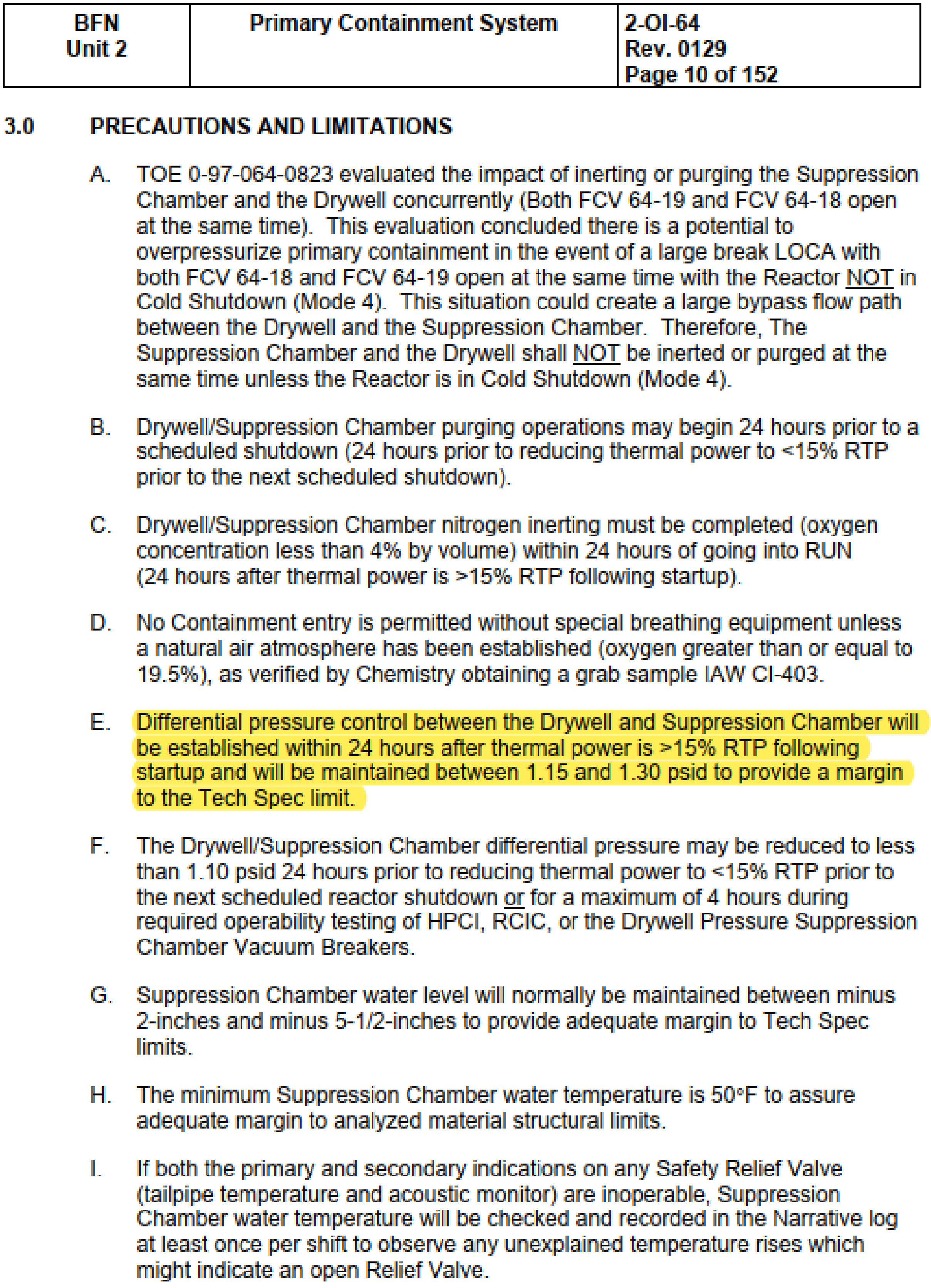
BFN Unit 2	**Primary Containment System**	**2-OI-64 Rev. 0129 Page 36 of 152**

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. 0129 Page 10 of 152

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.

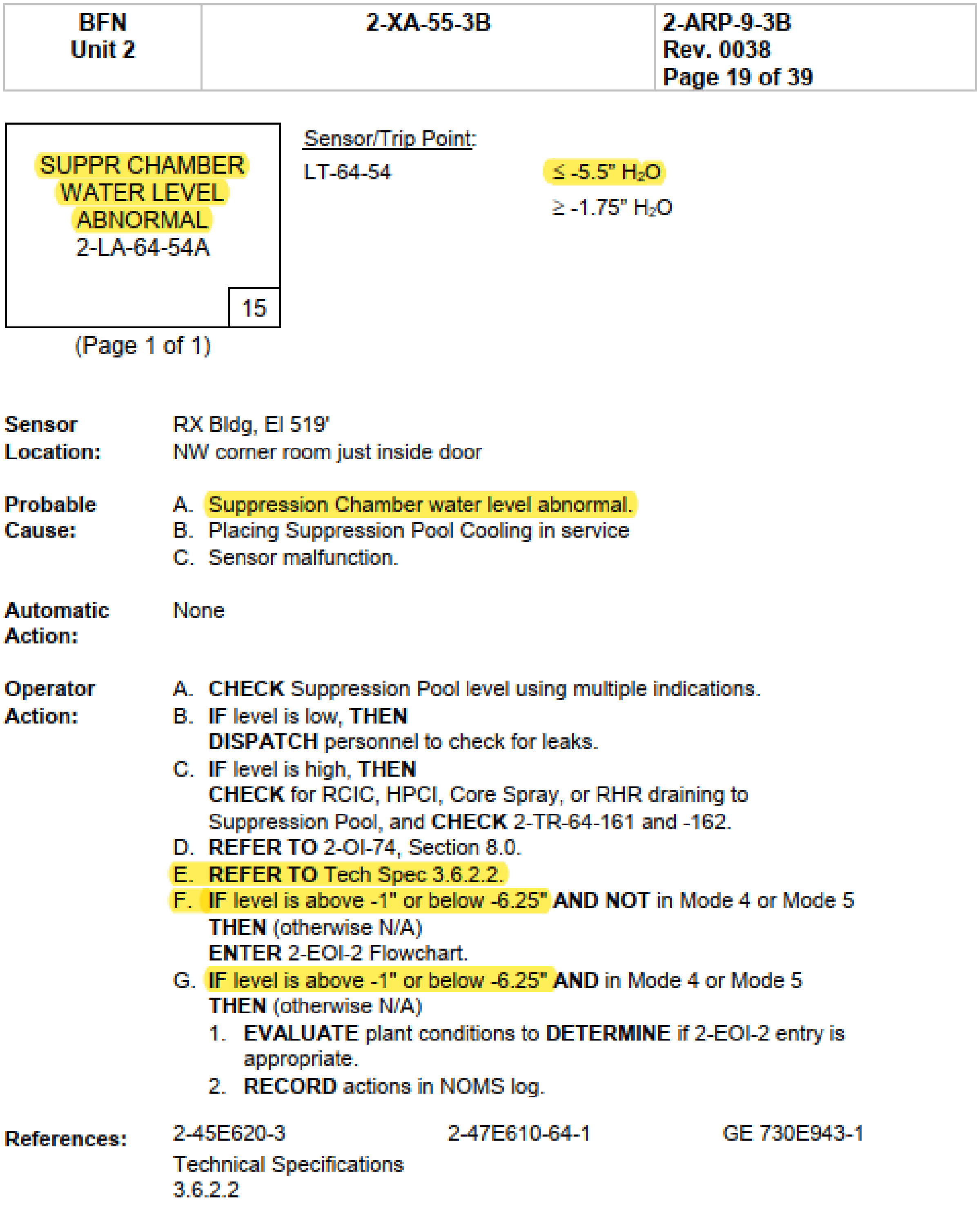
F. The Drywell/Suppression Chamber differential pressure may be reduced to less than 1.10 psid 24 hours prior to reducing thermal power to <15% RTP prior to the next scheduled reactor shutdown or for a maximum of 4 hours during required operability testing of HPCI, RCIC, or the Drywell Pressure Suppression 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: Supports Distractors A(1), B(1)



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)

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		

Examination Outline Cross-reference:	Level	RO	SRO
295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9	Tier #	1	------
EK3.06 (10CFR 55.41.5)	Group #	2	------
Knowledge of the reasons for the following responses or actions as they apply to High Secondary Containment Area Radiation Levels: • Operating ventilation systems	K/A #	295033EK3.06	
	Importance Rating	3.6	------

Proposed Question: **# 28**

Unit 3 is in MODE 5 when an event occurs resulting in the following:

- All Refueling Zone radiation monitors indicate 69 mR/hr
- All Reactor Zone radiation monitors indicate 73 mR/hr

Given the conditions above, the Unit 3 Ventilation System isolates ___**(1)**___ Zone Ventilation System(s) to limit the ___**(2)**___ .

A. (1) **ONLY** the Reactor
(2) release of fission products to the environment

B. (1) **ONLY** the Reactor
(2) spread of contamination within Secondary Containment

C. (1) the Refueling **AND** Reactor
(2) release of fission products to the environment

D. (1) the Refueling **AND** Reactor
(2) spread of contamination within Secondary Containment

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that it is given that Refuel radiation is 69 mR/hr and Reactor zone is 73 mR/hr (exceeded the 72 mR/hr isolation setpoint) which could lead the candidate to believe that only the Reactor Zone Ventilation isolates. 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 isolation of ventilation shuts down all Reactor and Refuel Zone fans. The candidate could reasonably believe the air flow stopping would be to prevent spreading of the radioactive particles as the flow path of air from outside of containment, through containment, and exhausted to the atmosphere covers the entire Secondary Containment.

C **CORRECT:** (*See Attached*) In accordance with 3-OI-90, Radiation Monitoring System, when Reactor Zone Exhaust Radiation Monitors reach 72 mr/hr high radiation, a 2 out 2 taken once logic PCIS Group 6 Isolation occurs for both Refueling and Reactor Zone Ventilation. Additionally, other automatic actions occur as a result of this signal such as Control Room Emergency Ventilation and Standby Gas Treatment auto start. For second part, in accordance with Tech Spec Bases 3.3.6.1, the isolation of containment upon a valid high radiation signal is to limit the release of fission products to the environment. Building ventilation is monitored and elevated, but it is not filtered. Standby Gas is filtered.

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 Reactor and Refuel Ventilation Systems response to a high radiation condition and the associated bases for that response. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s):	3-OI-90, Rev. 64	(Attach if not previously provided)
	3-AOI-64-2D, Rev.20	
	U3 Tech Spec 3.3.6.1, Amend. 213	

Proposed references to be provided to applicants during examination: **None**

Learning Objective: OPL171.017 Obj. 4 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1909 #33	(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: **# 33**

The following conditions are observed on Unit 1:

1-RM-90-140/142

- Reactor Zone 1-RM-90-142A indicates 65 mR/hr
- Reactor Zone 1-RM-90-142B indicates 67 mR/hr
- Refuel Zone 1-RM-90-140A indicates 75 mR/hr
- Refuel Zone 1-RM-90-140B indicates 78 mR/hr

1-RM-90-141/143

- Reactor Zone 1-RM-90-143A indicates 68 mR/hr
- Reactor Zone 1-RM-90-143B indicates downscale
- Refuel Zone 1-RM-90-141A indicates 70 mR/hr
- Refuel Zone 1-RM-90-141B indicates 69 mR/hr

Which **ONE** of the following identifies the Ventilation System response?

A. Refuel Zone isolation **ONLY**

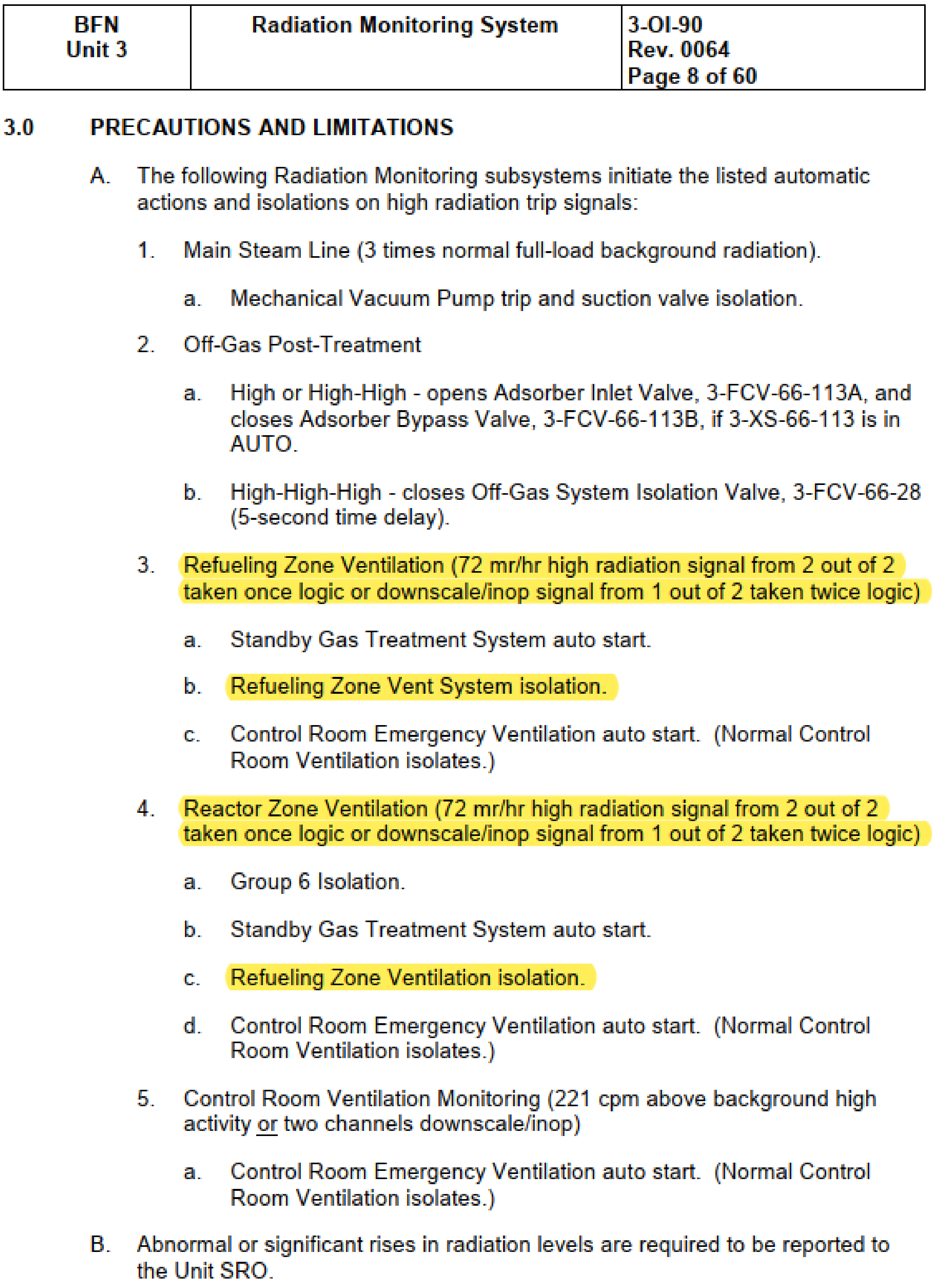
B. Reactor **AND** Refuel Zone isolation

C. Refuel Zone isolation **AND** CREV auto initiation

D. Reactor Zone isolation **AND** CREV auto initiation

Proposed Answer: **C**

Excerpts from 3-OI-90:



BFN Unit 3	**Radiation Monitoring System**	**3-OI-90 Rev. 0064 Page 8 of 60**

3.0 PRECAUTIONS AND LIMITATIONS

A. The following Radiation Monitoring subsystems initiate the listed automatic actions and isolations on high radiation trip signals:

1. Main Steam Line (3 times normal full-load background radiation).
   - a. Mechanical Vacuum Pump trip and suction valve isolation.
2. Off-Gas Post-Treatment
   - a. High or High-High - opens Adsorber Inlet Valve, 3-FCV-66-113A, and closes Adsorber Bypass Valve, 3-FCV-66-113B, if 3-XS-66-113 is in AUTO.
   - b. High-High-High - closes Off-Gas System Isolation Valve, 3-FCV-66-28 (5-second time delay).
3. Refueling Zone Ventilation (72 mr/hr high radiation signal from 2 out of 2 taken once logic or downscale/inop signal from 1 out of 2 taken twice logic)
   - a. Standby Gas Treatment System auto start.
   - b. Refueling Zone Vent System isolation.
   - c. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)
4. Reactor Zone Ventilation (72 mr/hr high radiation signal from 2 out of 2 taken once logic or downscale/inop signal from 1 out of 2 taken twice logic)
   - a. Group 6 Isolation.
   - b. Standby Gas Treatment System auto start.
   - c. Refueling Zone Ventilation isolation.
   - d. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)
5. Control Room Ventilation Monitoring (221 cpm above background high activity or two channels downscale/inop)
   - a. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)

B. Abnormal or significant rises in radiation levels are required to be reported to the Unit SRO.

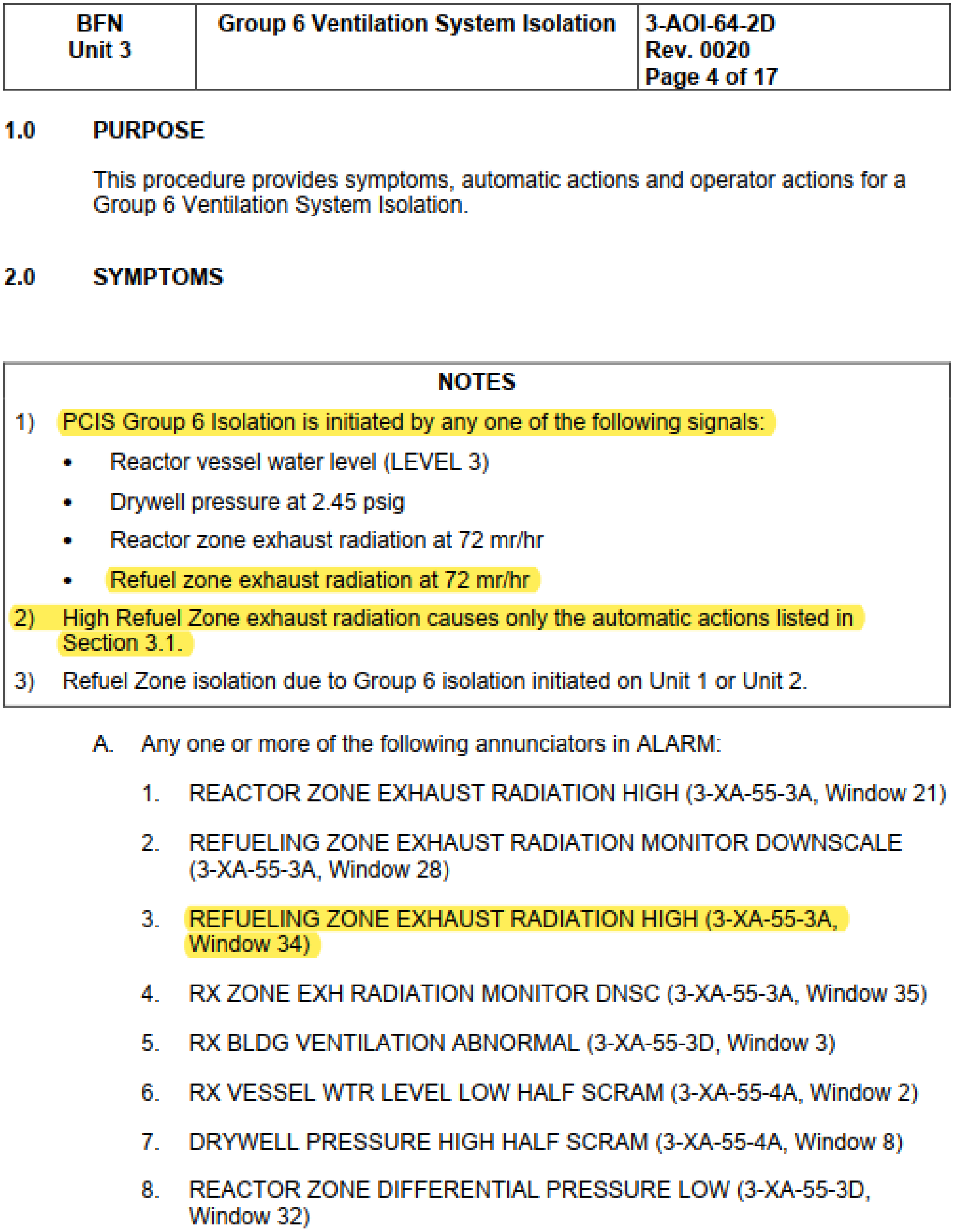
BFN **Unit 3**	**Radiation Monitoring System**	**3-OI-90** **Rev. 0064** **Page 43 of 60**

Attachment 1
(Page 2 of 3)

Radiation Monitoring System Operational Summary

Subsystem	**Operation**
Off-Gas Pretreatment Post-Treatment Vial Samplers	Two portable sampling units, **NOT** normally in service, can be used to draw Off-Gas samples for laboratory analysis during operation or while shutdown. Pretreatment samples can be drawn from the holdup volume inlet (or alternately from the SJAE second stage suction). Post-treatment samples can be drawn from the charcoal bed inlet and outlet. PNL 3-25-40 & 0-25-259
Reactor/Refueling Zone Exhaust Radiation Monitors 3-RE-90-140/142 3-RE-90-141/143	Each Control Room drawer contains all the monitors for that channel (both reactor and refuel zone). Each monitor has two detectors (i.e., 3-RM-90-141 Detector A and Detector B). The purpose of the second detector is to prevent spurious trips For an upscale trip to be initiated, both detectors (i.e., 3-RM-90-141 Detector A and Detector B) in a channel (A or B) must reach the trip setpoint. For a downscale/inop trip to be initiated, one detector for the associated zone in both channels must be in the downscale/inop state (i.e., 3-RM-90-140 Detector A or B and 3-RM-90-141 Detector A or B). Note that with this arrangement, one detector downscale/inop will render that channel inop for a high trip. **REFER TO** Tech Spec 3.3.6.2, for Reactor Bldg Vent Radiation Monitoring system channel(s) which are inoperable for functional testing or inoperable for calibration or maintenance. The inoperable channel is to be placed in the tripped condition. Note that placing the drawer in the inoperable state results in both radiation monitors (Reactor and Refuel Zone) being in the inoperable trip condition and is allowed by this note. High radiation in either of these systems will isolate the respective ventilation, start Standby Gas Treatment, and initiate emergency Control Room isolation and pressurization. In addition, high radiation in the reactor zone ventilation inserts a redundant refueling zone high radiation trip and initiates a Group 6 Isolation. High refueling zone radiation in any unit will cause a refueling zone isolation in all units.

Excerpts from 3-AOI-64-2D:



BFN **Unit 3**	**Group 6 Ventilation System Isolation**	**3-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.

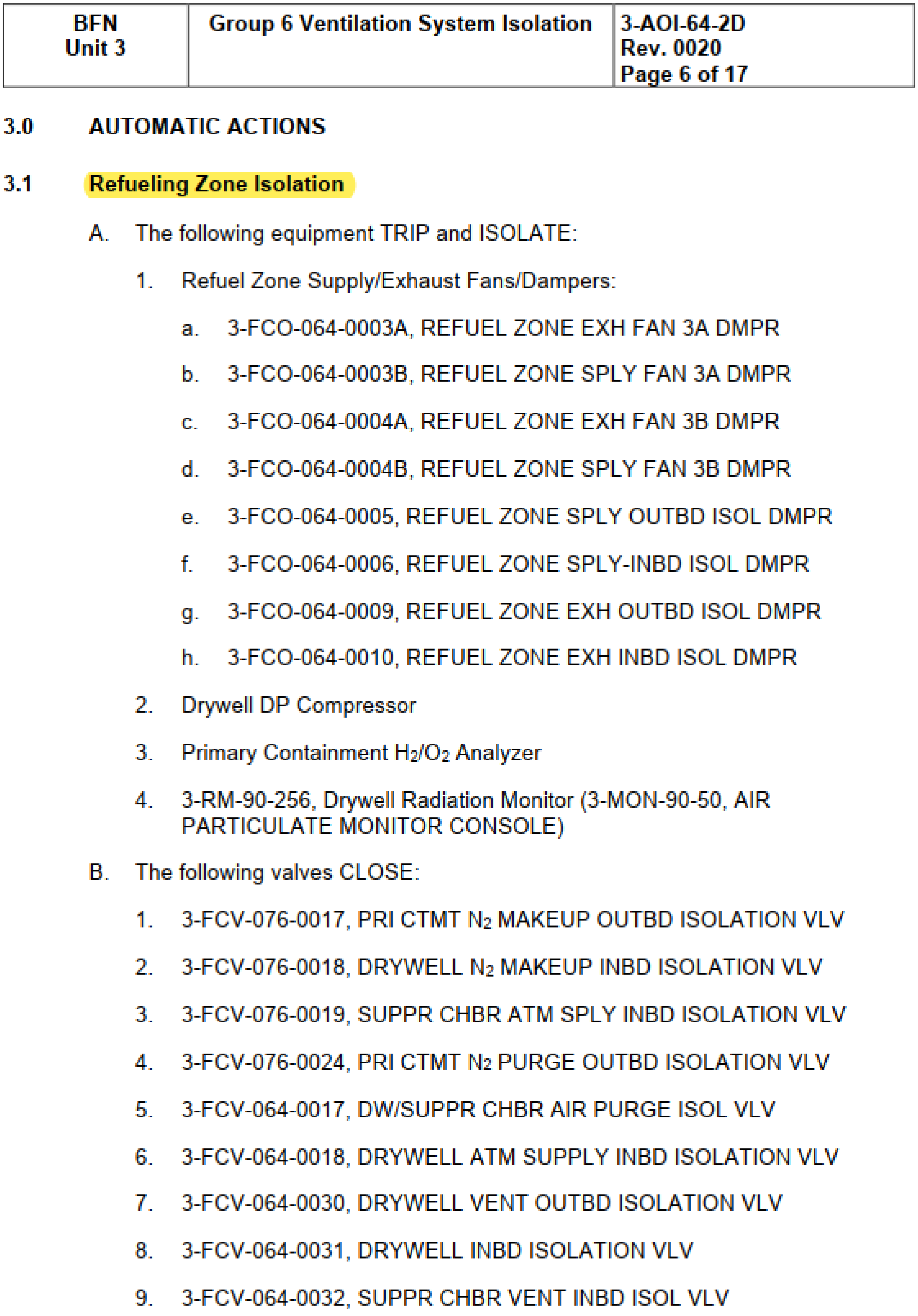
2.0 SYMPTOMS

NOTES

1) PCIS Group 6 Isolation is initiated by any one of the following signals:
   - Reactor vessel water level (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. 0020** **Page 6 of 17**

3.0 AUTOMATIC ACTIONS

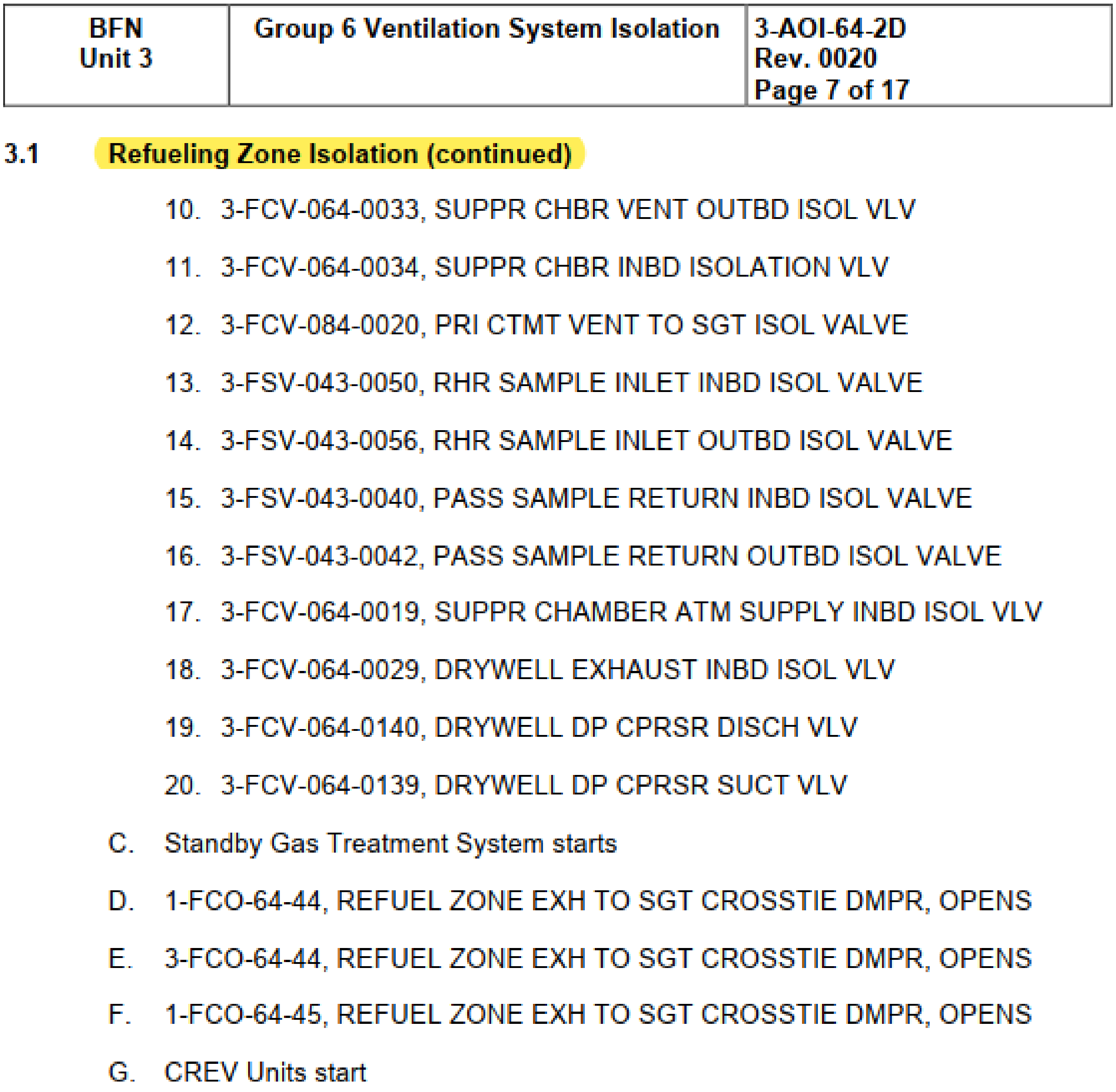
3.1 Refueling Zone Isolation

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. 0020** **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

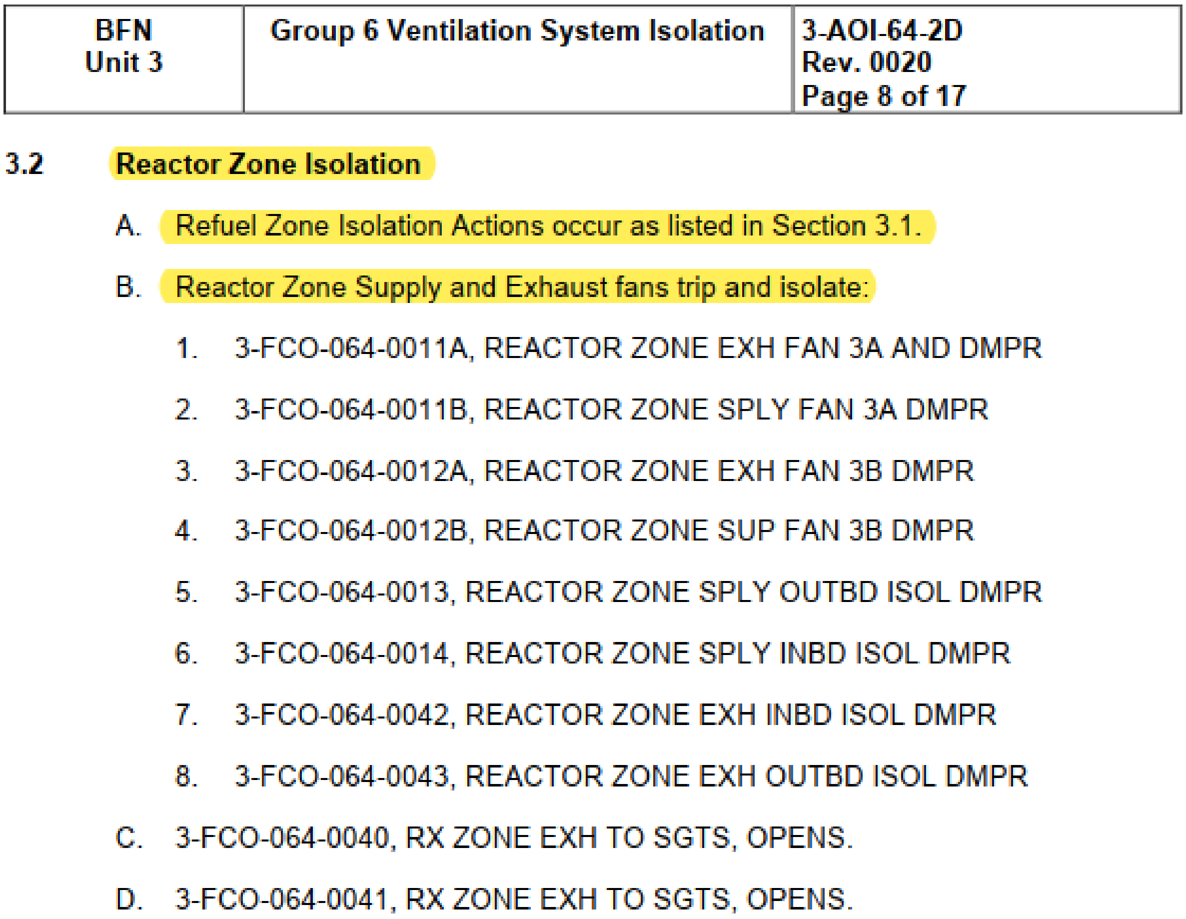
C. Standby Gas Treatment System starts

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. 0020** **Page 8 of 17**

3.2 Reactor Zone Isolation

A. Refuel Zone Isolation Actions occur as listed in Section 3.1.

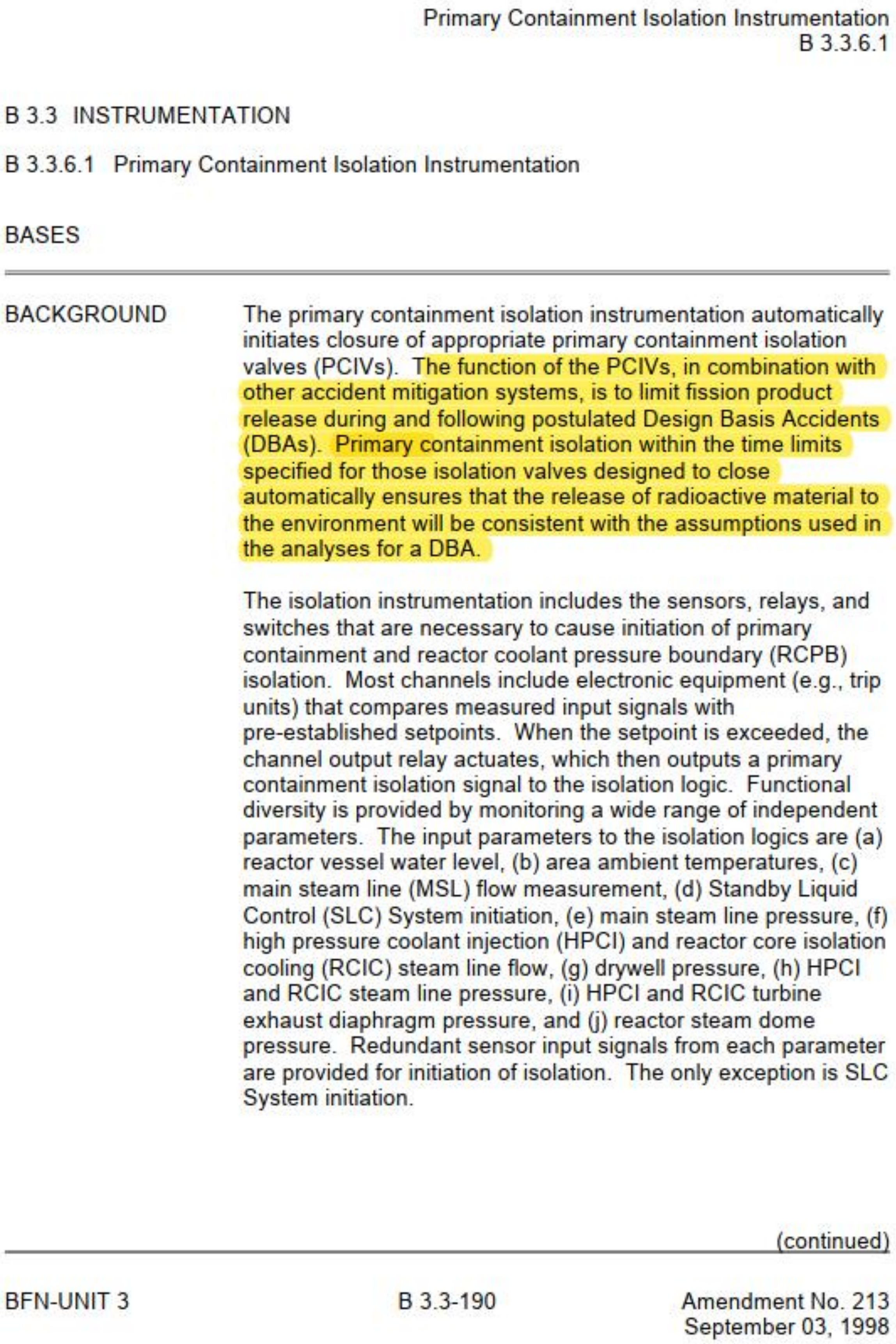
B. Reactor Zone Supply and Exhaust fans trip and isolate:

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.

Excerpt from U3 Tech Spec Bases 3.3.6.1:



Primary Containment Isolation Instrumentation
B 3.3.6.1

B 3.3 INSTRUMENTATION

B 3.3.6.1 Primary Containment Isolation Instrumentation

BASES

BACKGROUND

The primary containment isolation instrumentation automatically initiates closure of appropriate primary containment isolation valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary containment and reactor coolant pressure boundary (RCPB) isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a primary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logics are (a) reactor vessel water level, (b) area ambient temperatures, (c) main steam line (MSL) flow measurement, (d) Standby Liquid Control (SLC) System initiation, (e) main steam line pressure, (f) high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) steam line flow, (g) drywell pressure, (h) HPCI and RCIC steam line pressure, (i) HPCI and RCIC turbine exhaust diaphragm pressure, and (j) reactor steam dome pressure. Redundant sensor input signals from each parameter are provided for initiation of isolation. The only exception is SLC System initiation.

(continued)

BFN-UNIT 3 B 3.3-190 Amendment No. 213
September 03, 1998

Examination Outline Cross-reference:

295037 (EPE 14) SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1

EK3.02 (10CFR 55.41.5)

Knowledge of the reasons for the following responses or actions as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or UNKNOWN:

- Boron Injection

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295037EK3.02	
Importance Rating	4.2	------

Proposed Question: **# 29**

In accordance with EOI-1A, ATWS RPV Control and the associated EOI Program Manual, Standby Liquid Control (SLC) injection is required **BEFORE** Suppression Pool Temperature rises to a **MINIMUM** of **___(1)___** in order to **___(2)___**.

A. (1) 110 °F
(2) preclude Emergency Depressurization

B. (1) 110 °F
(2) prevent Thermal Hydraulic Instabilities (THI)

C. (1) 120 °F
(2) preclude Emergency Depressurization

D. (1) 120 °F
(2) prevent Thermal Hydraulic Instabilities (THI)

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 2-EOI-1A, ATWS RPV CONTROL, step ARC/Q-8 states BEFORE Suppression Pool Temperature rises to 110 °F, continue to step ARC/Q-9 – Boron Injection is Required. For second part, in accordance with EOIPM 0-V(D), EOI-1A, ATWS RPV Control Bases for step ARC/Q-8 states that if SLC is injected prior to 110 °F which equates to Boron Injection Initiation Temperature (BIIT), then Emergency RPV Depressurization may be precluded at lower Reactor Power levels.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible in that ARC/Q-7 states, WHEN periodic APRM oscillations greater than 25 % peak-to-peak persist then ARC/Q-9 states Boron Injection is Required, however core instability is not tied to 110 °F Suppression Pool Temperature.

C INCORRECT: First part is incorrect but plausible in that 120 °F is the Tech Spec 3.6.2.1 value for Suppression Pool Average Temperature at which the RPV is required to be depressurized to less than 200 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).*

RO Level Justification: Tests the candidate's knowledge of the reasons for Boron injection as it applies to ATWS 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 this knowledge and its meaning to predict the correct outcome especially given that the candidate must discern between Technical Specification and Emergency Operating Instruction Bases knowledge without any references provided.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): (Attach if not previously provided)
U2 Tech Spec 3.6.2.1, Amend. 253
2-EOI-1A, Rev. 3
EOIPM 0-V(D), Rev. 1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.202 Obj. 1 (As available)

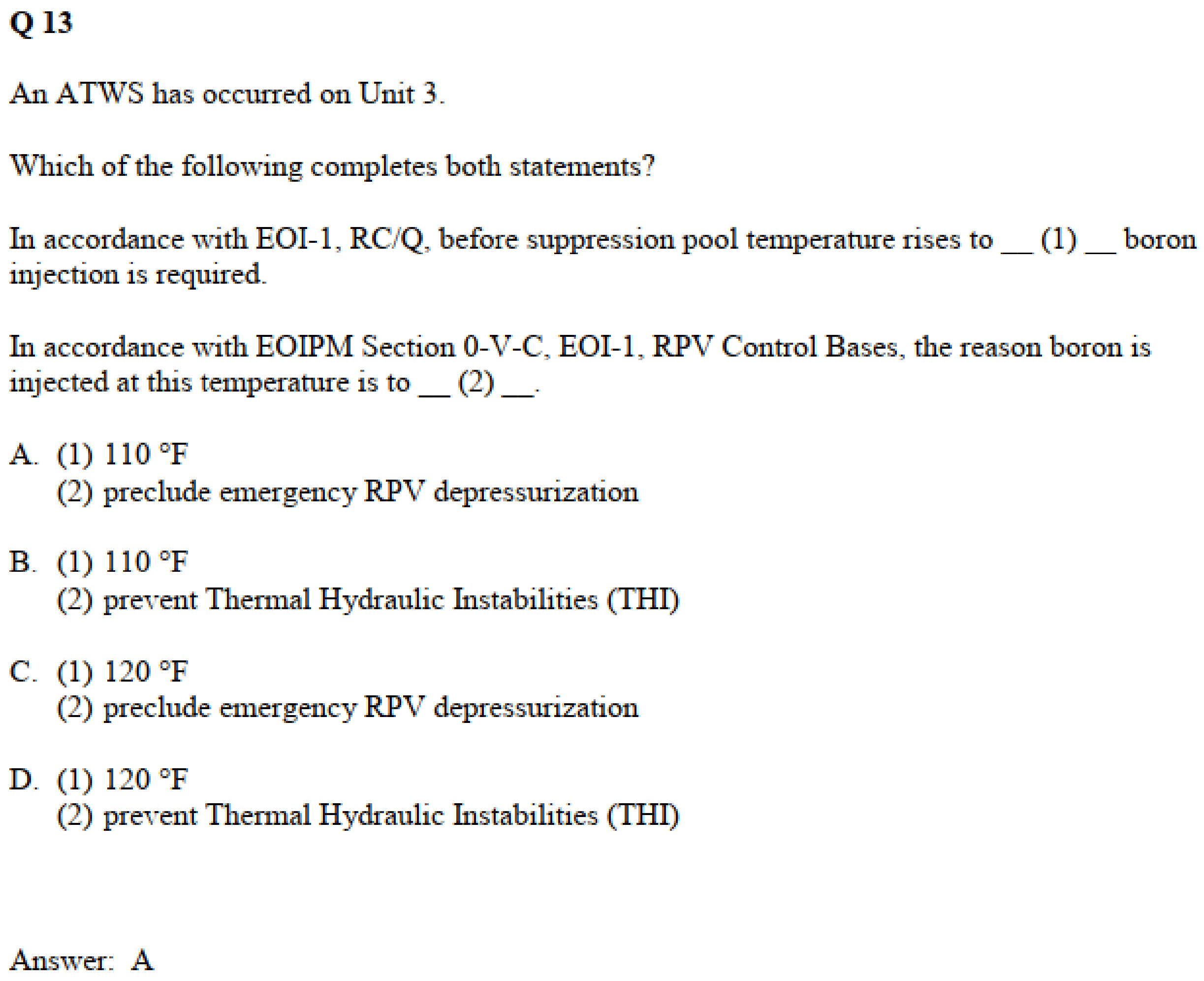
Question Source:	Bank #	ILT 1501 #13	
	Modified Bank #		(Note changes or attach parent)
Question History:	New		
	Last NRC Exam	2015	

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 13

An ATWS has occurred on Unit 3.

Which of the following completes both statements?

In accordance with EOI-1, RC/Q, before suppression pool temperature rises to __ (1) __ boron injection is required.

In accordance with EOIPM Section 0-V-C, EOI-1, RPV Control Bases, the reason boron is injected at this temperature is to __ (2) __.

A. (1) 110 °F
(2) preclude emergency RPV depressurization

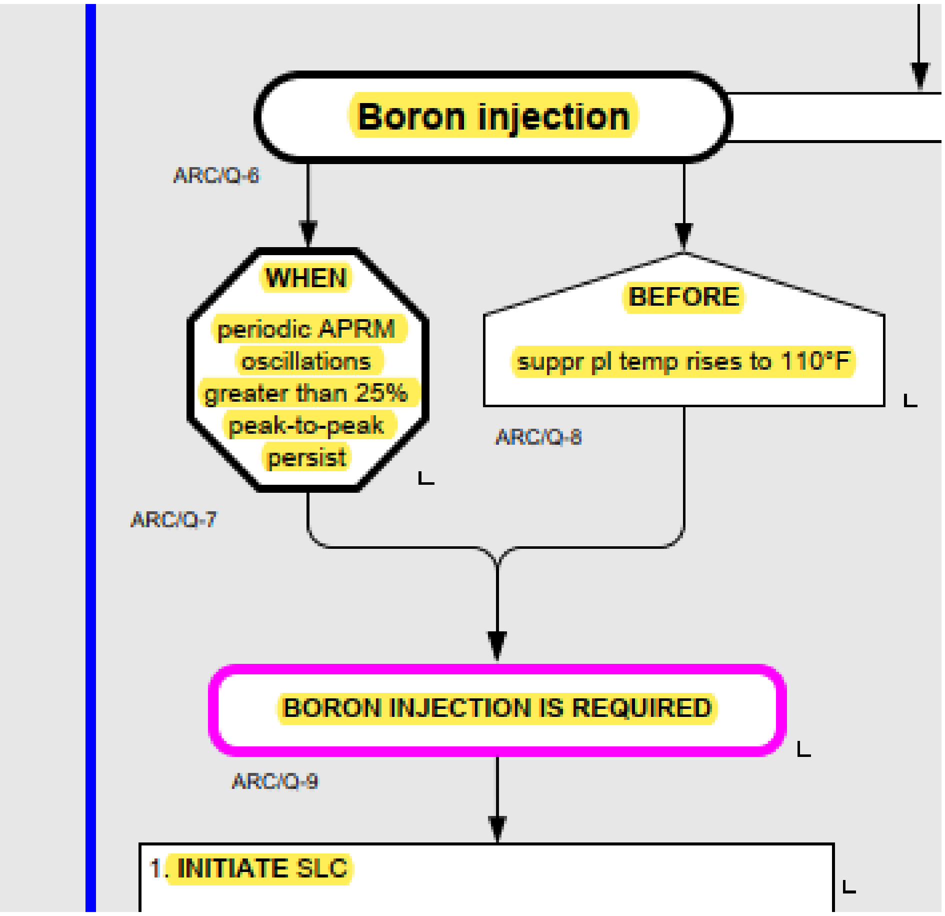
B. (1) 110 °F
(2) prevent Thermal Hydraulic Instabilities (THI)

C. (1) 120 °F
(2) preclude emergency RPV depressurization

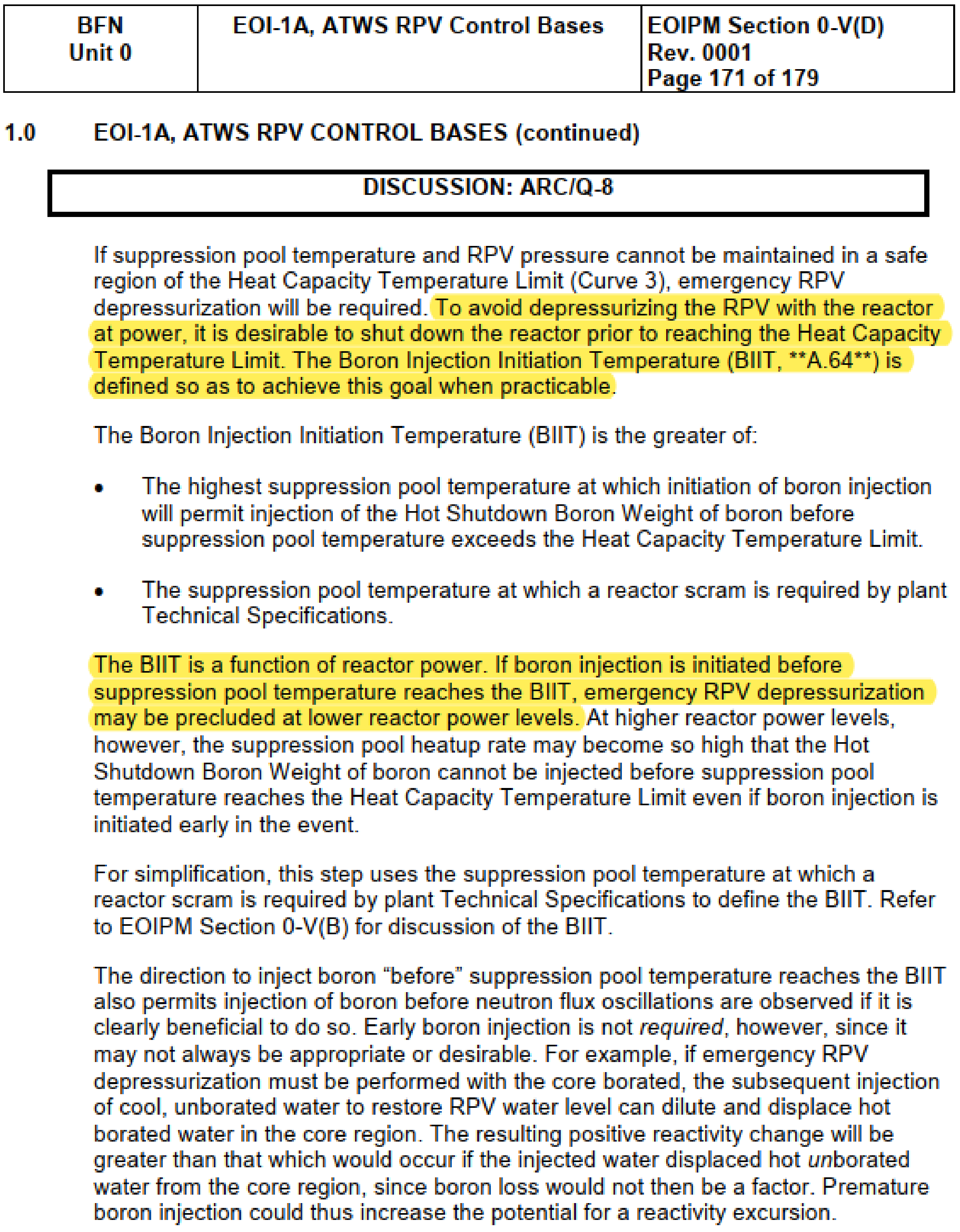
D. (1) 120 °F
(2) prevent Thermal Hydraulic Instabilities (THI)

Answer: A

Excerpt from 2-EOI-1A:



Excerpt from EOIPM 0-V(D):



BFN **Unit 0**	**EOI-1A, ATWS RPV Control Bases**	**EOIPM Section 0-V(D)** **Rev. 0001** **Page 171 of 179**

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:

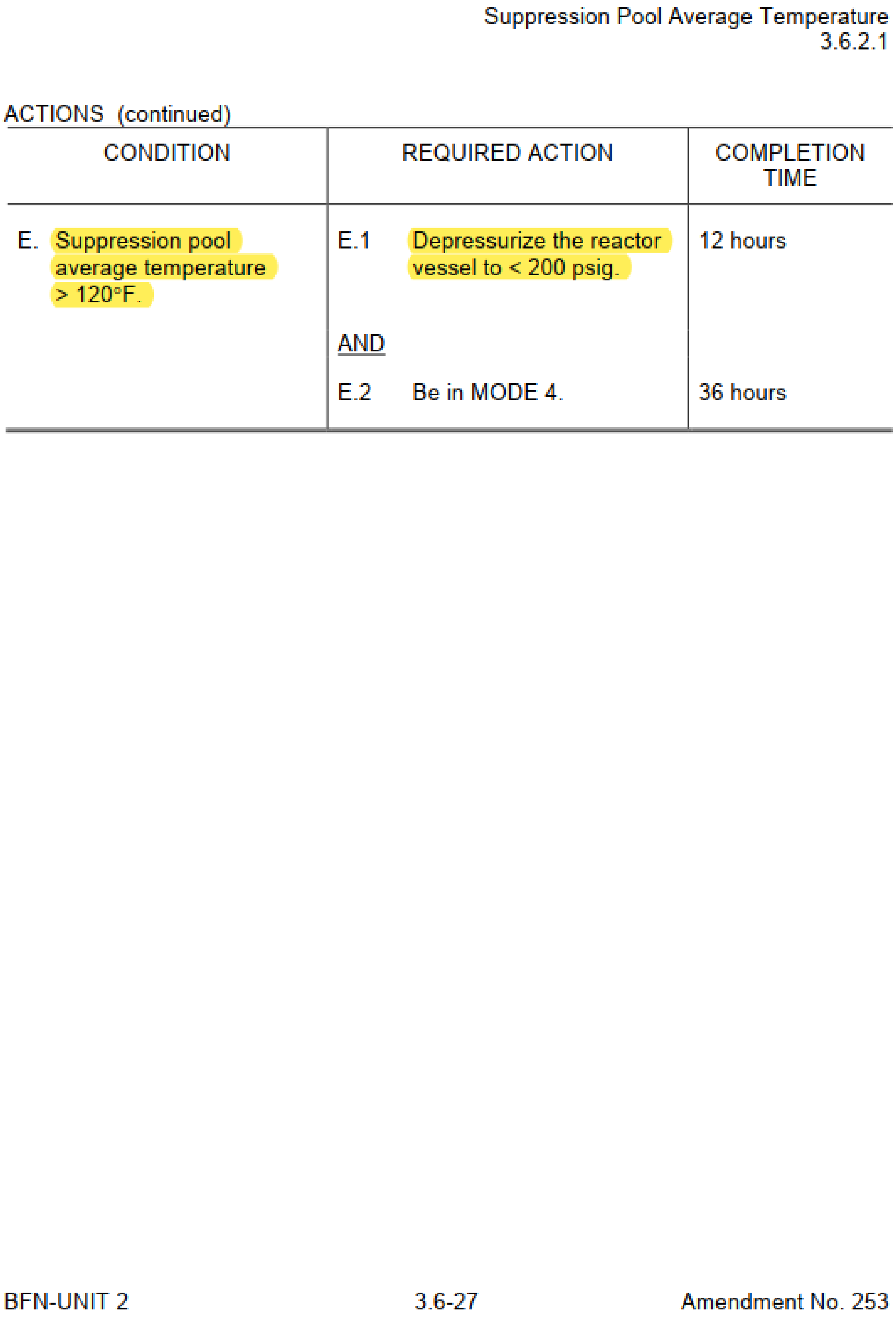
- The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.
- The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

The BIIT is a function of reactor power. If boron injection is initiated before suppression pool temperature reaches the BIIT, emergency RPV depressurization may be precluded at lower reactor power levels. At higher reactor power levels, however, the suppression pool heatup rate may become so high that the Hot Shutdown Boron Weight of boron cannot be injected before suppression pool temperature reaches the Heat Capacity Temperature Limit even if boron injection is initiated early in the event.

For simplification, this step uses the suppression pool temperature at which a reactor scram is required by plant Technical Specifications to define the BIIT. Refer to EOIPM Section 0-V(B) for discussion of the BIIT.

The direction to inject boron "before" suppression pool temperature reaches the BIIT also permits injection of boron before neutron flux oscillations are observed if it is clearly beneficial to do so. Early boron injection is not *required*, however, since it may not always be appropriate or desirable. For example, if emergency RPV depressurization must be performed with the core borated, the subsequent injection of cool, unborated water to restore RPV water level can dilute and displace hot borated water in the core region. The resulting positive reactivity change will be greater than that which would occur if the injected water displaced hot *un*borated water from the core region, since boron loss would not then be a factor. Premature boron injection could thus increase the potential for a reactivity excursion.

Excerpt from U2 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

BFN-UNIT 2 3.6-27 Amendment No. 253

Examination Outline Cross-reference:

295031 (EPE 8) Reactor Low Water Level / 2

EK2.06 (10CFR 55.41.7)

Knowledge of the relationship between Reactor Low Water Level and the following systems or components:

- High-pressure coolant injection (HPCI)

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295031EK2.06	
Importance Rating	4.1	------

Proposed Question: **# 30**

Unit 2 is operating at 100% RTP, with the following plant conditions:

- HPCI is operating for a surveillance as follows:
    - 2-FCV-73-35, HPCI PUMP CST TEST VALVE, is THROTTLED OPEN
    - 2-FCV-73-36, HPCI/RCIC CST TEST VALVE, is OPEN
    - 2-FCV-73-44, HPCI PUMP INJECTION VALVE, is CLOSED
- An event occurs resulting in a Reactor SCRAM
- Reactor Water Level indicates (-) 50 inches

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

2-FCV-73-36, HPCI/RCIC CST TEST VALVE will **(1)**.

2-FCV-73-44, HPCI PUMP INJECTION VALVE will **(2)**.

A. (1) remain OPEN
(2) travel OPEN

B. (1) remain OPEN
(2) remain CLOSED

C. (1) travel CLOSED
(2) travel OPEN

D. (1) travel CLOSED
(2) remain CLOSED

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that when HPCI automatically initiates, multiple valves close and multiple valves will open in accordance with 2-OI-73, HPCI System. Additionally, given this combination, a flow path for the system would be retained since the aligned surveillance has it already pumping water, potentially at rated flow. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible when HPCI automatically initiates, multiple valves close and multiple valves will open in accordance with 2-OI-73, HPCI System. Additionally, given this combination, a candidate may assume one other valve must reposition first, as does the suction valves before allowing them to reposition. This is the case for Suppression Pool suction valves where both must be full open prior to the Condensate Storage Tank suction path automatically closing.

C **CORRECT**: *(See attached)* In accordance with 2-OI-73, HPCI System, HPCI automatically initiates on Low Reactor Water Level at (-) 45 inches. The test flow path isolates itself upon an automatic initiation signal and is interlocked closed. If open (as given), 2-FCV-73-36, HPCI/RCIC CST TEST VALVE, will automatically travel CLOSED. For second part, 2-FCV-73-44, HPCI PUMP INJECTION VALVE is a normally closed valve that along with a check valve prevents over pressurizing the system and acts as a boundary to the vessel while testing. Upon reaching automatic initiation of (-) 45 inches Reactor Water Level, HPCI will align itself automatically and 2-FCV-73-44, HPCI PUMP INJECTION VALVE with travel OPEN.

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 relationship between Low Reactor Water Level and HPCI as it relates to automatic initiation setpoints and valve interlocks. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-OI-73, Rev. 101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

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

Question Source:

Bank #	OPL171.042-04.007 #1386	
Modified Bank #		(Note changes or attach parent)
New		

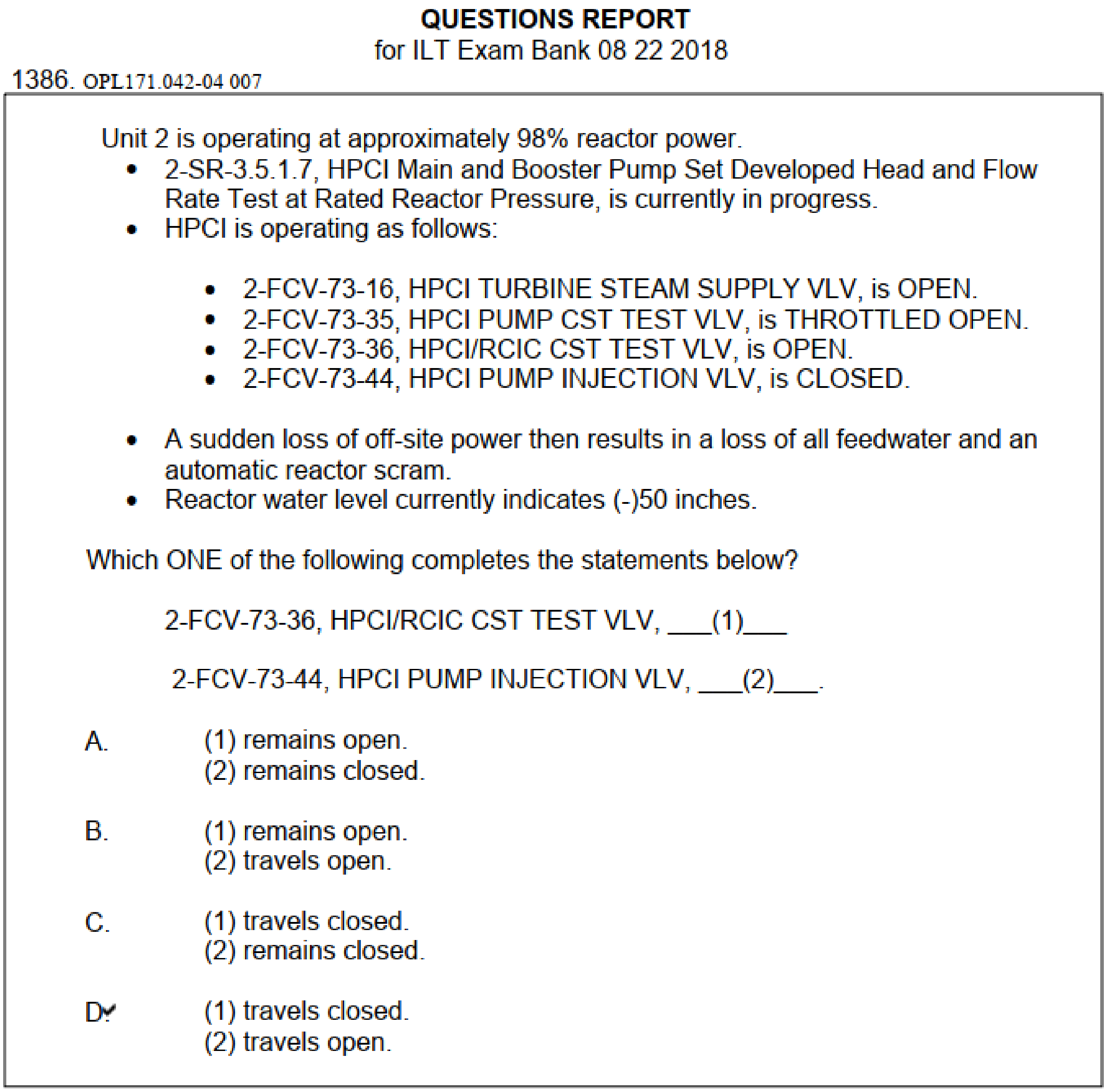
Question History:

Last NRC Exam	

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

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

Copy of Bank Question:



QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1386. OPL171.042-04 007

Unit 2 is operating at approximately 98% reactor power.

- 2-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure, is currently in progress.
- HPCI is operating as follows:
    - 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, is OPEN.
    - 2-FCV-73-35, HPCI PUMP CST TEST VLV, is THROTTLED OPEN.
    - 2-FCV-73-36, HPCI/RCIC CST TEST VLV, is OPEN.
    - 2-FCV-73-44, HPCI PUMP INJECTION VLV, is CLOSED.
- A sudden loss of off-site power then results in a loss of all feedwater and an automatic reactor scram.
- Reactor water level currently indicates (-)50 inches.

Which ONE of the following completes the statements below?

2-FCV-73-36, HPCI/RCIC CST TEST VLV, ___(1)___

2-FCV-73-44, HPCI PUMP INJECTION VLV, ___(2)___.

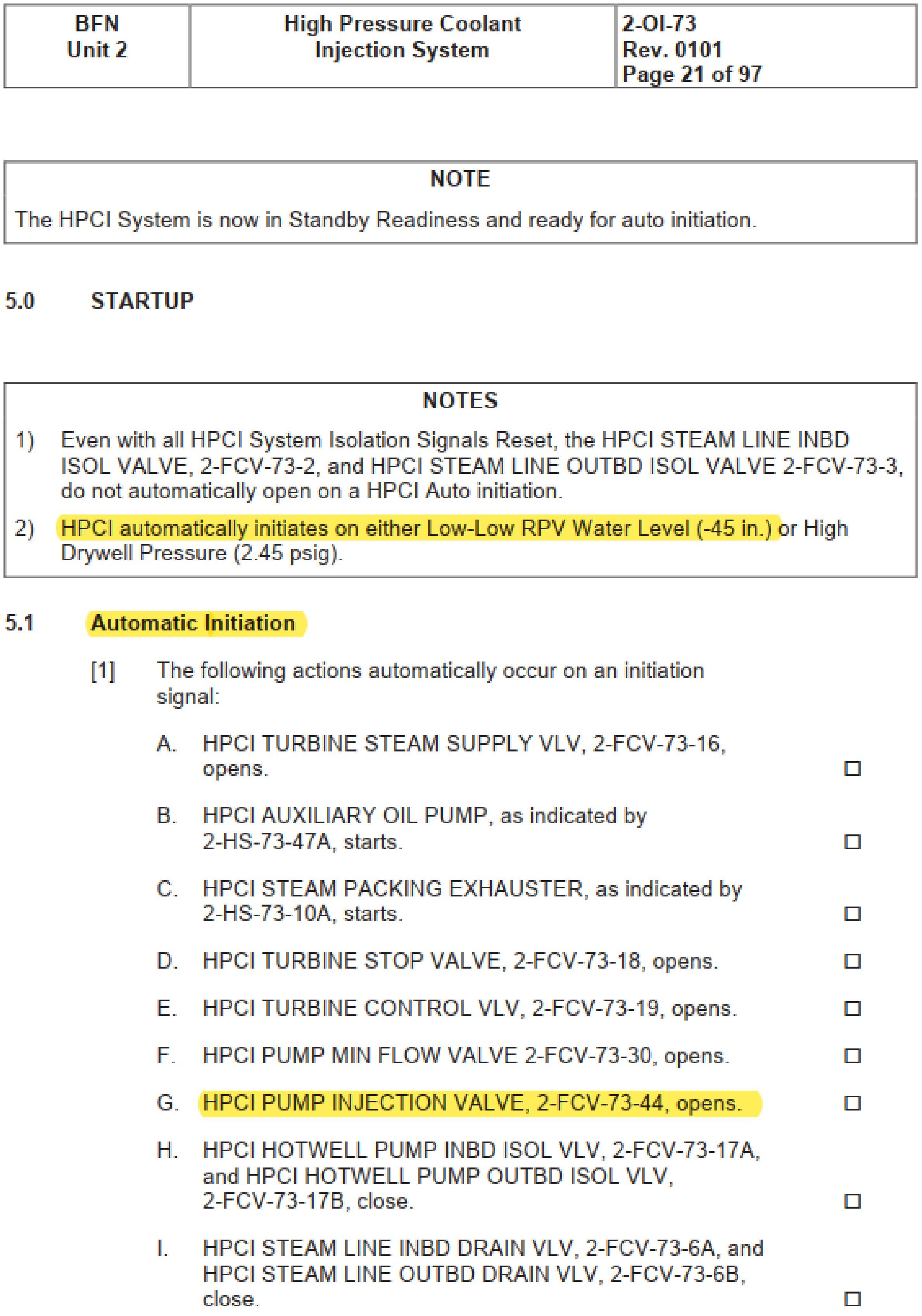
A. (1) remains open.
(2) remains closed.

B. (1) remains open.
(2) travels open.

C. (1) travels closed.
(2) remains closed.

D✓ (1) travels closed.
(2) travels open.

Excerpts from 2-OI-73:



BFN Unit 2	**High Pressure Coolant Injection System**	**2-OI-73 Rev. 0101 Page 21 of 97**

NOTE

The HPCI System is now in Standby Readiness and ready for auto initiation.

5.0 STARTUP

NOTES

1) Even with all HPCI System Isolation Signals Reset, the HPCI STEAM LINE INBD ISOL VALVE, 2-FCV-73-2, and HPCI STEAM LINE OUTBD ISOL VALVE 2-FCV-73-3, do not automatically open on a HPCI Auto initiation.

2) HPCI automatically initiates on either Low-Low RPV Water Level (-45 in.) or High Drywell Pressure (2.45 psig).

5.1 Automatic Initiation

[1] The following actions automatically occur on an initiation signal:

A. HPCI TURBINE STEAM SUPPLY VLV, 2-FCV-73-16, opens. ☐

B. HPCI AUXILIARY OIL PUMP, as indicated by 2-HS-73-47A, starts. ☐

C. HPCI STEAM PACKING EXHAUSTER, as indicated by 2-HS-73-10A, starts. ☐

D. HPCI TURBINE STOP VALVE, 2-FCV-73-18, opens. ☐

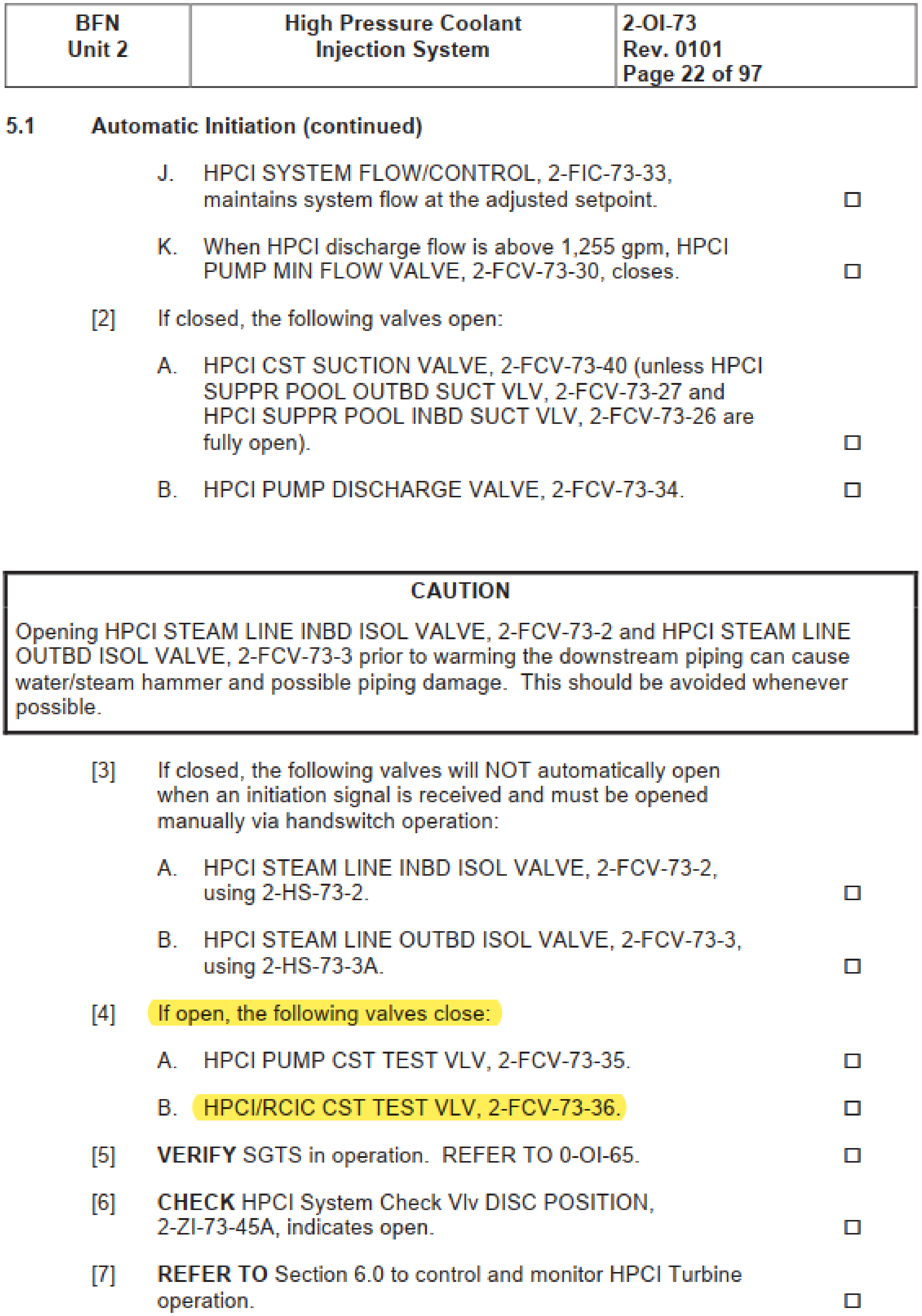
E. HPCI TURBINE CONTROL VLV, 2-FCV-73-19, opens. ☐

F. HPCI PUMP MIN FLOW VALVE 2-FCV-73-30, opens. ☐

G. HPCI PUMP INJECTION VALVE, 2-FCV-73-44, opens. ☐

H. HPCI HOTWELL PUMP INBD ISOL VLV, 2-FCV-73-17A, and HPCI HOTWELL PUMP OUTBD ISOL VLV, 2-FCV-73-17B, close. ☐

I. HPCI STEAM LINE INBD DRAIN VLV, 2-FCV-73-6A, and HPCI STEAM LINE OUTBD DRAIN VLV, 2-FCV-73-6B, close. ☐



BFN **Unit 2**	**High Pressure Coolant** **Injection System**	**2-OI-73** **Rev. 0101** **Page 22 of 97**

5.1 Automatic Initiation (continued)

J. HPCI SYSTEM FLOW/CONTROL, 2-FIC-73-33, maintains system flow at the adjusted setpoint. ☐

K. When HPCI discharge flow is above 1,255 gpm, HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, closes. ☐

[2] If closed, the following valves open:

A. HPCI CST SUCTION VALVE, 2-FCV-73-40 (unless HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27 and HPCI SUPPR POOL INBD SUCT VLV, 2-FCV-73-26 are fully open). ☐

B. HPCI PUMP DISCHARGE VALVE, 2-FCV-73-34. ☐

CAUTION

Opening HPCI STEAM LINE INBD ISOL VALVE, 2-FCV-73-2 and HPCI STEAM LINE OUTBD ISOL VALVE, 2-FCV-73-3 prior to warming the downstream piping can cause water/steam hammer and possible piping damage. This should be avoided whenever possible.

[3] If closed, the following valves will NOT automatically open when an initiation signal is received and must be opened manually via handswitch operation:

A. HPCI STEAM LINE INBD ISOL VALVE, 2-FCV-73-2, using 2-HS-73-2. ☐

B. HPCI STEAM LINE OUTBD ISOL VALVE, 2-FCV-73-3, using 2-HS-73-3A. ☐

[4] If open, the following valves close:

A. HPCI PUMP CST TEST VLV, 2-FCV-73-35. ☐

B. HPCI/RCIC CST TEST VLV, 2-FCV-73-36. ☐

[5] **VERIFY** SGTS in operation. REFER TO 0-OI-65. ☐

[6] **CHECK** HPCI System Check Vlv DISC POSITION, 2-ZI-73-45A, indicates open. ☐

[7] **REFER TO** Section 6.0 to control and monitor HPCI Turbine operation. ☐

Examination Outline Cross-reference:	Level	RO	SRO
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection **K3.04** (10CFR 55.41.7)	Tier #	2	------
Knowledge of the effect that a loss or malfunction of the High-Pressure Coolant Injection System will have on the following systems or system parameters:	Group #	1	------
	K/A #	206000K3.04	
- Reactor Power	Importance Rating	3.6	------

Proposed Question: **# 31**

Unit 3 was operating at 100% RTP when HPCI initiates and injects on an invalid signal.

Given the conditions above, Reactor Power as monitored on the Average Power Range Monitors (APRMs) on Panel 3-9-5 will ___**(1)**___ and **INDICATED** Total Steam Flow on Panel 3-9-5 will ___**(2)**___.

A. (1) rise
(2) rise

B. (1) rise
(2) lower

C. (1) lower
(2) rise

D. (1) lower
(2) lower

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** First part is correct *(See B).* Second part is incorrect in that HPCI steam supply comes from 'B' Main Steam line which taps off before the 4 Main Steam line flow element indications on Panel 3-9-5. The candidate could easily confuse this configuration to conclude a rise in indicated total steam flow on Panel 3-9-5 when HPCI initiates.

B **CORRECT**: *(See attached OE)* In accordance with 3-OI-6, Feedwater Heating and Misc Drains System, removing Feedwater heaters from service results in higher Reactor Power due to rising moderation of neutrons. When HPCI initiates steam is leaving the steam cycle, less steam is going to Feedwater heating, thus the Feedwater Temperature is lower. Average Power Range Monitors (APRMs) on Panel 3-9-5 in the Main Control Room is real Power as sensed by Local Power Range Monitor (LPRM) detectors. The colder Feedwater Temperature results in a rise in Reactor Power.
For second part, indicated Total Steam Flow on Panel 3-9-5 will lower when HPCI initiates. With Total Steam Flow coming from the 4 Main Steam line flow element indications on Panel 3-9-5, each tap off after the HPCI steam supply. HPCI is supplied from 'B' Main Steam line and the 4 Main Steam line flow indications each represent A, B, C, D lines respectively.

C INCORRECT**:** First part is incorrect but plausible in that Reactor Power Heat Balance will lower. BFN HPCI Inadvertent Injection OE *(see attached)* indicates Heat Balance Power lowered although actual Reactor Power rose since Heat Balance saw lower Steam Flow. It also saw lower Feedwater Temperatures but the lower Steam Flow was a larger variable and had more impact, thus indicated Reactor Power lowered. 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 effect on both actual and indicated Reactor Power during an inadvertent HPCI injection while the Reactor is at full Power. 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): 3-OI-6, Rev. 84 (Attach if not previously provided)
3-OI-3, Rev. 114
3-47E812-1, Rev. 73
OPL171.042, Rev. 24U1
OPL171.012, Rev. 16

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.042 Obj. 7 (As available)

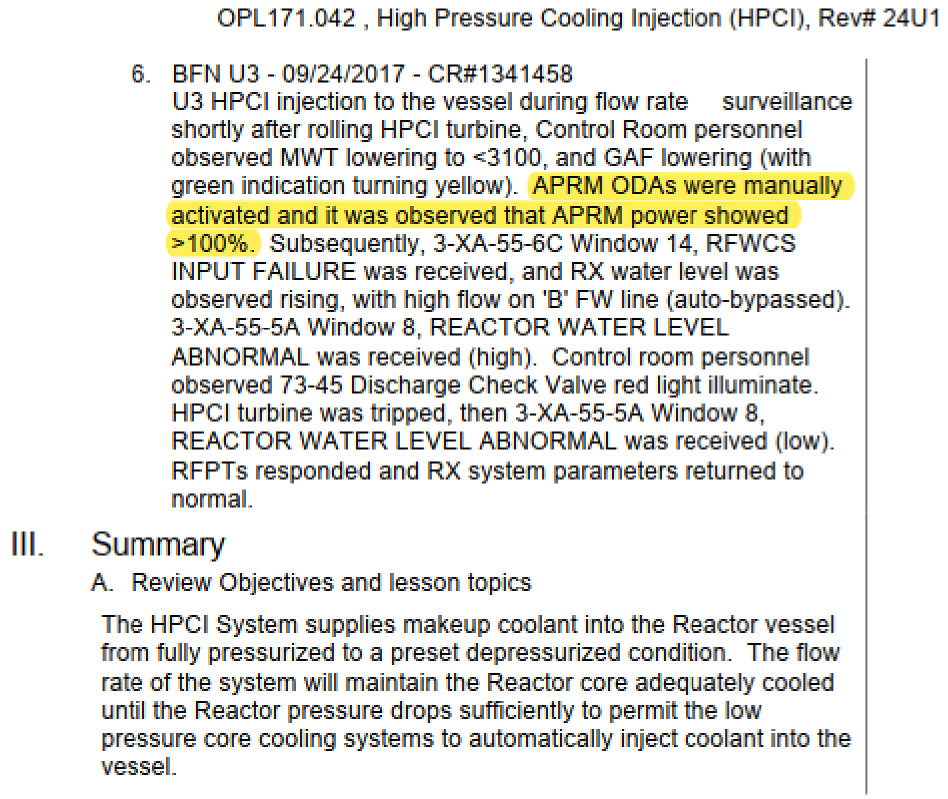
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

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

Comments:

Excerpt from OPL171.042 Lesson Plan:



OPL171.042 , High Pressure Cooling Injection (HPCI), Rev# 24U1

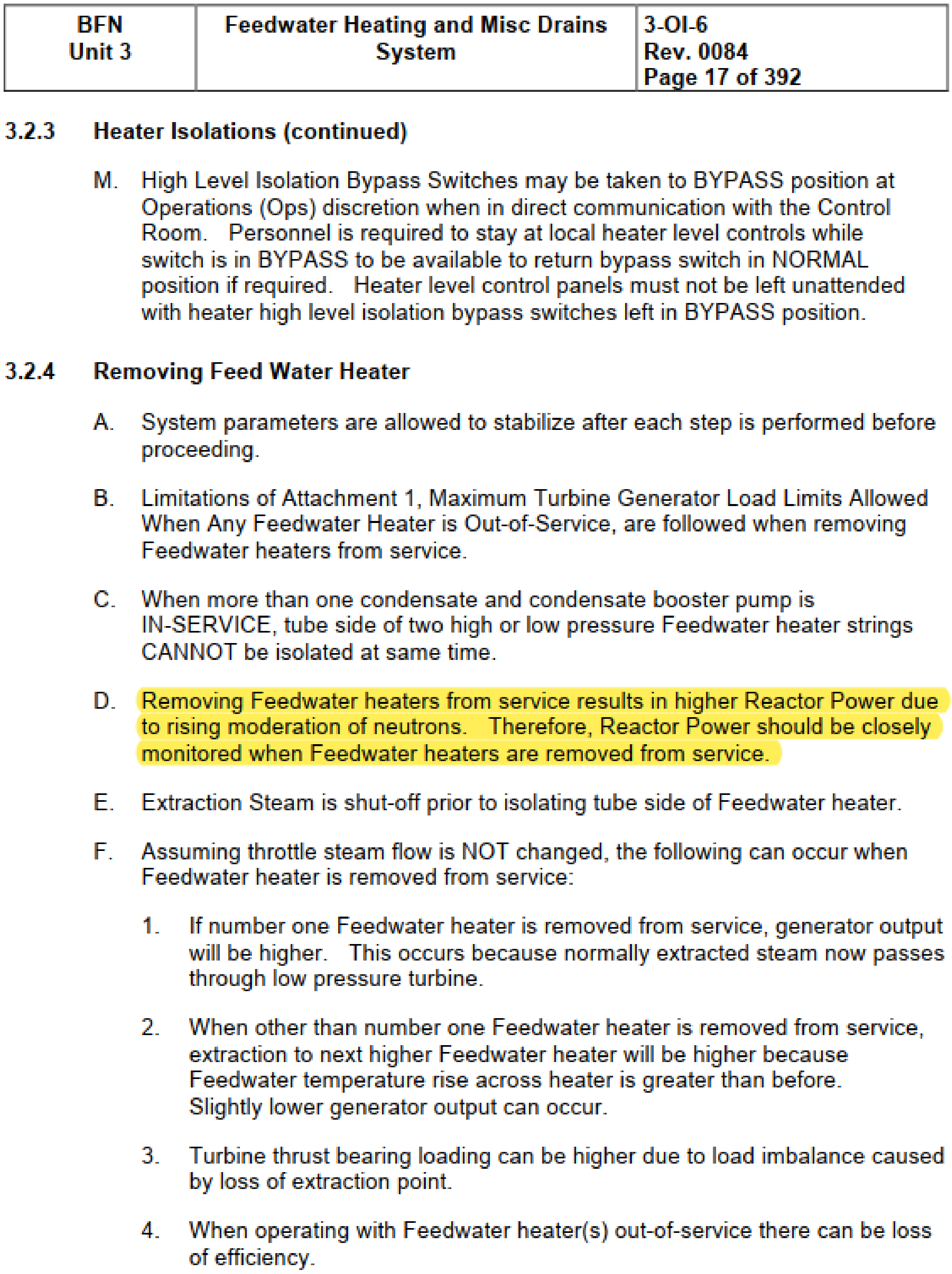
6. BFN U3 - 09/24/2017 - CR#1341458
U3 HPCI injection to the vessel during flow rate surveillance shortly after rolling HPCI turbine, Control Room personnel observed MWT lowering to <3100, and GAF lowering (with green indication turning yellow). APRM ODAs were manually activated and it was observed that APRM power showed >100%. Subsequently, 3-XA-55-6C Window 14, RFWCS INPUT FAILURE was received, and RX water level was observed rising, with high flow on 'B' FW line (auto-bypassed). 3-XA-55-5A Window 8, REACTOR WATER LEVEL ABNORMAL was received (high). Control room personnel observed 73-45 Discharge Check Valve red light illuminate. HPCI turbine was tripped, then 3-XA-55-5A Window 8, REACTOR WATER LEVEL ABNORMAL was received (low). RFPTs responded and RX system parameters returned to normal.

III. Summary

A. Review Objectives and lesson topics

The HPCI System supplies makeup coolant into the Reactor vessel from fully pressurized to a preset depressurized condition. The flow rate of the system will maintain the Reactor core adequately cooled until the Reactor pressure drops sufficiently to permit the low pressure core cooling systems to automatically inject coolant into the vessel.

Excerpt from 3-OI-6:



BFN **Unit 3**	**Feedwater Heating and Misc Drains System**	**3-OI-6** **Rev. 0084** **Page 17 of 392**

3.2.3 Heater Isolations (continued)

M. High Level Isolation Bypass Switches may be taken to BYPASS position at Operations (Ops) discretion when in direct communication with the Control Room. Personnel is required to stay at local heater level controls while switch is in BYPASS to be available to return bypass switch in NORMAL position if required. Heater level control panels must not be left unattended with heater high level isolation bypass switches left in BYPASS position.

3.2.4 Removing Feed Water Heater

A. System parameters are allowed to stabilize after each step is performed before proceeding.

B. Limitations of Attachment 1, Maximum Turbine Generator Load Limits Allowed When Any Feedwater Heater is Out-of-Service, are followed when removing Feedwater heaters from service.

C. When more than one condensate and condensate booster pump is IN-SERVICE, tube side of two high or low pressure Feedwater heater strings CANNOT be isolated at same time.

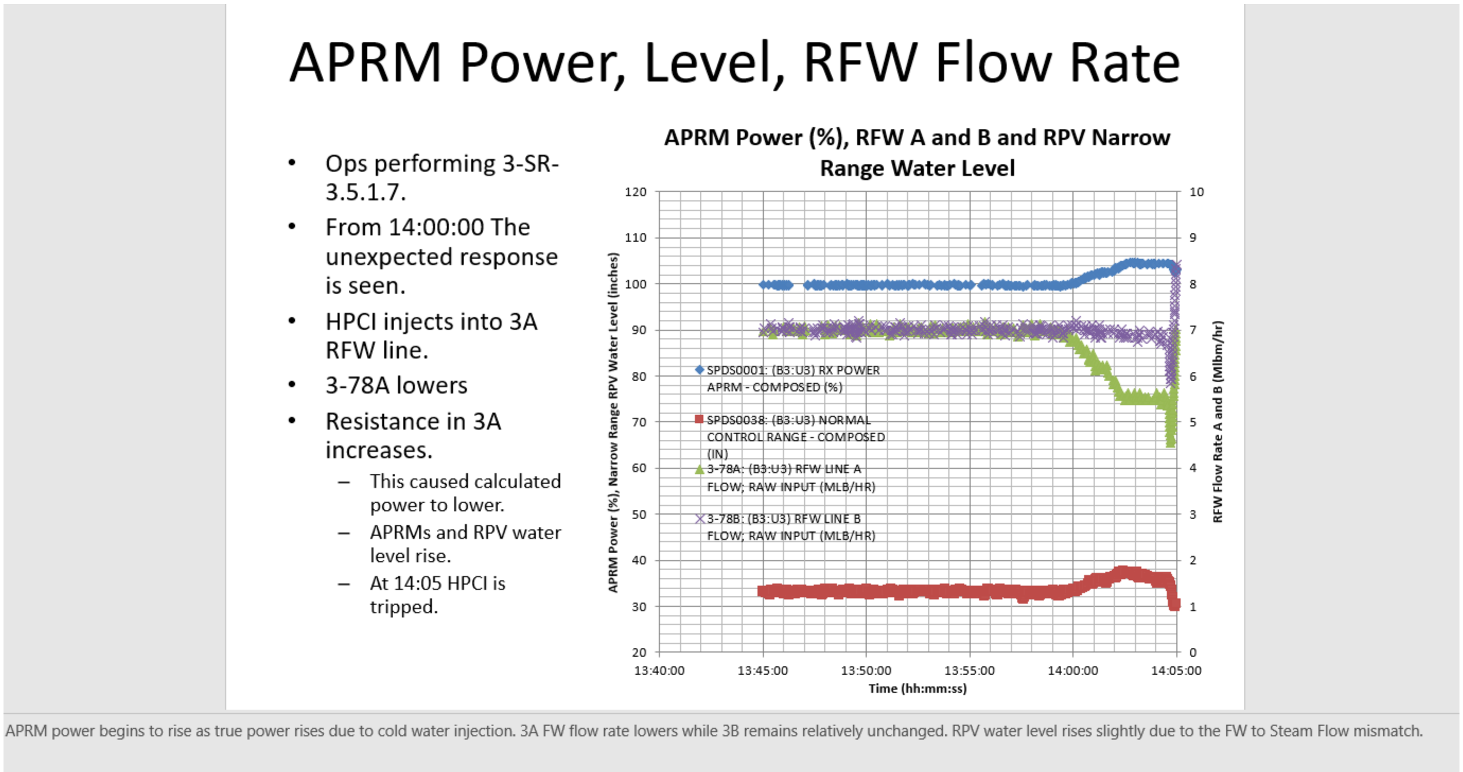
D. Removing Feedwater heaters from service results in higher Reactor Power due to rising moderation of neutrons. Therefore, Reactor Power should be closely monitored when Feedwater heaters are removed from service.

E. Extraction Steam is shut-off prior to isolating tube side of Feedwater heater.

F. Assuming throttle steam flow is NOT changed, the following can occur when Feedwater heater is removed from service:

1. If number one Feedwater heater is removed from service, generator output will be higher. This occurs because normally extracted steam now passes through low pressure turbine.
2. When other than number one Feedwater heater is removed from service, extraction to next higher Feedwater heater will be higher because Feedwater temperature rise across heater is greater than before. Slightly lower generator output can occur.
3. Turbine thrust bearing loading can be higher due to load imbalance caused by loss of extraction point.
4. When operating with Feedwater heater(s) out-of-service there can be loss of efficiency.
]

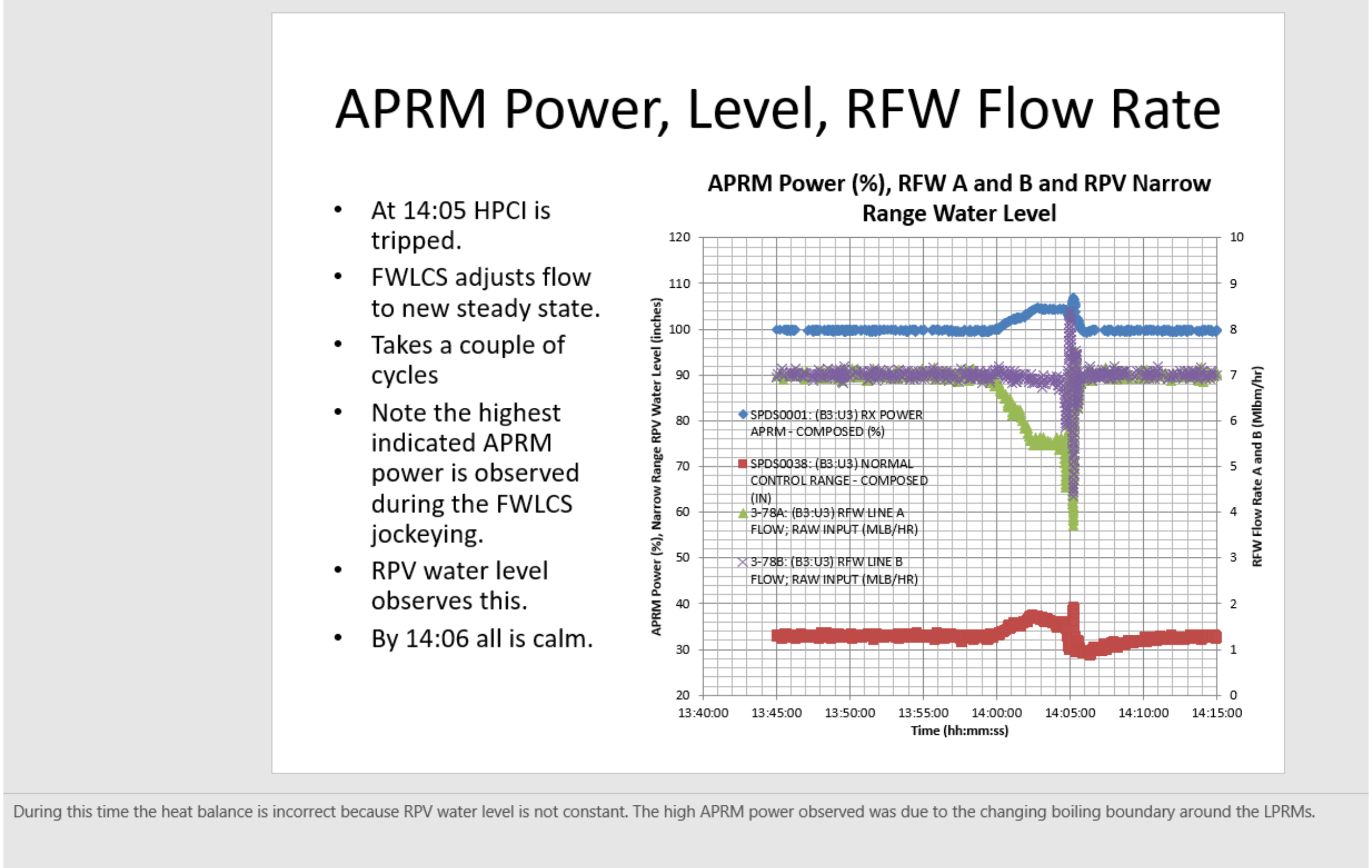
Excerpts from HPCI Inadvertent Injection analysis OE: Illustrates rise in APRM Power



APRM Power, Level, RFW Flow Rate

- Ops performing 3-SR-3.5.1.7.
- From 14:00:00 The unexpected response is seen.
- HPCI injects into 3A RFW line.
- 3-78A lowers
- Resistance in 3A increases.
  - This caused calculated power to lower.
  - APRMs and RPV water level rise.
  - At 14:05 HPCI is tripped.

APRM power begins to rise as true power rises due to cold water injection. 3A FW flow rate lowers while 3B remains relatively unchanged. RPV water level rises slightly due to the FW to Steam Flow mismatch.

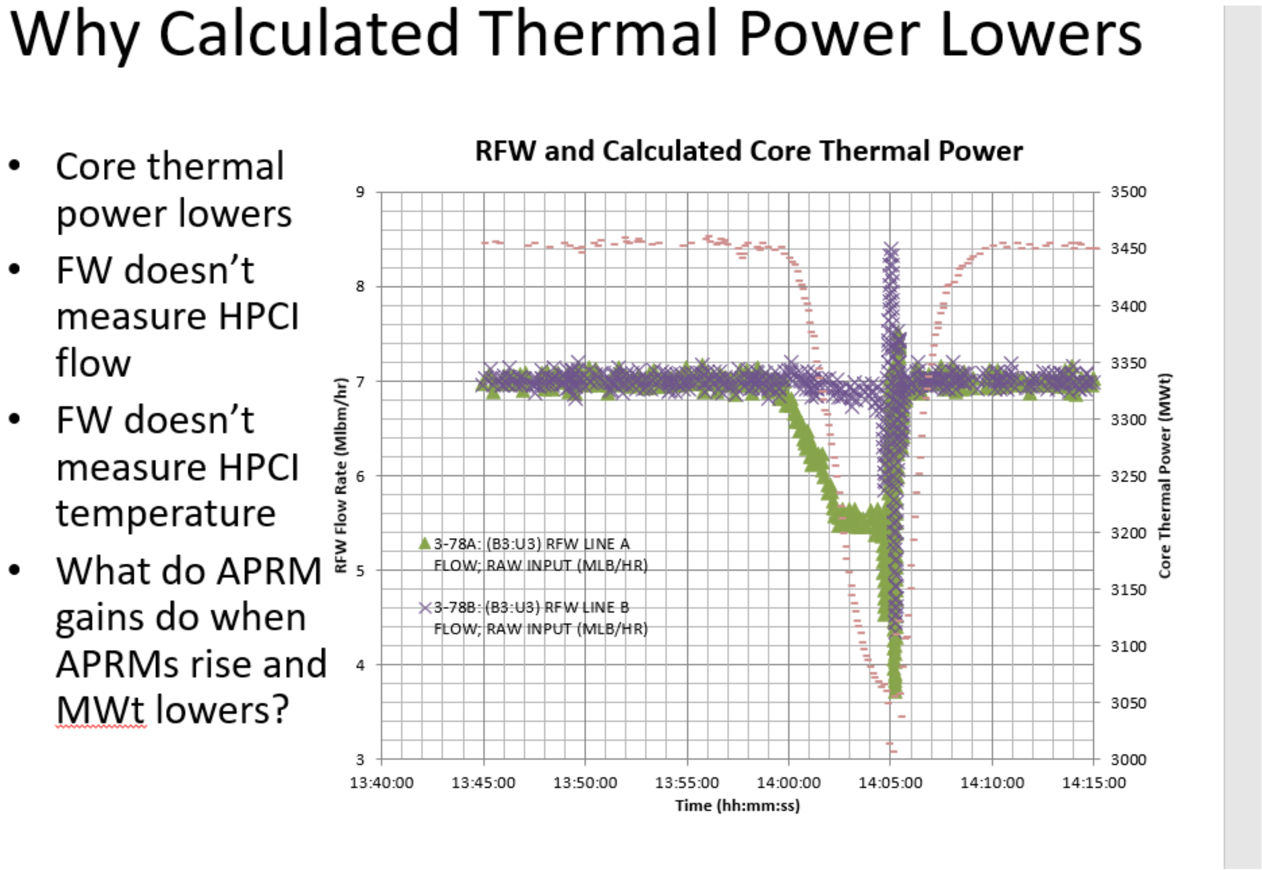


APRM Power, Level, RFW Flow Rate

- At 14:05 HPCI is tripped.
- FWLCS adjusts flow to new steady state.
- Takes a couple of cycles
- Note the highest indicated APRM power is observed during the FWLCS jockeying.
- RPV water level observes this.
- By 14:06 all is calm.

During this time the heat balance is incorrect because RPV water level is not constant. The high APRM power observed was due to the changing boiling boundary around the LPRMs.

Excerpt from HPCI Inadvertent Injection analysis OE: Supports Distractors C(1), D(1)



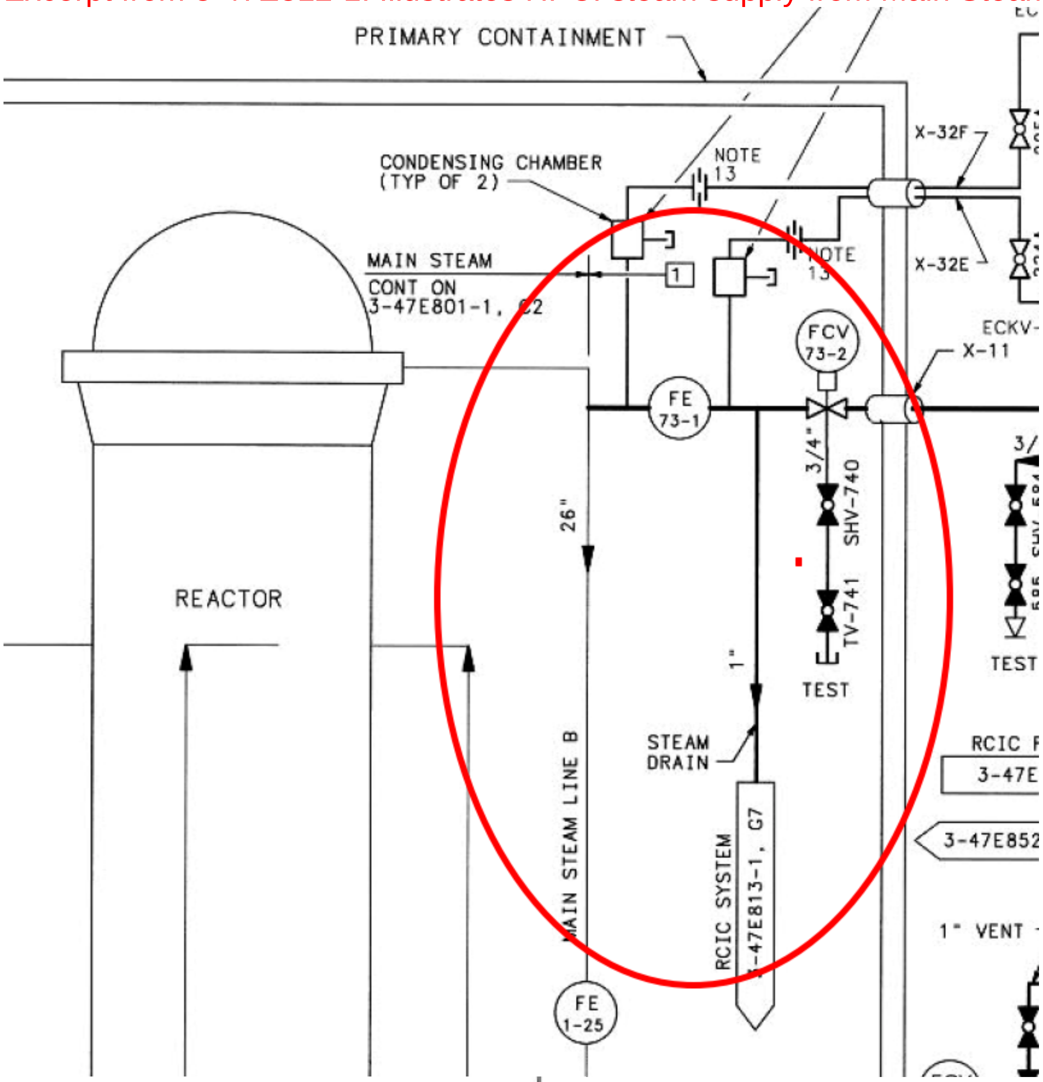
Why Calculated Thermal Power Lowers

- Core thermal power lowers
- FW doesn't measure HPCI flow
- FW doesn't measure HPCI temperature
- What do APRM gains do when APRMs rise and MWt lowers?

Takeaways

- When performing tests EXPECT Mwe and 1st pressure to lower and not increase until test is finished.
- Be aware of HPCI discharge pressure above RPV dome pressure. This is when inadvertent injection is possible.
- HPCI injection will be seen by rising APRM power, rising Mwe, rising 1st stage pressure, and lowering core thermal power and APRM GAFs.

Excerpt from 3-47E812-1: Illustrates HPCI steam supply from Main Steam line B



073	ADMIN	D.Stanford	ML Poole	BCampbell	4-10-18
REVISED PER RIMS MEMO R14 180409 105 (REF: BFCR 1347898)					
REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE

SCALE: NONE EXCEPT AS NOTED

POWERHOUSE
UNIT 3

SYSTEM NO.
73

FLOW DIAGRAM
HIGH PRESSURE COOLANT
INJECTION SYSTEM

BROWNS FERRY NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY

DESIGN		DISCIPLINE INTERFACE		ENGINEERING APPROVAL
DRAWN RSH & HMS	CHECKED BEN	1 N/A	2 N/A	J.R. PARRISH
DESIGNED HMS	REVIEWED ROY E. LYON	3 N/A	4 N/A	

KNOXVILLE	DATE 4-20-70	67	M	3-47E812-1	R073

CAD MAINTAINED DRAWING

CCD

Excerpt from 3-OI-3:

BFN **Unit 3**	**Reactor Feedwater System**	**3-OI-3** **Rev. 0114** **Page 315 of 330**

Attachment 8
(Page 2 of 7)

RFWCS Instrumentation

1.4 Failure Mechanisms

When RFWCS declares a level signal bad or invalid, then level instrument is automatically bypassed (amber light illuminates on the instrument). This will cause annunciation RFWCS INPUT FAILURE (3-XA-55-6C, window 14) to alarm.

When all four level instruments become bad or invalid, then RFW Control System will automatically transfer to SINGLE ELEMENT control and Reactor Water Level Control PDS, 3-LIC-46-5, trips to MANUAL mode. Annunciation RFWCS FAILED TO MANUAL (3-XA-55-6C, window 21) will alarm when this condition occurs.

2.0 MAIN STEAM LINE FLOW AND TURBINE FIRST STAGE PRESSURE

2.1 Components

3-FI-46-1

3-FI-46-2

3-FI-46-3

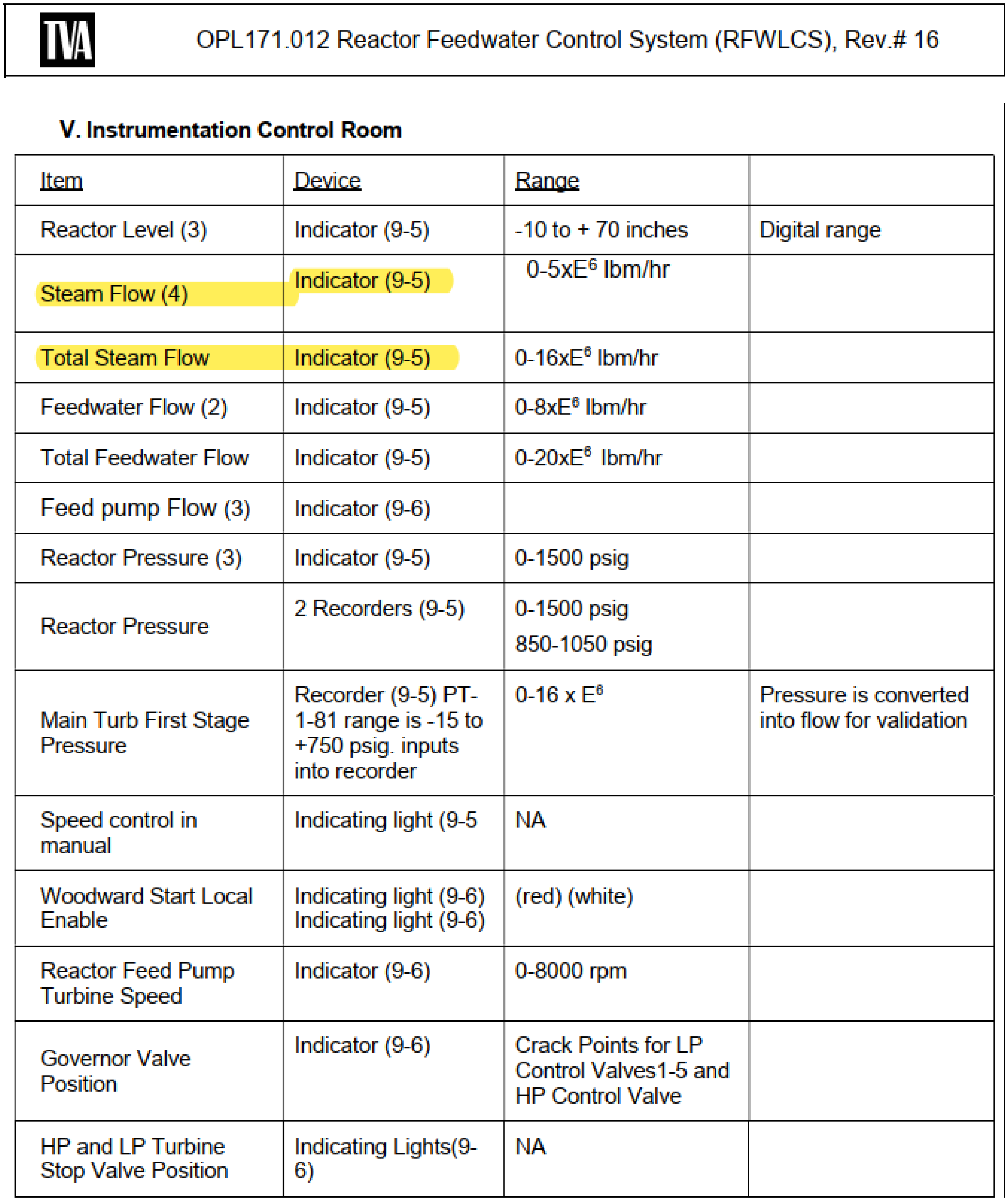
3-FI-46-4

2.2 Description

Instruments are located on Panel 3-9-5 but are manually bypassed at the Engineering Work Station in Unit 3 Computer Room. Steam line flows are compensated for changes in Reactor pressure. Flow signals are sent to the RFWC to produce a Total Steam Flow signal. Total Steam Flow is used for the following:

- Correcting level demand signal for RFWCS when system is in THREE ELEMENT control.
- Initiating Rod Worth Minimizer Low Power Alarm Point (<27% rated flow) and Enable setpoint (<22% rated flow).

Excerpt from OPL171.012 Lesson Plan:



OPL171.012 Reactor Feedwater Control System (RFWLCS), Rev.# 16

V. Instrumentation Control Room

Item	Device	Range	
Reactor Level (3)	Indicator (9-5)	-10 to + 70 inches	Digital range
Steam Flow (4)	Indicator (9-5)	0-5xE^6 lbm/hr	
Total Steam Flow	Indicator (9-5)	0-16xE^6 lbm/hr	
Feedwater Flow (2)	Indicator (9-5)	0-8xE^6 lbm/hr	
Total Feedwater Flow	Indicator (9-5)	0-20xE^6 lbm/hr	
Feed pump Flow (3)	Indicator (9-6)		
Reactor Pressure (3)	Indicator (9-5)	0-1500 psig	
Reactor Pressure	2 Recorders (9-5)	0-1500 psig 850-1050 psig	
Main Turb First Stage Pressure	Recorder (9-5) PT-1-81 range is -15 to +750 psig. inputs into recorder	0-16 x E^6	Pressure is converted into flow for validation
Speed control in manual	Indicating light (9-5	NA	
Woodward Start Local Enable	Indicating light (9-6) Indicating light (9-6)	(red) (white)	
Reactor Feed Pump Turbine Speed	Indicator (9-6)	0-8000 rpm	
Governor Valve Position	Indicator (9-6)	Crack Points for LP Control Valves1-5 and HP Control Valve	
HP and LP Turbine Stop Valve Position	Indicating Lights(9-6)	NA	

Examination Outline Cross-reference:	Level	RO	SRO
295021 (APE 21) Loss of Shutdown Cooling / 4	Tier #	1	------
AK1.03 (10CFR 55.41.9)	Group #	1	------
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to LOSS OF SHUTDOWN COOLING: • Adequate Core Cooling	K/A #	295021AK1.03	
	Importance Rating	4.4	------

Proposed Question: **# 32**

Preparations are underway to place Unit 2 in Cold Shutdown following a SCRAM when the following occurs:

- 2B RHR Pump was started for Shutdown Cooling (SDC)
- Reactor Water Level then lowered to 0 inches

Given the conditions above, SDC ___**(1)**___ isolate.

In accordance with 2-OI-74, Residual Heat Removal System (RHR), the normal Cold Shutdown Reactor Water Level band is maintained to prevent ___**(2)**___.

A. (1) will
   (2) thermal stratification

B. (1) will
   (2) jet pump cavitation

C. (1) will NOT
   (2) thermal stratification

D. (1) will NOT
   (2) jet pump cavitation

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 2-OI-74, RHR, if in Shutdown Cooling, a PCIS Group 2 Isolation occurs when Reactor Water Level lowers to (+) 2 inches. Given that Reactor Water Level lowered to 0 inches, Shutdown Cooling will isolate. For second part, 2-OI-74 states that Reactor Level Band should be maintained 70 to 90 inches to prevent thermal stratification with no forced flow due to allowing the core to communicate with the annulus and aiding in natural circulation.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible if the candidate confuses the Shutdown Cooling conditions given in the stem as indications of jet pump cavitation believing that higher Reactor Water Level will result in more margin to jet pump cavitation.

C INCORRECT: First part is incorrect but plausible in that Reactor Water Level 0 inches is below the setpoint of (+) 2 inches for the RHR Group 2 PCIS Isolation while also above the HPCI/RCIC automatic initiation setpoint of (-) 45 inches. 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 ensuring Adequate Core Cooling during a Loss of Shutdown Cooling event. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-OI-74, Rev. 189 (Attach if not previously provided)
OPL171.044, Rev. 22

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.044 Obj. 12 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1909 #36	(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:

Unit 1 is preparing for a Refueling outage with the following conditions:

- RHR SYS I FLOW is 7500 gpm for Shutdown Cooling (SDC)
- **NO** Recirc Pumps are in service
- Reactor Coolant Temperature is 175 °F
- **NO** other testing or evolutions are in progress

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

Given the conditions above, in accordance with 1-OI-74, Residual Heat Removal System, Reactor Water Level should be maintained ____**(1)**____.

The purpose of maintaining this SDC Reactor Water Level is to ____**(2)**____.

A. **(1)** < 70 inches
(2) prevent thermal stratification

B. **(1)** < 70 inches
(2) prevent jet pump cavitation

C. **(1)** 70 to 90 inches
(2) prevent thermal stratification

D. **(1)** 70 to 90 inches
(2) prevent jet pump cavitation

Excerpt from OPL171.044 Lesson Plan:

OPL171.044, Residual Heat Removal (RHR) System, Revision 22

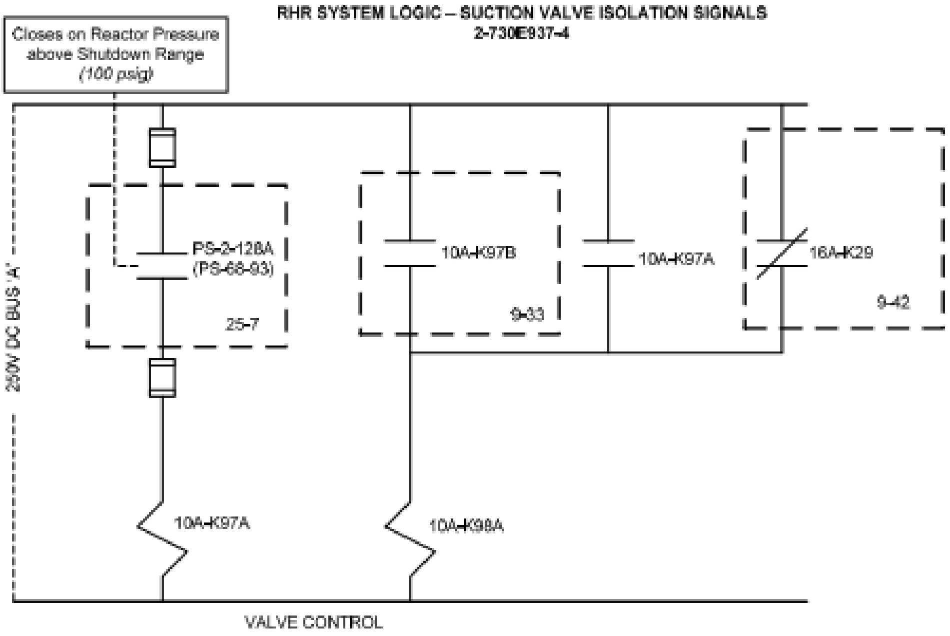


Figure 31 – RHR Shutdown Cooling Isolation Valve Isolation Logic

(2) Valve automatically closes (10A-K98A) on any one of the following signals:
   (a) Reactor Pressure >100 psig (10A-K97A/B)
   (b) PCIS Group 2 isolation signal (16A-K29)
      Drywell pressure 2.45 psig
      Reactor water level (+) 2 inches

Refer to OI-99 and AOI-74-1 for recovery actions for loss of RPS power if in Shutdown Cooling

(3) Switch HS-74-157 must be in SHUTDOWN position to operate FCV-74-48 from any location (Unit 1 and Unit 2 only).

(4) NORMAL/EMEREGNCY switch in EMERGENCY position allows operation at the breaker only and removes PCIS logic closure circuit.

Unit 3, 3-FCV-74-48 does not have a NORMAL / SHUTDOWN switch interlock

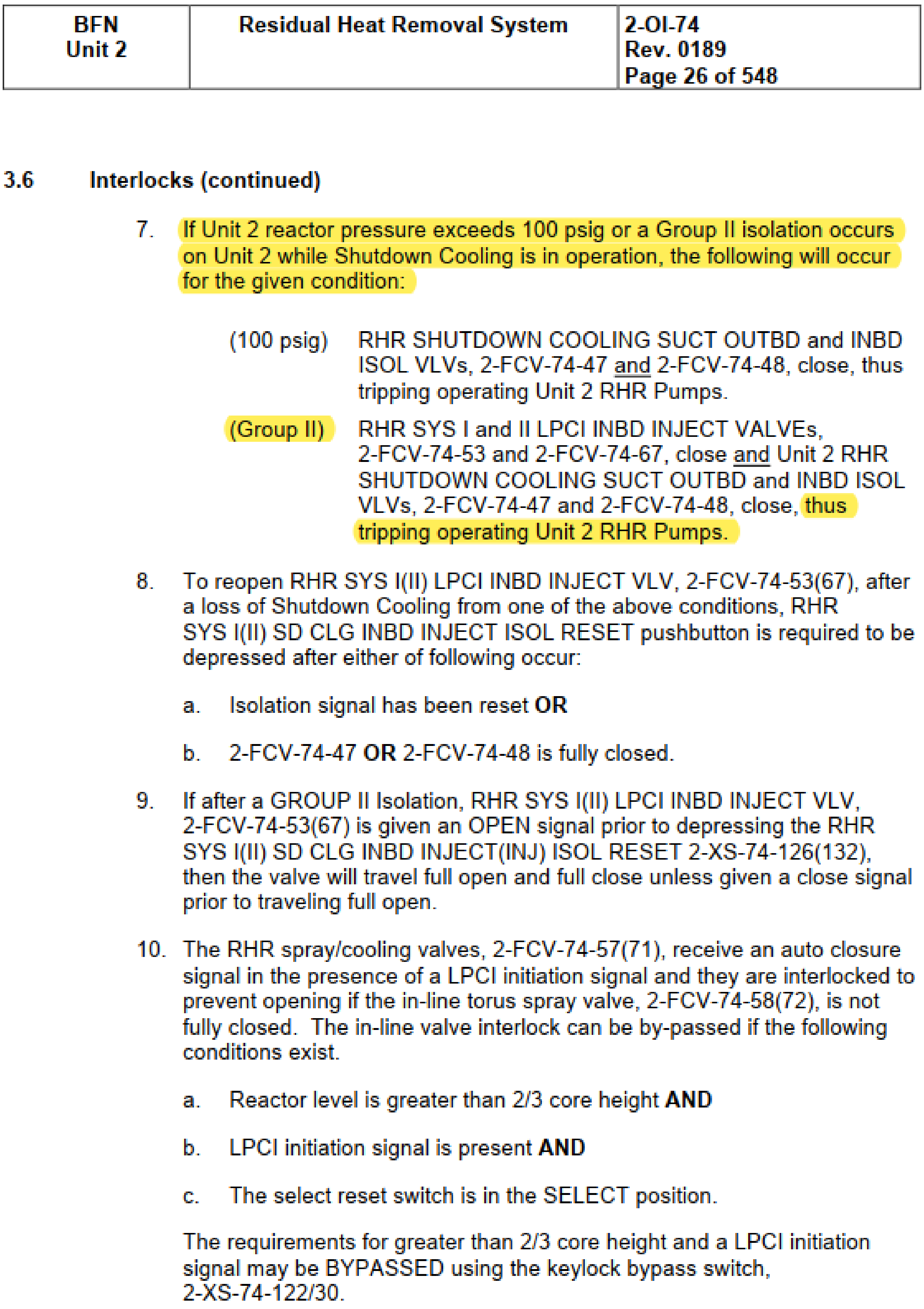
(5) Interlocks with RHR Pump – suction interlock – if RHR pump is running while in shutdown cooling, pump will trip if either 74-47 OR 74-48 leave their full open position.

LOR Objective 2
ILT Objective 6

i) RHR Pump Suction Cross-Tie Valves
   (1) Valves are labeled as follows:
      (a) Unit 1 – 1-FCV-74-98/99 (Loop II)
      (b) Unit 2 – 2-FCV-74-96/97 (Loop I)
      (c) Unit 2 – 2-FCV-74-98/99 (Loop II)
      (d) Unit 3 – 3-FCV-74-96/97 (Loop I)
   (2) No automatic opening or closing interlocks

ILT Objective 3k

Excerpts from 2-OI-74:



BFN **Unit 2**	**Residual Heat Removal System**	**2-OI-74** **Rev. 0189** **Page 26 of 548**

3.6 Interlocks (continued)

7. If Unit 2 reactor pressure exceeds 100 psig or a Group II isolation occurs on Unit 2 while Shutdown Cooling is in operation, the following will occur for the given condition:

(100 psig)	RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.
(Group II)	RHR SYS I and II LPCI INBD INJECT VALVEs, 2-FCV-74-53 and 2-FCV-74-67, close and Unit 2 RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.

8. To reopen RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT ISOL RESET pushbutton is required to be depressed after either of following occur:

   a. Isolation signal has been reset **OR**

   b. 2-FCV-74-47 **OR** 2-FCV-74-48 is fully closed.

9. If after a GROUP II Isolation, RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67) is given an OPEN signal prior to depressing the RHR SYS I(II) SD CLG INBD INJECT(INJ) ISOL RESET 2-XS-74-126(132), then the valve will travel full open and full close unless given a close signal prior to traveling full open.

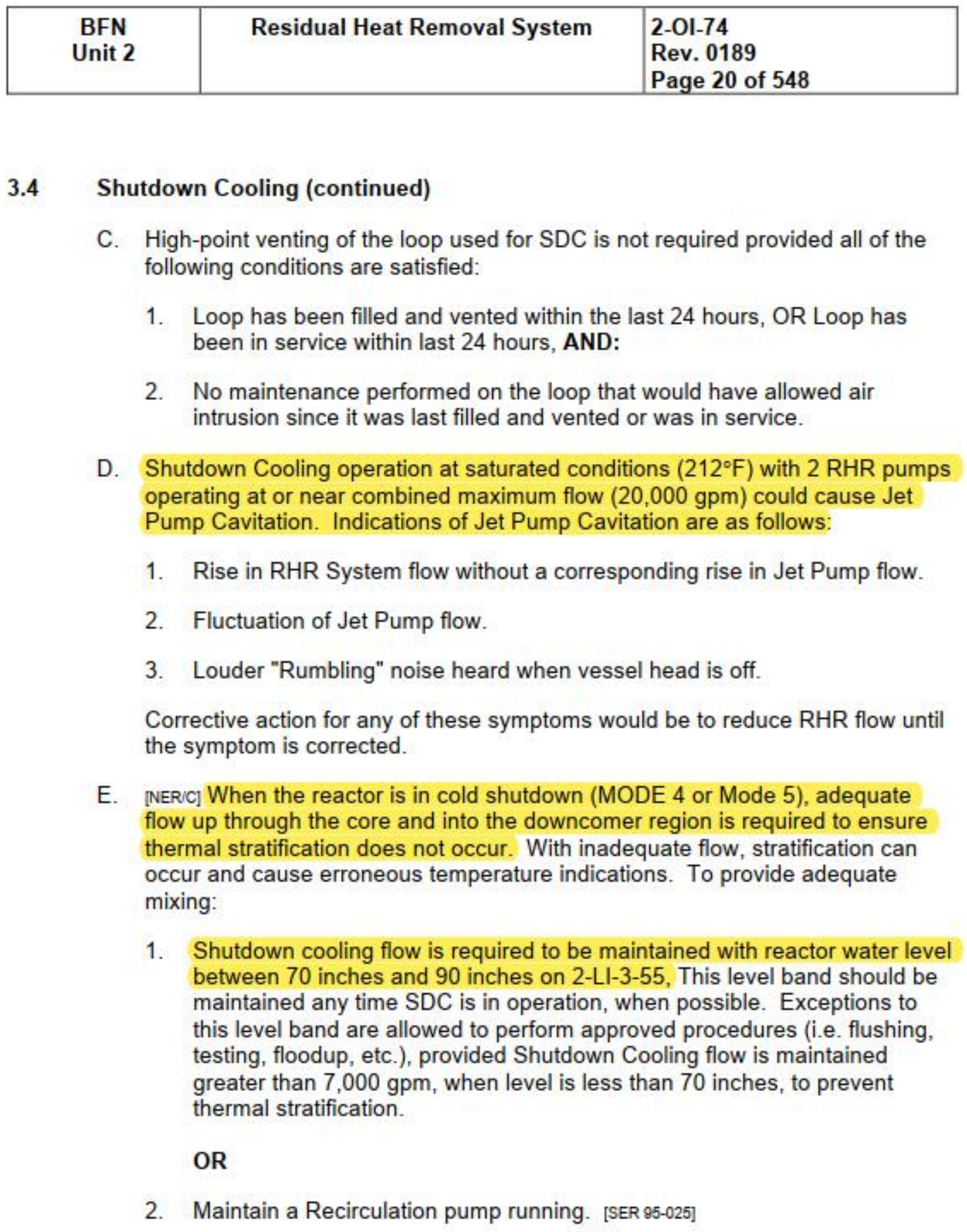
10. The RHR spray/cooling valves, 2-FCV-74-57(71), receive an auto closure signal in the presence of a LPCI initiation signal and they are interlocked to prevent opening if the in-line torus spray valve, 2-FCV-74-58(72), is not fully closed. The in-line valve interlock can be by-passed if the following conditions exist.

   a. Reactor level is greater than 2/3 core height **AND**

   b. LPCI initiation signal is present **AND**

   c. The select reset switch is in the SELECT position.

   The requirements for greater than 2/3 core height and a LPCI initiation signal may be BYPASSED using the keylock bypass switch, 2-XS-74-122/30.



BFN **Unit 2**	**Residual Heat Removal System**	**2-OI-74** **Rev. 0189** **Page 20 of 548**

3.4 Shutdown Cooling (continued)

C. High-point venting of the loop used for SDC is not required provided all of the following conditions are satisfied:

1. Loop has been filled and vented within the last 24 hours, OR Loop has been in service within last 24 hours, **AND:**

2. No maintenance performed on the loop that would have allowed air intrusion since it was last filled and vented or was in service.

D. Shutdown Cooling operation at saturated conditions (212°F) with 2 RHR pumps operating at or near combined maximum flow (20,000 gpm) could cause Jet Pump Cavitation. Indications of Jet Pump Cavitation are as follows:

1. Rise in RHR System flow without a corresponding rise in Jet Pump flow.

2. Fluctuation of Jet Pump flow.

3. Louder "Rumbling" noise heard when vessel head is off.

Corrective action for any of these symptoms would be to reduce RHR flow until the symptom is corrected.

E. [NER/C] When the reactor is in cold shutdown (MODE 4 or Mode 5), adequate flow up through the core and into the downcomer region is required to ensure thermal stratification does not occur. With inadequate flow, stratification can occur and cause erroneous temperature indications. To provide adequate mixing:

1. Shutdown cooling flow is required to be maintained with reactor water level between 70 inches and 90 inches on 2-LI-3-55, This level band should be maintained any time SDC is in operation, when possible. Exceptions to this level band are allowed to perform approved procedures (i.e. flushing, testing, floodup, etc.), provided Shutdown Cooling flow is maintained greater than 7,000 gpm, when level is less than 70 inches, to prevent thermal stratification.

OR

2. Maintain a Recirculation pump running. [SER 95-025]

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

Examination Outline Cross-reference:

209001 (SF2, SF4 LPCS) Low-Pressure Core Spray
K4.05 (10CFR 55.41.7)
Knowledge of Low-Pressure Core Spray System design features and/or interlocks that provide for the following:

- Pump minimum flow

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	209001K4.05	
Importance Rating	3.4	------

Proposed Question: **# 33**

In accordance with 2-OI-75, Core Spray System automatic interlocks, when system flow reaches the **SETPOINT** of ___**(1)**___ gpm, 2-FCV-75-9(37), CORE SPRAY SYS I(II) MINIMUM FLOW VALVEs OPEN and will CLOSE when system flow reaches the **SETPOINT** of ___**(2)**___ .

A. (1) 900 gpm
(2) 2600 gpm

B. (1) 900 gpm
(2) 5800 gpm

C. (1) 2200 gpm
(2) 2600 gpm

D. (1) 2200 gpm
(2) 5800 gpm

Proposed Answer: **C**

Explanation (Optional):		
	A	INCORRECT: First part is incorrect but plausible in that multiple safety systems at BFN each have their own flowrate setpoint to both open and close pump minimum flow valves. 900 gpm is the set point for HPCI minimum flow valve to open on a low flow condition. Second part is correct *(See C).*
	B	INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that 5800 gpm is the setpoint for the RHR minimum flow valve to open.
	C	**CORRECT**: *(See attached)* In accordance with 2-OI-75, Core Spray System, 2-FCV-75-9(37), CORE SPRAY SYS I(II) MINIMUM FLOW VALVEs will automatically open when flow reaches 2200 gpm lowering. For second part, Core Spray minimum flow valves will auto close when system flow reaches 2600 gpm rising.
	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 Core Spray System design features and pump minimum flow valve automatic interlocks. This question is rated as memory due to strictly recalling facts related to the Core Spray System design features.

Technical Reference(s): 2-OI-75, Rev. 118 (Attach if not previously provided)

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

Proposed references to be provided to applicants during examination: **NONE**

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

Question Source:	Bank #		
	Modified Bank #	OPL171.045-06 009 #1663	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

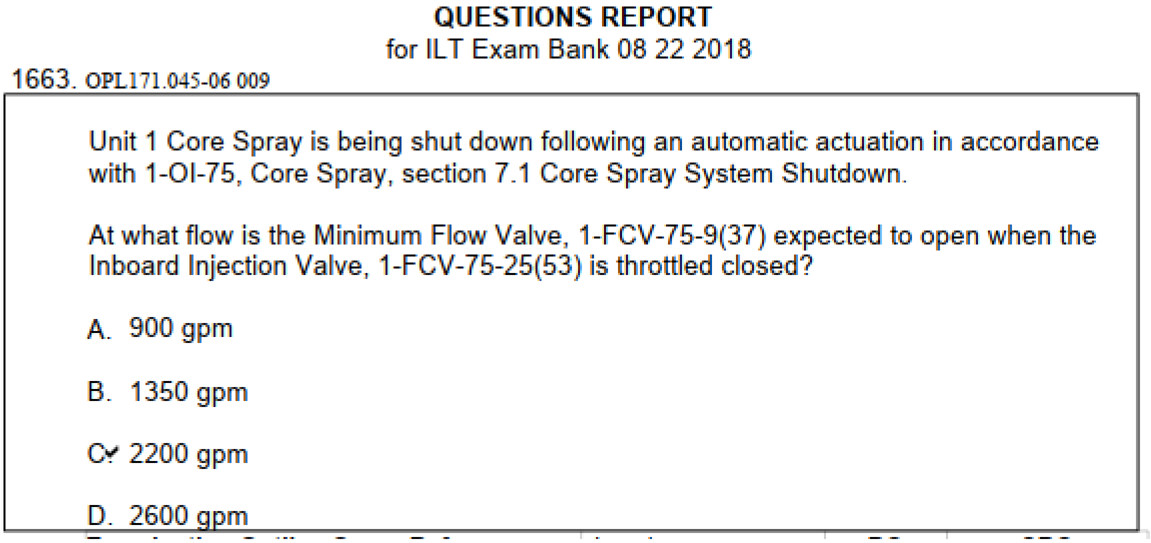
Comprehension or Analysis

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

55.43

Comments:

Copy of Bank Question:



QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1663. OPL171.045-06 009

Unit 1 Core Spray is being shut down following an automatic actuation in accordance with 1-OI-75, Core Spray, section 7.1 Core Spray System Shutdown.

At what flow is the Minimum Flow Valve, 1-FCV-75-9(37) expected to open when the Inboard Injection Valve, 1-FCV-75-25(53) is throttled closed?

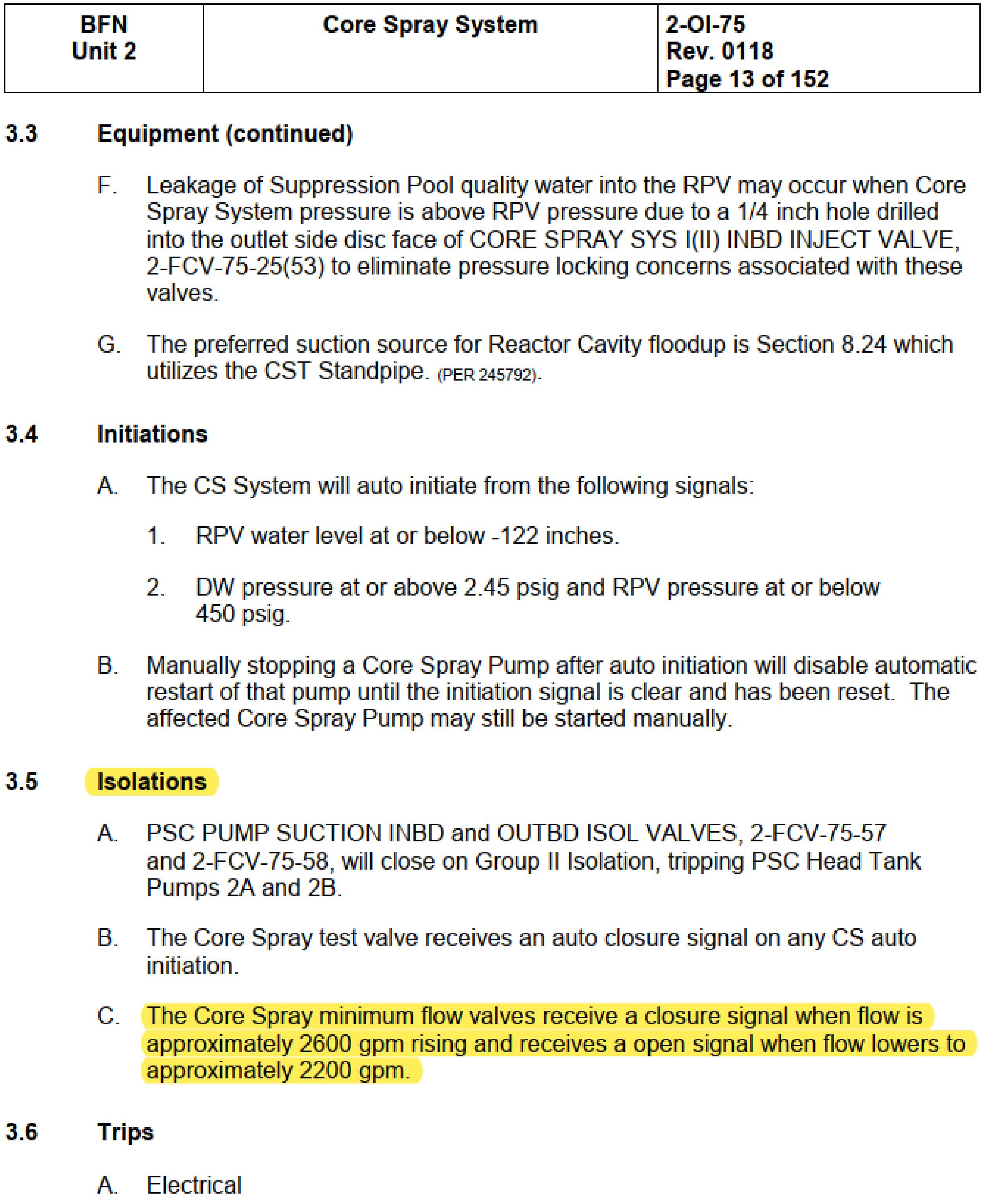
A. 900 gpm

B. 1350 gpm

C✓ 2200 gpm

D. 2600 gpm

Excerpt from 2-OI-75:



BFN **Unit 2**	**Core Spray System**	**2-OI-75** **Rev. 0118** **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

Examination Outline Cross-reference:

295038 (EPE 15) High Offsite Radioactivity Release Rate / 9
EA1.08 (10CFR 55.41.7)
Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RADIOACTIVITY RELEASE RATE:

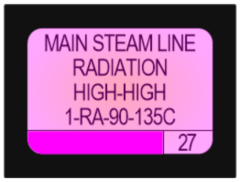
- MSIV leakage control

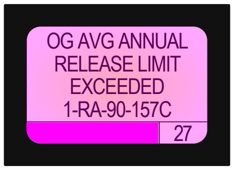
Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295038EA1.08	
Importance Rating	3.1	------

Proposed Question: **# 34**

Unit 1 was operating at 100% RTP when an event occurred, resulting in the following conditions:

- A manual SCRAM was inserted, **ALL** Control Rods are in
- MAIN STEAM LINE RADIATION HIGH-HIGH, (1-9-3A, Window 27) is sealed in
- OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED, (1-9-4C, Window 27) is sealed in





Given the conditions above, Main Steam Isolation Valves (MSIVs) require closure based on actions contained within Alarm Response Procedure (ARP) ___**(1)**___, Window 27.

The MAIN STEAM LINE RADIATION HIGH-HIGH alarms at radiation levels of ___**(2)**___ normal full power background.

A. (1) 1-9-3A
(2) 1.5 x

B. (1) 1-9-3A
(2) 3.0 x

C. (1) 1-9-4C
(2) 1.5 x

D. (1) 1-9-4C
(2) 3.0 x

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that 1.5 x normal full power background radiation is the alarm setpoint for MAIN STEAM LINE RADIATION HIGH, (1-9-3A, Window 7) which is not given in this question.

B **CORRECT:** *(See attached)* In accordance with the given sealed in (meaning valid) alarm MAIN STEAM LINE RADIATION HIGH-HIGH, (1-9-3A, Window 27), a core flow runback is to be inserted followed by a manual SCRAM. Additionally, the same ARP specifies that, if SLC injection per RC/Q of EOI-1 is NOT required, then MSIVs must be closed. Since all Control Rods are in (as given), this is specifically indicating that the Reactor is shut down under all conditions therefore closing the MSIVs is required. For second part, in accordance with the given MAIN STEAM LINE RADIATION HIGH-HIGH (1-9-3A, Window 27), the alarm setpoint is 3.0 x normal full power background radiation levels.

C INCORRECT: First part is incorrect but plausible in that candidates must determine the specifics between MAIN STEAM LINE RADIATION HIGH, (1-9-3A, Window 7) which is not given versus the given MAIN STEAM LINE RADIATION HIGH-HIGH, (1-9-3A, Window 27). Window 7 alarms at 1.5 x normal full power background and does NOT require MSIV closure versus Window 27 alarming at 3.0 x normal full power background and does require MSIV closure when SLC injection is not required. Additionally, the given OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED, (1-9-4C, Window 27) does NOT require MSIV closure.

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 operate equipment and monitor plant conditions to control releases of offsite radioactivity associated with procedural requirements related to MSIV operation. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 1-ARP-9-3A, Rev. 53 (Attach if not previously provided)
1-ARP-9-4C, Rev. 30

Proposed references to be provided to applicants during examination: **MAIN STEAM LINE RADIATION HIGH-HIGH (1-9-3A, Window 27), OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED (1-9-4C, Window 27)**

Learning Objective: OPL171.033 Obj. 3.a (As available)

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

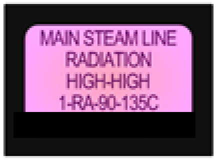
Question Source:	Bank #		
	Modified Bank #	BFN 1804 #98	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

Copy of Bank Question:

Unit 1 is at 100% RTP and in addition to other radiation alarms, the following is noted:

- MAIN STEAM LINE RADIATION HIGH-HIGH, (1-9-3A, Window 27) is sealed in
- OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED, (1-9-4C, Window 27) is sealed in



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

As Unit Supervisor, you are required to direct a manual Reactor SCRAM by the ___**(1)**___ Alarm Response Procedure (ARP).

As Unit Supervisor and upon hearing “ALL Rods In” during the SCRAM report, the MSIVs ___**(2)**___required to be directed closed.

A. **(1)** MAIN STEAM LINE RADIATION HIGH-HIGH, (1-9-3A, Window 27)
(2) are not

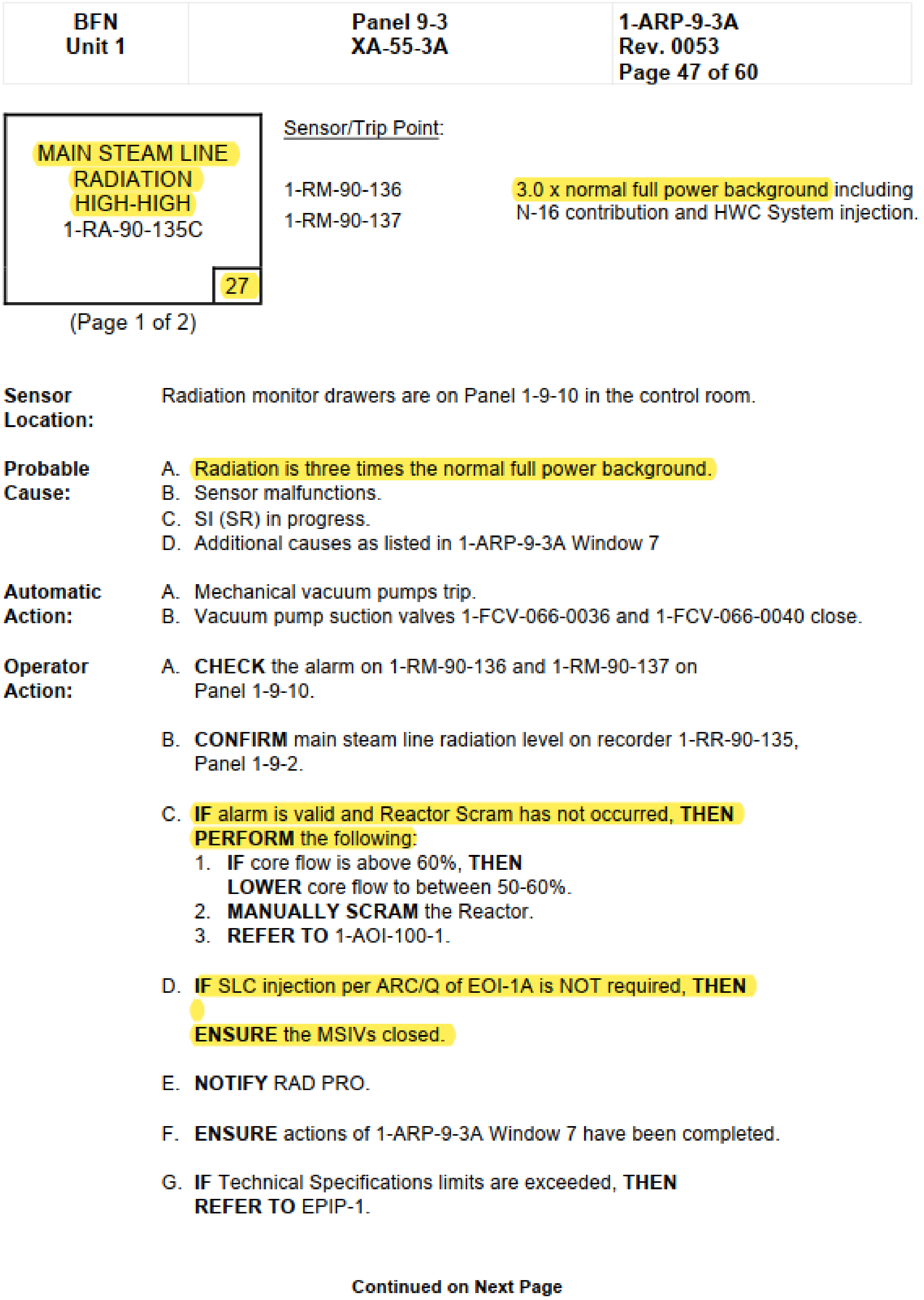
B. **(1)** OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED, (1-9-4C, Window 27)
(2) are not

C. **(1)** MAIN STEAM LINE RADIATION HIGH-HIGH, (1-9-3A, Window 27)
(2) are

D. **(1)** OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED, (1-9-4C, Window 27)
(2) are

Proposed Answer: C

Excerpts from 1-ARP-3A, WINDOW 27:



BFN **Unit 1**	**Panel 9-3** **XA-55-3A**	**1-ARP-9-3A** **Rev. 0053** **Page 47 of 60**

MAIN STEAM LINE
RADIATION
HIGH-HIGH
1-RA-90-135C
27

(Page 1 of 2)

Sensor/Trip Point:

1-RM-90-136 1-RM-90-137	3.0 x normal full power background including N-16 contribution and HWC System injection.

Sensor Location: Radiation monitor drawers are on Panel 1-9-10 in the control room.

Probable Cause:

A. Radiation is three times the normal full power background.
B. Sensor malfunctions.
C. SI (SR) in progress.
D. Additional causes as listed in 1-ARP-9-3A Window 7

Automatic Action:

A. Mechanical vacuum pumps trip.
B. Vacuum pump suction valves 1-FCV-066-0036 and 1-FCV-066-0040 close.

Operator Action:

A. **CHECK** the alarm on 1-RM-90-136 and 1-RM-90-137 on Panel 1-9-10.

B. **CONFIRM** main steam line radiation level on recorder 1-RR-90-135, Panel 1-9-2.

C. **IF** alarm is valid and Reactor Scram has not occurred, **THEN PERFORM** the following:
   1. **IF** core flow is above 60%, **THEN LOWER** core flow to between 50-60%.
   2. **MANUALLY SCRAM** the Reactor.
   3. **REFER TO** 1-AOI-100-1.

D. **IF** SLC injection per ARC/Q of EOI-1A is NOT required, **THEN**

ENSURE the MSIVs closed.

E. **NOTIFY** RAD PRO.

F. **ENSURE** actions of 1-ARP-9-3A Window 7 have been completed.

G. **IF** Technical Specifications limits are exceeded, **THEN REFER TO** EPIP-1.

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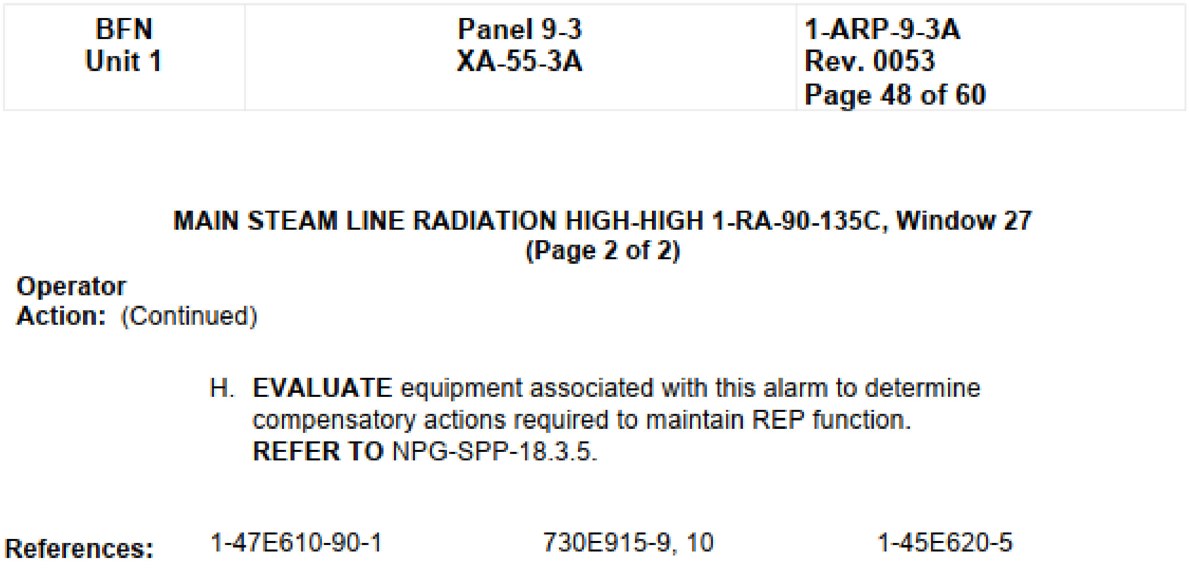


BFN **Unit 1**	**Panel 9-3** **XA-55-3A**	**1-ARP-9-3A** **Rev. 0053** **Page 48 of 60**

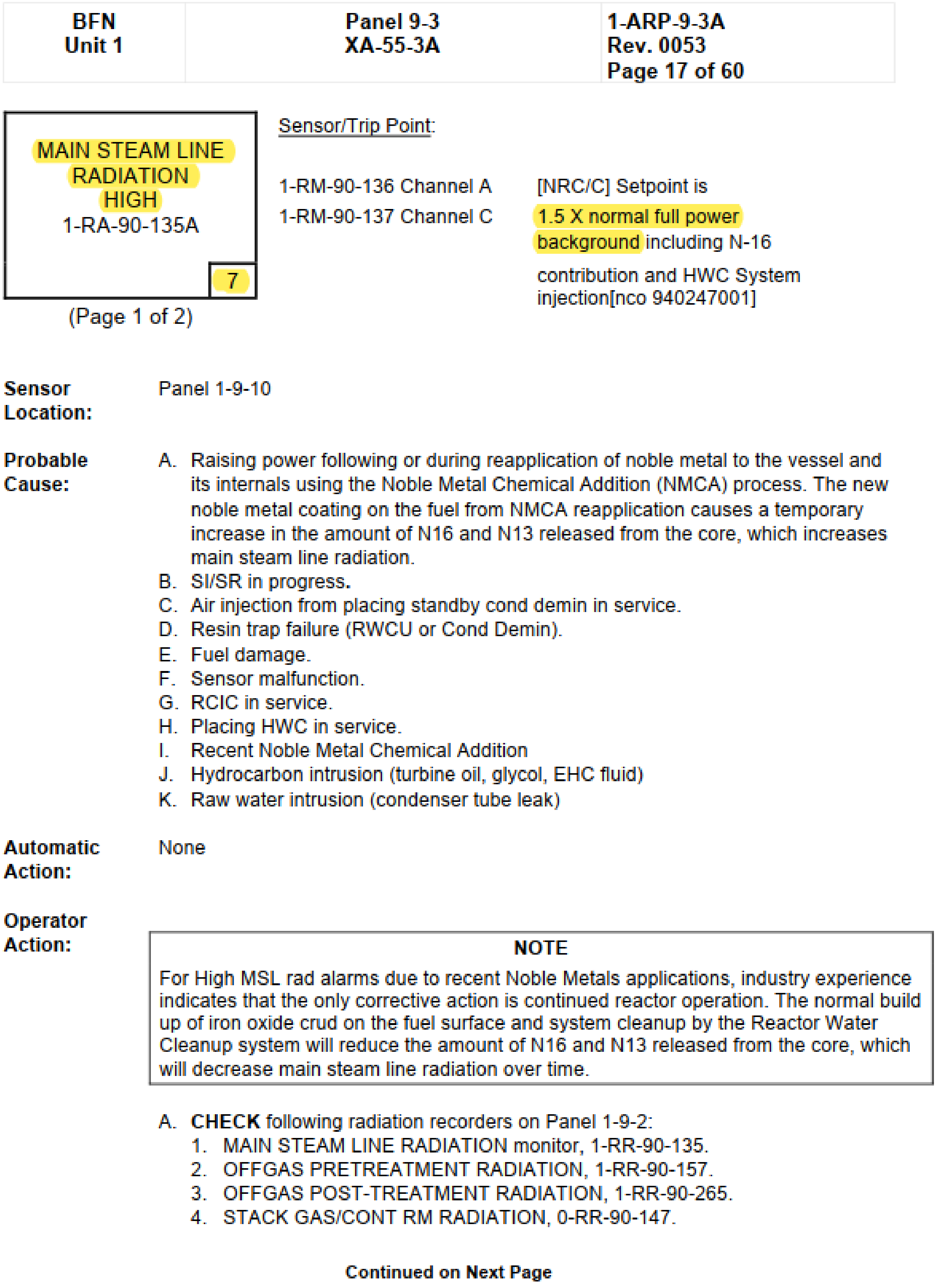
MAIN STEAM LINE RADIATION HIGH-HIGH 1-RA-90-135C, Window 27
(Page 2 of 2)

Operator Action: (Continued)

H. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References: 1-47E610-90-1 730E915-9, 10 1-45E620-5

Excerpts from 1-ARP-3A, WINDOW 7: Supports Distractors A(2), C(2)



BFN **Unit 1**	**Panel 9-3** **XA-55-3A**	**1-ARP-9-3A** **Rev. 0053** **Page 17 of 60**

MAIN STEAM LINE
RADIATION
HIGH
1-RA-90-135A
7

(Page 1 of 2)

Sensor/Trip Point:

1-RM-90-136 Channel A 1-RM-90-137 Channel C	[NRC/C] Setpoint is 1.5 X normal full power background including N-16 contribution and HWC System injection[nco 940247001]

Sensor Location: Panel 1-9-10

Probable Cause:

A. Raising power following or during reapplication of noble metal to the vessel and its internals using the Noble Metal Chemical Addition (NMCA) process. The new noble metal coating on the fuel from NMCA reapplication causes a temporary increase in the amount of N16 and N13 released from the core, which increases main steam line radiation.
B. SI/SR in progress.
C. Air injection from placing standby cond demin in service.
D. Resin trap failure (RWCU or Cond Demin).
E. Fuel damage.
F. Sensor malfunction.
G. RCIC in service.
H. Placing HWC in service.
I. Recent Noble Metal Chemical Addition
J. Hydrocarbon intrusion (turbine oil, glycol, EHC fluid)
K. Raw water intrusion (condenser tube leak)

Automatic Action: None

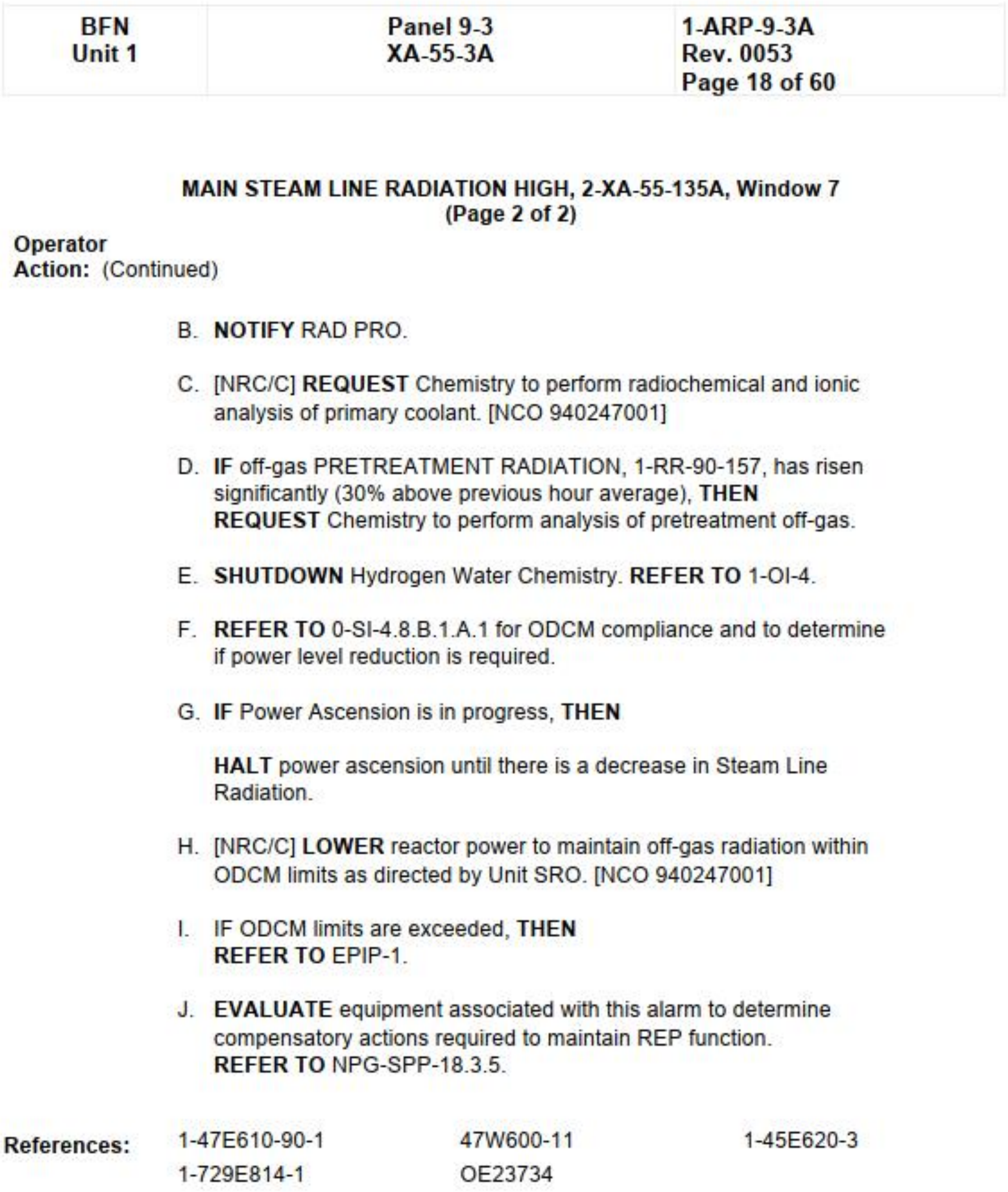
Operator Action:

NOTE

For High MSL rad alarms due to recent Noble Metals applications, industry experience indicates that the only corrective action is continued reactor operation. The normal build up of iron oxide crud on the fuel surface and system cleanup by the Reactor Water Cleanup system will reduce the amount of N16 and N13 released from the core, which will decrease main steam line radiation over time.

A. **CHECK** following radiation recorders on Panel 1-9-2:
   1. MAIN STEAM LINE RADIATION monitor, 1-RR-90-135.
   2. OFFGAS PRETREATMENT RADIATION, 1-RR-90-157.
   3. OFFGAS POST-TREATMENT RADIATION, 1-RR-90-265.
   4. STACK GAS/CONT RM RADIATION, 0-RR-90-147.

Continued on Next Page



BFN **Unit 1**	**Panel 9-3** **XA-55-3A**	**1-ARP-9-3A** **Rev. 0053** **Page 18 of 60**

MAIN STEAM LINE RADIATION HIGH, 2-XA-55-135A, Window 7
(Page 2 of 2)

Operator Action: (Continued)

B. **NOTIFY** RAD PRO.

C. [NRC/C] **REQUEST** Chemistry to perform radiochemical and ionic analysis of primary coolant. [NCO 940247001]

D. **IF** off-gas PRETREATMENT RADIATION, 1-RR-90-157, has risen significantly (30% above previous hour average), **THEN** **REQUEST** Chemistry to perform analysis of pretreatment off-gas.

E. **SHUTDOWN** Hydrogen Water Chemistry. **REFER TO** 1-OI-4.

F. **REFER TO** 0-SI-4.8.B.1.A.1 for ODCM compliance and to determine if power level reduction is required.

G. **IF** Power Ascension is in progress, **THEN**

HALT power ascension until there is a decrease in Steam Line Radiation.

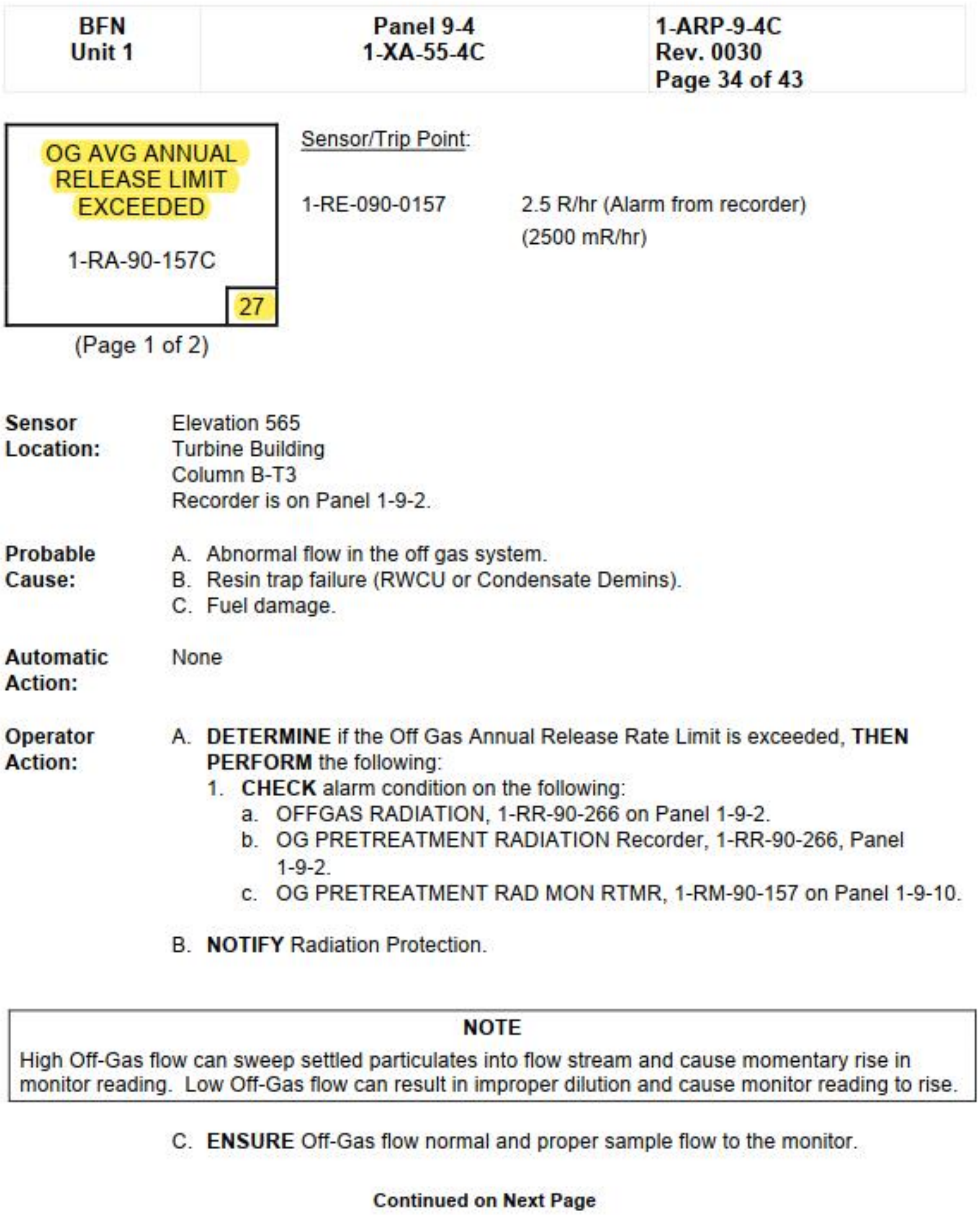
H. [NRC/C] **LOWER** reactor power to maintain off-gas radiation within ODCM limits as directed by Unit SRO. [NCO 940247001]

I. IF ODCM limits are exceeded, **THEN** **REFER TO** EPIP-1.

J. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References:	1-47E610-90-1	47W600-11	1-45E620-3
	1-729E814-1	OE23734	

Excerpts from 1-ARP-4C, WINDOW 27: Supports Distractors C(1), D(1)



BFN **Unit 1**	**Panel 9-4** **1-XA-55-4C**	**1-ARP-9-4C** **Rev. 0030** **Page 34 of 43**

OG AVG ANNUAL
RELEASE LIMIT
EXCEEDED

1-RA-90-157C

27

(Page 1 of 2)

Sensor/Trip Point:

1-RE-090-0157	2.5 R/hr (Alarm from recorder) (2500 mR/hr)

Sensor Location:
Elevation 565
Turbine Building
Column B-T3
Recorder is on Panel 1-9-2.

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. **DETERMINE** if the Off Gas Annual Release Rate Limit is exceeded, **THEN PERFORM** the following:
   1. **CHECK** alarm condition on the following:
      a. OFFGAS RADIATION, 1-RR-90-266 on Panel 1-9-2.
      b. OG PRETREATMENT RADIATION Recorder, 1-RR-90-266, Panel 1-9-2.
      c. OG PRETREATMENT RAD MON RTMR, 1-RM-90-157 on Panel 1-9-10.

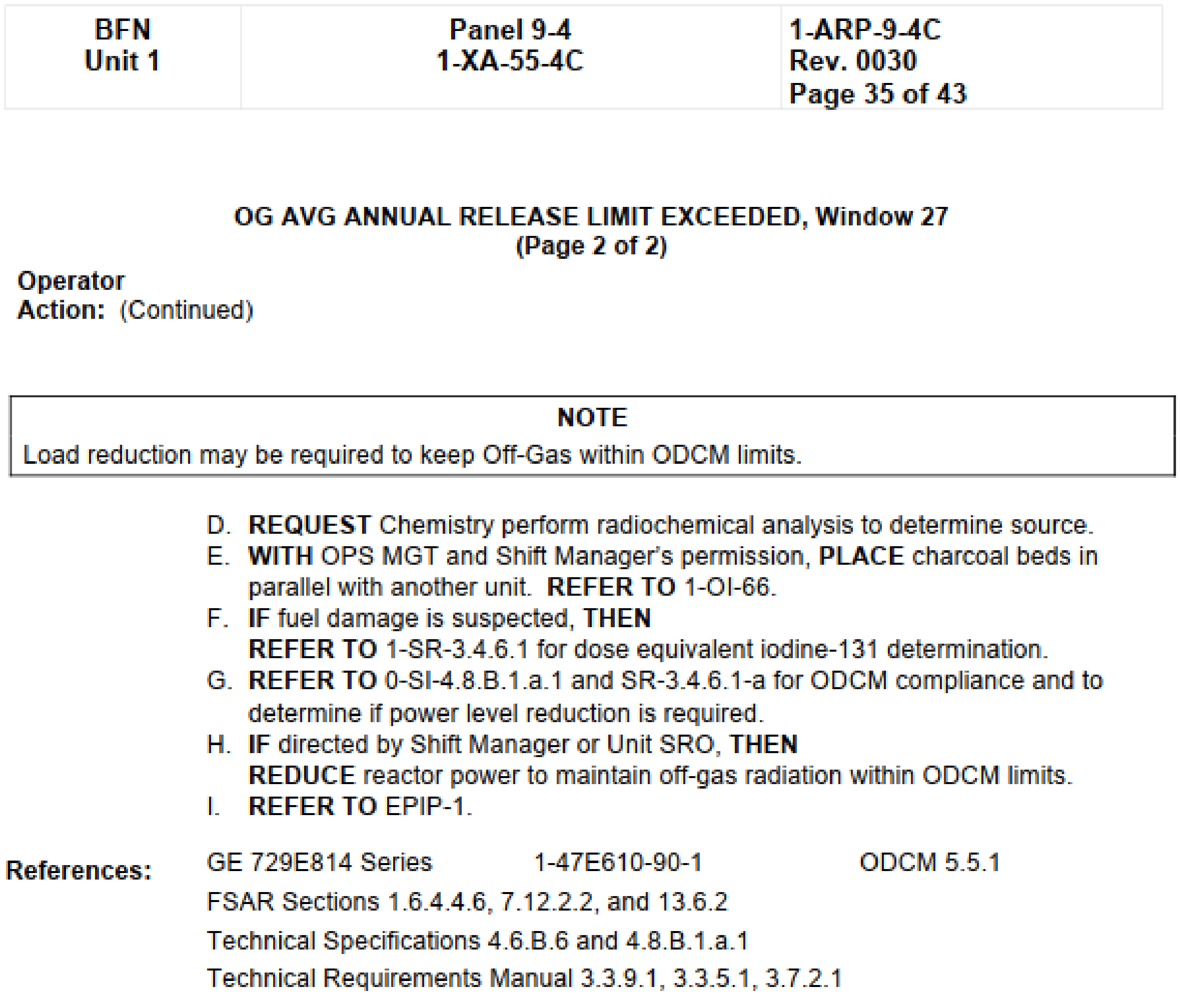
B. **NOTIFY** Radiation Protection.

NOTE

High Off-Gas flow can sweep settled particulates into flow stream and cause momentary rise in monitor reading. Low Off-Gas flow can result in improper dilution and cause monitor reading to rise.

C. **ENSURE** Off-Gas flow normal and proper sample flow to the monitor.

Continued on Next Page



BFN **Unit 1**	**Panel 9-4** **1-XA-55-4C**	**1-ARP-9-4C** **Rev. 0030** **Page 35 of 43**

OG AVG ANNUAL RELEASE LIMIT EXCEEDED, Window 27
(Page 2 of 2)

Operator
Action: (Continued)

> **NOTE**
> Load reduction may be required to keep Off-Gas within ODCM limits.

D. **REQUEST** Chemistry perform radiochemical analysis to determine source.
E. **WITH** OPS MGT and Shift Manager's permission, **PLACE** charcoal beds in parallel with another unit. **REFER TO** 1-OI-66.
F. **IF** fuel damage is suspected, **THEN**
REFER TO 1-SR-3.4.6.1 for dose equivalent iodine-131 determination.
G. **REFER TO** 0-SI-4.8.B.1.a.1 and 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 SRO, **THEN**
REDUCE reactor power to maintain off-gas radiation within ODCM limits.
I. **REFER TO** EPIP-1.

References: GE 729E814 Series 1-47E610-90-1 ODCM 5.5.1
FSAR Sections 1.6.4.4.6, 7.12.2.2, and 13.6.2
Technical Specifications 4.6.B.6 and 4.8.B.1.a.1
Technical Requirements Manual 3.3.9.1, 3.3.5.1, 3.7.2.1

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

Examination Outline Cross-reference:

212000 (SF7 RPS) Reactor Protection
K6.10 (10CFR 55.41.7)
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Reactor Protection System:

- Reactor/turbine pressure regulating system

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	212000K6.10	
Importance Rating	3.5	------

Proposed Question: **# 35**

In accordance with 1-OI-99, Reactor Protection System (RPS), which **ONE** of the following completes the statements below?

The Generator Load Reject Turbine Control Valve (TCV) Fast Closure RPS SCRAM signal is automatically bypassed at ____**(1)**____ RTP.

The pressure switch on either TCV 1 or 3 will send a trip signal to RPS Trip System ____**(2)**____.

A. (1) 50%
(2) A

B. (1) 50%
(2) B

C. (1) 5%
(2) A

D. (1) 5%
(2) B

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that there are 19 automatic Reactor SCRAM initiation signals the candidate must recall from memory as it relates to numerous system setpoints along with variations of logic and bypass conditions. Second part is correct *(See A).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that TCVs 1 and 3 input into RPS channels A1 and A2 while TCVs 2 and 4 input into RPS channels B1 and B2 respectively.

C **CORRECT:** *(See attached)* In accordance with 1-OI-99, Reactor Protection System, upon a Main Turbine trip, TCV fast closure is sensed, however it is automatically bypassed in the RPS SCRAM logic while Reactor Power is LESS than 26%. It is active greater than 26% to enable the RPS function due to being at a Reactor Power that's greater than the bypass valve capacity. For second part, there are 4 automatic strings of RPS logic channels, A1/A2 and B1/B2. TCV 1 inputs to A1 and TCV 3 inputs in to A2. Additionally, TCV 2 inputs to B1 and TCV 4 inputs to B2.

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 operation of the turbine pressure regulating system it relates to the Reactor Protection System. This question is rated as memory due to strictly recalling facts related to the turbine pressure regulating system and Reactor Protection System.

Technical Reference(s): 1-OI-99, Rev. 61 (Attach if not previously provided)

OPL171.028, Rev. 26

1-ARP-9-5B, Rev. 24

1-730E915-9, Rev. 23

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.028 Obj. 12j (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**

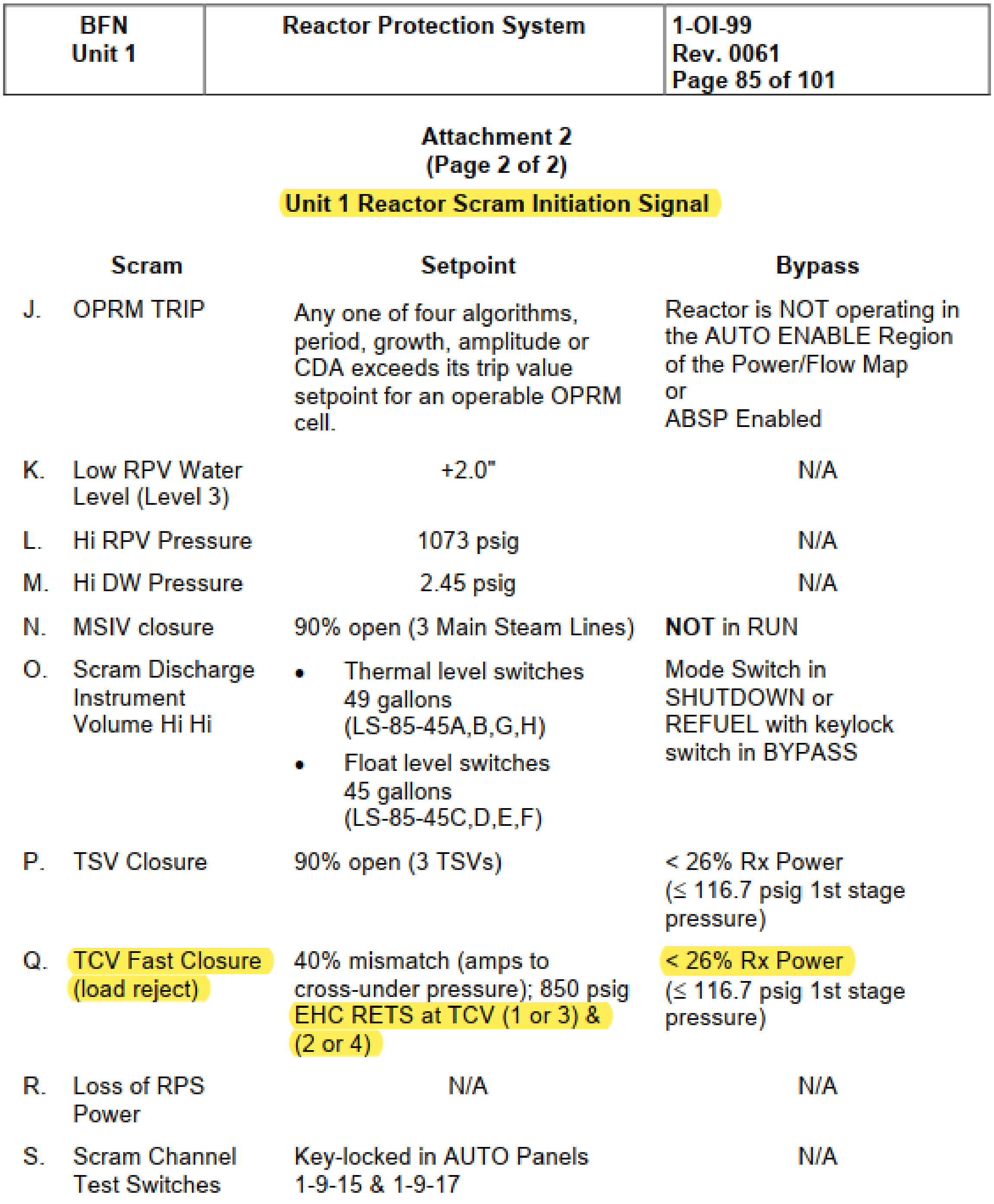
Comprehension or Analysis

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

55.43

Comments:

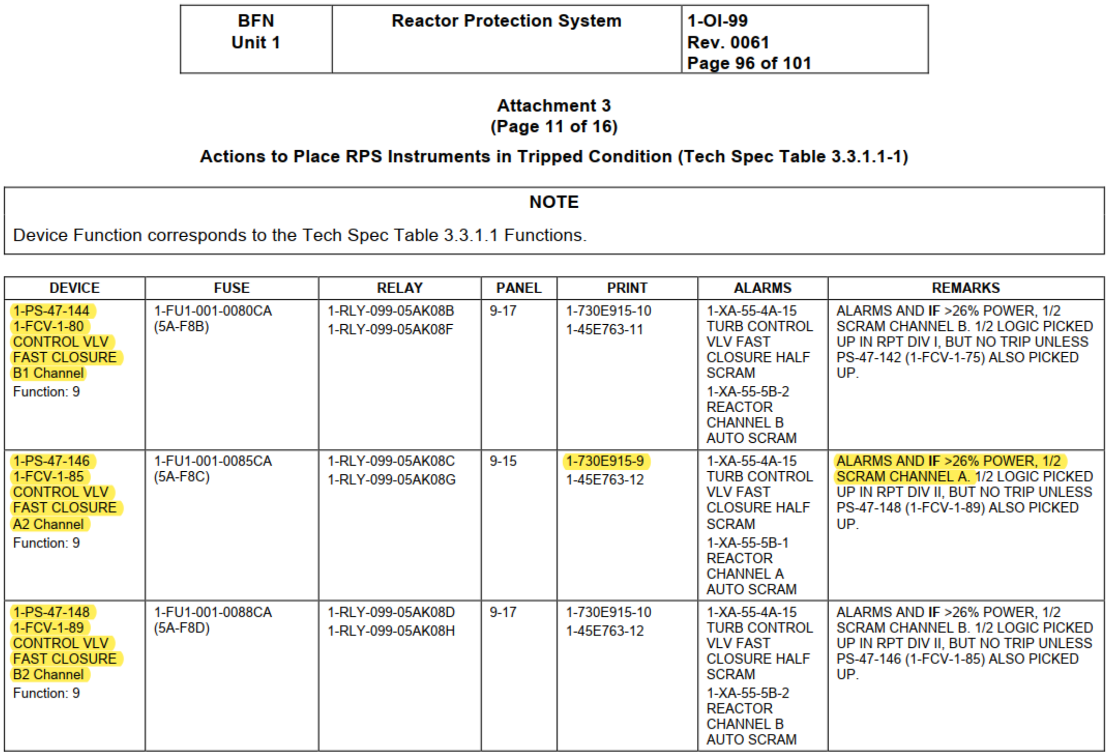
Excerpts from 1-OI-99:



BFN Unit 1	Reactor Protection System	1-OI-99 Rev. 0061 Page 85 of 101

Attachment 2
(Page 2 of 2)
Unit 1 Reactor Scram Initiation Signal

	Scram	Setpoint	Bypass
J.	OPRM TRIP	Any one of four algorithms, period, growth, amplitude or CDA exceeds its trip value setpoint for an operable OPRM cell.	Reactor is NOT operating in the AUTO ENABLE Region of the Power/Flow Map or ABSP Enabled
K.	Low RPV Water Level (Level 3)	+2.0"	N/A
L.	Hi RPV Pressure	1073 psig	N/A
M.	Hi DW Pressure	2.45 psig	N/A
N.	MSIV closure	90% open (3 Main Steam Lines)	**NOT** in RUN
O.	Scram Discharge Instrument Volume Hi Hi	• Thermal level switches 49 gallons (LS-85-45A,B,G,H) • Float level switches 45 gallons (LS-85-45C,D,E,F)	Mode Switch in SHUTDOWN or REFUEL with keylock switch in BYPASS
P.	TSV Closure	90% open (3 TSVs)	< 26% Rx Power (≤ 116.7 psig 1st stage pressure)
Q.	TCV Fast Closure (load reject)	40% mismatch (amps to cross-under pressure); 850 psig EHC RETS at TCV (1 or 3) & (2 or 4)	< 26% Rx Power (≤ 116.7 psig 1st stage pressure)
R.	Loss of RPS Power	N/A	N/A
S.	Scram Channel Test Switches	Key-locked in AUTO Panels 1-9-15 & 1-9-17	N/A



BFN Unit 1	**Reactor Protection System**	**1-OI-99 Rev. 0061 Page 96 of 101**

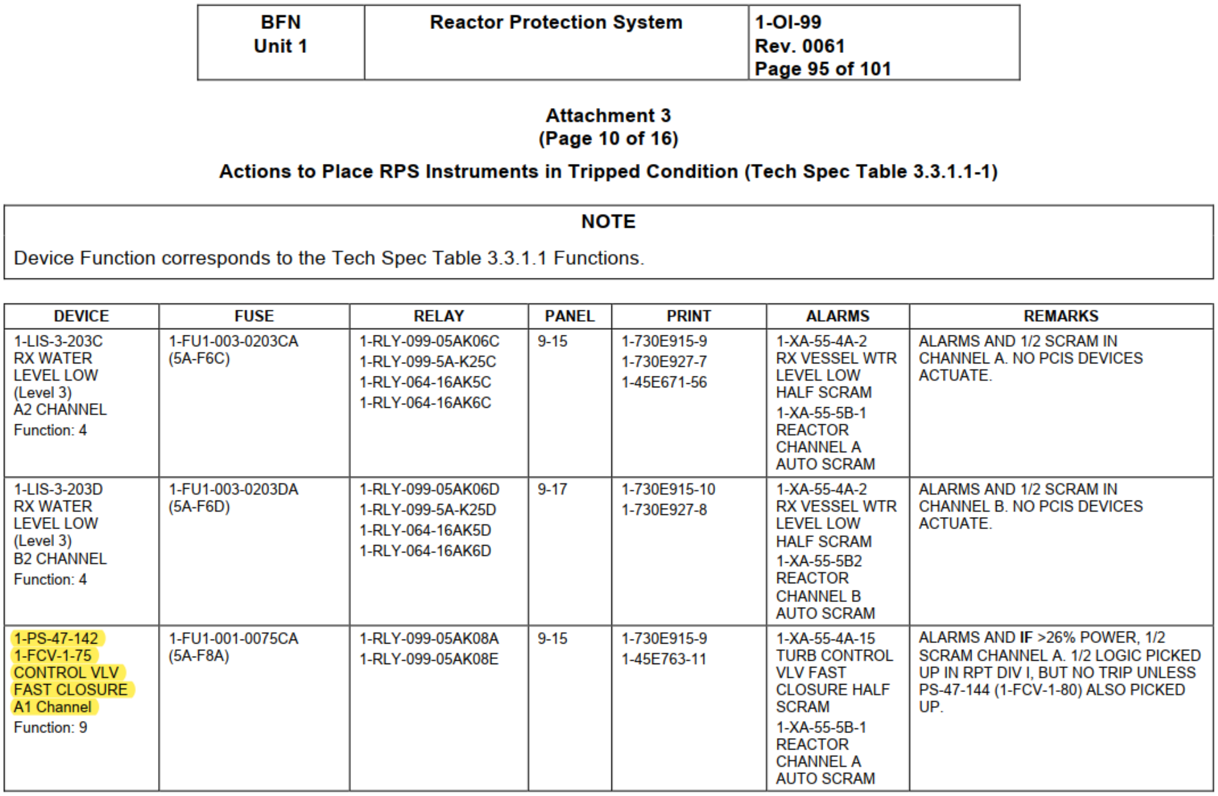
Attachment 3
(Page 11 of 16)

Actions to Place RPS Instruments in Tripped Condition (Tech Spec Table 3.3.1.1-1)

NOTE

Device Function corresponds to the Tech Spec Table 3.3.1.1 Functions.

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
1-PS-47-144 1-FCV-1-80 CONTROL VLV FAST CLOSURE B1 Channel Function: 9	1-FU1-001-0080CA (5A-F8B)	1-RLY-099-05AK08B 1-RLY-099-05AK08F	9-17	1-730E915-10 1-45E763-11	1-XA-55-4A-15 TURB CONTROL VLV FAST CLOSURE HALF SCRAM 1-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND **IF** >26% POWER, 1/2 SCRAM CHANNEL B. 1/2 LOGIC PICKED UP IN RPT DIV I, BUT NO TRIP UNLESS PS-47-142 (1-FCV-1-75) ALSO PICKED UP.
1-PS-47-146 1-FCV-1-85 CONTROL VLV FAST CLOSURE A2 Channel Function: 9	1-FU1-001-0085CA (5A-F8C)	1-RLY-099-05AK08C 1-RLY-099-05AK08G	9-15	1-730E915-9 1-45E763-12	1-XA-55-4A-15 TURB CONTROL VLV FAST CLOSURE HALF SCRAM 1-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND **IF** >26% POWER, 1/2 SCRAM CHANNEL A. 1/2 LOGIC PICKED UP IN RPT DIV II, BUT NO TRIP UNLESS PS-47-148 (1-FCV-1-89) ALSO PICKED UP.
1-PS-47-148 1-FCV-1-89 CONTROL VLV FAST CLOSURE B2 Channel Function: 9	1-FU1-001-0088CA (5A-F8D)	1-RLY-099-05AK08D 1-RLY-099-05AK08H	9-17	1-730E915-10 1-45E763-12	1-XA-55-4A-15 TURB CONTROL VLV FAST CLOSURE HALF SCRAM 1-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND **IF** >26% POWER, 1/2 SCRAM CHANNEL B. 1/2 LOGIC PICKED UP IN RPT DIV II, BUT NO TRIP UNLESS PS-47-146 (1-FCV-1-85) ALSO PICKED UP.



BFN Unit 1	**Reactor Protection System**	**1-OI-99 Rev. 0061 Page 95 of 101**

Attachment 3
(Page 10 of 16)

Actions to Place RPS Instruments in Tripped Condition (Tech Spec Table 3.3.1.1-1)

NOTE

Device Function corresponds to the Tech Spec Table 3.3.1.1 Functions.

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
1-LIS-3-203C RX WATER LEVEL LOW (Level 3) A2 CHANNEL Function: 4	1-FU1-003-0203CA (5A-F6C)	1-RLY-099-05AK06C 1-RLY-099-5A-K25C 1-RLY-064-16AK5C 1-RLY-064-16AK6C	9-15	1-730E915-9 1-730E927-7 1-45E671-56	1-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 1-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A. NO PCIS DEVICES ACTUATE.
1-LIS-3-203D RX WATER LEVEL LOW (Level 3) B2 CHANNEL Function: 4	1-FU1-003-0203DA (5A-F6D)	1-RLY-099-05AK06D 1-RLY-099-5A-K25D 1-RLY-064-16AK5D 1-RLY-064-16AK6D	9-17	1-730E915-10 1-730E927-8	1-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 1-XA-55-5B2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B. NO PCIS DEVICES ACTUATE.
1-PS-47-142 1-FCV-1-75 CONTROL VLV FAST CLOSURE A1 Channel Function: 9	1-FU1-001-0075CA (5A-F8A)	1-RLY-099-05AK08A 1-RLY-099-05AK08E	9-15	1-730E915-9 1-45E763-11	1-XA-55-4A-15 TURB CONTROL VLV FAST CLOSURE HALF SCRAM 1-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND **IF** >26% POWER, 1/2 SCRAM CHANNEL A. 1/2 LOGIC PICKED UP IN RPT DIV I, BUT NO TRIP UNLESS PS-47-144 (1-FCV-1-80) ALSO PICKED UP.

Excerpt from 1-ARP-9-5B:

BFN **Unit 1**	**Panel 9-5** **1-XA-55-5B**	**1-ARP-9-5B** **Rev. 0024** **Page 19 of 42**

TURB CV FAST CLOSURE
TURB SV CLOSURE
SCRAM/RPT TRIP
LOGIC BYPASS
16

(Page 1 of 1)

Sensor/Trip Point:

Relays		
5A-K9A	1-PIS-001-0091B	Turbine first stage pressure ≤ 118 psig.
5A-K9C	1-PIS-001-0081B	
5A-K9B	1-PIS-001-0091A	
5A-K9D	1-PIS-001-0081A	

NOTE

All pressure values include 11.5 psig static head pressure correction.

Sensor Location: Elevation 586
1-PIS-001-0081A(B) on 1-LPNL-925-0111 j-T2
1-PIS-001-0091A(B) on 1-LPNL-925-0110 h-T5

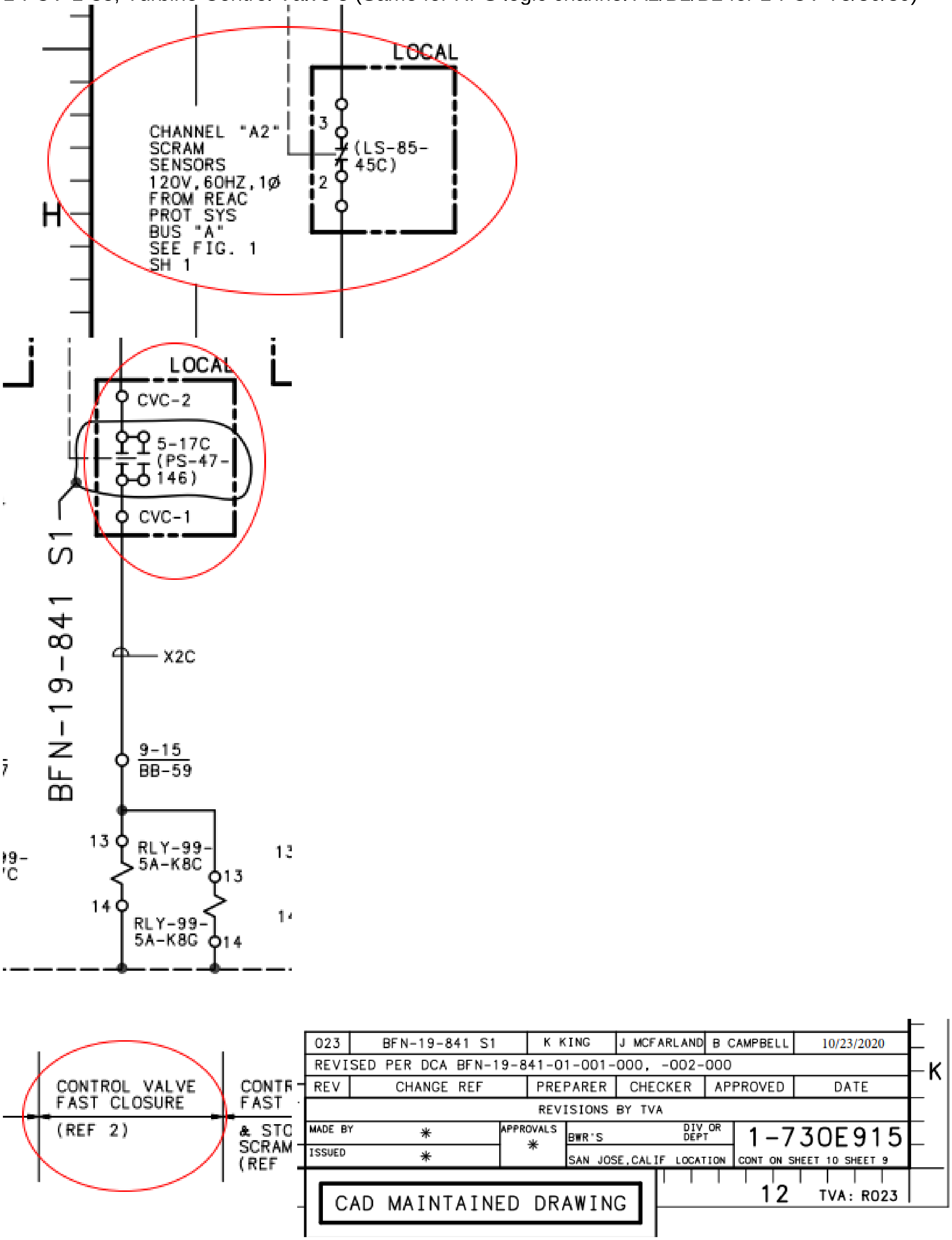
Probable Cause:
A. Power is less than 26% (first stage turbine pressure is ≤ 118 psig).
B. SI (or SR) in progress.
C. Sensor malfunction.

Automatic Action:
A. The control valve fast closure scram and the turbine stop valve scram is bypassed.
B. RPT Trip is bypassed.

Operator Action:
A. **CHECK** first stage pressure with 1-PI-1-79 on Panel 1-9-7.
B. **IF** pressure is above 128.2 psig, **THEN**
REFER TO Tech Spec Table 3.3.1.1-1, Section 3.3.4.1, TRM Section 3.3.1.
C. **IF** alarm remains sealed in with reactor power greater than 26%, **THEN**
NOTIFY Reactor Engineering to determine if it is necessary to apply a thermal limit penalty due to Power Load Unbalance trip being out of service.(PLOOS).

References: 1-45E620-6-2 1-730E915-9 and 10 1-47E610-1-2

Excerpt from 1-730E915-9: Illustrates RPS logic channel A2 as it relates to pressure switch for 1-FCV-1-85, Turbine Control Valve 3 (Same for RPS logic channel A1/B1/B2 for 1-FCV-75/80/89)



Excerpt from OPL171.028 Lesson:

OPL171.028 , Reactor Protection System, Rev.# 26

e) Circuitry is designed such that the pressure switch on:
   (1) Either Control Valve 1 or 3 will trip RPS Channel A.
   (2) Either Control Valve 2 or 4 will trip RPS Channel B.

2-730E915-9, 10

f) These switches will also provide a SCRAM signal on loss of hydraulic fluid pressure when a load reject signal is not present. The loss of hydraulic fluid pressure can result in a fast closure of the Control Valves.

4. SCRAM discharge volume high level:

IL-6

a) Initiates a SCRAM while adequate volume is available to receive SCRAM discharge water to assure that all operable drives will fully insert.

Drawing
2-730E915-9, 10,
2-730E915RF-11
2-730E915RF-12

b) Level is sensed by two mechanical float switches and two electronic level switches (RTD's) in each instrument volume.
   (1) East Instrument Volume
      (a) LS-85-45E (A1-float)
      (b) LS-85-45F (B1-float)
      (c) LS-85-45G (A2-thermal)
      (d) LS-85-45H (B2-thermal)
   (2) West Instrument Volume
      (a) LS-85-45A (A1-thermal)
      (b) LS-85-45B (B1-thermal)
      (c) LS-85-45C (A2-float)
      (d) LS-85-45D (B2-float)

45 gal – float

49 gal – thermal

49 gal – thermal

45 gal – float

c) When the instrument volume fills up to the setpoint, the sensors open contacts in both RPS Trip Systems. Therefore, one SDV that is full would initiate a full SCRAM.

5. Main Steam Isolation Valve (MSIV) closure, (90% full open)

IL-7

a) The SCRAM anticipates the pressure transient resulting from closure and helps mitigate flux spike to prevent exceeding MCPR.
b) Each MSIV has two RPS limit switches - one from the A trip logic, the other from the B trip logic.
c) The MSIV red/green lamp indication is NOT supplied by the RPS limit switches, but instead by a third limit switch.

Limit Switch Setting: LS-3 and LS-4 actuate at <90% open

GREEN LIGHT Limit Switch (LS-5) setting is 85%
2-730E915-9, 10

Examination Outline Cross-reference:

700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6

G2.1.20 (10CFR 55.41.10)

Ability to interpret and execute procedure steps.

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	700000G2.1.20	
Importance Rating	4.6	------

Proposed Question: **# 36**

Unit 3 is operating at 80% RTP with the following conditions:

- The Transmission System Operator (TOp) notified the Shift Manager (SM) that the 500 KV System voltage is 542 KV
- Grid frequency is 60.03 Hz
- Unit 3 Main Generator Reactive Load is 53 MVARs outgoing
- 0-AOI-57-1E, Grid Instability has been entered by all Units

Given the conditions above, which ONE of the following identifies how to restore to normal limits in accordance with 0-AOI-57-1E?

A. Raise **Reactive** Power using the Voltage Regulator.

B. Lower **Reactive** Power using the Voltage Regulator.

C. Raise **Reactor** Power by approximately 1% / minute using Recirculation Flow.

D. Lower **Reactor** Power by approximately 1% / minute using Recirculation Flow.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that if System voltage is lower than 510 KV then RAISE Reactive Power until System voltage returns to 510 KV OR UNTIL Generator Reactive Power reaches (+) 300 MVAR (maximum outgoing/lagging Reactive limit).

B **CORRECT**: (*See attached*) In accordance with 0-AOI-57-1E, if system voltage is greater than 540KV then LOWER Reactive Power until System voltage is 530KV (normal limits of 520 +/- 10 KV) OR UNTIL Generator Reactive Power reaches (-) 150 MVAR (Incoming/leading Reactive limit). In accordance with 3-OI-47, to adjust GENERATOR MVAR in the negative/leading direction, PLACE VOLTAGE REGULATOR LOWER/RAISE ADJUST in LOWER until desired MVAR is indicated.

C INCORRECT**:** Incorrect but plausible in that if System frequency was lower than 59.85 Hz AND Reactor power is less than rated power, THEN RAISE Reactor Power by approximately 1%/minute (10 MWe/minute) UNTIL System frequency returns to 59.98 Hz.

D INCORRECT: Incorrect but plausible in that if System frequency was greater than 60.15 Hz THEN LOWER Reactor power by approximately 1%/minute (10 MWe/minute) UNTIL system frequency returns to 60.03 Hz.

RO Level Justification: Tests the candidate's ability to recognize the need and method to adjust the generator voltage regulator controls to lower REACTIVE power during grid instability 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.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 0-AOI-57-1E, Rev. 11 (Attach if not previously provided)
3-OI-47, Rev. 130

Proposed references to be provided to applicants during examination: **NONE**

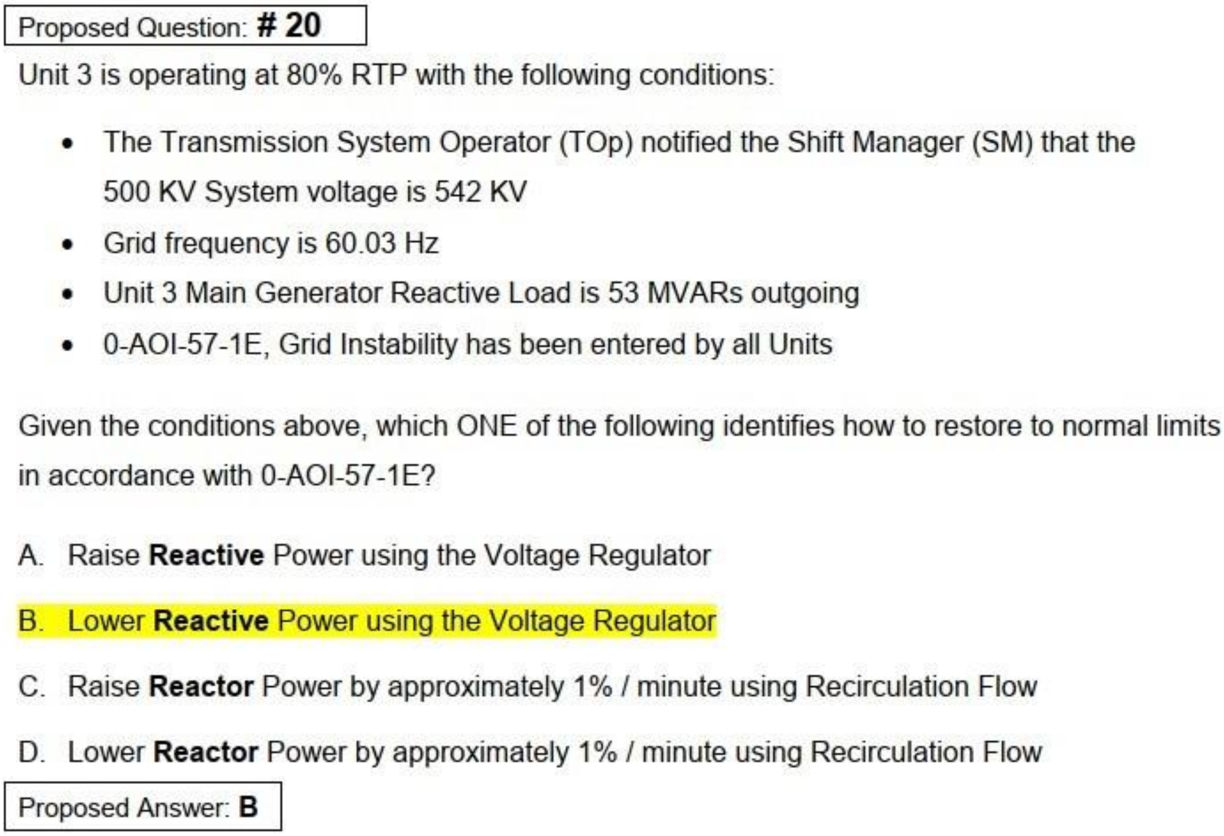
Learning Objective: OPL171.134 Obj. 9 (As available)

Question Source:	Bank #	BFN 1804 NRC #20	
	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:



Proposed Question: **# 20**

Unit 3 is operating at 80% RTP with the following conditions:

- The Transmission System Operator (TOp) notified the Shift Manager (SM) that the 500 KV System voltage is 542 KV
- Grid frequency is 60.03 Hz
- Unit 3 Main Generator Reactive Load is 53 MVARs outgoing
- 0-AOI-57-1E, Grid Instability has been entered by all Units

Given the conditions above, which ONE of the following identifies how to restore to normal limits in accordance with 0-AOI-57-1E?

A. Raise **Reactive** Power using the Voltage Regulator

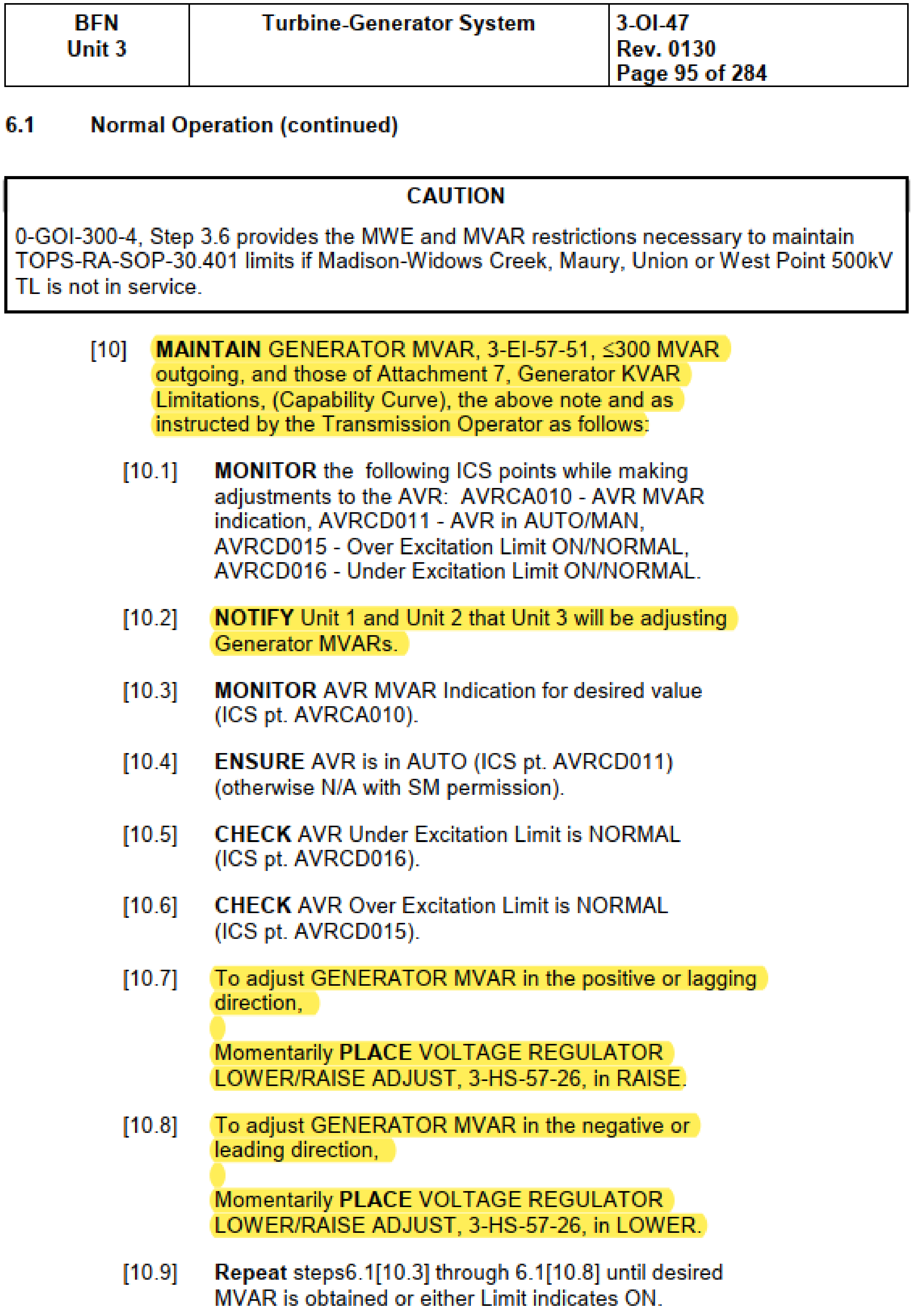
B. Lower **Reactive** Power using the Voltage Regulator

C. Raise **Reactor** Power by approximately 1% / minute using Recirculation Flow

D. Lower **Reactor** Power by approximately 1% / minute using Recirculation Flow

Proposed Answer: **B**

Excerpts from 3-OI-47:



BFN **Unit 3**	**Turbine-Generator System**	**3-OI-47** **Rev. 0130** **Page 95 of 284**

6.1 Normal Operation (continued)

CAUTION

0-GOI-300-4, Step 3.6 provides the MWE and MVAR restrictions necessary to maintain TOPS-RA-SOP-30.401 limits if Madison-Widows Creek, Maury, Union or West Point 500kV TL is not in service.

[10] **MAINTAIN** GENERATOR MVAR, 3-EI-57-51, ≤300 MVAR outgoing, and those of Attachment 7, Generator KVAR Limitations, (Capability Curve), the above note and as instructed by the Transmission Operator as follows:

[10.1] **MONITOR** the following ICS points while making adjustments to the AVR: AVRCA010 - AVR MVAR indication, AVRCD011 - AVR in AUTO/MAN, AVRCD015 - Over Excitation Limit ON/NORMAL, AVRCD016 - Under Excitation Limit ON/NORMAL.

[10.2] **NOTIFY** Unit 1 and Unit 2 that Unit 3 will be adjusting Generator MVARs.

[10.3] **MONITOR** AVR MVAR Indication for desired value (ICS pt. AVRCA010).

[10.4] **ENSURE** AVR is in AUTO (ICS pt. AVRCD011) (otherwise N/A with SM permission).

[10.5] **CHECK** AVR Under Excitation Limit is NORMAL (ICS pt. AVRCD016).

[10.6] **CHECK** AVR Over Excitation Limit is NORMAL (ICS pt. AVRCD015).

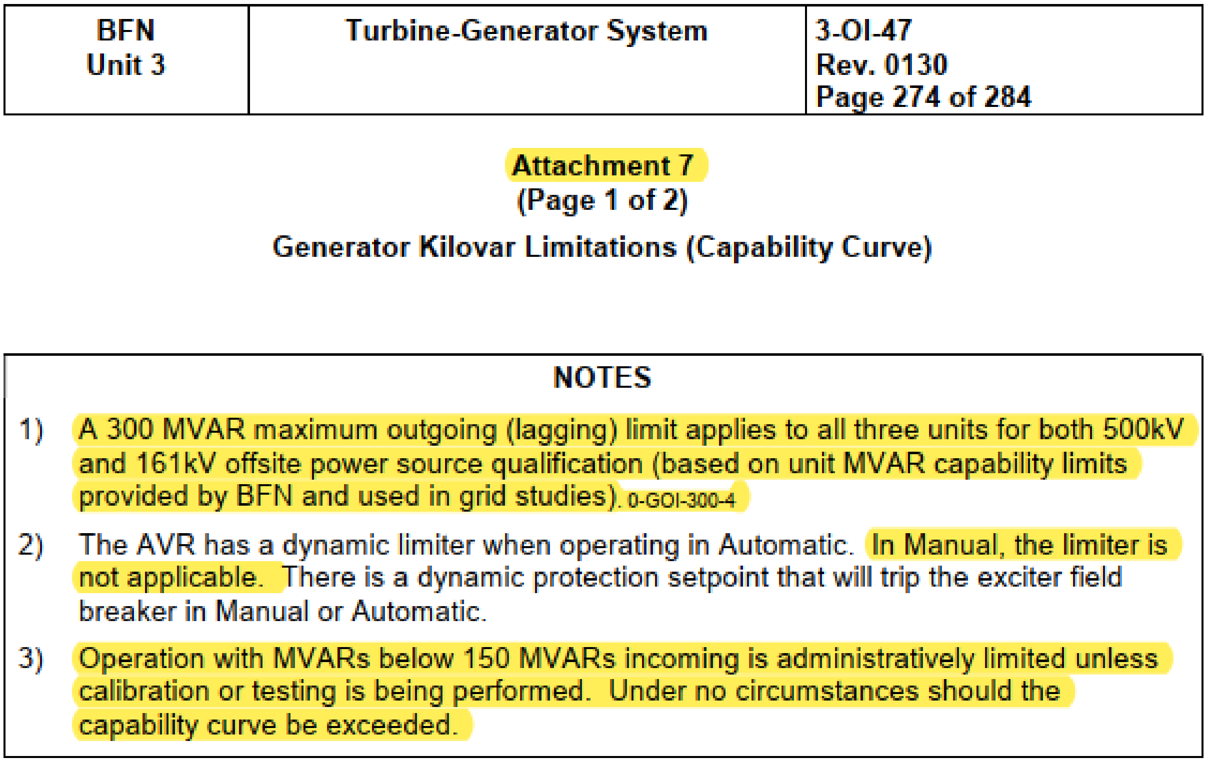
[10.7] To adjust GENERATOR MVAR in the positive or lagging direction,

Momentarily **PLACE** VOLTAGE REGULATOR LOWER/RAISE ADJUST, 3-HS-57-26, in RAISE.

[10.8] To adjust GENERATOR MVAR in the negative or leading direction,

Momentarily **PLACE** VOLTAGE REGULATOR LOWER/RAISE ADJUST, 3-HS-57-26, in LOWER.

[10.9] **Repeat** steps6.1[10.3] through 6.1[10.8] until desired MVAR is obtained or either Limit indicates ON.



BFN Unit 3	Turbine-Generator System	3-OI-47 Rev. 0130 Page 274 of 284

Attachment 7
(Page 1 of 2)

Generator Kilovar Limitations (Capability Curve)

NOTES

1) A 300 MVAR maximum outgoing (lagging) limit applies to all three units for both 500kV and 161kV offsite power source qualification (based on unit MVAR capability limits provided by BFN and used in grid studies). 0-GOI-300-4

2) The AVR has a dynamic limiter when operating in Automatic. In Manual, the limiter is not applicable. There is a dynamic protection setpoint that will trip the exciter field breaker in Manual or Automatic.

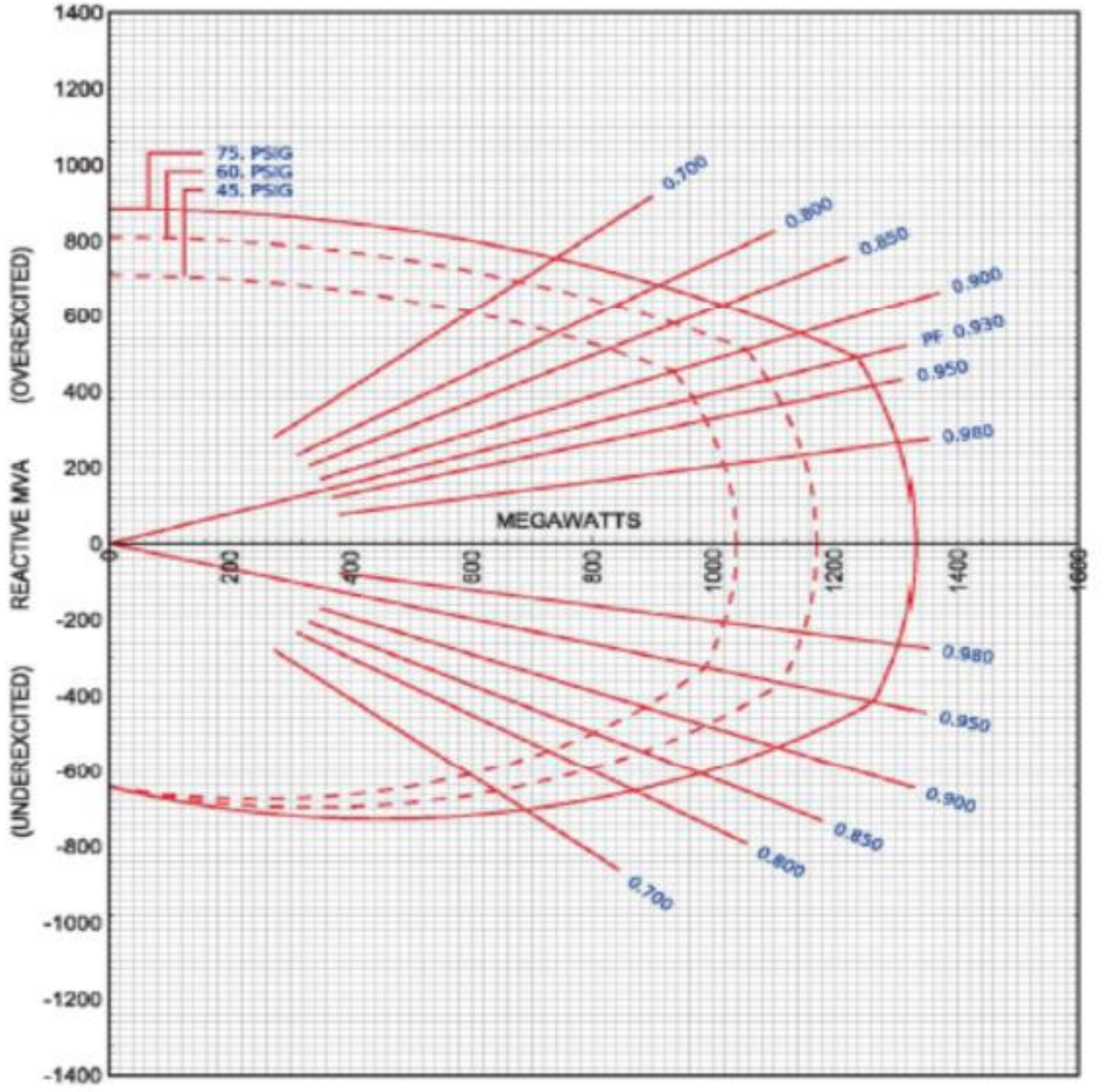
3) Operation with MVARs below 150 MVARs incoming is administratively limited unless calibration or testing is being performed. Under no circumstances should the capability curve be exceeded.

BFN **Unit 3**	**Turbine-Generator System**	**3-OI-47** **Rev. 0130** **Page 275 of 284**

Attachment 7
(Page 2 of 2)

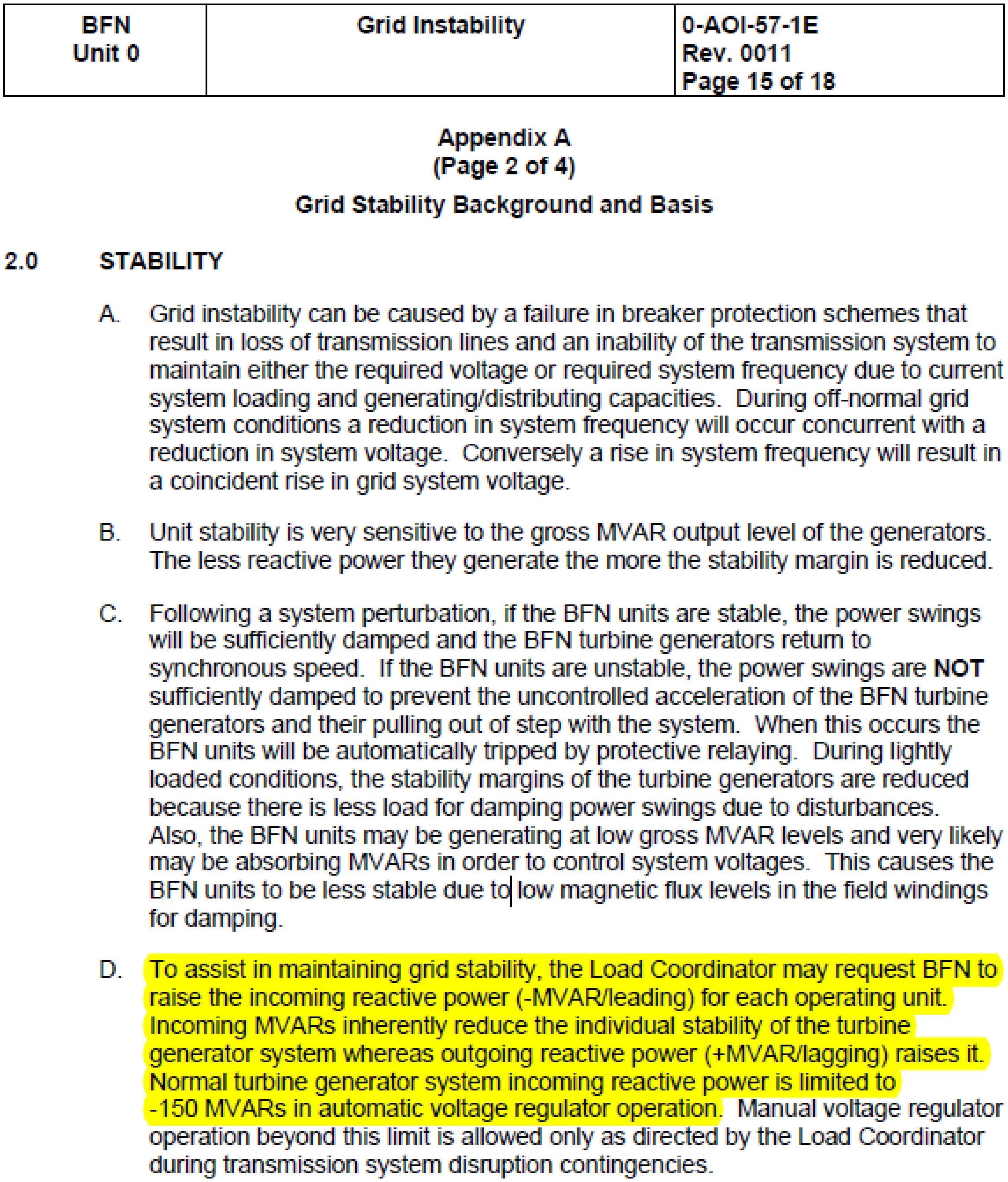
SIEMENS ENERGY, INC
CALCULATED CAPABILITY CURVES
AT 100 PERCENT VOLTAGE (22.0 kV)
WITH WATER COOLED STATOR
HYDROGEN INNER-COOLED TURBINE GENERATOR

CURVE NO. 090506GLC



1332.0 MVA PF 0.930 22.0 kV 34955 A
3 PHASE 60 Hz 1800 RPM SCR 0.58 75 PSIG
RATED COLD GAS 46 °C RATED STATOR WATER 50°C
DATE - 2015-06-26 CURVE NO. - 090506GLC

Excerpts from 0-AOI-57-1E:



BFN Unit 0	Grid Instability	0-AOI-57-1E Rev. 0011 Page 15 of 18

Appendix A
(Page 2 of 4)

Grid Stability Background and Basis

2.0 STABILITY

A. Grid instability can be caused by a failure in breaker protection schemes that result in loss of transmission lines and an inability of the transmission system to maintain either the required voltage or required system frequency due to current system loading and generating/distributing capacities. During off-normal grid system conditions a reduction in system frequency will occur concurrent with a reduction in system voltage. Conversely a rise in system frequency will result in a coincident rise in grid system voltage.

B. Unit stability is very sensitive to the gross MVAR output level of the generators. The less reactive power they generate the more the stability margin is reduced.

C. Following a system perturbation, if the BFN units are stable, the power swings will be sufficiently damped and the BFN turbine generators return to synchronous speed. If the BFN units are unstable, the power swings are **NOT** sufficiently damped to prevent the uncontrolled acceleration of the BFN turbine generators and their pulling out of step with the system. When this occurs the BFN units will be automatically tripped by protective relaying. During lightly loaded conditions, the stability margins of the turbine generators are reduced because there is less load for damping power swings due to disturbances. Also, the BFN units may be generating at low gross MVAR levels and very likely may be absorbing MVARs in order to control system voltages. This causes the BFN units to be less stable due to low magnetic flux levels in the field windings for damping.

D. To assist in maintaining grid stability, the Load Coordinator may request BFN to raise the incoming reactive power (-MVAR/leading) for each operating unit. Incoming MVARs inherently reduce the individual stability of the turbine generator system whereas outgoing reactive power (+MVAR/lagging) raises it. Normal turbine generator system incoming reactive power is limited to -150 MVARs in automatic voltage regulator operation. Manual voltage regulator operation beyond this limit is allowed only as directed by the Load Coordinator during transmission system disruption contingencies.

BFN **Unit 0**	**Grid Instability**	**0-AOI-57-1E** **Rev. 0011** **Page 7 of 18**

4.2 Subsequent Action (continued)

[6] **IF** grid instability is characterized by system voltage being maintained outside the normal limits of 520 ± 10 KV, **THEN**

PERFORM the following steps:

[6.1] **IF** system voltage is greater than 540KV, **THEN**

[6.1.1] **LOWER** reactive power to system voltage returns to 530KV, **OR UNTIL** Generator Reactive power reaches -150 MVAR. ☐

[6.1.2] **CHECK** 161KV Cap Banks are Out of Service and **EVALUATE** conditions to determine appropriate actions. **REFER TO** 0-GOI-300-4. ☐

[6.2] **IF** system voltage is lower than 510KV, **THEN**

PERFORM the following:

[6.2.1] **RAISE** reactive power to system voltage returns to 510 KV **OR UNTIL** Generator Reactive Power reaches +300 MVAR, ☐

BFN **Unit 0**	**Grid Instability**	**0-AOI-57-1E** **Rev. 0011** **Page 6 of 18**

4.2 Subsequent Action (continued)

NOTES

1) Changes in reactor power must comply with thermal power limits, rate of change limits and maximum power limits as specified in the unitized GOI-100-12.

2) Fluctuating Frequency and/or Voltages by ± 0.15 Hz or ± 5kV is indicative of too much generation for the system load. The Generator will have a tendency to pulsate due to coupling and momentum (with speed/frequency rising and lowering).

[4] **IF** System Frequency and/or Voltages are continually fluctuating by ± 0.15 Hertz or ± 5kV, **THEN**

VERIFY/INITIATE Recirc System upper power runback by DEPRESSING 1(2)(3)-HS-42, UPPER POWER RUNBACK. ☐

[5] **IF** grid instability is characterized by system frequency being maintained outside the normal limits of 60.0 ± 0.05 Hz, **THEN**

PERFORM the following:

[5.1] **IF** system frequency is greater than 60.15 Hz, **THEN**

LOWER reactor power by approximately 1%/minute (10 MW(e)/minute) **UNTIL** system frequency returns to 60.03 Hz. ☐

[5.2] **IF** system frequency is lower than 59.85 Hz **AND** reactor power is less than rated power, **THEN**

RAISE reactor power by approximately 1%/minute (10 MW(e)/minute) **UNTIL** system frequency returns to 59.98 Hz. ☐

Examination Outline Cross-reference:
295017 (APE 17) High Off-Site Radioactive Release Rate / 9
G2.1.30 (10CFR 55.41.7)
Ability to locate and operate components, including local controls

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295017G2.1.30	
Importance Rating	4.4	------

Proposed Question: **# 37**

Unit 2 is operating at 100% RTP when 2-RM-90-**265A**, Off-Gas Post-Treatment Radiation Monitor, fails **UPSCALE**.

One minute after the given conditions above, 2-FCV-66-28, Off-Gas System Isolation Valve, ___**(1)**___ **AUTOMATICALLY** close.

If directed to manually restrain the 2-FCV-66-28, Off-Gas System Isolation Valve, the AUO will be dispatched to the ___**(2)**___.

A. **(1)** will
(2) Turbine Building

B. **(1)** will
(2) Off Gas Stack

C. **(1)** will NOT
(2) Turbine Building

D. **(1)** will **NOT**
(2) Off Gas Stack

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the stem states that only one instrument is indicating upscale. The alarm OFFGAS POST TREATMENT RADIATION HIGH-HIGH, (2-9-4C, Window 34), does not state that 2-FCV-66-28 will automatically close. The candidate could easily confuse the required trip logic along with the purpose of the mechanical restraining device associated with 2-FCV-66-28. Second part is incorrect but plausible in that a significant amount of the Off-Gas system components are in the Turbine Building in various locations and on multiple elevations. Additionally, the Off-Gas system can be isolated in the Turbine Building.

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 alarm OFFGAS POST TREATMENT RADIATION MONITOR HI-HI-HI/INOP, (2-9-4C, Window 35), 2-FCV-66-28, OFFGAS SYSTEM ISOLATION VALVE closes after a 5-second time delay. 2-OI-66, Off-Gas System states that the automatic isolation of 2-FCV-66-28 occurs on any combination of 'HI-HI-HI', downscale, or inoperable trip simultaneously in both (two-out-of-two logic) trip channels (channel A - RM-90-266A and Channel B - RM-90-265A) of the Post-Treatment Radiation Monitoring System.
For second part, 2-FCV-66-28 is located near the end of the Off-Gas System flow path and is specifically located on the ground floor of the Off-Gas Stack in its own room.

RO Level Justification: Tests the candidate's ability to locate and operate Off-Gas System components including local controls as it relates to the Radiation Monitoring System and High Off-Site Radioactive Release Rate. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the isolation logic to predict an outcome. The candidate must mentally use this knowledge and its meaning to predict the correct outcome.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 2-ARP-9-4C, Rev. 36 (Attach if not previously provided)
2-OI-66, Rev. 120

Proposed references to be provided to applicants during examination: **NONE**

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

Question Source:	Bank #		
	Modified Bank #	BFN 2104 #15	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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

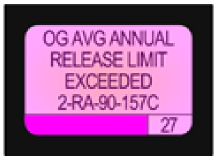
10 CFR Part 55 Content: 55.41 **X**
55.43

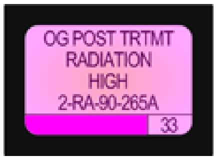
Comments:

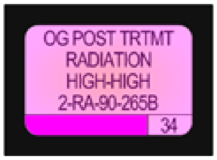
Copy of Bank Question:

Proposed Question: **# 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.


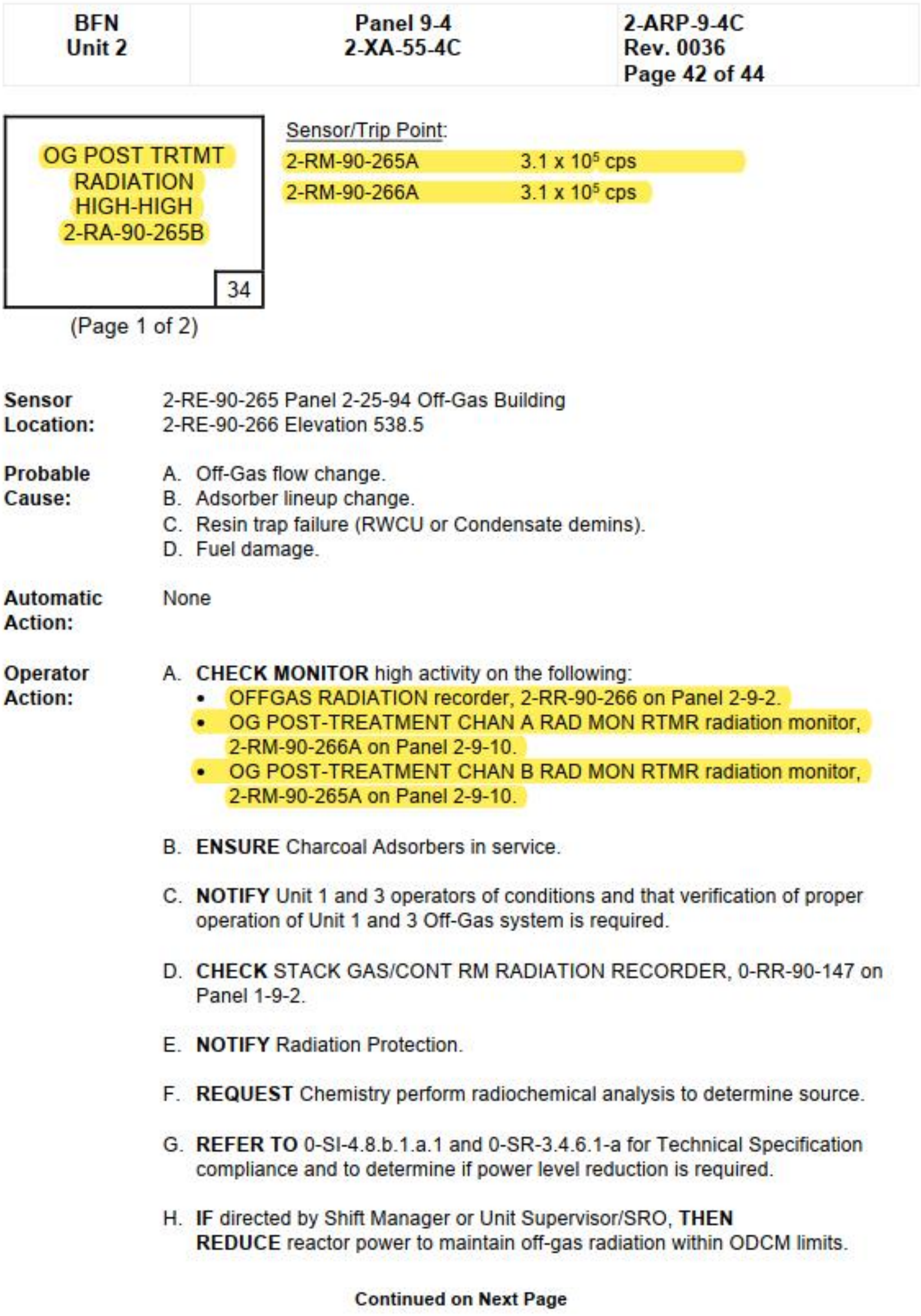
B.


C.


D.


Proposed Answer: **D**

Excerpts from 2-ARP-9-4C:



BFN Unit 2	**Panel 9-4 2-XA-55-4C**	**2-ARP-9-4C Rev. 0036 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 x 10^5 cps
2-RM-90-266A	3.1 x 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.

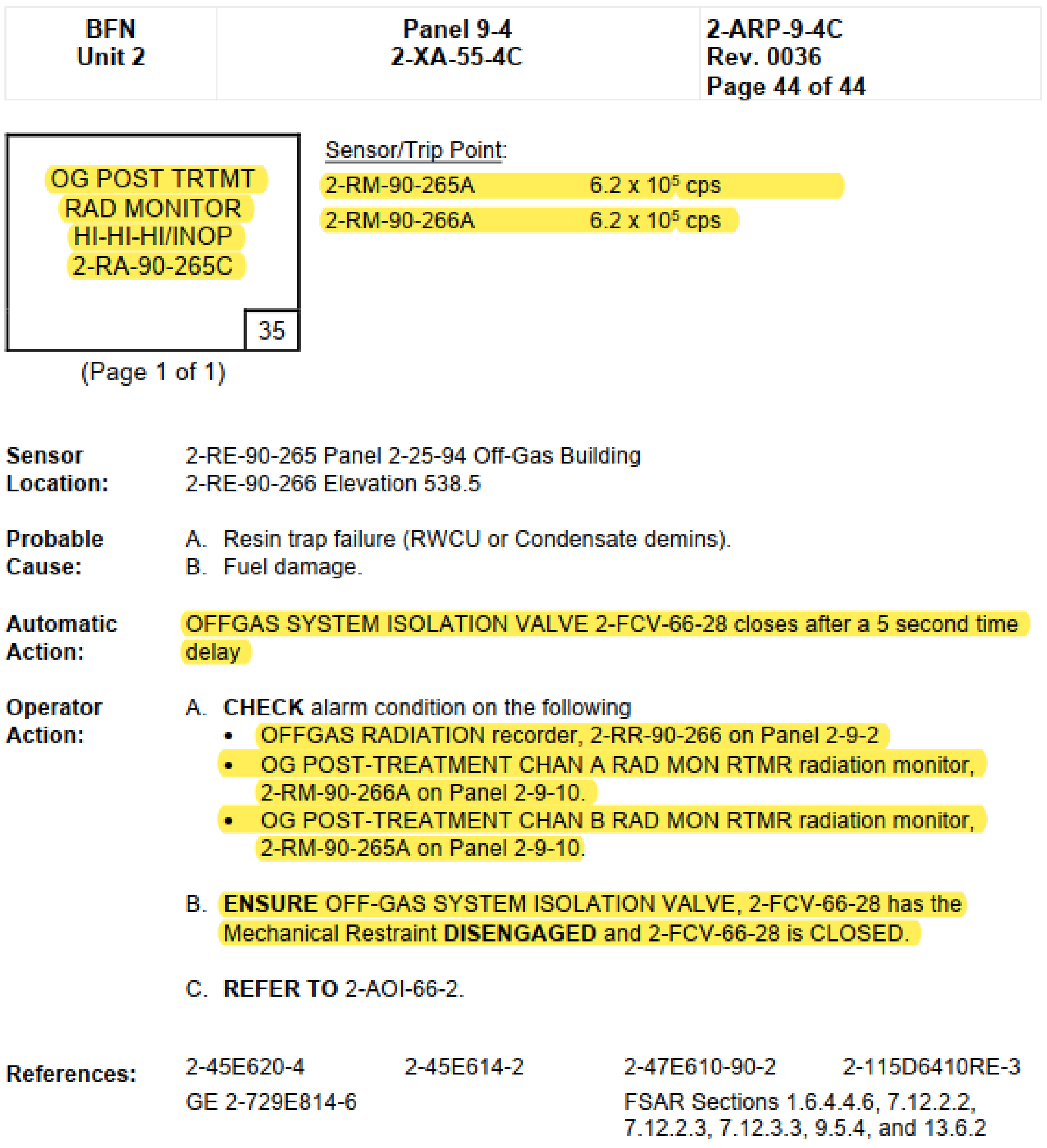
E. **NOTIFY** Radiation Protection.

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.

Continued on Next Page



BFN **Unit 2**	**Panel 9-4** **2-XA-55-4C**	**2-ARP-9-4C** **Rev. 0036** **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 x 10^5 cps
2-RM-90-266A	6.2 x 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.

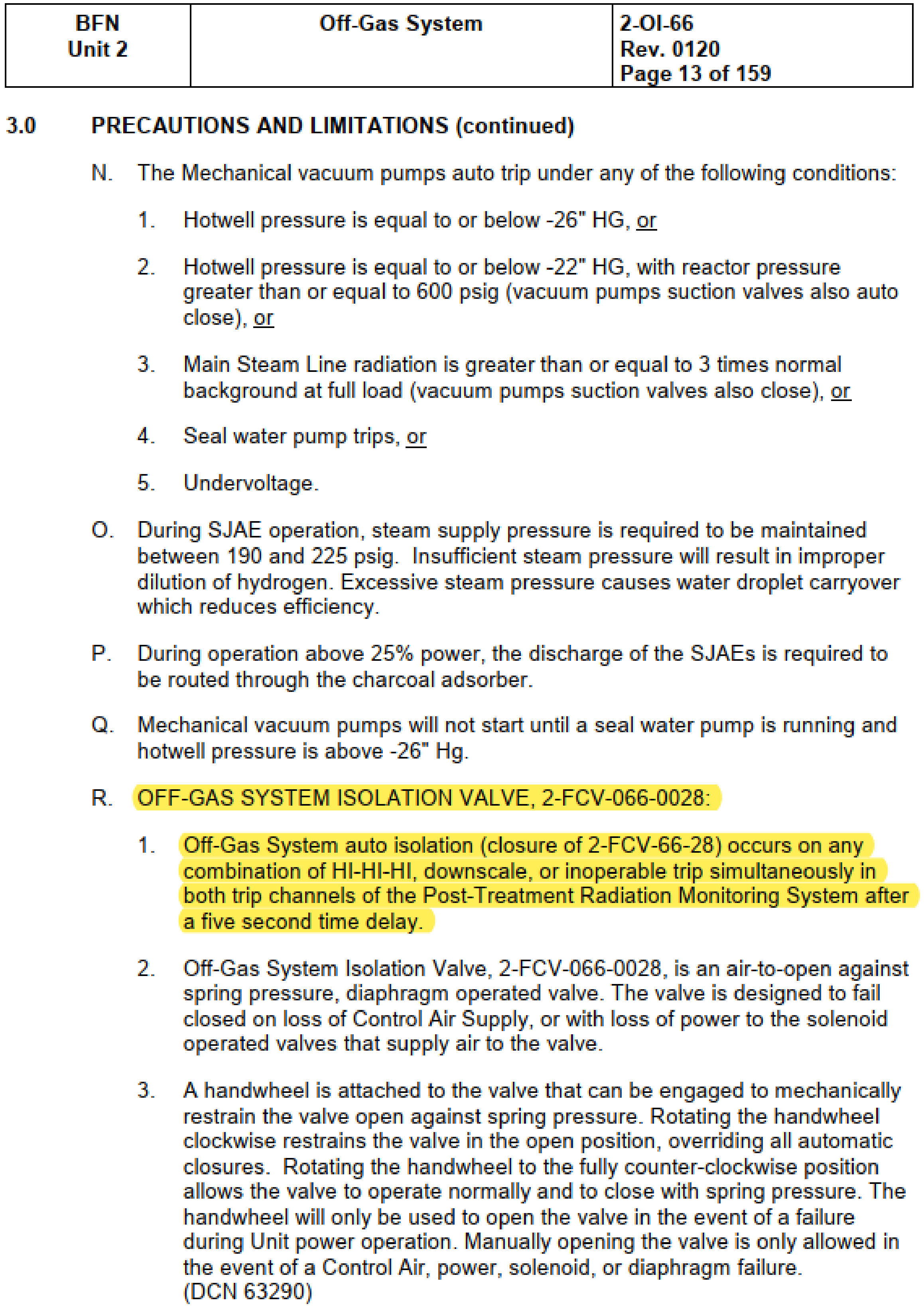
B. **ENSURE** OFF-GAS SYSTEM ISOLATION VALVE, 2-FCV-66-28 has the Mechanical Restraint **DISENGAGED** and 2-FCV-66-28 is CLOSED.

C. **REFER TO** 2-AOI-66-2.

References:

2-45E620-4	2-45E614-2	2-47E610-90-2	2-115D6410RE-3
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	

Excerpt from 2-OI-66:



BFN **Unit 2**	**Off-Gas System**	**2-OI-66** **Rev. 0120** **Page 13 of 159**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

N. The Mechanical vacuum pumps auto trip under any of the following conditions:

1. Hotwell pressure is equal to or below -26" HG, or
2. Hotwell pressure is equal to or below -22" HG, with reactor pressure greater than or equal to 600 psig (vacuum pumps suction valves also auto close), or
3. Main Steam Line radiation is greater than or equal to 3 times normal background at full load (vacuum pumps suction valves also close), or
4. Seal water pump trips, or
5. Undervoltage.

O. During SJAE operation, steam supply pressure is required to be maintained between 190 and 225 psig. Insufficient steam pressure will result in improper dilution of hydrogen. Excessive steam pressure causes water droplet carryover which reduces efficiency.

P. During operation above 25% power, the discharge of the SJAEs is required to be routed through the charcoal adsorber.

Q. Mechanical vacuum pumps will not start until a seal water pump is running and hotwell pressure is above -26" Hg.

R. OFF-GAS SYSTEM ISOLATION VALVE, 2-FCV-066-0028:

1. Off-Gas System auto isolation (closure of 2-FCV-66-28) occurs on any combination of HI-HI-HI, downscale, or inoperable trip simultaneously in both trip channels of the Post-Treatment Radiation Monitoring System after a five second time delay.
2. Off-Gas System Isolation Valve, 2-FCV-066-0028, is an air-to-open against spring pressure, diaphragm operated valve. The valve is designed to fail closed on loss of Control Air Supply, or with loss of power to the solenoid operated valves that supply air to the valve.
3. A handwheel is attached to the valve that can be engaged to mechanically restrain the valve open against spring pressure. Rotating the handwheel clockwise restrains the valve in the open position, overriding all automatic closures. Rotating the handwheel to the fully counter-clockwise position allows the valve to operate normally and to close with spring pressure. The handwheel will only be used to open the valve in the event of a failure during Unit power operation. Manually opening the valve is only allowed in the event of a Control Air, power, solenoid, or diaphragm failure.
(DCN 63290)

Examination Outline Cross-reference:	Level	RO	SRO
218000 (SF3 ADS) Automatic Depressurization	Tier #	2	------
G2.1.23 (10CFR 55.41.10)	Group #	1	------
Ability to perform general or normal operating procedures during any plant condition.	K/A #	218000G2.1.23	
	Importance Rating	4.3	------

Proposed Question: **# 38**

Unit 1 was operating at 100% RTP when the following conditions occur:

- An Automatic Depressurization System (ADS) valve failed open and **CANNOT** be closed
- OATC reports Reactor Power is 8% following the SCRAM
- Plant conditions require Emergency Depressurization

In accordance with EOIs, ADS ___**(1)**___ required to be inhibited.

In accordance with the EOI Program Manual, once directed to open 6 ADS Main Steam Relief Valves (SRV) for Emergency Depressurization, the Reactor Operator will open ___**(2)**___ additional SRVs.

A. (1) is
(2) 5

B. (1) is
(2) 6

C. (1) is NOT
(2) 5

D. (1) is NOT
(2) 6

Proposed Answer: **B**

Explanation (Optional):	A	INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that it is given that an ADS valve has failed open and cannot be closed, the candidate could believe that failed open ADS valve will count as 1 of the 6 and only 5 additional SRVs need to be opened.

B **CORRECT:** *(See attached)* In accordance with 1-EOI-1A, ATWS RPV CONTROL, ADS is required to be inhibited given that Reactor Power is above 5% in preparation of lowering Reactor Water Level to mitigate the ATWS conditions. For second part, in accordance with EOIPM 0-V(D), EOI-1A, ATWS RPV Control Bases, if one or more ADS valves cannot be opened, other MSRVs are required to be opened to effect the desired RPV depressurization until the total number of open MSRVs equals the number of MSRVs dedicated to ADS function which is 6. Additionally, in accordance with EOIPM 0-V(C), the direction to 'OPEN 6 MSRVs (ADS valves preferred)' requires manual action, even if the valves are already open on high pressure. Direct manual control must be established to ensure that the valves remain open as RPV Pressure lowers and will open if necessary, to prevent RPV repressurization.

C INCORRECT: First part is incorrect but plausible in that it is given that Emergency Depressurization is required, the candidate could believe that the ADS valves are going to be opened at some point at time so no need to inhibit ADS. 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 perform operating procedures during emergency conditions as it relates to Automatic Depressurization of the RPV. 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.

Technical Reference(s): EOIPM Section 0-V(C), Rev.1 (Attach if not previously provided)
EOIPM Section 0-V(D), Rev.1
1-EOI-1A, Rev. 1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.043 Obj. 6 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1804 #42	(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: **# 42**

Emergency Depressurization is required.

Which ONE of the following completes the statements below?

For each Unit at BFN, there are ___**(1)**___ ADS valves equipped with alternate power supplies.

In the event that one of the ADS valves previously failed open, that SRV ___**(2)**___ count as one of the required SRVs needed for Emergency Depressurization in accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation.

A. **(1)** Four (4)
(2) will

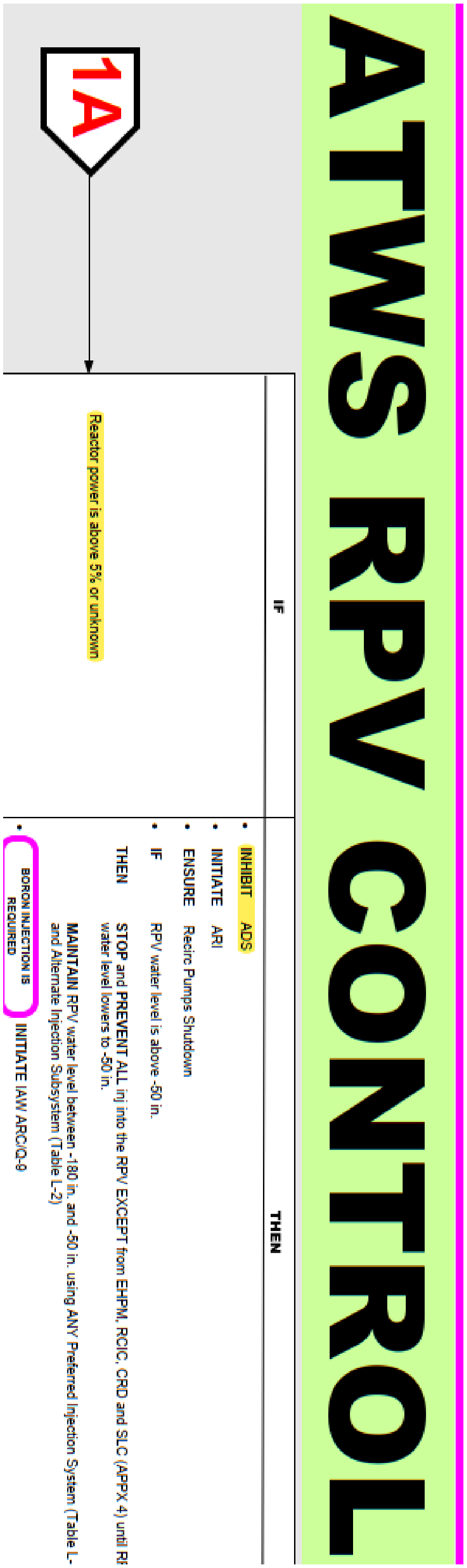
B. **(1)** Four (4)
(2) will **NOT**

C. **(1)** Five (5)
(2) will

D. **(1)** Five (5)
(2) will **NOT**

Proposed Answer: **B**

Excerpt from 1-EOI-1A:

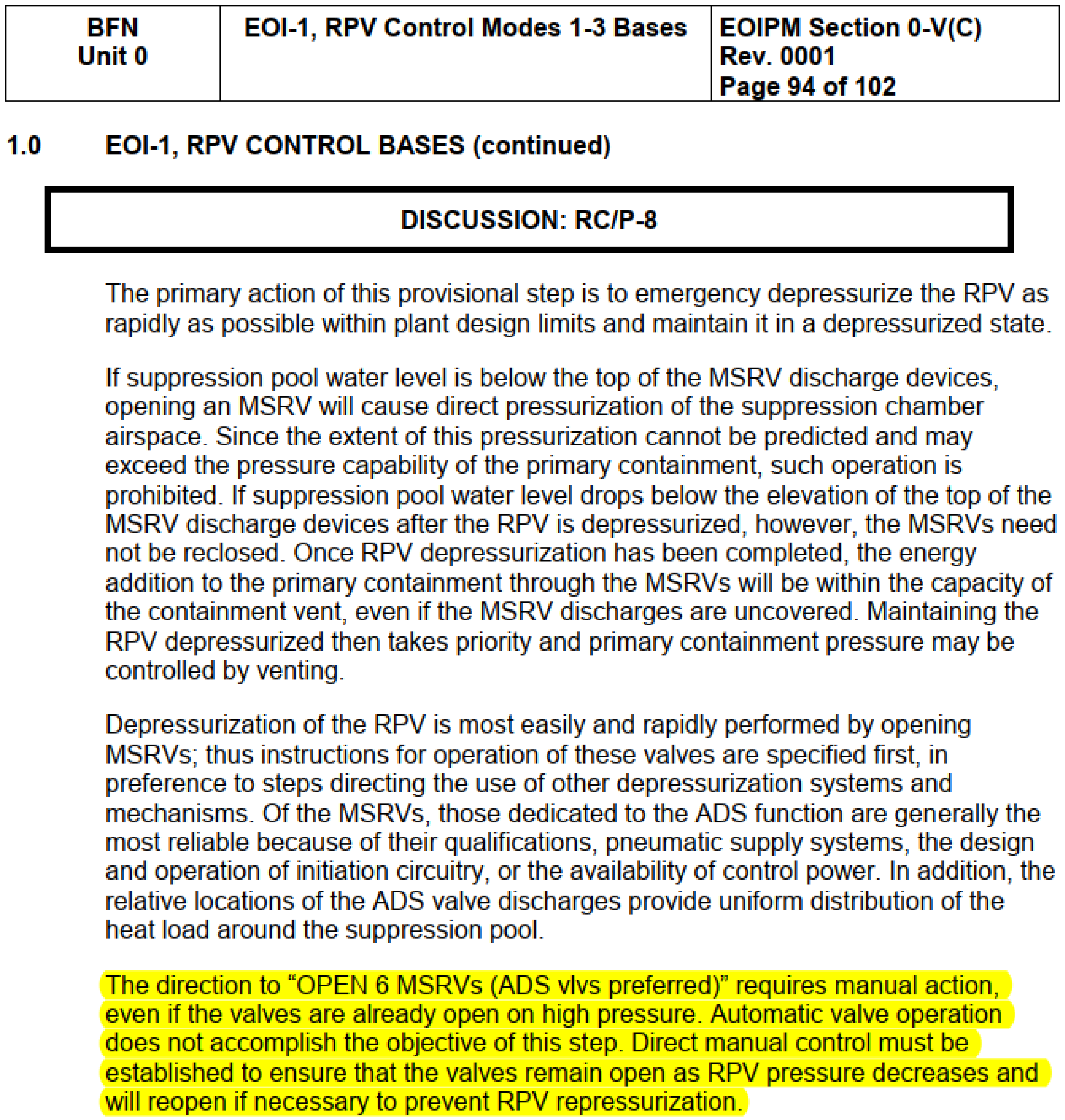


ATWS RPV CONTROL

1A

IF	THEN
Reactor power is above 5% or unknown	• INHIBIT ADS • INITIATE ARI • ENSURE Recirc Pumps Shutdown • IF RPV water level is above -50 in. THEN **STOP** and **PREVENT ALL** inj into the RPV EXCEPT from EHPM, RCIC, CRD and SLC (APPX 4) until RP water level lowers to -50 in. **MAINTAIN** RPV water level between -180 in. and -50 in. using ANY Preferred Injection System (Table L- and Alternate Injection Subsystem (Table L-2) • BORON INJECTION IS REQUIRED INITIATE IAW ARC/Q-9

Excerpt from EOIPM Section 0-V(C):



BFN **Unit 0**	**EOI-1, RPV Control Modes 1-3 Bases**	**EOIPM Section 0-V(C)** **Rev. 0001** **Page 94 of 102**

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/P-8

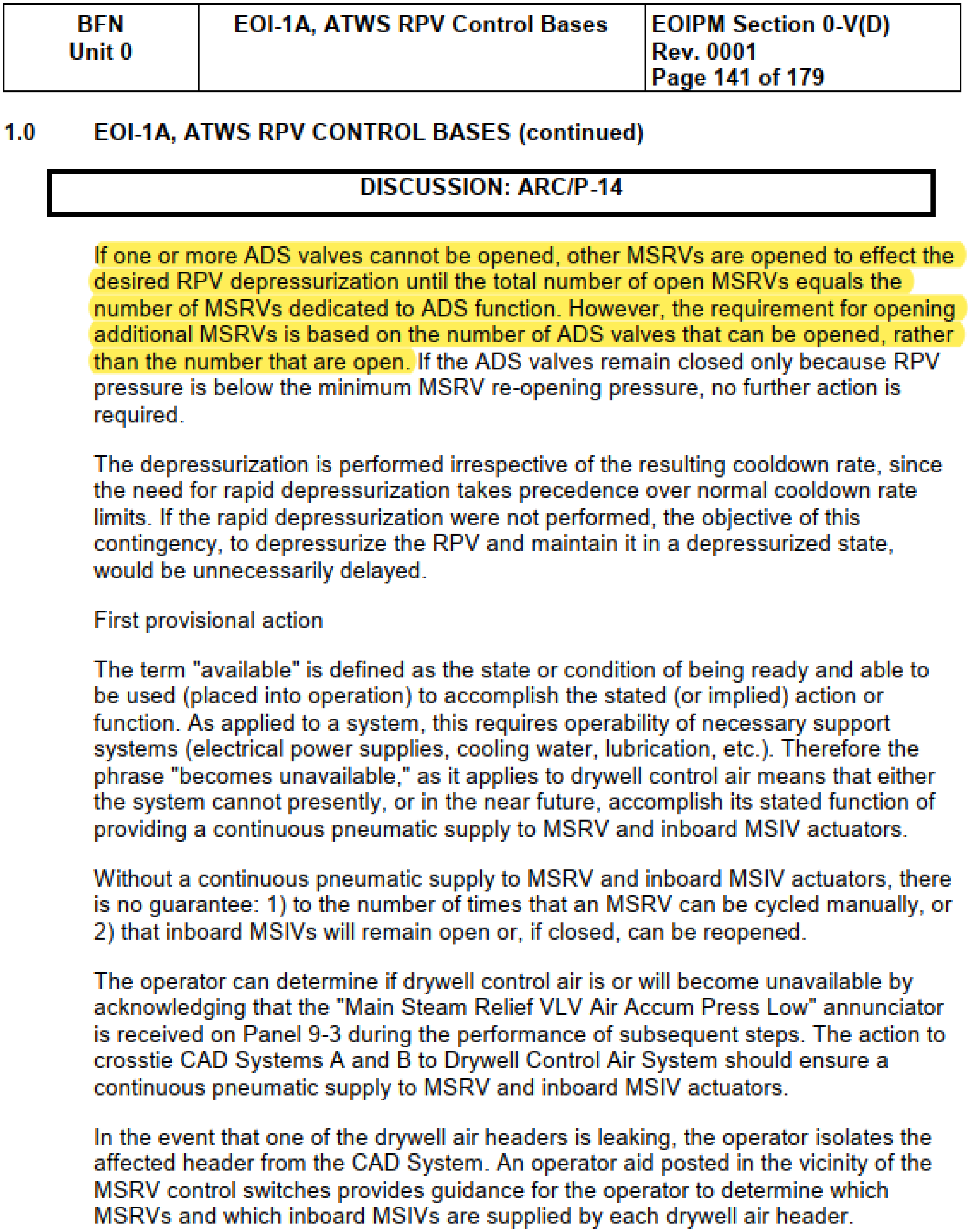
The primary action of this provisional step is to emergency depressurize the RPV as rapidly as possible within plant design limits and maintain it in a depressurized state.

If suppression pool water level is below the top of the MSRV discharge devices, opening an MSRV will cause direct pressurization of the suppression chamber airspace. Since the extent of this pressurization cannot be predicted and may exceed the pressure capability of the primary containment, such operation is prohibited. If suppression pool water level drops below the elevation of the top of the MSRV discharge devices after the RPV is depressurized, however, the MSRVs need not be reclosed. Once RPV depressurization has been completed, the energy addition to the primary containment through the MSRVs will be within the capacity of the containment vent, even if the MSRV discharges are uncovered. Maintaining the RPV depressurized then takes priority and primary containment pressure may be controlled by venting.

Depressurization of the RPV is most easily and rapidly performed by opening MSRVs; thus instructions for operation of these valves are specified first, in preference to steps directing the use of other depressurization systems and mechanisms. Of the MSRVs, those dedicated to the ADS function are generally the most reliable because of their qualifications, pneumatic supply systems, the design and operation of initiation circuitry, or the availability of control power. In addition, the relative locations of the ADS valve discharges provide uniform distribution of the heat load around the suppression pool.

The direction to “OPEN 6 MSRVs (ADS vlvs preferred)” requires manual action, even if the valves are already open on high pressure. Automatic valve operation does not accomplish the objective of this step. Direct manual control must be established to ensure that the valves remain open as RPV pressure decreases and will reopen if necessary to prevent RPV repressurization.

Excerpt from EOIPM Section 0-V(D):



BFN **Unit 0**	**EOI-1A, ATWS RPV Control Bases**	**EOIPM Section 0-V(D)** **Rev. 0001** **Page 141 of 179**

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/P-14

If one or more ADS valves cannot be opened, other MSRVs are opened to effect the desired RPV depressurization until the total number of open MSRVs equals the number of MSRVs dedicated to ADS function. However, the requirement for opening additional MSRVs is based on the number of ADS valves that can be opened, rather than the number that are open. If the ADS valves remain closed only because RPV pressure is below the minimum MSRV re-opening pressure, no further action is required.

The depressurization is performed irrespective of the resulting cooldown rate, since the need for rapid depressurization takes precedence over normal cooldown rate limits. If the rapid depressurization were not performed, the objective of this contingency, to depressurize the RPV and maintain it in a depressurized state, would be unnecessarily delayed.

First provisional action

The term "available" is defined as the state or condition of being ready and able to be used (placed into operation) to accomplish the stated (or implied) action or function. As applied to a system, this requires operability of necessary support systems (electrical power supplies, cooling water, lubrication, etc.). Therefore the phrase "becomes unavailable," as it applies to drywell control air means that either the system cannot presently, or in the near future, accomplish its stated function of providing a continuous pneumatic supply to MSRV and inboard MSIV actuators.

Without a continuous pneumatic supply to MSRV and inboard MSIV actuators, there is no guarantee: 1) to the number of times that an MSRV can be cycled manually, or 2) that inboard MSIVs will remain open or, if closed, can be reopened.

The operator can determine if drywell control air is or will become unavailable by acknowledging that the "Main Steam Relief VLV Air Accum Press Low" annunciator is received on Panel 9-3 during the performance of subsequent steps. The action to crosstie CAD Systems A and B to Drywell Control Air System should ensure a continuous pneumatic supply to MSRV and inboard MSIV actuators.

In the event that one of the drywell air headers is leaking, the operator isolates the affected header from the CAD System. An operator aid posted in the vicinity of the MSRV control switches provides guidance for the operator to determine which MSRVs and which inboard MSIVs are supplied by each drywell air header.

Examination Outline Cross-reference:	Level	RO	SRO
290003 (SF9 CRV) Control Room Ventilation	Tier #	2	------
K3.06 (10CFR 55.41.7)	Group #	2	------
Knowledge of the effect that a loss or malfunction of the Control Room Ventilation will have on the following systems or system parameters:	K/A #	290003K3.06	
- Control room radioactivity	Importance Rating	3.5	------

Proposed Question: **# 39**

Given the following conditions:

- A LOCA has occurred on Unit 2
- Control Bay Ventilation has isolated
- Control Room Emergency Ventilation System (CREV) ‘A’ is currently selected

If CREV ‘A’ fails to start, then CREV ‘B’ ___**(1)**___ start **AUTOMATICALLY**.

CREV is designed to maintain Control Room differential pressure ___**(2)**___ to limit the spread of radioactive contamination.

A. (1) will
(2) positive

B. (1) will
(2) negative

C. (1) will NOT
(2) positive

D. (1) will NOT
(2) negative

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (*See attached)* In accordance with 0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System, the CREV UNIT PRIMARY SELECTOR switch may be placed in the ‘A’ or ‘B’ train position to operate on a designed time delay. ‘A’ train in the preferred selected train with ‘B’ as the preferred standby. The standby CREV Unit (in this case, CREV ‘B’) start timer also counts down to start on an initiation signal, but with a 30 second time delay. If the selected CREV Unit fails to start within approximately 75 seconds, the standby CREV Unit (B) will automatically start due to a low differential pressure signal. For second part, in accordance with Technical Specification Bases 3.7.3, a single CREV Unit is designed to pressurize the Control Room to minimize air infiltration from outside the Control Room to maintain habitability.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that the Reactor Building Ventilation System is designed to prevent the release of radioactivity to the adjacent zones, which could be considered negative pressure to minimize the outflow to other zones.

C INCORRECT: First part is incorrect but plausible in that there is a selector switch in the CREV System to determine which CREV Unit is selected to start in the event that a valid CREV actuation signal is received. It is plausible that the candidate would expect that only the selected CREV Unit would start on an initiation signal. 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 start sequence of the Control Room Ventilation Systems and how Control Room radioactivity is controlled by the CREV System. 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): 0-OI-31, Rev. 166 (Attach if not previously provided)
OPL171.067, Rev. 22
FSAR 10.12, Rev. 22

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.067, Obj. 3, 18 (As available)

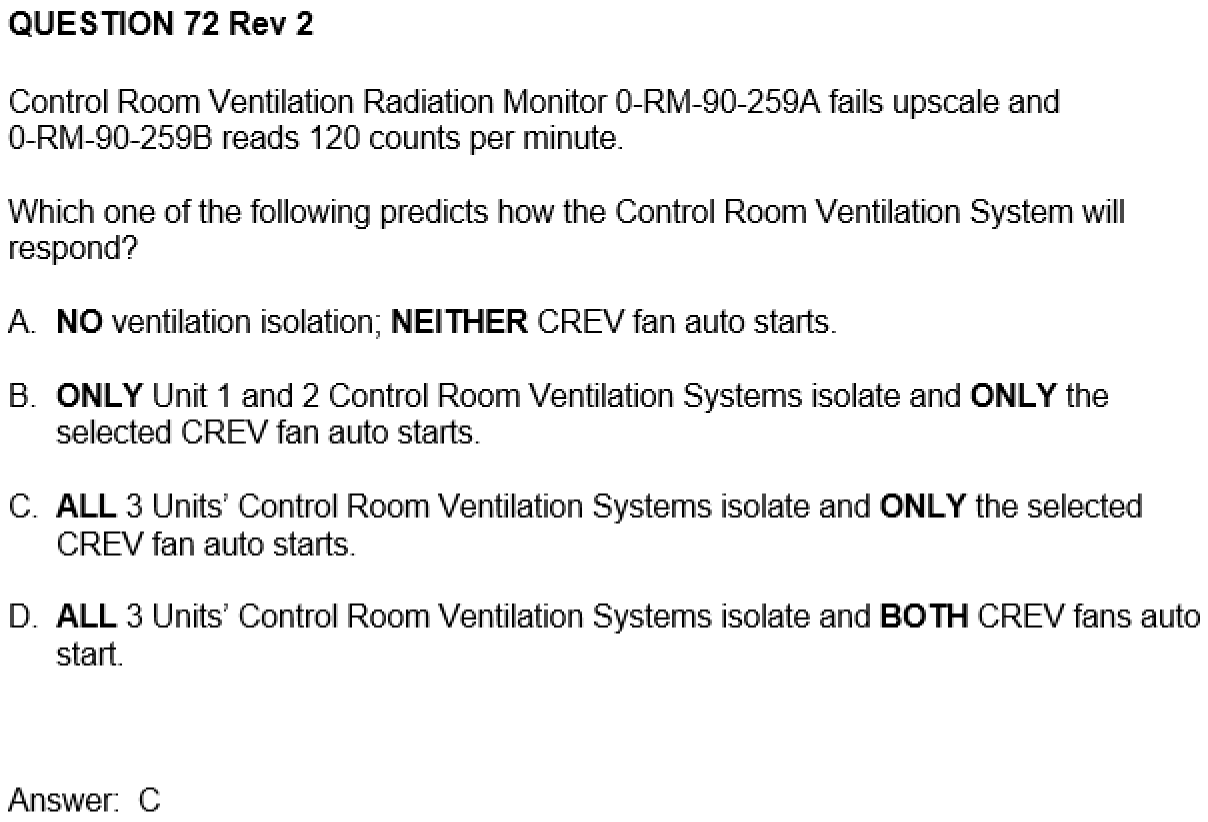
Question Source:	Bank #		
	Modified Bank #	BFN NRC 17-03 #72	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2017	

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 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.

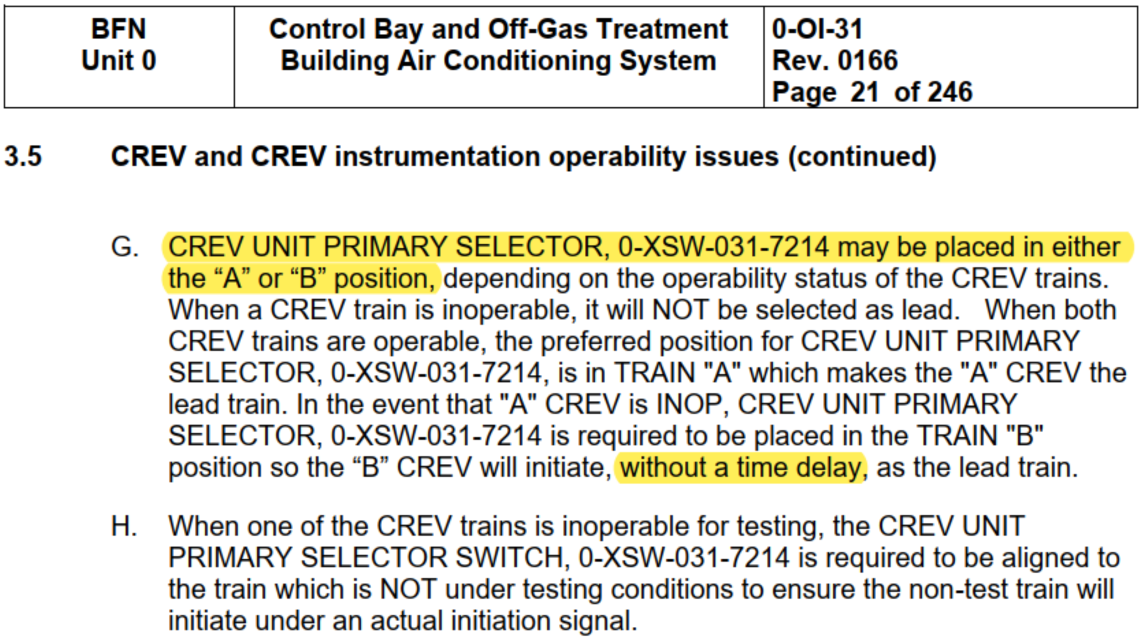
B. **ONLY** Unit 1 and 2 Control Room Ventilation Systems isolate and **ONLY** the selected CREV fan auto starts.

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

Excerpts from 0-OI-31:

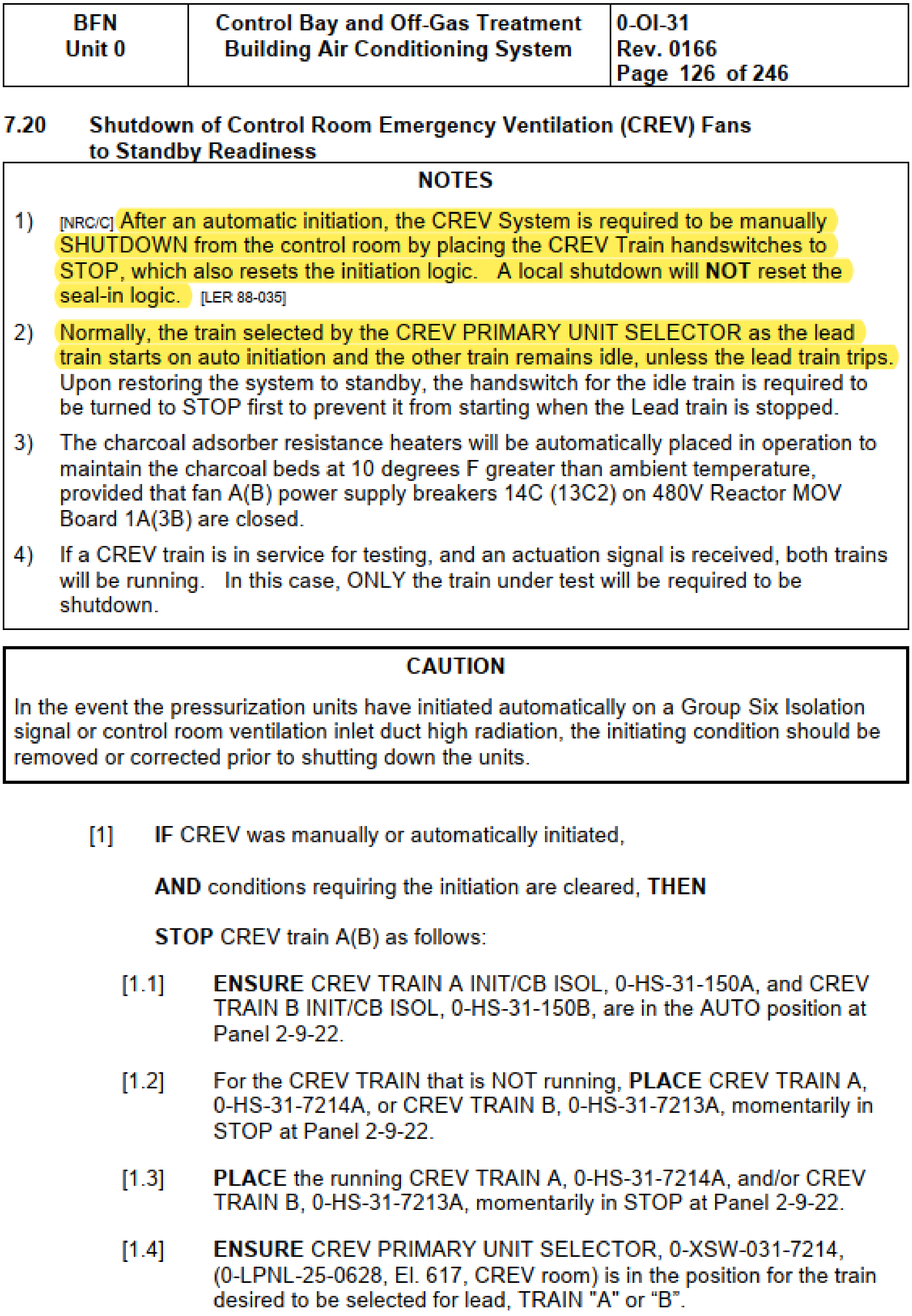


BFN **Unit 0**	**Control Bay and Off-Gas Treatment Building Air Conditioning System**	**0-OI-31** **Rev. 0166** **Page 21 of 246**

3.5 CREV and CREV instrumentation operability issues (continued)

G. CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 may be placed in either the “A” or “B” position, depending on the operability status of the CREV trains. When a CREV train is inoperable, it will NOT be selected as lead. When both CREV trains are operable, the preferred position for CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214, is in TRAIN "A" which makes the "A" CREV the lead train. In the event that "A" CREV is INOP, CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 is required to be placed in the TRAIN "B" position so the “B” CREV will initiate, without a time delay, as the lead train.

H. When one of the CREV trains is inoperable for testing, the CREV UNIT PRIMARY SELECTOR SWITCH, 0-XSW-031-7214 is required to be aligned to the train which is NOT under testing conditions to ensure the non-test train will initiate under an actual initiation signal.



BFN **Unit 0**	**Control Bay and Off-Gas Treatment Building Air Conditioning System**	**0-OI-31** **Rev. 0166** **Page 126 of 246**

7.20 Shutdown of Control Room Emergency Ventilation (CREV) Fans to Standby Readiness

NOTES

1) [NRC/C] After an automatic initiation, the CREV System is required to be manually SHUTDOWN from the control room by placing the CREV Train handswitches to STOP, which also resets the initiation logic. A local shutdown will **NOT** reset the seal-in logic. [LER 88-035]

2) Normally, the train selected by the CREV PRIMARY UNIT SELECTOR as the lead train starts on auto initiation and the other train remains idle, unless the lead train trips. Upon restoring the system to standby, the handswitch for the idle train is required to be turned to STOP first to prevent it from starting when the Lead train is stopped.

3) The charcoal adsorber resistance heaters will be automatically placed in operation to maintain the charcoal beds at 10 degrees F greater than ambient temperature, provided that fan A(B) power supply breakers 14C (13C2) on 480V Reactor MOV Board 1A(3B) are closed.

4) If a CREV train is in service for testing, and an actuation signal is received, both trains will be running. In this case, ONLY the train under test will be required to be shutdown.

CAUTION

In the event the pressurization units have initiated automatically on a Group Six Isolation signal or control room ventilation inlet duct high radiation, the initiating condition should be removed or corrected prior to shutting down the units.

[1] **IF** CREV was manually or automatically initiated,

AND conditions requiring the initiation are cleared, **THEN**

STOP CREV train A(B) as follows:

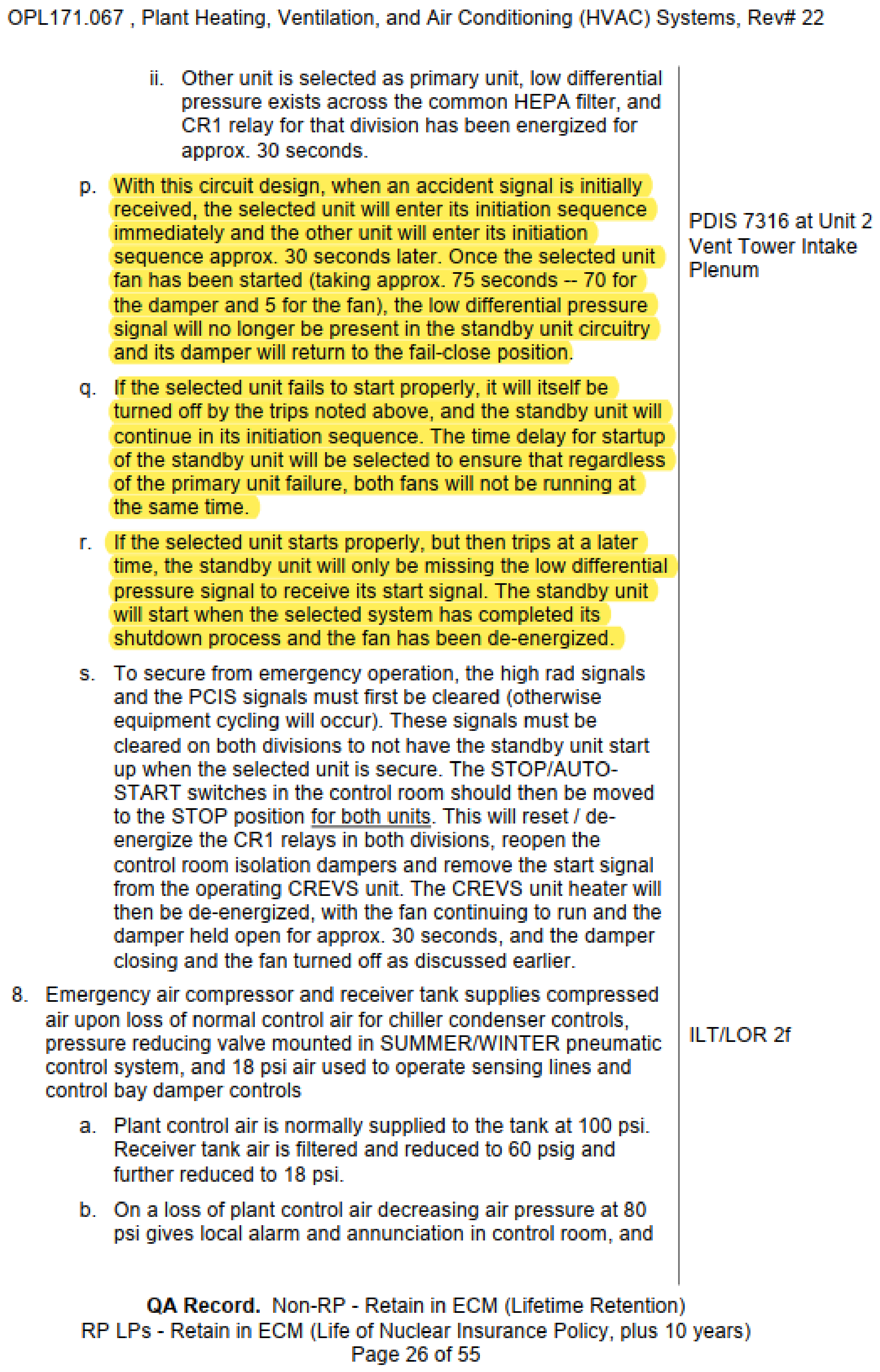
[1.1] **ENSURE** CREV TRAIN A INIT/CB ISOL, 0-HS-31-150A, and CREV TRAIN B INIT/CB ISOL, 0-HS-31-150B, are in the AUTO position at Panel 2-9-22.

[1.2] For the CREV TRAIN that is NOT running, **PLACE** CREV TRAIN A, 0-HS-31-7214A, or CREV TRAIN B, 0-HS-31-7213A, momentarily in STOP at Panel 2-9-22.

[1.3] **PLACE** the running CREV TRAIN A, 0-HS-31-7214A, and/or CREV TRAIN B, 0-HS-31-7213A, momentarily in STOP at Panel 2-9-22.

[1.4] **ENSURE** CREV PRIMARY UNIT SELECTOR, 0-XSW-031-7214, (0-LPNL-25-0628, El. 617, CREV room) is in the position for the train desired to be selected for lead, TRAIN "A" or "B".

Excerpt from OPL171.067 Lesson Plan:



OPL171.067 , Plant Heating, Ventilation, and Air Conditioning (HVAC) Systems, Rev# 22

   ii. Other unit is selected as primary unit, low differential pressure exists across the common HEPA filter, and CR1 relay for that division has been energized for approx. 30 seconds.

  p. With this circuit design, when an accident signal is initially received, the selected unit will enter its initiation sequence immediately and the other unit will enter its initiation sequence approx. 30 seconds later. Once the selected unit fan has been started (taking approx. 75 seconds -- 70 for the damper and 5 for the fan), the low differential pressure signal will no longer be present in the standby unit circuitry and its damper will return to the fail-close position.

PDIS 7316 at Unit 2 Vent Tower Intake Plenum

  q. If the selected unit fails to start properly, it will itself be turned off by the trips noted above, and the standby unit will continue in its initiation sequence. The time delay for startup of the standby unit will be selected to ensure that regardless of the primary unit failure, both fans will not be running at the same time.

  r. If the selected unit starts properly, but then trips at a later time, the standby unit will only be missing the low differential pressure signal to receive its start signal. The standby unit will start when the selected system has completed its shutdown process and the fan has been de-energized.

  s. To secure from emergency operation, the high rad signals and the PCIS signals must first be cleared (otherwise equipment cycling will occur). These signals must be cleared on both divisions to not have the standby unit start up when the selected unit is secure. The STOP/AUTO-START switches in the control room should then be moved to the STOP position for both units. This will reset / de-energize the CR1 relays in both divisions, reopen the control room isolation dampers and remove the start signal from the operating CREVS unit. The CREVS unit heater will then be de-energized, with the fan continuing to run and the damper held open for approx. 30 seconds, and the damper closing and the fan turned off as discussed earlier.

8. Emergency air compressor and receiver tank supplies compressed air upon loss of normal control air for chiller condenser controls, pressure reducing valve mounted in SUMMER/WINTER pneumatic control system, and 18 psi air used to operate sensing lines and control bay damper controls

ILT/LOR 2f

  a. Plant control air is normally supplied to the tank at 100 psi. Receiver tank air is filtered and reduced to 60 psig and further reduced to 18 psi.

  b. On a loss of plant control air decreasing air pressure at 80 psi gives local alarm and annunciation in control room, and

OPL171.067 , Plant Heating, Ventilation, and Air Conditioning (HVAC) Systems, Rev# 22

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

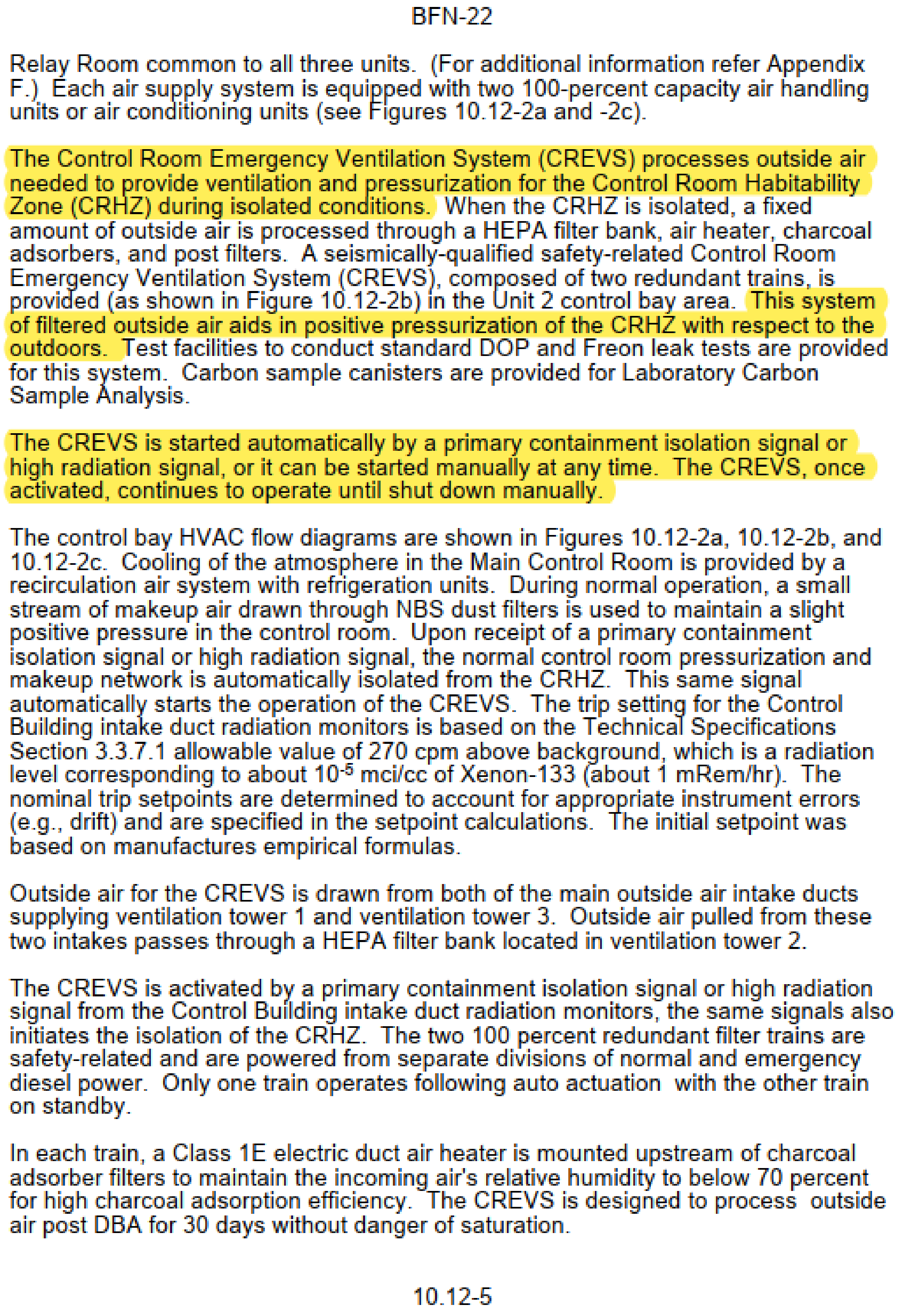
a. A CREV may be started manually from control room Panel 2-9-22 if local control switch is in AUTO position via a 3 position, spring-return to center switch. (STOP-AUTO-START). Actuates only the CREVS unit & associated damper, not the isolation dampers.

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.

Excerpt from FSAR, 10.12:



Relay Room common to all three units. (For additional information refer Appendix F.) Each air supply system is equipped with two 100-percent capacity air handling units or air conditioning units (see Figures 10.12-2a and -2c).

The Control Room Emergency Ventilation System (CREVS) processes outside air needed to provide ventilation and pressurization for the Control Room Habitability Zone (CRHZ) during isolated conditions. When the CRHZ is isolated, a fixed amount of outside air is processed through a HEPA filter bank, air heater, charcoal adsorbers, and post filters. A seismically-qualified safety-related Control Room Emergency Ventilation System (CREVS), composed of two redundant trains, is provided (as shown in Figure 10.12-2b) in the Unit 2 control bay area. This system of filtered outside air aids in positive pressurization of the CRHZ with respect to the outdoors. Test facilities to conduct standard DOP and Freon leak tests are provided for this system. Carbon sample canisters are provided for Laboratory Carbon Sample Analysis.

The CREVS is started automatically by a primary containment isolation signal or high radiation signal, or it can be started manually at any time. The CREVS, once activated, continues to operate until shut down manually.

The control bay HVAC flow diagrams are shown in Figures 10.12-2a, 10.12-2b, and 10.12-2c. Cooling of the atmosphere in the Main Control Room is provided by a recirculation air system with refrigeration units. During normal operation, a small stream of makeup air drawn through NBS dust filters is used to maintain a slight positive pressure in the control room. Upon receipt of a primary containment isolation signal or high radiation signal, the normal control room pressurization and makeup network is automatically isolated from the CRHZ. This same signal automatically starts the operation of the CREVS. The trip setting for the Control Building intake duct radiation monitors is based on the Technical Specifications Section 3.3.7.1 allowable value of 270 cpm above background, which is a radiation level corresponding to about 10^{-5} mci/cc of Xenon-133 (about 1 mRem/hr). The nominal trip setpoints are determined to account for appropriate instrument errors (e.g., drift) and are specified in the setpoint calculations. The initial setpoint was based on manufactures empirical formulas.

Outside air for the CREVS is drawn from both of the main outside air intake ducts supplying ventilation tower 1 and ventilation tower 3. Outside air pulled from these two intakes passes through a HEPA filter bank located in ventilation tower 2.

The CREVS is activated by a primary containment isolation signal or high radiation signal from the Control Building intake duct radiation monitors, the same signals also initiates the isolation of the CRHZ. The two 100 percent redundant filter trains are safety-related and are powered from separate divisions of normal and emergency diesel power. Only one train operates following auto actuation with the other train on standby.

In each train, a Class 1E electric duct air heater is mounted upstream of charcoal adsorber filters to maintain the incoming air's relative humidity to below 70 percent for high charcoal adsorption efficiency. The CREVS is designed to process outside air post DBA for 30 days without danger of saturation.

Examination Outline Cross-reference:	Level	RO	SRO
223002 (SF5 PCIS) Primary Containment Isolation / Nuclear Steam Supply Shutoff **K1.04** (10CFR 55.41.7)	Tier #	2	------
Knowledge of the physical connections and/or cause and effect relationships between the Primary Containment Isolation System/Nuclear Steam Supply Shutoff and the following systems:	Group #	1	------
	K/A #	223002K1.04	
- HPCI	Importance Rating	4.2	------

Proposed Question: **# 40**

Unit 2 is operating at 100% RTP and the HPCI System surveillance test was started.

A seismic event occurs that results in a HPCI steam leak with the following conditions:

- 250V RMOV BOARD 2A de-energized
- HPCI room temperature is 190 °F and rising
- Highest steam rate achieved was 160% of rated flow

Given the above conditions, HPCI received a PCIS Isolation signal from high **___(1)___**.

BOTH 2-FCV-73-2, HPCI STEAM LINE INBOARD ISOLATION VALVE **AND** 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE **___(2)___** close on the PCIS isolation signal.

A. (1) temperature
(2) will

B. (1) temperature
(2) will NOT

C. (1) steam flow rate
(2) will

D. (1) steam flow rate
(2) will NOT

Proposed Answer**:** **B**

Explanation (Optional):	A	INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that although 250V RMOV Board 2A is de-energized, HPCI will still isolate on high area temperature or high steam flow. 2-FCV-73-2, HPCI STEAM LINE INBOARD ISOLATION VALVE will isolate on the PCIS signal, but 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE will not because it has lost power (2A 250V DC RMOV Board). HPCI Logic has two power supplies: 250V DC RMOV Board 2B (HPCI Logic Bus A) and 250V DC RMOV Board 2A (HPCI Logic Bus B). A loss of 250V DC RMOV Board 2B will result in a loss of ALL isolation capability.

B **CORRECT:** (*See attached*) In accordance with 2-OI-73, High Pressure Coolant Injection System, area temperature has exceeded the isolation setpoint for the HPCI Room (185º F). For second part, although 2-FCV-73-2, HPCI STEAM LINE INBOARD ISOLATION VALVE will close on the temperature isolation signal, 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE has lost its power supply and therefore will not automatically close on the PCIS Isolation signal.

C INCORRECT: First part is incorrect but plausible in that the given steam flow does not exceed the HPCI PCIS Isolation setpoint (200%), but it does exceed the setpoint for a RCIC Isolation (150% in accordance with 2-OI-71, Reactor Core Isolation Cooling System). HPCI and RCIC have similar trips and isolations, but the values are different between the two systems and the two systems are often confused. 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: Given HPCI parameters, tests the candidate's knowledge of the effect of exceeding a PCIS Isolation setpoint for HPCI and the effect of a loss of HPCI Isolation Logic Power on the inboard and outboard HPCI Steam Supply Valves. 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. The candidate must know the HPCI PCIS Isolation setpoints and integrate that with the loss of power to one Division of HPCI Isolation Logic and determine how the isolation valves will respond.

Technical Reference(s): 2-OI-73, Rev. 101 (Attach if not previously provided)
2-AOI-64-2B, Rev.17
2-OI-71, Rev.75
OPL171.042, Rev. 24U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.043 Obj. 6 (As available)

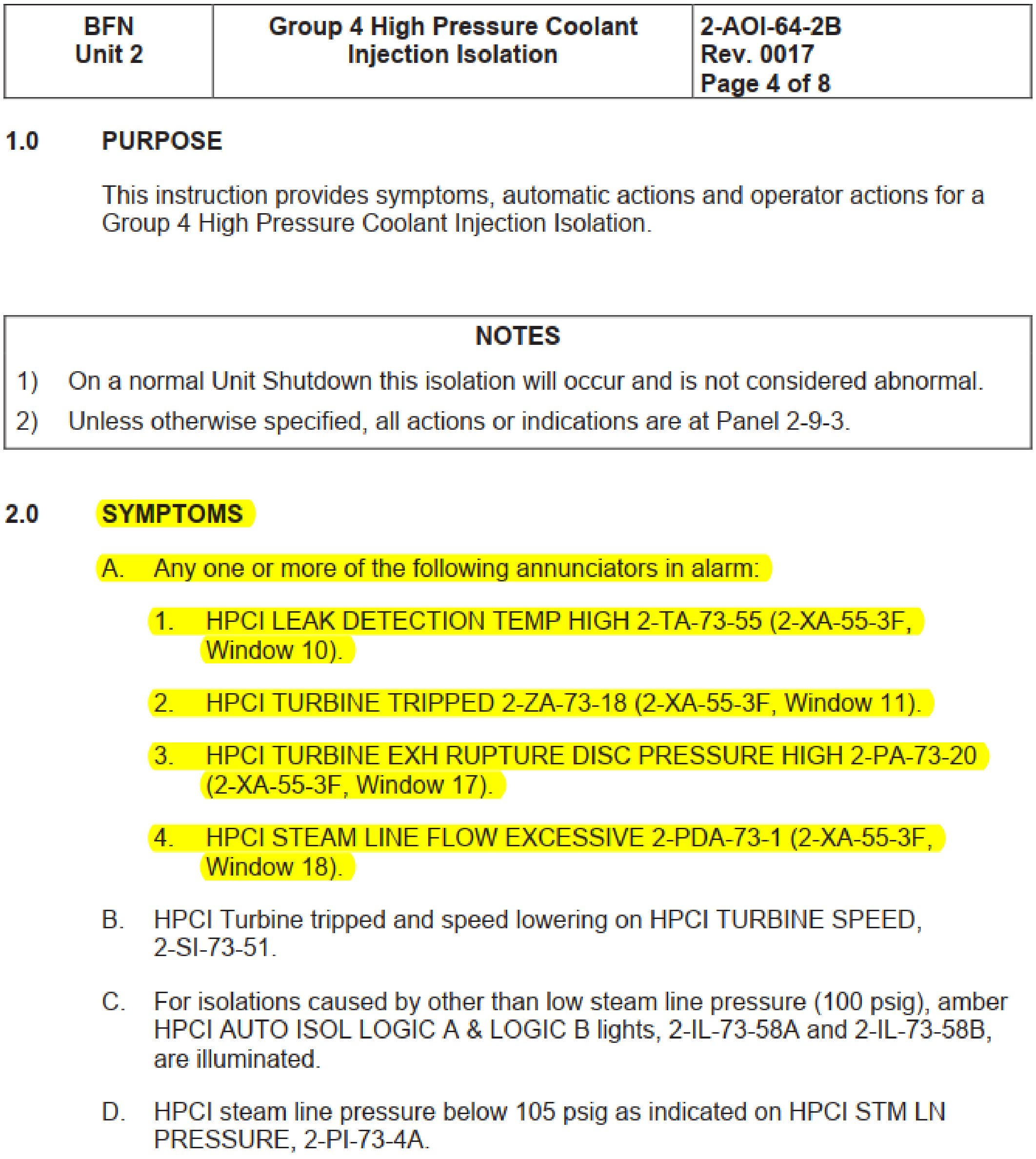
Question Source:	Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

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

Comments:

Excerpt from 2-AOI-64-2B:



BFN **Unit 2**	**Group 4 High Pressure Coolant Injection Isolation**	**2-AOI-64-2B** **Rev. 0017** **Page 4 of 8**

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 4 High Pressure Coolant Injection Isolation.

NOTES

1) On a normal Unit Shutdown this isolation will occur and is not considered abnormal.

2) Unless otherwise specified, all actions or indications are at Panel 2-9-3.

2.0 SYMPTOMS

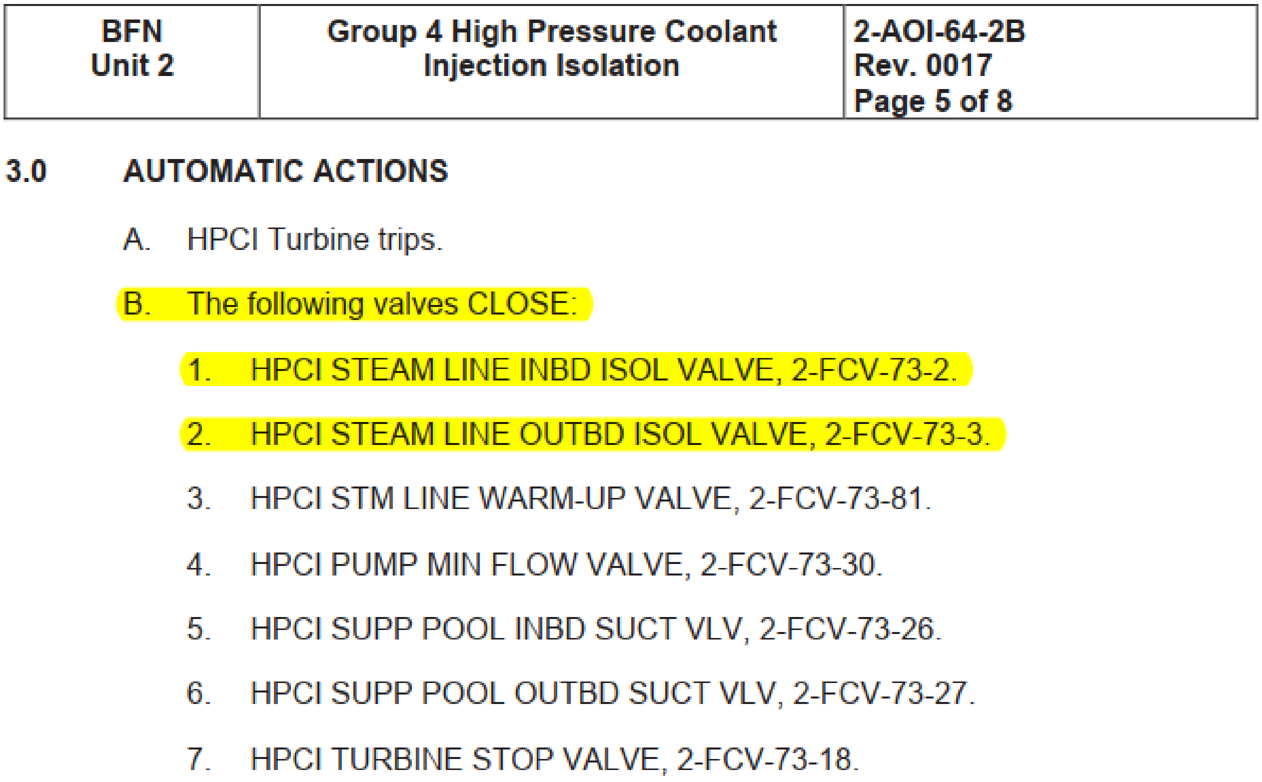
A. Any one or more of the following annunciators in alarm:

1. HPCI LEAK DETECTION TEMP HIGH 2-TA-73-55 (2-XA-55-3F, Window 10).
2. HPCI TURBINE TRIPPED 2-ZA-73-18 (2-XA-55-3F, Window 11).
3. HPCI TURBINE EXH RUPTURE DISC PRESSURE HIGH 2-PA-73-20 (2-XA-55-3F, Window 17).
4. HPCI STEAM LINE FLOW EXCESSIVE 2-PDA-73-1 (2-XA-55-3F, Window 18).

B. HPCI Turbine tripped and speed lowering on HPCI TURBINE SPEED, 2-SI-73-51.

C. For isolations caused by other than low steam line pressure (100 psig), amber HPCI AUTO ISOL LOGIC A & LOGIC B lights, 2-IL-73-58A and 2-IL-73-58B, are illuminated.

D. HPCI steam line pressure below 105 psig as indicated on HPCI STM LN PRESSURE, 2-PI-73-4A.



BFN Unit 2	**Group 4 High Pressure Coolant Injection Isolation**	**2-AOI-64-2B Rev. 0017 Page 5 of 8**

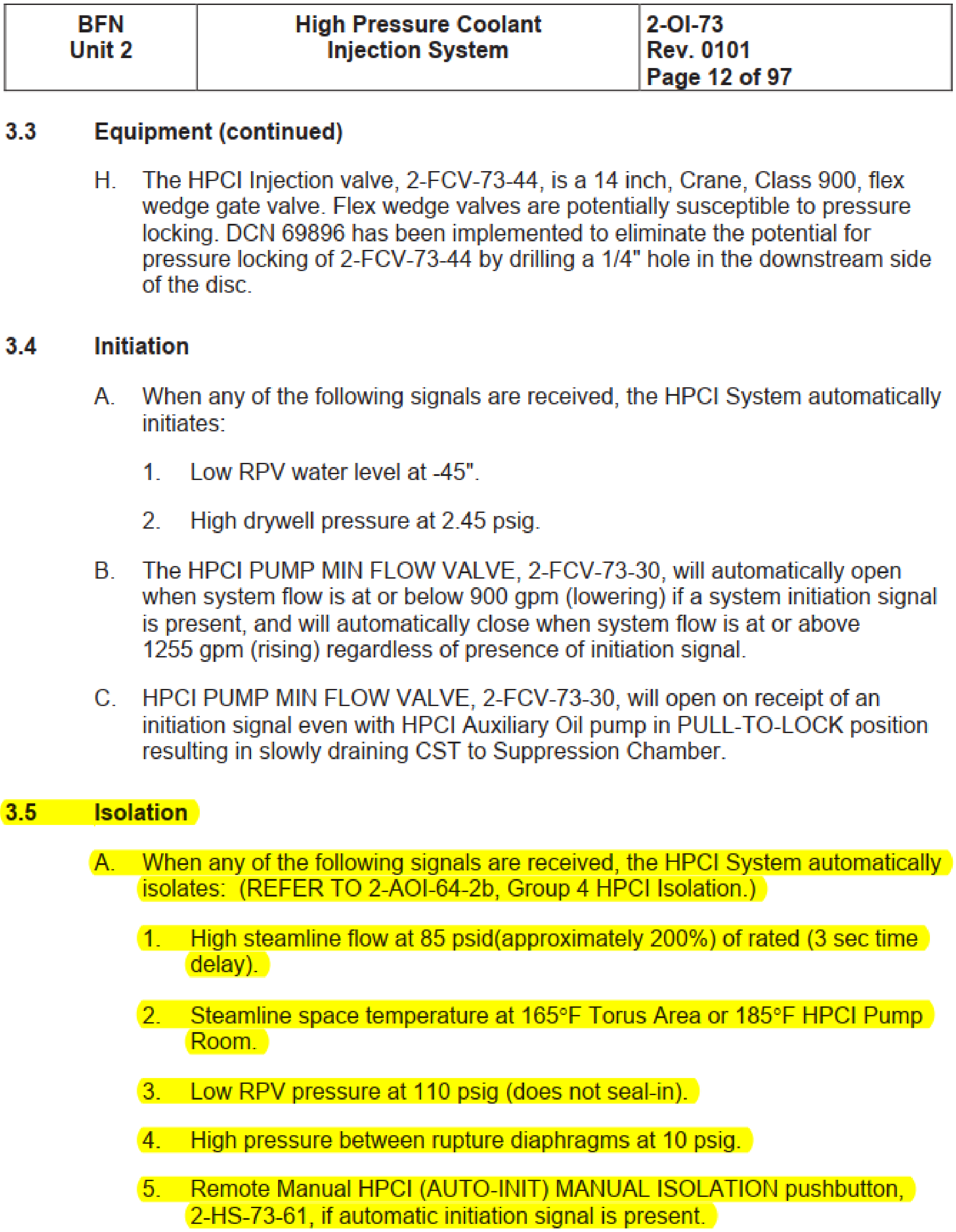
3.0 AUTOMATIC ACTIONS

A. HPCI Turbine trips.

B. The following valves CLOSE:

1. HPCI STEAM LINE INBD ISOL VALVE, 2-FCV-73-2.
2. HPCI STEAM LINE OUTBD ISOL VALVE, 2-FCV-73-3.
3. HPCI STM LINE WARM-UP VALVE, 2-FCV-73-81.
4. HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30.
5. HPCI SUPP POOL INBD SUCT VLV, 2-FCV-73-26.
6. HPCI SUPP POOL OUTBD SUCT VLV, 2-FCV-73-27.
7. HPCI TURBINE STOP VALVE, 2-FCV-73-18.

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.)

1. High steamline flow at 85 psid(approximately 200%) of rated (3 sec time delay).

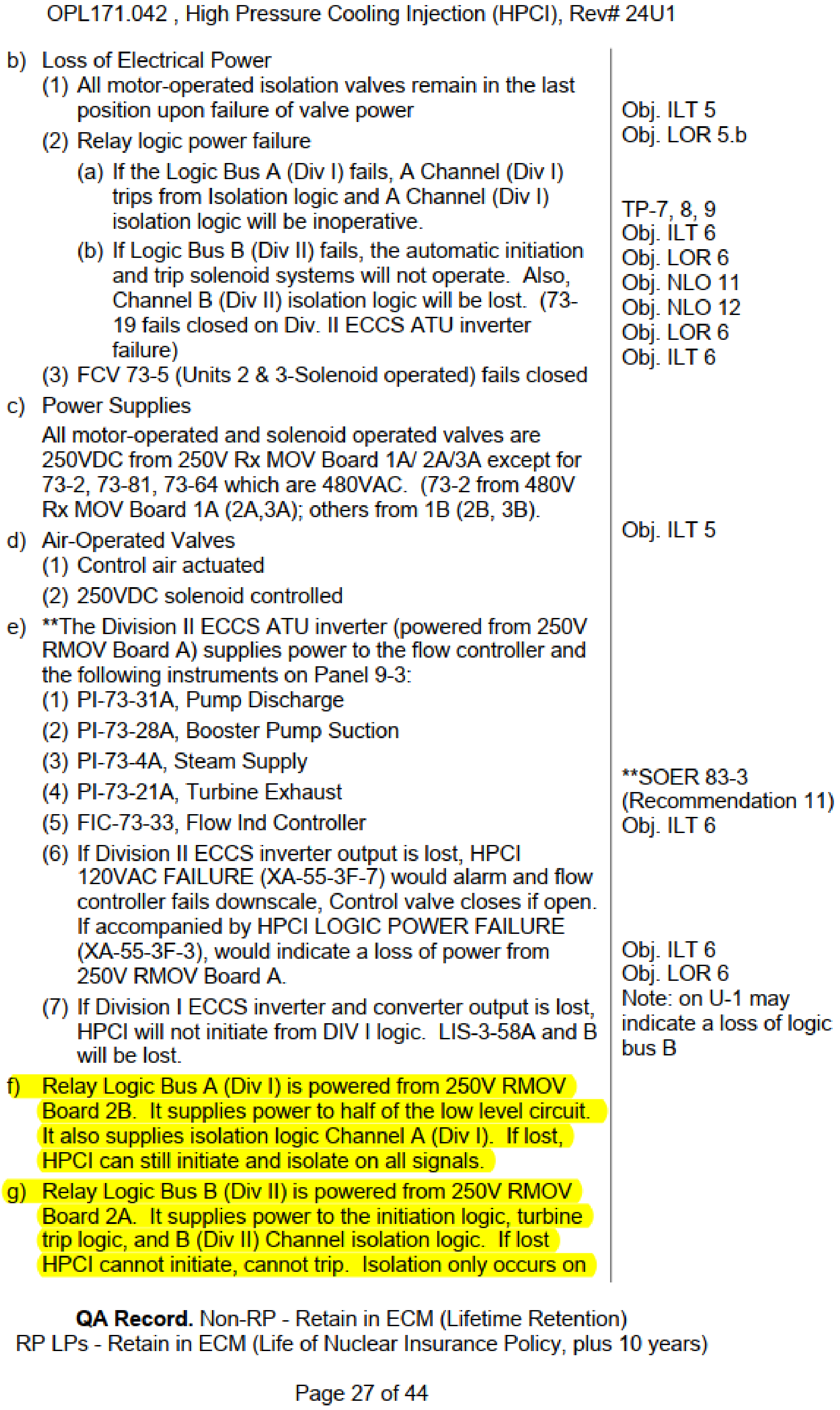
2. Steamline space temperature at 165°F Torus Area or 185°F HPCI Pump Room.

3. Low RPV pressure at 110 psig (does not seal-in).

4. High pressure between rupture diaphragms at 10 psig.

5. Remote Manual HPCI (AUTO-INIT) MANUAL ISOLATION pushbutton, 2-HS-73-61, if automatic initiation signal is present.

Excerpts from OPL171.042 Lesson Plan:



OPL171.042 , High Pressure Cooling Injection (HPCI), Rev# 24U1

b) Loss of Electrical Power
   (1) All motor-operated isolation valves remain in the last position upon failure of valve power
   (2) Relay logic power failure

Obj. ILT 5
Obj. LOR 5.b

      (a) If the Logic Bus A (Div I) fails, A Channel (Div I) trips from Isolation logic and A Channel (Div I) isolation logic will be inoperative.
      (b) If Logic Bus B (Div II) fails, the automatic initiation and trip solenoid systems will not operate. Also, Channel B (Div II) isolation logic will be lost. (73-19 fails closed on Div. II ECCS ATU inverter failure)
   (3) FCV 73-5 (Units 2 & 3-Solenoid operated) fails closed

TP-7, 8, 9
Obj. ILT 6
Obj. LOR 6
Obj. NLO 11
Obj. NLO 12
Obj. LOR 6
Obj. ILT 6

c) Power Supplies
   All motor-operated and solenoid operated valves are 250VDC from 250V Rx MOV Board 1A/ 2A/3A except for 73-2, 73-81, 73-64 which are 480VAC. (73-2 from 480V Rx MOV Board 1A (2A,3A); others from 1B (2B, 3B).

d) Air-Operated Valves
   (1) Control air actuated
   (2) 250VDC solenoid controlled

Obj. ILT 5

e) **The Division II ECCS ATU inverter (powered from 250V RMOV Board A) supplies power to the flow controller and the following instruments on Panel 9-3:
   (1) PI-73-31A, Pump Discharge
   (2) PI-73-28A, Booster Pump Suction
   (3) PI-73-4A, Steam Supply
   (4) PI-73-21A, Turbine Exhaust
   (5) FIC-73-33, Flow Ind Controller

**SOER 83-3
(Recommendation 11)
Obj. ILT 6

   (6) If Division II ECCS inverter output is lost, HPCI 120VAC FAILURE (XA-55-3F-7) would alarm and flow controller fails downscale, Control valve closes if open. If accompanied by HPCI LOGIC POWER FAILURE (XA-55-3F-3), would indicate a loss of power from 250V RMOV Board A.
   (7) If Division I ECCS inverter and converter output is lost, HPCI will not initiate from DIV I logic. LIS-3-58A and B will be lost.

Obj. ILT 6
Obj. LOR 6
Note: on U-1 may indicate a loss of logic bus B

f) Relay Logic Bus A (Div I) is powered from 250V RMOV Board 2B. It supplies power to half of the low level circuit. It also supplies isolation logic Channel A (Div I). If lost, HPCI can still initiate and isolate on all signals.

g) Relay Logic Bus B (Div II) is powered from 250V RMOV Board 2A. It supplies power to the initiation logic, turbine trip logic, and B (Div II) Channel isolation logic. If lost HPCI cannot initiate, cannot trip. Isolation only occurs on high steam flow or high area temperature (from Div I logic).

6. Technical Specifications/TRM
   a) Tech Spec Section 3.5.1, 3.3.5.1, 3.3.6.1, 3.6.1.3
   b) TRM Section 3.3.3.4, 3.74
7. Procedure Review
   a) Review 1/2/3-OI-73
   b) Review 1/2/3-AOI-64-2B
8. Plant Observation Session:

   Locate the components listed in Objective NLO-14

Obj. ILT 10
Obj. LOR 10
Obj. NLOR 10

Obj. NLO 14

G. Plant/Operator Experience:

(Also see INPO: https://apps.inpo.org/xICES)

1. BFN U2 - 09/16/15 - #318401

   Description

   The leak occurred following 2-SR-3.6.1.3.5 HPCI System MOV Operability, in which the U2 HPCI turbine steam supply valve was the last valve cycled. Operations received an auto-start of the A and B electric fire pumps, and a report was made from the field that U2 HPCI room had a significant amount of steam present. No Area Radiation Monitoring or PCIS area high temp alarms were received and no auto isolation setpoints were reached. HPCI was declared inop per Technical Specification (TS) LCO 3.5.1 ECCS - Operating, Condition C.

2. Browns Ferry Unit 2- #11024

   Description

   While performing U2 HPCI rated flow test, HPCI suction valves auto transferred to torus, a pipe break occurred in 20" condensate test return line to CST.

   The pipe failure was caused by combination of an inadequate weld and pipe movement and excessive loading caused by several rapid changes in HPCI flow.

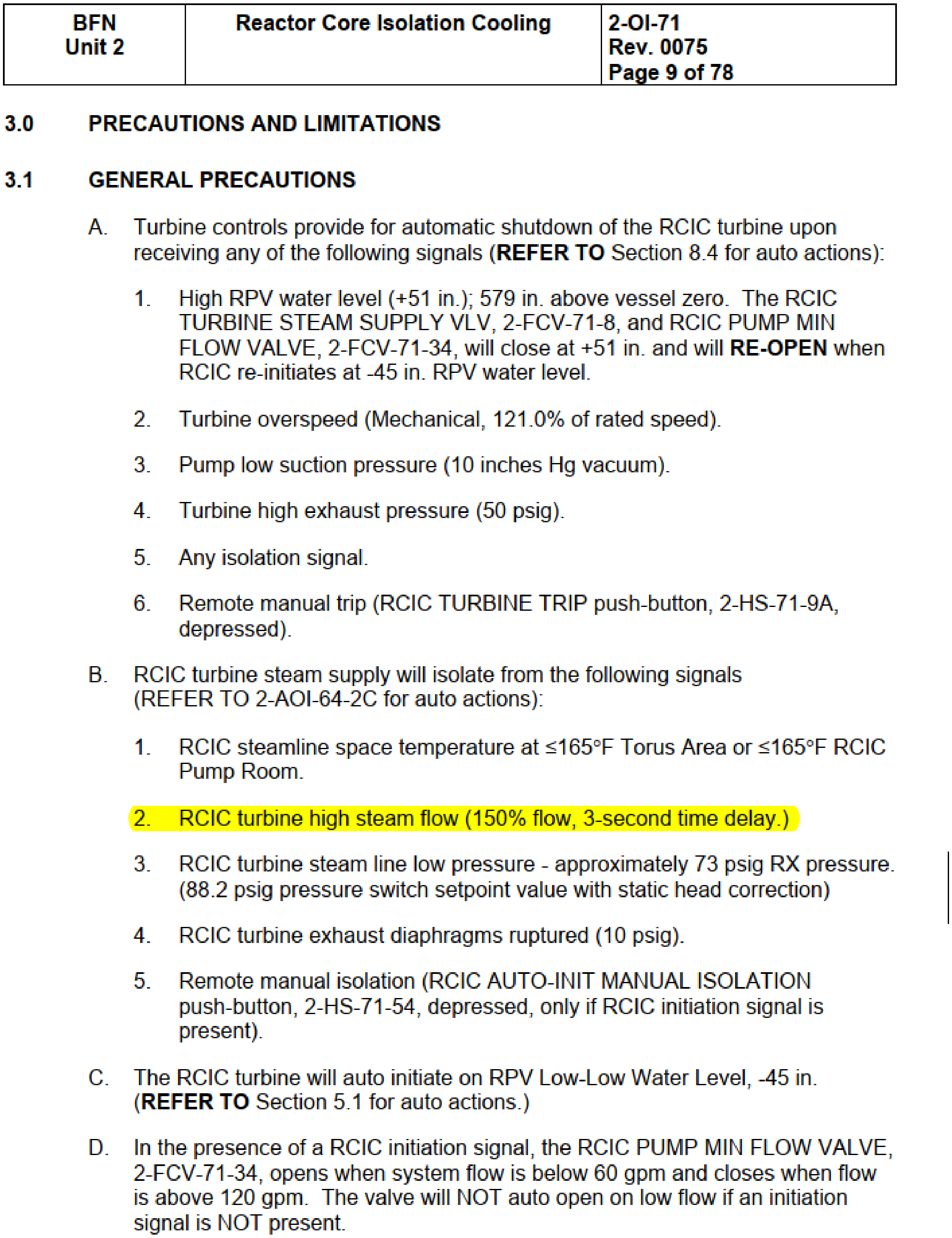
   To help prevent from happening again, HPCI flow rate SR has been modified to ensure that flow and pressure are stabilized prior to making any adjustments on test return valve (73-35). Also, steps have been added to SR to lift seal-in leads in 73-36 valve control circuit to allow it to be throttled when testing HPCI flow. This "double throttling" makes HPCI flow much less sensitive to adjustments on test return valve 73-35.

3. BFN U2 - 06/17/16 - #323263

   Description

   While performing U2's HPCI Time Delay Relay Calibration SR, electrical maintenance received an abnormal indication of no voltage to RLY 23A-K43. The electricians backed out of the procedure, informed Operations, and initiated a CR. Later that day, a different crew performed the remaining sections of the

Excerpt from 2-OI-71: Supports Distractors C(1), D(1)



BFN **Unit 2**	**Reactor Core Isolation Cooling**	**2-OI-71** **Rev. 0075** **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).

B. RCIC turbine steam supply will isolate from the following signals (REFER TO 2-AOI-64-2C for auto actions):

1. RCIC steamline space temperature at ≤165°F Torus Area or ≤165°F RCIC Pump Room.
2. RCIC turbine high steam flow (150% flow, 3-second time delay.)
3. RCIC turbine steam line low pressure - approximately 73 psig RX pressure. (88.2 psig pressure switch setpoint value with static head correction)
4. RCIC turbine exhaust diaphragms ruptured (10 psig).
5. Remote manual isolation (RCIC AUTO-INIT MANUAL ISOLATION push-button, 2-HS-71-54, depressed, only if RCIC initiation signal is present).

C. The RCIC turbine will auto initiate on RPV Low-Low Water Level, -45 in. (**REFER TO** Section 5.1 for auto actions.)

D. In the presence of a RCIC initiation signal, the RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, opens when system flow is below 60 gpm and closes when flow is above 120 gpm. The valve will NOT auto open on low flow if an initiation signal is NOT present.

Examination Outline Cross-reference:

295005 (APE 5) Main Turbine Generator Trip /3

AK2.03 (10CFR 55.41.7)

Knowledge of the relationship between Main Turbine Generator trip and the following systems and components:

- Recirculation System

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295005AK2.03	
Importance Rating	3.5	------

Proposed Question: **# 41**

Unit 1 is operating at 100% RTP.

Given the condition above and pertaining to End-of-Cycle Recirculation Pump Trips (EOC-RPT), the **NORMAL** plant configuration of 1-HS-099-5A-S14A and 1-HS-099-5A-S14B, RECIRC PUMP TRIP (RPT) LOGIC A/B TEST SWITCHES in the Aux Instrument Room, are in the **(1)** position.

IF the above switches are in the NORMAL position, **THEN** a Recirculation Pump trip will occur as a result of the **(2)** Valves being less than 90% open.

A. (1) NORMAL
   (2) Turbine Stop

B. (1) NORMAL
   (2) Turbine Control

C. (1) OUT OF SERVICE
   (2) Turbine Stop

D. (1) OUT OF SERVICE
   (2) Turbine Control

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate confuses the EOC-RPT NORMAL plant configuration as the NORMAL position of the RPT LOGIC A/B TEST SWITCHES, however this is not the case at Browns Ferry. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible if the RPT LOGIC A/B TEST SWITCHES are in the NORMAL position, then a Recirc Pump Trip (RPT) occurs when Emergency Trips Supply Oil Pressure is less than 850 psig. This is a Turbine Control Valve Fast Closure (load reject) which also results in a Reactor SCRAM signal.

C **CORRECT:** (*See attached*) In accordance with 2-OI-68, Reactor Recirculation System, the End-of-Cycle Recirculation Pump Trips (EOC-RPT) logic are bypassed (disabled) from the Aux Instrument Room and may remain out of service for any length of time. This is performed by placing the both of the RPT LOGIC A/B TEST SWITCHES in the OUT OF SERVICE position. At Browns Ferry, EOC-RPT is normally bypassed on all three Units due to better fuel designs and the ability to better calculate thermal limits. This allows operation with greater margins from limits, without EOC-RPT, thermal limits would not be exceeded on a Main Turbine Trip or load reject. Adding to the complexity, ATWS-RPT is a separate Recirc System logic that is normally active and not bypassed.
For second part, if the RPT LOGIC A/B TEST SWITCHES are in the NORMAL position, then a Recirc Pump Trip (RPT) will occur when Main Turbine First Stage Pressure is greater than 121 psig should the Turbine Stop Valves be less than 90% open as indicated by logic resulting in a Reactor SCRAM signal.

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 relationship between Main Turbine Generator trip and Recirculation pertaining to End-of-Cycle Recirculation Pump Trips (EOC-RPT) logic. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-OI-68, Rev. 164 (Attach if not previously provided)
OPL171.007, Rev. 31U2

Proposed references to be provided to applicants during examination: **NONE**

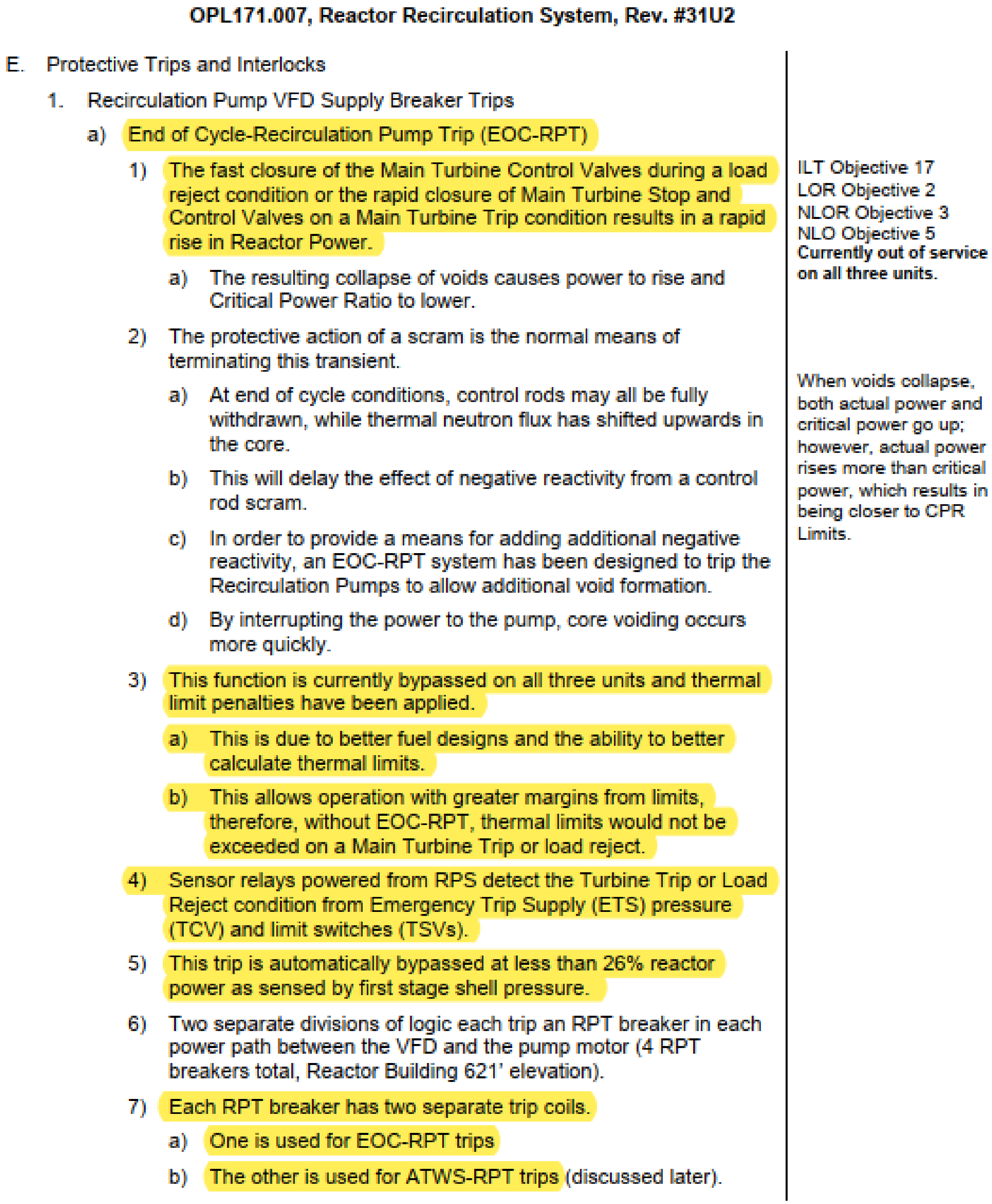
Learning Objective: OPL171.007 Obj. 17 (As available)

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 **X**
55.43

Excerpts from OPL171.007 Lesson Plan: (lists the difference between EOC-RPT and ATWS-RPT)



OPL171.007, Reactor Recirculation System, Rev. #31U2

E. Protective Trips and Interlocks

1. Recirculation Pump VFD Supply Breaker Trips
   - a) End of Cycle-Recirculation Pump Trip (EOC-RPT)
     - 1) The fast closure of the Main Turbine Control Valves during a load reject condition or the rapid closure of Main Turbine Stop and Control Valves on a Main Turbine Trip condition results in a rapid rise in Reactor Power.
       - a) The resulting collapse of voids causes power to rise and Critical Power Ratio to lower.

ILT Objective 17
LOR Objective 2
NLOR Objective 3
NLO Objective 5
Currently out of service on all three units.

     - 2) The protective action of a scram is the normal means of terminating this transient.
       - a) At end of cycle conditions, control rods may all be fully withdrawn, while thermal neutron flux has shifted upwards in the core.
       - b) This will delay the effect of negative reactivity from a control rod scram.
       - c) In order to provide a means for adding additional negative reactivity, an EOC-RPT system has been designed to trip the Recirculation Pumps to allow additional void formation.
       - d) By interrupting the power to the pump, core voiding occurs more quickly.

When voids collapse, both actual power and critical power go up; however, actual power rises more than critical power, which results in being closer to CPR Limits.

     - 3) This function is currently bypassed on all three units and thermal limit penalties have been applied.
       - a) This is due to better fuel designs and the ability to better calculate thermal limits.
       - b) This allows operation with greater margins from limits, therefore, without EOC-RPT, thermal limits would not be exceeded on a Main Turbine Trip or load reject.
     - 4) Sensor relays powered from RPS detect the Turbine Trip or Load Reject condition from Emergency Trip Supply (ETS) pressure (TCV) and limit switches (TSVs).
     - 5) This trip is automatically bypassed at less than 26% reactor power as sensed by first stage shell pressure.
     - 6) Two separate divisions of logic each trip an RPT breaker in each power path between the VFD and the pump motor (4 RPT breakers total, Reactor Building 621' elevation).
     - 7) Each RPT breaker has two separate trip coils.
       - a) One is used for EOC-RPT trips
       - b) The other is used for ATWS-RPT trips (discussed later).

OPL171.007, Reactor Recirculation System, Rev. #31U2

c) Two divisions of trip logic comprise the EOC-RPT logic scheme.

  (1) Either division will trip both Recirculation Pumps.

d) When first stage pressure is greater than 121 psig (closing contact K9A-D), should the Turbine Stop Valves become less than 90% open (closing contacts K10J-M), **OR** ETS oil pressure lower below 850 psig (Turbine Control Valve fast closure, relays K8E, K8F, K8H, K8J), the trip coils would energize, tripping the RPT breakers.

8) The RPT Breakers are normally tripped only automatically.

  a) A mechanical trip would have to be initiated locally at the breaker.

  b) The RPT breakers can be closed from Panel 9-4 following a trip.

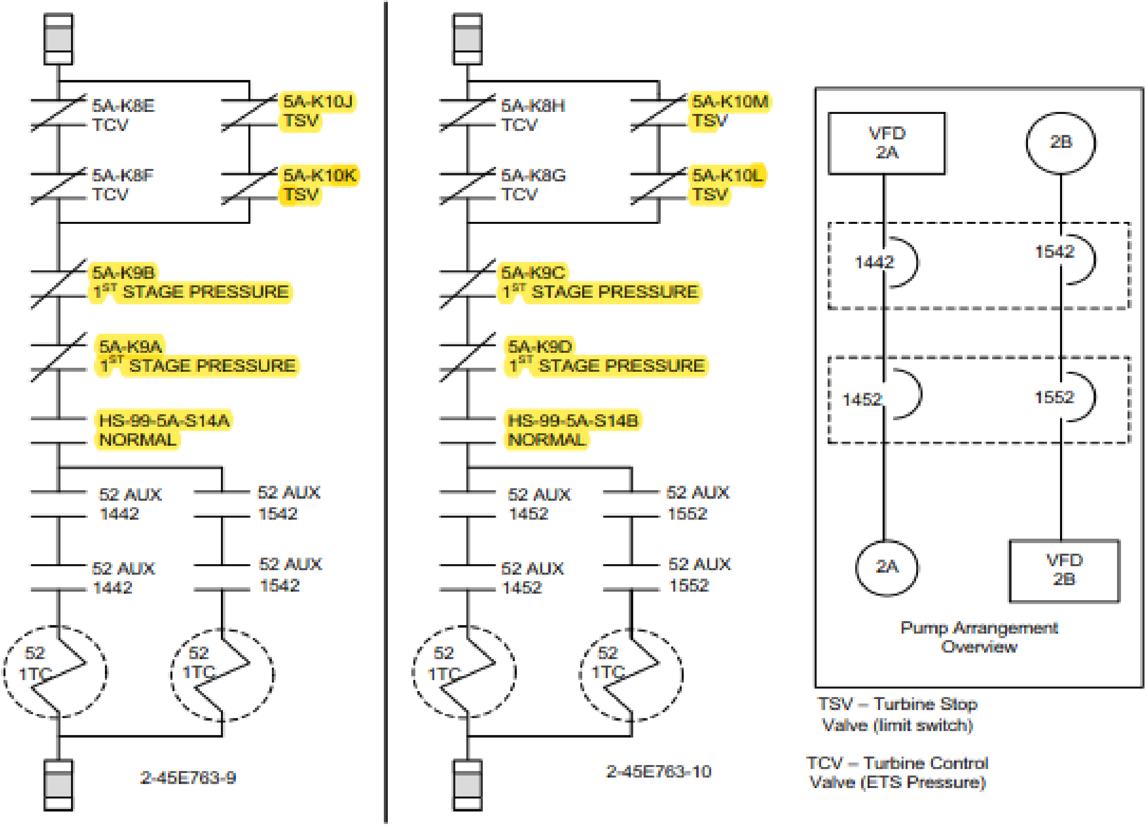
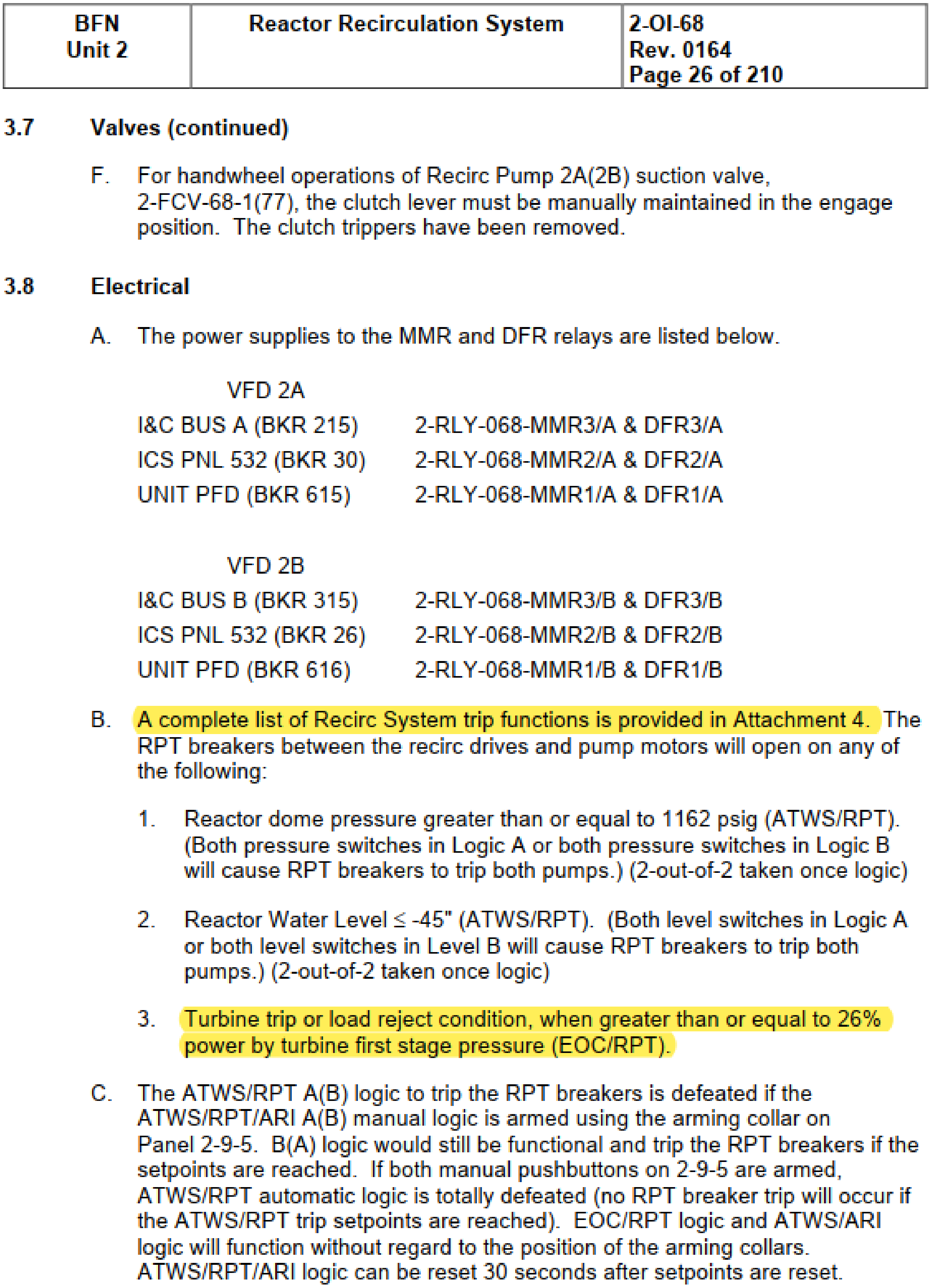


Figure-10 - EOC-RPT Logic

Excerpts from 2-OI-68:



BFN **Unit 2**	**Reactor Recirculation System**	**2-OI-68** **Rev. 0164** **Page 26 of 210**

3.7 Valves (continued)

F. For handwheel operations of Recirc Pump 2A(2B) suction valve, 2-FCV-68-1(77), the clutch lever must be manually maintained in the engage position. The clutch trippers have been removed.

3.8 Electrical

A. The power supplies to the MMR and DFR relays are listed below.

VFD 2A	
I&C BUS A (BKR 215)	2-RLY-068-MMR3/A & DFR3/A
ICS PNL 532 (BKR 30)	2-RLY-068-MMR2/A & DFR2/A
UNIT PFD (BKR 615)	2-RLY-068-MMR1/A & DFR1/A

VFD 2B	
I&C BUS B (BKR 315)	2-RLY-068-MMR3/B & DFR3/B
ICS PNL 532 (BKR 26)	2-RLY-068-MMR2/B & DFR2/B
UNIT PFD (BKR 616)	2-RLY-068-MMR1/B & DFR1/B

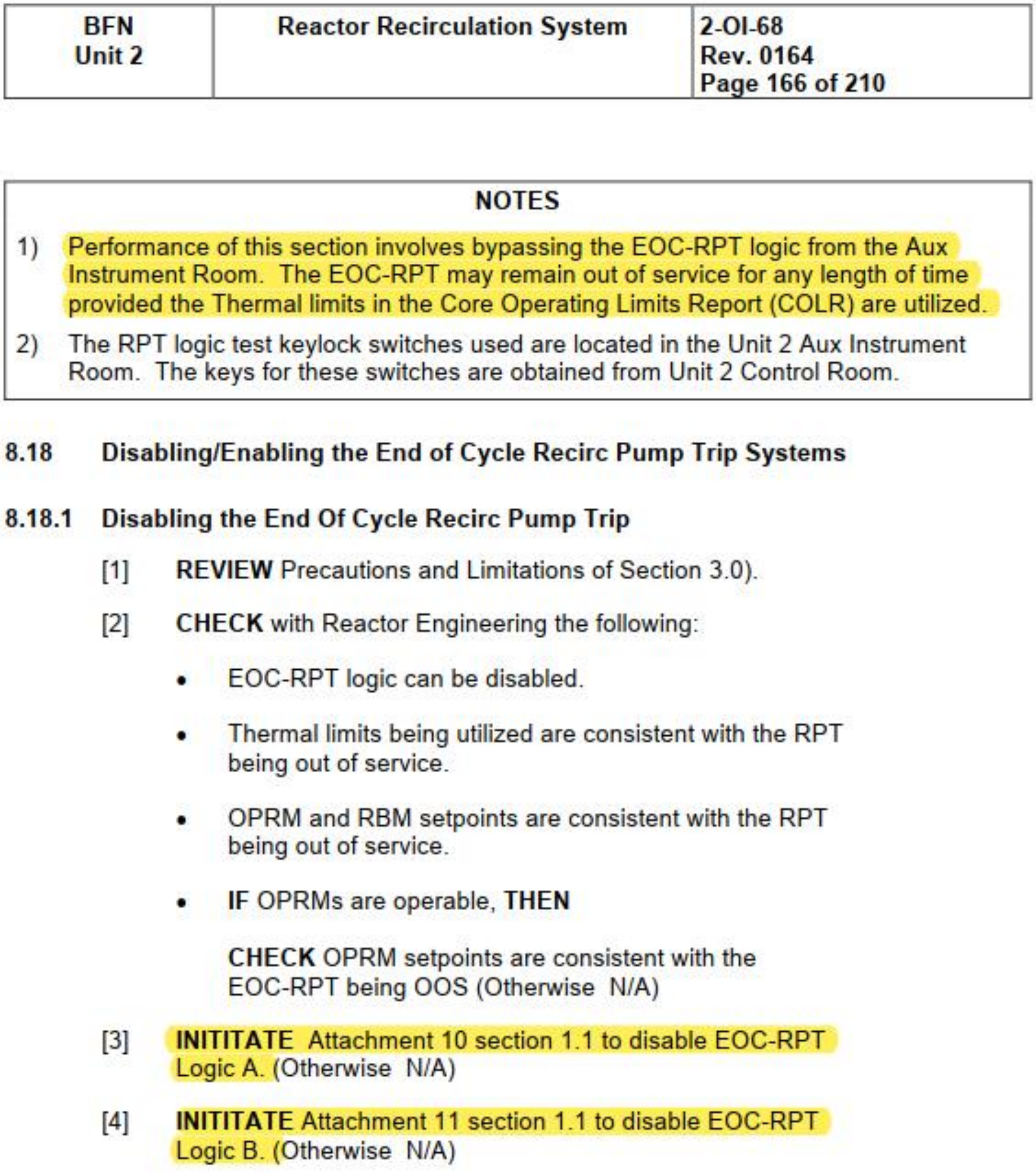
B. A complete list of Recirc System trip functions is provided in Attachment 4. The RPT breakers between the recirc drives and pump motors will open on any of the following:

1. Reactor dome pressure greater than or equal to 1162 psig (ATWS/RPT). (Both pressure switches in Logic A or both pressure switches in Logic B will cause RPT breakers to trip both pumps.) (2-out-of-2 taken once logic)

2. Reactor Water Level ≤ -45" (ATWS/RPT). (Both level switches in Logic A or both level switches in Level B will cause RPT breakers to trip both pumps.) (2-out-of-2 taken once logic)

3. Turbine trip or load reject condition, when greater than or equal to 26% power by turbine first stage pressure (EOC/RPT).

C. The ATWS/RPT A(B) logic to trip the RPT breakers is defeated if the ATWS/RPT/ARI A(B) manual logic is armed using the arming collar on Panel 2-9-5. B(A) logic would still be functional and trip the RPT breakers if the setpoints are reached. If both manual pushbuttons on 2-9-5 are armed, ATWS/RPT automatic logic is totally defeated (no RPT breaker trip will occur if the ATWS/RPT trip setpoints are reached). EOC/RPT logic and ATWS/ARI logic will function without regard to the position of the arming collars. ATWS/RPT/ARI logic can be reset 30 seconds after setpoints are reset.



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NOTES

1) Performance of this section involves bypassing the EOC-RPT logic from the Aux Instrument Room. The EOC-RPT may remain out of service for any length of time provided the Thermal limits in the Core Operating Limits Report (COLR) are utilized.

2) The RPT logic test keylock switches used are located in the Unit 2 Aux Instrument Room. The keys for these switches are obtained from Unit 2 Control Room.

8.18 Disabling/Enabling the End of Cycle Recirc Pump Trip Systems

8.18.1 Disabling the End Of Cycle Recirc Pump Trip

[1] **REVIEW** Precautions and Limitations of Section 3.0).

[2] **CHECK** with Reactor Engineering the following:

- EOC-RPT logic can be disabled.
- Thermal limits being utilized are consistent with the RPT being out of service.
- OPRM and RBM setpoints are consistent with the RPT being out of service.
- **IF** OPRMs are operable, **THEN**

  CHECK OPRM setpoints are consistent with the EOC-RPT being OOS (Otherwise N/A)

[3] **INITITATE** Attachment 10 section 1.1 to disable EOC-RPT Logic A. (Otherwise N/A)

[4] **INITITATE** Attachment 11 section 1.1 to disable EOC-RPT Logic B. (Otherwise N/A)

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Attachment 4
(Page 4 of 4)

Recirc System Trips/Interlocks

1.0 RECIRC SYSTEM TRIPS/INTERLOCKS (continued)

G. RPT Breakers Open (Trips the feeder breakers)

1. EOC-RPT
   a. Turbine Stop Valve Closure, Power greater than 26% by Turbine 1st Stage Pressure
   b. Turbine Control Valve Fast Closure, Power greater than 26% by Turbine 1st Stage
2. ATWS-RPT
   a. RPV Pressure greater than 1162 psig (2 of 2 pressure switches in Logic A or B)
   b. RPV Water Level < -45 inches (2 of 2 level switches in Logic A or B)
3. Local Trip

H. Feeder Breaker Shut and No Voltage 0n Recirc Board

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Attachment 10
(Page 1 of 3)

Disabling/Enabling the RPT LOGIC 2A EOC Recirc Pump Trip

1.0 DISABLING/ENABLING 2A RECIRC PUMP RPT

1.1 Disabling the End Of Cycle Recirc Pump Trip

[1] **ENSURE** Section 8.18.1 has been performed prior to disabling the 2A End of Cycle Recirc Pump Trip system.

Initials Date

[2] **PLACE** RPT LOGIC A TEST SWITCH (keylock switch), in the RPT LOGIC A OUT OF SERVICE position on Panel 2-9-15.

Initials Date

[3] **CHECK** EOC-RPT SYS A ACTIVE, 2-XA-68-74, 2-XA-55-4A Window 19, is reset on Panel 2-9-6.

Initials Date

[4] **RECORD** in Narrative Log that EOC-RPT Logic A has been disabled.

Initials Date

[5] **IF** EOC-RPT SYS A ACTIVE, 2-XA-68-74, 2-XA-55-4A Window 19 will be disabled, **THEN**

PERFORM Attachment 12, Section 1.1.(Otherwise N/A)

Initials Date

[6] **PLACE** this Attachment in 2-OI-68 System Status Folder.

Initials Date

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Attachment 10
(Page 2 of 3)
Disabling/Enabling the RPT LOGIC 2A EOC Recirc Pump Trip

1.2 Enabling the End Of Cycle Recirc Pump Trip

[1] **ENSURE** Section 8.18.2 has been performed prior to enabling the 2A End of Cycle Recirc Pump Trip system.

Initials Date

[2] **IF** EOC-RPT SYS A ACTIVE, 2-XA-68-74, 2-XA-55-4A, Window 19 is disabled, **THEN**

PERFORM Attachment 12, Section 2.1 to enable annunciator 2-XA-55-4A Window 19. (Otherwise N/A)

Initials Date

[3] **CHECK** the EOC-RPT SYS A ACTIVE, 2-XA-68-74, 2-XA-55-4A, Window 19 is reset:

Initials Date

[4] **PLACE** RPT LOGIC A TEST SWITCH (keylock switch), in the NORMAL position on Panel 2-9-15.

Initials Date

[5] **RETURN** the keys for the RPT LOGIC A TEST SWITCH. keylock switches to the key cabinet in Unit 2 Control Room.

Initials Date

[6] **CHECK** EOC-RPT SYS A ACTIVE, 2-XA-68-74, 2-XA-55-4A Window 19 annunciator is in ALARM.

Initials Date

Examination Outline Cross-reference:	Level	RO	SRO
295013 (APE 13 IRM) High Suppression Pool Temperature / 5	Tier #	1	------
AA2.01 (10CFR 55.41.10)	Group #	2	------
Ability to determine and/or interpret the following as they apply to High Suppression Pool Water Temperature: • Suppression Pool Temperature	K/A #	295013AA2.01	
	Importance Rating	4.3	------

Proposed Question: **# 42**

Unit 1 is operating at 95% RTP with **NO** equipment out of service.

While HPCI testing is in progress, 1-EOI-2, Primary Containment Control entry is required upon first reaching ___**(1)**___ Suppression Pool Temperature.

When 1-EOI-2 is entered on high Suppression Pool Temperature, the Operator is required to place ___**(2)**___ loop(s) of Suppression Pool Cooling in service.

A. (1) 95 °F
(2) **ONLY** 1

B. (1) 95 °F
(2) **ALL** available

C. (1) 105 °F
(2) **ONLY** 1

D. (1) 105 °F
(2) **ALL** available

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that 1-EOI-2 states to monitor and control Suppression Pool Temperature below 95 °F using available Suppression Pool Cooling. The candidate could easily believe this to believe that only 1 loop of RHR is currently required for Suppression Pool Cooling.

B **CORRECT**: *(See attached)* In accordance with 1-EOI-2, Primary Containment Control, the entry condition for high Suppression Pool Temperature is 95 °F. For second part, 1-EOI-2 states to operate all available Suppression Pool Cooling using only RHR Pumps not required for continuous injection. Given that Reactor Power is 95 %, RHR is not currently needed for injection.

C INCORRECT: First part is incorrect but plausible in that 105 °F is the Tech Spec 3.6.2.1 number for Suppression Pool Temperature at which testing that adds heat to the Suppression Pool must be suspended. 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 determine and interpret Suppression Pool Temperature as it applies to EOI entry conditions and the required actions to mitigate the condition. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to emergency plant conditions.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): U1 Tech Spec 3.6.2.1, Amend. 234 (Attach if not previously provided)
1-EOI-2, Rev. 8

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.203 Obj. 2 (As available)

Question Source:	Bank #	
	Modified Bank #	
	New	**X**
Question History:	Last NRC Exam	

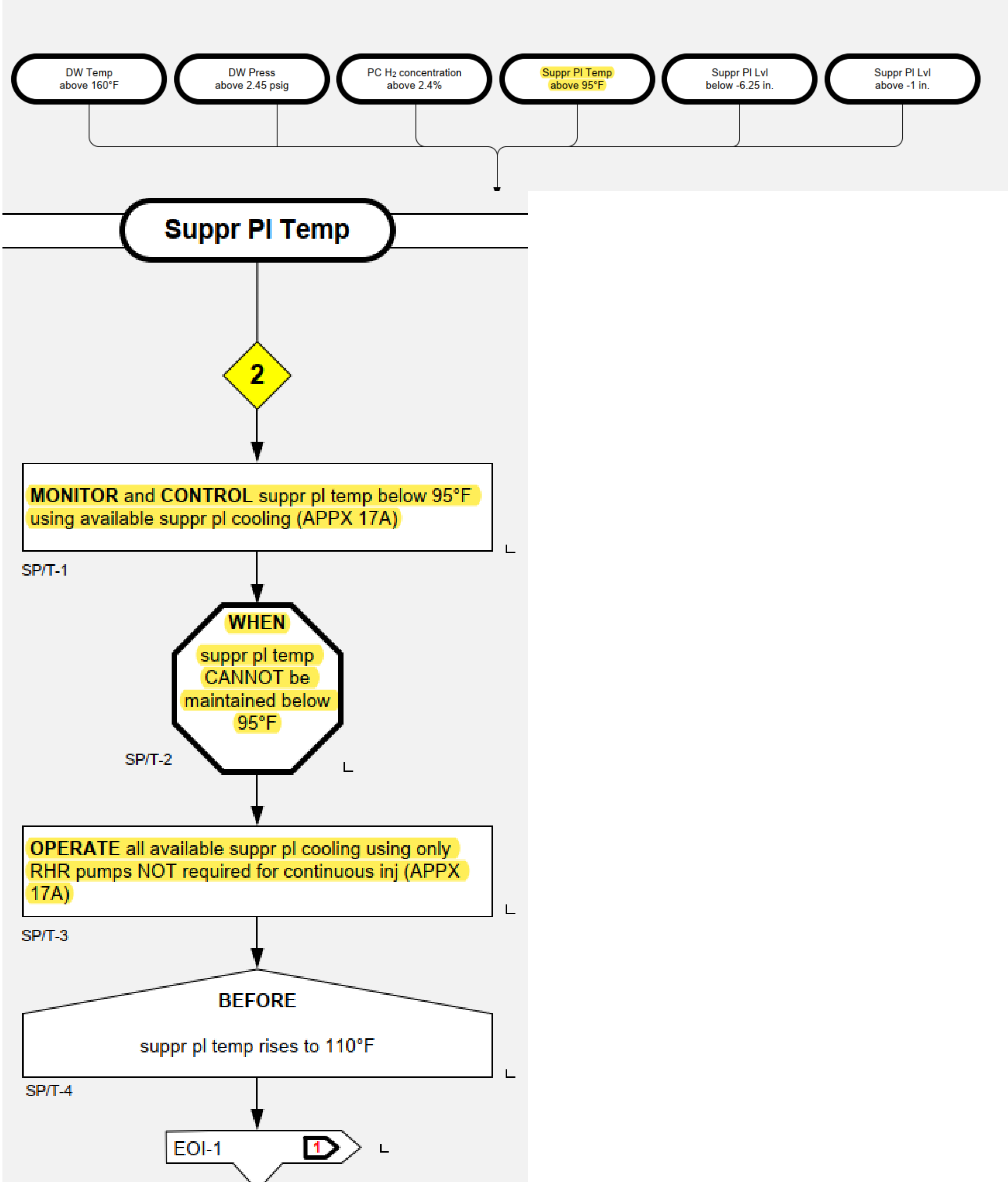
(Note changes or attach parent)

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

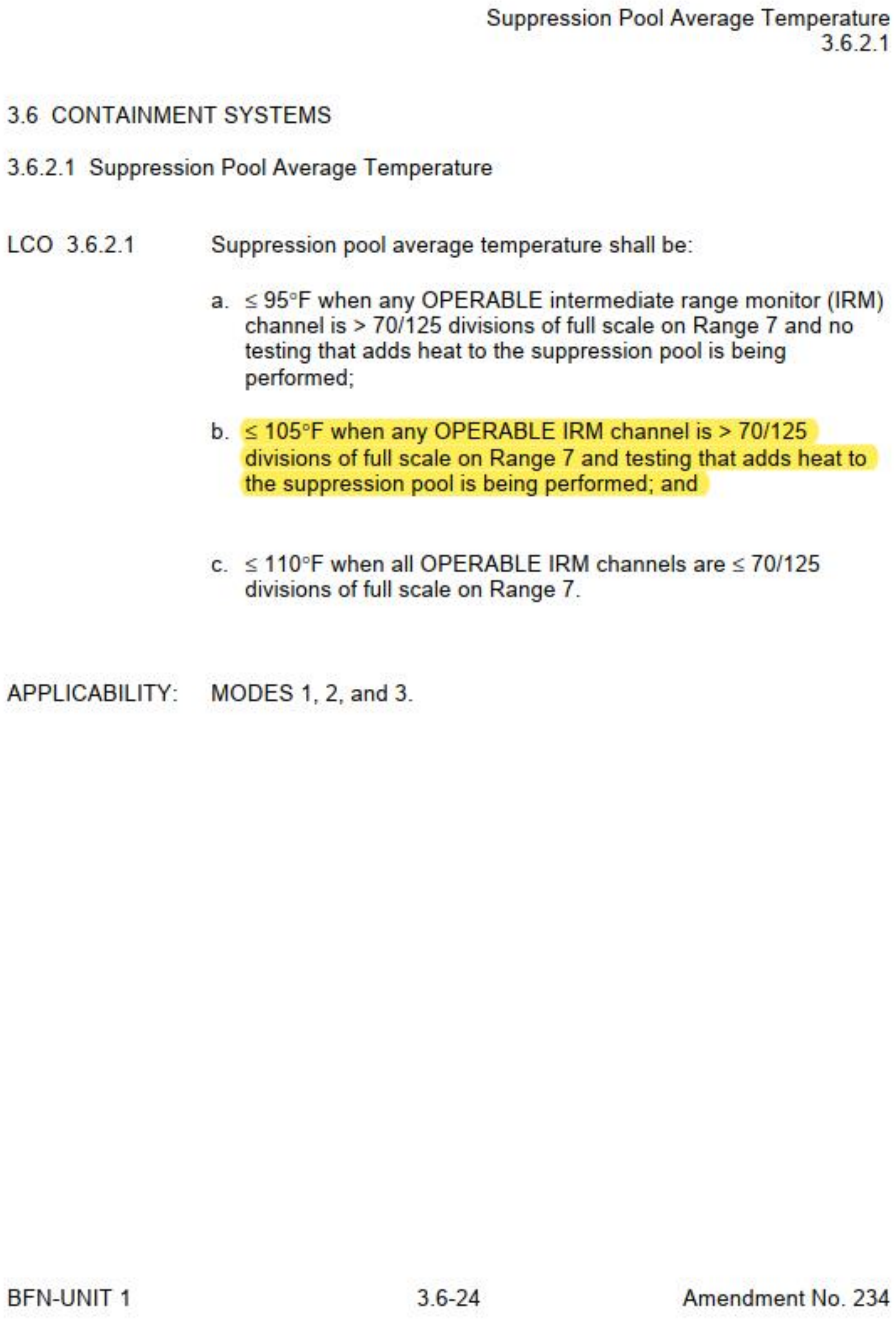
10 CFR Part 55 Content:	55.41	**X**
	55.43	

Comments:

Excerpts from 1-EOI-2:



Excerpt from U1 Tech Spec 3.6.2.1:



Suppression Pool Average Temperature
3.6.2.1

3.6 CONTAINMENT SYSTEMS

3.6.2.1 Suppression Pool Average Temperature

LCO 3.6.2.1 Suppression pool average temperature shall be:

a. ≤ 95°F when any OPERABLE intermediate range monitor (IRM) channel is > 70/125 divisions of full scale on Range 7 and no testing that adds heat to the suppression pool is being performed;

b. ≤ 105°F when any OPERABLE IRM channel is > 70/125 divisions of full scale on Range 7 and testing that adds heat to the suppression pool is being performed; and

c. ≤ 110°F when all OPERABLE IRM channels are ≤ 70/125 divisions of full scale on Range 7.

APPLICABILITY: MODES 1, 2, and 3.

BFN-UNIT 1 3.6-24 Amendment No. 234

Examination Outline Cross-reference:

215004 (SF 7 SRMS) Source-Range Monitor

A2.02 (10CFR 55.41.5)

Ability to (a) predict the impacts of the following on the Source Range Monitor System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- SRMS inoperable condition

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	215004A2.02	
Importance Rating	3.4	------

Proposed Question: **# 43**

Unit 3 is in MODE 2 with the following plant conditions:

- SRM ‘B’ is bypassed due to excessive noise and spiking
- All IRMs are on Range 3

Subsequently, SRM ‘A’ fails **HIGH** and has been declared INOPERABLE.

Given the conditions above and in accordance with 3-OI-92, Source Range Monitors, a Control Rod Block ___**(1)**___ occur and more than one SRM ___**(2)**___ be bypassed by a Control Room Operator.

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: First part is correct *(See B).* Second part is incorrect but plausible in that 4 SRM channels exist, A, B, C, D and IRMs have 2 bypass joysticks allowing for 2 IRMs to be bypassed at a time.

B **CORRECT**: *(See attached)* In accordance with 3-OI-92, Source Range Monitors, a Control Rod Block will occur given SRM ‘A’ fails HIGH unless IRMs are on Range 8 (or higher) or REACTOR MODE SWITCH is in RUN. The MODE SWITCH is in STARTUP/HOT STANDBY since the stem states MODE 2. For second part, only one SRM channel can be bypassed at a time.

C INCORRECT: First part is incorrect but plausible in that the candidate could confuse the SRM channel and trip logic associated with a Control Rod Block and/or a full Reactor Protection System SCRAM. Second part is incorrect but plausible *(See A).*

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

RO Level Justification: Tests the candidate's ability to predict the impacts of INOPERABLE Source Range Monitors and the associated required procedural actions. This question is rated as Memory due to the requirement to strictly recall facts related to operation of the SRMs.

Technical Reference(s):	3-OI-92, Rev. 17	(Attach if not previously provided)
	OPL171.019, Rev. 15	

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.019 Obj. 4a (As available)

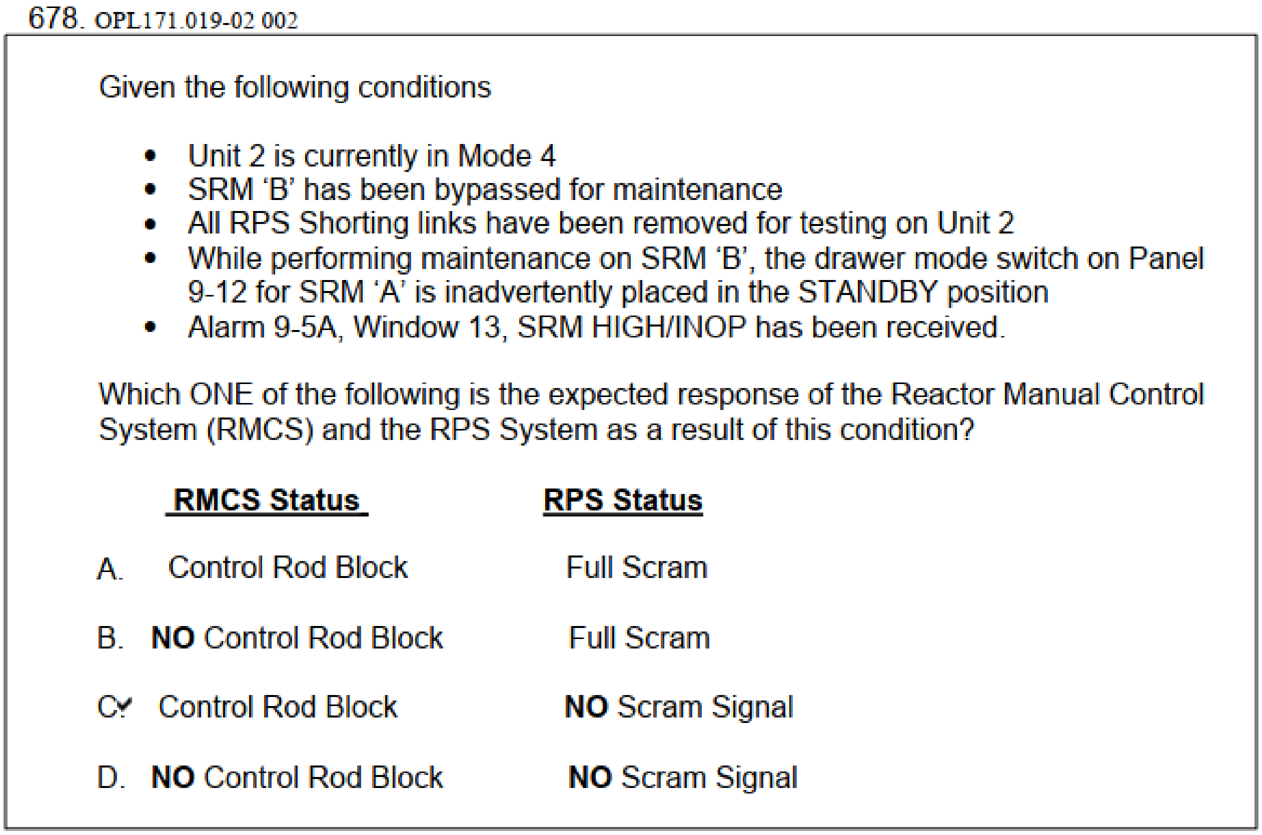
Question Source:	Bank #		
	Modified Bank #	OPL171.019-02 002 #678	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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:



678. OPL171.019-02 002

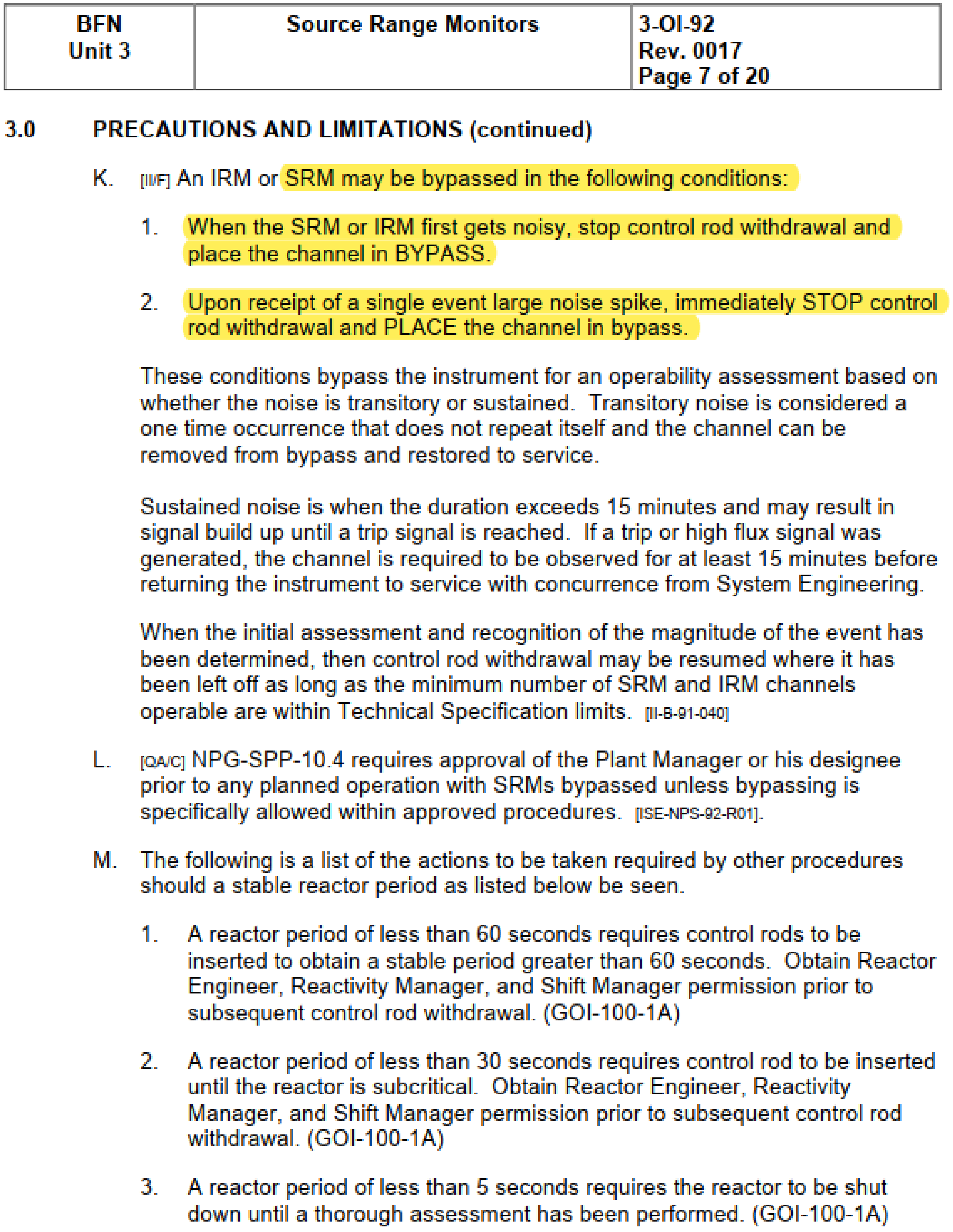
Given the following conditions

- Unit 2 is currently in Mode 4
- SRM 'B' has been bypassed for maintenance
- All RPS Shorting links have been removed for testing on Unit 2
- While performing maintenance on SRM 'B', the drawer mode switch on Panel 9-12 for SRM 'A' is inadvertently placed in the STANDBY position
- Alarm 9-5A, Window 13, SRM HIGH/INOP has been received.

Which ONE of the following is the expected response of the Reactor Manual Control System (RMCS) and the RPS System as a result of this condition?

	RMCS Status	**RPS Status**
A.	Control Rod Block	Full Scram
B.	**NO** Control Rod Block	Full Scram
C✓	Control Rod Block	**NO** Scram Signal
D.	**NO** Control Rod Block	**NO** Scram Signal

Excerpts from 3-OI-92:



BFN **Unit 3**	**Source Range Monitors**	**3-OI-92** **Rev. 0017** **Page 7 of 20**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

K. [II/F] An IRM or SRM may be bypassed in the following conditions:

1. When the SRM or IRM first gets noisy, stop control rod withdrawal and place the channel in BYPASS.
2. Upon receipt of a single event large noise spike, immediately STOP control rod withdrawal and PLACE the channel in bypass.

These conditions bypass the instrument for an operability assessment based on whether the noise is transitory or sustained. Transitory noise is considered a one time occurrence that does not repeat itself and the channel can be removed from bypass and restored to service.

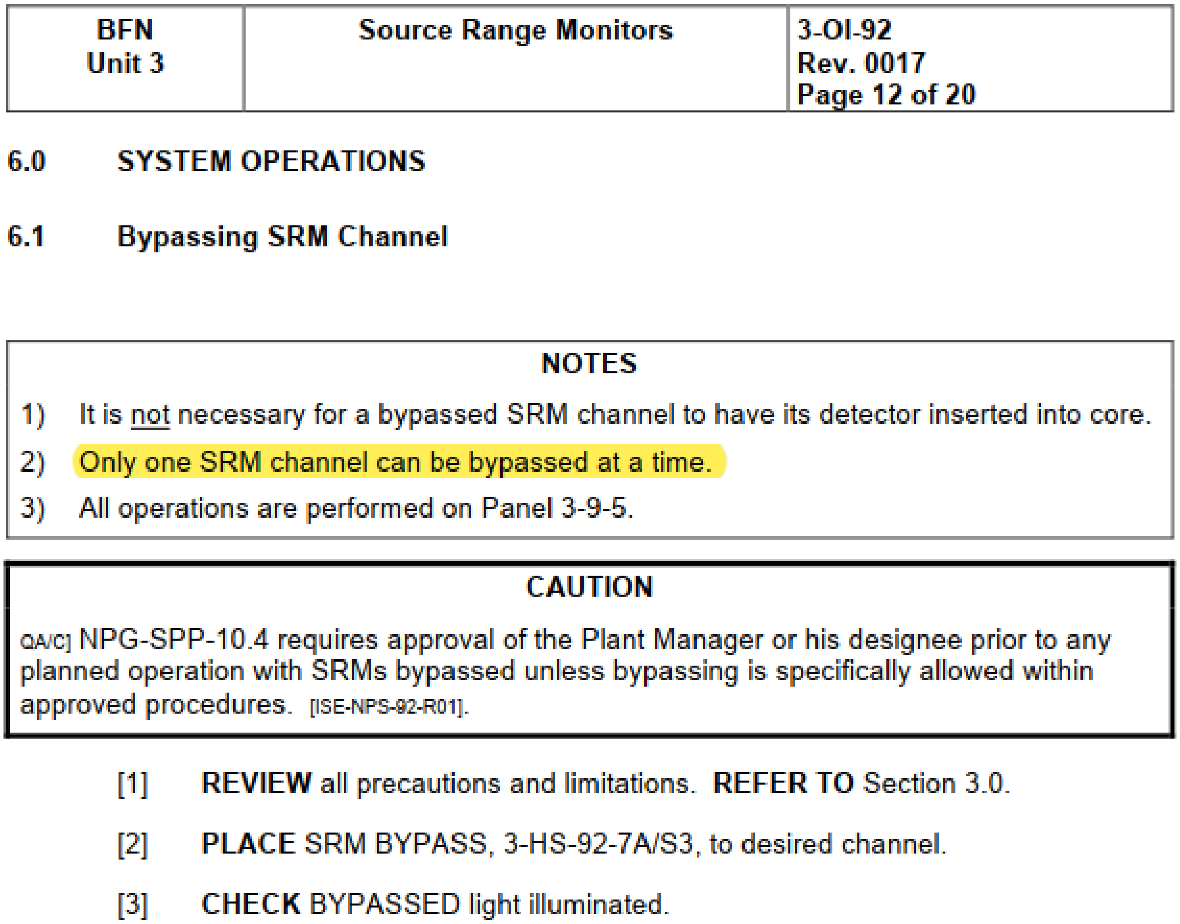
Sustained noise is when the duration exceeds 15 minutes and may result in signal build up until a trip signal is reached. If a trip or high flux signal was generated, the channel is required to be observed for at least 15 minutes before returning the instrument to service with concurrence from System Engineering.

When the initial assessment and recognition of the magnitude of the event has been determined, then control rod withdrawal may be resumed where it has been left off as long as the minimum number of SRM and IRM channels operable are within Technical Specification limits. [II-B-91-040]

L. [QA/C] NPG-SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with SRMs bypassed unless bypassing is specifically allowed within approved procedures. [ISE-NPS-92-R01].

M. The following is a list of the actions to be taken required by other procedures should a stable reactor period as listed below be seen.

1. A reactor period of less than 60 seconds requires control rods to be inserted to obtain a stable period greater than 60 seconds. Obtain Reactor Engineer, Reactivity Manager, and Shift Manager permission prior to subsequent control rod withdrawal. (GOI-100-1A)
2. A reactor period of less than 30 seconds requires control rod to be inserted until the reactor is subcritical. Obtain Reactor Engineer, Reactivity Manager, and Shift Manager permission prior to subsequent control rod withdrawal. (GOI-100-1A)
3. A reactor period of less than 5 seconds requires the reactor to be shut down until a thorough assessment has been performed. (GOI-100-1A)



BFN Unit 3	**Source Range Monitors**	**3-OI-92 Rev. 0017 Page 12 of 20**

6.0 SYSTEM OPERATIONS

6.1 Bypassing SRM Channel

NOTES

1) It is not necessary for a bypassed SRM channel to have its detector inserted into core.

2) Only one SRM channel can be bypassed at a time.

3) All operations are performed on Panel 3-9-5.

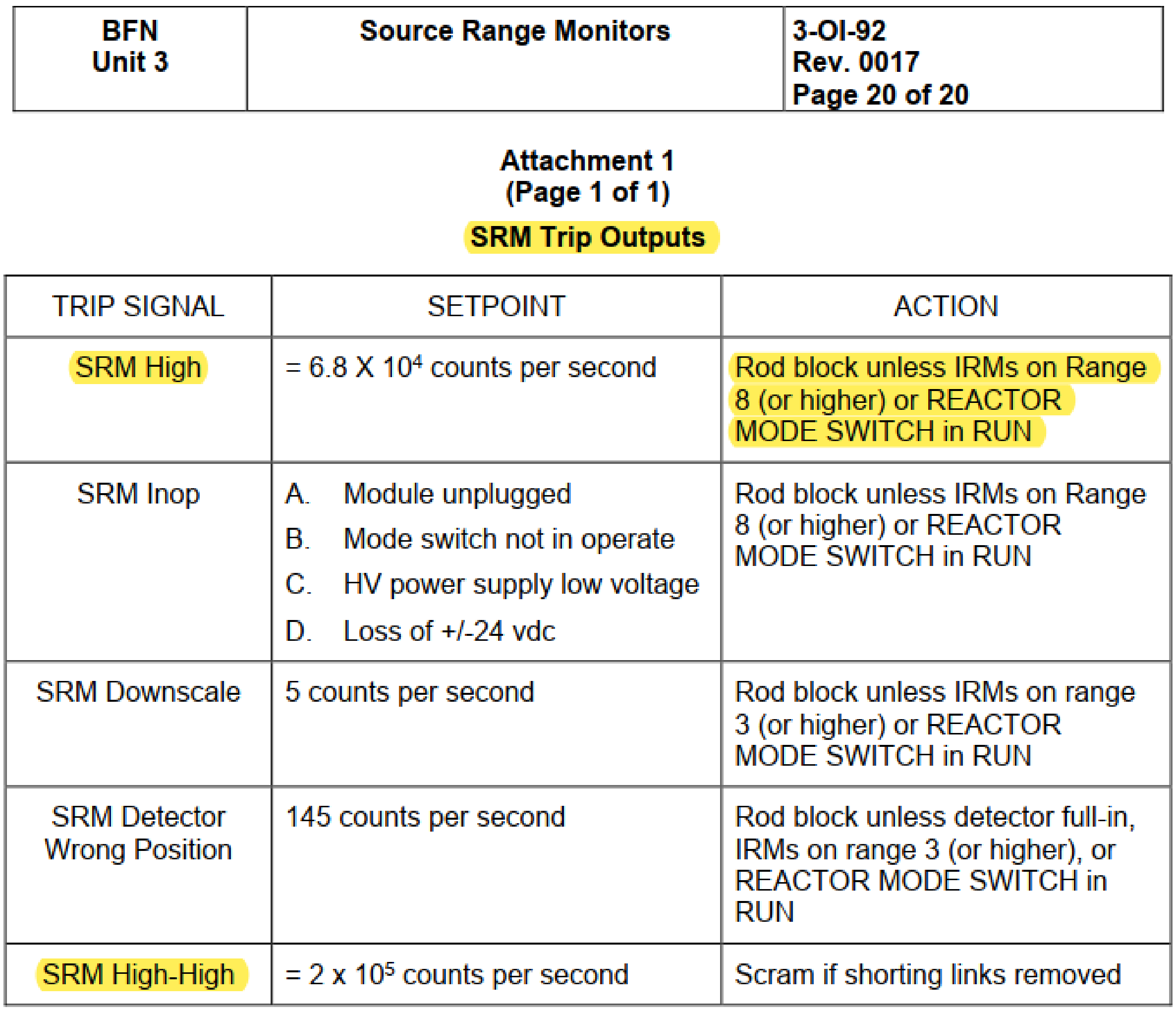
CAUTION

QA/C] NPG-SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with SRMs bypassed unless bypassing is specifically allowed within approved procedures. [ISE-NPS-92-R01].

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

[2] **PLACE** SRM BYPASS, 3-HS-92-7A/S3, to desired channel.

[3] **CHECK** BYPASSED light illuminated.



BFN **Unit 3**	**Source Range Monitors**	**3-OI-92** **Rev. 0017** **Page 20 of 20**

Attachment 1
(Page 1 of 1)

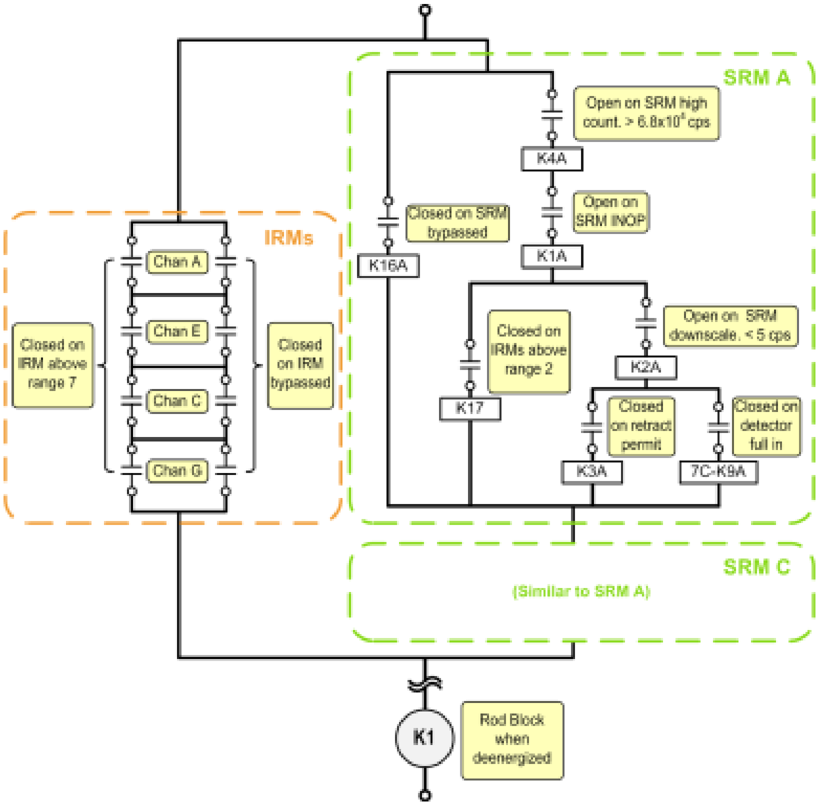
SRM Trip Outputs

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

Excerpts from OPL171.019 Lesson Plan: Supports Distractors C(1), D(1)

Simplified SRM Rod Block Logic

A more complete version of Rod Block Logic, including the mode switch, refuel related contacts, IRMs and PRNMs, can be found in the back of this Student Guide.



Trips

SRM trips (used during refueling operations) require the removal of shorting links in the Reactor Protection System (RPS) logic to activate the associated scram. The Reactor Protection System in conjunction with the Neutron Monitoring System (SRM and IRM) has non-coincident (only one detector must trip) logic if all eight shorting links are removed. If the yellow, green, and red shorting links (six total) are removed, the SRM High-High trips will be placed in a coincidence (one-out-of-two taken twice) logic. These RPS logic modifications are controlled by plant procedures.

Examination Outline Cross-reference:

259002 (SF2 RWLCS) Reactor Water Level Control

A2.02 (10CFR55.41.5)

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

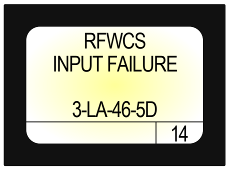
- Loss of any number of Reactor feedwater flow inputs

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	259002A2.02	
Importance Rating	3.8	------

Proposed Question: **# 44**

Unit 3 is operating at 100% RTP with the following conditions:

- RFWCS INPUT FAILURE (3-9-6C, Window 14) alarms
- The OATC reports 3-FI-3-78A, FW LINE A FLOW on Panel 3-9-5 indicates UPSCALE



Given the conditions above, 3-FI-3-78A, FW LINE A FLOW **___(1)____** automatically bypass.

Once FW LINE A FLOW is bypassed, Feedwater Level Control will be in **___(2)____** element control.

A. (1) will
(2) three

B. (1) will
(2) single

C. (1) will NOT
(2) three

D. (1) will NOT
(2) single

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 3-OI-3, Reactor Feedwater System, Attachment 8, 3-FI-3-78A, FEEDWATER LINE A FLOW will be declared invalid and automatically bypassed when it deviates by more than 0.8 Mlbm/hr from the total of the Reactor Feedwater Pump (RFP) discharge flows. For second part, in accordance with 3-OI-3, only one valid total Feedwater Flow is required for three element control as long as there are three valid RFP Flow signals.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that in accordance with 3-OI-3, several combinations of bypassed/failed Feedwater Level Control System (FWLCS) instruments will cause Reactor Water Level Control to switch to single element.

C INCORRECT: First part is incorrect but plausible in that 3-OI-3 contains guidance on manually bypassing FWLCS Instrumentation using push-buttons on Panel 3-9-5, and it is reasonable to assume that this is the only way to bypass faulty FWLCS Instruments. 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 method the bypass a faulty instrument and the effect that a loss or malfunction of one Feedwater Flow Instrument will have on the Feedwater Level Control 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): 3-OI-3, Rev. 114 (Attach if not previously provided)
3-ARP-9-6C, Rev. 29

Proposed references to be provided to applicants during examination: **RFWCS INPUT FAILURE (3-ARP-9-6C, Window 14)**

Learning Objective: OPL171. 012, Obj. 6 (As available)

Question Source:

Bank #		
Modified Bank #		(Note changes or attach parent)
New	**X**	

Question History:

Last NRC Exam	

Question Cognitive Level:

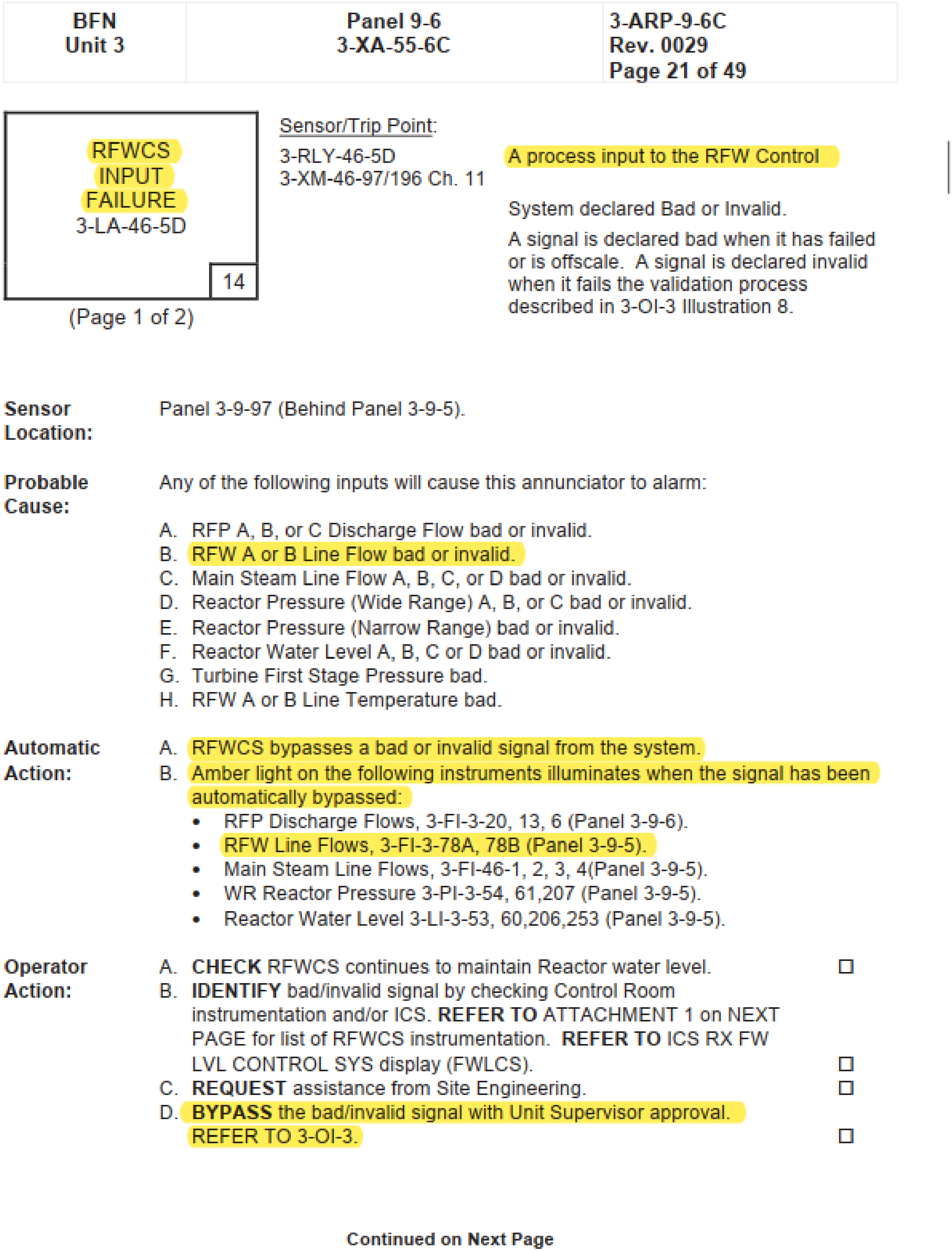
Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Excerpt from 3-ARP-9-6C:



BFN **Unit 3**	**Panel 9-6** **3-XA-55-6C**	**3-ARP-9-6C** **Rev. 0029** **Page 21 of 49**

RFWCS
INPUT
FAILURE
3-LA-46-5D
14

(Page 1 of 2)

Sensor/Trip Point:

3-RLY-46-5D
3-XM-46-97/196 Ch. 11

A process input to the RFW Control System declared Bad or Invalid.

A signal is declared bad when it has failed or is offscale. A signal is declared invalid when it fails the validation process described in 3-OI-3 Illustration 8.

Sensor Location: Panel 3-9-97 (Behind Panel 3-9-5).

Probable Cause: Any of the following inputs will cause this annunciator to alarm:

A. RFP A, B, or C Discharge Flow bad or invalid.
B. RFW A or B Line Flow bad or invalid.
C. Main Steam Line Flow A, B, C, or D bad or invalid.
D. Reactor Pressure (Wide Range) A, B, or C bad or invalid.
E. Reactor Pressure (Narrow Range) bad or invalid.
F. Reactor Water Level A, B, C or D bad or invalid.
G. Turbine First Stage Pressure bad.
H. RFW A or B Line Temperature bad.

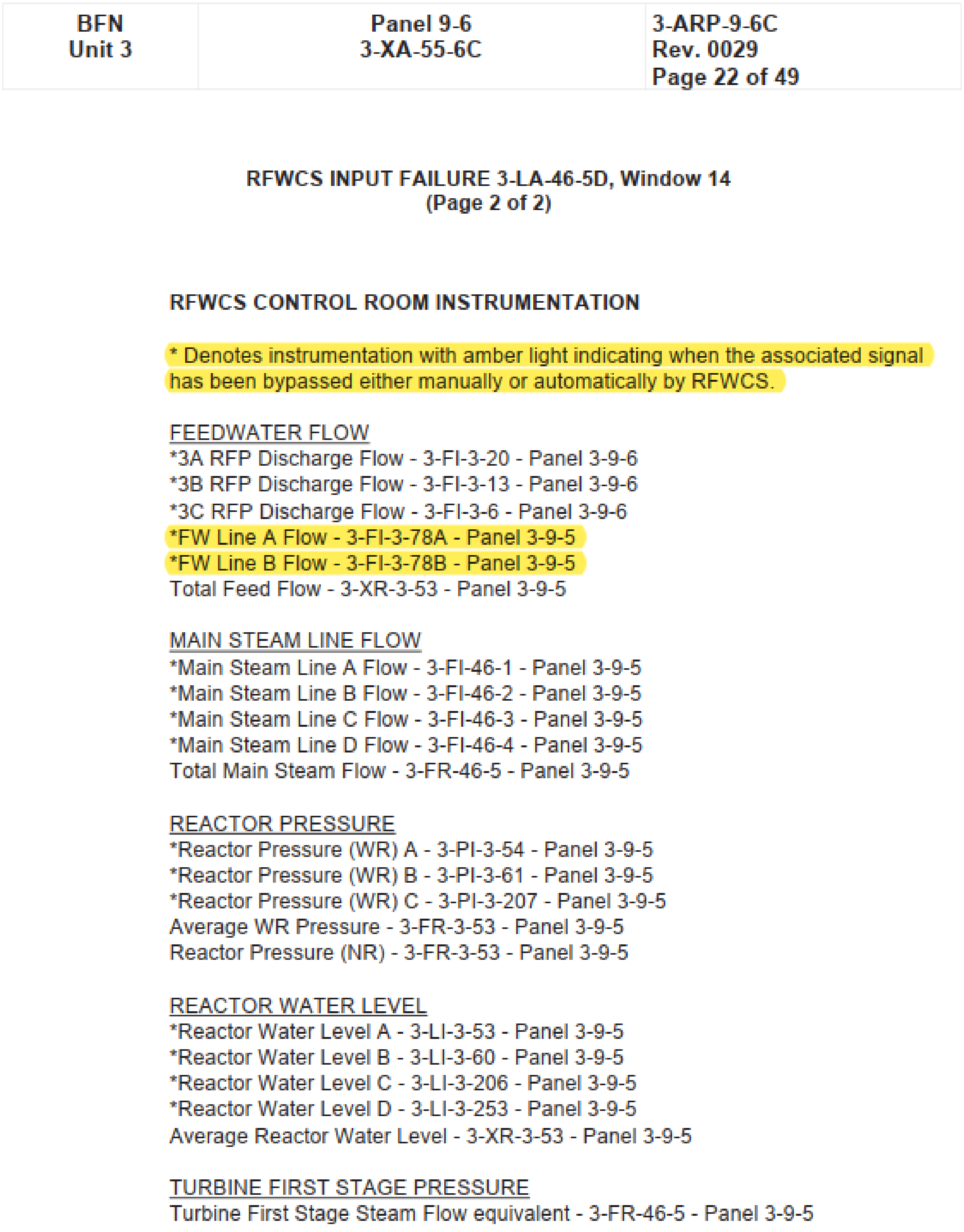
Automatic Action:

A. RFWCS bypasses a bad or invalid signal from the system.
B. Amber light on the following instruments illuminates when the signal has been automatically bypassed:
  - RFP Discharge Flows, 3-FI-3-20, 13, 6 (Panel 3-9-6).
  - RFW Line Flows, 3-FI-3-78A, 78B (Panel 3-9-5).
  - Main Steam Line Flows, 3-FI-46-1, 2, 3, 4(Panel 3-9-5).
  - WR Reactor Pressure 3-PI-3-54, 61,207 (Panel 3-9-5).
  - Reactor Water Level 3-LI-3-53, 60,206,253 (Panel 3-9-5).

Operator Action:

A. **CHECK** RFWCS continues to maintain Reactor water level. ☐
B. **IDENTIFY** bad/invalid signal by checking Control Room instrumentation and/or ICS. **REFER TO** ATTACHMENT 1 on NEXT PAGE for list of RFWCS instrumentation. **REFER TO** ICS RX FW LVL CONTROL SYS display (FWLCS). ☐
C. **REQUEST** assistance from Site Engineering. ☐
D. **BYPASS** the bad/invalid signal with Unit Supervisor approval. REFER TO 3-OI-3. ☐

Continued on Next Page



BFN Unit 3	Panel 9-6 3-XA-55-6C	3-ARP-9-6C Rev. 0029 Page 22 of 49

RFWCS INPUT FAILURE 3-LA-46-5D, Window 14
(Page 2 of 2)

RFWCS CONTROL ROOM INSTRUMENTATION

* Denotes instrumentation with amber light indicating when the associated signal has been bypassed either manually or automatically by RFWCS.

FEEDWATER FLOW
*3A RFP Discharge Flow - 3-FI-3-20 - Panel 3-9-6
*3B RFP Discharge Flow - 3-FI-3-13 - Panel 3-9-6
*3C RFP Discharge Flow - 3-FI-3-6 - Panel 3-9-6
*FW Line A Flow - 3-FI-3-78A - Panel 3-9-5
*FW Line B Flow - 3-FI-3-78B - Panel 3-9-5
Total Feed Flow - 3-XR-3-53 - Panel 3-9-5

MAIN STEAM LINE FLOW
*Main Steam Line A Flow - 3-FI-46-1 - Panel 3-9-5
*Main Steam Line B Flow - 3-FI-46-2 - Panel 3-9-5
*Main Steam Line C Flow - 3-FI-46-3 - Panel 3-9-5
*Main Steam Line D Flow - 3-FI-46-4 - Panel 3-9-5
Total Main Steam Flow - 3-FR-46-5 - Panel 3-9-5

REACTOR PRESSURE
*Reactor Pressure (WR) A - 3-PI-3-54 - Panel 3-9-5
*Reactor Pressure (WR) B - 3-PI-3-61 - Panel 3-9-5
*Reactor Pressure (WR) C - 3-PI-3-207 - Panel 3-9-5
Average WR Pressure - 3-FR-3-53 - Panel 3-9-5
Reactor Pressure (NR) - 3-FR-3-53 - Panel 3-9-5

REACTOR WATER LEVEL
*Reactor Water Level A - 3-LI-3-53 - Panel 3-9-5
*Reactor Water Level B - 3-LI-3-60 - Panel 3-9-5
*Reactor Water Level C - 3-LI-3-206 - Panel 3-9-5
*Reactor Water Level D - 3-LI-3-253 - Panel 3-9-5
Average Reactor Water Level - 3-XR-3-53 - Panel 3-9-5

TURBINE FIRST STAGE PRESSURE
Turbine First Stage Steam Flow equivalent - 3-FR-46-5 - Panel 3-9-5

References: 3-729E895-10 3-45E620-7

Excerpts from 3-OI-3:

BFN Unit 3	**Reactor Feedwater System**	**3-OI-3 Rev. 0114 Page 317 of 330**

Attachment 8
(Page 4 of 7)

RFWCS Instrumentation

3.0 FEEDWATER LINE FLOW, TEMPERATURE, AND RFP DISCHARGE FLOW

3.1 Components

3-FI-3-78A, FW Line A Flow (Panel 3-9-5)

3-FI-3-78B, FW Line B Flow (Panel 3-9-5)

3-FI-3-20, RFP 3A Flow (Panel 3-9-6)

3-FI-3-13, RFP 3B Flow (Panel 3-9-6)

3-FI-3-6, RFP 3C Flow (Panel 3-9-6)

3-TI-3-48, RFP Line A Temp (Panel 3-9-6)

3-TI-3-50, RFP Line B Temp (Panel 3-9-6)

3.2 Description

Feedwater line flow and RFP flow instruments can be manually bypassed in Unit 3 Computer Room. These instruments each have amber lights which illuminate when the instrument has been bypassed automatically by the RFW Control System or manually in Unit 3 Computer Room. Feedwater line flows are density compensated based on average line temperatures. These temperatures are detected by four RTDs, two mounted on each Feedwater line. Feedwater line flows are combined by the RFWCS to produce a Total Feedwater Flow signal used for the following:

- RFWCS Steam Flow/Feed flow mismatch signal (when in THREE ELEMENT control).
- Rod Worth Minimizer Enable setpoint (<22% rated flow).
- Recirc Pump Runback (<16% rated flow).
- Total Feedwater Flow displayed on recorder 3-XR-3-53 on Panel 3-9-5.

Attachment 8
(Page 5 of 7)

RFWCS Instrumentation

3.3 System Operation

RFW Control System will use a Feedwater line flow signal provided the signal is good and valid. A GOOD line flow signal is one that has **NOT** failed and is on scale. A VALID signal is determined by a validation process described in the next paragraph.

Individual Feedwater Line Flows are validated by comparison to each other. If line flows deviate by more than 0.8 Mlbm/hr and Total Steam Flow is >16% rated, then each Feedwater line flow is validated against one half of the total of valid individual RFP discharge flows. If either Feedwater Line Flow signal deviates from the total of the RFP discharge flows by more than 0.8 Mlbm/hr, then Feedwater Line Flow signal is invalid and bypassed by the system.

RFP discharge flows are used for the auto flow balancing feature of the RFW Control System. Individual flows are subtracted from operator supplied flow bias with resultant error signal sent to individual RFP flow balance blocks in RFWCS.

RFP discharge flow signals are also used for controlling RFP Minimum Flow Valves. The associated minimum flow valve will open when RFP discharge flow falls below 2000 gpm, and closes when RFP discharge flow exceeds 3000 gpm. RFP Discharge Flows are utilized by the Recirc System for 75% Pump Runback (any individual RFP Discharge Flow < 16% and Reactor Level < 27 inches).

3.4 Failure Mechanisms

When RFWCS declares a Feedwater Line Flow or a RFP Discharge Flow signal bad or invalid, then the signal is automatically bypassed (amber light illuminates on instrument) and RFWCS INPUT FAILURE annunciation (3-XA-55-6C, window 14) will alarm.

When RFWCS declares a RFW Line Temperature signal bad, then RFWCS INPUT FAILURE annunciation (3-XA-9-6C, window 14) will alarm.

If both Feedwater Line Flows are declared bad or invalid, then RFWCS will automatically transfer to SINGLE ELEMENT control.

If both temperature inputs are lost from associated Feedwater Line or there is a deviation between two sensors on same Feedwater Line of greater than 5°F, then the average temperature signal is declared bad by the system. A default temperature signal of ~ 380°F is produced for density compensation.

If RFP Discharge Flow signal is bad, then minimum flow valve will open.

BFN **Unit 3**	**Reactor Feedwater System**	**3-OI-3** **Rev. 0114** **Page 301 of 330**

Attachment 4
(Page 2 of 3)

Single/Three Element Control

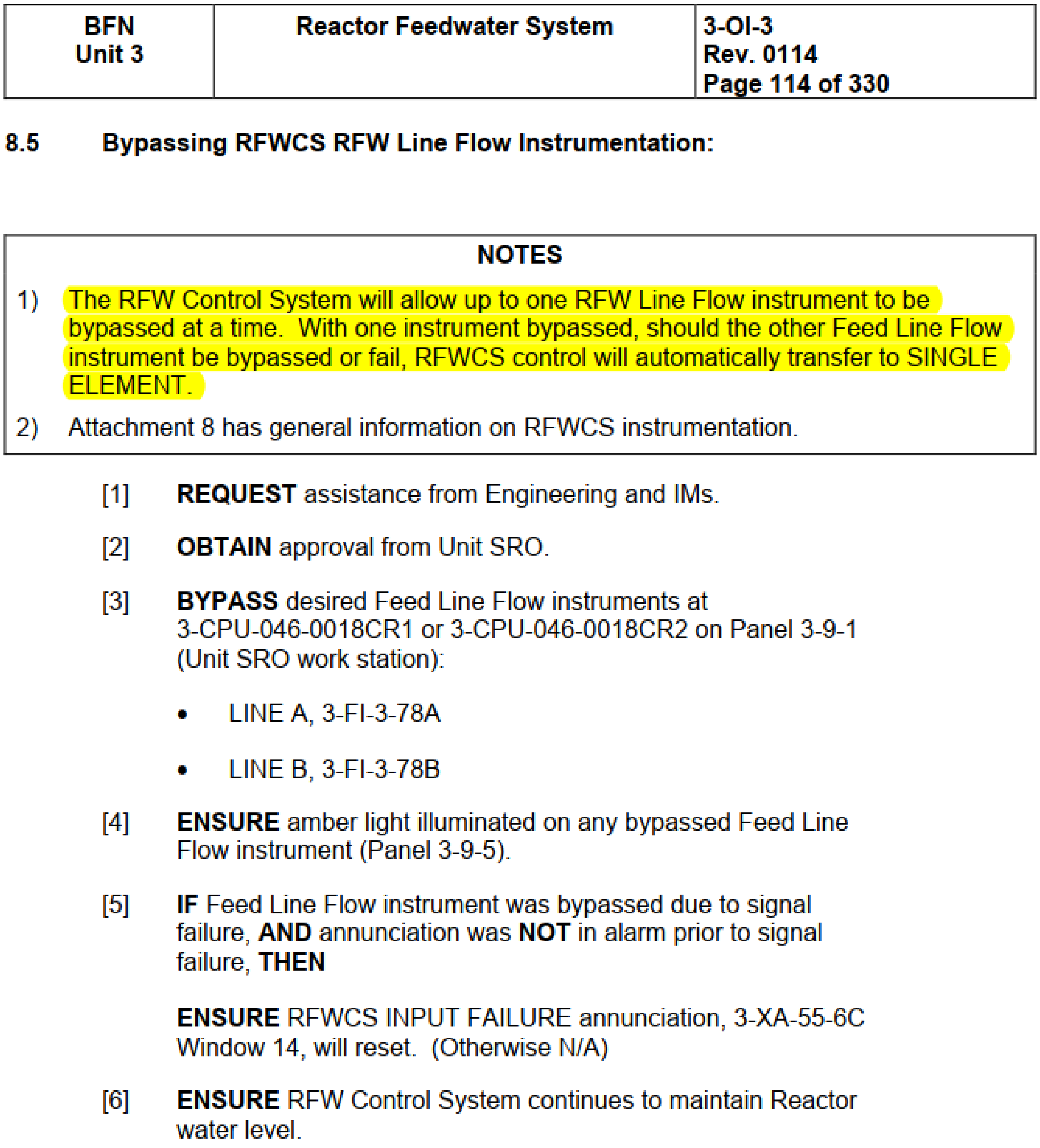
3.0 DESCRIPTION

SINGLE ELEMENT An automatic operating mode where one feedback input is used, Narrow Range Reactor water level. In SINGLE ELEMENT control, the operator adjusted Reactor water level setpoint is compared to the average of the valid Narrow Range level signals. SINGLE ELEMENT is used with automatic operation of RFW Startup Level Control PDS and Reactor Water Level Control PDS (with at least one individual RFPT Speed Control PDS in AUTO).

THREE ELEMENT An automatic operating mode using three feedback inputs: Narrow Range water level, Total Steam Flow, and Total Feedwater Flow. In THREE ELEMENT control, the operator adjusted Reactor level setpoint is compared to the average of the valid Narrow Range Reactor water level signals. A resultant level error signal is generated. This level error signal is summed with a Total Steam Flow signal and sent to the Steam Flow - Feed Flow Mismatch control block. The control block takes demand signal and Total Feedwater Flow signal and produces a Steam Flow/Feed Flow signal and is processed through the single element control logic. The demand signal is then sent to RFPT Control logics, where it is combined with automatic flow balance (flow biasing) inputs.

The RFW Control System will **NOT** allow THREE ELEMENT operation until all of the following permissives are met:

- Reactor Water Level Control (PDS), 3-LIC-46-5, in AUTO and at least one RFPT Speed Control (PDS) in AUTO.
- A valid Total Steam Flow signal exists (Four good steam line flows, or at least two good steam line flow signals and a valid Turbine First Stage pressure signal).
- A valid Total Feedwater line flow signal exists (Two good Feedwater line flow signals, or one good Feedwater line flow and three valid individual RFP flow signals).
- Total Steam Flow >16%.
- At least one valid Narrow Range Level signal.



BFN **Unit 3**	**Reactor Feedwater System**	**3-OI-3** **Rev. 0114** **Page 114 of 330**

8.5 Bypassing RFWCS RFW Line Flow Instrumentation:

NOTES

1) The RFW Control System will allow up to one RFW Line Flow instrument to be bypassed at a time. With one instrument bypassed, should the other Feed Line Flow instrument be bypassed or fail, RFWCS control will automatically transfer to SINGLE ELEMENT.

2) Attachment 8 has general information on RFWCS instrumentation.

[1] **REQUEST** assistance from Engineering and IMs.

[2] **OBTAIN** approval from Unit SRO.

[3] **BYPASS** desired Feed Line Flow instruments at 3-CPU-046-0018CR1 or 3-CPU-046-0018CR2 on Panel 3-9-1 (Unit SRO work station):

- LINE A, 3-FI-3-78A
- LINE B, 3-FI-3-78B

[4] **ENSURE** amber light illuminated on any bypassed Feed Line Flow instrument (Panel 3-9-5).

[5] **IF** Feed Line Flow instrument was bypassed due to signal failure, **AND** annunciation was **NOT** in alarm prior to signal failure, **THEN**

ENSURE RFWCS INPUT FAILURE annunciation, 3-XA-55-6C Window 14, will reset. (Otherwise N/A)

[6] **ENSURE** RFW Control System continues to maintain Reactor water level.

Examination Outline Cross-reference:	Level	RO	SRO
262001 (SF6 AC) AC Electrical Distribution	Tier #	2	------
K5.02 (10CFR 55.41.5)	Group #	1	------
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AC Electrical Distribution: - Breaker control power	K/A #	262001K5.02	
	Importance Rating	3.5	------

Proposed Question: **# 45**

Unit 1 and Unit 2 are operating at 100% RTP with the following conditions:

- Unit 1 has RHR Loop I in Suppression Pool Cooling
- Unit 2 has RHR Loop I in Suppression Pool Cooling
- 250V Shutdown Battery ‘A’ de-energizes due to a fault

Given the conditions above, RHR Pump(s) ____**(1)**____ control power is(are) affected by this failure.

The affected 4KV Shutdown Board breaker(s) control power is provided by a(an) ____**(2)**____ source.

A. (1) 1A **ONLY**
(2) DC

B. (1) 1A **ONLY**
(2) AC

C. (1) 1A **AND** 2A
(2) DC

D. (1) 1A **AND** 2A
(2) AC

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that both 1A and 2A RHR Pumps are powered from 4KV Shutdown Board A where 3A RHR pump does not share the same power supply. It’s powered from the 4KV Shutdown Board 3EA. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that some electrical boards and panels do have control power provided by AC sources.

C **CORRECT:** *(See attached)* In accordance with 2-OI-74, Residual Heat Removal System, RHR Pumps 1A and 2A both receive power from the same power supply which is 4KV Shutdown Board A. Given that 250 Shutdown Battery 'A' has de-energized, the control power for 4KV Shutdown Board A has been lost due to the failure. For second part, in accordance with 0-OI-57D, DC Electrical System, 250VDC Shutdown Battery 'A' provides control power to the affected 4KV Shutdown Board 'A'.

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 operational implications of breaker control power as it relates to specific AC Electrical Distribution 4KV boards. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions related to the complex BFN Electrical Systems both AC and DC. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-OI-74, Rev. 113 (Attach if not previously provided)
2-OI-74, Rev. 189
0-OI-57D, Rev. 179
0-OI-57B, Rev. 200

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.036, Obj. 6c (As available)

Question Source:

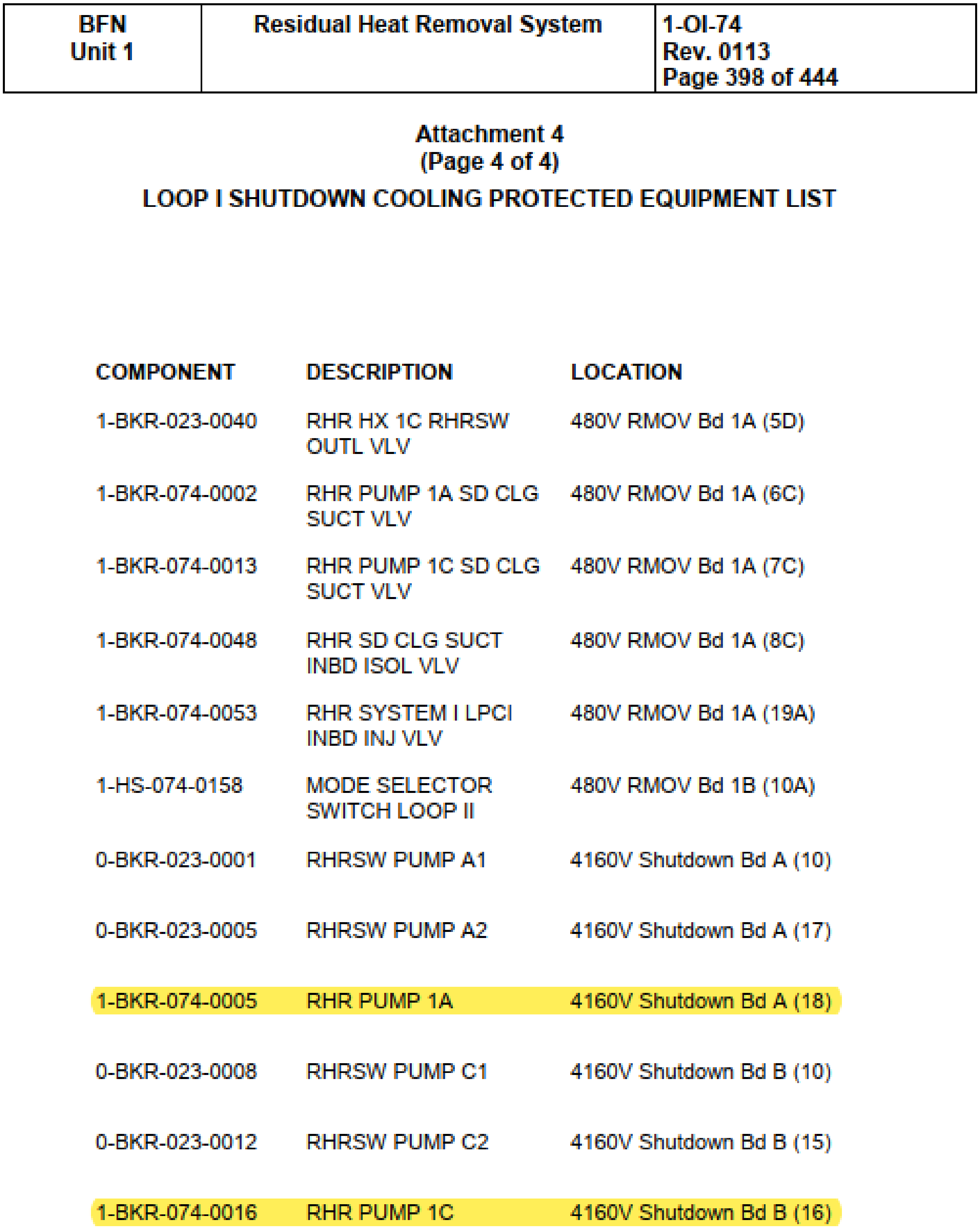
Bank #		
Modified Bank #		(Note changes or attach parent)
New	**X**	

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-OI-74:

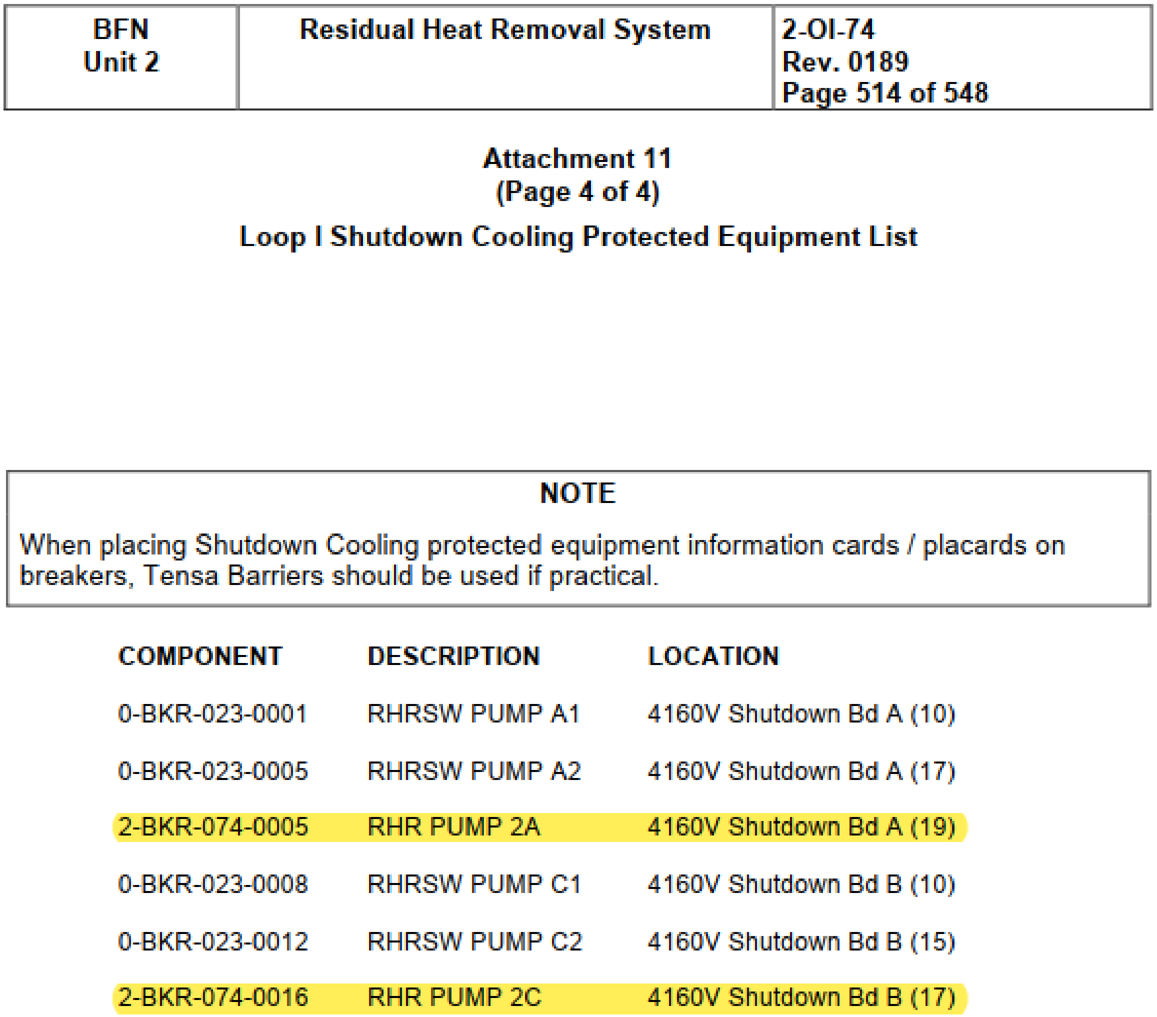


BFN Unit 1	Residual Heat Removal System	1-OI-74 Rev. 0113 Page 398 of 444

Attachment 4
(Page 4 of 4)
LOOP I SHUTDOWN COOLING PROTECTED EQUIPMENT LIST

COMPONENT	DESCRIPTION	LOCATION
1-BKR-023-0040	RHR HX 1C RHRSW OUTL VLV	480V RMOV Bd 1A (5D)
1-BKR-074-0002	RHR PUMP 1A SD CLG SUCT VLV	480V RMOV Bd 1A (6C)
1-BKR-074-0013	RHR PUMP 1C SD CLG SUCT VLV	480V RMOV Bd 1A (7C)
1-BKR-074-0048	RHR SD CLG SUCT INBD ISOL VLV	480V RMOV Bd 1A (8C)
1-BKR-074-0053	RHR SYSTEM I LPCI INBD INJ VLV	480V RMOV Bd 1A (19A)
1-HS-074-0158	MODE SELECTOR SWITCH LOOP II	480V RMOV Bd 1B (10A)
0-BKR-023-0001	RHRSW PUMP A1	4160V Shutdown Bd A (10)
0-BKR-023-0005	RHRSW PUMP A2	4160V Shutdown Bd A (17)
1-BKR-074-0005	RHR PUMP 1A	4160V Shutdown Bd A (18)
0-BKR-023-0008	RHRSW PUMP C1	4160V Shutdown Bd B (10)
0-BKR-023-0012	RHRSW PUMP C2	4160V Shutdown Bd B (15)
1-BKR-074-0016	RHR PUMP 1C	4160V Shutdown Bd B (16)

Excerpt from 2-OI-74:



BFN **Unit 2**	**Residual Heat Removal System**	**2-OI-74** **Rev. 0189** **Page 514 of 548**

Attachment 11
(Page 4 of 4)

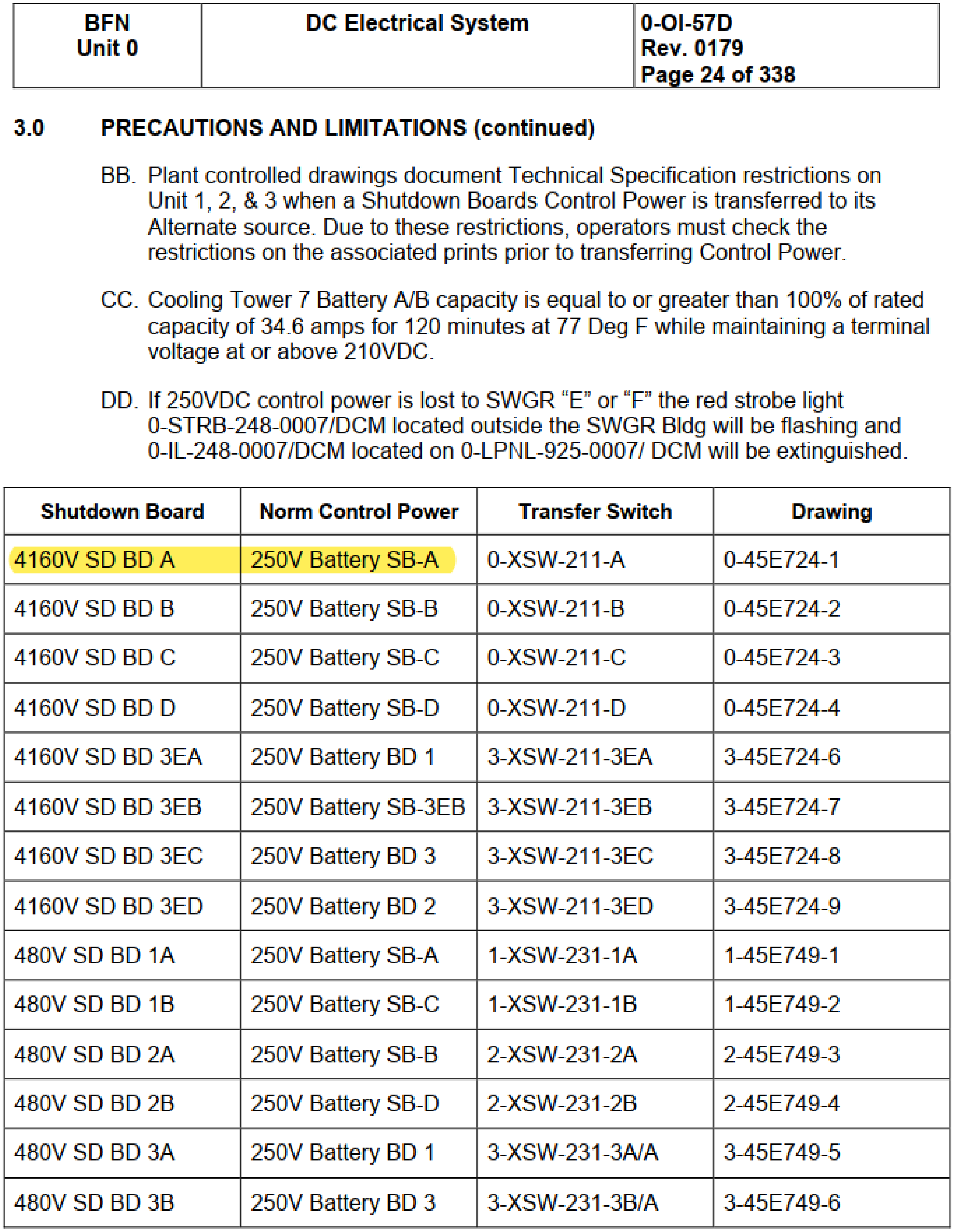
Loop I Shutdown Cooling Protected Equipment List

NOTE

When placing Shutdown Cooling protected equipment information cards / placards on breakers, Tensa Barriers should be used if practical.

COMPONENT	DESCRIPTION	LOCATION
0-BKR-023-0001	RHRSW PUMP A1	4160V Shutdown Bd A (10)
0-BKR-023-0005	RHRSW PUMP A2	4160V Shutdown Bd A (17)
2-BKR-074-0005	RHR PUMP 2A	4160V Shutdown Bd A (19)
0-BKR-023-0008	RHRSW PUMP C1	4160V Shutdown Bd B (10)
0-BKR-023-0012	RHRSW PUMP C2	4160V Shutdown Bd B (15)
2-BKR-074-0016	RHR PUMP 2C	4160V Shutdown Bd B (17)

Excerpt from 0-OI-57D:



BFN **Unit 0**	**DC Electrical System**	**0-OI-57D** **Rev. 0179** **Page 24 of 338**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

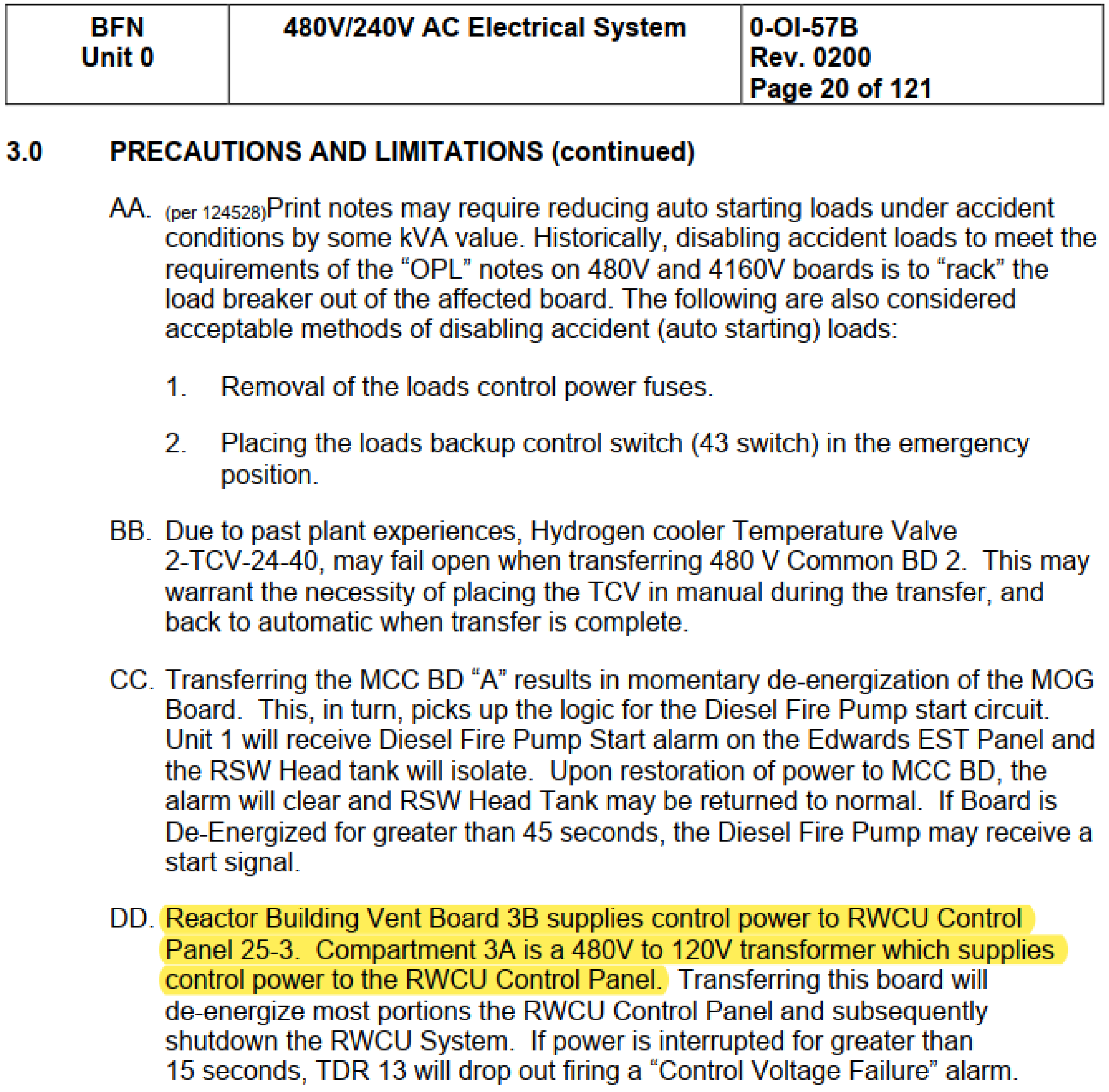
BB. Plant controlled drawings document Technical Specification restrictions on Unit 1, 2, & 3 when a Shutdown Boards Control Power is transferred to its Alternate source. Due to these restrictions, operators must check the restrictions on the associated prints prior to transferring Control Power.

CC. Cooling Tower 7 Battery A/B capacity is equal to or greater than 100% of rated capacity of 34.6 amps for 120 minutes at 77 Deg F while maintaining a terminal voltage at or above 210VDC.

DD. If 250VDC control power is lost to SWGR “E” or “F” the red strobe light 0-STRB-248-0007/DCM located outside the SWGR Bldg will be flashing and 0-IL-248-0007/DCM located on 0-LPNL-925-0007/ DCM will be extinguished.

Shutdown Board	Norm Control Power	Transfer Switch	Drawing
4160V SD BD A	250V Battery SB-A	0-XSW-211-A	0-45E724-1
4160V SD BD B	250V Battery SB-B	0-XSW-211-B	0-45E724-2
4160V SD BD C	250V Battery SB-C	0-XSW-211-C	0-45E724-3
4160V SD BD D	250V Battery SB-D	0-XSW-211-D	0-45E724-4
4160V SD BD 3EA	250V Battery BD 1	3-XSW-211-3EA	3-45E724-6
4160V SD BD 3EB	250V Battery SB-3EB	3-XSW-211-3EB	3-45E724-7
4160V SD BD 3EC	250V Battery BD 3	3-XSW-211-3EC	3-45E724-8
4160V SD BD 3ED	250V Battery BD 2	3-XSW-211-3ED	3-45E724-9
480V SD BD 1A	250V Battery SB-A	1-XSW-231-1A	1-45E749-1
480V SD BD 1B	250V Battery SB-C	1-XSW-231-1B	1-45E749-2
480V SD BD 2A	250V Battery SB-B	2-XSW-231-2A	2-45E749-3
480V SD BD 2B	250V Battery SB-D	2-XSW-231-2B	2-45E749-4
480V SD BD 3A	250V Battery BD 1	3-XSW-231-3A/A	3-45E749-5
480V SD BD 3B	250V Battery BD 3	3-XSW-231-3B/A	3-45E749-6

Excerpt from 0-OI-57B: Supports Distractors B(2), D(2)



BFN **Unit 0**	**480V/240V AC Electrical System**	**0-OI-57B** **Rev. 0200** **Page 20 of 121**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

AA. (per 124528)Print notes may require reducing auto starting loads under accident conditions by some kVA value. Historically, disabling accident loads to meet the requirements of the “OPL” notes on 480V and 4160V boards is to “rack” the load breaker out of the affected board. The following are also considered acceptable methods of disabling accident (auto starting) loads:

1. Removal of the loads control power fuses.
2. Placing the loads backup control switch (43 switch) in the emergency position.

BB. Due to past plant experiences, Hydrogen cooler Temperature Valve 2-TCV-24-40, may fail open when transferring 480 V Common BD 2. This may warrant the necessity of placing the TCV in manual during the transfer, and back to automatic when transfer is complete.

CC. Transferring the MCC BD “A” results in momentary de-energization of the MOG Board. This, in turn, picks up the logic for the Diesel Fire Pump start circuit. Unit 1 will receive Diesel Fire Pump Start alarm on the Edwards EST Panel and the RSW Head tank will isolate. Upon restoration of power to MCC BD, the alarm will clear and RSW Head Tank may be returned to normal. If Board is De-Energized for greater than 45 seconds, the Diesel Fire Pump may receive a start signal.

DD. Reactor Building Vent Board 3B supplies control power to RWCU Control Panel 25-3. Compartment 3A is a 480V to 120V transformer which supplies control power to the RWCU Control Panel. Transferring this board will de-energize most portions the RWCU Control Panel and subsequently shutdown the RWCU System. If power is interrupted for greater than 15 seconds, TDR 13 will drop out firing a “Control Voltage Failure” alarm.

Examination Outline Cross-reference:

295023 (APE 23) Refueling Accidents / 8

AK2.01 (10CFR 55.41.7)

Knowledge of the relationship between Refueling Accidents and the following systems or components:

- Fuel Handling Equipment

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295023AK2.01	
Importance Rating	3.5	------

Proposed Question: **# 46**

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 it full up
- The fuel bundle was then moved towards the Reactor Core

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, in accordance with 0-GOI-100-3A, 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: First part is correct (*See A*). Second part is incorrect but plausible in that the Refueling Rod Block and Platform interlocks are often and easily confused. A Control Rod Block will NOT occur if grapple was unloaded.

C INCORRECT: First part is incorrect but plausible in accordance with Attachment 32, Rod, Bridge, and Hoist Blocks, Block Diagram, since the following interlocks are not met in order for a Refueling Bridge Block to stop before it reaches the Core: one Control Rod withdrawn and a second Control Rod was selected or as stated in this question, the hoist is loaded and would require at least one rod to not be full in once the bridge goes over the core to stop bridge motion. All other interlocks are met as given in the stem. 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 to help prevent Refueling Accidents. 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.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 0-GOI-100-3A, Rev. 92 (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 #	BFN 21-04 #48	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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: **# 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

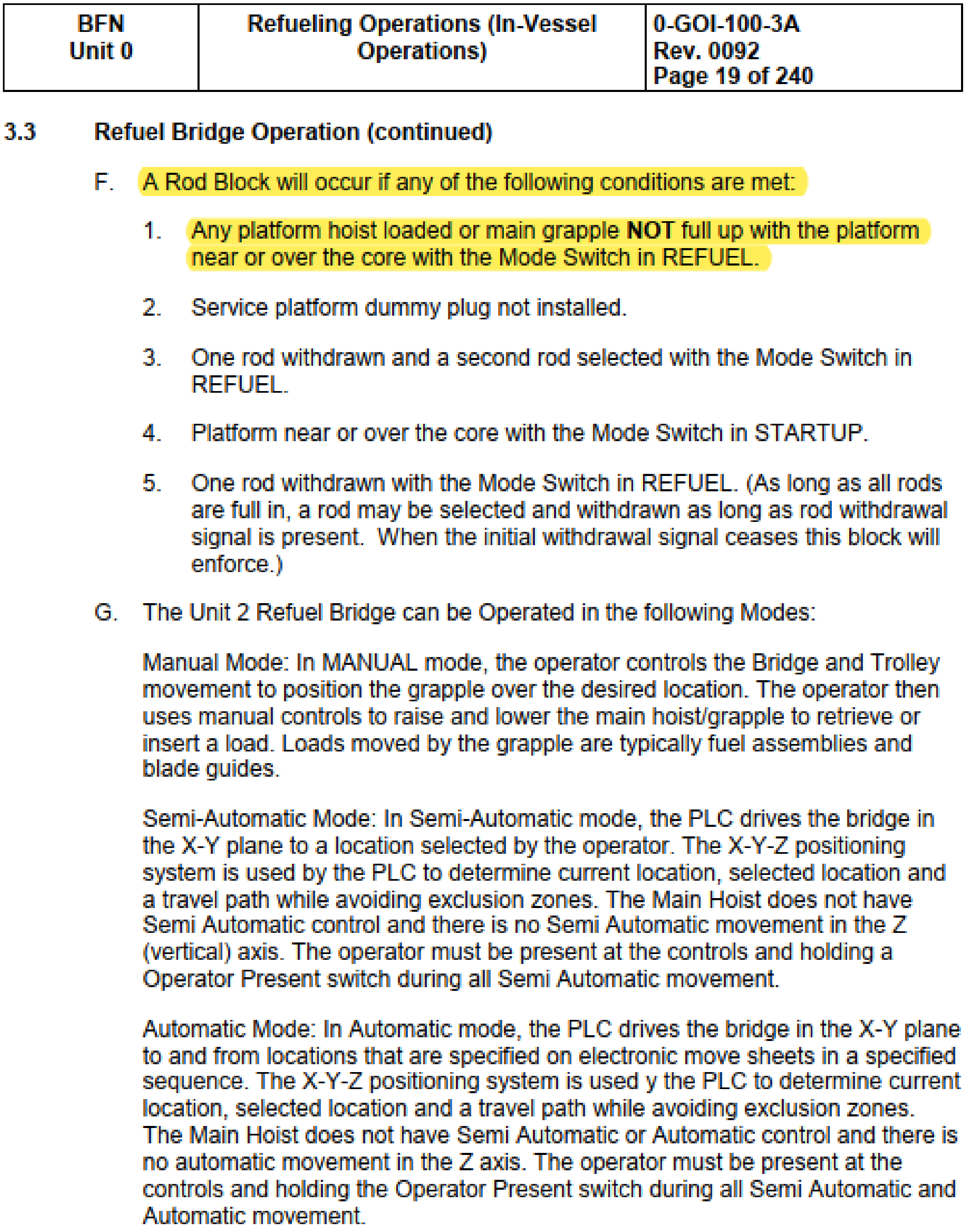
B. (1) continues over
(2) will NOT

C. (1) stops before it reaches
(2) will

D. (1) stops before it reaches
(2) will NOT

Proposed Answer: **A**

Excerpts from 0-GOI-100-3A:



BFN Unit 0	**Refueling Operations (In-Vessel Operations)**	**0-GOI-100-3A Rev. 0092 Page 19 of 240**

3.3 Refuel Bridge Operation (continued)

F. A Rod Block will occur if any of the following conditions are met:

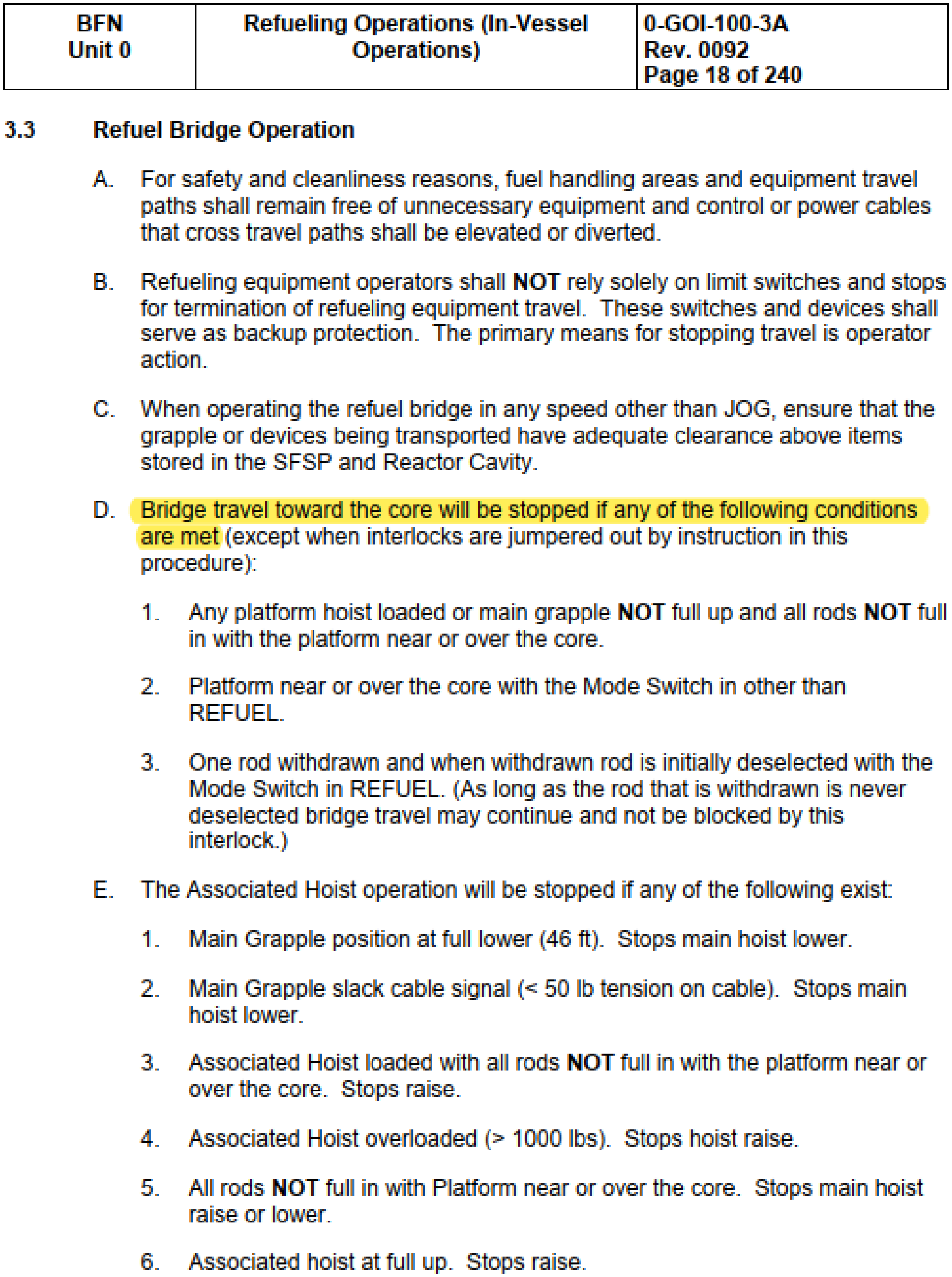
1. Any platform hoist loaded or main grapple **NOT** full up with the platform near or over the core with the Mode Switch in REFUEL.
2. Service platform dummy plug not installed.
3. One rod withdrawn and a second rod selected with the Mode Switch in REFUEL.
4. Platform near or over the core with the 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.)

G. The Unit 2 Refuel Bridge can be Operated in the following Modes:

Manual Mode: In MANUAL mode, the operator controls the Bridge and Trolley movement to position the grapple over the desired location. The operator then uses manual controls to raise and lower the main hoist/grapple to retrieve or insert a load. Loads moved by the grapple are typically fuel assemblies and blade guides.

Semi-Automatic Mode: In Semi-Automatic mode, the PLC drives the bridge in the X-Y plane to a location selected by the operator. The X-Y-Z positioning system is used by the PLC to determine current location, selected location and a travel path while avoiding exclusion zones. The Main Hoist does not have Semi Automatic control and there is no Semi Automatic movement in the Z (vertical) axis. The operator must be present at the controls and holding a Operator Present switch during all Semi Automatic movement.

Automatic Mode: In Automatic mode, the PLC drives the bridge in the X-Y plane to and from locations that are specified on electronic move sheets in a specified sequence. The X-Y-Z positioning system is used y the PLC to determine current location, selected location and a travel path while avoiding exclusion zones. The Main Hoist does not have Semi Automatic or Automatic control and there is no automatic movement in the Z axis. The operator must be present at the controls and holding the Operator Present switch during all Semi Automatic and Automatic movement.



BFN Unit 0	**Refueling Operations (In-Vessel Operations)**	**0-GOI-100-3A Rev. 0092 Page 18 of 240**

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.)

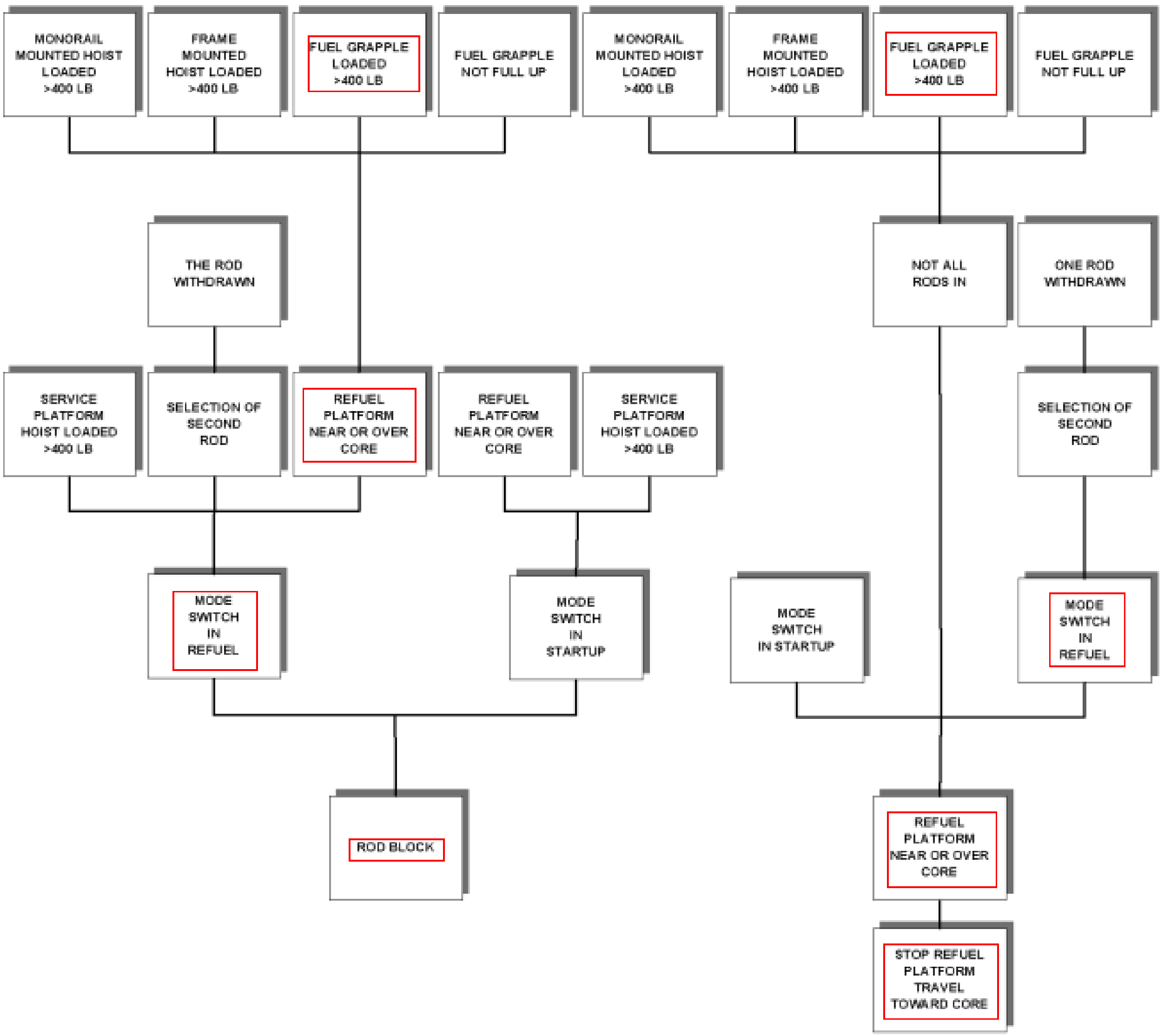
E. The Associated Hoist operation will be stopped if any of the following exist:

1. Main Grapple position at full lower (46 ft). Stops main hoist lower.
2. Main Grapple slack cable signal (< 50 lb tension on cable). Stops main hoist lower.
3. Associated Hoist loaded with all rods **NOT** full in with the platform near or over the core. Stops raise.
4. Associated Hoist overloaded (> 1000 lbs). Stops hoist raise.
5. All rods **NOT** full in with Platform near or over the core. Stops main hoist raise or lower.
6. Associated hoist at full up. Stops raise.

BFN **Unit 0**	**Refueling Operations (In-Vessel Operations)**	**0-GOI-100-3A** **Rev. 0092** **Page 233 of 240**

Attachment 32
(Page 2 of 2)

Rod, Bridge, and Hoist Blocks, Block Diagram



Examination Outline Cross-reference:

295026 (EPE 3) Suppression Pool High Water Temperature / 5

EA2.03 (10CFR 55.41.10)

Ability to determine and/or interpret the following as they apply to Suppression Pool High Water Temperature:

- Reactor Pressure

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295026EA2.03	
Importance Rating	3.5	------

Proposed Question: **# 47**

Unit 1 suffered a LOCA with the following condition:

- Suppression Pool Temperature is 200 °F

In accordance with EOI Curve 3, Heat Capacity Temp Limit (HCTL), the **REQUIRED ACTION** area is entered if Reactor Pressure reaches **(1)** when Suppression Pool Level is **(2)** .

[REFERENCE PROVIDED]

A. (1) 500 psig
(2) 14 feet

B. (1) 500 psig
(2) 15 feet

C. (1) 700 psig
(2) 14 feet

D. (1) 700 psig
(2) 15 feet

Proposed Answer: **D**

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 14 feet and the intersecting point with 500 psig from first part, Suppression Pool Temperature would have to be above 215 °F to exceed Heat Capacity Temperature Limit (HCTL). With the given Suppression Pool Temperature at 200 °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 15 feet and the intersecting point from 500 psig in first part, Suppression Pool Temperature would have to be closer to 210 °F to exceed HCTL. With the given Suppression Pool Temperature at 200 °F, HCTL is not exceeded, therefore action is not required.

C 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 14 feet and the intersecting point from 700 psig in first part, Suppression Pool Temperature would have to be at or above 205 °F to exceed HCTL. With the given Suppression Pool Temperature at 200 °F, HCTL is not exceeded, therefore action is not required.

D **CORRECT:** *(See attached)* In accordance with 1-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 200 °F, follow the 700 psig Reactor Pressure curve to find the intersecting points. Second part, then follow the given 200 °F Suppression Pool Temperature line to where it intersects Suppression Pool Level of 15 feet. The resulting intersecting point is **above** the existing Reactor Pressure indicating that action is REQUIRED since HCTL has been exceeded.

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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 1-EOI-2, Rev. 9 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **1-EOI-2, Curve 3 – Heat Capacity Temperature Limit**

Learning Objective: OPL171.203 Obj. 12 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 2104 #21	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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: **# 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

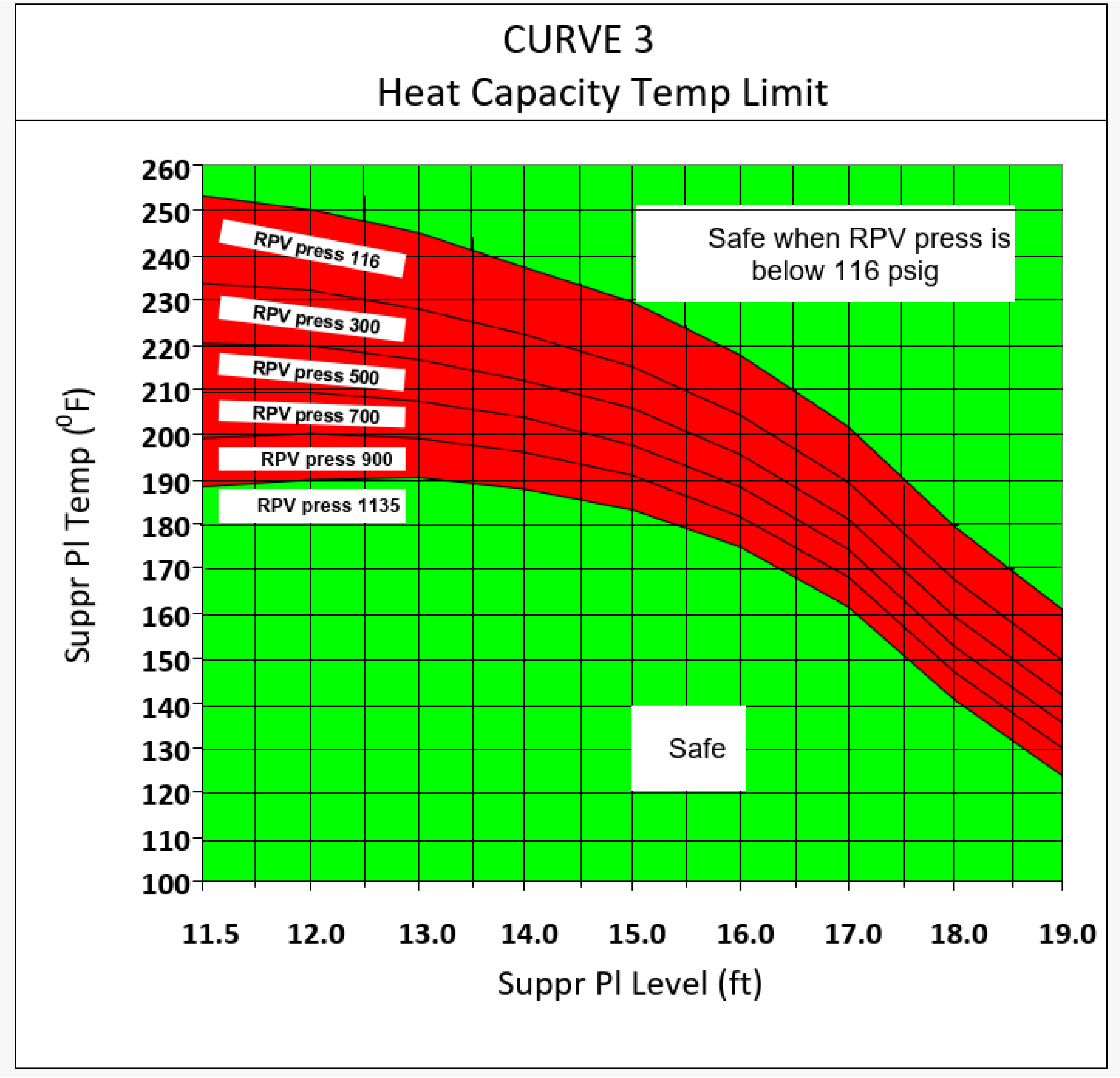
B. (1) 500 psig
(2) 16.5 feet

C. (1) 700 psig
(2) 17.5 feet

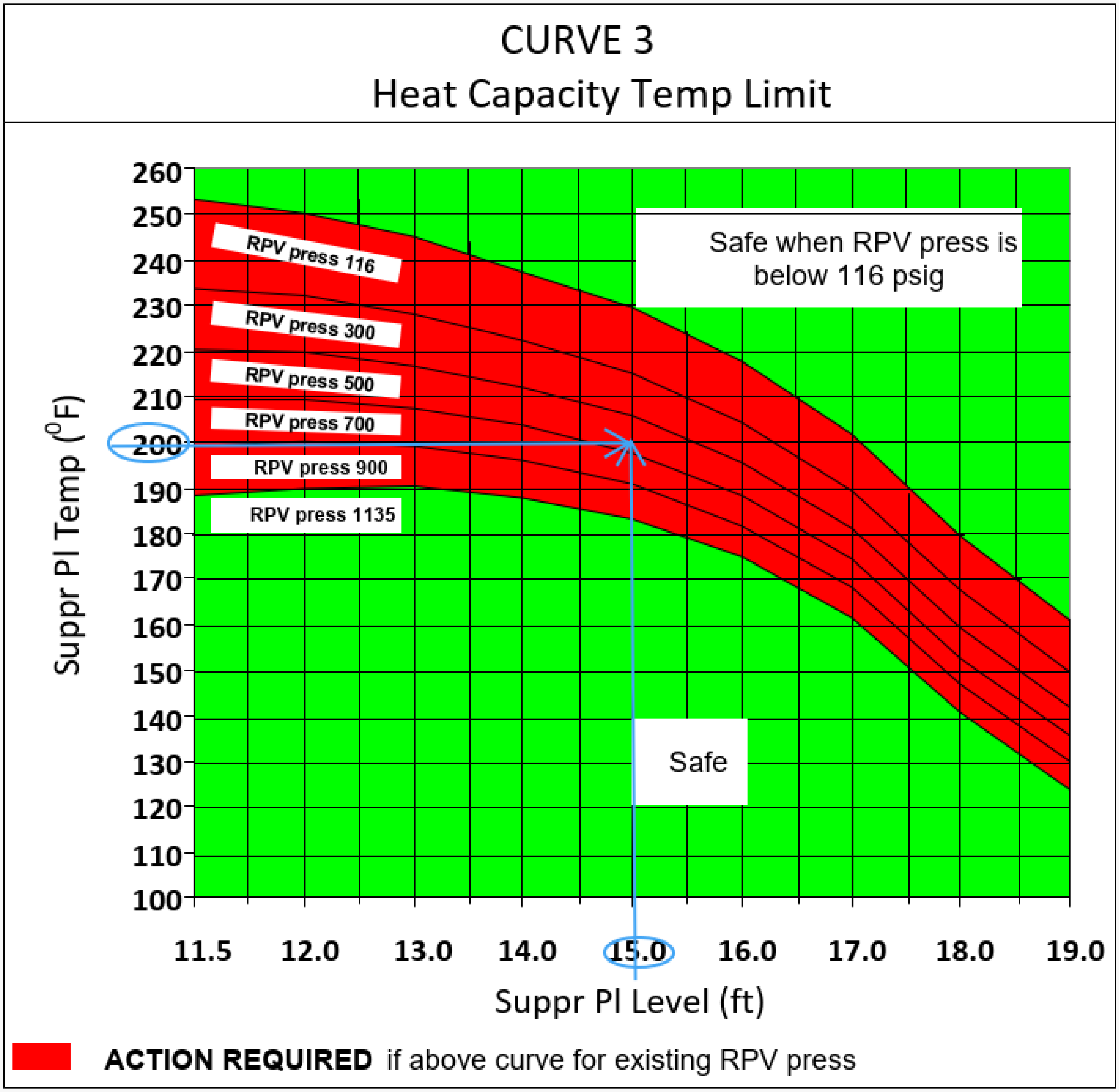
D. (1) 700 psig
(2) 16.5 feet

Proposed Answer: **C**

REFERENCE PROVIDED to candidate: 1-EOI-2, Curve 3, HCTL without ACTION REQUIRED statement



Excerpt from 1-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



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

Examination Outline Cross-reference:

211000 (SF1, SLCS) Standby Liquid Control

K5.01 (10CFR 55.41.5)

Knowledge of the operational implications and cause and effect relationships of the following concepts as they apply to the Standby Liquid Control System:

- Effects of moderator temperature coefficient of reactivity on boron

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	211000K5.01	
Importance Rating	3.0	------

Proposed Question: **# 48**

What effect does moderator temperature have on the ability of boron to shut down the Reactor, if any?

A. At lower moderator temperatures, less boron is needed.

B. At higher moderator temperatures, less boron is needed.

C. At higher moderator temperatures, more boron is needed.

D. Moderator temperature has no effect on the amount of boron needed.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that as moderator temperature lowers, more boron is required to offset the positive reactivity added by the colder moderator temperature. This equates to Cold Shutdown Boron Weight (CSBW) which is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the Reactor shutdown under all conditions. This weight is utilized to assure the Reactor will shutdown irrespective of Control Rod position or RPV temperature.

B **CORRECT**: *(See attached)* As moderator temperature rises, the positive reactivity added by the moderator is less, therefore less boron is required. In accordance with EOIPM 0-V(B), Hot Shutdown Boron Weight (HSBW) is the least weight of soluble boron which, if injected (tank level lowered by 12%) into the RPV and mixed uniformly, will maintain the Reactor shutdown under hot standby conditions. This applies directly to an ATWS condition.

C INCORRECT: Incorrect but plausible in that as moderator temperature rises, the reactivity added by the moderator is less, so less boron would be needed. *(See B)*

D INCORRECT: Incorrect but plausible in that moderator temperature has a direct impact on the amount of boron needed to shut down a Reactor. *(See A and B)*

RO Level Justification: Tests the candidate’s knowledge of the operational implications of moderator temperature coefficient of reactivity on boron as it relates to the Standby Liquid Control System. This question is rated as memory due to strictly recalling facts related to the Standard Liquid Control.

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

Technical Reference(s): EOIPM 0-V(D), Rev. 1 (Attach if not previously provided)

OPL171.039, Rev. 24

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.039 Obj. 11a (As available)

Question Source:

Bank #	Hope Creek 2010 NRC #10
Modified Bank #	
New	

(Note changes or attach parent)

Question History: Last NRC Exam 2010

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

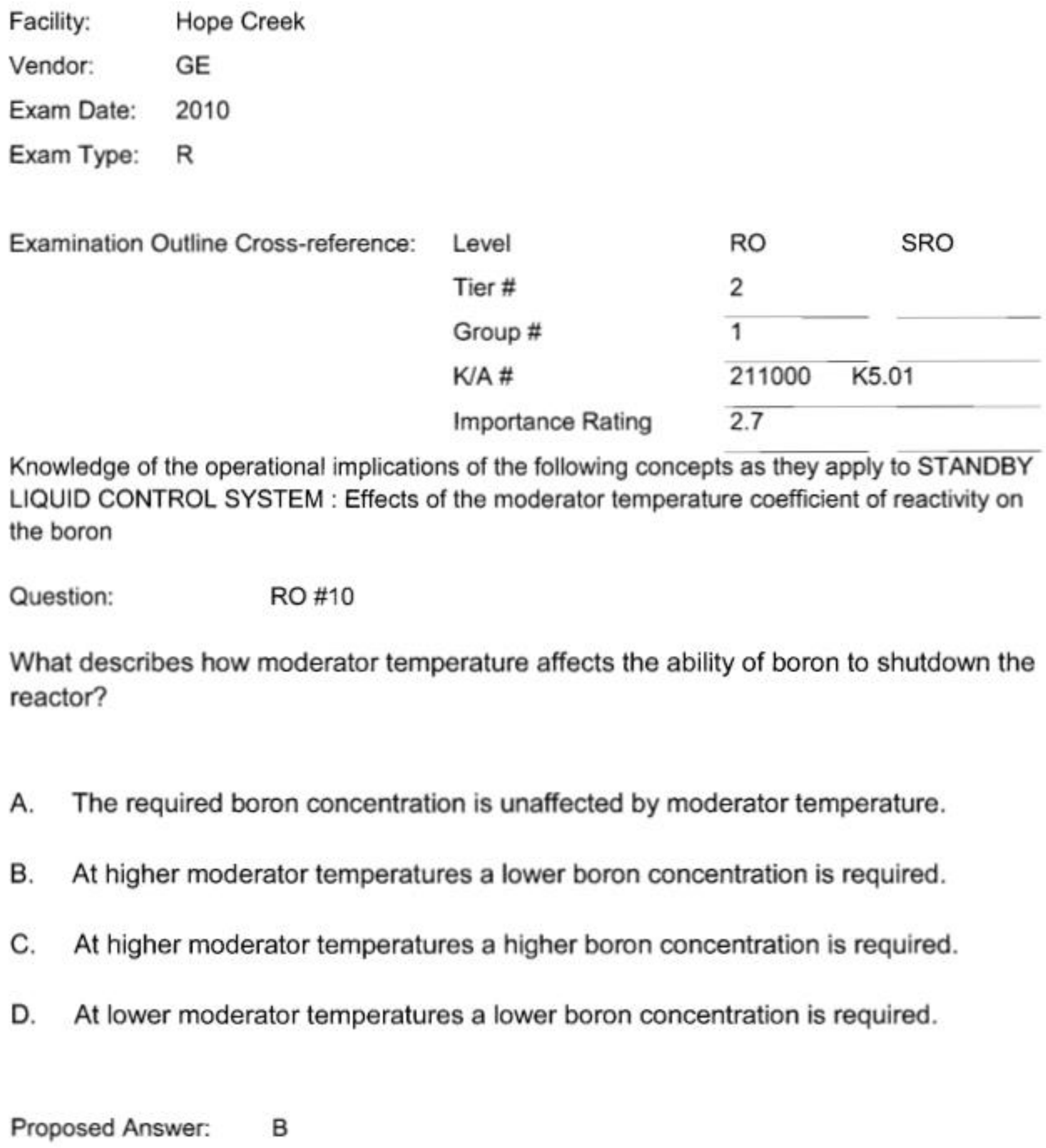
10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Copy of Bank Question:



Facility: Hope Creek

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	211000 K5.01	
Importance Rating	2.7	

Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM : Effects of the moderator temperature coefficient of reactivity on the boron

Question: RO #10

What describes how moderator temperature affects the ability of boron to shutdown the reactor?

A. The required boron concentration is unaffected by moderator temperature.

B. At higher moderator temperatures a lower boron concentration is required.

C. At higher moderator temperatures a higher boron concentration is required.

D. At lower moderator temperatures a lower boron concentration is required.

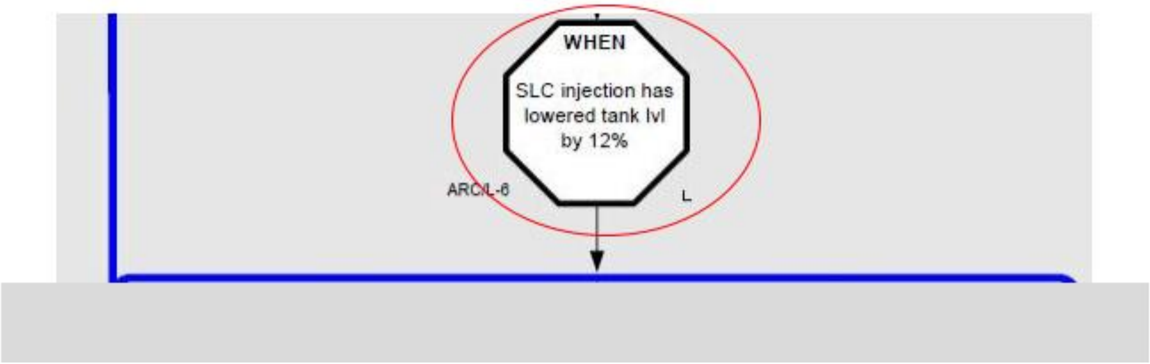
Proposed Answer: B

Excerpts from EOIPM 0-V(D):

BFN **Unit 0**	**EOI-1A, ATWS RPV Control Bases**	**EOIPM Section 0-V(D)** **Rev. 0001** **Page 66 of 179**

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

ARC/L-6



BFN **Unit 0**	**EOI-1A, ATWS RPV Control Bases**	**EOIPM Section 0-V(D)** **Rev. 0001** **Page 67 of 179**

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/L-6

Once the Hot Shutdown Boron Weight has been injected, Step ARC/L-8 will restore RPV water level to reestablish natural circulation flow and distribute boron that has accumulated in the lower plenum throughout the core. This strategy minimizes the integrated primary containment heatup by maintaining reactor power as low as possible during the time of boron injection.

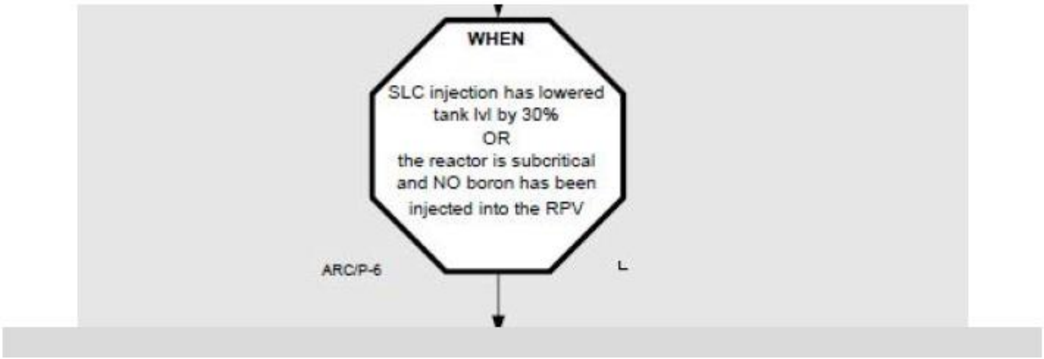
The Hot Shutdown Boron Weight (HSBW) is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under hot standby conditions.

Supports Distractor A and D:

BFN Unit 0	**EOI-1A, ATWS RPV Control Bases**	**EOIPM Section 0-V(D) Rev. 0001 Page 118 of 179**

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

ARC/P-6



BFN Unit 0	**EOI-1A, ATWS RPV Control Bases**	**EOIPM Section 0-V(D) Rev. 0001 Page 119 of 179**

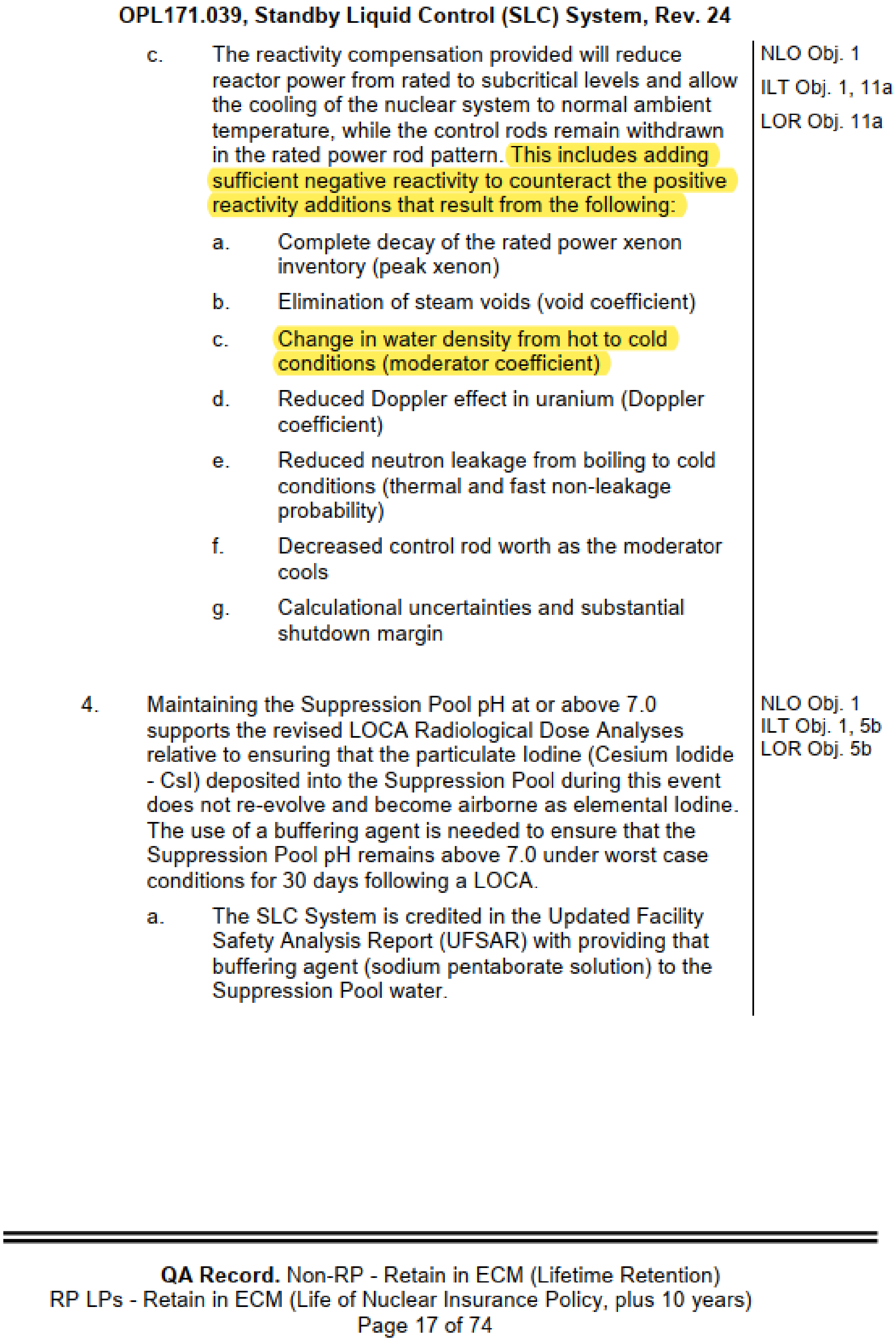
1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/P-6

RPV depressurization and cooldown in subsequent steps may not proceed until at least one of the three conditions listed in this step is satisfied.

- Injection of the Cold Shutdown Boron Weight (CSBW) of boron into the RPV also provides adequate assurance that the reactor is and will remain shutdown. The CSBW is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions. This weight is utilized to assure the reactor will remain shutdown irrespective of control rod position or RPV temperature. Refer to EOIPM Section 0-V(B) for discussion of the CSBW. If any amount of boron less than the CSBW has been injected into the RPV, cooldown is not permitted unless it can be determined that control rod insertion alone assures the reactor will remain subcritical under all conditions. The core reactivity response from cooldown in a partially borated core is unpredictable and subsequent steps may not prescribe the correct actions for such conditions if criticality were to occur.

Excerpt from OPL171.039 Lesson Plan:



OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

c. The reactivity compensation provided will reduce reactor power from rated to subcritical levels and allow the cooling of the nuclear system to normal ambient temperature, while the control rods remain withdrawn in the rated power rod pattern. This includes adding sufficient negative reactivity to counteract the positive reactivity additions that result from the following:

NLO Obj. 1
ILT Obj. 1, 11a
LOR Obj. 11a

  a. Complete decay of the rated power xenon inventory (peak xenon)
  b. Elimination of steam voids (void coefficient)
  c. Change in water density from hot to cold conditions (moderator coefficient)
  d. Reduced Doppler effect in uranium (Doppler coefficient)
  e. Reduced neutron leakage from boiling to cold conditions (thermal and fast non-leakage probability)
  f. Decreased control rod worth as the moderator cools
  g. Calculational uncertainties and substantial shutdown margin

4. Maintaining the Suppression Pool pH at or above 7.0 supports the revised LOCA Radiological Dose Analyses relative to 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 use of a buffering agent is needed to ensure that the Suppression Pool pH remains above 7.0 under worst case conditions for 30 days following a LOCA.

NLO Obj. 1
ILT Obj. 1, 5b
LOR Obj. 5b

  a. The SLC System is credited in the Updated Facility Safety Analysis Report (UFSAR) with providing that buffering agent (sodium pentaborate solution) to the Suppression Pool water.

QA Record. Non-RP - Retain in ECM (Lifetime Retention)
RP LPs - Retain in ECM (Life of Nuclear Insurance Policy, plus 10 years)
Page 17 of 74

Examination Outline Cross-reference:	Level	RO	SRO
400000 (SF8 CCS) Component Cooling Water	Tier #	2	------
A4.01 (10CFR 55.41.7)	Group #	1	------
Ability to manually operate and/or monitor in the control room: • CCW indications and control	K/A #	400000A4.01	
	Importance Rating	3.8	------

Proposed Question: **# 49**

Unit 2 is operating at 100% RTP with the following plant conditions:

- ‘2A’ Reactor Building Closed Cooling Water (RBCCW) Pump is tagged for bearing replacement
- The Spare RBCCW Pump ‘1C’ is aligned to Unit 2
- 'A' EDG is supplying its 4KV Shutdown Board as the only source

Subsequently, a steam line break inside containment occurs, causing Drywell Pressure to reach 3.0 psig and Reactor Pressure to lower to 440 psig.

Note: **NO** Operator actions have been taken.

Given the above conditions, ‘2B’ RBCCW pump restarts in ___**(1)**___ and 60 seconds later, ‘1C’ RBCCW pump ____**(2)**_____ running.

A. **(1)** 40 seconds
(2) is NOT

B. **(1)** 40 seconds
(2) is

C. **(1)** 43 seconds
(2) is NOT

D. **(1)** 43 seconds
(2) is

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that after Load Shed initiates, the ‘2A’ RBCCW Pump would auto restart in 40 seconds (currently tagged out) and ‘2B’ RBCCW Pump will auto start after a (3) three second time delay if ‘2A’ fails to restart. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that common misconceptions exist pertaining to Load Shed Logic between the Unit 1 and 2 as compared to Unit 3. Plausible to believe that '1C' RBCCW Pump (used as the spare between Units 1 and 2) is not affected by Load Shed therefore it will be running if aligned after 60 seconds.

C **CORRECT**: (*See attached*) In accordance with 2-OI-70, RBCCW System, RBCCW is affected by 480V Load Shed. 480V Load Shedding Logic is initiated by a Common Accident Signal (CAS) from the Unit's Core Spray logic on Reactor Water Level (-) 122 inches or High Drywell Pressure (+) 2.45 psig with Reactor Pressure below 450 psig when **ANY** Unit 1 or 2 Diesel Generator is tied to its Shutdown Board as the only source of power. After Load Shed initiates, '2A' RBCCW Pump would auto restart in 40 seconds (given as tagged out) and '2B' RBCCW Pump will auto start after a (3) three second time delay if '2A' fails to restart which gives the 43 seconds. For second part, in accordance with 1/2-AOI-57-1D, 480V Load Shed, '1C' RBCCW Pump (used as the spare between Units 1 and 2) also trips during Load Shed and does NOT have auto restart capability, but can be restarted manually after a 40 second time delay. However, given in the stem that no Operator action was taken, '1C' RBCCW Pump would not have been manually started, therefore it is not running.

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 monitor Main Control Room RBCCW indications and controls when 480V Load Shed Logic initiates. 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-70, Rev. 71 (Attach if not previously provided)
1/2-AOI-57-1D, Rev. 5
OPL171.047, Rev. 14

Proposed references to be provided to applicants during examination: **NONE**

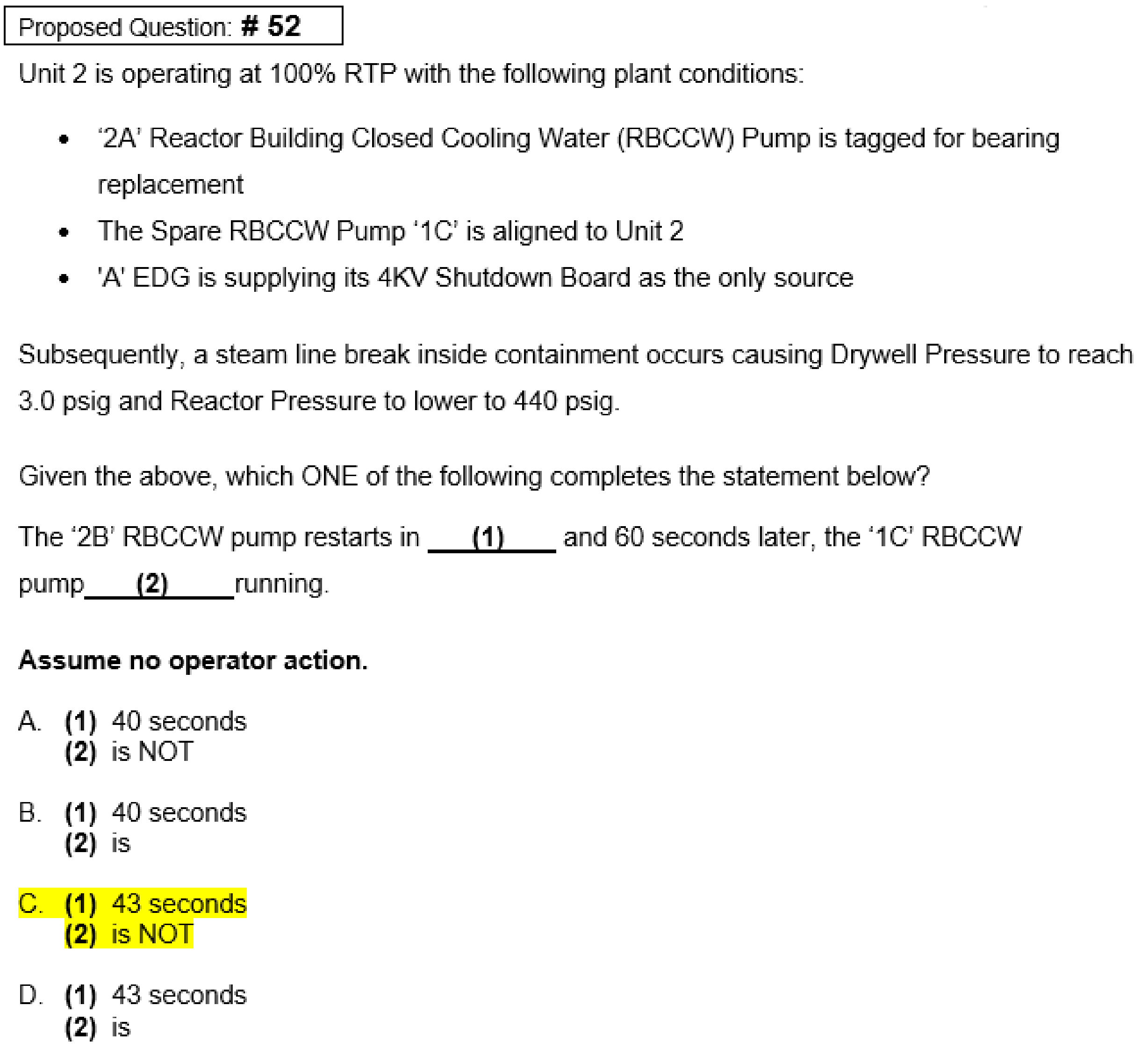
Learning Objective: OPL171.047 Obj. 5 (As available)

Question Source:	Bank #	BFN 1804 NRC #52	
	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

Copy of Bank Question:



Proposed Question: **# 52**

Unit 2 is operating at 100% RTP with the following plant conditions:

- '2A' Reactor Building Closed Cooling Water (RBCCW) Pump is tagged for bearing replacement
- The Spare RBCCW Pump '1C' is aligned to Unit 2
- 'A' EDG is supplying its 4KV Shutdown Board as the only source

Subsequently, a steam line break inside containment occurs causing Drywell Pressure to reach 3.0 psig and Reactor Pressure to lower to 440 psig.

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

The '2B' RBCCW pump restarts in ____**(1)**____ and 60 seconds later, the '1C' RBCCW pump____**(2)**_____running.

Assume no operator action.

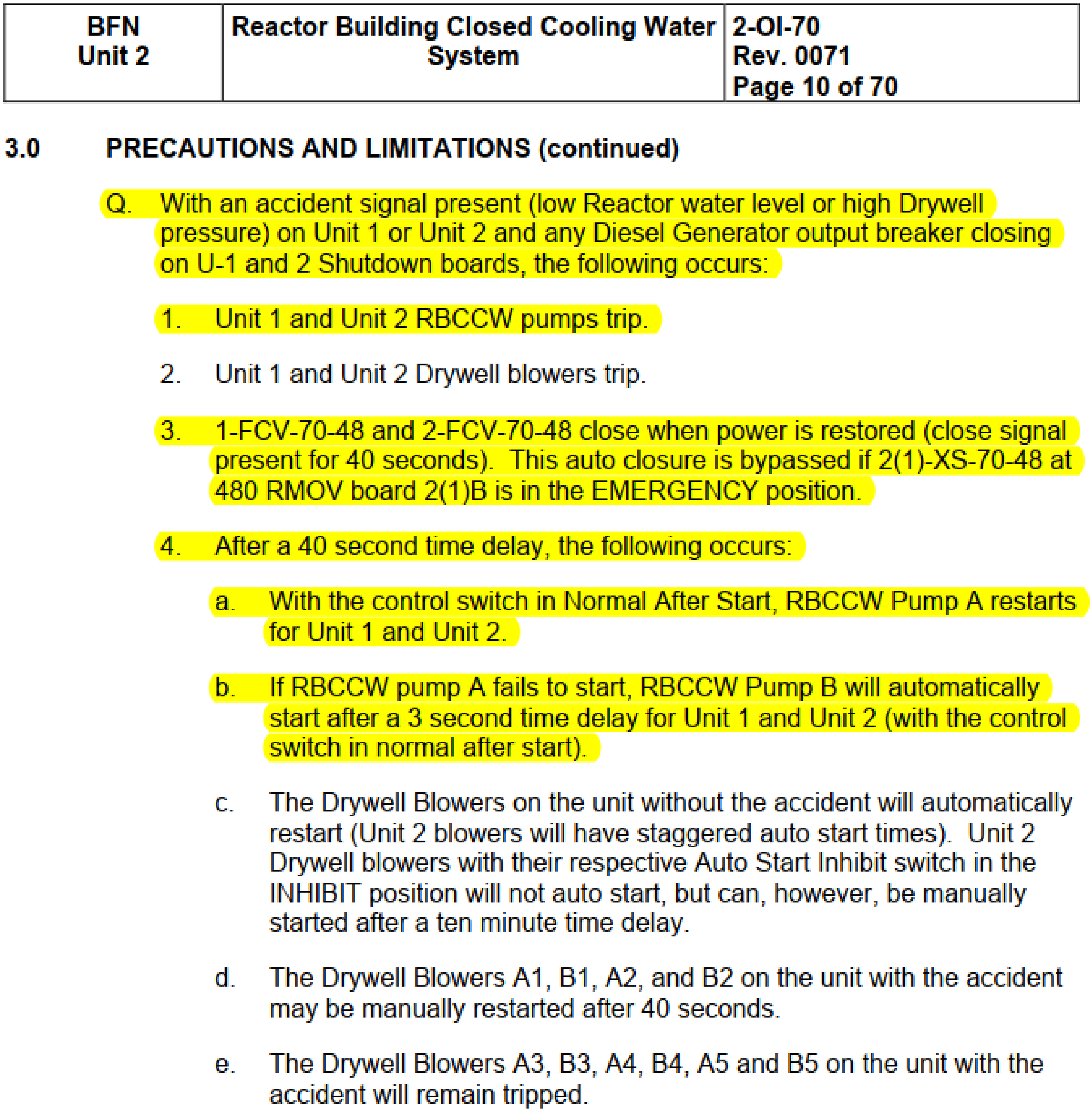
A. **(1)** 40 seconds
(2) is NOT

B. **(1)** 40 seconds
(2) is

C. **(1)** 43 seconds
(2) is NOT

D. **(1)** 43 seconds
(2) is

Excerpts from 2-OI-70:



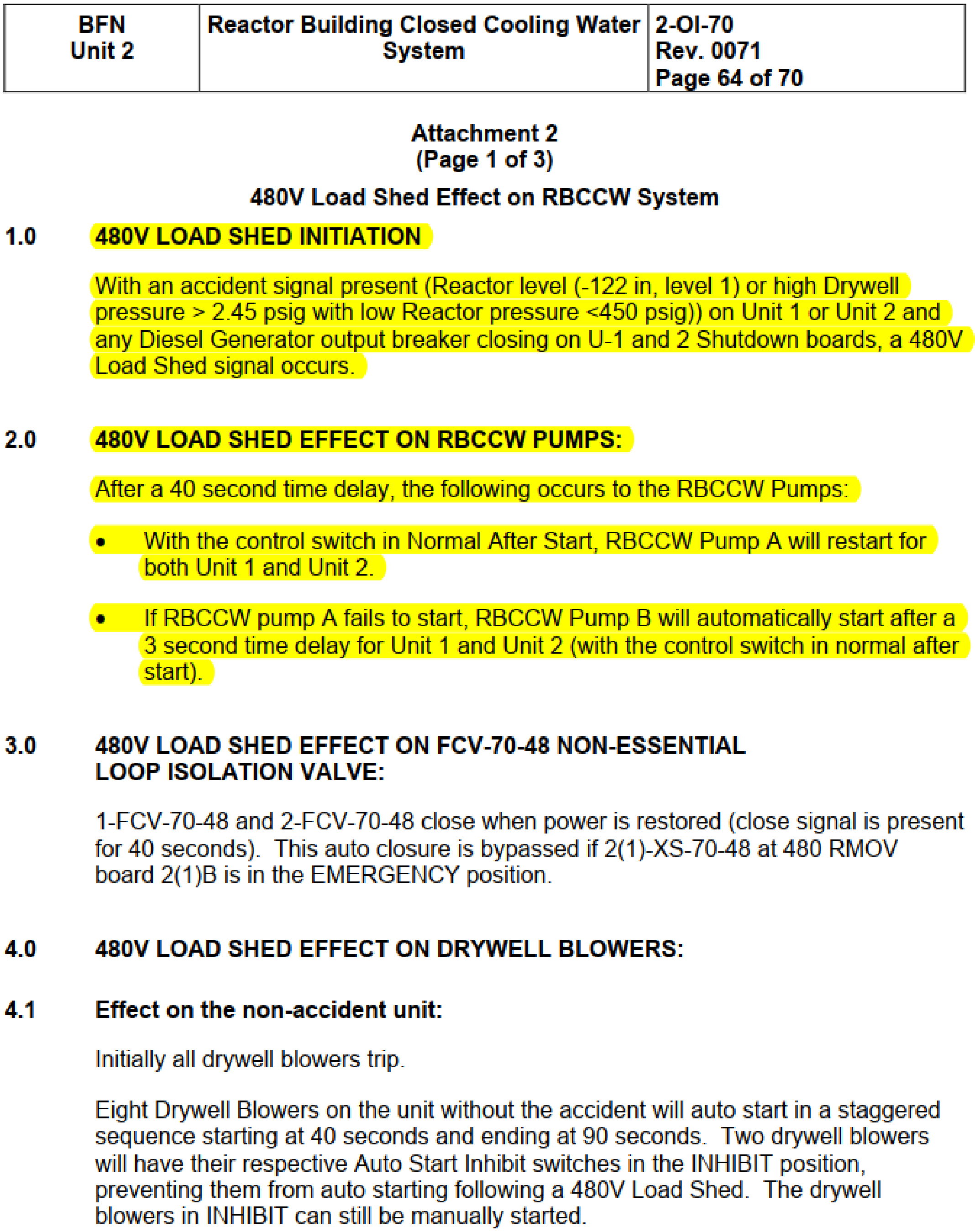
BFN **Unit 2**	**Reactor Building Closed Cooling Water System**	**2-OI-70** **Rev. 0071** **Page 10 of 70**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Q. With an accident signal present (low Reactor water level or high Drywell pressure) on Unit 1 or Unit 2 and any Diesel Generator output breaker closing on U-1 and 2 Shutdown boards, the following occurs:

1. Unit 1 and Unit 2 RBCCW pumps trip.
2. Unit 1 and Unit 2 Drywell blowers trip.
3. 1-FCV-70-48 and 2-FCV-70-48 close when power is restored (close signal present for 40 seconds). This auto closure is bypassed if 2(1)-XS-70-48 at 480 RMOV board 2(1)B is in the EMERGENCY position.
4. After a 40 second time delay, the following occurs:
    - a. With the control switch in Normal After Start, RBCCW Pump A restarts for Unit 1 and Unit 2.
    - b. If RBCCW pump A fails to start, RBCCW Pump B will automatically start after a 3 second time delay for Unit 1 and Unit 2 (with the control switch in normal after start).
    - c. The Drywell Blowers on the unit without the accident will automatically restart (Unit 2 blowers will have staggered auto start times). Unit 2 Drywell blowers with their respective Auto Start Inhibit switch in the INHIBIT position will not auto start, but can, however, be manually started after a ten minute time delay.
    - d. The Drywell Blowers A1, B1, A2, and B2 on the unit with the accident may be manually restarted after 40 seconds.
    - e. The Drywell Blowers A3, B3, A4, B4, A5 and B5 on the unit with the accident will remain tripped.

Supports Distractors A(1), B(1):



BFN Unit 2	**Reactor Building Closed Cooling Water System**	**2-OI-70 Rev. 0071 Page 64 of 70**

Attachment 2
(Page 1 of 3)

480V Load Shed Effect on RBCCW System

1.0 480V LOAD SHED INITIATION

With an accident signal present (Reactor level (-122 in, level 1) or high Drywell pressure > 2.45 psig with low Reactor pressure <450 psig)) on Unit 1 or Unit 2 and any Diesel Generator output breaker closing on U-1 and 2 Shutdown boards, a 480V Load Shed signal occurs.

2.0 480V LOAD SHED EFFECT ON RBCCW PUMPS:

After a 40 second time delay, the following occurs to the RBCCW Pumps:

- With the control switch in Normal After Start, RBCCW Pump A will restart for both Unit 1 and Unit 2.
- If RBCCW pump A fails to start, RBCCW Pump B will automatically start after a 3 second time delay for Unit 1 and Unit 2 (with the control switch in normal after start).

3.0 480V LOAD SHED EFFECT ON FCV-70-48 NON-ESSENTIAL LOOP ISOLATION VALVE:

1-FCV-70-48 and 2-FCV-70-48 close when power is restored (close signal is present for 40 seconds). This auto closure is bypassed if 2(1)-XS-70-48 at 480 RMOV board 2(1)B is in the EMERGENCY position.

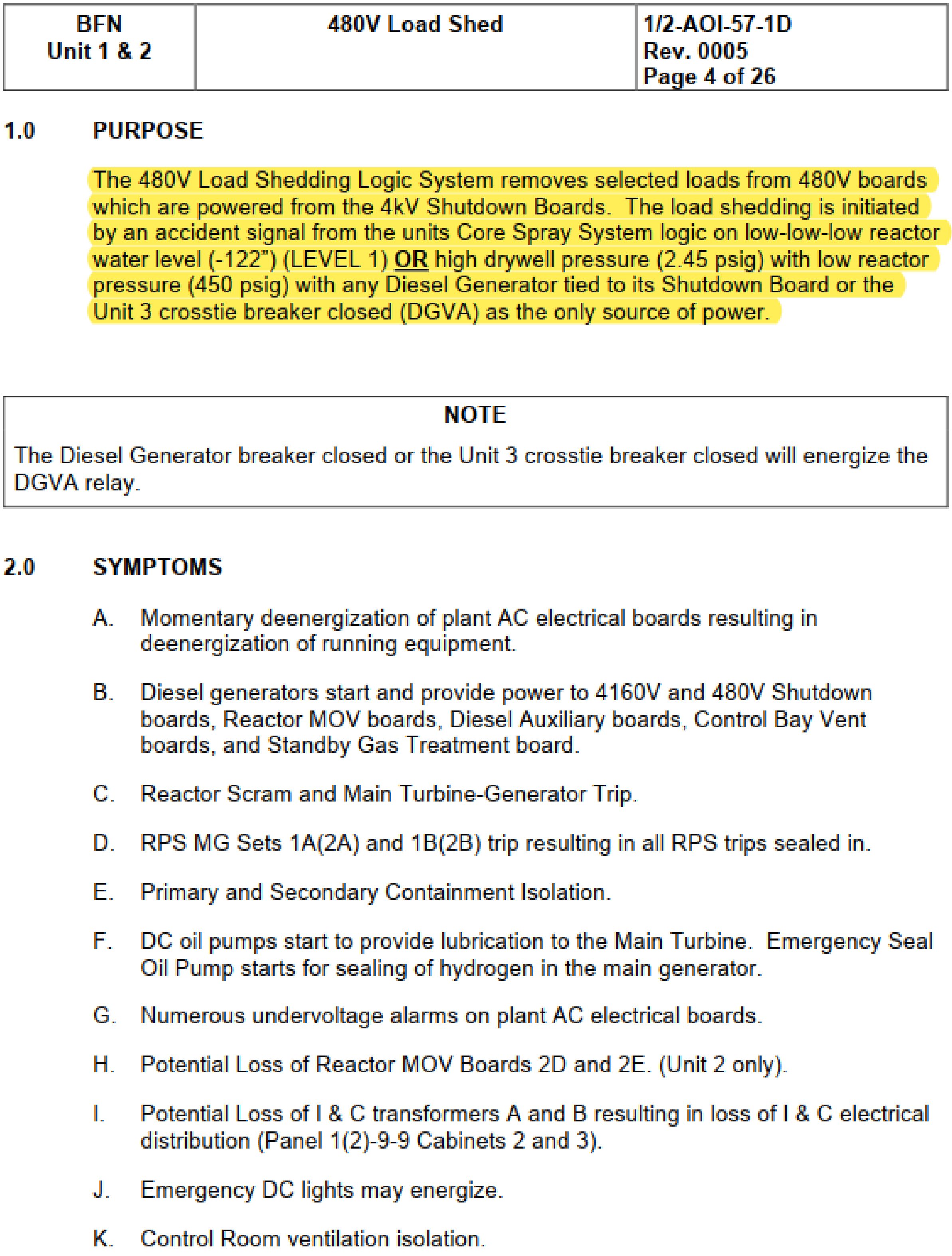
4.0 480V LOAD SHED EFFECT ON DRYWELL BLOWERS:

4.1 Effect on the non-accident unit:

Initially all drywell blowers trip.

Eight Drywell Blowers on the unit without the accident will auto start in a staggered sequence starting at 40 seconds and ending at 90 seconds. Two drywell blowers will have their respective Auto Start Inhibit switches in the INHIBIT position, preventing them from auto starting following a 480V Load Shed. The drywell blowers in INHIBIT can still be manually started.

Excerpts from 1/2-AOI-57-1D:



BFN Unit 1 & 2	**480V Load Shed**	**1/2-AOI-57-1D Rev. 0005 Page 4 of 26**

1.0 PURPOSE

The 480V Load Shedding Logic System removes selected loads from 480V boards which are powered from the 4kV Shutdown Boards. The load shedding is initiated by an accident signal from the units Core Spray System logic on low-low-low reactor water level (-122") (LEVEL 1) **OR** high drywell pressure (2.45 psig) with low reactor pressure (450 psig) with any Diesel Generator tied to its Shutdown Board or the Unit 3 crosstie breaker closed (DGVA) as the only source of power.

NOTE

The Diesel Generator breaker closed or the Unit 3 crosstie breaker closed will energize the DGVA relay.

2.0 SYMPTOMS

A. Momentary deenergization of plant AC electrical boards resulting in deenergization of running equipment.

B. Diesel generators start and provide power to 4160V and 480V Shutdown boards, Reactor MOV boards, Diesel Auxiliary boards, Control Bay Vent boards, and Standby Gas Treatment board.

C. Reactor Scram and Main Turbine-Generator Trip.

D. RPS MG Sets 1A(2A) and 1B(2B) trip resulting in all RPS trips sealed in.

E. Primary and Secondary Containment Isolation.

F. DC oil pumps start to provide lubrication to the Main Turbine. Emergency Seal Oil Pump starts for sealing of hydrogen in the main generator.

G. Numerous undervoltage alarms on plant AC electrical boards.

H. Potential Loss of Reactor MOV Boards 2D and 2E. (Unit 2 only).

I. Potential Loss of I & C transformers A and B resulting in loss of I & C electrical distribution (Panel 1(2)-9-9 Cabinets 2 and 3).

J. Emergency DC lights may energize.

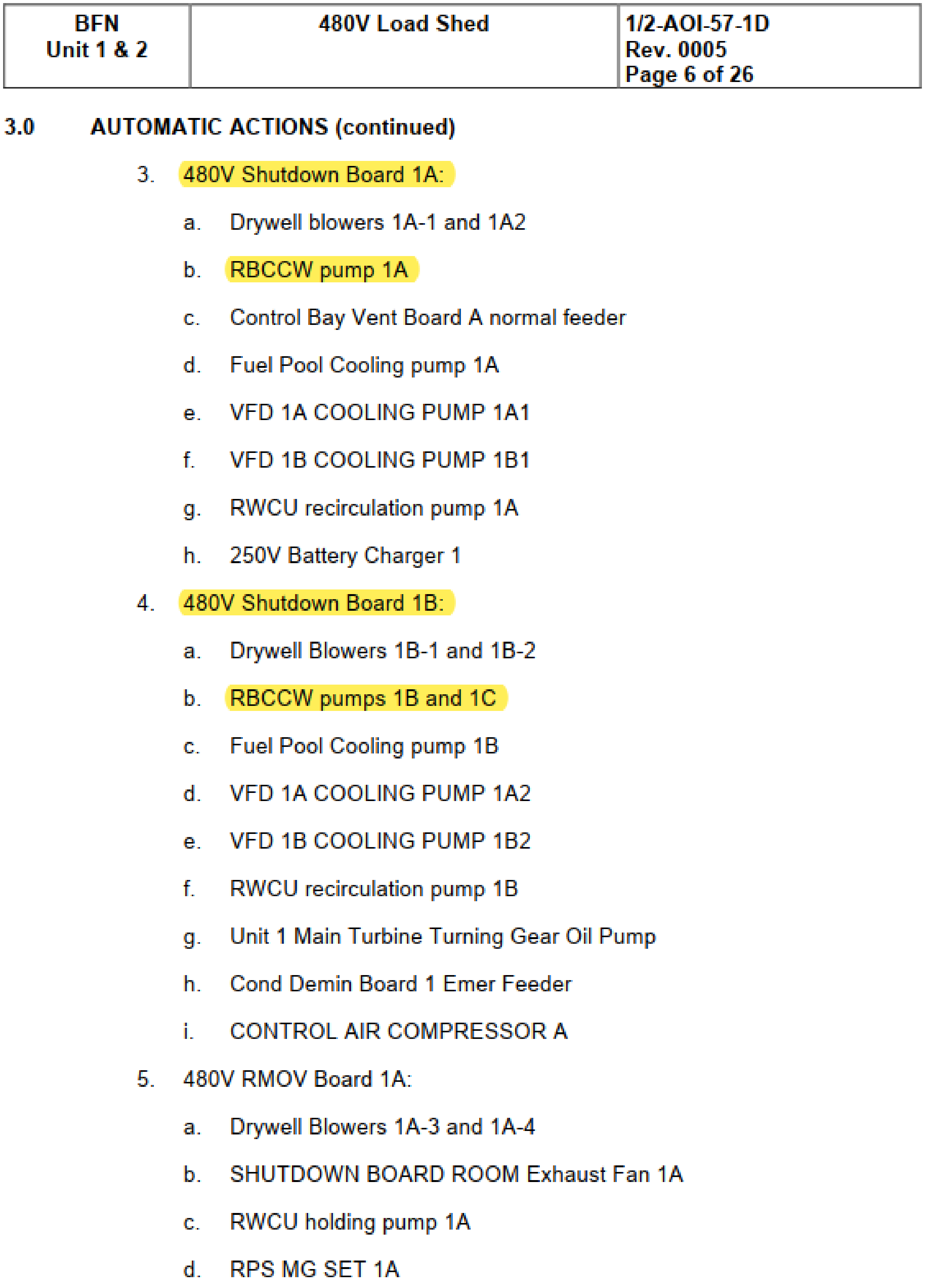
K. Control Room ventilation isolation.

BFN **Unit 1 & 2**	**480V Load Shed**	**1/2-AOI-57-1D** **Rev. 0005** **Page 5 of 26**

3.0 AUTOMATIC ACTIONS

A. The following loads will be shed:

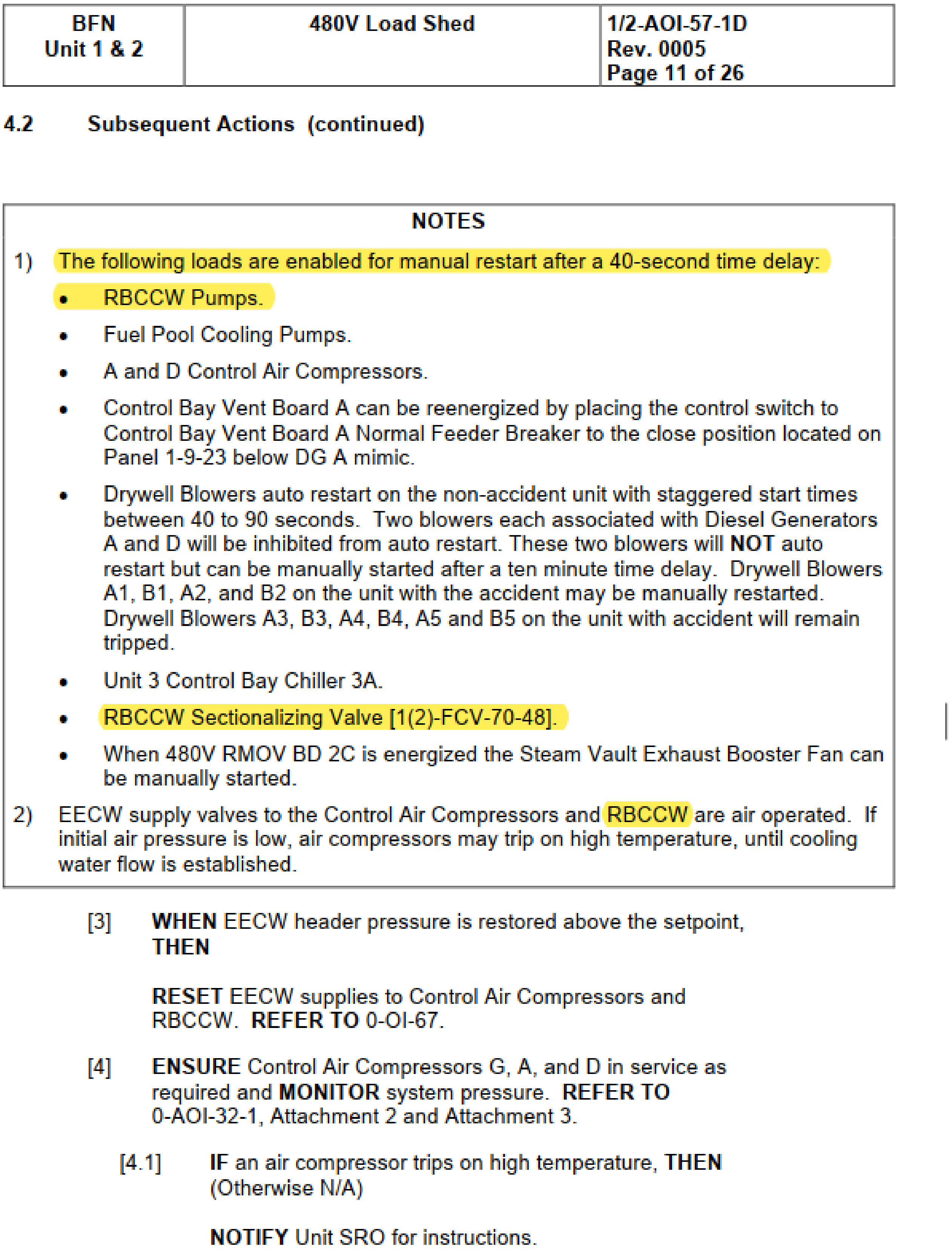
1. 480V Diesel Auxiliary Board A:
   - a. D/G A, B, C and D air compressor right bank
   - b. Standby Gas Treatment Fan A
   - c. Standby Gas Treatment humidity control heater A
   - d. Off Gas Dilution Fans 1A, 2A and 3A
   - e. Diesel Generator A & B room heater
   - f. Standby Gas Treatment Building Heater A
   - g. CO_2 Generator Purging Vaporizer
   - h. Control Bay Chilled Water Pump A
2. 480V Diesel Auxiliary Board B:
   - a. D/G A, B, C and D air compressor left bank
   - b. Standby Gas Treatment Fan B
   - c. Standby Gas Treatment humidity control heater B
   - d. Off Gas Dilution Fans 1B, 2B and 3B
   - e. Stack Sample room heater
   - f. Sewage Treatment Plant
   - g. Diesel Generator Building Power outlet
   - h. Diesel Generator C & D room heater
   - i. Standby Gas Treatment Building heater B
   - j. CO_2 refrigeration unit
   - k. Stack lighting transformer



BFN **Unit 1 & 2**	**480V Load Shed**	**1/2-AOI-57-1D** **Rev. 0005** **Page 6 of 26**

3.0 AUTOMATIC ACTIONS (continued)

3. 480V Shutdown Board 1A:
   - a. Drywell blowers 1A-1 and 1A2
   - b. RBCCW pump 1A
   - c. Control Bay Vent Board A normal feeder
   - d. Fuel Pool Cooling pump 1A
   - e. VFD 1A COOLING PUMP 1A1
   - f. VFD 1B COOLING PUMP 1B1
   - g. RWCU recirculation pump 1A
   - h. 250V Battery Charger 1
4. 480V Shutdown Board 1B:
   - a. Drywell Blowers 1B-1 and 1B-2
   - b. RBCCW pumps 1B and 1C
   - c. Fuel Pool Cooling pump 1B
   - d. VFD 1A COOLING PUMP 1A2
   - e. VFD 1B COOLING PUMP 1B2
   - f. RWCU recirculation pump 1B
   - g. Unit 1 Main Turbine Turning Gear Oil Pump
   - h. Cond Demin Board 1 Emer Feeder
   - i. CONTROL AIR COMPRESSOR A
5. 480V RMOV Board 1A:
   - a. Drywell Blowers 1A-3 and 1A-4
   - b. SHUTDOWN BOARD ROOM Exhaust Fan 1A
   - c. RWCU holding pump 1A
   - d. RPS MG SET 1A



BFN **Unit 1 & 2**	**480V Load Shed**	**1/2-AOI-57-1D** **Rev. 0005** **Page 11 of 26**

4.2 Subsequent Actions (continued)

NOTES

1) The following loads are enabled for manual restart after a 40-second time delay:
   - RBCCW Pumps.
   - Fuel Pool Cooling Pumps.
   - A and D Control Air Compressors.
   - Control Bay Vent Board A can be reenergized by placing the control switch to Control Bay Vent Board A Normal Feeder Breaker to the close position located on Panel 1-9-23 below DG A mimic.
   - Drywell Blowers auto restart on the non-accident unit with staggered start times between 40 to 90 seconds. Two blowers each associated with Diesel Generators A and D will be inhibited from auto restart. These two blowers will **NOT** auto restart but can be manually started after a ten minute time delay. Drywell Blowers A1, B1, A2, and B2 on the unit with the accident may be manually restarted. Drywell Blowers A3, B3, A4, B4, A5 and B5 on the unit with accident will remain tripped.
   - Unit 3 Control Bay Chiller 3A.
   - RBCCW Sectionalizing Valve [1(2)-FCV-70-48].
   - When 480V RMOV BD 2C is energized the Steam Vault Exhaust Booster Fan can be manually started.

2) EECW supply valves to the Control Air Compressors and RBCCW are air operated. If initial air pressure is low, air compressors may trip on high temperature, until cooling water flow is established.

[3] **WHEN** EECW header pressure is restored above the setpoint,
THEN

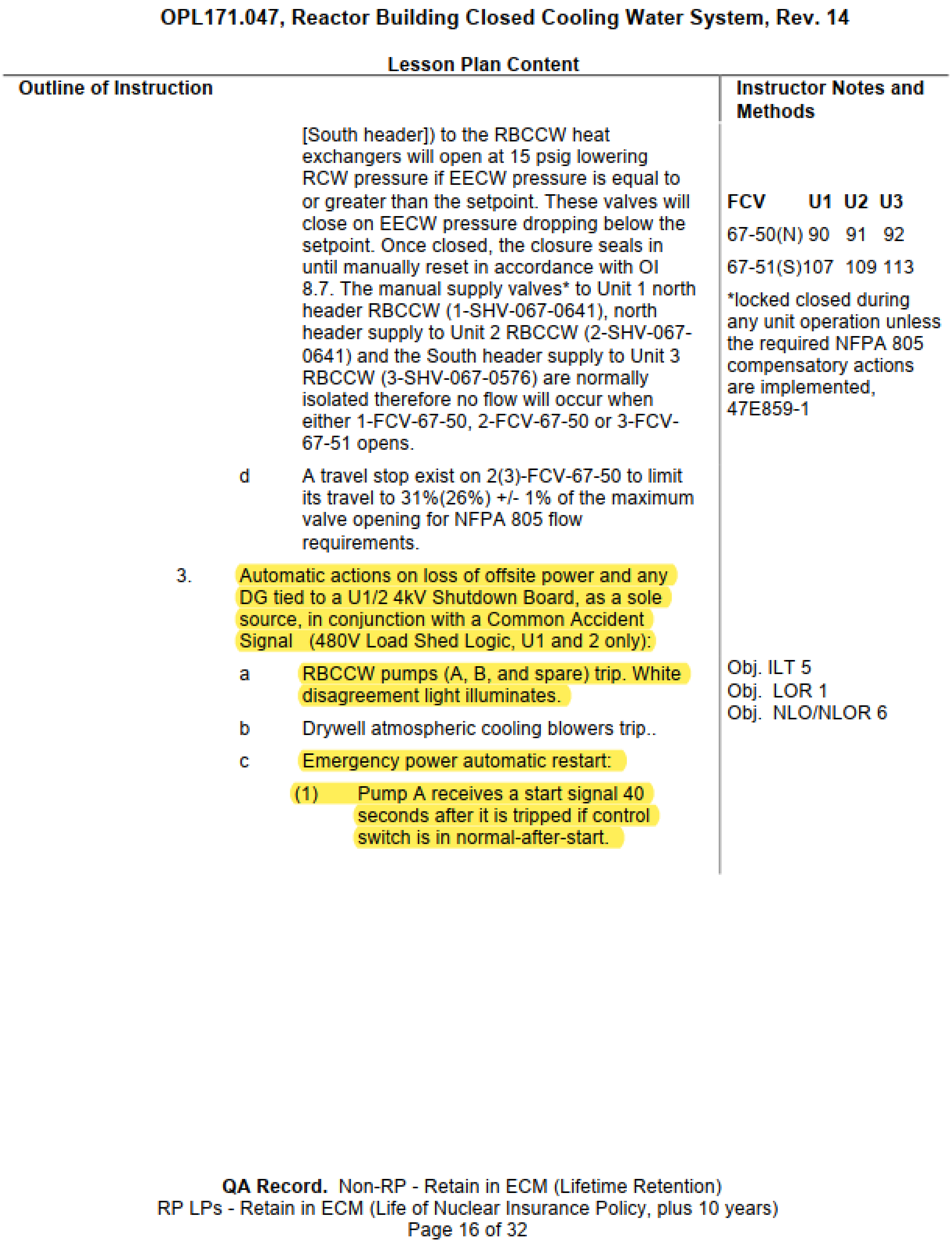
RESET EECW supplies to Control Air Compressors and RBCCW. **REFER TO** 0-OI-67.

[4] **ENSURE** Control Air Compressors G, A, and D in service as required and **MONITOR** system pressure. **REFER TO** 0-AOI-32-1, Attachment 2 and Attachment 3.

[4.1] **IF** an air compressor trips on high temperature, **THEN** (Otherwise N/A)

NOTIFY Unit SRO for instructions.

Excerpt from RBCCW Lesson Plan:



OPL171.047, Reactor Building Closed Cooling Water System, Rev. 14

Lesson Plan Content

Outline of Instruction	**Instructor Notes and Methods**
[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 OI 8.7. The manual supply valves* to Unit 1 north header RBCCW (1-SHV-067-0641), north header supply to Unit 2 RBCCW (2-SHV-067-0641) and the South header supply to Unit 3 RBCCW (3-SHV-067-0576) are normally isolated therefore no flow will occur when either 1-FCV-67-50, 2-FCV-67-50 or 3-FCV-67-51 opens.	**FCV U1 U2 U3** 67-50(N) 90 91 92 67-51(S)107 109 113 *locked closed during any unit operation unless the required NFPA 805 compensatory actions are implemented, 47E859-1
d A travel stop exist on 2(3)-FCV-67-50 to limit its travel to 31%(26%) +/- 1% of the maximum valve opening for NFPA 805 flow requirements.	
3. Automatic actions on loss of offsite power and any DG tied to a U1/2 4kV Shutdown Board, as a sole source, in conjunction with a Common Accident Signal (480V Load Shed Logic, U1 and 2 only):	
a RBCCW pumps (A, B, and spare) trip. White disagreement light illuminates.	Obj. ILT 5 Obj. LOR 1 Obj. NLO/NLOR 6
b Drywell atmospheric cooling blowers trip..	
c Emergency power automatic restart:	
(1) Pump A receives a start signal 40 seconds after it is tripped if control switch is in normal-after-start.	

OPL171.047, Reactor Building Closed Cooling Water System, Rev. 14

Lesson Plan Content

Outline of Instruction	**Instructor Notes and Methods**
(2) Pump B receives a start signal 3 seconds after the Pump A start signal if Pump A fails to start and B control switch is in normal-after-start. Note: Spare pump has no auto restart	
(3) Drywell atmosphere cooling coil blowers will auto restart on the non-accident unit after 40 seconds.	
(4) The accident unit's AI, A2, B1 and B2 blowers can be manually restarted after 40 seconds.	
(5) The other blowers on the accident unit are locked out until the signal clears.	
d The non-essential loop isolation valve, FCV-70-48 closes when power is restored. On loss of power to a 480V Shutdown Board with no Common Accident Signal , the RBCCW pumps "coast" to a stop but the supply breaker does not trip. They will restart when power is restored, and there is no other effect on system operation (other than FCV-70-48 closing on low header pressure). A Common Accident Signal with no loss of power does not affect system operation at all.	Obj. ILT 6 Obj. LOR 2 Obj. NLO/NLOR 7
4. Unit 3 Load Shed	Obj. ILT 5, LOR 1
a U3 480V load shed is divisionalized; such that if either of the following conditions exists with U3 Common Accident Signal present :	Obj. NLO/NLOR 8
(1) Division 1 480V load shedding: 3EA D/G or the U-1/2 crosstie breaker is the only source of power to 3EA 4KV S/D Bd (DGVA-A) and 3A 480V S/D Bd normal feeder Bkr is closed	U3, FCV-70-48 does not automatically close from load shed logic.
OR	

Examination Outline Cross-reference:

300000 (SF8 IA) Instrument Air

A3.04 (10CFR 55.41.8)

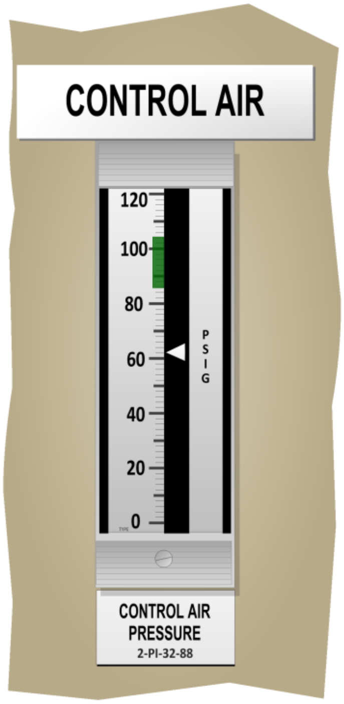
Ability to monitor automatic operation of the Instrument Air System, including:

- Automatic Isolation

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	300000A3.04	
Importance Rating	3.4	------

Proposed Question: **# 50**

Unit 2 is operating at 100% RTP when a Control Air leak develops, resulting in the following indication:



Given the indication, which **ONE** of the following identifies the correct plant status in accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors?

A. 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE, is **CLOSED**

B. 2-PCV-032-3901, CONTROL AIR CROSSTIE, is **CLOSED**

C. 2-FCV-2-130, CONDENSATE DEMIN BYP VALVE, is **OPEN**

D. OUTBOARD MAIN STEAM ISOLATION VALVES are **CLOSED**

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that this valve has both an automatic opening and closing function associated with Control Air Pressure. 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE opens when Control Air Pressure lowers to 85 psig as a backup supply and fails closed at a Control Air Pressure of 30 psig.

B **CORRECT**: (*See attached*) In accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors, the Unit 2 to Unit 3 Crosstie Valve, 2-PCV-032-3901, CONTROL AIR CROSSTIE, closes when Control Air Pressure is equal to or less than 65 psig and will **not** re-open until Control Air Pressure is 85 psig.

C INCORRECT: Incorrect but plausible in that 2-FCV-2-130, CONDENSATE DEMIN BYP VALVE fails open when Control Air Pressure is less than 50 psig, however that setpoint has not yet been reached.

D INCORRECT: Incorrect but plausible in that if the loss of Control Air is instantaneous, when pressure drops to < 45 psig, the Main Steam Isolation Valve accumulator air will be routed to close the outboard MSIVs.

RO Level Justification: Tests the candidate's ability to diagnose alarms and readings given for a Loss of Control Air and determine the operational effect on system pneumatically operated valves. 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.

Technical Reference(s): 0-OI-32, Rev. 143 (Attach if not previously provided)
2-ARP-9-5B, Rev. 33
0-AOI-32-1, Rev. 56
2-AOI-32-2, Rev. 39

Proposed references to be provided to applicants during examination: **Panel 2-9-20, 2-PI-32-88, Control Air Pressure**

Learning Objective: OPL171.54, Obj. 8 (As available)

Question Source:	Bank #	BFN 1909 NRC #57	
	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**

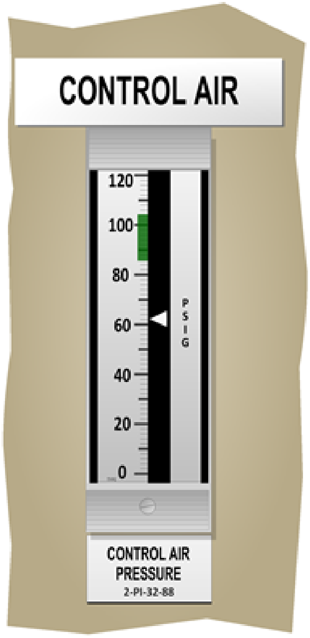
10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

Proposed Question: **# 57**

Unit 2 is operating at 100% RTP when a Control Air leak develops, resulting in the following indication:



Given the indication, which **ONE** of the following identifies the correct plant status in accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors?

A. 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE, is CLOSED

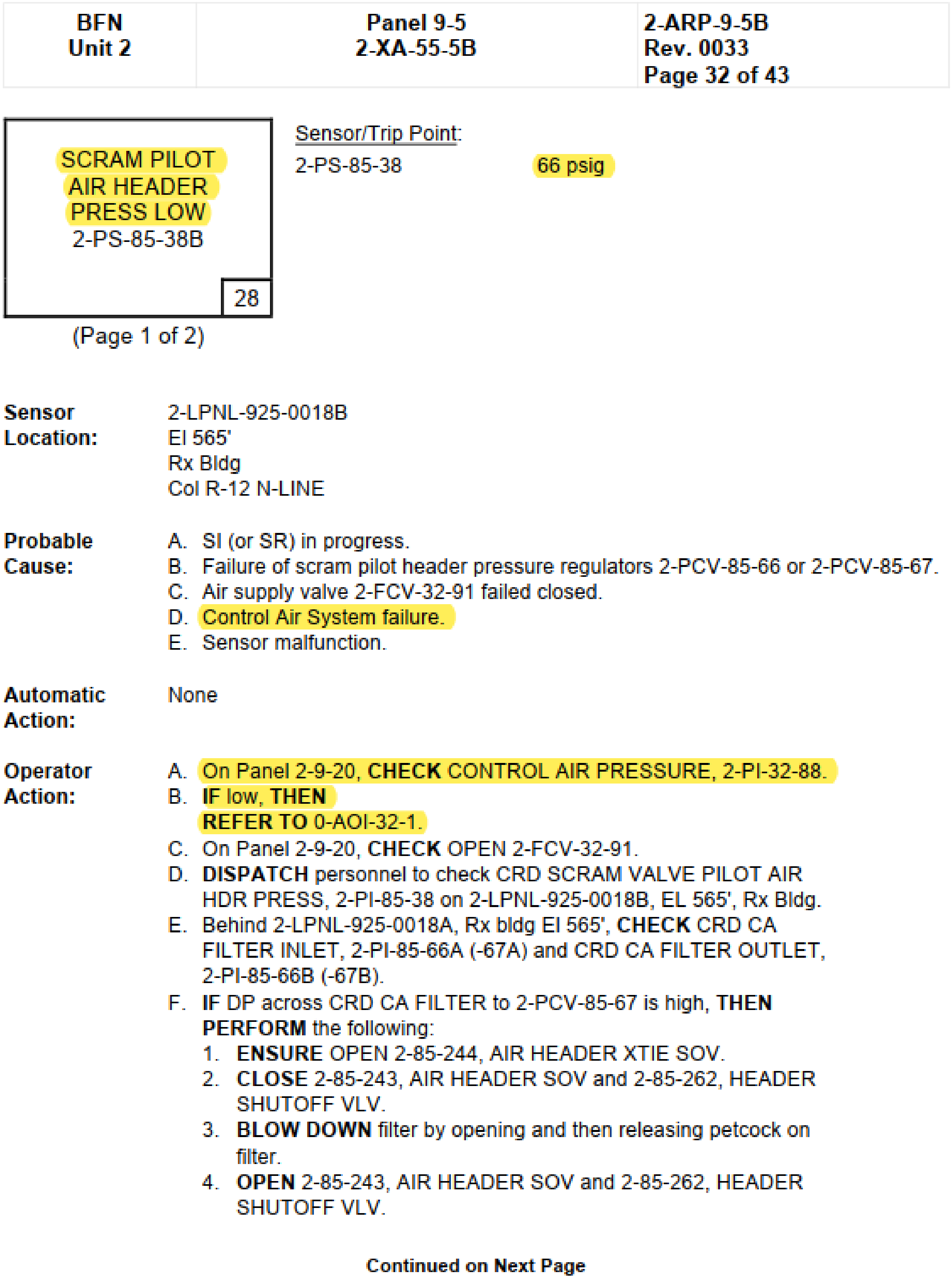
B. 2-PCV-032-3901, CONTROL AIR CROSSTIE, is CLOSED

C. 2-FCV-2-130, CONDENSATE DEMIN BYP VALVE, is OPEN

D. OUTBOARD MAIN STEAM ISOLATION VALVES are CLOSED

Proposed Answer: **B**

Excerpt from 2-ARP-9-5B:



BFN **Unit 2**	**Panel 9-5** **2-XA-55-5B**	**2-ARP-9-5B** **Rev. 0033** **Page 32 of 43**

SCRAM PILOT
AIR HEADER
PRESS LOW
2-PS-85-38B
28

(Page 1 of 2)

Sensor/Trip Point:

2-PS-85-38	66 psig

Sensor Location:
2-LPNL-925-0018B
El 565'
Rx Bldg
Col R-12 N-LINE

Probable Cause:

A. SI (or SR) in progress.
B. Failure of scram pilot header pressure regulators 2-PCV-85-66 or 2-PCV-85-67.
C. Air supply valve 2-FCV-32-91 failed closed.
D. Control Air System failure.
E. Sensor malfunction.

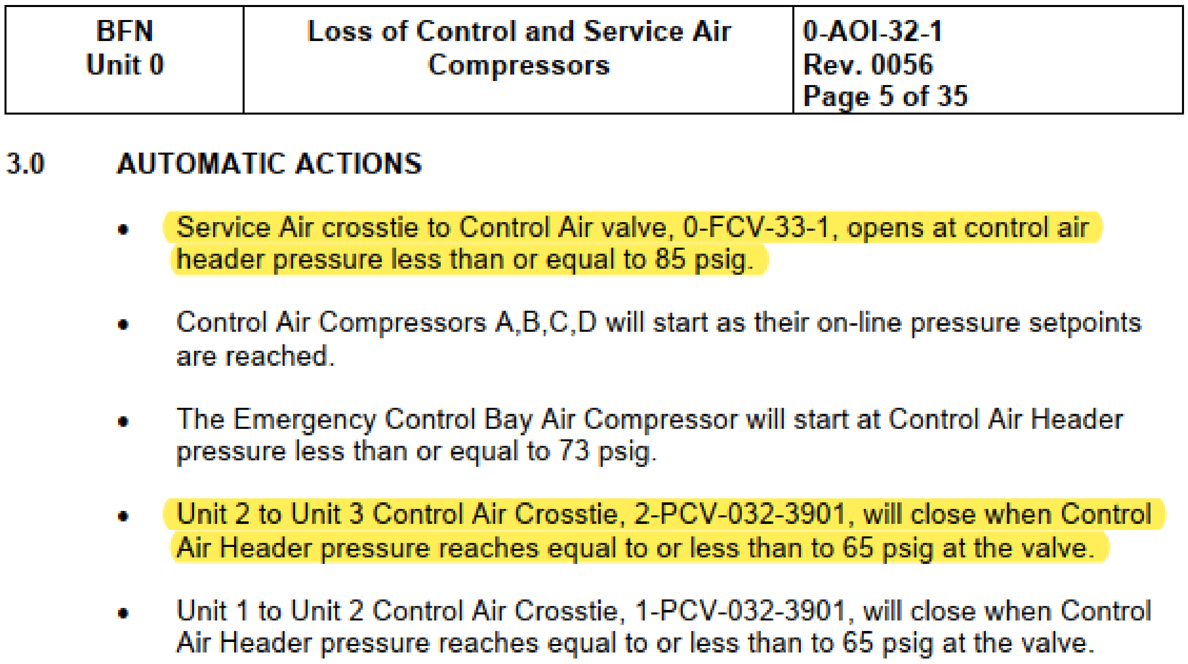
Automatic Action: None

Operator Action:

A. On Panel 2-9-20, **CHECK** CONTROL AIR PRESSURE, 2-PI-32-88.
B. **IF** low, **THEN**
REFER TO 0-AOI-32-1.
C. On Panel 2-9-20, **CHECK** OPEN 2-FCV-32-91.
D. **DISPATCH** personnel to check CRD SCRAM VALVE PILOT AIR HDR PRESS, 2-PI-85-38 on 2-LPNL-925-0018B, EL 565', Rx Bldg.
E. Behind 2-LPNL-925-0018A, Rx bldg El 565', **CHECK** CRD CA FILTER INLET, 2-PI-85-66A (-67A) and CRD CA FILTER OUTLET, 2-PI-85-66B (-67B).
F. **IF** DP across CRD CA FILTER to 2-PCV-85-67 is high, **THEN PERFORM** the following:
    1. **ENSURE** OPEN 2-85-244, AIR HEADER XTIE SOV.
    2. **CLOSE** 2-85-243, AIR HEADER SOV and 2-85-262, HEADER SHUTOFF VLV.
    3. **BLOW DOWN** filter by opening and then releasing petcock on filter.
    4. **OPEN** 2-85-243, AIR HEADER SOV and 2-85-262, HEADER SHUTOFF VLV.

Continued on Next Page

Excerpt from 0-AOI-32-1:



BFN **Unit 0**	**Loss of Control and Service Air Compressors**	**0-AOI-32-1** **Rev. 0056** **Page 5 of 35**

3.0 AUTOMATIC ACTIONS

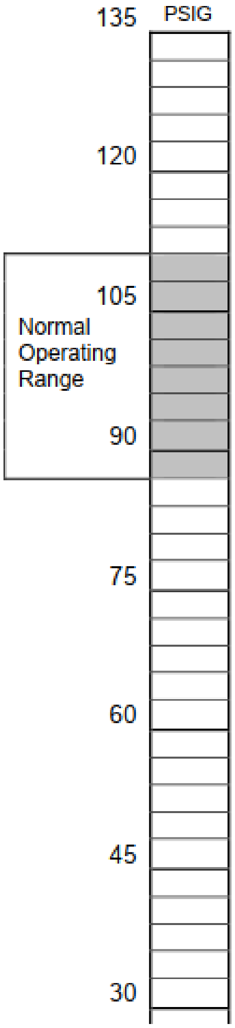
- Service Air crosstie to Control Air valve, 0-FCV-33-1, opens at control air header pressure less than or equal to 85 psig.
- Control Air Compressors A,B,C,D will start as their on-line pressure setpoints are reached.
- The Emergency Control Bay Air Compressor will start at Control Air Header pressure 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.

Excerpt from 0-OI-32:

BFN **Unit 0**	**Control Air System**	**0-OI-32** **Rev. 0143** **Page 73 of 117**

Attachment 1
(Page 1 of 1)

Control Air System Pressure Spectrum



↑ 132 Compressors A, B C, D Discharge Relief Valves lift.

↑ 115 Control Air Receiver Relief Valves lift, Compressor G Discharge Relief Valve lifts.

↓ 98 Compressors A, B C, D in LEAD loads (UNLOAD @ 108 psig)
↓ 96 Compressors A, B C, D in first LAG loads (UNLOAD @ 106 psig)
↓ 94 Compressors A, B C, D in second LAG loads (UNLOAD @ 104 psig)

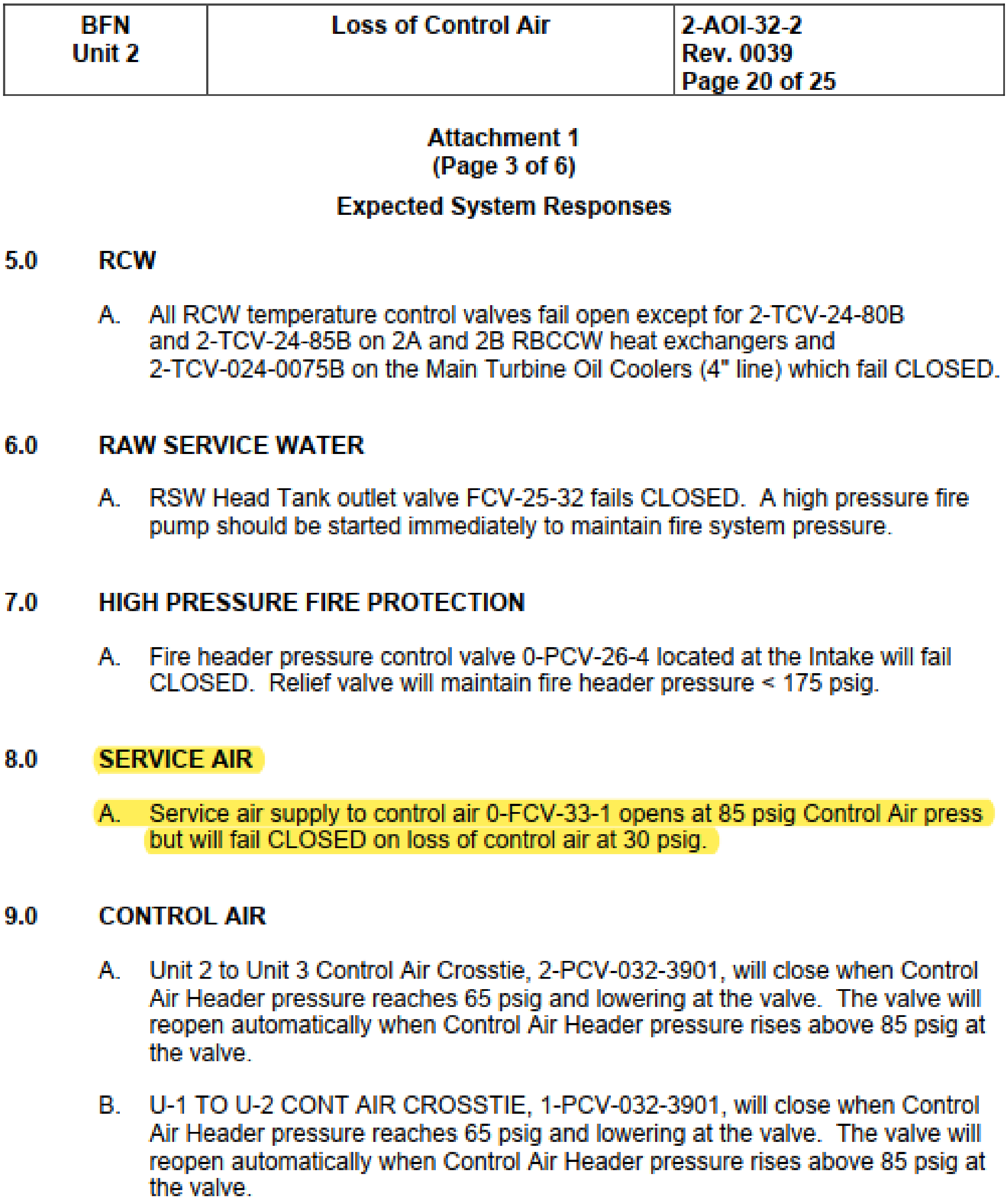
↓ 85 Service Air Crosstie, 0-PCV-33-1, opens to backup Control Air System
↓ 82 MS Relief Valve Air Accumulator Pressure Low annunciation PNL 9-3D

↓ 75 U2 CAD/CA Flow Select 2-PCV-84-654 aligns Nitrogen from CAD Tank A to FSV-64-20 & 21
↓ 73 Control Bay Emergency Compressor starts.
↓ 70 Control Air Pressure Low annunciation(PNL 9-20B and 0-LPNL-25-0118A)
↓ 66 Scram Pilot Air Header Pressure Low annunciation (PNL 9-5B)
↓ 65 Unit Crosstie, PCV-032-3901, closes (reopens at 85 psig).
Startup Bypass control Valve, FCV-3-53, Fails as is.
The valve cannot be controlled by PDS until pressure returns to >65psig.

↓ 50 Condensate Demin Bypass FCV-2-130, fails open

↓ 30 If open, Service Air Crosstie, 0-PCV-33-1 closes

Excerpts from 2-AOI-32-2: Supports Distractor A



BFN **Unit 2**	**Loss of Control Air**	**2-AOI-32-2** **Rev. 0039** **Page 20 of 25**

Attachment 1
(Page 3 of 6)

Expected System Responses

5.0 RCW

A. All RCW temperature control valves fail open except for 2-TCV-24-80B and 2-TCV-24-85B on 2A and 2B RBCCW heat exchangers and 2-TCV-024-0075B on the Main Turbine Oil Coolers (4" line) which fail CLOSED.

6.0 RAW SERVICE WATER

A. RSW Head Tank outlet valve FCV-25-32 fails CLOSED. A high pressure fire pump should be started immediately to maintain fire system pressure.

7.0 HIGH PRESSURE FIRE PROTECTION

A. Fire header pressure control valve 0-PCV-26-4 located at the Intake will fail CLOSED. Relief valve will maintain fire header pressure < 175 psig.

8.0 SERVICE AIR

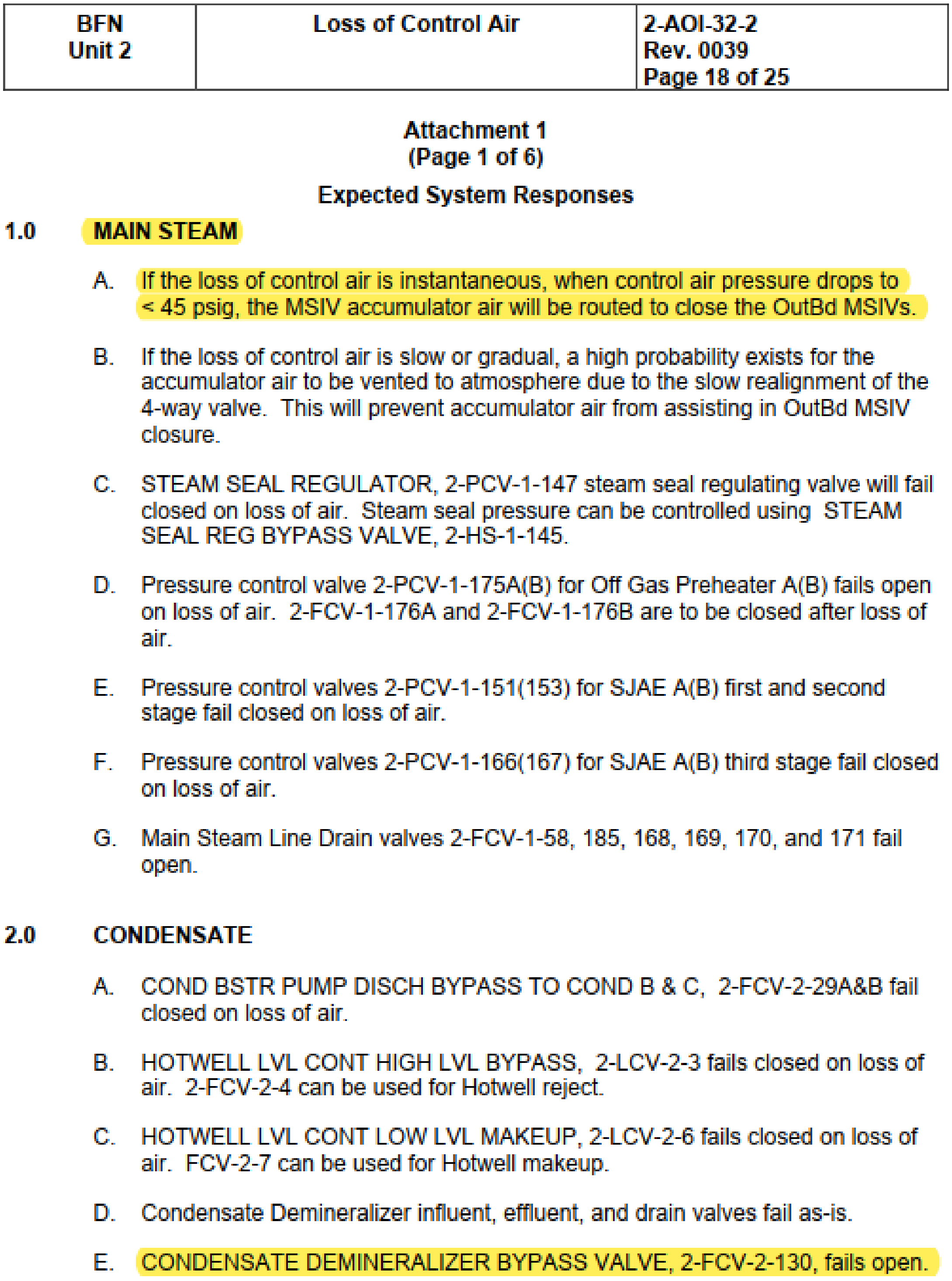
A. Service air supply to control air 0-FCV-33-1 opens at 85 psig Control Air press but will fail CLOSED on loss of control air at 30 psig.

9.0 CONTROL AIR

A. Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901, will close when Control Air Header pressure reaches 65 psig and lowering at the valve. The valve will reopen automatically when Control Air Header pressure rises above 85 psig at the valve.

B. U-1 TO U-2 CONT AIR CROSSTIE, 1-PCV-032-3901, will close when Control Air Header pressure reaches 65 psig and lowering at the valve. The valve will reopen automatically when Control Air Header pressure rises above 85 psig at the valve.

Supports Distractors C and D:



BFN **Unit 2**	**Loss of Control Air**	**2-AOI-32-2** **Rev. 0039** **Page 18 of 25**

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.

B. If the loss of control air is slow or gradual, a high probability exists for the accumulator air to be vented to atmosphere due to the slow realignment of the 4-way valve. This will prevent accumulator air from assisting in OutBd MSIV closure.

C. STEAM SEAL REGULATOR, 2-PCV-1-147 steam seal regulating valve will fail closed on loss of air. Steam seal pressure can be controlled using STEAM SEAL REG BYPASS VALVE, 2-HS-1-145.

D. Pressure control valve 2-PCV-1-175A(B) for Off Gas Preheater A(B) fails open on loss of air. 2-FCV-1-176A and 2-FCV-1-176B are to be closed after loss of air.

E. Pressure control valves 2-PCV-1-151(153) for SJAE A(B) first and second stage fail closed on loss of air.

F. Pressure control valves 2-PCV-1-166(167) for SJAE A(B) third stage fail closed on loss of air.

G. Main Steam Line Drain valves 2-FCV-1-58, 185, 168, 169, 170, and 171 fail open.

2.0 CONDENSATE

A. COND BSTR PUMP DISCH BYPASS TO COND B & C, 2-FCV-2-29A&B fail closed on loss of air.

B. HOTWELL LVL CONT HIGH LVL BYPASS, 2-LCV-2-3 fails closed on loss of air. 2-FCV-2-4 can be used for Hotwell reject.

C. HOTWELL LVL CONT LOW LVL MAKEUP, 2-LCV-2-6 fails closed on loss of air. FCV-2-7 can be used for Hotwell makeup.

D. Condensate Demineralizer influent, effluent, and drain valves fail as-is.

E. CONDENSATE DEMINERALIZER BYPASS VALVE, 2-FCV-2-130, fails open.

Examination Outline Cross-reference:

510000 (SF4 SWS*) Service Water (Normal and Emergency)

G2.1.2(10CFR 55.41.10)

Knowledge of operator responsibilities during any mode of plant operation.

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	510000G2.1.2	
Importance Rating	4.1	------

Proposed Question: **# 51**

In accordance with OPDP-1, Conduct of Operations, a plant announcement ___ **(1)** ___ required prior to starting A2 RHRSW Pump.

In accordance with 0-OI-23, Residual Heat Removal Service Water System, after starting A2 RHRSW Pump, a minimum flow of ___ **(2)** ___ is required to prevent pump damage.

A. (1) is
(2) 1350 gpm

B. (1) is
(2) 1700 gpm

C. (1) is NOT
(2) 1350 gpm

D. (1) is NOT
(2) 1700 gpm

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that the required minimum flow was different for RHRSW Pumps A1, B1, and D1, but all 12 RHRSW Pumps were all recently upgraded. The minimum flow limit for three of the old pumps was 1350 gpm.

B **CORRECT**: (*See attached*) In accordance with OPDP-1, Conduct of Operations, plant announcements shall be made before changing the status of any major equipment such as starting or stopping pumps. For second part, in accordance with 0-OI-23, Residual Heat Removal Service Water System, the minimum flow for RHRSW Pumps is 1700 gpm.

C INCORRECT: First part is incorrect but plausible in that RHRSW Pumps are located in their own building near the intake structure, and it is plausible that a plant announcement for starting equipment that is not located in the Reactor or Turbine Buildings would not be required. 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 operator responsibilities during RHRSW Pump operation to prevent personnel injury and equipment damage. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): 0-OI-23, Rev. 109 and Rev. 105 (Attach if not previously provided)

OPDP-1, Rev. 53

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.046, Obj. 5a, 5b (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**

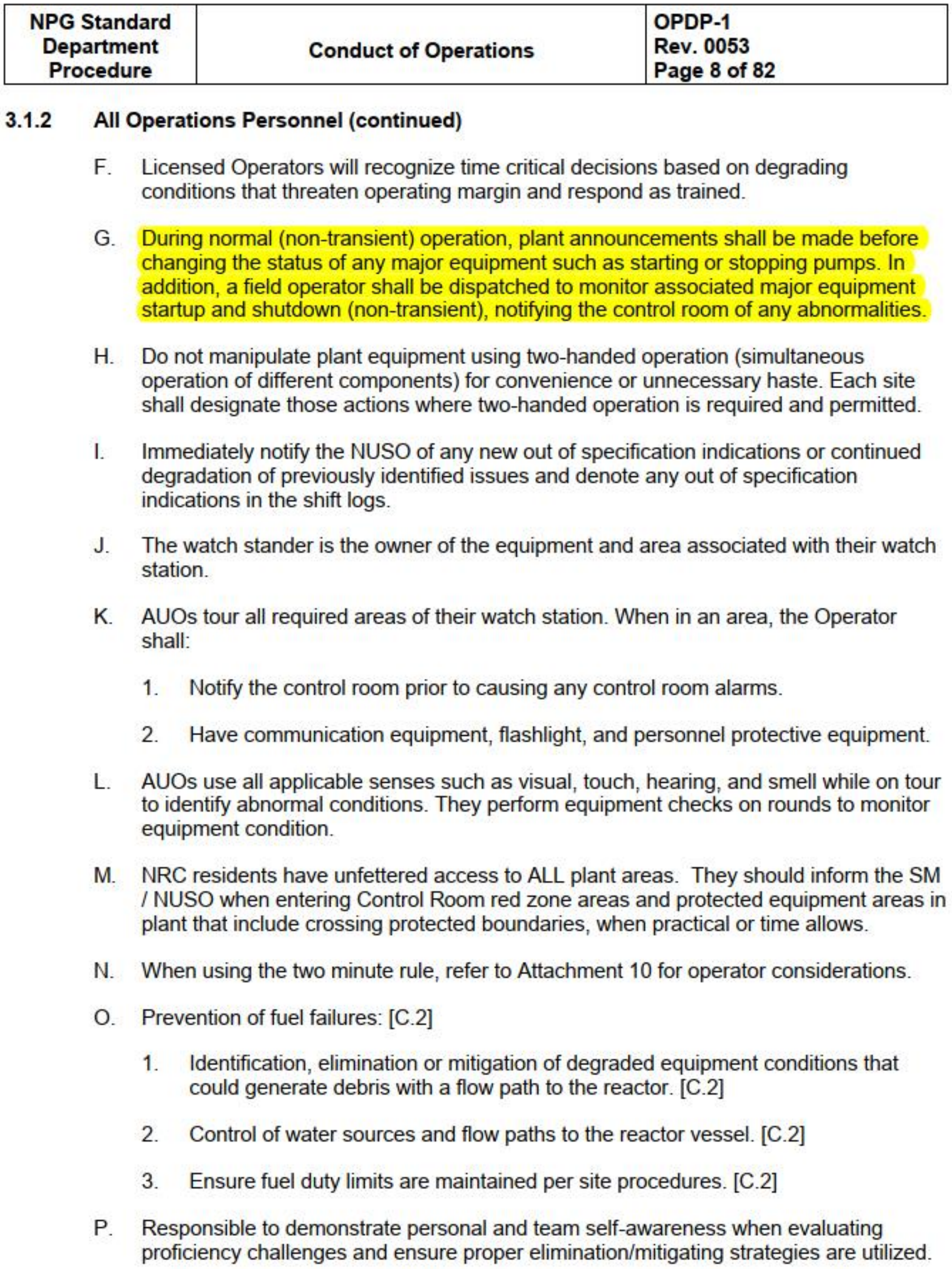
Comprehension or Analysis

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

55.43

Comments:

Excerpt from OPDP-1:



NPG Standard Department Procedure	**Conduct of Operations**	**OPDP-1** **Rev. 0053** **Page 8 of 82**

3.1.2 All Operations Personnel (continued)

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.

H. Do not manipulate plant equipment using two-handed operation (simultaneous operation of different components) for convenience or unnecessary haste. Each site shall designate those actions where two-handed operation is required and permitted.

I. Immediately notify the NUSO of any new out of specification indications or continued degradation of previously identified issues and denote any out of specification indications in the shift logs.

J. The watch stander is the owner of the equipment and area associated with their watch station.

K. AUOs tour all required areas of their watch station. When in an area, the Operator shall:

1. Notify the control room prior to causing any control room alarms.
2. Have communication equipment, flashlight, and personnel protective equipment.

L. AUOs use all applicable senses such as visual, touch, hearing, and smell while on tour to identify abnormal conditions. They perform equipment checks on rounds to monitor equipment condition.

M. NRC residents have unfettered access to ALL plant areas. They should inform the SM / NUSO when entering Control Room red zone areas and protected equipment areas in plant that include crossing protected boundaries, when practical or time allows.

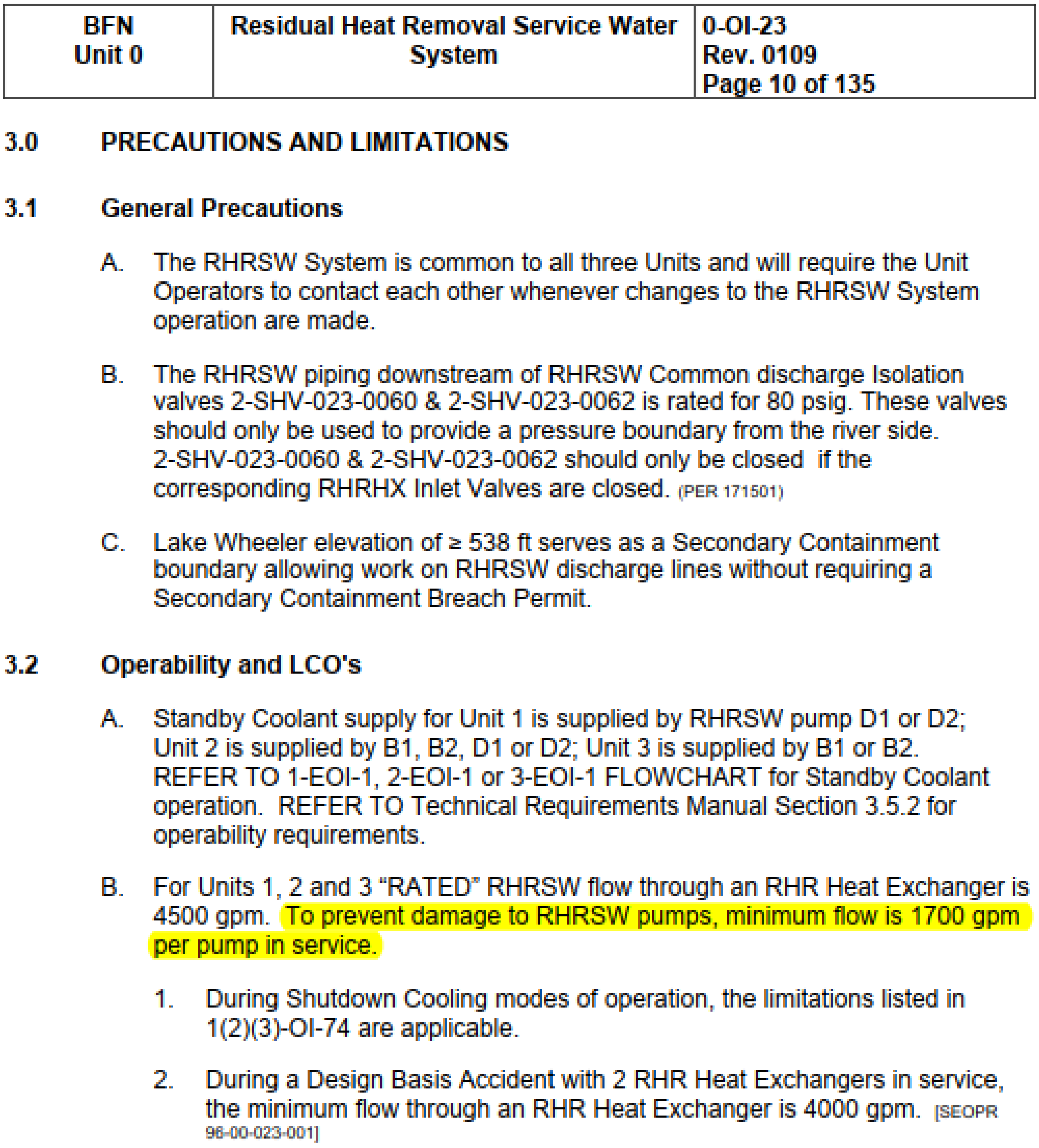
N. When using the two minute rule, refer to Attachment 10 for operator considerations.

O. Prevention of fuel failures: [C.2]

1. Identification, elimination or mitigation of degraded equipment conditions that could generate debris with a flow path to the reactor. [C.2]
2. Control of water sources and flow paths to the reactor vessel. [C.2]
3. Ensure fuel duty limits are maintained per site procedures. [C.2]

P. Responsible to demonstrate personal and team self-awareness when evaluating proficiency challenges and ensure proper elimination/mitigating strategies are utilized.

Excerpt from 0-OI-23: Current revision



BFN **Unit 0**	**Residual Heat Removal Service Water System**	**0-OI-23** **Rev. 0109** **Page 10 of 135**

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions

A. The RHRSW System is common to all three Units and will require the Unit Operators to contact each other whenever changes to the RHRSW System operation are made.

B. The RHRSW piping downstream of RHRSW Common discharge Isolation valves 2-SHV-023-0060 & 2-SHV-023-0062 is rated for 80 psig. These valves should only be used to provide a pressure boundary from the river side. 2-SHV-023-0060 & 2-SHV-023-0062 should only be closed if the corresponding RHRHX Inlet Valves are closed. (PER 171501)

C. Lake Wheeler elevation of ≥ 538 ft serves as a Secondary Containment boundary allowing work on RHRSW discharge lines without requiring a Secondary Containment Breach Permit.

3.2 Operability and LCO's

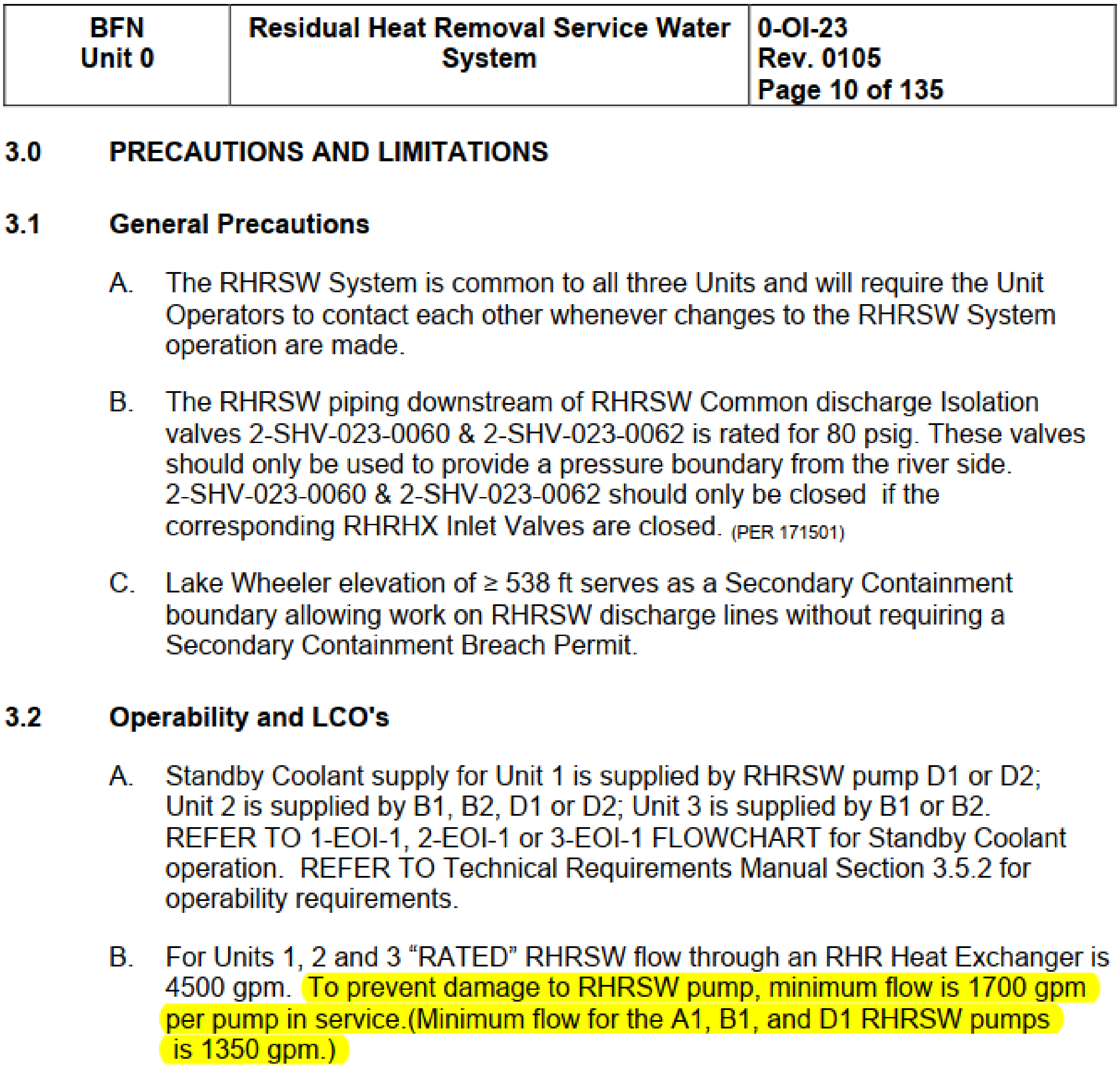
A. Standby Coolant supply for Unit 1 is supplied by RHRSW pump D1 or D2; Unit 2 is supplied by B1, B2, D1 or D2; Unit 3 is supplied by B1 or B2. REFER TO 1-EOI-1, 2-EOI-1 or 3-EOI-1 FLOWCHART for Standby Coolant operation. REFER TO Technical Requirements Manual Section 3.5.2 for operability requirements.

B. For Units 1, 2 and 3 "RATED" RHRSW flow through an RHR Heat Exchanger is 4500 gpm. To prevent damage to RHRSW pumps, minimum flow is 1700 gpm per pump in service.

   1. During Shutdown Cooling modes of operation, the limitations listed in 1(2)(3)-OI-74 are applicable.

   2. During a Design Basis Accident with 2 RHR Heat Exchangers in service, the minimum flow through an RHR Heat Exchanger is 4000 gpm. [SEOPR 96-00-023-001]

Excerpt from 0-OI-23 (old revision (rev 105)): Supports Distractors A(2), C(2)



BFN Unit 0	**Residual Heat Removal Service Water System**	**0-OI-23 Rev. 0105 Page 10 of 135**

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions

A. The RHRSW System is common to all three Units and will require the Unit Operators to contact each other whenever changes to the RHRSW System operation are made.

B. The RHRSW piping downstream of RHRSW Common discharge Isolation valves 2-SHV-023-0060 & 2-SHV-023-0062 is rated for 80 psig. These valves should only be used to provide a pressure boundary from the river side. 2-SHV-023-0060 & 2-SHV-023-0062 should only be closed if the corresponding RHRHX Inlet Valves are closed. (PER 171501)

C. Lake Wheeler elevation of ≥ 538 ft serves as a Secondary Containment boundary allowing work on RHRSW discharge lines without requiring a Secondary Containment Breach Permit.

3.2 Operability and LCO's

A. Standby Coolant supply for Unit 1 is supplied by RHRSW pump D1 or D2; Unit 2 is supplied by B1, B2, D1 or D2; Unit 3 is supplied by B1 or B2. REFER TO 1-EOI-1, 2-EOI-1 or 3-EOI-1 FLOWCHART for Standby Coolant operation. REFER TO Technical Requirements Manual Section 3.5.2 for operability requirements.

B. For Units 1, 2 and 3 “RATED” RHRSW flow through an RHR Heat Exchanger is 4500 gpm. To prevent damage to RHRSW pump, minimum flow is 1700 gpm per pump in service.(Minimum flow for the A1, B1, and D1 RHRSW pumps is 1350 gpm.)

Examination Outline Cross-reference:

214000 (SF7 RPIS) Rod Position Information

K6.01 (10CFR 55.41.7)

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Rod Position Information System:

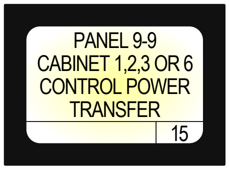
- RPIS Power Supply

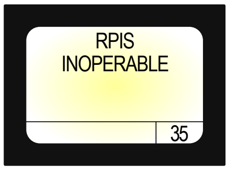
Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	214000K6.01	
Importance Rating	3.2	------

Proposed Question: **# 52**

Unit 2 is operating at 100% RTP, when the following conditions occur:

- **ALL** Control Rod positions on Panel 2-9-5 are lost
- PANEL 2-9-9 CABINET 1, 2, 3, or 6 CONTROL POWER TRANSFER (2-9-7A, Window 15) alarms
- RPIS INOPERABLE (2-9-5A, Window 35) alarms





Given the conditions above, the power supply to **(1)** has tripped.

During the failure, movement of Control Rods can be accomplished by **(2)**.

A. (1) Unit Preferred, Panel 2-9-9 Cabinet 6
(2) SCRAM **ONLY**

B. (1) Unit Preferred, Panel 2-9-9 Cabinet 6
(2) SCRAM **OR** 2-HS-85-48, CRD CONTROL SWITCH

C. (1) I & C ‘A’, Panel 2-9-9 Cabinet 2
(2) SCRAM **ONLY**

D. (1) I & C ‘A’, Panel 2-9-9 Cabinet 2
(2) SCRAM **OR** 2-HS-85-48, CRD CONTROL SWITCH

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with the given alarms, if both I & C ‘A’ and Unit Preferred were lost, PANEL 2-9-9 CABINET 1, 2, 3, or 6 CONTROL POWER TRANSFER (2-9-7A, Window 15) would alarm. However, RPIS INOPERABLE (2-9-5A, Window 35) would alarm from a loss of Unit Preferred only. For second part, as referenced in the given (2-9-5A, Window 35) and in accordance with 2-AOI-57-4, Loss of Unit Preferred, all Control Rod select relays are de-energized following the loss of Unit Preferred. With the loss of Control Rod position indication on Panel 2-9-5, the only method of Control Rod movement is to manually SCRAM the Reactor.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible in that with the two given alarms being received at the same time as a result of a singular event, indicate a loss of Unit Preferred. Following a loss of Unit Preferred, Control Rods cannot be selected and with an inability to select, Control Rods then cannot be moved by the CRD CONTROL SWITCH.

C INCORRECT: First part is incorrect, but plausible in that there are numerous 120V power supplies that bring in numerous alarms. Some of those alarms are shared. The given alarm, PANEL 2-9-9 CABINET 1, 2, 3, or 6 CONTROL POWER TRANSFER (2-9-7A, Window 15) is a shared alarm. I & C 'A' and Unit Preferred both input to this alarm. 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 loss of RPIS power supply impact on Control Rod indication and the required actions to mitigate the abnormal conditions. 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.

Technical Reference(s): 2-ARP-9-7A, Rev. 35 (Attach if not previously provided)

2-ARP-9-5A, Rev. 61

2-AOI-57-4, Rev. 49

2-AOI-85-4, Rev. 21

Proposed references to be provided to applicants during examination: **PANEL 2-9-9 CABINET 1, 2, 3, or 6 CONTROL POWER TRANSFER (2-9-7A, Window 15), RPIS INOPERABLE (2-9-5A, Window 35)**

Learning Objective: OPL171.029, Obj. 7e (As available)

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

Question Cognitive Level:

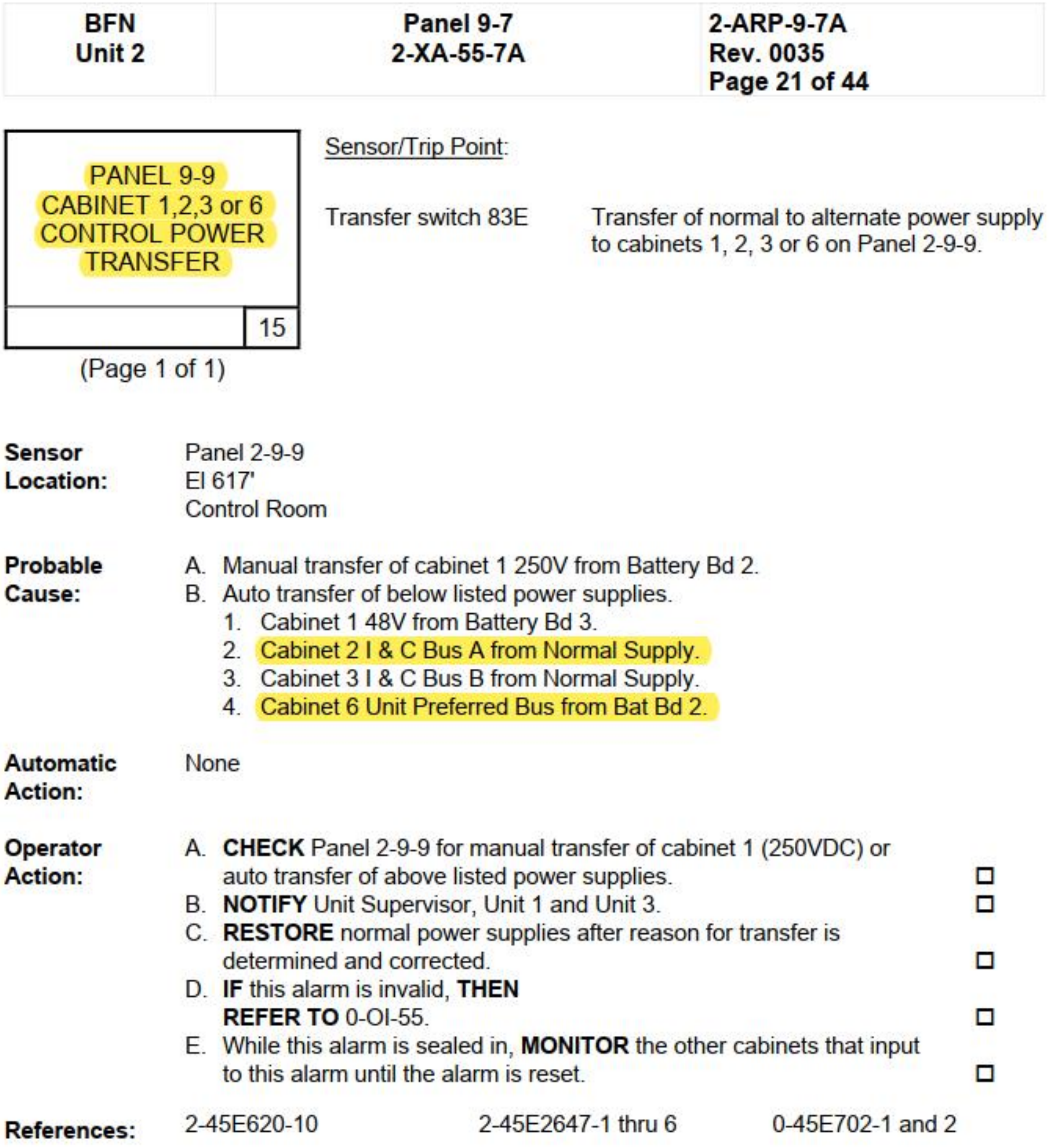
Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

10 CFR Part 55 Content:

55.41 **X**

Comments:

Excerpt from 2-ARP-9-7A:



BFN **Unit 2**	**Panel 9-7** **2-XA-55-7A**	**2-ARP-9-7A** **Rev. 0035** **Page 21 of 44**

PANEL 9-9
CABINET 1,2,3 or 6
CONTROL POWER
TRANSFER

15

(Page 1 of 1)

Sensor/Trip Point:

Transfer switch 83E	Transfer of normal to alternate power supply to cabinets 1, 2, 3 or 6 on Panel 2-9-9.

Sensor Location: Panel 2-9-9
El 617'
Control Room

Probable Cause:

A. Manual transfer of cabinet 1 250V from Battery Bd 2.
B. Auto transfer of below listed power supplies.
  1. Cabinet 1 48V from Battery Bd 3.
  2. Cabinet 2 I & C Bus A from Normal Supply.
  3. Cabinet 3 I & C Bus B from Normal Supply.
  4. Cabinet 6 Unit Preferred Bus from Bat Bd 2.

Automatic Action: None

Operator Action:

A. **CHECK** Panel 2-9-9 for manual transfer of cabinet 1 (250VDC) or auto transfer of above listed power supplies. ☐
B. **NOTIFY** Unit Supervisor, Unit 1 and Unit 3. ☐
C. **RESTORE** normal power supplies after reason for transfer is determined and corrected. ☐
D. **IF** this alarm is invalid, **THEN** **REFER TO** 0-OI-55. ☐
E. While this alarm is sealed in, **MONITOR** the other cabinets that input to this alarm until the alarm is reset. ☐

References: 2-45E620-10 2-45E2647-1 thru 6 0-45E702-1 and 2

Excerpt from 2-ARP-9-5A:

BFN **Unit 2**	**Panel 9-5** **2-XA-55-5A**	**2-ARP-9-5A** **Rev. 0061** **Page 48 of 48**

RPIS
INOPERABLE
35

(Page 1 of 1)

Sensor/Trip Point:

Relay 3A-K5

Alarms on electronic malfunction such as:

A. Card pulled.
B. Internal logic stall.

Sensor Location:

Panel 2-9-28
El 593'
Aux Instrument Room

Probable Cause:

A. 120V Unit Preferred breaker 612 on Panel 2-9-9 tripped.
B. 2-PX-85-5X(5Y)(6X)(6Y) fuse cleared or internal breaker open in Panel 2-9-27, Aux Instr Room.
C. Malfunction of a card in Panel 2-9-27.
D. Spurious trip of sensor.

Automatic Action:

A. RWM rod block.
B. Rod select block on Panel 2-9-5 controls.

Operator Action:

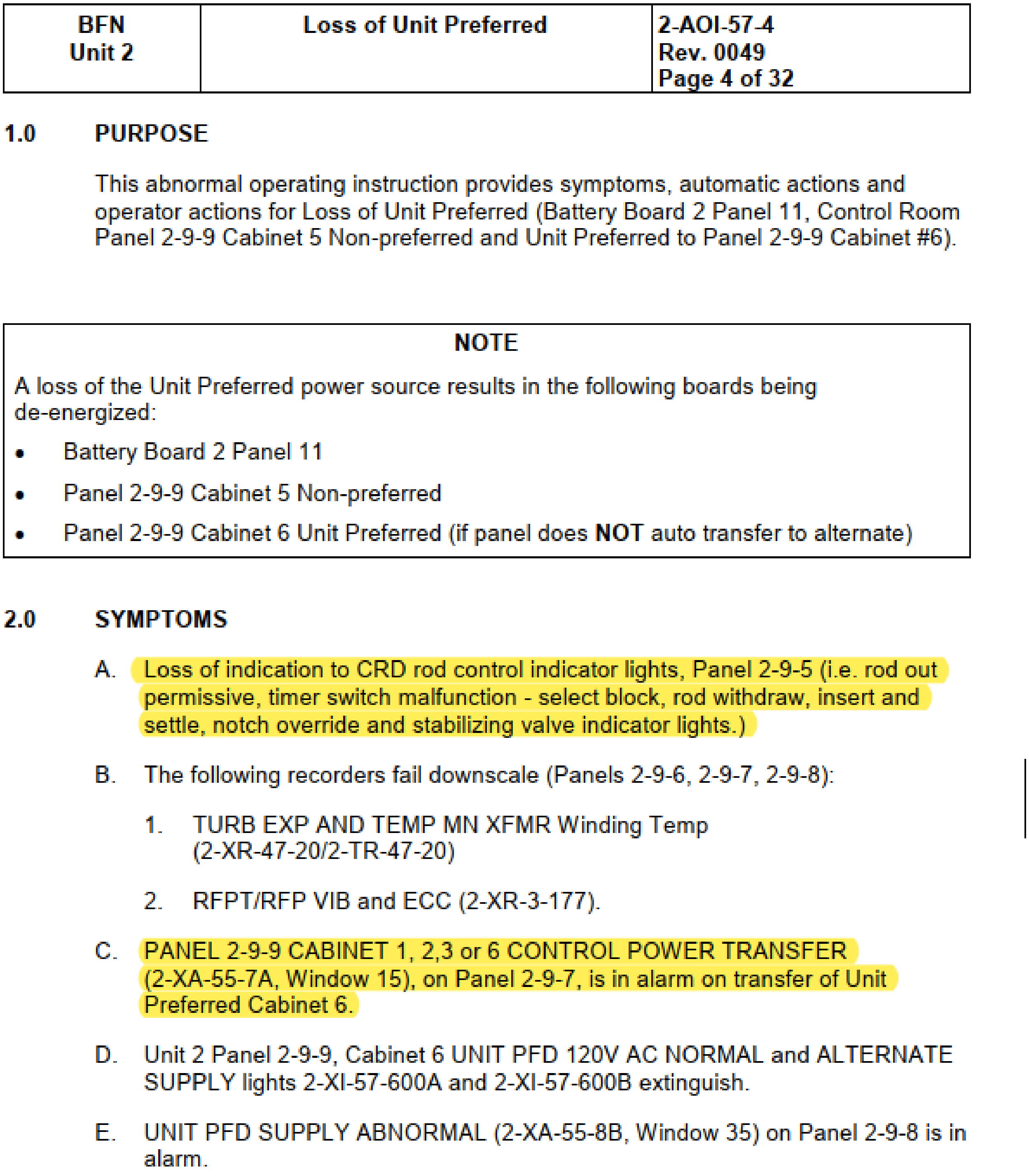
A. **CHECK** alarm as follows:
   1. Loss of position indication on Panel 2-9-5.
   2. ICS will lose position information.
   3. RWM rod block.
   4. Rod Select block on Panel 2-9-5 controls.

B. **IF** alarm is valid, **THEN**
   REFER TO 2-AOI-85-4.
C. **REFER TO** Tech Spec 3.9.3, 3.9.4, 3.10.3, 3.10.4, 3.10.5, 3.10.6, TRM Table 3.3.5-1.

References:

2-45E620-6
2-730E321-3
2-730E321-11
2-AOI-85-4
Technical Specifications
Technical Requirements Manual-TRM

Excerpts from 2-AOI-57-4:



BFN Unit 2	Loss of Unit Preferred	2-AOI-57-4 Rev. 0049 Page 4 of 32

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for Loss of Unit Preferred (Battery Board 2 Panel 11, Control Room Panel 2-9-9 Cabinet 5 Non-preferred and Unit Preferred to Panel 2-9-9 Cabinet #6).

NOTE

A loss of the Unit Preferred power source results in the following boards being de-energized:

- Battery Board 2 Panel 11
- Panel 2-9-9 Cabinet 5 Non-preferred
- Panel 2-9-9 Cabinet 6 Unit Preferred (if panel does **NOT** auto transfer to alternate)

2.0 SYMPTOMS

A. Loss of indication to CRD rod control indicator lights, Panel 2-9-5 (i.e. rod out permissive, timer switch malfunction - select block, rod withdraw, insert and settle, notch override and stabilizing valve indicator lights.)

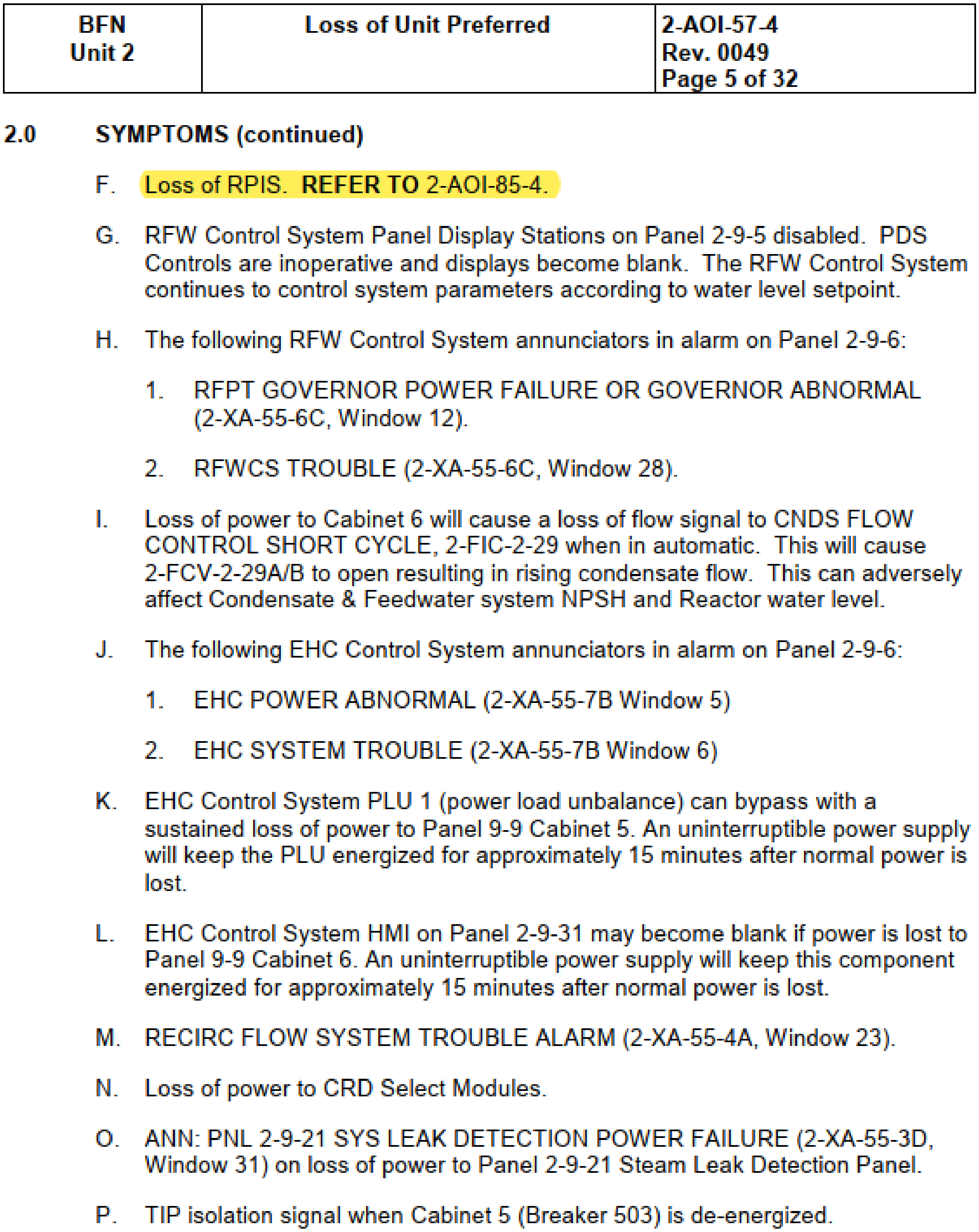
B. The following recorders fail downscale (Panels 2-9-6, 2-9-7, 2-9-8):

1. TURB EXP AND TEMP MN XFMR Winding Temp (2-XR-47-20/2-TR-47-20)
2. RFPT/RFP VIB and ECC (2-XR-3-177).

C. PANEL 2-9-9 CABINET 1, 2,3 or 6 CONTROL POWER TRANSFER (2-XA-55-7A, Window 15), on Panel 2-9-7, is in alarm on transfer of Unit Preferred Cabinet 6.

D. Unit 2 Panel 2-9-9, Cabinet 6 UNIT PFD 120V AC NORMAL and ALTERNATE SUPPLY lights 2-XI-57-600A and 2-XI-57-600B extinguish.

E. UNIT PFD SUPPLY ABNORMAL (2-XA-55-8B, Window 35) on Panel 2-9-8 is in alarm.



BFN Unit 2	**Loss of Unit Preferred**	**2-AOI-57-4 Rev. 0049 Page 5 of 32**

2.0 SYMPTOMS (continued)

F. Loss of RPIS. **REFER TO** 2-AOI-85-4.

G. RFW Control System Panel Display Stations on Panel 2-9-5 disabled. PDS Controls are inoperative and displays become blank. The RFW Control System continues to control system parameters according to water level setpoint.

H. The following RFW Control System annunciators in alarm on Panel 2-9-6:

1. RFPT GOVERNOR POWER FAILURE OR GOVERNOR ABNORMAL (2-XA-55-6C, Window 12).
2. RFWCS TROUBLE (2-XA-55-6C, Window 28).

I. Loss of power to Cabinet 6 will cause a loss of flow signal to CNDS FLOW CONTROL SHORT CYCLE, 2-FIC-2-29 when in automatic. This will cause 2-FCV-2-29A/B to open resulting in rising condensate flow. This can adversely affect Condensate & Feedwater system NPSH and Reactor water level.

J. The following EHC Control System annunciators in alarm on Panel 2-9-6:

1. EHC POWER ABNORMAL (2-XA-55-7B Window 5)
2. EHC SYSTEM TROUBLE (2-XA-55-7B Window 6)

K. 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.

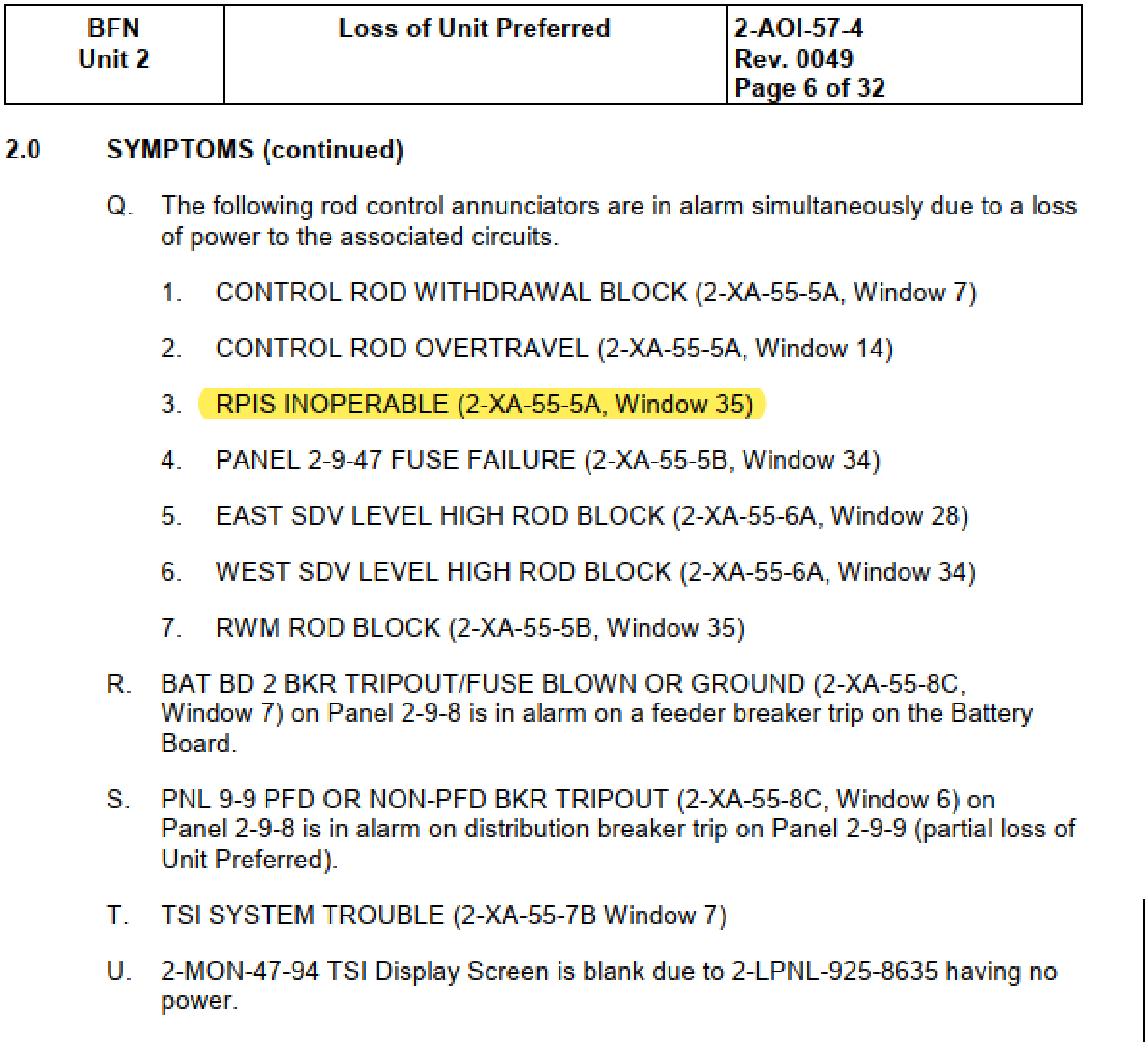
L. EHC Control System HMI on Panel 2-9-31 may become blank if power is lost to Panel 9-9 Cabinet 6. An uninterruptible power supply will keep this component energized for approximately 15 minutes after normal power is lost.

M. RECIRC FLOW SYSTEM TROUBLE ALARM (2-XA-55-4A, Window 23).

N. Loss of power to CRD Select Modules.

O. ANN: PNL 2-9-21 SYS LEAK DETECTION POWER FAILURE (2-XA-55-3D, Window 31) on loss of power to Panel 2-9-21 Steam Leak Detection Panel.

P. TIP isolation signal when Cabinet 5 (Breaker 503) is de-energized.



BFN **Unit 2**	**Loss of Unit Preferred**	**2-AOI-57-4** **Rev. 0049** **Page 6 of 32**

2.0 SYMPTOMS (continued)

Q. The following rod control annunciators are in alarm simultaneously due to a loss of power to the associated circuits.

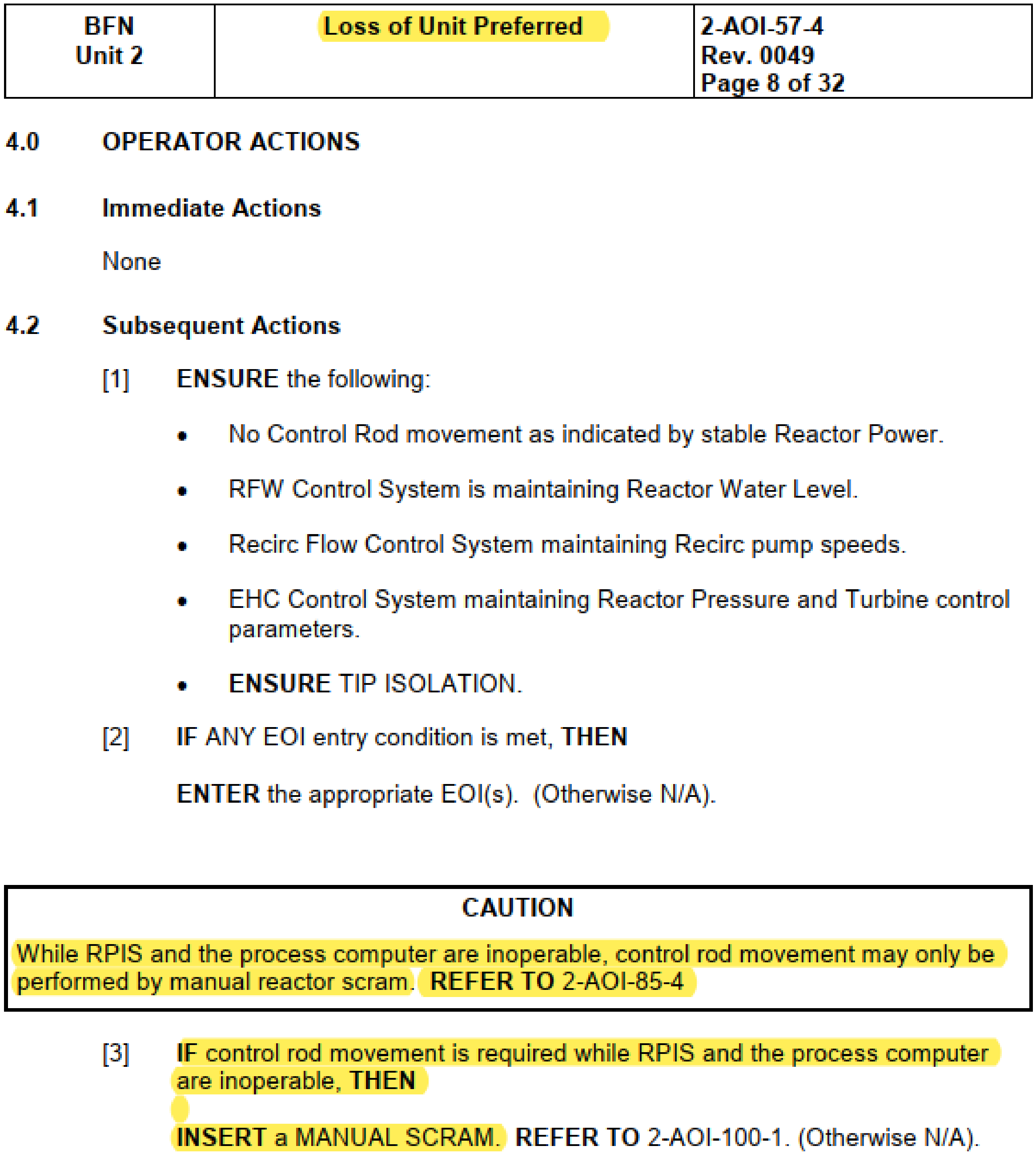
1. CONTROL ROD WITHDRAWAL BLOCK (2-XA-55-5A, Window 7)
2. CONTROL ROD OVERTRAVEL (2-XA-55-5A, Window 14)
3. RPIS INOPERABLE (2-XA-55-5A, Window 35)
4. PANEL 2-9-47 FUSE FAILURE (2-XA-55-5B, Window 34)
5. EAST SDV LEVEL HIGH ROD BLOCK (2-XA-55-6A, Window 28)
6. WEST SDV LEVEL HIGH ROD BLOCK (2-XA-55-6A, Window 34)
7. RWM ROD BLOCK (2-XA-55-5B, Window 35)

R. BAT BD 2 BKR TRIPOUT/FUSE BLOWN OR GROUND (2-XA-55-8C, Window 7) on Panel 2-9-8 is in alarm on a feeder breaker trip on the Battery Board.

S. PNL 9-9 PFD OR NON-PFD BKR TRIPOUT (2-XA-55-8C, Window 6) on Panel 2-9-8 is in alarm on distribution breaker trip on Panel 2-9-9 (partial loss of Unit Preferred).

T. TSI SYSTEM TROUBLE (2-XA-55-7B Window 7)

U. 2-MON-47-94 TSI Display Screen is blank due to 2-LPNL-925-8635 having no power.



BFN Unit 2	**Loss of Unit Preferred**	**2-AOI-57-4 Rev. 0049 Page 8 of 32**

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **ENSURE** the following:

- No Control Rod movement as indicated by stable Reactor Power.
- RFW Control System is maintaining Reactor Water Level.
- Recirc Flow Control System maintaining Recirc pump speeds.
- EHC Control System maintaining Reactor Pressure and Turbine control parameters.
- **ENSURE** TIP ISOLATION.

[2] **IF** ANY EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s). (Otherwise N/A).

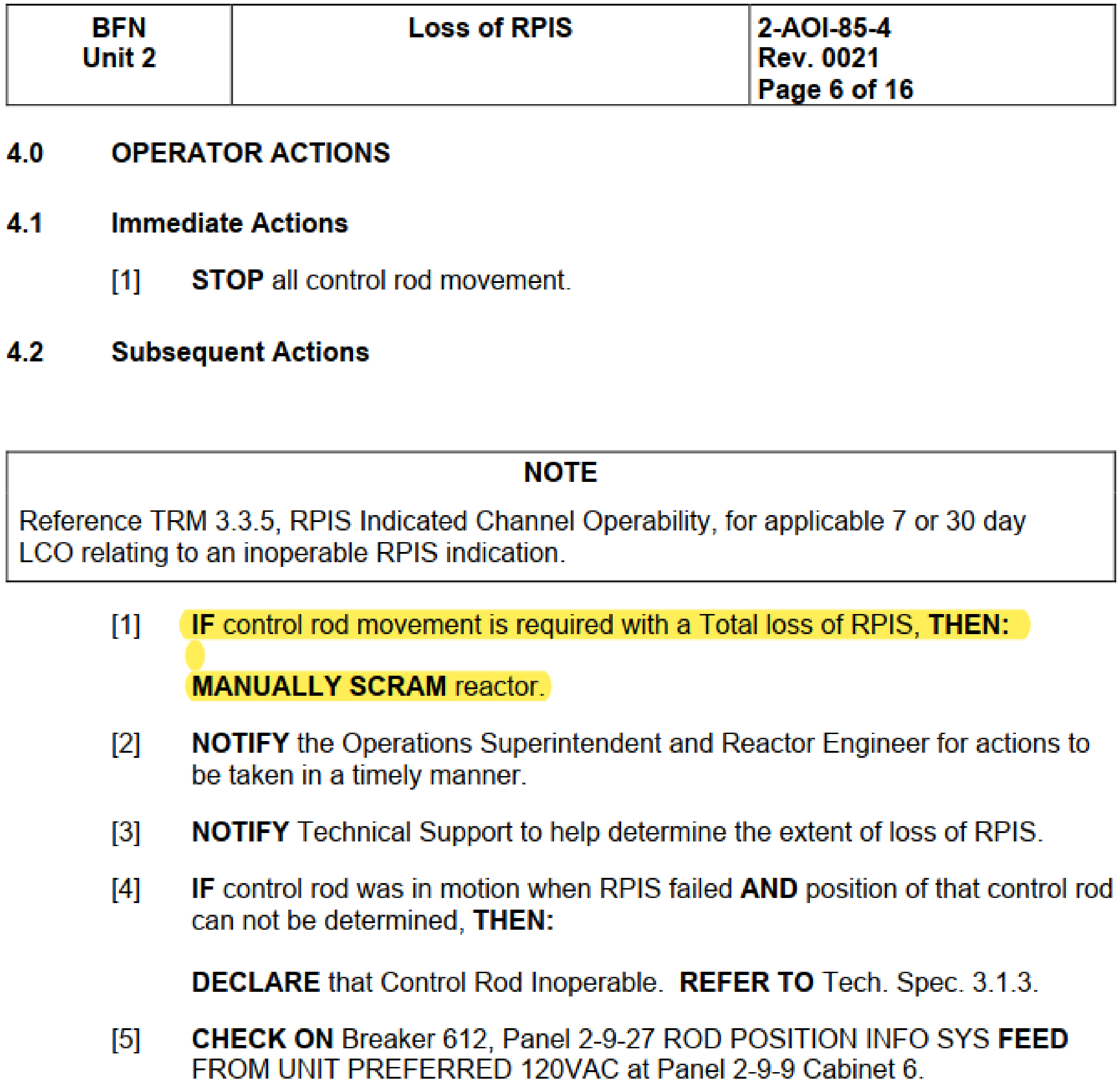
CAUTION

While RPIS and the process computer are inoperable, control rod movement may only be performed by manual reactor scram. **REFER TO** 2-AOI-85-4

[3] **IF** control rod movement is required while RPIS and the process computer are inoperable, **THEN**

INSERT a MANUAL SCRAM. **REFER TO** 2-AOI-100-1. (Otherwise N/A).

Excerpt from 2-AOI-85-4:



BFN **Unit 2**	**Loss of RPIS**	**2-AOI-85-4** **Rev. 0021** **Page 6 of 16**

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **STOP** all control rod movement.

4.2 Subsequent Actions

NOTE

Reference TRM 3.3.5, RPIS Indicated Channel Operability, for applicable 7 or 30 day LCO relating to an inoperable RPIS indication.

[1] **IF** control rod movement is required with a Total loss of RPIS, **THEN:**

MANUALLY SCRAM reactor.

[2] **NOTIFY** the Operations Superintendent and Reactor Engineer for actions to be taken in a timely manner.

[3] **NOTIFY** Technical Support to help determine the extent of loss of RPIS.

[4] **IF** control rod was in motion when RPIS failed **AND** position of that control rod can not be determined, **THEN:**

DECLARE that Control Rod Inoperable. **REFER TO** Tech. Spec. 3.1.3.

[5] **CHECK ON** Breaker 612, Panel 2-9-27 ROD POSITION INFO SYS **FEED** FROM UNIT PREFERRED 120VAC at Panel 2-9-9 Cabinet 6.

Examination Outline Cross-reference:

201006 (SF7 RWMS) Rod Worth Minimizer

A4.02 (10CFR 55.41.7)

Ability to manually operate and/or monitor in the control room:

- Pushbutton indicating switches

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	201006A4.02	
Importance Rating	3.2	------

Proposed Question: **# 53**

In accordance with 2-OI-85, Control Rod Drive System, Rod Worth Minimizer (RWM) is manually bypassed using a **(1)** switch.

RWM will automatically bypass when Total **(2)** is (are) greater than 22%.

A. (1) keylock
(2) Feedwater Flow **ONLY**

B. (1) keylock
(2) Feedwater **AND** Steam Flow

C. (1) pushbutton
(2) Feedwater Flow **ONLY**

D. (1) pushbutton
(2) Feedwater **AND** Steam Flow

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that in accordance with 2-OI-85, Control Rod Drive System, there are several combinations of Feedwater and Steam Flow that provide bypassing, alarm, and Rod Block features and any combination could be plausible for enabling or disabling any feature of the Rod Worth Minimizer (RWM). Additionally, Rod Worth Minimizer will enforce when Feedwater Flow or Total Steam Flow is less than 22%.

B **CORRECT**: (*See attached*) In accordance with 2-OI-85, Control Rod Drive System, RWM is manually bypassed by placing keylock switch 2-XS-85-9025, RWM SWITCH PANEL, in BYPASS located on Panel 2-9-5. For second part, Total Feedwater Flow and Total Steam Flow must be greater than 22% to automatically take the RWM out of service.

C INCORRECT: First part is incorrect but plausible in that RWM has several hardware and software push button controls that perform various functions, and one of those pushbutton features could plausibly be a bypass feature. Second part is incorrect but plausible (*See A*).

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

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

RO Level Justification: Tests the candidate's ability to monitor automatic bypass signals and operate pushbuttons and switches to manually bypass Rod Worth Minimizer. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): 2-OI-85, Rev. 151 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.024 Obj. 11 (As available)

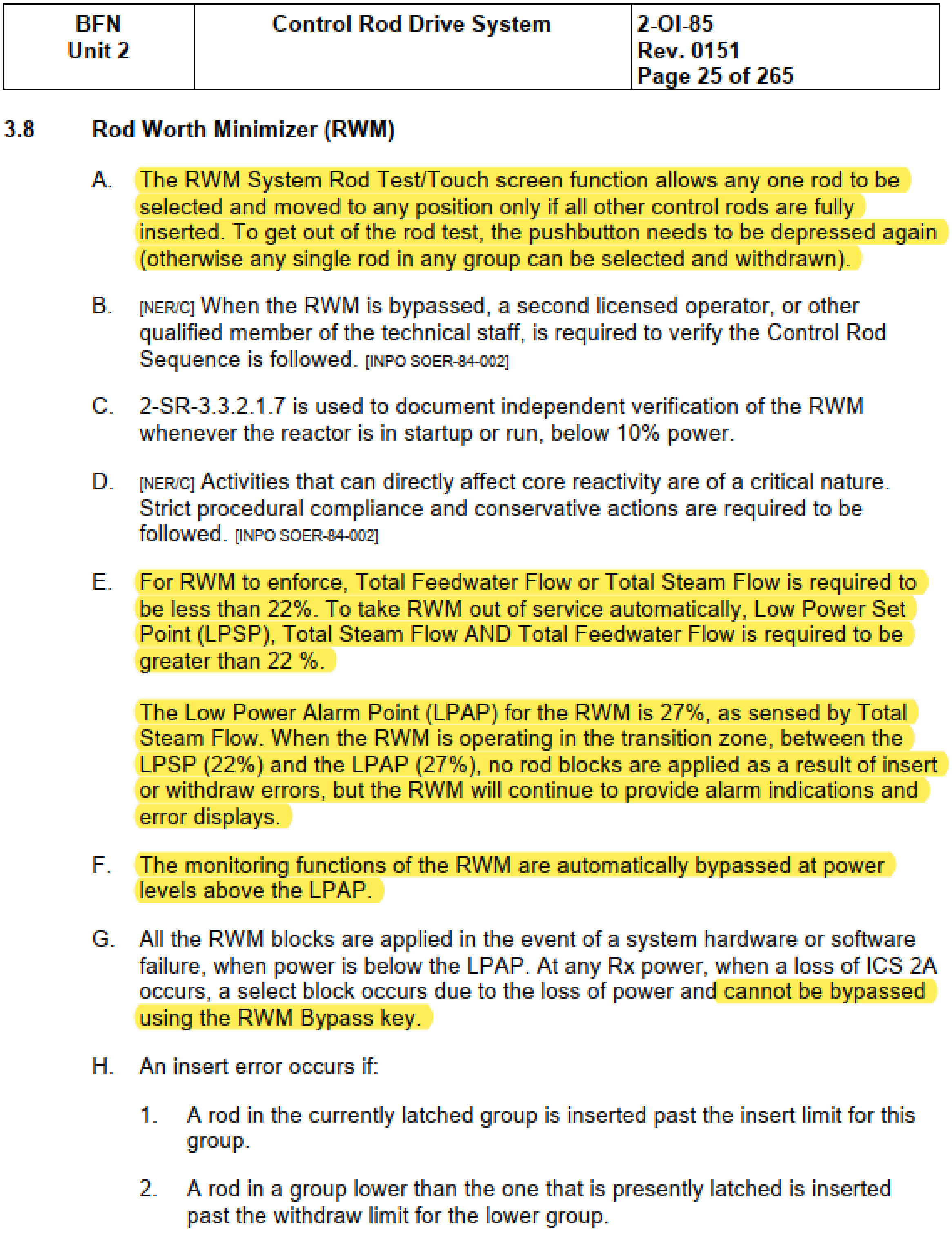
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

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

Comments:

Excerpts from 2-OI-85:



BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0151 Page 25 of 265

3.8 Rod Worth Minimizer (RWM)

A. The RWM System Rod Test/Touch screen function allows any one rod to be selected and moved to any position only if all other control rods are fully inserted. To get out of the rod test, the pushbutton needs to be depressed again (otherwise any single rod in any group can be selected and withdrawn).

B. [NER/C] When the RWM is bypassed, a second licensed operator, or other qualified member of the technical staff, is required to verify the Control Rod Sequence is followed. [INPO SOER-84-002]

C. 2-SR-3.3.2.1.7 is used to document independent verification of the RWM whenever the reactor is in startup or run, below 10% power.

D. [NER/C] Activities that can directly affect core reactivity are of a critical nature. Strict procedural compliance and conservative actions are required to be followed. [INPO SOER-84-002]

E. For RWM to enforce, Total Feedwater Flow or Total Steam Flow is required to be less than 22%. To take RWM out of service automatically, Low Power Set Point (LPSP), Total Steam Flow AND Total Feedwater Flow is required to be greater than 22 %.

The Low Power Alarm Point (LPAP) for the RWM is 27%, as sensed by Total Steam Flow. When the RWM is operating in the transition zone, between the LPSP (22%) and the LPAP (27%), no rod blocks are applied as a result of insert or withdraw errors, but the RWM will continue to provide alarm indications and error displays.

F. The monitoring functions of the RWM are automatically bypassed at power levels above the LPAP.

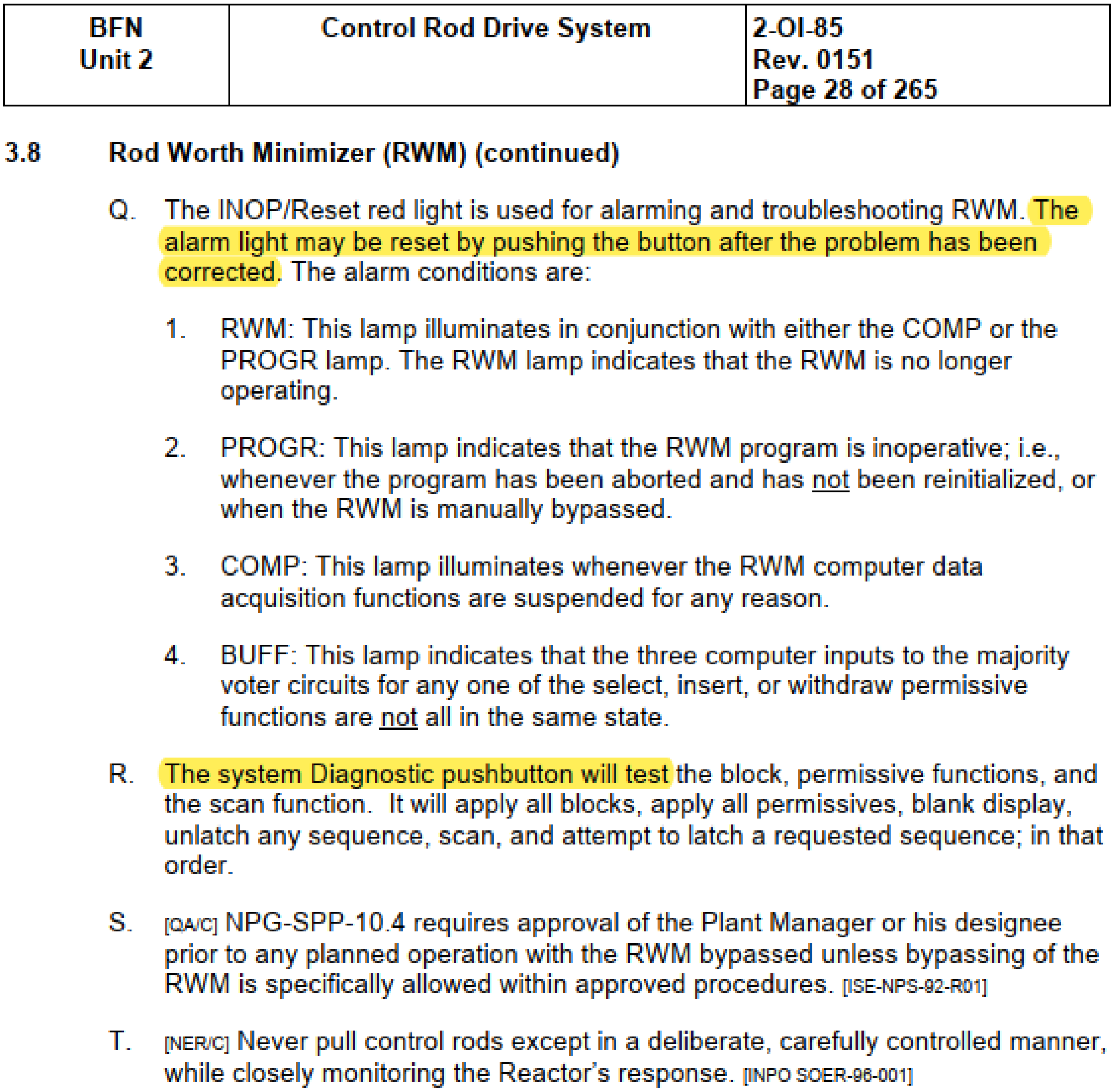
G. All the RWM blocks are applied in the event of a system hardware or software failure, when power is below the LPAP. At any Rx power, when a loss of ICS 2A occurs, a select block occurs due to the loss of power and cannot be bypassed using the RWM Bypass key.

H. An insert error occurs if:

1. A rod in the currently latched group is inserted past the insert limit for this group.

2. A rod in a group lower than the one that is presently latched is inserted past the withdraw limit for the lower group.

Supports Distractors C(1), D(1)



BFN Unit 2	**Control Rod Drive System**	**2-OI-85 Rev. 0151 Page 28 of 265**

3.8 Rod Worth Minimizer (RWM) (continued)

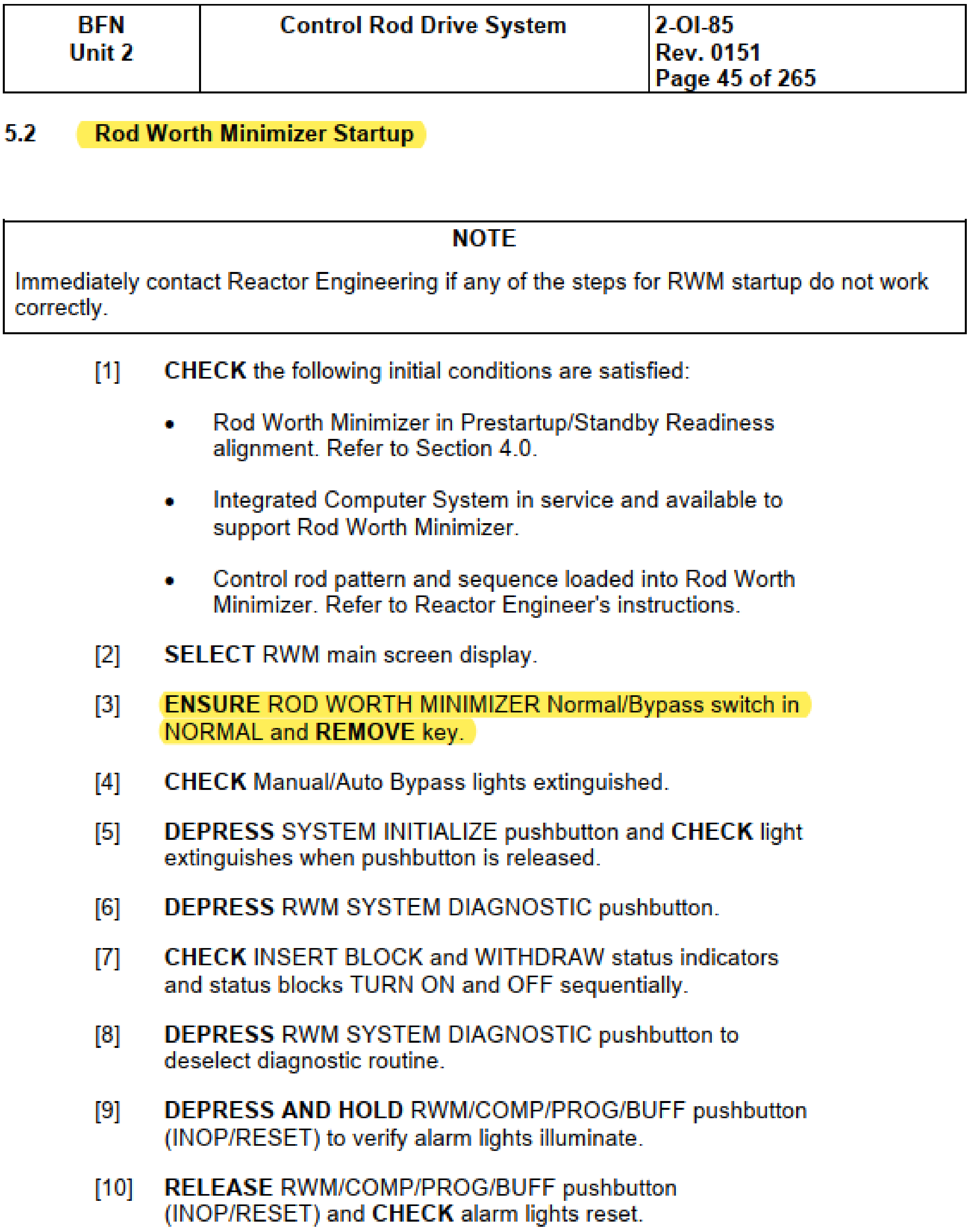
Q. The INOP/Reset red light is used for alarming and troubleshooting RWM. The alarm light may be reset by pushing the button after the problem has been corrected. The alarm conditions are:

1. RWM: This lamp illuminates in conjunction with either the COMP or the PROGR lamp. The RWM lamp indicates that the RWM is no longer operating.
2. PROGR: This lamp indicates that the RWM program is inoperative; i.e., whenever the program has been aborted and has not been reinitialized, or when the RWM is manually bypassed.
3. COMP: This lamp illuminates whenever the RWM computer data acquisition functions are suspended for any reason.
4. BUFF: This lamp indicates that the three computer inputs to the majority voter circuits for any one of the select, insert, or withdraw permissive functions are not all in the same state.

R. The system Diagnostic pushbutton will test the block, permissive functions, and the scan function. It will apply all blocks, apply all permissives, blank display, unlatch any sequence, scan, and attempt to latch a requested sequence; in that order.

S. [QA/C] NPG-SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with the RWM bypassed unless bypassing of the RWM is specifically allowed within approved procedures. [ISE-NPS-92-R01]

T. [NER/C] Never pull control rods except in a deliberate, carefully controlled manner, while closely monitoring the Reactor's response. [INPO SOER-96-001]



BFN Unit 2	**Control Rod Drive System**	**2-OI-85 Rev. 0151 Page 45 of 265**

5.2 Rod Worth Minimizer Startup

NOTE

Immediately contact Reactor Engineering if any of the steps for RWM startup do not work correctly.

[1] **CHECK** the following initial conditions are satisfied:

- Rod Worth Minimizer in Prestartup/Standby Readiness alignment. Refer to Section 4.0.
- Integrated Computer System in service and available to support Rod Worth Minimizer.
- Control rod pattern and sequence loaded into Rod Worth Minimizer. Refer to Reactor Engineer's instructions.

[2] **SELECT** RWM main screen display.

[3] **ENSURE** ROD WORTH MINIMIZER Normal/Bypass switch in NORMAL and **REMOVE** key.

[4] **CHECK** Manual/Auto Bypass lights extinguished.

[5] **DEPRESS** SYSTEM INITIALIZE pushbutton and **CHECK** light extinguishes when pushbutton is released.

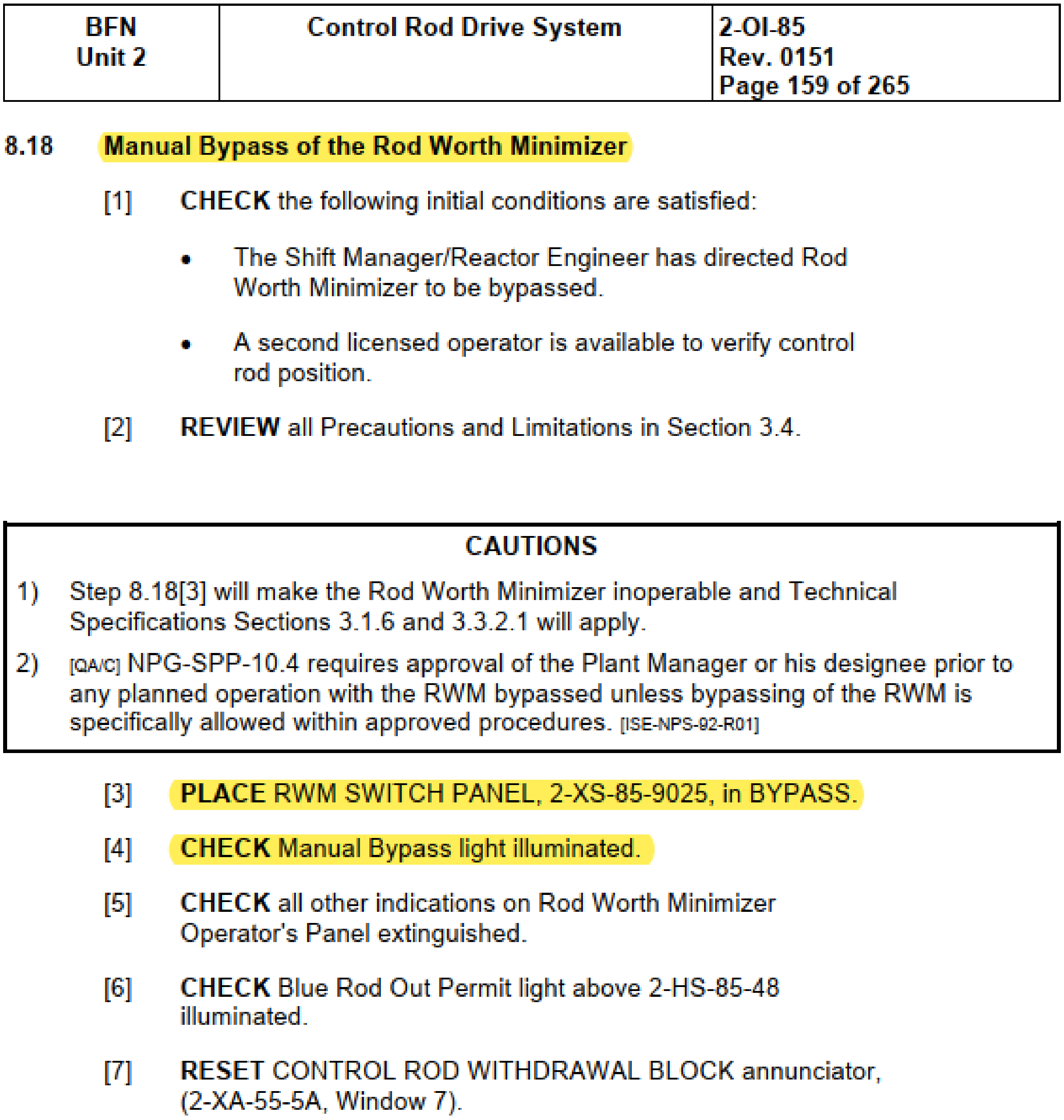
[6] **DEPRESS** RWM SYSTEM DIAGNOSTIC pushbutton.

[7] **CHECK** INSERT BLOCK and WITHDRAW status indicators and status blocks TURN ON and OFF sequentially.

[8] **DEPRESS** RWM SYSTEM DIAGNOSTIC pushbutton to deselect diagnostic routine.

[9] **DEPRESS AND HOLD** RWM/COMP/PROG/BUFF pushbutton (INOP/RESET) to verify alarm lights illuminate.

[10] **RELEASE** RWM/COMP/PROG/BUFF pushbutton (INOP/RESET) and **CHECK** alarm lights reset.



BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0151 Page 159 of 265

8.18 Manual Bypass of the Rod Worth Minimizer

[1] **CHECK** the following initial conditions are satisfied:

- The Shift Manager/Reactor Engineer has directed Rod Worth Minimizer to be bypassed.
- A second licensed operator is available to verify control rod position.

[2] **REVIEW** all Precautions and Limitations in Section 3.4.

CAUTIONS

1) Step 8.18[3] will make the Rod Worth Minimizer inoperable and Technical Specifications Sections 3.1.6 and 3.3.2.1 will apply.

2) [QA/C] NPG-SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with the RWM bypassed unless bypassing of the RWM is specifically allowed within approved procedures. [ISE-NPS-92-R01]

[3] **PLACE** RWM SWITCH PANEL, 2-XS-85-9025, in BYPASS.

[4] **CHECK** Manual Bypass light illuminated.

[5] **CHECK** all other indications on Rod Worth Minimizer Operator's Panel extinguished.

[6] **CHECK** Blue Rod Out Permit light above 2-HS-85-48 illuminated.

[7] **RESET** CONTROL ROD WITHDRAWAL BLOCK annunciator, (2-XA-55-5A, Window 7).

Examination Outline Cross-reference:

295007 (APE 7) High Reactor Pressure / 3
AA1.06 (10CFR 55.41.7)
Ability to operate and/or monitor the following as they apply to High Reactor Pressure:

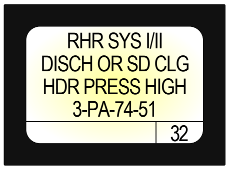
- Shutdown cooling system (RHR shutdown cooling mode)

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295007AA1.06	
Importance Rating	3.6	------

Proposed Question: **# 54**

Unit 3 is in MODE 3 with Loop I RHR in Shutdown Cooling when the following conditions occur:

- A Drywell leak occurs
- Drywell Pressure rises to 2.7 psig
- RHR SYS I/II DISCHARGE OR SHUTDOWN COOLING HEADER PRESSURE HIGH (3-9-3E, Window 32) alarms
- Reactor Pressure is 105 psig



In accordance with the appropriate ARP, ___**(1)**___ automatically closed when Reactor Pressure reached ___**(2)**___.

Note: 3-FCV-74-47, RHR SHUTDOWN COOLING SUCTION OUTBOARD ISOLATION VALVE
3-FCV-74-48, RHR SHUTDOWN COOLING SUCTION INBOARD ISOLATION VALVE

A. (1) **ONLY** 3-FCV-74-47
(2) 95 psig

B. (1) **ONLY** 3-FCV-74-47
(2) 100 psig

C. (1) 3-FCV-74-47 **AND** 3-FCV-74-48
(2) 95 psig

D. (1) 3-FCV-74-47 **AND** 3-FCV-74-48
(2) 100 psig

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that there are failures that would result in the closure of only 3-FCV-74-47 or 3-FCV-74-48. A loss of RPS-A closes 3-FCV-74-47 and a loss of RPS-B closes 3-FCV-74-48. Additionally, a High Reactor Pressure isolation closes both 3-FCV-74-47 and 3-FCV-74-48, where an isolation due to High Drywell Pressure closes all valves that are open. Second part is incorrect but plausible in that a multitude of Reactor Pressure isolation setpoints exist such as HPCI isolates at 110 psig and RCIC isolates at 70 psig (86 psig pressure switch setpoint value with static head correction).

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 given alarm RHR SYS I/II DISCHARGE OR SHUTDOWN COOLING HEADER PRESSURE HIGH (3-9-3E, Window 32), both 3-FCV-74-47 and 3-FCV-74-48 (along with other valves) will automatically close at 100 psig Reactor Pressure when in Shutdown Cooling. Although not an option, the stem also states that Drywell Pressure was 2.7 psig which is a PCIS Group II Isolation signal (2.45 psig) closes the same valves. For second part, in accordance with 3-OI-74, RHR System, if Unit 3 Reactor Pressure exceeds 100 psig or a Group II Isolation occurs on Unit 3 while Shutdown Cooling is in operation, both 3-FCV-74-47 and 3-FCV-74-48 will automatically close at 100 psig Reactor Pressure and trip the operating RHR Pumps in order to protect the low pressure RHR piping.

RO Level Justification: The question tests the candidate's ability to monitor and predict the outcome of High Reactor Pressure impacts on Shutdown Cooling mode of RHR. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 3-ARP-9-3E, Rev. 34 (Attach if not previously provided)
3-OI-74, Rev. 132
3-OI-71, Rev. 65
3-OI-73, Rev. 63

Proposed references to be provided to applicants during examination: **RHR SYS I/II DISCHARGE OR SHUTDOWN COOLING HEADER PRESSURE HIGH (3-9-3E, Window 32)**

Learning Objective: OPL171.044 Obj. 4i As available)

Question Source:	Bank #		
	Modified Bank #	OPL171.044-01 002 #1474	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

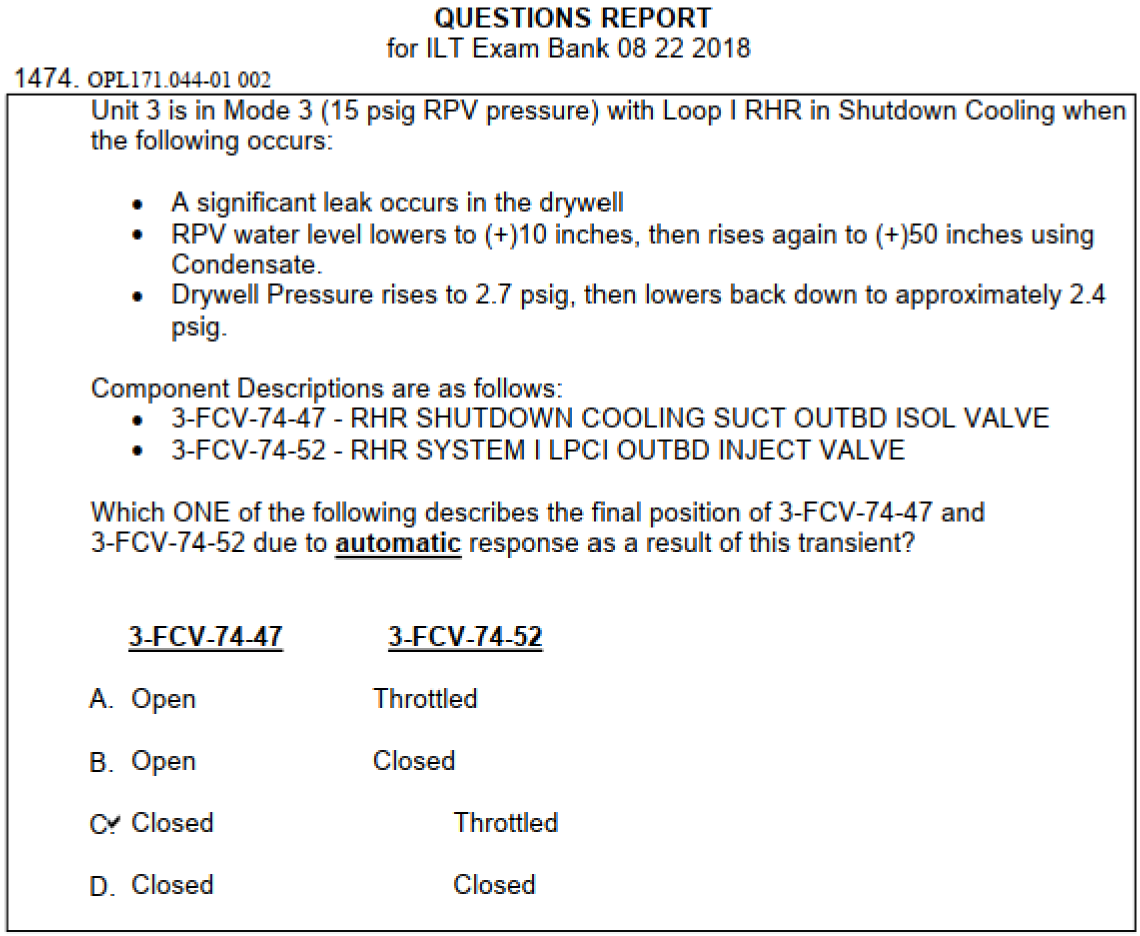
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

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

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

55.43

Copy of Bank Question:



QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1474. OPL171.044-01 002

Unit 3 is in Mode 3 (15 psig RPV pressure) with Loop I RHR in Shutdown Cooling when the following occurs:

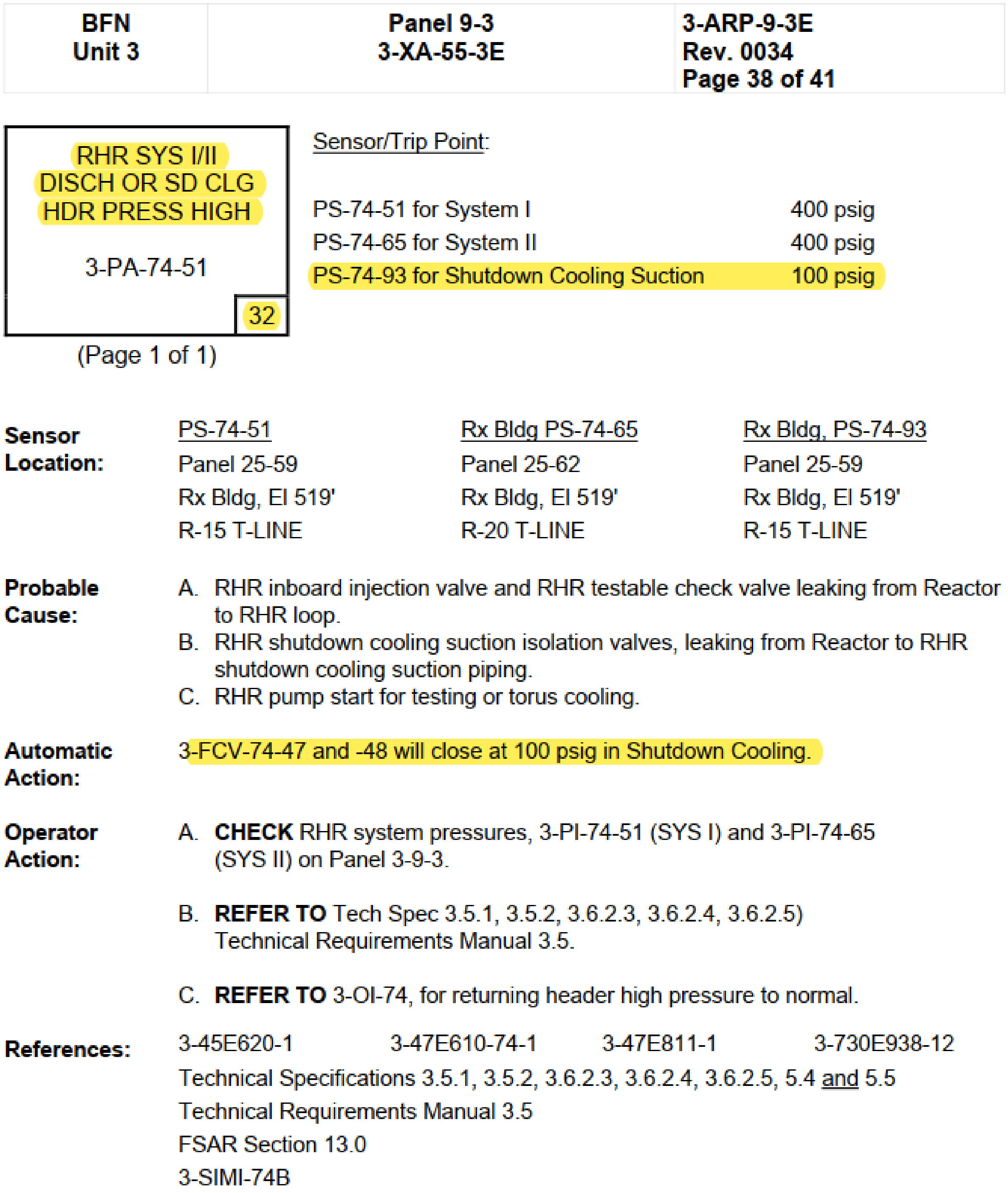
- A significant leak occurs in the drywell
- RPV water level lowers to (+)10 inches, then rises again to (+)50 inches using Condensate.
- Drywell Pressure rises to 2.7 psig, then lowers back down to approximately 2.4 psig.

Component Descriptions are as follows:
- 3-FCV-74-47 - RHR SHUTDOWN COOLING SUCT OUTBD ISOL VALVE
- 3-FCV-74-52 - RHR SYSTEM I LPCI OUTBD INJECT VALVE

Which ONE of the following describes the final position of 3-FCV-74-47 and 3-FCV-74-52 due to **automatic** response as a result of this transient?

	3-FCV-74-47	3-FCV-74-52
A.	Open	Throttled
B.	Open	Closed
C✓	Closed	Throttled
D.	Closed	Closed

Excerpt from 3-ARP-9-3E:



BFN **Unit 3**	**Panel 9-3** **3-XA-55-3E**	**3-ARP-9-3E** **Rev. 0034** **Page 38 of 41**

RHR SYS I/II
DISCH OR SD CLG
HDR PRESS HIGH

3-PA-74-51

32

(Page 1 of 1)

Sensor/Trip Point:

PS-74-51 for System I	400 psig
PS-74-65 for System II	400 psig
PS-74-93 for Shutdown Cooling Suction	100 psig

Sensor Location:	PS-74-51	Rx Bldg PS-74-65	Rx Bldg, PS-74-93
	Panel 25-59	Panel 25-62	Panel 25-59
	Rx Bldg, El 519'	Rx Bldg, El 519'	Rx Bldg, El 519'
	R-15 T-LINE	R-20 T-LINE	R-15 T-LINE

Probable Cause:

A. RHR inboard injection valve and RHR testable check valve leaking from Reactor to RHR loop.
B. RHR shutdown cooling suction isolation valves, leaking from Reactor to RHR shutdown cooling suction piping.
C. RHR pump start for testing or torus cooling.

Automatic Action: 3-FCV-74-47 and -48 will close at 100 psig in Shutdown Cooling.

Operator Action:

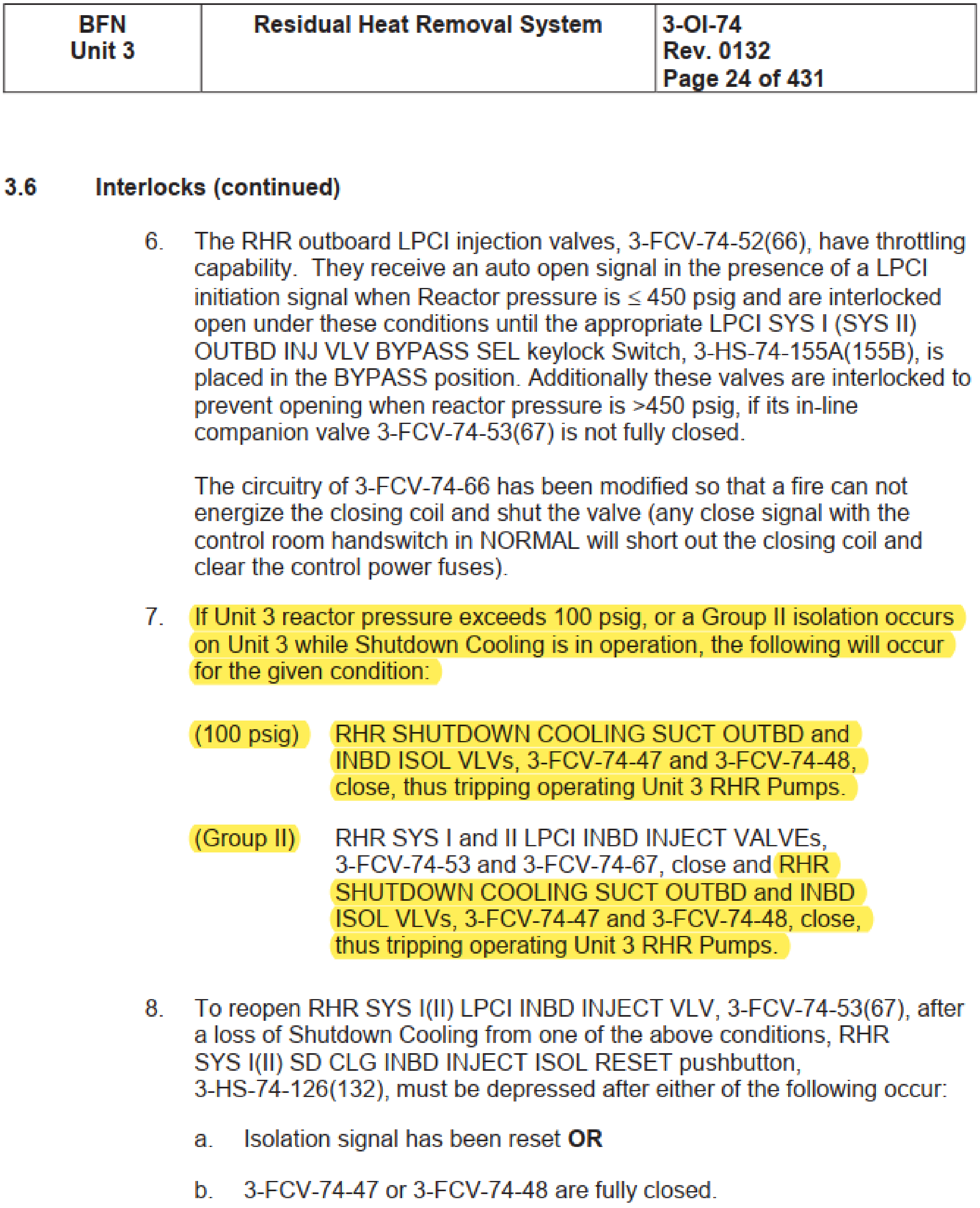
A. **CHECK** RHR system pressures, 3-PI-74-51 (SYS I) and 3-PI-74-65 (SYS II) on Panel 3-9-3.

B. **REFER TO** Tech Spec 3.5.1, 3.5.2, 3.6.2.3, 3.6.2.4, 3.6.2.5) Technical Requirements Manual 3.5.

C. **REFER TO** 3-OI-74, for returning header high pressure to normal.

References: 3-45E620-1 3-47E610-74-1 3-47E811-1 3-730E938-12
Technical Specifications 3.5.1, 3.5.2, 3.6.2.3, 3.6.2.4, 3.6.2.5, 5.4 and 5.5
Technical Requirements Manual 3.5
FSAR Section 13.0
3-SIMI-74B

Excerpt from 3-OI-74:



BFN **Unit 3**	**Residual Heat Removal System**	**3-OI-74** **Rev. 0132** **Page 24 of 431**

3.6 Interlocks (continued)

6. The RHR outboard LPCI injection valves, 3-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 (SYS II) OUTBD INJ VLV BYPASS SEL keylock Switch, 3-HS-74-155A(155B), is placed in the BYPASS position. Additionally these valves are interlocked to prevent opening when reactor pressure is >450 psig, if its in-line companion valve 3-FCV-74-53(67) is not fully closed.

   The circuitry of 3-FCV-74-66 has been modified so that a fire can not energize the closing coil and shut the valve (any close signal with the control room handswitch in NORMAL will short out the closing coil and clear the control power fuses).

7. If Unit 3 reactor pressure exceeds 100 psig, or a Group II isolation occurs on Unit 3 while Shutdown Cooling is in operation, the following will occur for the given condition:

   (100 psig) RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 3-FCV-74-47 and 3-FCV-74-48, close, thus tripping operating Unit 3 RHR Pumps.

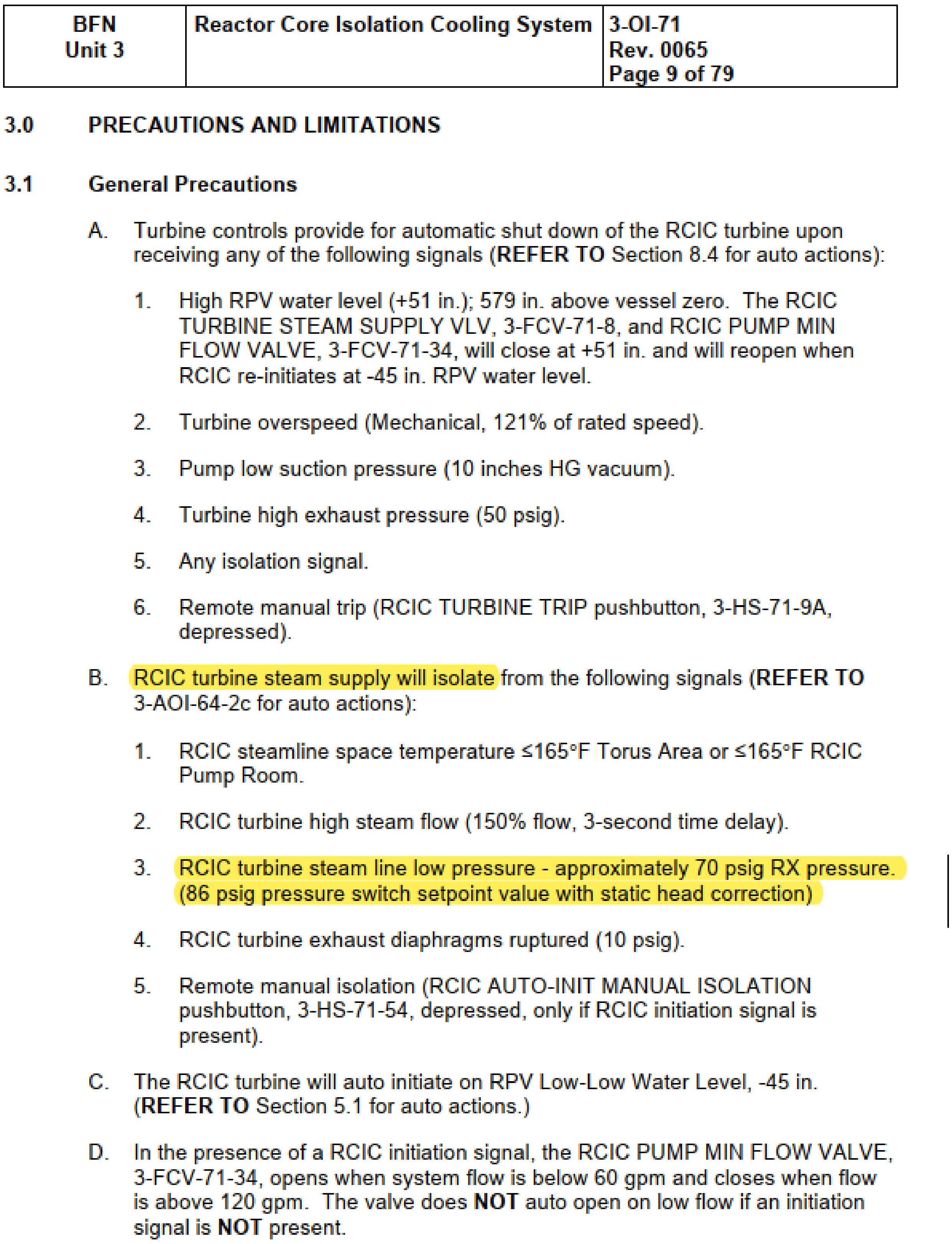
   (Group II) RHR SYS I and II LPCI INBD INJECT VALVEs, 3-FCV-74-53 and 3-FCV-74-67, close and RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 3-FCV-74-47 and 3-FCV-74-48, close, thus tripping operating Unit 3 RHR Pumps.

8. To reopen RHR SYS I(II) LPCI INBD INJECT VLV, 3-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT ISOL RESET pushbutton, 3-HS-74-126(132), must be depressed after either of the following occur:

   a. Isolation signal has been reset **OR**

   b. 3-FCV-74-47 or 3-FCV-74-48 are fully closed.

Excerpt from 3-OI-71: Supports Distractors A(2), C(2)



BFN **Unit 3**	**Reactor Core Isolation Cooling System**	**3-OI-71** **Rev. 0065** **Page 9 of 79**

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions

A. Turbine controls provide for automatic shut down 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, 3-FCV-71-8, and RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34, will close at +51 in. and will reopen when RCIC re-initiates at -45 in. RPV water level.

2. Turbine overspeed (Mechanical, 121% 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 pushbutton, 3-HS-71-9A, depressed).

B. RCIC turbine steam supply will isolate from the following signals (**REFER TO** 3-AOI-64-2c for auto actions):

1. RCIC steamline space temperature ≤165°F Torus Area or ≤165°F RCIC Pump Room.

2. RCIC turbine high steam flow (150% flow, 3-second time delay).

3. RCIC turbine steam line low pressure - approximately 70 psig RX pressure. (86 psig pressure switch setpoint value with static head correction)

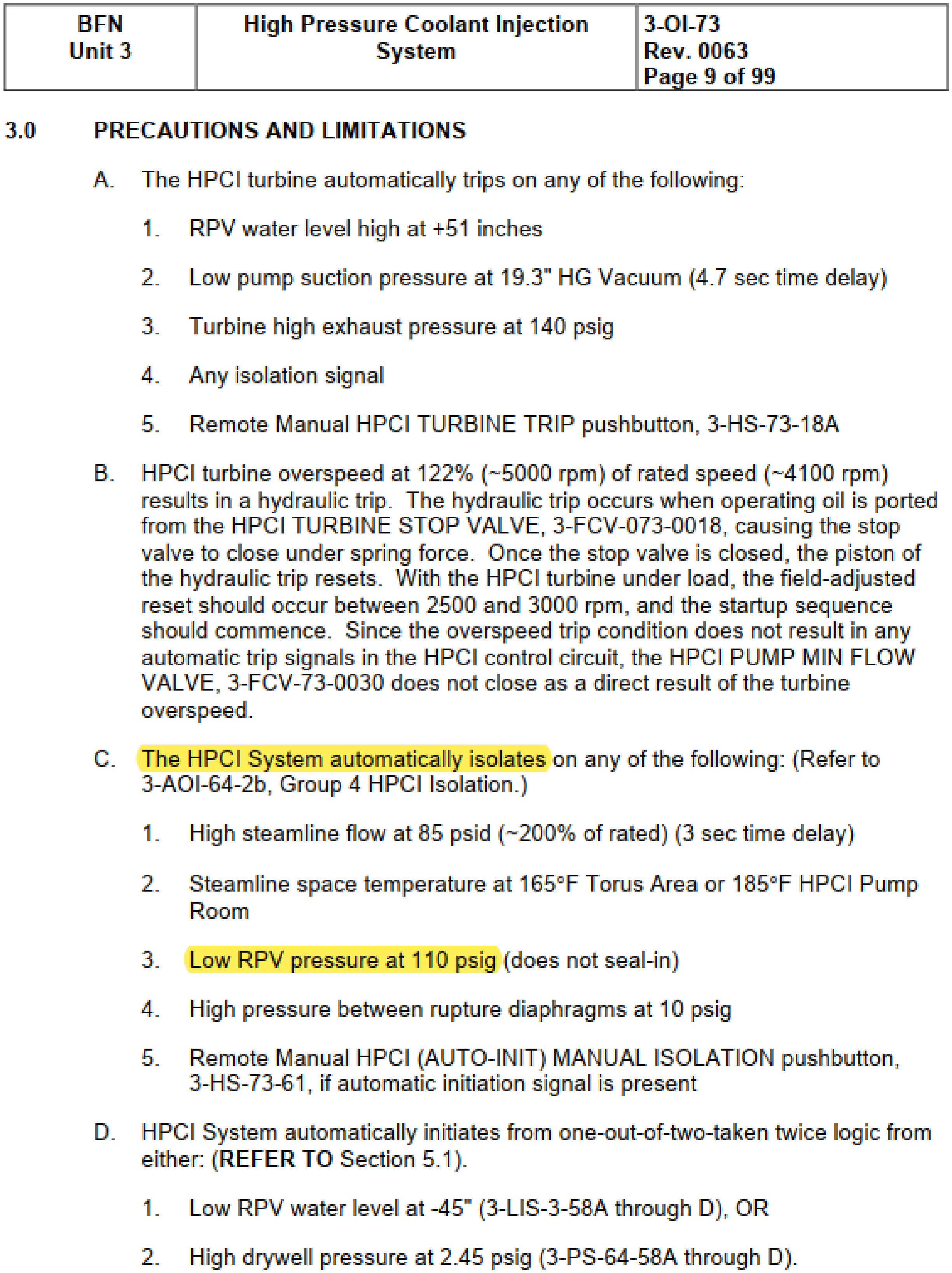
4. RCIC turbine exhaust diaphragms ruptured (10 psig).

5. Remote manual isolation (RCIC AUTO-INIT MANUAL ISOLATION pushbutton, 3-HS-71-54, depressed, only if RCIC initiation signal is present).

C. The RCIC turbine will auto initiate on RPV Low-Low Water Level, -45 in. (**REFER TO** Section 5.1 for auto actions.)

D. In the presence of a RCIC initiation signal, the RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34, opens when system flow is below 60 gpm and closes when flow is above 120 gpm. The valve does **NOT** auto open on low flow if an initiation signal is **NOT** present.

Excerpt from 3-OI-73: Supports Distractors A(2), C(2)



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.

C. The HPCI System automatically isolates on any of the following: (Refer to 3-AOI-64-2b, Group 4 HPCI Isolation.)

1. High steamline flow at 85 psid (~200% of rated) (3 sec time delay)
2. Steamline space temperature at 165°F Torus Area or 185°F HPCI Pump Room
3. Low RPV pressure at 110 psig (does not seal-in)
4. High pressure between rupture diaphragms at 10 psig
5. Remote Manual HPCI (AUTO-INIT) MANUAL ISOLATION pushbutton, 3-HS-73-61, if automatic initiation signal is present

D. HPCI System automatically initiates from one-out-of-two-taken twice logic from either: (**REFER TO** Section 5.1).

1. Low RPV water level at -45" (3-LIS-3-58A through D), OR
2. High drywell pressure at 2.45 psig (3-PS-64-58A through D).

Examination Outline Cross-reference:

264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG

A2.08 (10CFR 55.41.5)

Ability to predict the impacts of the following on the Emergency Generators and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- Initiation of emergency generator room fire protection system

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	264000A2.08	

Importance Rating	3.1	

Proposed Question: **# 55**

In accordance with 0-OI-39, CO_2 System, the initiation of CO_2 into the EDG rooms is **(1)** .

If the CO_2 System has initiated in an EDG room, then the respective EDG **(2)** receive an automatic trip signal.

A. (1) automatic **ONLY**
(2) does

B. (1) automatic **ONLY**
(2) does **NOT**

C. (1) automatic **OR** manual
(2) does

D. (1) automatic **OR** manual
(2) does **NOT**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 0-OI-39 has directions for manual or automatic initiation, however upon a fire condition, CO_2 will automatically initiate. Second part is incorrect but plausible to think that upon a CO_2 initiation, an EDG trip will occur. The ventilation to and from the room is secured as a result. However, the initiation does not input directly into the EDG logic.

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 Fire Protection Requirements Manual, we do have installed CO_2 Fire Suppression Systems at BFN in the Diesel Generator Buildings for all three Units. It can be operated in Manual or Automatic modes. For second part, while ventilation is secured in the EDG room, that would directly affect the ability to maintain the EDG operating. However, there is not an automatic trip of the EDG due to CO_2 initiating automatically or manually.

RO Level Justification: Tests the candidate's ability to predict the impacts of the initiation of Emergency Diesel Generator Room Fire Protection System and the associated procedures used to mitigate the conditions. This question is rated as Memory due to the requirement to strictly recall facts regarding the initiation of carbon dioxide in the Emergency Diesel Generator Room.

Technical Reference(s):	0-OI-39, Rev. 38	(Attach if not previously provided)
	0-OI-82, Rev. 175	
	NFPA 805 Fire Protection Requirements Manual, Rev. 15	

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.038, Obj 14_ (As available)

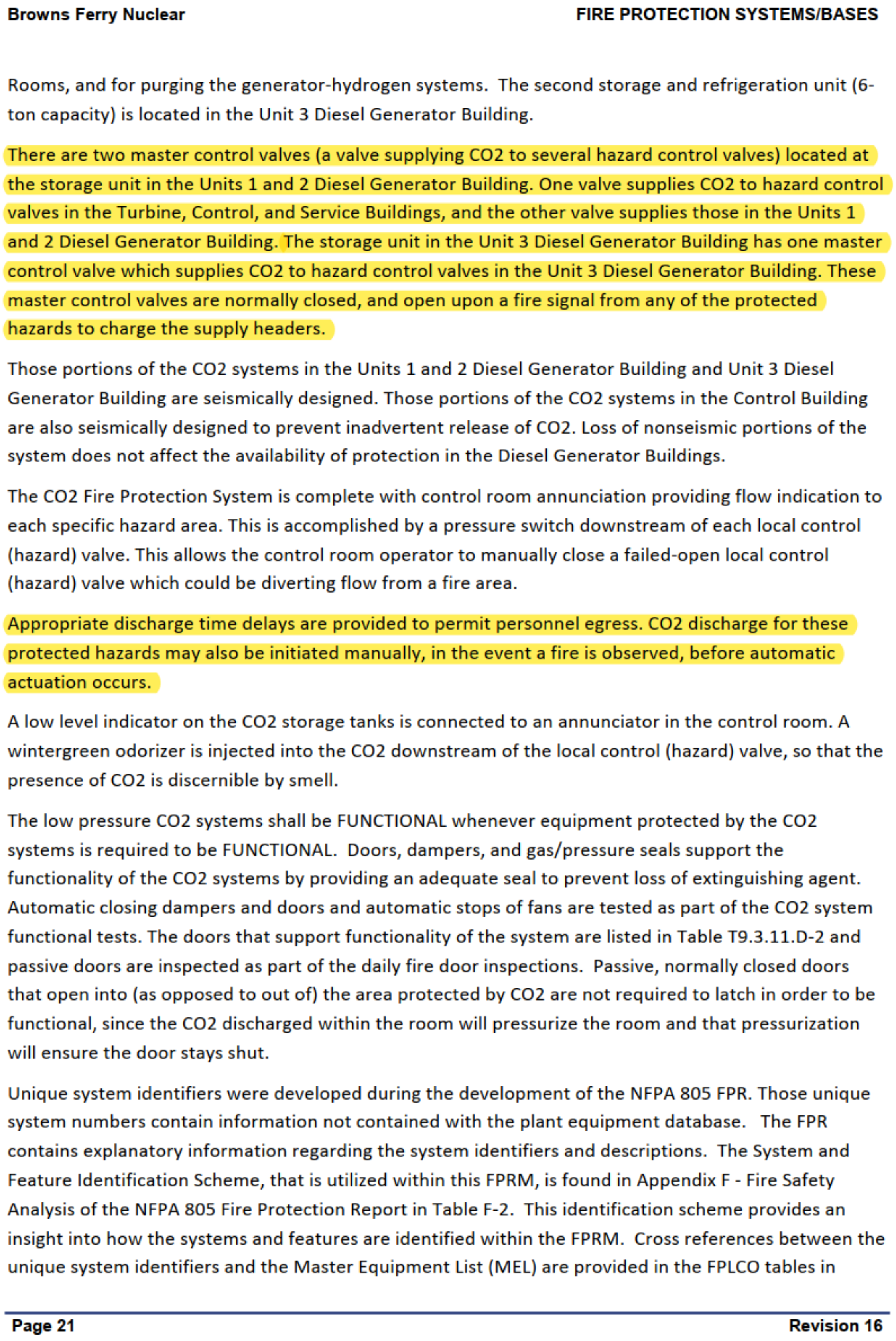
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

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

Comments:

Excerpts from NFPA 805 Fire Protection Requirements Manual:



Browns Ferry Nuclear **FIRE PROTECTION SYSTEMS/BASES**

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. The System and Feature Identification Scheme, that is utilized within this FPRM, is found in Appendix F - Fire Safety Analysis of the NFPA 805 Fire Protection Report in Table F-2. This identification scheme provides an insight into how the systems and features are identified within the FPRM. Cross references between the unique system identifiers and the Master Equipment List (MEL) are provided in the FPLCO tables in

Page 21 **Revision 16**

Excerpts from 0-OI-39:

BFN Unit 0	CO_2 System	0-OI-39 Rev. 0038 Page 8 of 29

3.0 PRECAUTIONS AND LIMITATIONS

A. [NRC/C] This Operating Instruction is used for three units. Valves, electrical boards, switches, and instruments have a specific unit prefix designated unless common to more than one unit, in which case "0" prefix is used. [RPT 82-13]

B. Minimum required CO_2 tank level is 5 units on the associated level indicator. (Unit 1 & 2, 5 = 8.5 tons; Unit 3, 5 = 3 tons)

C. CO_2 Fire Protection is required to be manually initiated for the following areas:

   1. Auxiliary Instrument Rooms.

   2. Computer Rooms.

D. Upon CO_2 initiation, an alarm will sound. Personnel then have 60 seconds (20 seconds for the Lube Oil Purification Room) to evacuate the area before CO_2 is dispensed. Evacuation times are posted at the entrance of each room. For detection purposes a wintergreen odor is injected into the CO_2 discharge.

E. Oxygen Deficiency Monitors 0-O2A-39-24 for Unit 1/2 CO_2 Storage Tank Room, and 3-O2A -39-24 for Unit 3 CO_2 Storage Tank Room, provide local audible and visual indication of low oxygen levels ($\leq$19.5%).

F. When work such as welding, cutting, or burning occurs in an area protected by CO_2, permission is required to be obtained from Unit SRO and Fire Protection Supervisor, or his designee, to place local CO_2 cutout switch in OFF.

G. When the refrigeration unit is SHUT DOWN for an extended period, approximately 6 to 20 pounds of CO_2 per hour will be lost through the respective bleeder valve located on the pilot supply line on the Unit 1 and 2 Cardox Unit and on the 3 way relief selector valve on the Unit 3 Cardox Unit.

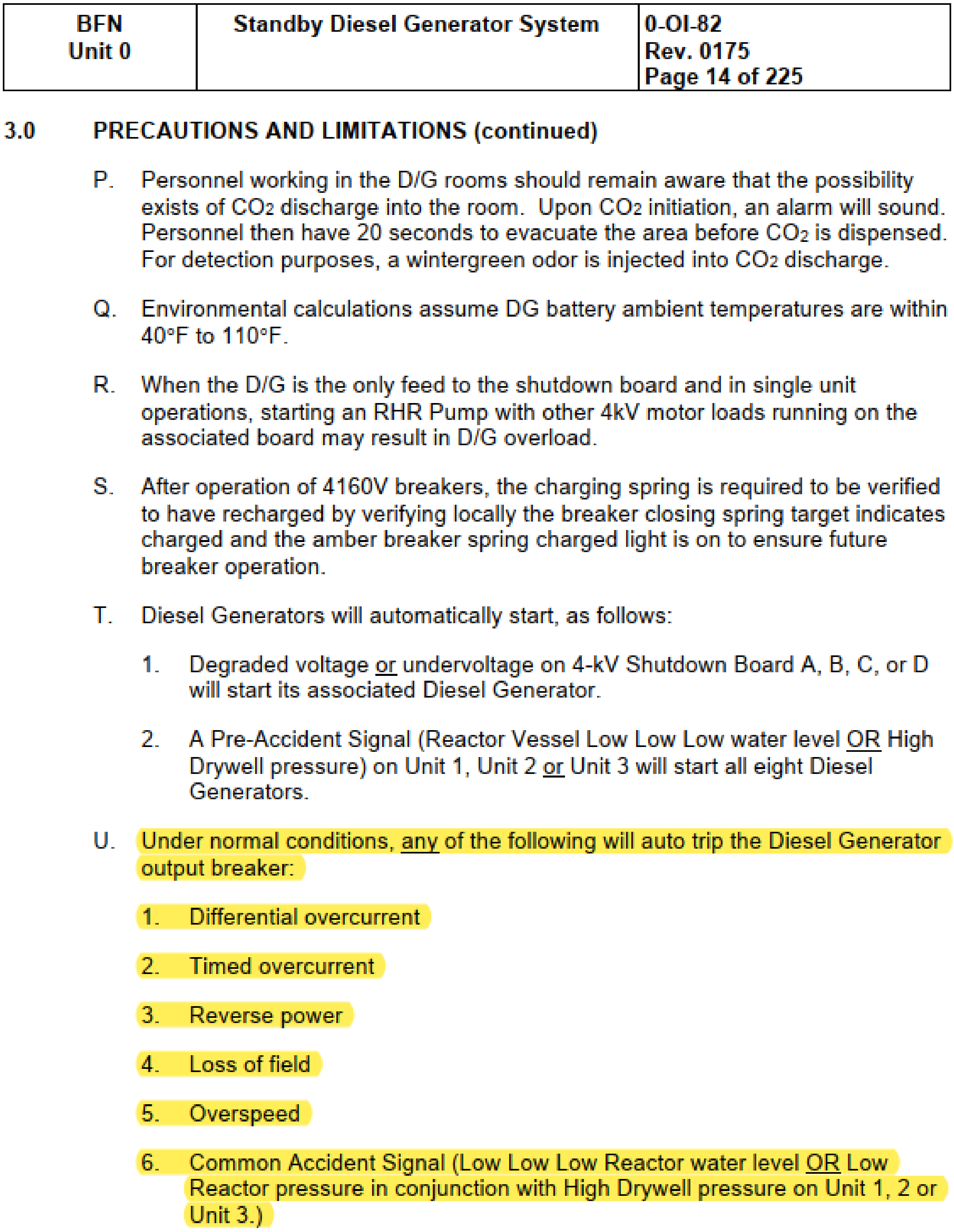
H. If CO_2 fire protection is lost to any diesel generator building area, a fire watch is required to be established immediately in accordance with the Fire Protection Report/Fire Protection Requirement Manual and is required to be continued until CO_2 fire protection is restored.

I. Prior to operating the CO_2 System from the pilot control valve station, make sure the area that CO_2 will discharge into is evacuated, as manual operation of the pilot control valve will bypass the alarm and 60 second time delay (20 second time delay for Lube Oil Purification Room).

J. Manual operation of the CO_2 System is performed only under the direction of the Unit SRO.

K. CO_2 discharge into the Auxiliary Instrument Rooms may have an adverse effect on the instrumentation.

Excerpts from 0-OI-82:



BFN Unit 0	**Standby Diesel Generator System**	**0-OI-82 Rev. 0175 Page 14 of 225**

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.

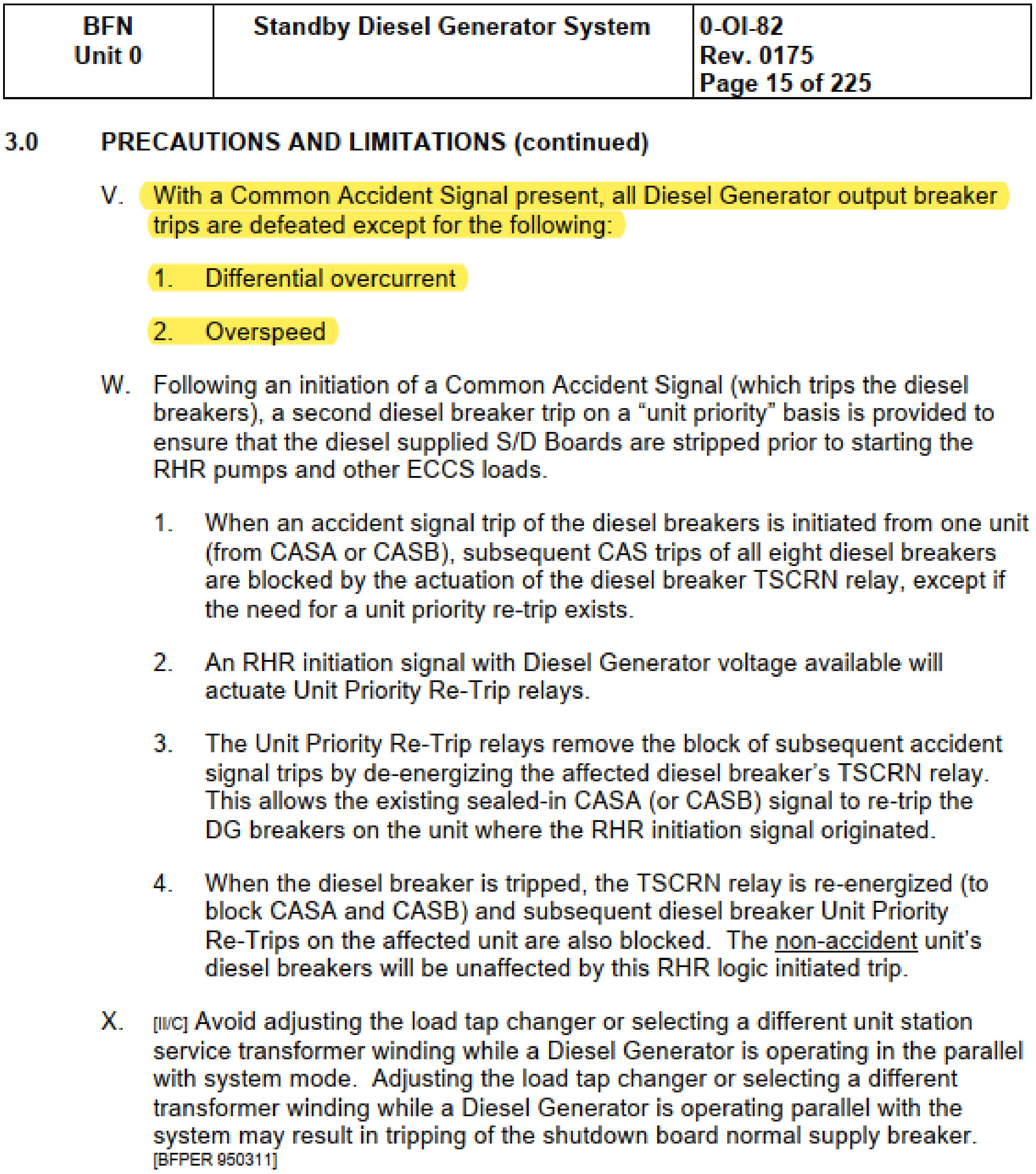
S. After operation of 4160V breakers, the charging spring is required to be verified to have recharged by verifying locally the breaker closing spring target indicates charged and the amber breaker spring charged light is on to ensure future breaker operation.

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.)



BFN **Unit 0**	**Standby Diesel Generator System**	**0-OI-82** **Rev. 0175** **Page 15 of 225**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

V. With a Common Accident Signal present, all Diesel Generator output breaker trips are defeated except for the following:

1. Differential overcurrent

2. Overspeed

W. Following an initiation of a Common Accident Signal (which trips the diesel breakers), a second diesel breaker trip on a “unit priority” basis is provided to ensure that the diesel supplied S/D Boards are stripped prior to starting the RHR pumps and other ECCS loads.

1. When an accident signal trip of the diesel breakers is initiated from one unit (from CASA or CASB), subsequent CAS trips of all eight diesel breakers are blocked by the actuation of the diesel breaker TSCRN relay, except if the need for a unit priority re-trip exists.

2. An RHR initiation signal with Diesel Generator voltage available will actuate Unit Priority Re-Trip relays.

3. The Unit Priority Re-Trip relays remove the block of subsequent accident signal trips by de-energizing the affected diesel breaker’s TSCRN relay. This allows the existing sealed-in CASA (or CASB) signal to re-trip the DG breakers on the unit where the RHR initiation signal originated.

4. When the diesel breaker is tripped, the TSCRN relay is re-energized (to block CASA and CASB) and subsequent diesel breaker Unit Priority Re-Trips on the affected unit are also blocked. The non-accident unit’s diesel breakers will be unaffected by this RHR logic initiated trip.

X. [II/C] Avoid adjusting the load tap changer or selecting a different unit station service transformer winding while a Diesel Generator is operating in the parallel with system mode. Adjusting the load tap changer or selecting a different transformer winding while a Diesel Generator is operating parallel with the system may result in tripping of the shutdown board normal supply breaker. [BFPER 950311]

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

Examination Outline Cross-reference:

223001 (SF5 PCS) Primary Containment and Auxiliaries

A3.04 (10CFR 55.41.7)

Ability to monitor automatic operation of the Primary Containment System and Auxiliaries, including:

- Containment/drywell response during LOCA

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	223001A3.04	
Importance Rating	4.2	------

Proposed Question: **# 56**

The Drywell to Suppression Chamber Vacuum Breakers will automatically open when differential pressure reaches a maximum of ___**(1)**____ (Drywell Pressure relative to Suppression Chamber Pressure).

Drywell to Suppression Chamber Vacuum Breaker position can be monitored on Panel ___**(2)**____.

A. (1) 0.5 psid
(2) 2-9-3

B. (1) 0.5 psid
(2) 2-9-4

C. (1) 0.25 psid
(2) 2-9-3

D. (1) 0.25 psid
(2) 2-9-4

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 2-SR-3.6.1.6.3, Drywell Pressure Suppression Chamber Vacuum Breaker Inspection and Test, each Drywell to Suppression Chamber vacuum breaker (12) shall be tested to verify the differential pressure required to open each breaker is less than or equal to 0.5 psid. Thus, 0.5 psid is the maximum differential pressure allowed by the Tech Spec for operation of the vacuum breaker. For second part, Drywell to Suppression Chamber Vacuum Breaker position is monitored on Panel 2-9-3. There are lights on the upper portion of the panel. There are also lights on the intermediate portion with test pushbuttons.

B INCORRECT: First part is correct *(See A).* Second part is incorrect, but plausible in that there are numerous position indications located on Panel 2-9-4 that are directly adjacent to the vacuum breaker position indications and controls on Panel 2-9-3. Panel 2-9-4 has position indications for numerous containment valves. They have inputs for most PCIVs and SCIVs for PCIS success for each PCIS group.

C INCORRECT**:** First part is incorrect but plausible in that prior to purging the Drywell/Suppression Chamber, the pressure in the Drywell/Suppression Chamber shall be reduced to less than 0.25 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).*

RO Level Justification: Tests the candidate's ability to monitor automatic operation of the Drywell to Suppression Chamber Vacuum Breakers during a LOCA. This question is rated as Memory due to the requirement to strictly recall facts related to the Primary Containment System design and components.

Technical Reference(s):	U2 Tech Spec 3.6.1.6, Amend. 338	(Attach if not previously provided)
	2-SR-3.6.1.6.3, Rev. 8	
	OPL171.016, Rev. 25U1	
	2-OI-64, Rev. 129	

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.016, Obj. 4a (As available)

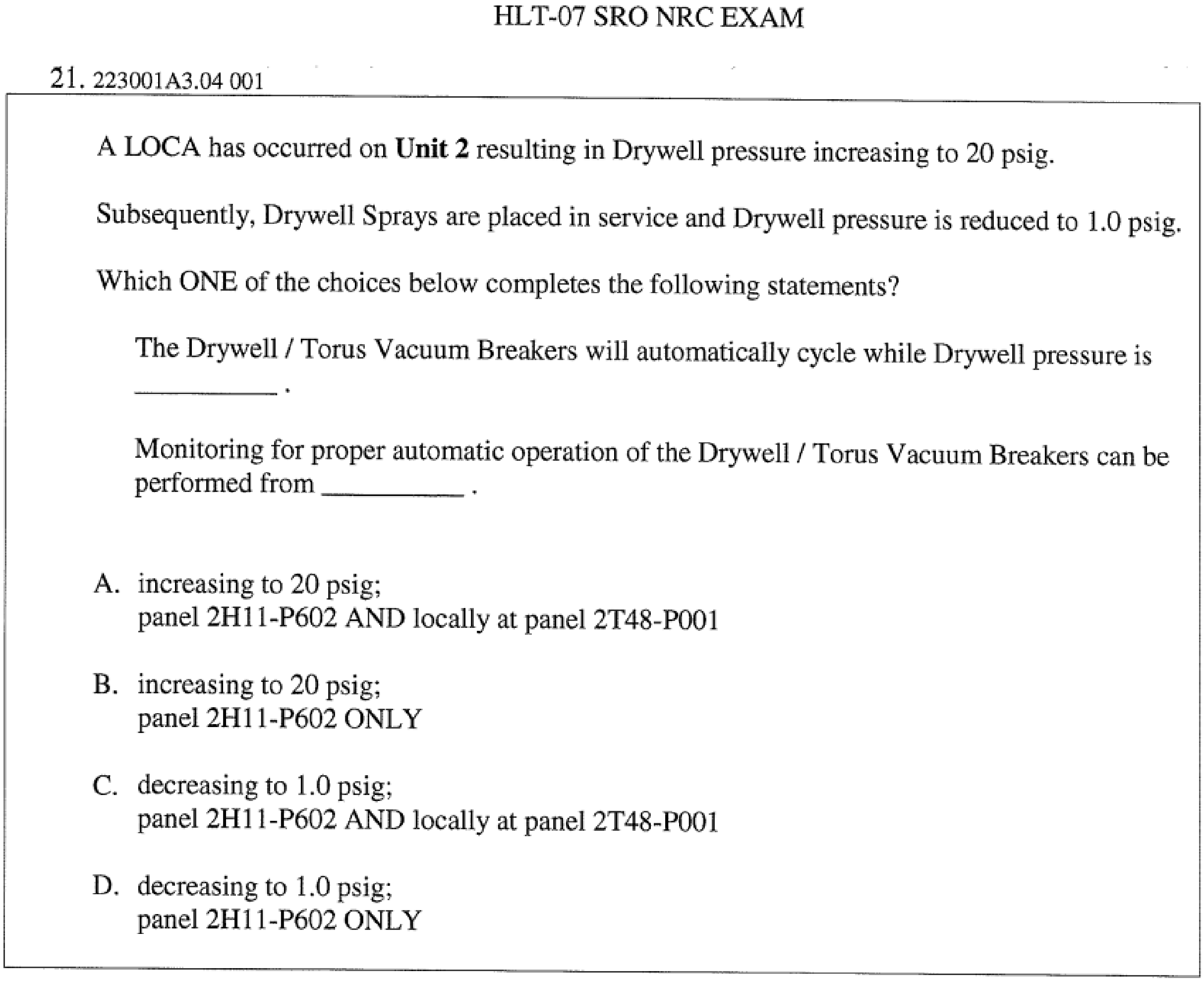
Question Source:	Bank #	Hatch HLT07 #21	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2012	

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

10 CFR Part 55 Content:	55.41	X
	55.43	

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

Copy of Bank Question:



HLT-07 SRO NRC EXAM

21. 223001A3.04 001

A LOCA has occurred on **Unit 2** resulting in Drywell pressure increasing to 20 psig.

Subsequently, Drywell Sprays are placed in service and Drywell pressure is reduced to 1.0 psig.

Which ONE of the choices below completes the following statements?

The Drywell / Torus Vacuum Breakers will automatically cycle while Drywell pressure is ___________ .

Monitoring for proper automatic operation of the Drywell / Torus Vacuum Breakers can be performed from ___________ .

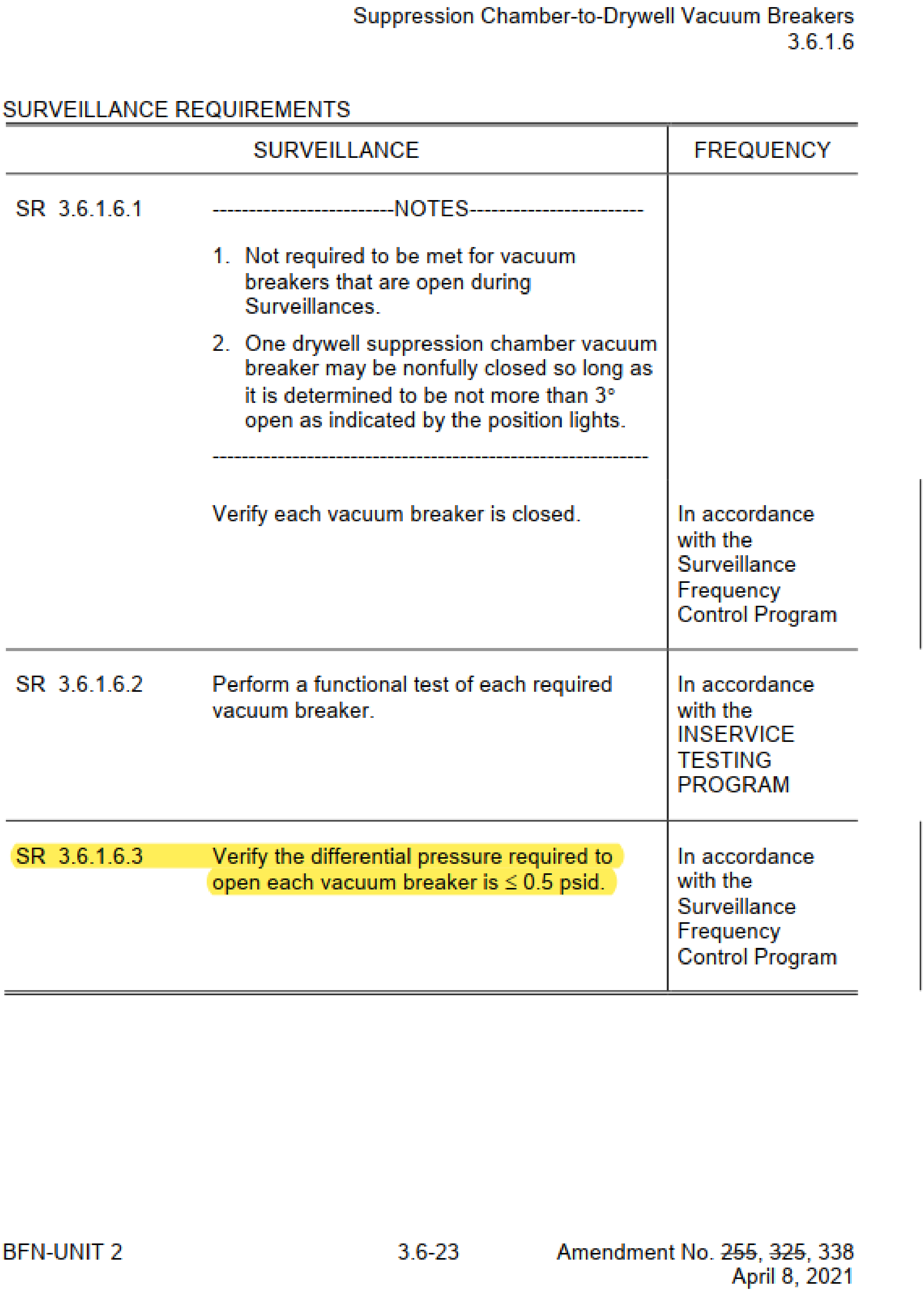
A. increasing to 20 psig;
panel 2H11-P602 AND locally at panel 2T48-P001

B. increasing to 20 psig;
panel 2H11-P602 ONLY

C. decreasing to 1.0 psig;
panel 2H11-P602 AND locally at panel 2T48-P001

D. decreasing to 1.0 psig;
panel 2H11-P602 ONLY

Excerpt from U2 Tech Spec 3.6.1.6:



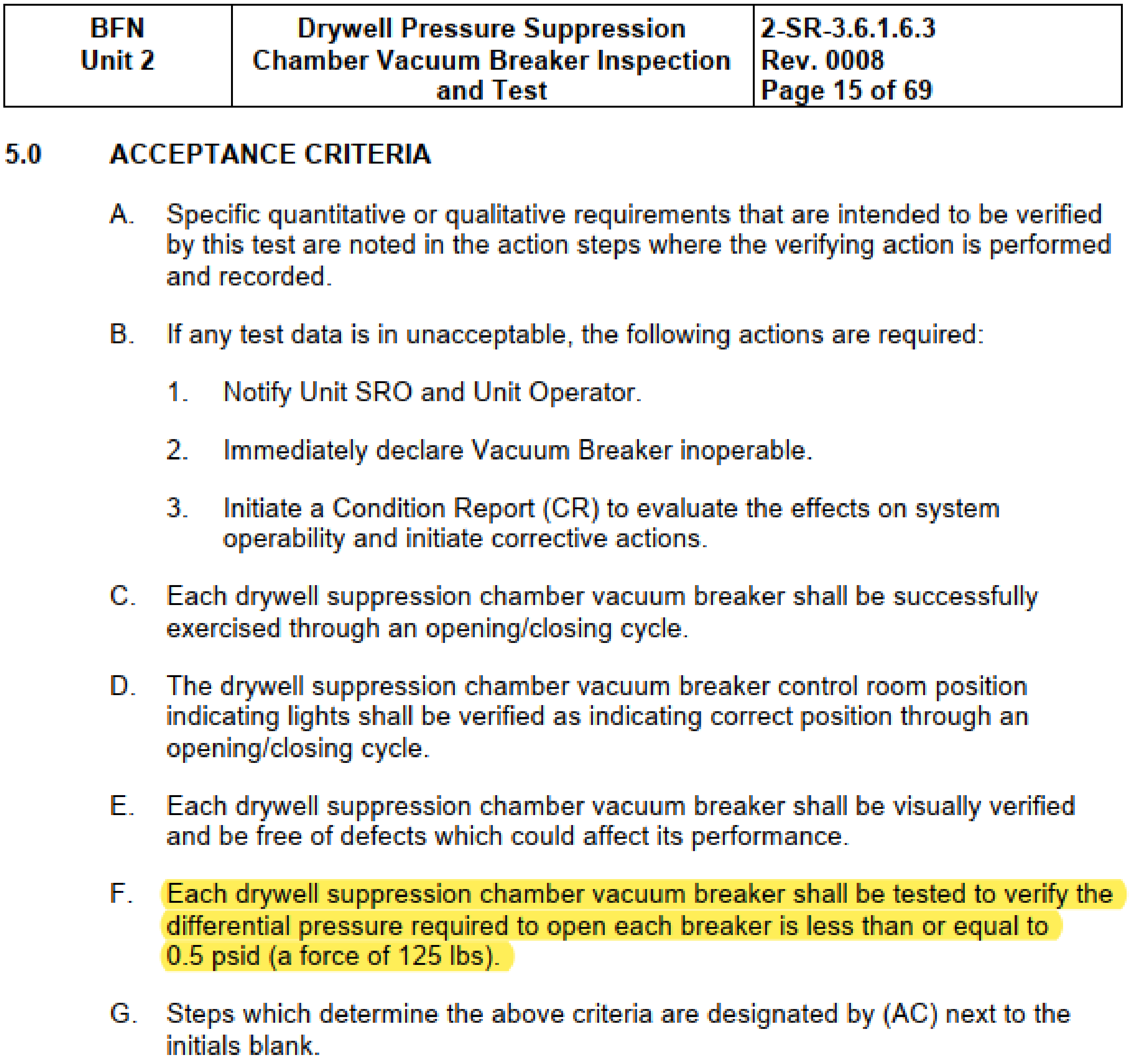
Suppression Chamber-to-Drywell Vacuum Breakers
3.6.1.6

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.6.1	-------------------------NOTES------------------------ 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. One drywell suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights. --- Verify each vacuum breaker is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.6.2	Perform a functional test of each required vacuum breaker.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.6.3	Verify the differential pressure required to open each vacuum breaker is ≤ 0.5 psid.	In accordance with the Surveillance Frequency Control Program

BFN-UNIT 2 3.6-23 Amendment No. ~~255~~, ~~325~~, 338
April 8, 2021

Excerpts from 2-SR-3.6.1.6.3:



BFN **Unit 2**	**Drywell Pressure Suppression Chamber Vacuum Breaker Inspection and Test**	**2-SR-3.6.1.6.3** **Rev. 0008** **Page 15 of 69**

5.0 ACCEPTANCE CRITERIA

A. Specific quantitative or qualitative requirements that are intended to be verified by this test are noted in the action steps where the verifying action is performed and recorded.

B. If any test data is in unacceptable, the following actions are required:

   1. Notify Unit SRO and Unit Operator.

   2. Immediately declare Vacuum Breaker inoperable.

   3. Initiate a Condition Report (CR) to evaluate the effects on system operability and initiate corrective actions.

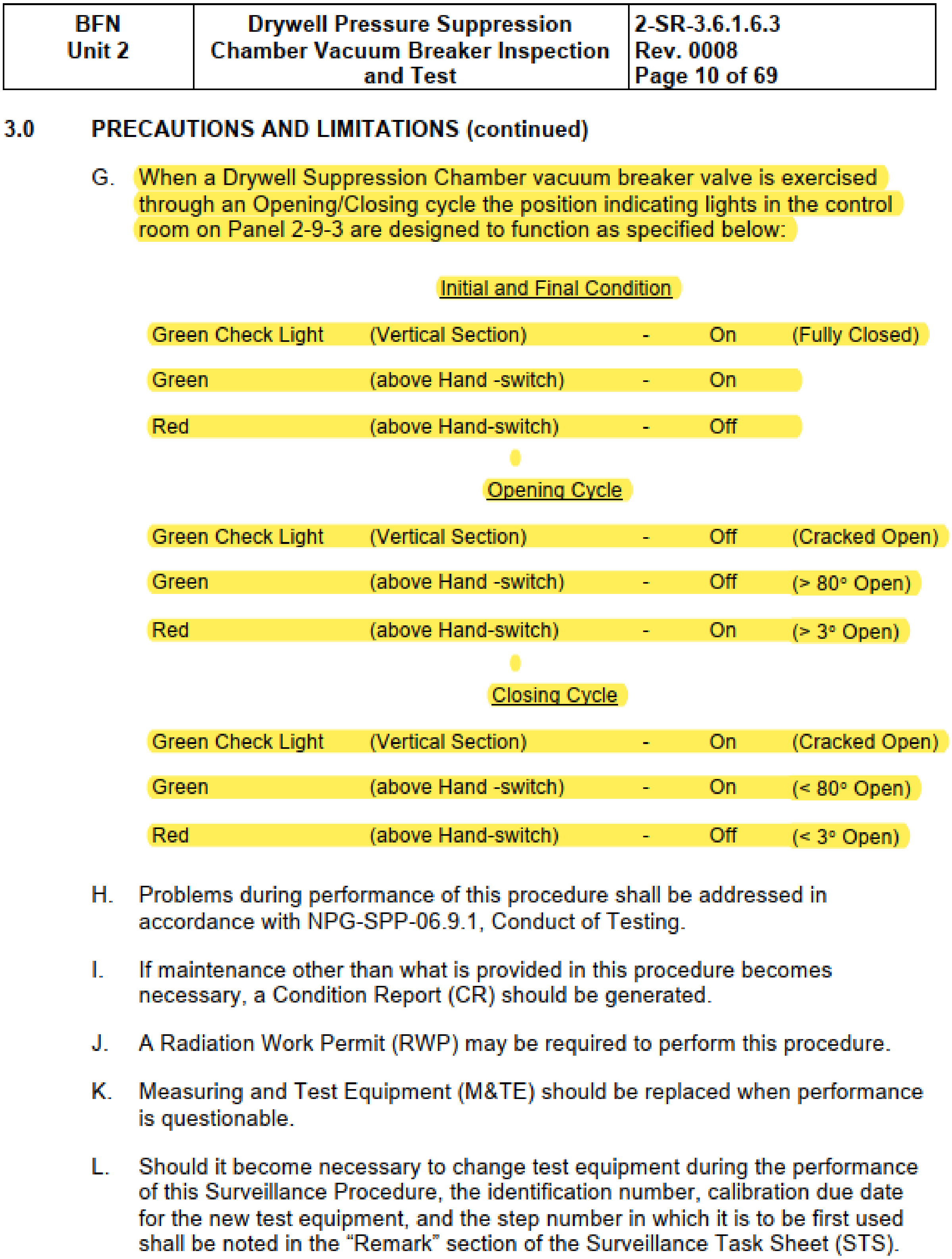
C. Each drywell suppression chamber vacuum breaker shall be successfully exercised through an opening/closing cycle.

D. The drywell suppression chamber vacuum breaker control room position indicating lights shall be verified as indicating correct position through an opening/closing cycle.

E. Each drywell suppression chamber vacuum breaker shall be visually verified and be free of defects which could affect its performance.

F. Each drywell suppression chamber vacuum breaker shall be tested to verify the differential pressure required to open each breaker is less than or equal to 0.5 psid (a force of 125 lbs).

G. Steps which determine the above criteria are designated by (AC) next to the initials blank.



BFN Unit 2	**Drywell Pressure Suppression Chamber Vacuum Breaker Inspection and Test**	**2-SR-3.6.1.6.3 Rev. 0008 Page 10 of 69**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

G. When a Drywell Suppression Chamber vacuum breaker valve is exercised through an Opening/Closing cycle the position indicating lights in the control room on Panel 2-9-3 are designed to function as specified below:

Initial and Final Condition

Green Check Light	(Vertical Section)	-	On	(Fully Closed)
Green	(above Hand -switch)	-	On	
Red	(above Hand-switch)	-	Off	

Opening Cycle

Green Check Light	(Vertical Section)	-	Off	(Cracked Open)
Green	(above Hand -switch)	-	Off	(> 80° Open)
Red	(above Hand-switch)	-	On	(> 3° Open)

Closing Cycle

Green Check Light	(Vertical Section)	-	On	(Cracked Open)
Green	(above Hand -switch)	-	On	(< 80° Open)
Red	(above Hand-switch)	-	Off	(< 3° Open)

H. Problems during performance of this procedure shall be addressed in accordance with NPG-SPP-06.9.1, Conduct of Testing.

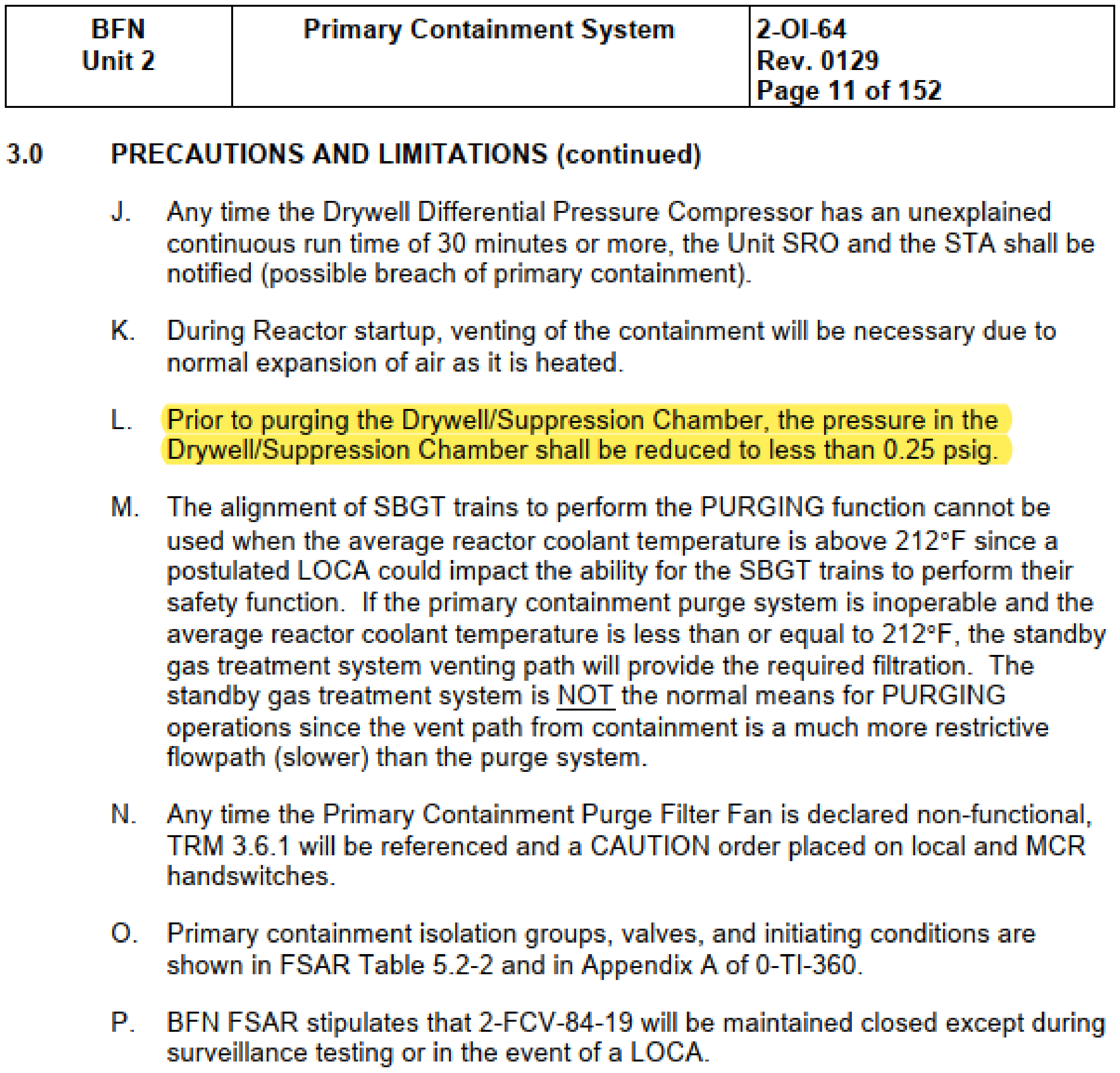
I. If maintenance other than what is provided in this procedure becomes necessary, a Condition Report (CR) should be generated.

J. A Radiation Work Permit (RWP) may be required to perform this procedure.

K. Measuring and Test Equipment (M&TE) should be replaced when performance is questionable.

L. Should it become necessary to change test equipment during the performance of this Surveillance Procedure, the identification number, calibration due date for the new test equipment, and the step number in which it is to be first used shall be noted in the “Remark” section of the Surveillance Task Sheet (STS).

Excerpt from 2-OI-64: Supports Distractors C(1), D(1)



BFN **Unit 2**	**Primary Containment System**	**2-OI-64** **Rev. 0129** **Page 11 of 152**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

J. Any time the Drywell Differential Pressure Compressor has an unexplained continuous run time of 30 minutes or more, the Unit SRO and the STA shall be notified (possible breach of primary containment).

K. During Reactor startup, venting of the containment will be necessary due to normal expansion of air as it is heated.

L. Prior to purging the Drywell/Suppression Chamber, the pressure in the Drywell/Suppression Chamber shall be reduced to less than 0.25 psig.

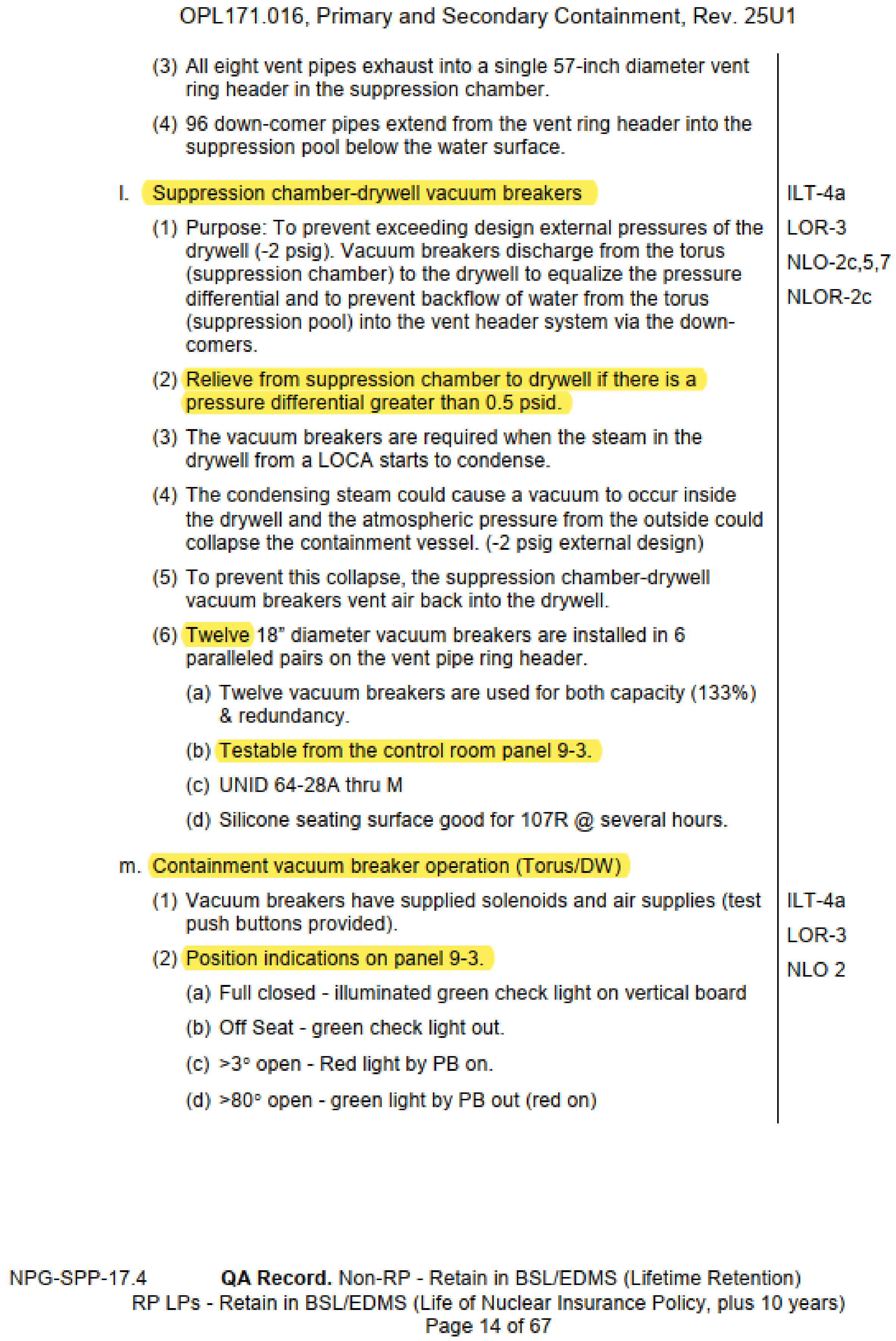
M. The alignment of SBGT trains to perform the PURGING function cannot be used when the average reactor coolant temperature is above 212°F since a postulated LOCA could impact the ability for the SBGT trains to perform their safety function. If the primary containment purge system is inoperable and the 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.

N. Any time the Primary Containment Purge Filter Fan is declared non-functional, TRM 3.6.1 will be referenced and a CAUTION order placed on local and MCR handswitches.

O. Primary containment isolation groups, valves, and initiating conditions are shown in FSAR Table 5.2-2 and in Appendix A of 0-TI-360.

P. BFN FSAR stipulates that 2-FCV-84-19 will be maintained closed except during surveillance testing or in the event of a LOCA.

Excerpt from OPL171.016 Lesson Plan:



OPL171.016, Primary and Secondary Containment, Rev. 25U1

(3) All eight vent pipes exhaust into a single 57-inch diameter vent ring header in the suppression chamber.

(4) 96 down-comer pipes extend from the vent ring header into the suppression pool below the water surface.

l. Suppression chamber-drywell vacuum breakers

ILT-4a

LOR-3

NLO-2c,5,7

NLOR-2c

(1) Purpose: To prevent exceeding design external pressures of the drywell (-2 psig). Vacuum breakers discharge from the torus (suppression chamber) to the drywell to equalize the pressure differential and to prevent backflow of water from the torus (suppression pool) into the vent header system via the down-comers.

(2) Relieve from suppression chamber to drywell if there is a pressure differential greater than 0.5 psid.

(3) The vacuum breakers are required when the steam in the drywell from a LOCA starts to condense.

(4) The condensing steam could cause a vacuum to occur inside the drywell and the atmospheric pressure from the outside could collapse the containment vessel. (-2 psig external design)

(5) To prevent this collapse, the suppression chamber-drywell vacuum breakers vent air back into the drywell.

(6) Twelve 18" diameter vacuum breakers are installed in 6 paralleled pairs on the vent pipe ring header.

  (a) Twelve vacuum breakers are used for both capacity (133%) & redundancy.

  (b) Testable from the control room panel 9-3.

  (c) UNID 64-28A thru M

  (d) Silicone seating surface good for 107R @ several hours.

m. Containment vacuum breaker operation (Torus/DW)

ILT-4a

LOR-3

NLO 2

(1) Vacuum breakers have supplied solenoids and air supplies (test push buttons provided).

(2) Position indications on panel 9-3.

  (a) Full closed - illuminated green check light on vertical board

  (b) Off Seat - green check light out.

  (c) >3° open - Red light by PB on.

  (d) >80° open - green light by PB out (red on)

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

Examination Outline Cross-reference:	Level	RO	SRO
230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Spray Mode	Tier #	2	------
K1.01 (10CFR 55.41.7)	Group #	2	------
Knowledge of the physical connections and/or cause and effect relationships between the RHR/LPCI: Torus/Suppression Pool Spray Mode and the following systems:	K/A #	230000K1.01	
- Primary containment	Importance Rating	3.9	------

Proposed Question: **# 57**

Unit 2 is operating at 100% RTP with the following plant conditions:

- A LOCA has occurred
- Reactor Water Level is (-) 75 inches and lowering
- Reactor Pressure is 350 psig and lowering
- Drywell Pressure is 13 psig and lowering
- Suppression Chamber Pressure is 14 psig and lowering

Subsequently:

- 2-XS-74-122, RHR SYSTEM I LPCI 2/3 CORE HEIGHT OVERRIDE has been placed in MANUAL OVERRIDE
- 2-XS-74-121, RHR SYS I CONTAINMENT SPRAY/COOLING VALVE SELECT has been placed in SELECT

The RHR Loop I Containment Spray Valves will **AUTOMATICALLY** close when __________.

A. RPV Level drops below (-) 122 inches

B. RPV Level drops below (-) 183 inches

C. Drywell Pressure drops below 1.96 psig

D. Suppression Chamber Pressure drops below 1.96 psig

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that an Accident Signal is required to close the Containment Spray Valves automatically, however it will not occur by itself. An Accident/LPCI Signal in conjunction with Drywell Pressure below 1.96 psig is required.

B INCORRECT: Incorrect but plausible in that RPV Level less than (-) 183 inches is a permissive for spraying the Drywell and Suppression Chamber. If RPV Water Level is below (-) 183 inches, 2-XS-74-122, RHR SYSTEM I LPCI 2/3 CORE HEIGHT OVERRIDE must be placed in OVERRIDE. However, this will not result in automatic closure of Containment Sprays if already spraying and the switch is placed in OVERRIDE as stated in the stem.

C **CORRECT**: *(See attached)* While spraying the Drywell, there are two interlocks that must be met for Containment Spray Valves to automatically close. Given a LPCI Initiation signal is present (Drywell Pressure > 2.45 psig and Reactor Pressure < 450 psig), the RHR Loop I Containment Spray Valves will automatically close when Drywell Pressure drops below 1.96 psig. This is referenced in both 2-OI-74, RHR System and 2-EOI Appendix-17B, RHR System Operation Drywell Sprays and illustrated on PIP-95-136, Containment Spray Interlock which is on Panel 2-9-3 above RHR Loop II.

D INCORRECT: Incorrect but plausible if the candidate does not recall that the Containment Spray permissive signal is Drywell Pressure, not Suppression Chamber Pressure which does not input into the logic.

RO Level Justification: Tests the candidate's knowledge of the cause and effect relationship between RHR/LPCI and Containment Spray interlocks. This question is rated as C/A due to the requirement to correctly assemble given Primary Containment parameters from abnormal plant conditions with a LOCA present related to Containment Spray permissive logic. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-74, Rev. 189 (Attach if not previously provided)
PIP-95-136, Rev. 2
2-EOI Appendix-17B, Rev. 18

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.044, Obj. 4g (As available)

Question Source:

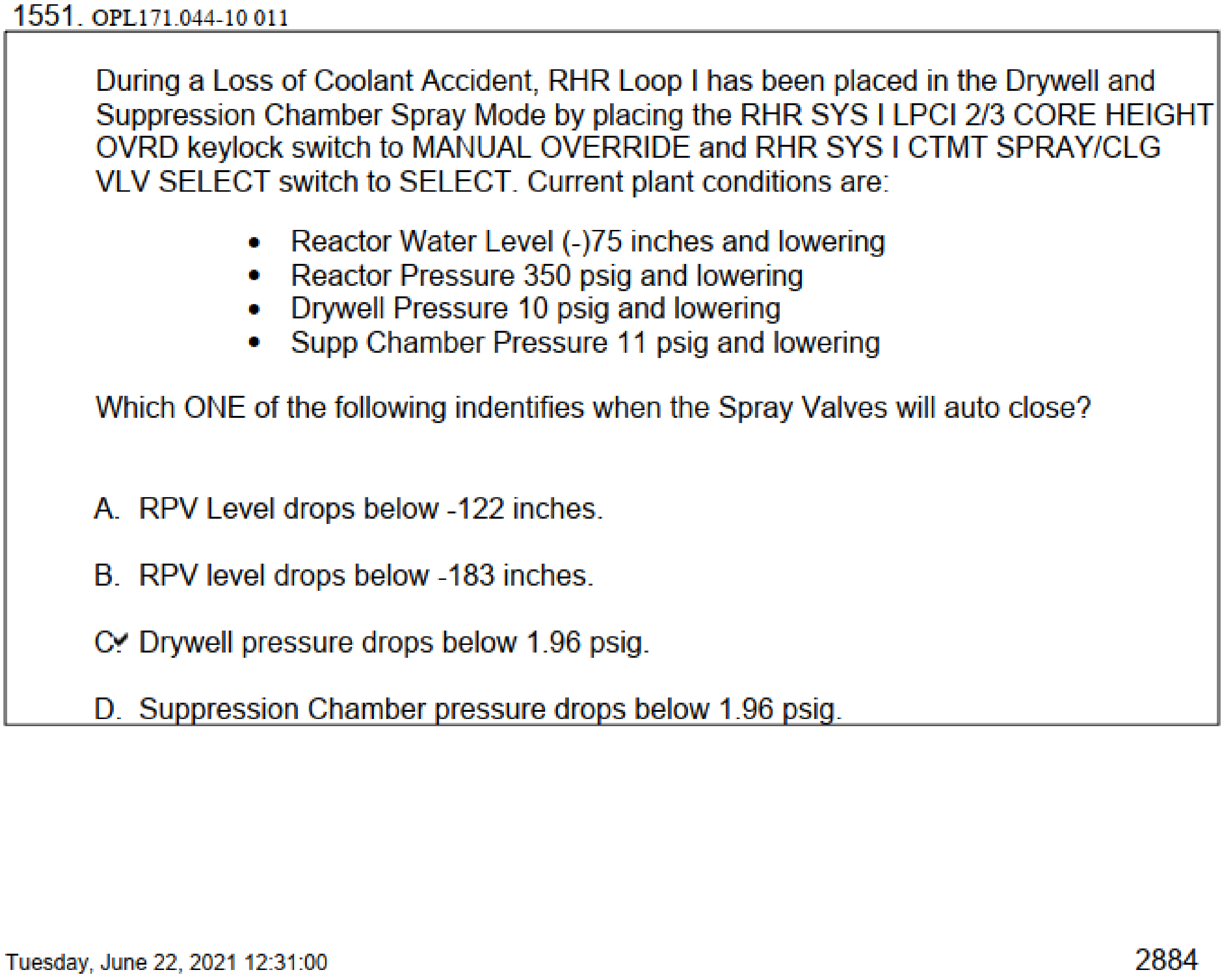
Question Source:	Bank #	OPL171.044-10 001 #1551	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

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

Comments:

Copy of Bank Question:



1551. OPL171.044-10 011

During a Loss of Coolant Accident, RHR Loop I has been placed in the Drywell and Suppression Chamber Spray Mode by placing the RHR SYS I LPCI 2/3 CORE HEIGHT OVRD keylock switch to MANUAL OVERRIDE and RHR SYS I CTMT SPRAY/CLG VLV SELECT switch to SELECT. Current plant conditions are:

- Reactor Water Level (-)75 inches and lowering
- Reactor Pressure 350 psig and lowering
- Drywell Pressure 10 psig and lowering
- Supp Chamber Pressure 11 psig and lowering

Which ONE of the following indentifies when the Spray Valves will auto close?

A. RPV Level drops below -122 inches.

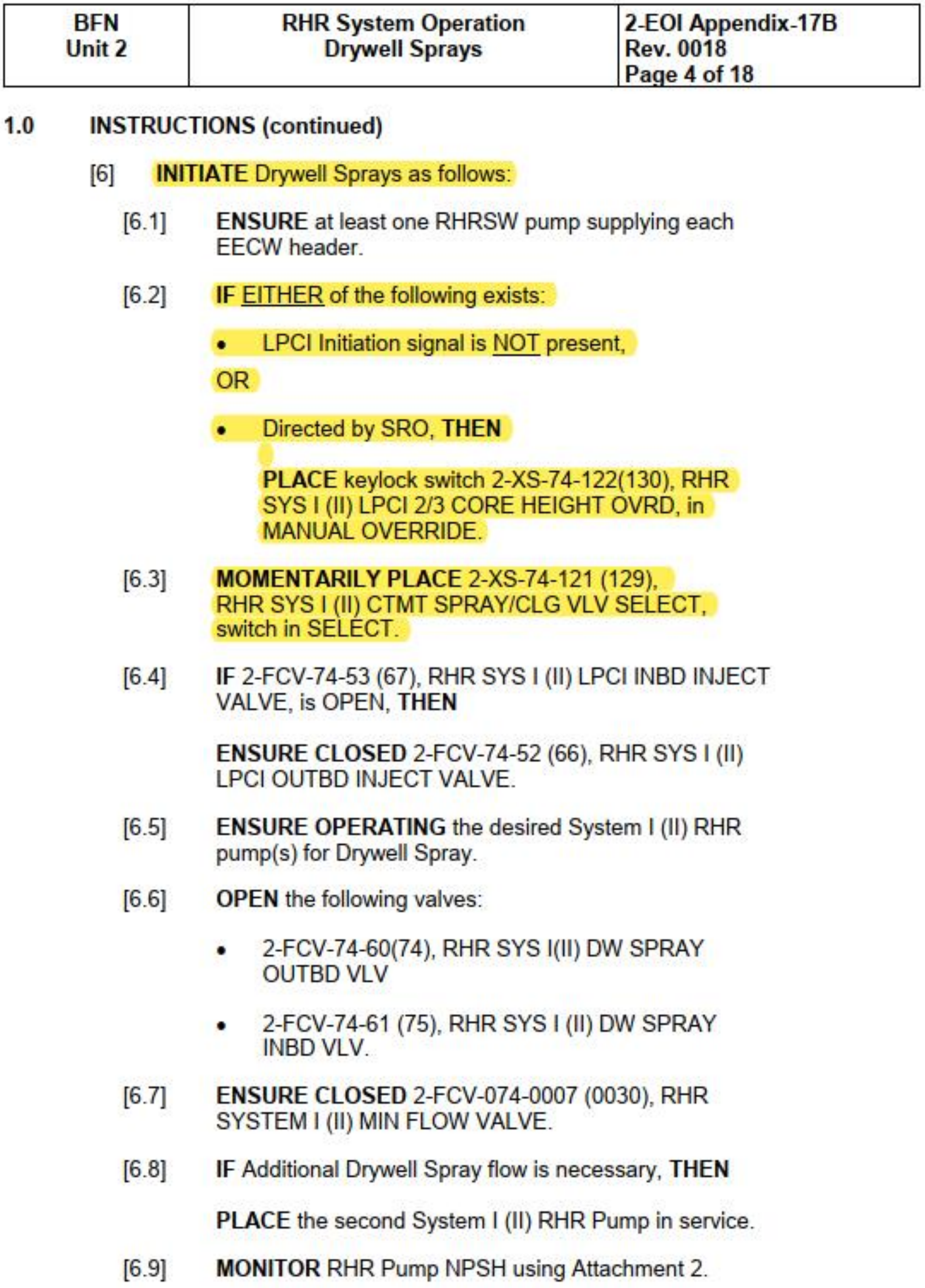
B. RPV level drops below -183 inches.

C✓ Drywell pressure drops below 1.96 psig.

D. Suppression Chamber pressure drops below 1.96 psig.

Tuesday, June 22, 2021 12:31:00 2884

Excerpt from 2-EOI Appendix-17B:



BFN **Unit 2**	**RHR System Operation** **Drywell Sprays**	**2-EOI Appendix-17B** **Rev. 0018** **Page 4 of 18**

1.0 INSTRUCTIONS (continued)

[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 SYS I (II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE.

[6.3] **MOMENTARILY PLACE** 2-XS-74-121 (129), RHR SYS I (II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT.

[6.4] **IF** 2-FCV-74-53 (67), RHR SYS I (II) LPCI INBD INJECT VALVE, is OPEN, **THEN**

ENSURE CLOSED 2-FCV-74-52 (66), RHR SYS I (II) LPCI OUTBD INJECT 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 SYS I(II) DW SPRAY OUTBD VLV
- 2-FCV-74-61 (75), RHR SYS I (II) DW SPRAY INBD VLV.

[6.7] **ENSURE CLOSED** 2-FCV-074-0007 (0030), RHR SYSTEM I (II) MIN FLOW VALVE.

[6.8] **IF** Additional Drywell Spray flow is necessary, **THEN**

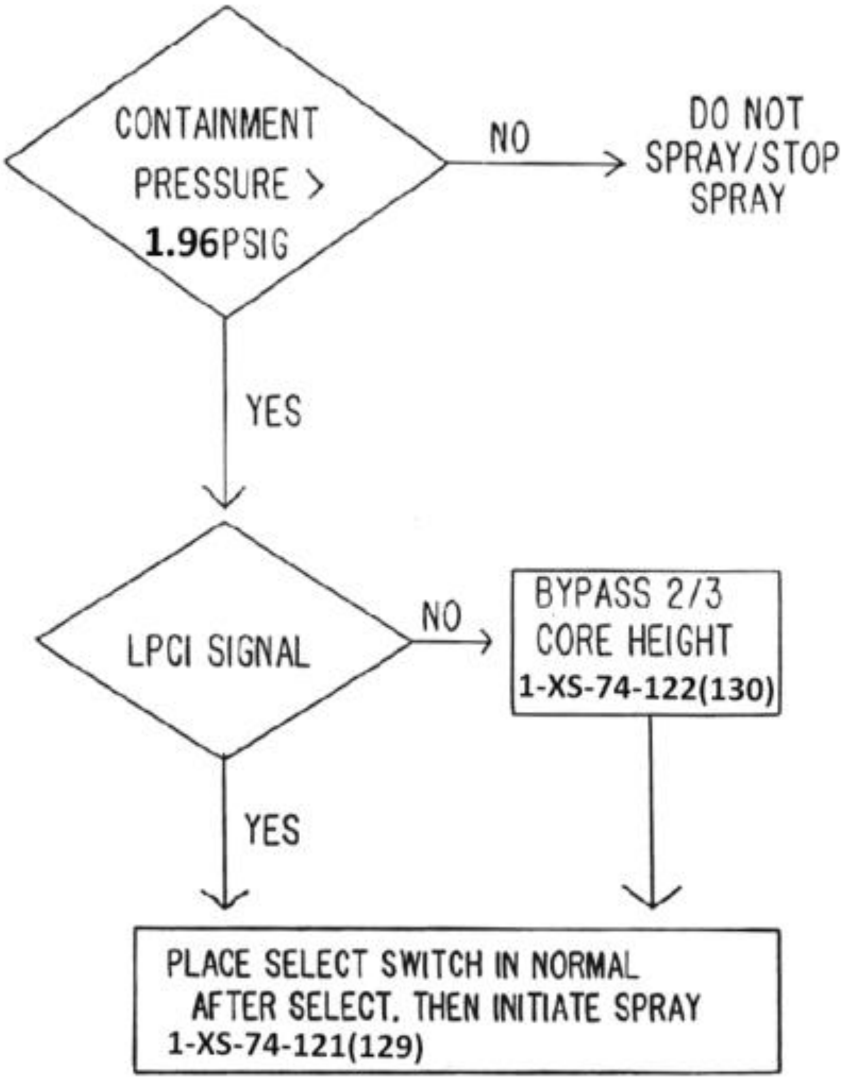
PLACE the second System I (II) RHR Pump in service.

[6.9] **MONITOR** RHR Pump NPSH using Attachment 2.

Excerpt from PIP-95-136: Illustrates the Containment Spray logic (located on Panel 2-9-3 above RHR Loop II)

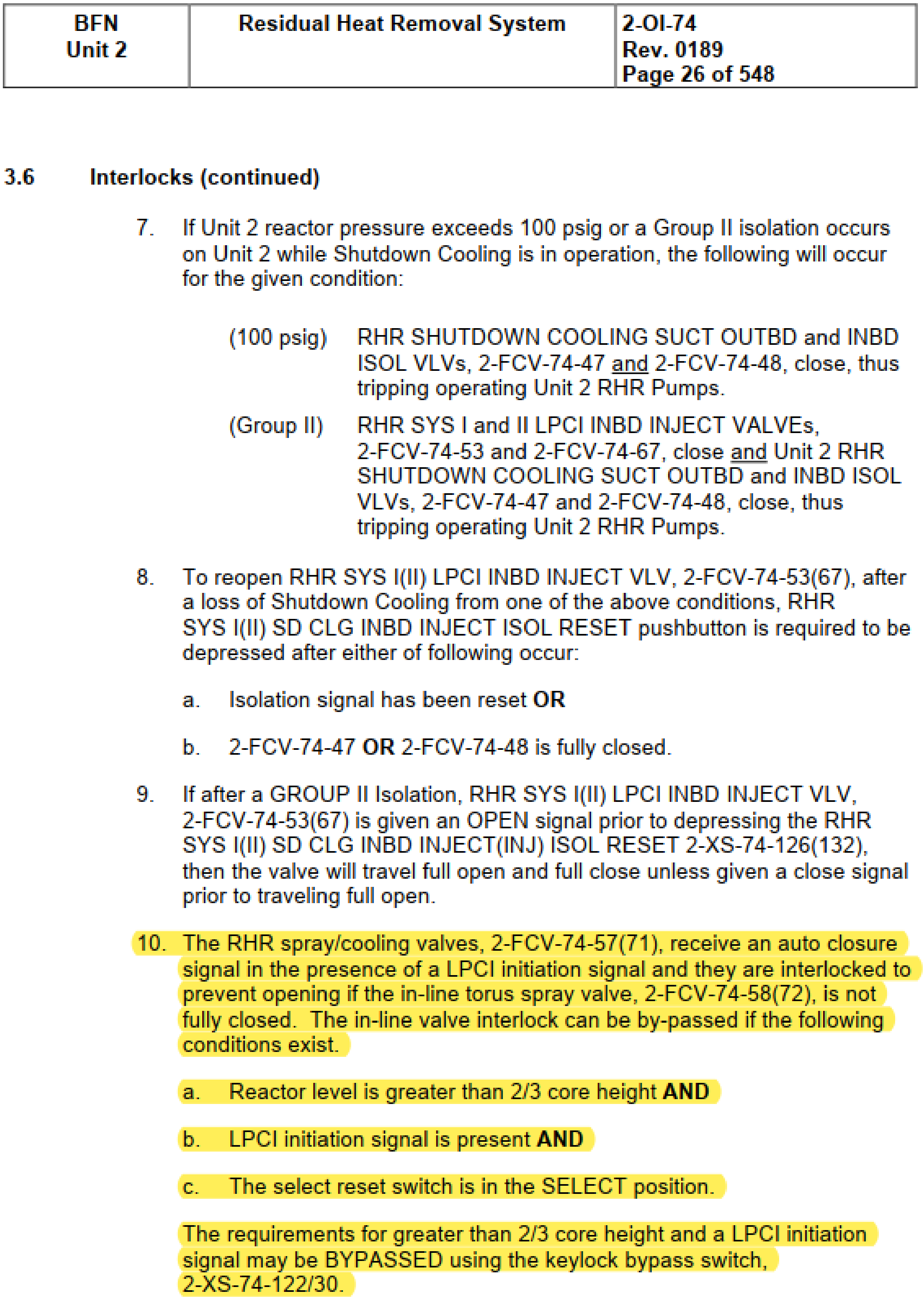
BFN **Unit 0**	**Component Labeling, Signs, Operator Aids, and Permanent Information Postings**	**0-TI-414** **Rev. 0006** **Page 28 of 67**

CONTAINMENT SPRAY INTERLOCK



PIP-95-136

Excerpts from 2-OI-74: Also supports Distractors A, B, and D



BFN Unit 2	**Residual Heat Removal System**	**2-OI-74 Rev. 0189 Page 26 of 548**

3.6 Interlocks (continued)

7. If Unit 2 reactor pressure exceeds 100 psig or a Group II isolation occurs on Unit 2 while Shutdown Cooling is in operation, the following will occur for the given condition:

(100 psig)	RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.
(Group II)	RHR SYS I and II LPCI INBD INJECT VALVEs, 2-FCV-74-53 and 2-FCV-74-67, close and Unit 2 RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.

8. To reopen RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT ISOL RESET pushbutton is required to be depressed after either of following occur:

   a. Isolation signal has been reset **OR**

   b. 2-FCV-74-47 **OR** 2-FCV-74-48 is fully closed.

9. If after a GROUP II Isolation, RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67) is given an OPEN signal prior to depressing the RHR SYS I(II) SD CLG INBD INJECT(INJ) ISOL RESET 2-XS-74-126(132), then the valve will travel full open and full close unless given a close signal prior to traveling full open.

10. The RHR spray/cooling valves, 2-FCV-74-57(71), receive an auto closure signal in the presence of a LPCI initiation signal and they are interlocked to prevent opening if the in-line torus spray valve, 2-FCV-74-58(72), is not fully closed. The in-line valve interlock can be by-passed if the following conditions exist.

    a. Reactor level is greater than 2/3 core height **AND**

    b. LPCI initiation signal is present **AND**

    c. The select reset switch is in the SELECT position.

    The requirements for greater than 2/3 core height and a LPCI initiation signal may be BYPASSED using the keylock bypass switch, 2-XS-74-122/30.

3.6 Interlocks (continued)

11. If primary containment cooling is desired with reactor level at less than2/3 core height, the keylock bypass switch is required to be placed in BYPASS before the select reset switch is placed in SELECT to ensure relay logic is made up.\

12. The RHR torus spray valves, 2-FCV-74-58(72), have the same in-line valve interlocks as those outlined in 3.6A 10 for the torus spray/cooling valves. Additionally these valves have an interlock preventing opening unless drywell pressure is ≥1.96 psig which cannot be bypassed.

13. The RHR torus cooling/test valves, 2-FCV-74-59(73), receive an auto closure signal in the presence of a LPCI initiation signal. Auto closure may be bypassed by the same conditions/actions outlined in Step 3.6A.10

14. The RHR containment spray valves, 2-FCV-74-60(74) and 61(75), have in-line valve interlocks similar to these described in Step 3.6A.10 through 3.6A.12 for the RHR torus spray valves 2-FCV-74-57(58) and 71(72).

15. If 2-FCV-74-59(73) LOCA CLOSURE TIME light (2-IL-74-59Y Loop I; 2-IL-74-73Y Loop II) on Panel 2-9-3 is extinguished due to its associated valve being opened, that Loop is inoperable for LPCI.

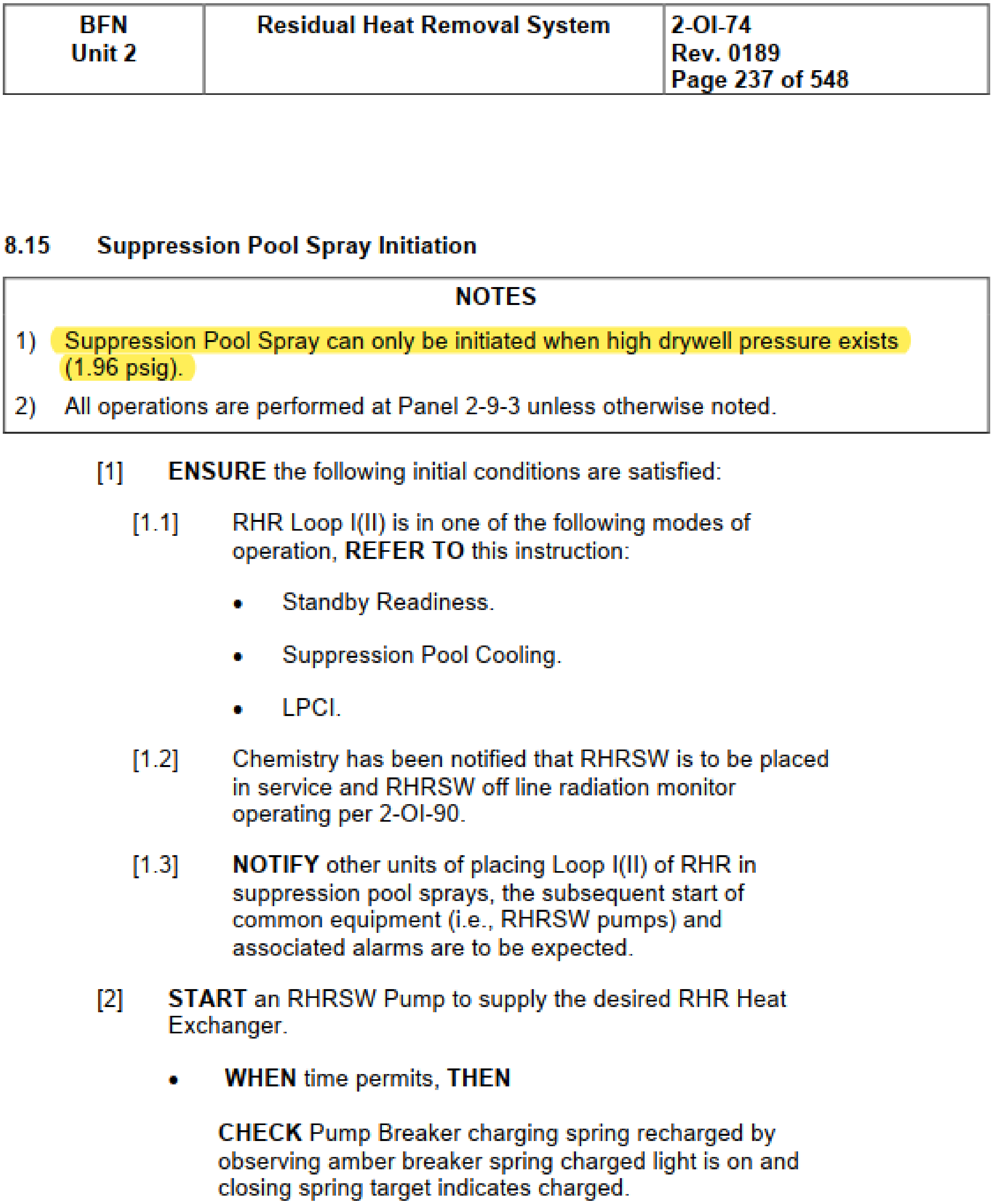
16. If 2-HS-74-148(149) RHR SYSTEM I (II) MIN FLOW INHIBIT switch is in the INHIBIT position, the pumps on that loop do not have automatic minimum flow protection.

17. If RHR pumps 2A(B) and 2C(D) SD COOLING SUCT VLV, 2-FCV-74-2(25) AND 13(36) are not fully closed, the RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV, 2-FCV-74-57(71) cannot be opened.

    a. 2-HS-074-0057D placed in BYPASS position, defeats the interlocks on 2-FCV-74-57 and the SD COOLING SUCT VLV 2-FCV-74-2 AND 13.

    b. 2-HS-074-0071C placed in BYPASS position, defeats the interlocks on 2-FCV-74-71 and the SD COOLING SUCT VLV 2-FCV-74-25 AND 36.

18. If RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV, 2-FCV-74-57(71) is not fully closed, the RHR pumps 2A(B) and 2C(D) SD COOLING SUCT VLV, 2-FCV-74-2(25) AND 13(36) cannot be opened.



BFN **Unit 2**	**Residual Heat Removal System**	**2-OI-74** **Rev. 0189** **Page 237 of 548**

8.15 Suppression Pool Spray Initiation

NOTES
1) Suppression Pool Spray can only be initiated when high drywell pressure exists (1.96 psig). 2) All operations are performed at Panel 2-9-3 unless otherwise noted.

[1] **ENSURE** the following initial conditions are satisfied:

[1.1] RHR Loop I(II) is in one of the following modes of operation, **REFER TO** this instruction:

- Standby Readiness.
- Suppression Pool Cooling.
- LPCI.

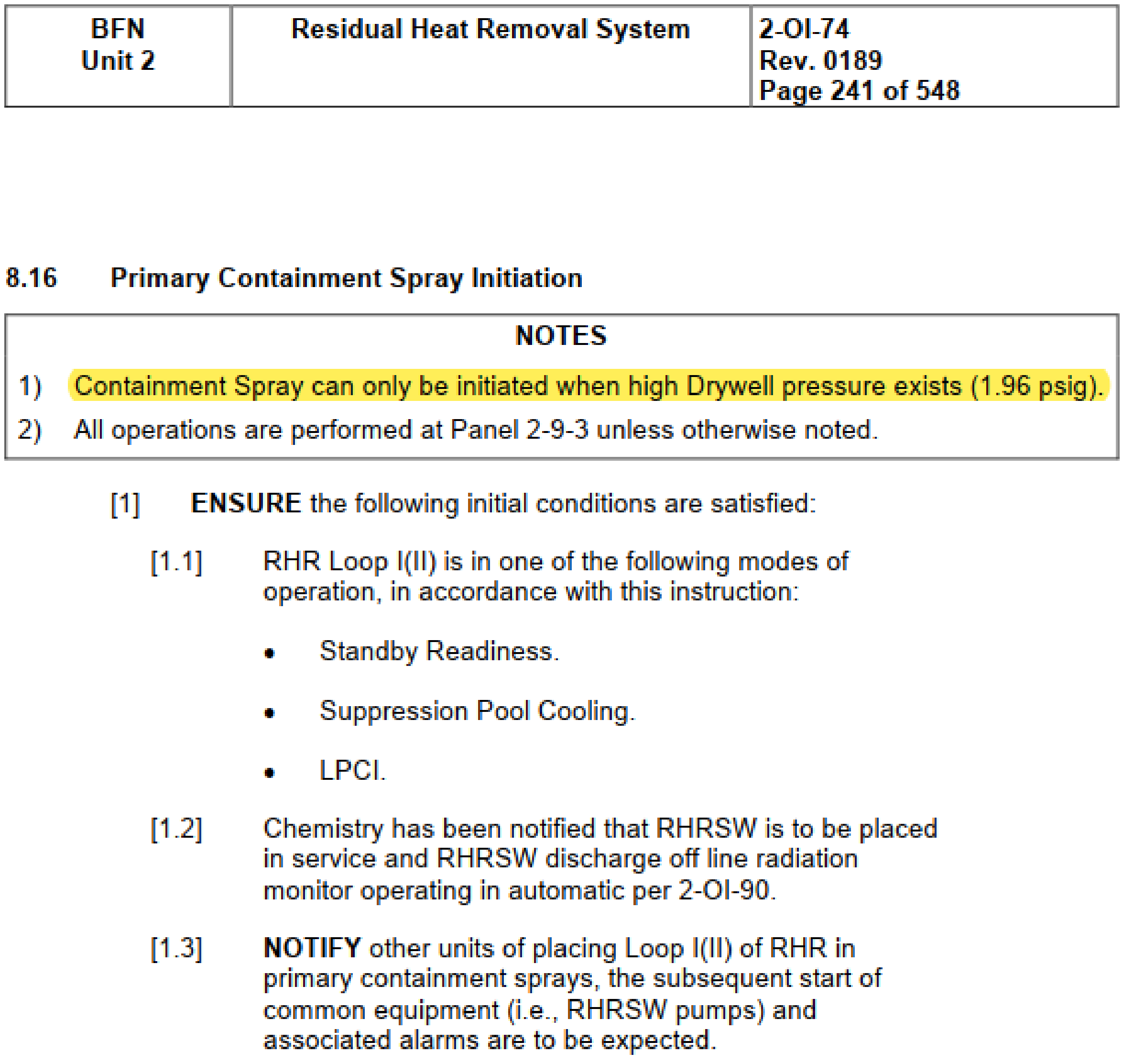
[1.2] Chemistry has been notified that RHRSW is to be placed in service and RHRSW off line radiation monitor operating per 2-OI-90.

[1.3] **NOTIFY** other units of placing Loop I(II) of RHR in suppression pool sprays, the subsequent start of common equipment (i.e., RHRSW pumps) and associated alarms are to be expected.

[2] **START** an RHRSW Pump to supply the desired RHR Heat Exchanger.

- **WHEN** time permits, **THEN**

  CHECK Pump Breaker charging spring recharged by observing amber breaker spring charged light is on and closing spring target indicates charged.



BFN **Unit 2**	**Residual Heat Removal System**	**2-OI-74** **Rev. 0189** **Page 241 of 548**

8.16 Primary Containment Spray Initiation

NOTES

1) Containment Spray can only be initiated when high Drywell pressure exists (1.96 psig).

2) All operations are performed at Panel 2-9-3 unless otherwise noted.

[1] **ENSURE** the following initial conditions are satisfied:

[1.1] RHR Loop I(II) is in one of the following modes of operation, in accordance with this instruction:

- Standby Readiness.
- Suppression Pool Cooling.
- LPCI.

[1.2] Chemistry has been notified that RHRSW is to be placed in service and RHRSW discharge off line radiation monitor operating in automatic per 2-OI-90.

[1.3] **NOTIFY** other units of placing Loop I(II) of RHR in primary containment sprays, the subsequent start of common equipment (i.e., RHRSW pumps) and associated alarms are to be expected.

Examination Outline Cross-reference:

233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup

A2.09 (10CFR 55.41.5)

Ability to (a) predict the impacts of the following on the Fuel Pool Cooling and Cleanup and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- AC electrical power failures

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	233000A2.09	
Importance Rating	3.4	------

Proposed Question: **# 58**

A trip of 4KV Shutdown Board **(1)** would cause a loss of 1B Fuel Pool Cooling Pump.

The **MINIMUM** Fuel Pool Temperature which requires entry into 1-EOI-3, Secondary Containment Control, is **(2)** .

A. (1) B
(2) 125 ºF

B. (1) B
(2) 150 ºF

C. (1) C
(2) 125 ºF

D. (1) C
(2) 150 ºF

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that it is reasonable to assume that the power supply to 480V Shutdown Board 1B which provides power to 1B Fuel Pool Cooling Pump would be 4KV Shutdown Board B. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that Fuel Pool Water Temperature is limited to 150 ºF in accordance with Technical Requirements Manual (TRM) 3.9.2.

C **CORRECT**: (*See attached*) In accordance with 0-OI-57B, 480V/240V AC Electrical System, 4KV Shutdown Board C is the normal power supply to 480V Shutdown Board 1B which provides power to 1B Fuel Pool Cooling Pump. For second part, in accordance with 1-EOI-3, Secondary Containment Control, entry is required when Fuel Pool Temperature reaches 125 ºF.

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 predict the impact of a loss of AC Electrical Power on Fuel Pool Cooling Pumps and when the related Emergency Operating Instructions (EOIs) would be entered to mitigate the associated high Fuel Pool Temperature. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): (Attach if not previously provided)
1-EOI-3, Rev. 6
0-OI-57B, Rev. 200
TRM 3.9.2, Rev.0
0-AOI-57-1A, Rev. 114
OPL171.052, Rev. 15U2
PIP-02-03, Rev. 9/16/2020

Proposed references to be provided to applicants during examination: **NONE**

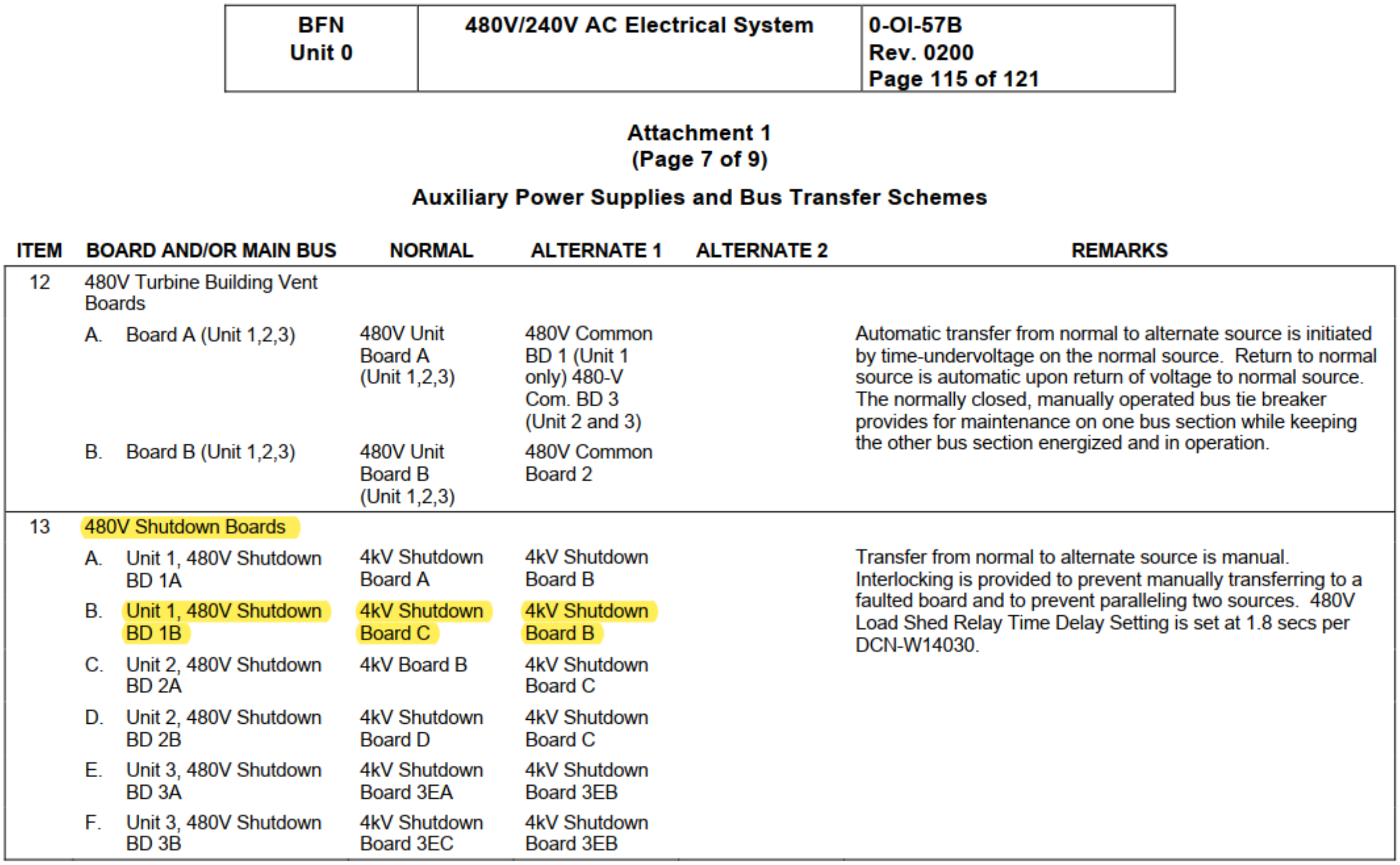
Learning Objective: OPL171.052 Obj. 8d (As available)

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 0-OI-57B: Illustrates 480V Shutdown Board 1B normal power supply



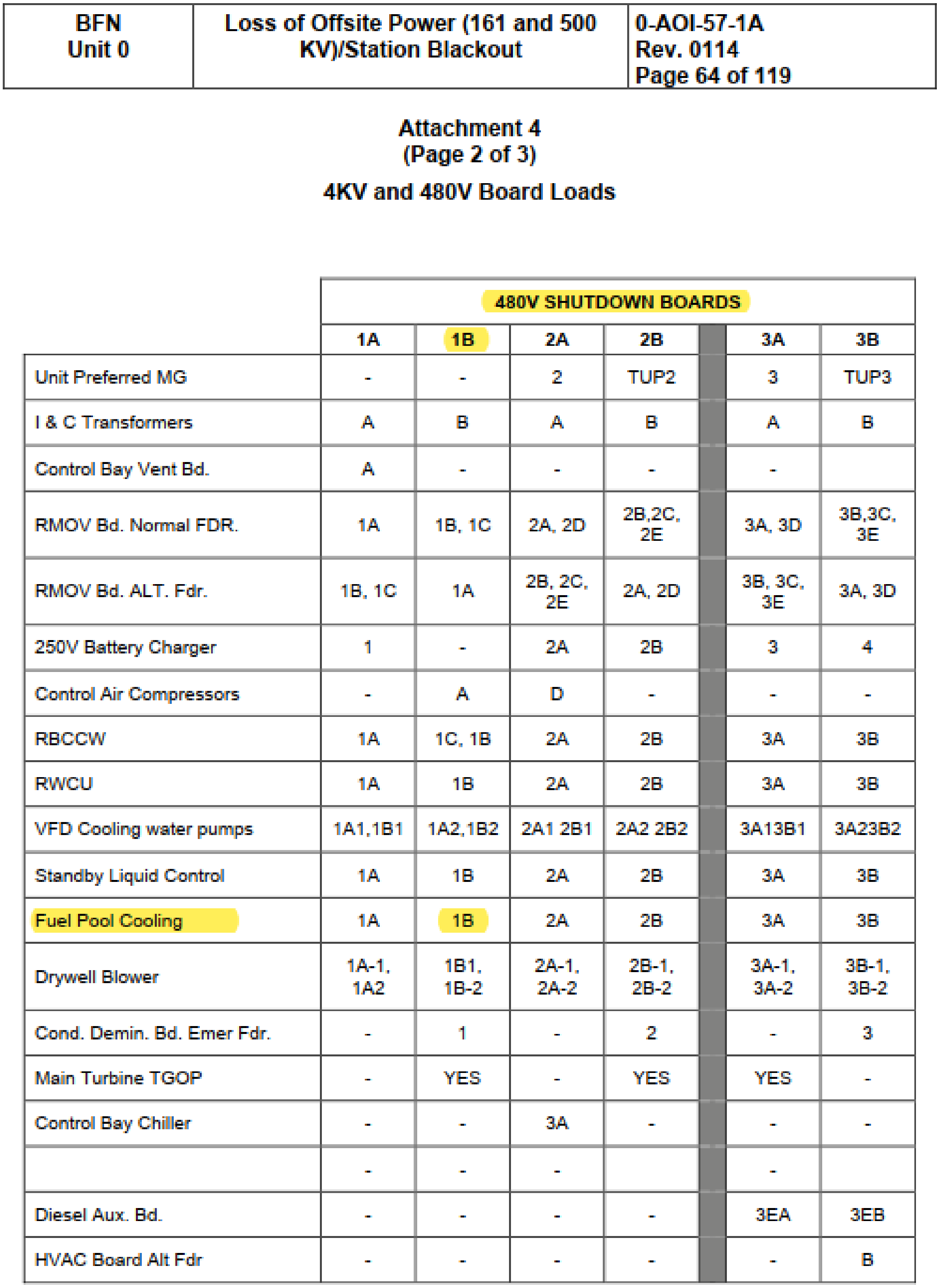
BFN **Unit 0**	**480V/240V AC Electrical System**	**0-OI-57B** **Rev. 0200** **Page 115 of 121**

Attachment 1
(Page 7 of 9)

Auxiliary Power Supplies and Bus Transfer Schemes

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE 1	ALTERNATE 2	REMARKS
12	480V Turbine Building Vent Boards				
	A. Board A (Unit 1,2,3)	480V Unit Board A (Unit 1,2,3)	480V Common BD 1 (Unit 1 only) 480-V Com. BD 3 (Unit 2 and 3)		Automatic transfer from normal to alternate source is initiated by time-undervoltage on the normal source. Return to normal source is automatic upon return of voltage to normal source. The normally closed, manually operated bus tie breaker provides for maintenance on one bus section while keeping the other bus section energized and in operation.
	B. Board B (Unit 1,2,3)	480V Unit Board B (Unit 1,2,3)	480V Common Board 2		
13	480V Shutdown Boards				
	A. Unit 1, 480V Shutdown BD 1A	4kV Shutdown Board A	4kV Shutdown Board B		Transfer from normal to alternate source is manual. Interlocking is provided to prevent manually transferring to a faulted board and to prevent paralleling two sources. 480V Load Shed Relay Time Delay Setting is set at 1.8 secs per DCN-W14030.
	B. Unit 1, 480V Shutdown BD 1B	4kV Shutdown Board C	4kV Shutdown Board B		
	C. Unit 2, 480V Shutdown BD 2A	4kV Board B	4kV Shutdown Board C		
	D. Unit 2, 480V Shutdown BD 2B	4kV Shutdown Board D	4kV Shutdown Board C		
	E. Unit 3, 480V Shutdown BD 3A	4kV Shutdown Board 3EA	4kV Shutdown Board 3EB		
	F. Unit 3, 480V Shutdown BD 3B	4kV Shutdown Board 3EC	4kV Shutdown Board 3EB		

Excerpt from 0-AOI-57-1A: Illustrates the Fuel Pool Cooling Pump power supplies



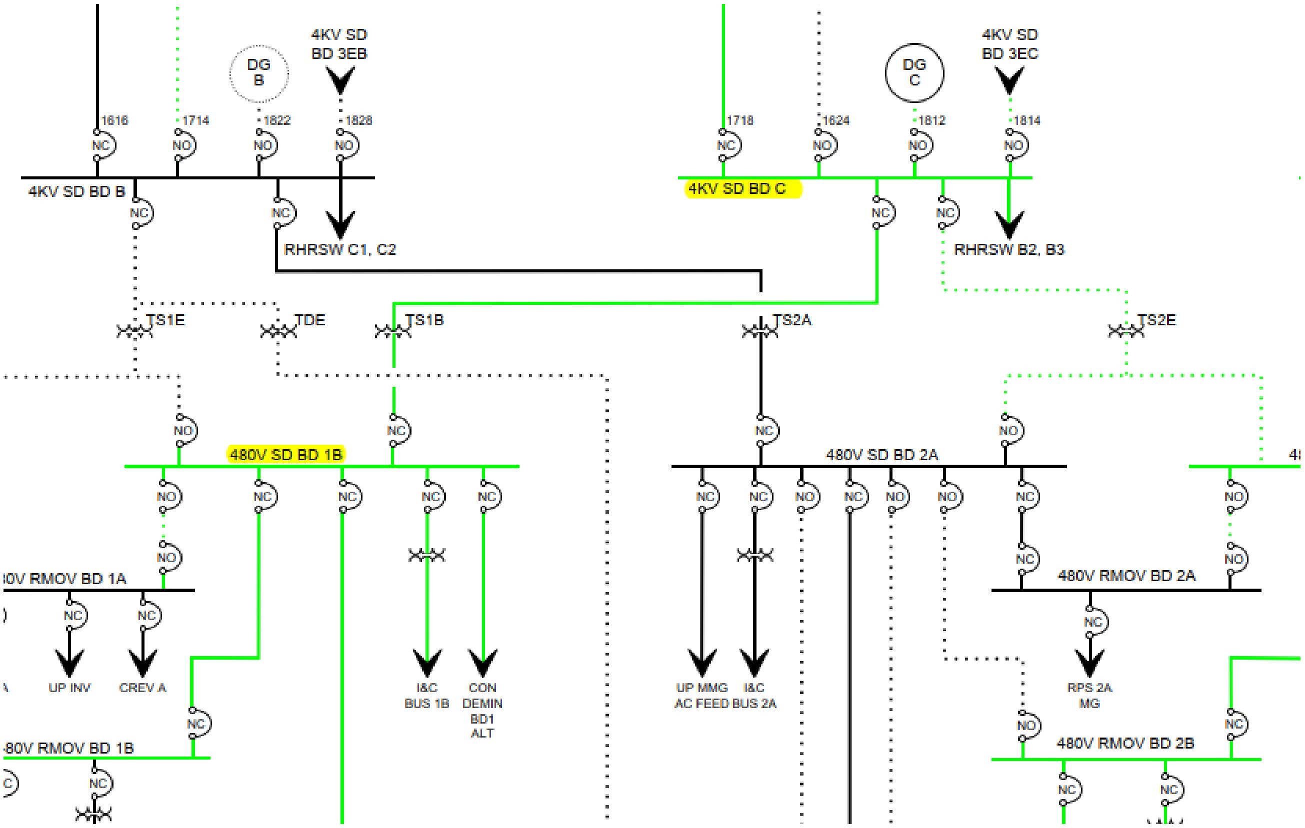
BFN Unit 0	Loss of Offsite Power (161 and 500 KV)/Station Blackout	0-AOI-57-1A Rev. 0114 Page 64 of 119

Attachment 4
(Page 2 of 3)

4KV and 480V Board Loads

	480V SHUTDOWN BOARDS					
	1A	1B	2A	2B	3A	3B
Unit Preferred MG	-	-	2	TUP2	3	TUP3
I & C Transformers	A	B	A	B	A	B
Control Bay Vent Bd.	A	-	-	-	-	
RMOV Bd. Normal FDR.	1A	1B, 1C	2A, 2D	2B,2C, 2E	3A, 3D	3B,3C, 3E
RMOV Bd. ALT. Fdr.	1B, 1C	1A	2B, 2C, 2E	2A, 2D	3B, 3C, 3E	3A, 3D
250V Battery Charger	1	-	2A	2B	3	4
Control Air Compressors	-	A	D	-	-	-
RBCCW	1A	1C, 1B	2A	2B	3A	3B
RWCU	1A	1B	2A	2B	3A	3B
VFD Cooling water pumps	1A1,1B1	1A2,1B2	2A1 2B1	2A2 2B2	3A13B1	3A23B2
Standby Liquid Control	1A	1B	2A	2B	3A	3B
Fuel Pool Cooling	1A	1B	2A	2B	3A	3B
Drywell Blower	1A-1, 1A2	1B1, 1B-2	2A-1, 2A-2	2B-1, 2B-2	3A-1, 3A-2	3B-1, 3B-2
Cond. Demin. Bd. Emer Fdr.	-	1	-	2	-	3
Main Turbine TGOP	-	YES	-	YES	YES	-
Control Bay Chiller	-	-	3A	-	-	-
	-	-	-		-	
Diesel Aux. Bd.	-	-	-	-	3EA	3EB
HVAC Board Alt Fdr	-	-	-	-	-	B

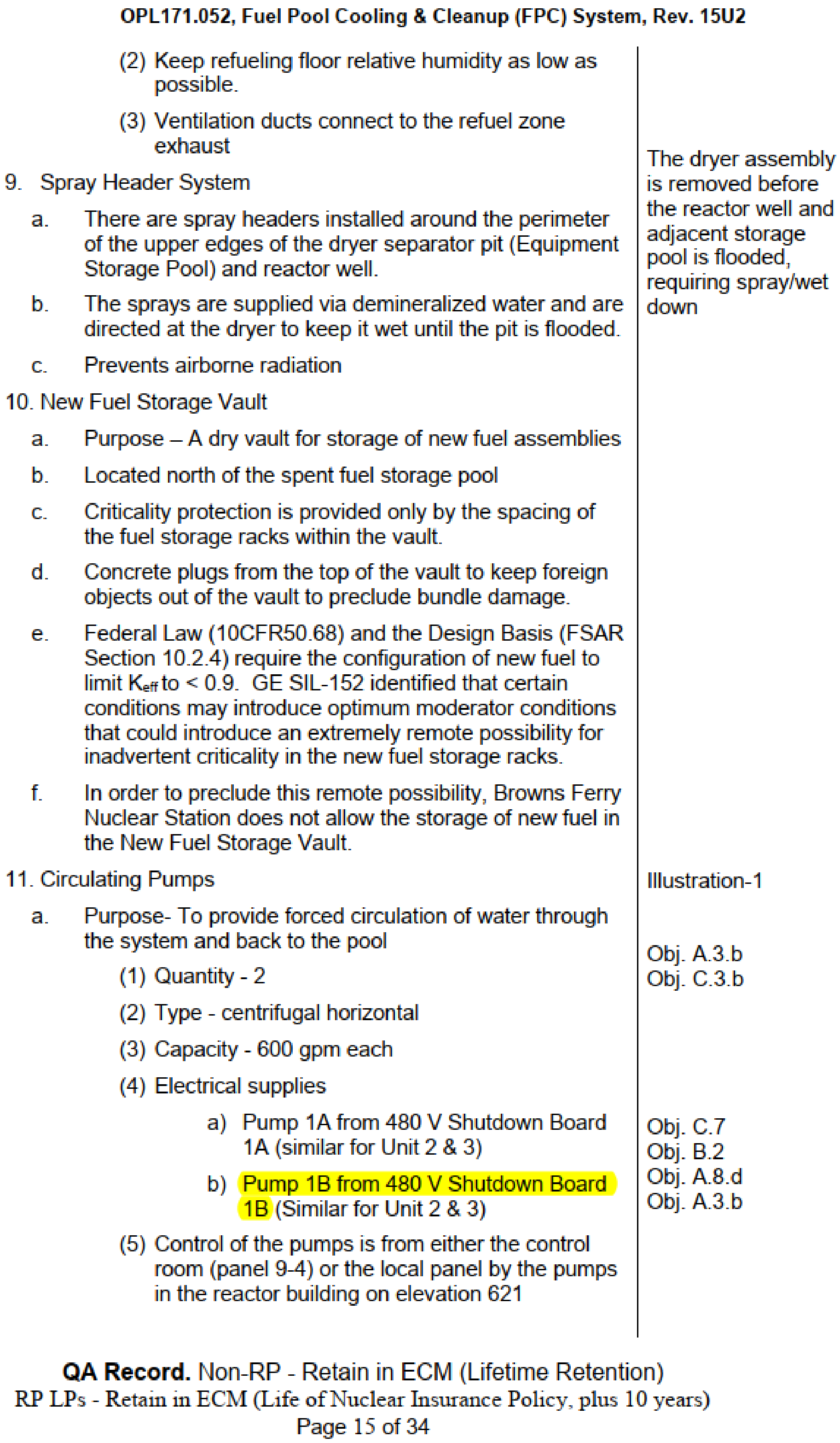
Excerpt from AC Electrical Distribution PIP 02-03:



AC ELECTRICAL DISTRIBUTION SYSTEM
BROWNS FERRY NUCLEAR PLANT

PIP-02-03 09/16/2020

Excerpt from OPL171.052 Lesson Plan:



OPL171.052, Fuel Pool Cooling & Cleanup (FPC) System, Rev. 15U2

(2) Keep refueling floor relative humidity as low as possible.

(3) Ventilation ducts connect to the refuel zone exhaust

9. Spray Header System

The dryer assembly is removed before the reactor well and adjacent storage pool is flooded, requiring spray/wet down

a. There are spray headers installed around the perimeter of the upper edges of the dryer separator pit (Equipment Storage Pool) and reactor well.

b. The sprays are supplied via demineralized water and are directed at the dryer to keep it wet until the pit is flooded.

c. Prevents airborne radiation

10. New Fuel Storage Vault

a. Purpose – A dry vault for storage of new fuel assemblies

b. Located north of the spent fuel storage pool

c. Criticality protection is provided only by the spacing of the fuel storage racks within the vault.

d. Concrete plugs from the top of the vault to keep foreign objects out of the vault to preclude bundle damage.

e. Federal Law (10CFR50.68) and the Design Basis (FSAR Section 10.2.4) require the configuration of new fuel to limit K_{eff} to < 0.9. GE SIL-152 identified that certain conditions may introduce optimum moderator conditions that could introduce an extremely remote possibility for inadvertent criticality in the new fuel storage racks.

f. In order to preclude this remote possibility, Browns Ferry Nuclear Station does not allow the storage of new fuel in the New Fuel Storage Vault.

11. Circulating Pumps

Illustration-1

a. Purpose- To provide forced circulation of water through the system and back to the pool

Obj. A.3.b
Obj. C.3.b

(1) Quantity - 2

(2) Type - centrifugal horizontal

(3) Capacity - 600 gpm each

(4) Electrical supplies

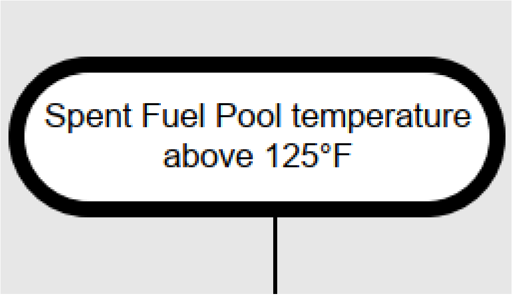
a) Pump 1A from 480 V Shutdown Board 1A (similar for Unit 2 & 3)

b) Pump 1B from 480 V Shutdown Board 1B (Similar for Unit 2 & 3)

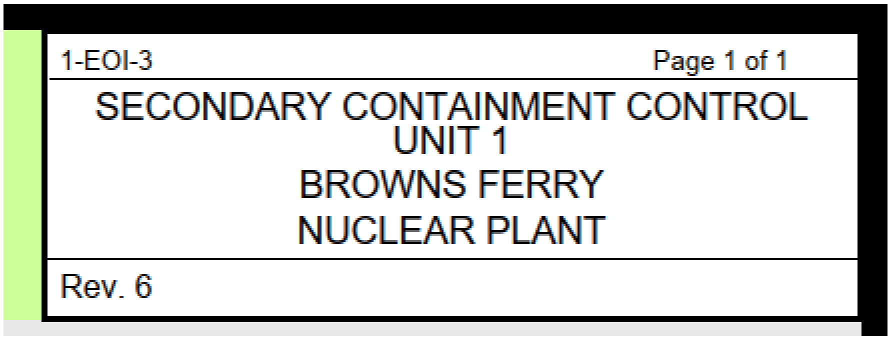
Obj. C.7
Obj. B.2
Obj. A.8.d
Obj. A.3.b

(5) Control of the pumps is from either the control room (panel 9-4) or the local panel by the pumps in the reactor building on elevation 621

Excerpt from 1-EOI-3: Illustrating related entry condition



Spent Fuel Pool temperature above 125°F

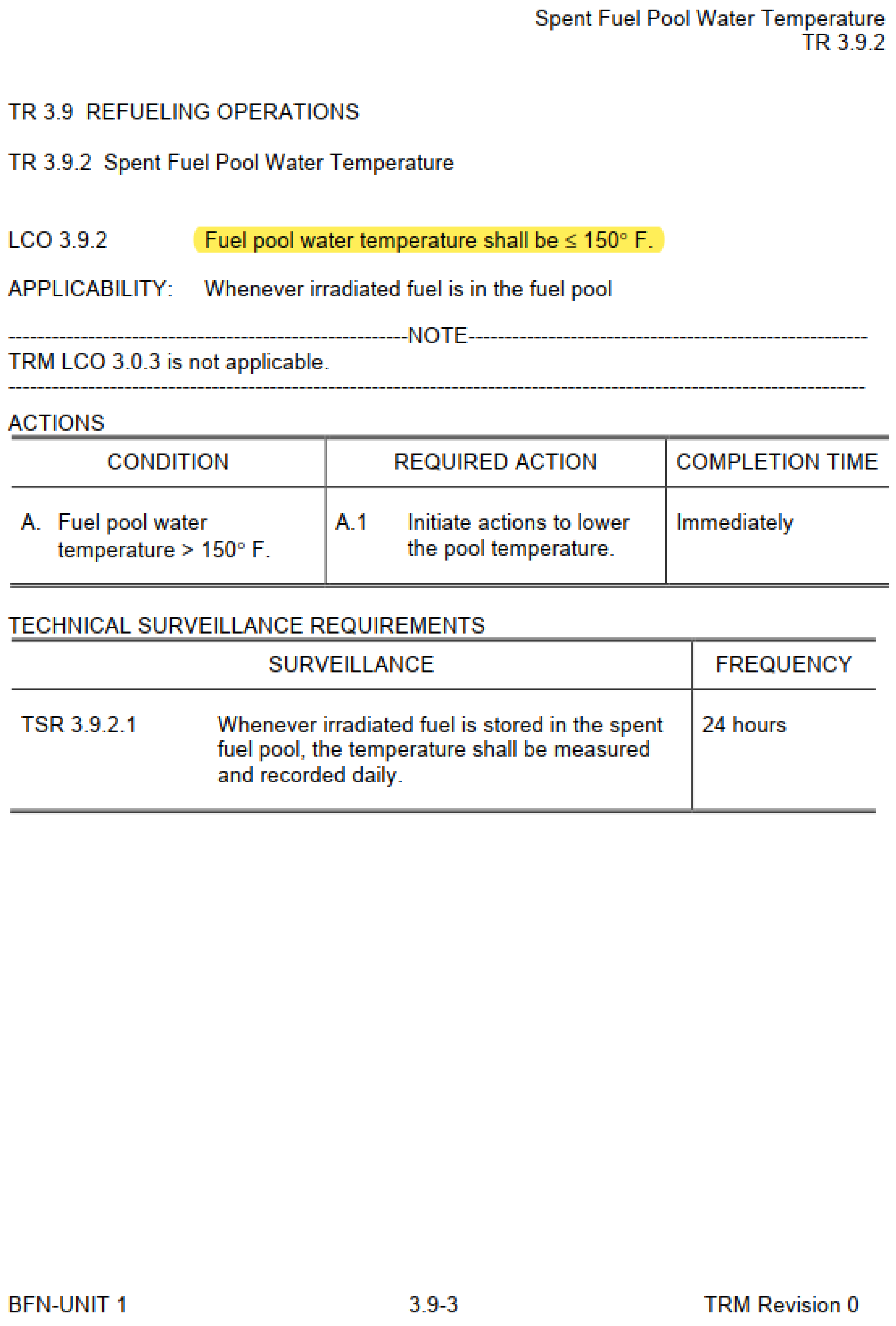


1-EOI-3 Page 1 of 1

SECONDARY CONTAINMENT CONTROL
UNIT 1
BROWNS FERRY
NUCLEAR PLANT

Rev. 6

Excerpt from TRM 3.9.2: Supports Distractors B(2), D(2)



Spent Fuel Pool Water Temperature
TR 3.9.2

TR 3.9 REFUELING OPERATIONS

TR 3.9.2 Spent Fuel Pool Water Temperature

LCO 3.9.2 Fuel pool water temperature shall be ≤ 150° F.

APPLICABILITY: Whenever irradiated fuel is in the fuel pool

---NOTE---
TRM LCO 3.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel pool water temperature > 150° F.	A.1 Initiate actions to lower the pool temperature.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.9.2.1	Whenever irradiated fuel is stored in the spent fuel pool, the temperature shall be measured and recorded daily.	24 hours

BFN-UNIT 1 3.9-3 TRM Revision 0

Examination Outline Cross-reference:	Level	RO	SRO
259001 (SF2 FWS) Feedwater	Tier #	2	------
K4.12 (10CFR 55.41.7)	Group #	2	------
Knowledge of Feedwater System design features and/or interlocks that provide for the following: • RFP start permissives	K/A #	259001K4.12	
	Importance Rating	3.1	------

Proposed Question: **# 59**

Unit 2 is performing a startup at 30% RTP when an event occurs resulting in the following conditions:

- Reactor Water Level peaked at (+) 50 inches
- Main Condenser Vacuum reached 7 inches mercury (Hg)
- All Reactor Feedwater Pump Turbines (RFPTs) tripped

Given the conditions above and in accordance with 2-OI-3, Reactor Feedwater System, the RFPTs tripped due to ___**(1)**____.

Subsequently, conditions were restored to normal. In order to meet a RFP start permissive, the respective ___**(2)**____ valve must be OPEN.

A. (1) High Reactor Water Level
(2) discharge

B. (1) High Reactor Water Level
(2) suction

C. (1) Low Main Condenser Vacuum
(2) discharge

D. (1) Low Main Condenser Vacuum
(2) suction

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that RFPTs have a sealed in trip signal at a Reactor Water Level of 55 inches, however the stated 50 inches would not inhibit operation of a RFPT. Second part is incorrect, but plausible if the candidate confuses the start/trip logic permissive. It would be plausible to believe that the discharge valve position inputs into the start and trip logic of the RFPTs. There is no throttle capability of the discharge valve, but the pump can be operated with the discharge valve closed in regards to interlocks.

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-OI-3, Reactor Feedwater System, Reactor Feedwater Pump Turbines trip on Condenser Low Vacuum at 7 inches mercury (Hg) and seals in until that interlock has cleared. For second part, if the RFPT suction valve is not full open, it will be interlocked tripped and all steam supply valves will remain closed. Then each respective RFPT suction must be OPEN prior to starting each RFPT, then each RFPT discharge will be OPEN for the pump being placed into service.

RO Level Justification: Tests the candidate's knowledge of Feedwater System interlocks that provide RFP start permissives and RFPT automatic trips. 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-3, Rev. 162 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

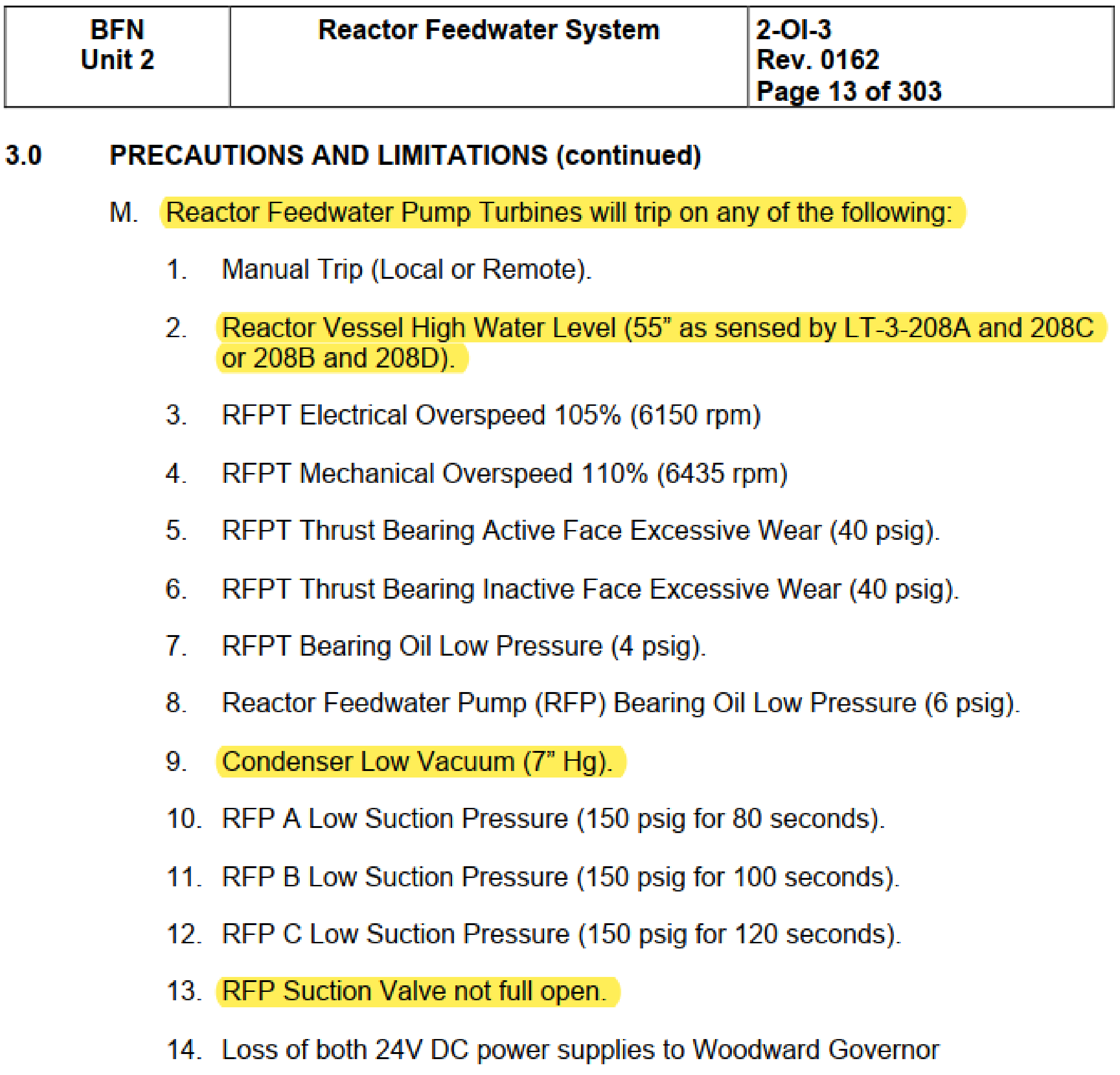
Learning Objective: OPL171.026, Obj. 4f (As available)

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	

Excerpts from 2-OI-3:

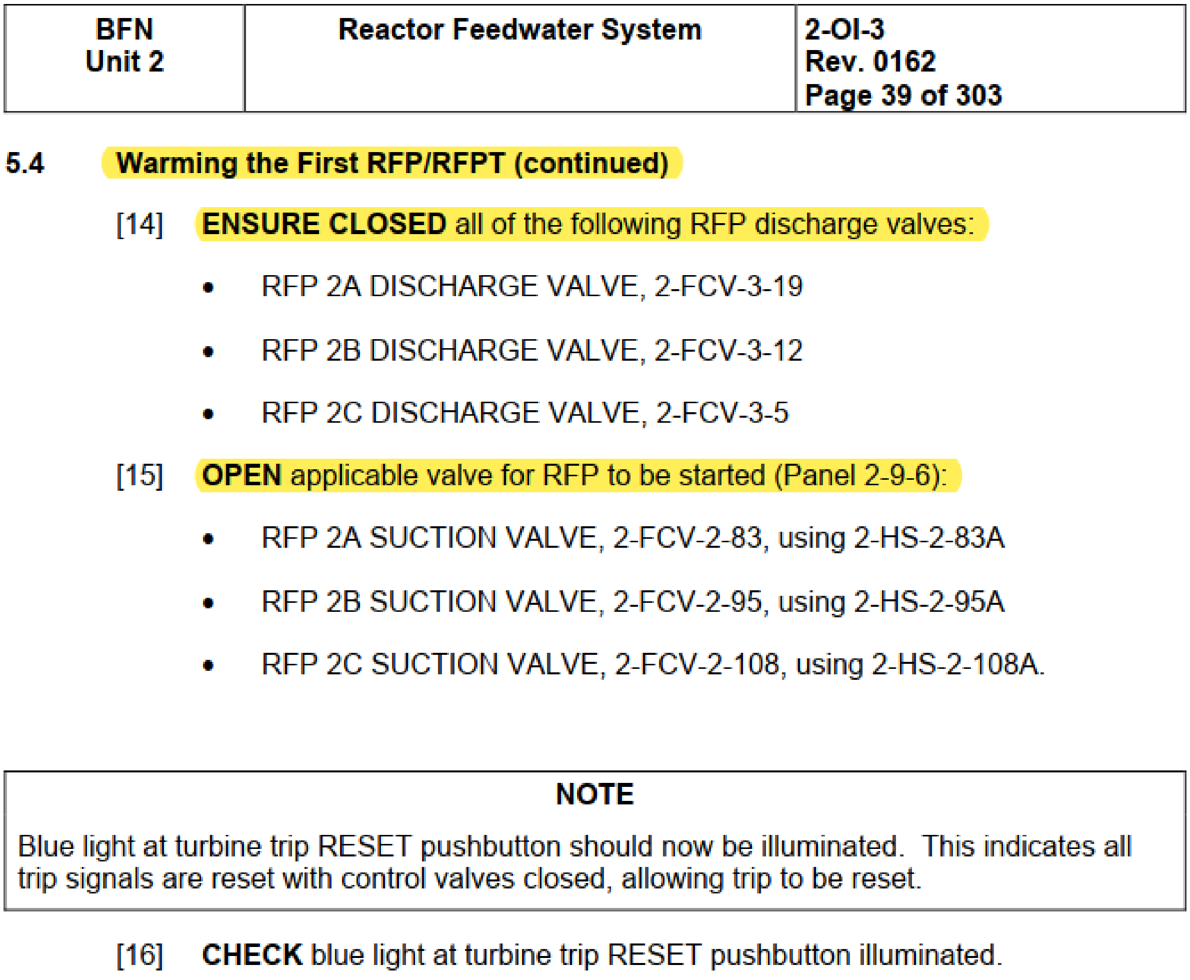


BFN Unit 2	Reactor Feedwater System	2-OI-3 Rev. 0162 Page 13 of 303

3.0 PRECAUTIONS AND LIMITATIONS (continued)

M. Reactor Feedwater Pump Turbines will trip on any of the following:

1. Manual Trip (Local or Remote).
2. Reactor Vessel High Water Level (55” as sensed by LT-3-208A and 208C or 208B and 208D).
3. RFPT Electrical Overspeed 105% (6150 rpm)
4. RFPT Mechanical Overspeed 110% (6435 rpm)
5. RFPT Thrust Bearing Active Face Excessive Wear (40 psig).
6. RFPT Thrust Bearing Inactive Face Excessive Wear (40 psig).
7. RFPT Bearing Oil Low Pressure (4 psig).
8. Reactor Feedwater Pump (RFP) Bearing Oil Low Pressure (6 psig).
9. Condenser Low Vacuum (7” Hg).
10. RFP A Low Suction Pressure (150 psig for 80 seconds).
11. RFP B Low Suction Pressure (150 psig for 100 seconds).
12. RFP C Low Suction Pressure (150 psig for 120 seconds).
13. RFP Suction Valve not full open.
14. Loss of both 24V DC power supplies to Woodward Governor



BFN **Unit 2**	**Reactor Feedwater System**	**2-OI-3** **Rev. 0162** **Page 39 of 303**

5.4 Warming the First RFP/RFPT (continued)

[14] **ENSURE CLOSED** all of the following RFP discharge valves:

- RFP 2A DISCHARGE VALVE, 2-FCV-3-19
- RFP 2B DISCHARGE VALVE, 2-FCV-3-12
- RFP 2C DISCHARGE VALVE, 2-FCV-3-5

[15] **OPEN** applicable valve for RFP to be started (Panel 2-9-6):

- RFP 2A SUCTION VALVE, 2-FCV-2-83, using 2-HS-2-83A
- RFP 2B SUCTION VALVE, 2-FCV-2-95, using 2-HS-2-95A
- RFP 2C SUCTION VALVE, 2-FCV-2-108, using 2-HS-2-108A.

NOTE

Blue light at turbine trip RESET pushbutton should now be illuminated. This indicates all trip signals are reset with control valves closed, allowing trip to be reset.

[16] **CHECK** blue light at turbine trip RESET pushbutton illuminated.

5.4 Warming the First RFP/RFPT (continued)

NOTE

Normal operating range for RFP lube oil to the bearings is 110°F to 120°F (local). For additional lube oil temperature control, Attachment 7 has information for controlling Raw Cooling Water through RFP lube oil cooler.

[30] **WARM** lube oil to bearings as follows (Panels 2-9-5 and 2-9-6):

[30.1] **ENSURE** RFPT Speed Control position.

- RFPT 2A SPEED CONT RAISE/LOWER switch, 2-HS-46-8A is depressed in MANUAL GOVERNOR position and amber light at switch is illuminated.
- RFPT 2B SPEED CONT RAISE/LOWER switch, 2-HS-46-9A is depressed in MANUAL GOVERNOR position and amber light at switch is illuminated.
- RFPT 2C SPEED CONT RAISE/LOWER switch, 2-HS-46-10A is depressed in MANUAL GOVERNOR position and amber light at switch is illuminated.

[30.2] **START** RFPT from Panel 2-9-6 as follows:

- **PLACE** RFPT 2A START/LOCAL ENABLE, 2-HS-46-112A, in START, and **OBSERVE** RFPT accelerates to approximately 600 rpm on RFPT 2A SPEED, 2-SI-46-8A.
- **PLACE** RFPT 2B START/LOCAL ENABLE, 2-HS-46-138A, in START, and **OBSERVE** RFPT accelerates to approximately 600 rpm on RFPT 2B SPEED, 2-SI-46-9A.
- **PLACE** RFPT 2C START/LOCAL ENABLE, 2-HS-46-163A, in START, and **OBSERVE** RFPT accelerates to approximately 600 rpm on RFPT 2C SPEED, 2-SI-46-10A.

BFN **Unit 2**	**Reactor Feedwater System**	**2-OI-3** **Rev. 0162** **Page 49 of 303**

5.5 Placing the First RFP/RFPT in Service (continued)

NOTE

In the following step, it may take a position demand of 50% to 80% before pressure rise is noted on HP HTR OUTLET, 2-PI-3-45.

CAUTION

Reactor water level should be closely monitored when Start-Up Level Controller is adjusted.

Start of Critical Steps

[8.4] **ENSURE** RFW START-UP LEVEL CONTROL, 2-LIC-3-53, in MANUAL (amber light illuminated) and Column 3 (CO) selected.

[8.5] **SLOWLY OPEN** RFW START-UP LCV, 2-LCV-3-53, using Ramp Raise/Ramp Lower pushbuttons, UNTIL rising pressure is noted on Panel 2-9-6, HP HTR OUTLET, 2-PI-3-45.

[8.6] **WHEN** indicated pressure on HP HTR OUTLET, 2-PI-3-45, is within 50 psig of indicated pressure on CNDS BSTR PMP DISCH HDR PRESS, 2-PI-2-70, **THEN**

OPEN RFP discharge valve for pump being placed in service:

- RFP 2A DISCHARGE VALVE, 2-FCV-3-19
- RFP 2B DISCHARGE VALVE, 2-FCV-3-12
- RFP 2C DISCHARGE VALVE, 2-FCV-3-5

[9] **ENSURE** the following on RFW START-UP LEVEL CONTROL, 2-LIC-3-53:

A. Column 3 (CO) selected and in MANUAL (amber light illuminated).

B. Valve position demand set at 0 or slightly negative value using Ramp LOWER pushbutton.

Examination Outline Cross-reference:

295018 (APE 18) Partial or Complete Loss of CCW / 8

AA2.02 (10CFR 55.41.10)

Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Component Cooling Water:

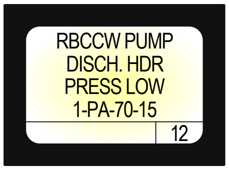
- Cooling Water Temperature

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295018AA2.02	
Importance Rating	3.7	------

Proposed Question: **# 60**

Unit 1 is operating at 100% RTP when the following conditions occur:

- RBCCW Suction Header Temperature is 100 °F and rising
- RBCCW PUMP DISCHARGE HEADER PRESSURE LOW (1-9-4C, Window 12)
- **NO** Operator actions have been taken



Given the conditions above and in accordance with 1-AOI-70-1, Loss of Reactor Building Closed Cooling Water, the Operator will **IMMEDIATELY** ___**(1)**___ and **ENSURE** 1-FCV-70-48, RBCCW SECTIONALIZING VALVE automatically ___**(2)**___.

A. (1) secure RWCU pumps
(2) closed

B. (1) secure RWCU pumps
(2) opened

C. (1) start the spare RBCCW pump
(2) closed

D. (1) start the spare RBCCW pump
(2) opened

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 1-AOI-70-1, Loss of Reactor Building Closed Cooling Water there are two Immediate Actions. Operators are to (1) secure the RWCU pumps and (2) ensure the RBCCW Sectionalizing Valve closes. For second part, in accordance with the given alarm RBCCW PUMP DISCHARGE HEADER PRESSURE LOW (1-9-4C, Window 12), the setpoint is 57 psig. In accordance with 1-AOI-70-1, Loss of Reactor Building Closed Cooling Water, RBCCW Pump Discharge Pressure of 57 psig causes 1-FCV-70-48, RBCCW SECTIONALIZING VALVE to automatically close.

B INCORRECT**:** First part is correct (*See A*). Second part is incorrect but plausible if the candidate confuses the normal position of 1-FCV-70-48, RBCCW SECTIONALIZING VALVE to be closed and thereby automatically opening when RBCCW Pump discharge pressure reaches 57 psig.

C INCORRECT: First part is incorrect but plausible since 1-AOI-70-1, Loss of Reactor Building Closed Cooling Water, has Subsequent Actions that direct starting the spare RBCCW pump. Additionally, given that RBCCW Suction Header Temperature is 100 °F and rising, RBCCW SUCTION HEADER TEMPERATURE HIGH (1-9-4C, Window 12) alarms (not given) which states to place the spare RBCCW pump in service. However, it is not an Immediate Action of 1-AOI-70-1 as the question is asking. 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 determine the impact that a partial or complete loss of Component Cooling Water and what Cooling Water Temperature has on the affected supported systems. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 1-AOI-70-1, Rev. 17 (Attach if not previously provided)
1-ARP-9-4C, Rev. 30

Proposed references to be provided to applicants during examination: **RBCCW PUMP DISCHARGE HEADER PRESSURE LOW (1-9-4C, Window 12)**

Learning Objective: OPL171.013 Obj.10a (As available)

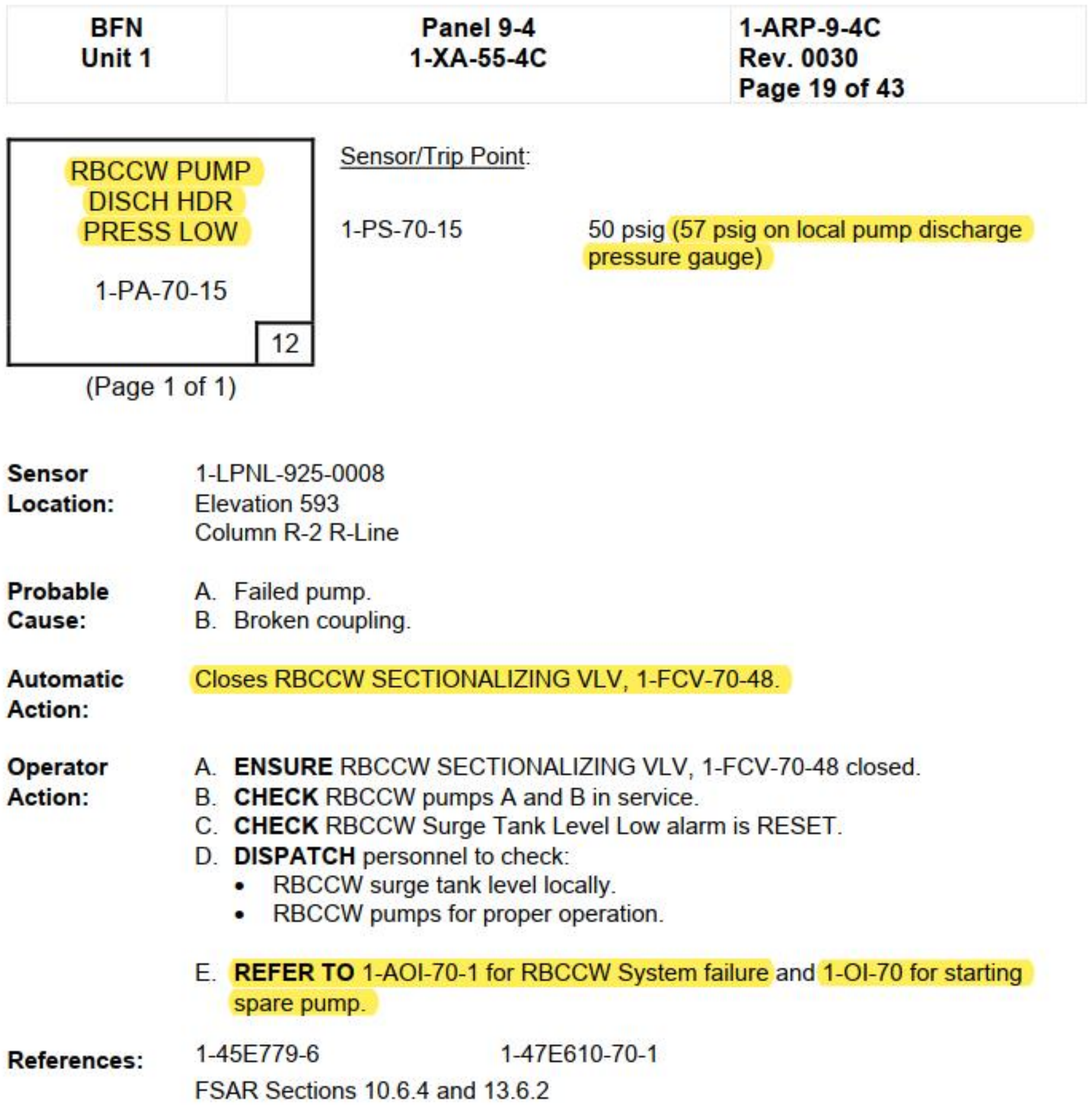
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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-9-4C:



BFN **Unit 1**	**Panel 9-4** **1-XA-55-4C**	**1-ARP-9-4C** **Rev. 0030** **Page 19 of 43**

RBCCW PUMP
DISCH HDR
PRESS LOW

1-PA-70-15

12

(Page 1 of 1)

Sensor/Trip Point:

1-PS-70-15	50 psig (57 psig on local pump discharge pressure gauge)

Sensor Location: 1-LPNL-925-0008
Elevation 593
Column R-2 R-Line

Probable Cause:
A. Failed pump.
B. Broken coupling.

Automatic Action: Closes RBCCW SECTIONALIZING VLV, 1-FCV-70-48.

Operator Action:
A. **ENSURE** RBCCW SECTIONALIZING VLV, 1-FCV-70-48 closed.
B. **CHECK** RBCCW pumps A and B in service.
C. **CHECK** RBCCW Surge Tank Level Low alarm is RESET.
D. **DISPATCH** personnel to check:
- RBCCW surge tank level locally.
- RBCCW pumps for proper operation.

E. **REFER TO** 1-AOI-70-1 for RBCCW System failure and 1-OI-70 for starting spare pump.

References: 1-45E779-6 1-47E610-70-1
FSAR Sections 10.6.4 and 13.6.2

Supports Distractors C(1), D(1)

BFN **Unit 1**	**Panel 9-4** **1-XA-55-4C**	**1-ARP-9-4C** **Rev. 0030** **Page 11 of 43**

RBCCW PUMP
SUCT HDR
TEMP HIGH

1-TA-70-3

5

(Page 1 of 1)

Sensor/Trip Point:

1-TIS-70-3 100°F

Sensor Location:	1-TIS-70-3	Panel 1-9-4	Main Control Room
	1-TM-070-003	1-PNLA-009-0019	Auxiliary Instrument Room
	1-TE-70-3	On Suction Headers on RBCCW Pumps A and B.	

Probable Cause:

A. Temperature control valve failure.
B. RWCU operating at elevated flow rate (270 - 340 gpm).
C. Low flow.

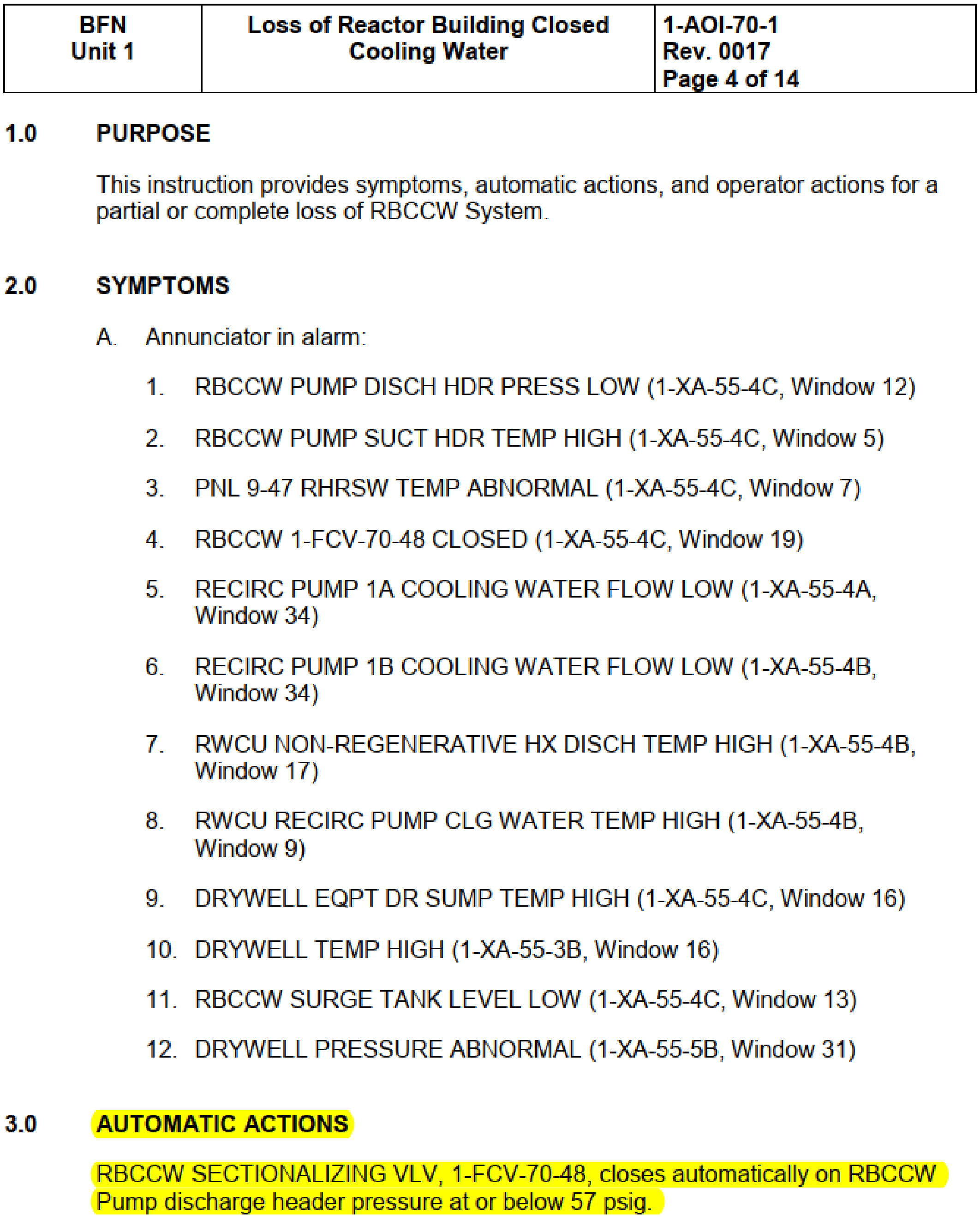
Automatic Action: None

Operator Action:

A. **CHECK** RBCCW PUMP SUCTION HDR TEMP, 1-TIS-70-3, for rising temperature, on Panel 1-9-4.
B. **IF** RWCU is operating at elevated flow rate (> 270 gpm), **THEN**
REDUCE RWCU flow rate, as necessary, to maintain RBCCW temperature below alarm setpoint.
C. **IF** RBCCW Pump Suction Header high temperature is valid and cannot be maintained <110°F,**THEN**
MANUALLY SCRAM the Reactor and **TRIP** both Recirc pumps
D. **EVALUATE** the need to lower RWCU flow.
E. **CHECK** Raw Cooling Water pressure using PI-24-18, on Panel 1-9-20.
F. **DISPATCH** personnel to check Raw Water Cooling temperature control valves operating properly on elevation 565 AND **ADJUST** as necessary.
G. **PLACE** spare RBCCW heat exchanger in service. **REFER TO** 1-OI-70 and 1-AOI-70-1.
H. **PLACE** additional RCW pumps in service as needed.

References: 1-45N620-4 1-47E610-70-1
FSAR Sections 10.6.4 and 13.6.2

Excerpts from 1-AOI-70-1:



BFN Unit 1	**Loss of Reactor Building Closed Cooling Water**	**1-AOI-70-1 Rev. 0017 Page 4 of 14**

1.0 PURPOSE

This instruction provides symptoms, automatic actions, and operator actions for a partial or complete loss of RBCCW System.

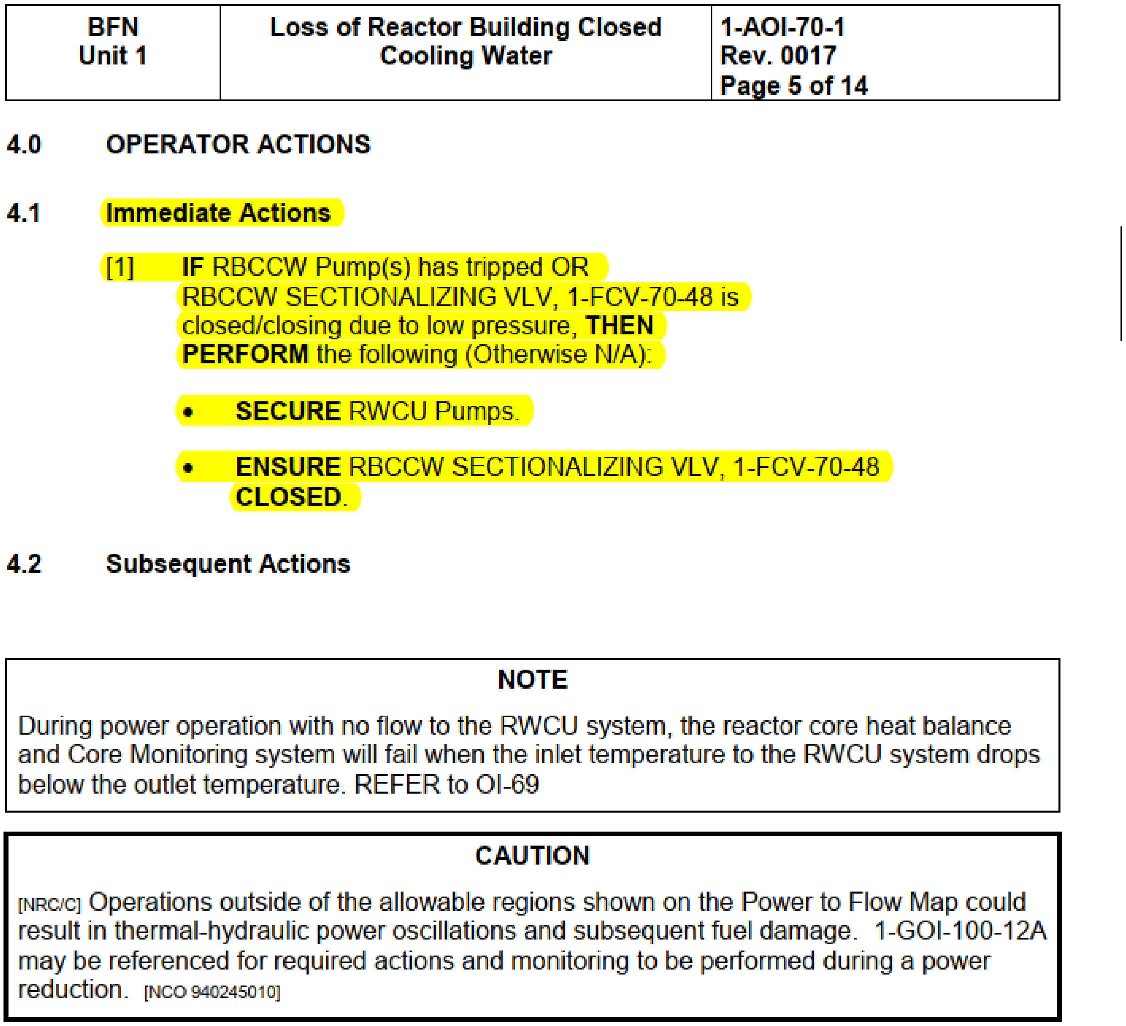
2.0 SYMPTOMS

A. Annunciator in alarm:

1. RBCCW PUMP DISCH HDR PRESS LOW (1-XA-55-4C, Window 12)
2. RBCCW PUMP SUCT HDR TEMP HIGH (1-XA-55-4C, Window 5)
3. PNL 9-47 RHRSW TEMP ABNORMAL (1-XA-55-4C, Window 7)
4. RBCCW 1-FCV-70-48 CLOSED (1-XA-55-4C, Window 19)
5. RECIRC PUMP 1A COOLING WATER FLOW LOW (1-XA-55-4A, Window 34)
6. RECIRC PUMP 1B COOLING WATER FLOW LOW (1-XA-55-4B, Window 34)
7. RWCU NON-REGENERATIVE HX DISCH TEMP HIGH (1-XA-55-4B, Window 17)
8. RWCU RECIRC PUMP CLG WATER TEMP HIGH (1-XA-55-4B, Window 9)
9. DRYWELL EQPT DR SUMP TEMP HIGH (1-XA-55-4C, Window 16)
10. DRYWELL TEMP HIGH (1-XA-55-3B, Window 16)
11. RBCCW SURGE TANK LEVEL LOW (1-XA-55-4C, Window 13)
12. DRYWELL PRESSURE ABNORMAL (1-XA-55-5B, Window 31)

3.0 AUTOMATIC ACTIONS

RBCCW SECTIONALIZING VLV, 1-FCV-70-48, closes automatically on RBCCW Pump discharge header pressure at or below 57 psig.



BFN **Unit 1**	**Loss of Reactor Building Closed Cooling Water**	**1-AOI-70-1** **Rev. 0017** **Page 5 of 14**

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** RBCCW Pump(s) has tripped OR RBCCW SECTIONALIZING VLV, 1-FCV-70-48 is closed/closing due to low pressure, **THEN PERFORM** the following (Otherwise N/A):

- **SECURE** RWCU Pumps.
- **ENSURE** RBCCW SECTIONALIZING VLV, 1-FCV-70-48 **CLOSED**.

4.2 Subsequent Actions

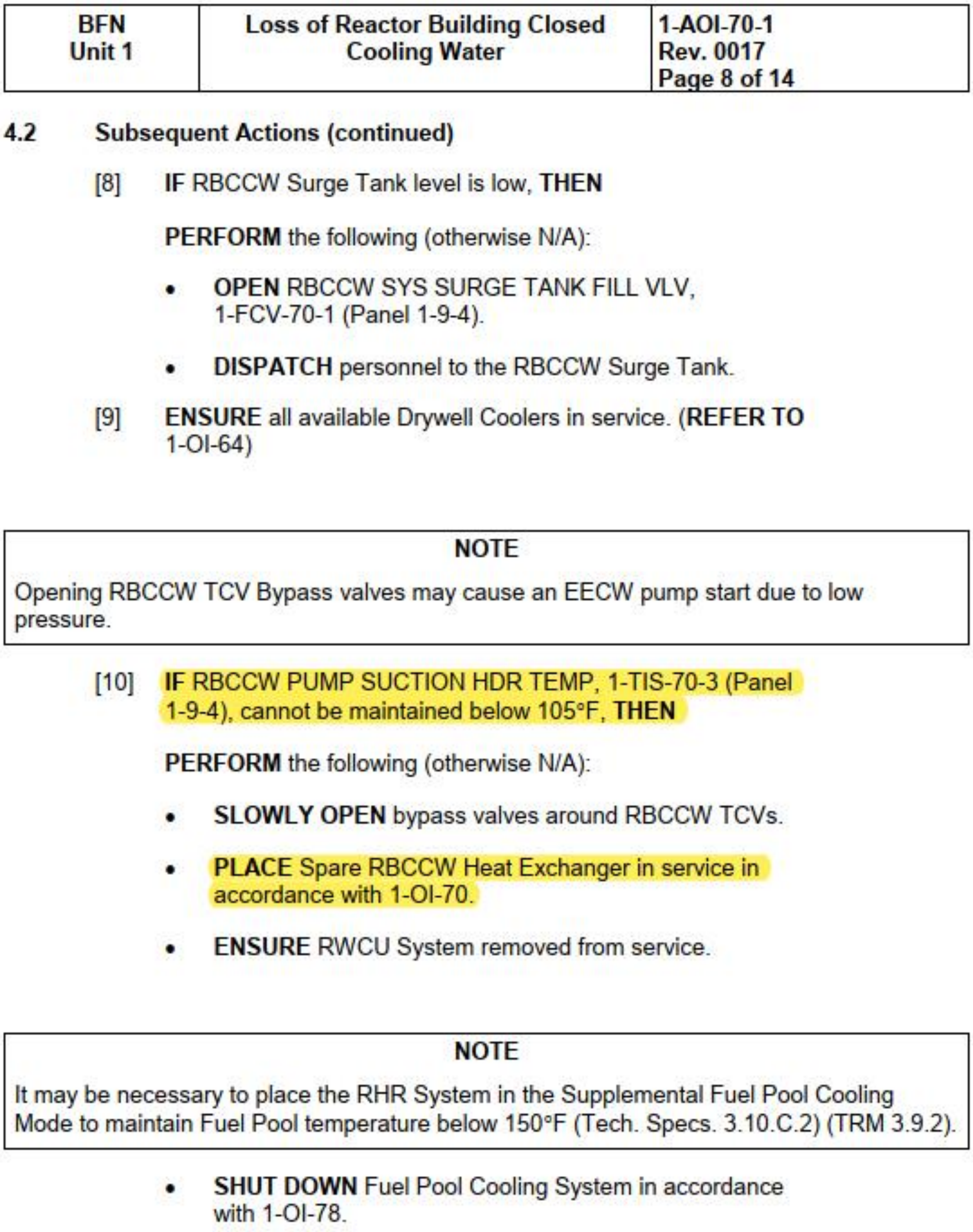
NOTE

During power operation with no flow to the RWCU system, the reactor core heat balance and Core Monitoring system will fail when the inlet temperature to the RWCU system drops below the outlet temperature. REFER to OI-69

CAUTION

[NRC/C] Operations outside of the allowable regions shown on the Power to Flow Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. 1-GOI-100-12A may be referenced for required actions and monitoring to be performed during a power reduction. [NCO 940245010]

Supports Distractors C(1), D(1):



BFN **Unit 1**	**Loss of Reactor Building Closed Cooling Water**	**1-AOI-70-1** **Rev. 0017** **Page 8 of 14**

4.2 Subsequent Actions (continued)

[8] **IF** RBCCW Surge Tank level is low, **THEN**

PERFORM the following (otherwise N/A):

- **OPEN** RBCCW SYS SURGE TANK FILL VLV, 1-FCV-70-1 (Panel 1-9-4).
- **DISPATCH** personnel to the RBCCW Surge Tank.

[9] **ENSURE** all available Drywell Coolers in service. (**REFER TO** 1-OI-64)

NOTE

Opening RBCCW TCV Bypass valves may cause an EECW pump start due to low pressure.

[10] **IF** RBCCW PUMP SUCTION HDR TEMP, 1-TIS-70-3 (Panel 1-9-4), cannot be maintained below 105°F, **THEN**

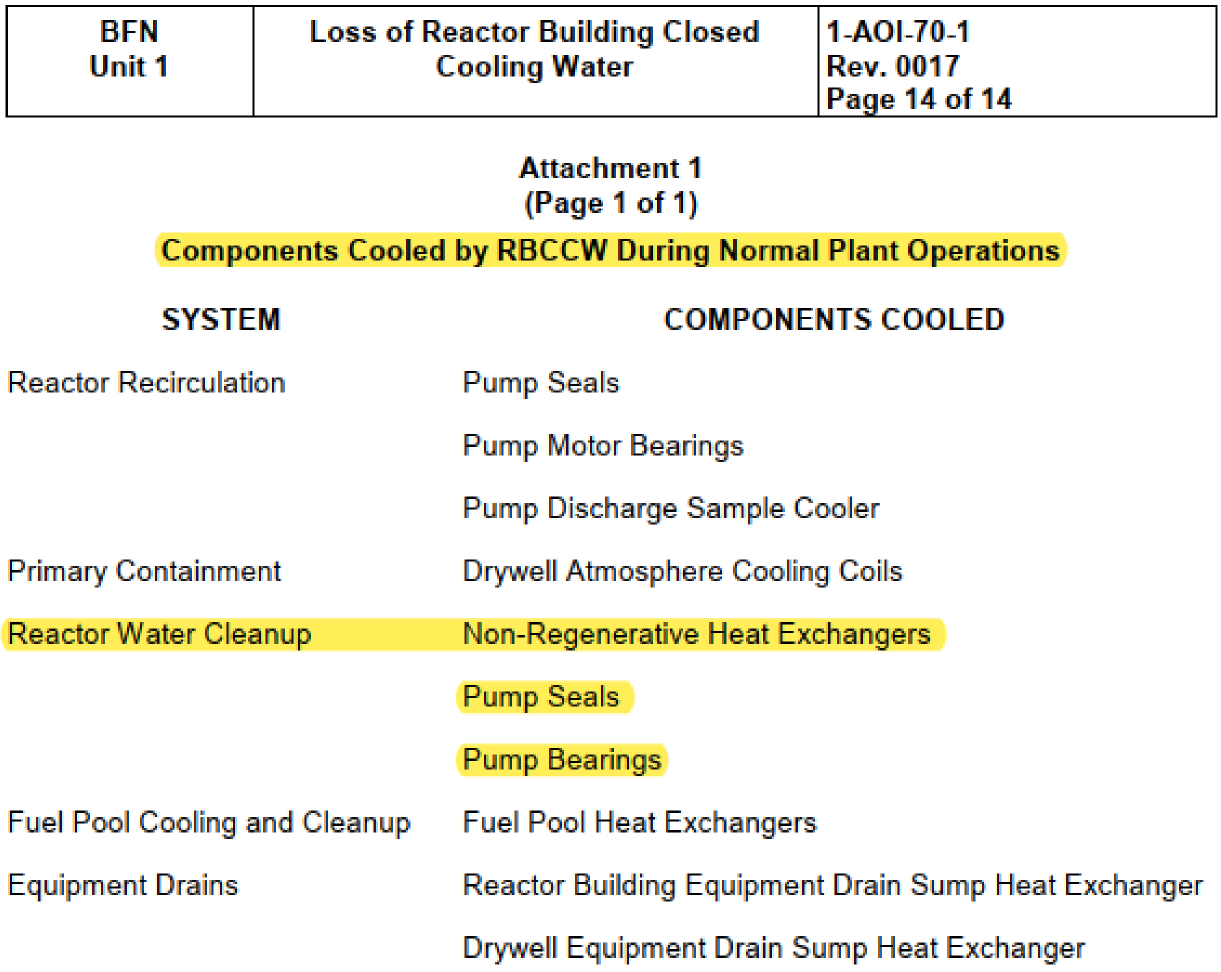
PERFORM the following (otherwise N/A):

- **SLOWLY OPEN** bypass valves around RBCCW TCVs.
- **PLACE** Spare RBCCW Heat Exchanger in service in accordance with 1-OI-70.
- **ENSURE** RWCU System removed from service.

NOTE

It may be necessary to place the RHR System in the Supplemental Fuel Pool Cooling Mode to maintain Fuel Pool temperature below 150°F (Tech. Specs. 3.10.C.2) (TRM 3.9.2).

- **SHUT DOWN** Fuel Pool Cooling System in accordance with 1-OI-78.



BFN Unit 1	Loss of Reactor Building Closed Cooling Water	1-AOI-70-1 Rev. 0017 Page 14 of 14

Attachment 1
(Page 1 of 1)

Components Cooled by RBCCW During Normal Plant Operations

SYSTEM	COMPONENTS COOLED
Reactor Recirculation	Pump Seals
	Pump Motor Bearings
	Pump Discharge Sample Cooler
Primary Containment	Drywell Atmosphere Cooling Coils
Reactor Water Cleanup	Non-Regenerative Heat Exchangers
	Pump Seals
	Pump Bearings
Fuel Pool Cooling and Cleanup	Fuel Pool Heat Exchangers
Equipment Drains	Reactor Building Equipment Drain Sump Heat Exchanger
	Drywell Equipment Drain Sump Heat Exchanger

Examination Outline Cross-reference:	Level	RO	SRO
Emergency Procedures/Plan	Tier #	3	------
G2.4.32 (10CFR 55.41.10)	Group #	------	------
Knowledge of operator response to loss of annunciators.	K/A #	G2.4.32	
	Importance Rating	3.6	------

Proposed Question: **# 61**

Unit 2 is operating at 100% RTP when the following occurs:

An annunciator with an invalid input has been disabled in accordance with OPDP-4, Annunciator Disablement.

The annunciator window is required to be flagged with a ___**(1)**___ magnetic border and a narrative log entry ___**(2)**___ required to made.

A. (1) white
(2) is NOT

B. (1) white
(2) is

C. (1) blue
(2) is NOT

D. (1) blue
(2) is

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that in accordance with OPDP-4, Annunciator Disablement, a white border will be placed on an alarm that is out of service for maintenance or testing. Second part is incorrect but plausible in that Operators are not required to log an out of service alarm by OPDP-4.

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 OPDP-4, Annunciator Disablement, a disabled annunciator indicator will be placed on the alarm window by the Unit Operator that is a blue magnetic border labeled 'Disabled Alarm Input'. For second part, in accordance with OPDP-4, an entry concerning the disablement of an alarm will be placed in the narrative log to include alarm location, method used to disable alarm, date and time removed, justification for disablement, and Tech Spec action requirements, if applicable.

RO Level Justification: Tests the candidate's knowledge of operator response to a disabled annunciator in accordance with OPDP-4, Annunciator Disablement. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): OPDP-4, Rev. 9 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

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

Question Source:	Bank #	BFN NRC 1703 #69	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2017	

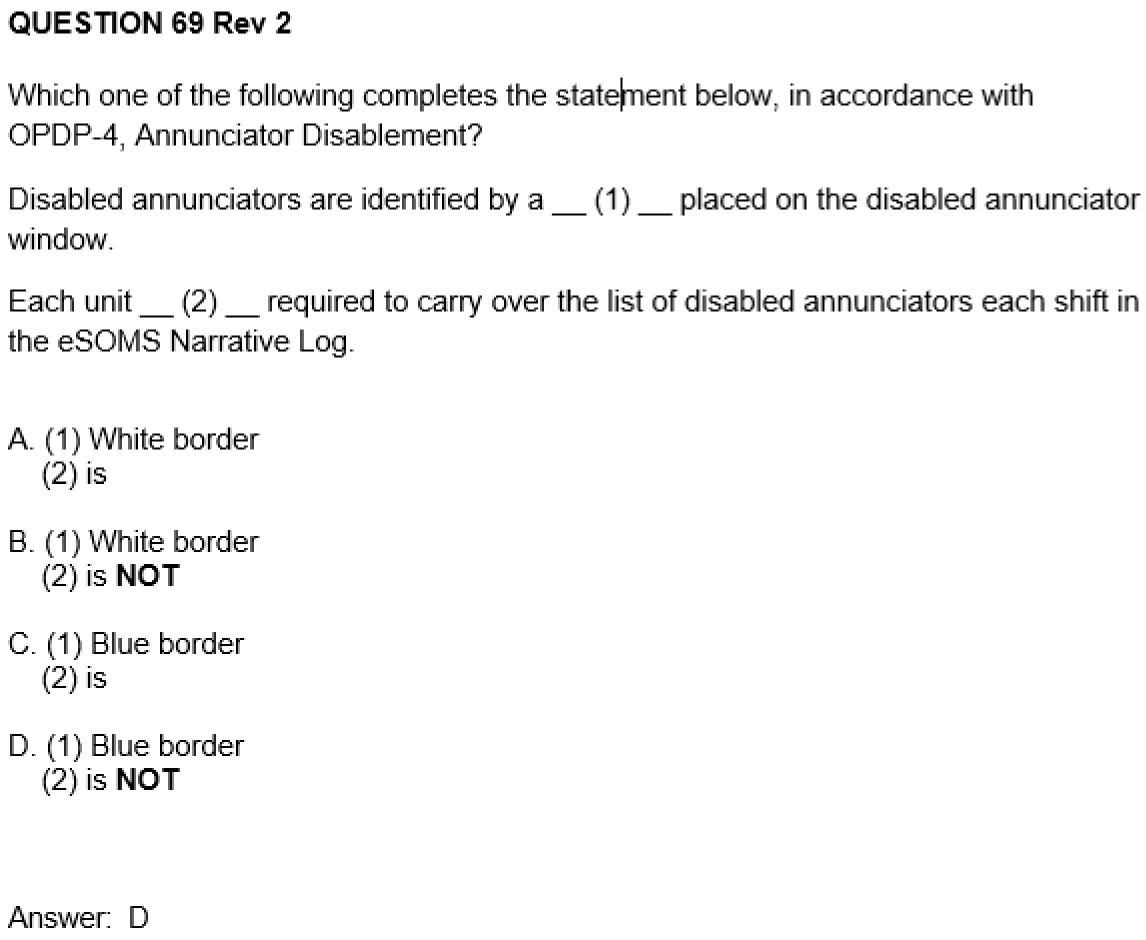
Question Cognitive Level:	Memory or Fundamental Knowledge	**X**
	Comprehension or Analysis	

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

Comments:

Form 4.2-1 Written Examination Question Worksheet

Copy of Bank Question:



QUESTION 69 Rev 2

Which one of the following completes the statement below, in accordance with OPDP-4, Annunciator Disablement?

Disabled annunciators are identified by a __ (1) __ placed on the disabled annunciator window.

Each unit __ (2) __ required to carry over the list of disabled annunciators each shift in the eSOMS Narrative Log.

A. (1) White border
(2) is

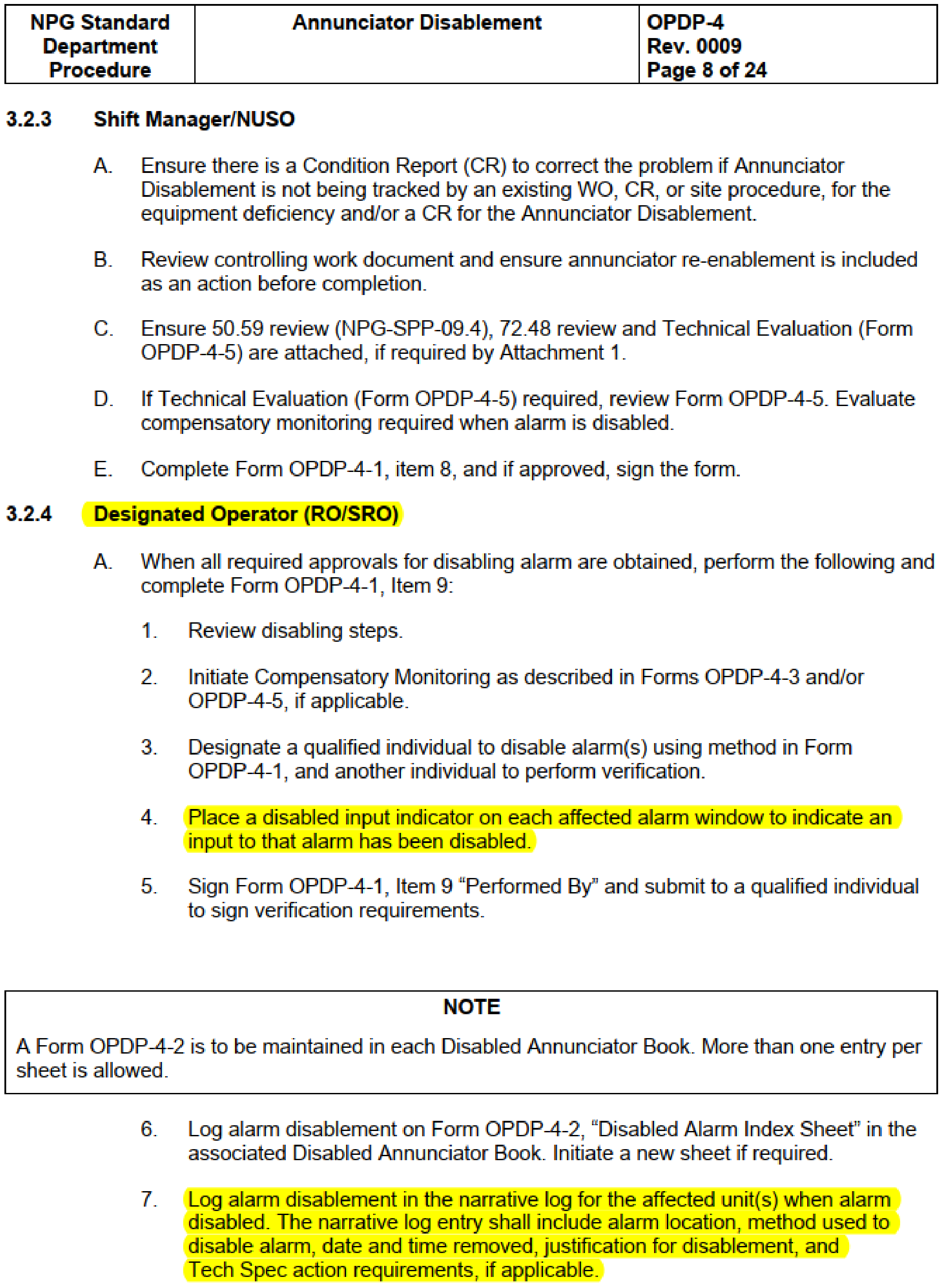
B. (1) White border
(2) is **NOT**

C. (1) Blue border
(2) is

D. (1) Blue border
(2) is **NOT**

Answer: D

Excerpts from OPDP-4:



NPG Standard Department Procedure	**Annunciator Disablement**	**OPDP-4** **Rev. 0009** **Page 8 of 24**

3.2.3 Shift Manager/NUSO

A. Ensure there is a Condition Report (CR) to correct the problem if Annunciator Disablement is not being tracked by an existing WO, CR, or site procedure, for the equipment deficiency and/or a CR for the Annunciator Disablement.

B. Review controlling work document and ensure annunciator re-enablement is included as an action before completion.

C. Ensure 50.59 review (NPG-SPP-09.4), 72.48 review and Technical Evaluation (Form OPDP-4-5) are attached, if required by Attachment 1.

D. If Technical Evaluation (Form OPDP-4-5) required, review Form OPDP-4-5. Evaluate compensatory monitoring required when alarm is disabled.

E. Complete Form OPDP-4-1, item 8, and if approved, sign the form.

3.2.4 Designated Operator (RO/SRO)

A. When all required approvals for disabling alarm are obtained, perform the following and complete Form OPDP-4-1, Item 9:

1. Review disabling steps.
2. Initiate Compensatory Monitoring as described in Forms OPDP-4-3 and/or OPDP-4-5, if applicable.
3. Designate a qualified individual to disable alarm(s) using method in Form OPDP-4-1, and another individual to perform verification.
4. Place a disabled input indicator on each affected alarm window to indicate an input to that alarm has been disabled.
5. Sign Form OPDP-4-1, Item 9 “Performed By” and submit to a qualified individual to sign verification requirements.

NOTE

A Form OPDP-4-2 is to be maintained in each Disabled Annunciator Book. More than one entry per sheet is allowed.

6. Log alarm disablement on Form OPDP-4-2, “Disabled Alarm Index Sheet” in the associated Disabled Annunciator Book. Initiate a new sheet if required.
7. Log alarm disablement in the narrative log for the affected unit(s) when alarm disabled. The narrative log entry shall include alarm location, method used to disable alarm, date and time removed, justification for disablement, and Tech Spec action requirements, if applicable.

3.5 Review and Audit (continued)

2. Monitoring plan

3. Length of time the alarm is disabled

4.0 RECORDS

4.1 QA-Records

A. Disabled Alarm Checklist OPDP-4-1

B. Annunciator Disablement Technical Evaluation OPDP-4-5

4.2 Non-QA Records

A. Disabled Alarm Index Sheet OPDP-4-2

B. Disabled Alarm Compensatory Measures OPDP-4-3

5.0 DEFINITIONS

Disabled Input Indicator -

- BFN - A blue magnetic border labeled “Disabled Alarm Input.”
- SQN- A blue dot (sticker) or numbered square attached to the window with the applicable details listed in the MCR Turnover Package.
- WBN- An orange plastic lens cover labeled “Disabled Alarm” which snaps over the affected window and a blue plastic lens cover labeled “Disabled Input.”

Out-of-Service Indicator -

- BFN - A white magnetic border labeled “Testing/Maintenance.”
- SQN - An orange sticker attached to the window.
- WBN - A green plastic lens cover labeled “Maintenance” which snaps over the affected window.

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

Examination Outline Cross-reference:

Equipment Control

G2.2.22 (10CFR 55.41.5)

Knowledge of limiting conditions for operation and safety limits

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.2.22	
Importance Rating	4.0	------

Proposed Question: **# 62**

Unit 1 was operating at 100% RTP when an event resulted in the following conditions:

- Reactor Pressure peaked at 1300 psig
- Reactor Water Level indicates (-) 170 inches

Given the conditions above, the Tech Spec Safety Limit for ___**(1)**___ has been violated.

The REQUIRED ACTION is to restore within the Reactor Safety Limit and insert **ALL** insertable Control Rods within ___**(2)**___ .

A. (1) Reactor Pressure
(2) 2 hours

B. (1) Reactor Pressure
(2) 10 hours

C. (1) Reactor Water Level
(2) 2 hours

D. (1) Reactor Water Level
(2) 10 hours

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that in accordance with Technical Specifications 2.0, Safety Limits, the Reactor Coolant System Pressure Safety Limit is ≤ 1325 psig. 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 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 that all insertable Control Rods should be inserted and the plant placed in MODE 2 within 10 hours in accordance with LCO 3.0.3.

C **CORRECT:** (*See attached*) In accordance with Technical Specifications 2.0, Safety Limits, Reactor Water Level shall be greater than the top of active irradiated fuel, or TAF (-) 162 inches. For second part, in accordance with Tech Specs, when any Safety Limit has been violated, all insertable Control Rods shall be inserted within 2 hours.

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

RO Level Justification: Tests candidate's knowledge of Limiting Conditions for Operation and Safety Limits for Reactor Pressure and Reactor Water Level and the associated actions for exceeding a Technical Specification Safety Limit. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): (Attach if not previously provided)

Unit 1 Tech Spec 2.0, Amend. 299
Unit 1 Tech Spec 3.0, Amend. 239
U1 Tech Spec Bases 2.0, Rev. 0
OPDP-8, Rev. 28
0-TI-394, Attachment 7, Rev. 13

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.087, Obj. 14 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 21-04 #64	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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:

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

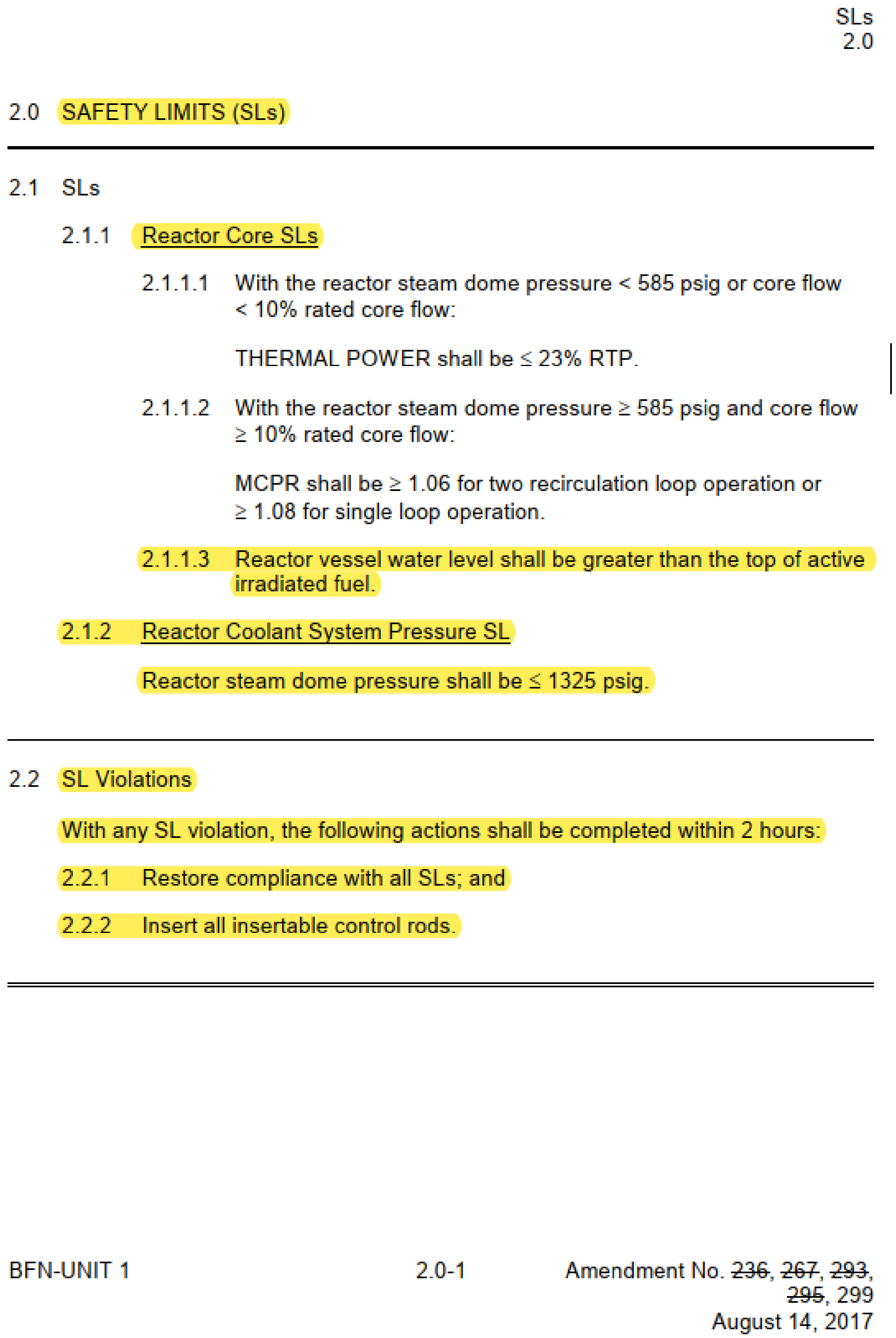
B. (1) 1325 psig
(2) 10 hours

C. (1) 1375 psig
(2) 2 hours

D. (1) 1375 psig
(2) 10 hours

Proposed Answer: **A**

Excerpt from Tech Spec 2.0:



SLs
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 ≤ 23% 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

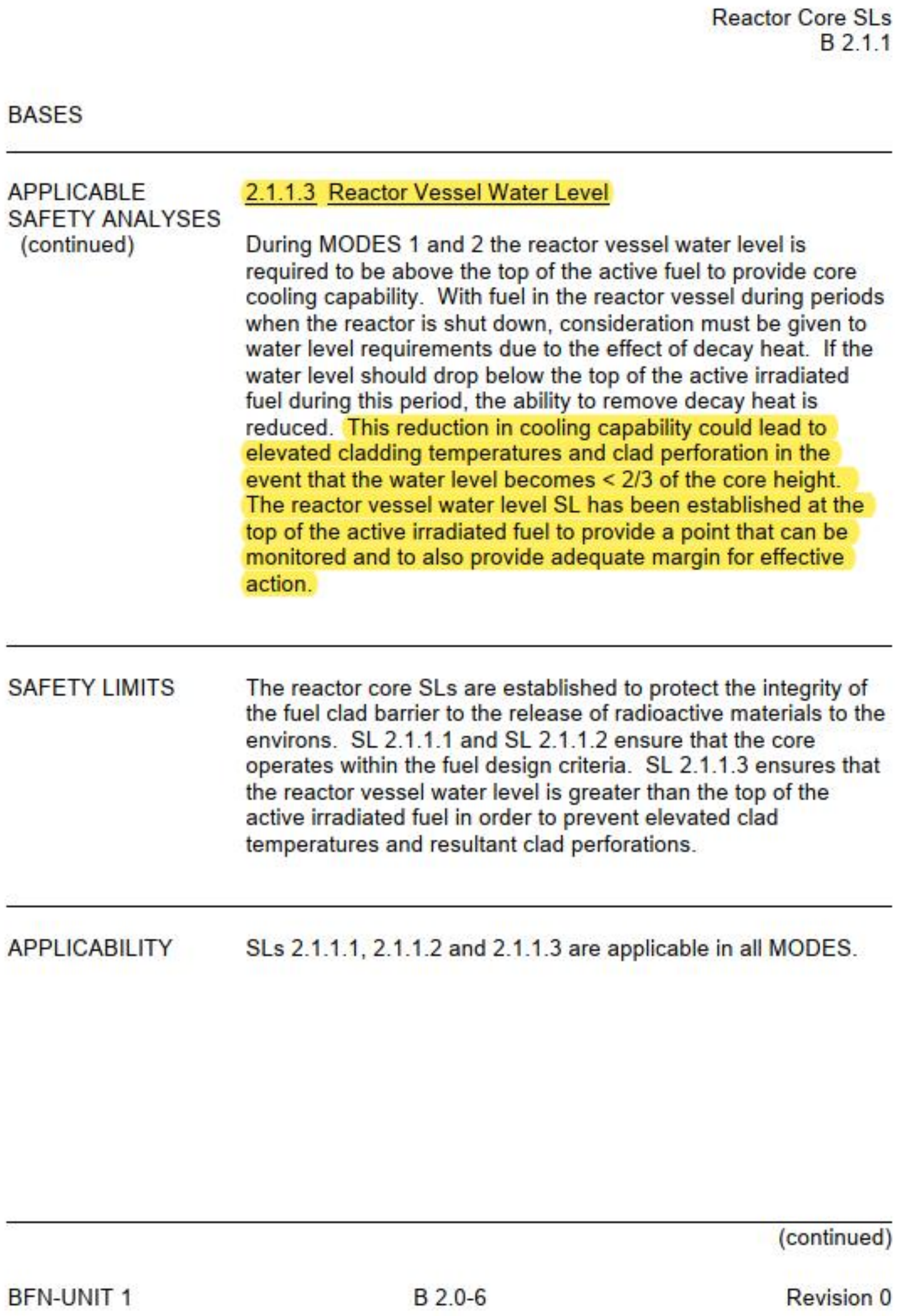
With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

BFN-UNIT 1 2.0-1 Amendment No. ~~236~~, ~~267~~, ~~293~~,
~~295~~, 299
August 14, 2017

Excerpts from Tech Spec 2.0 Bases:



Reactor Core SLs
B 2.1.1

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

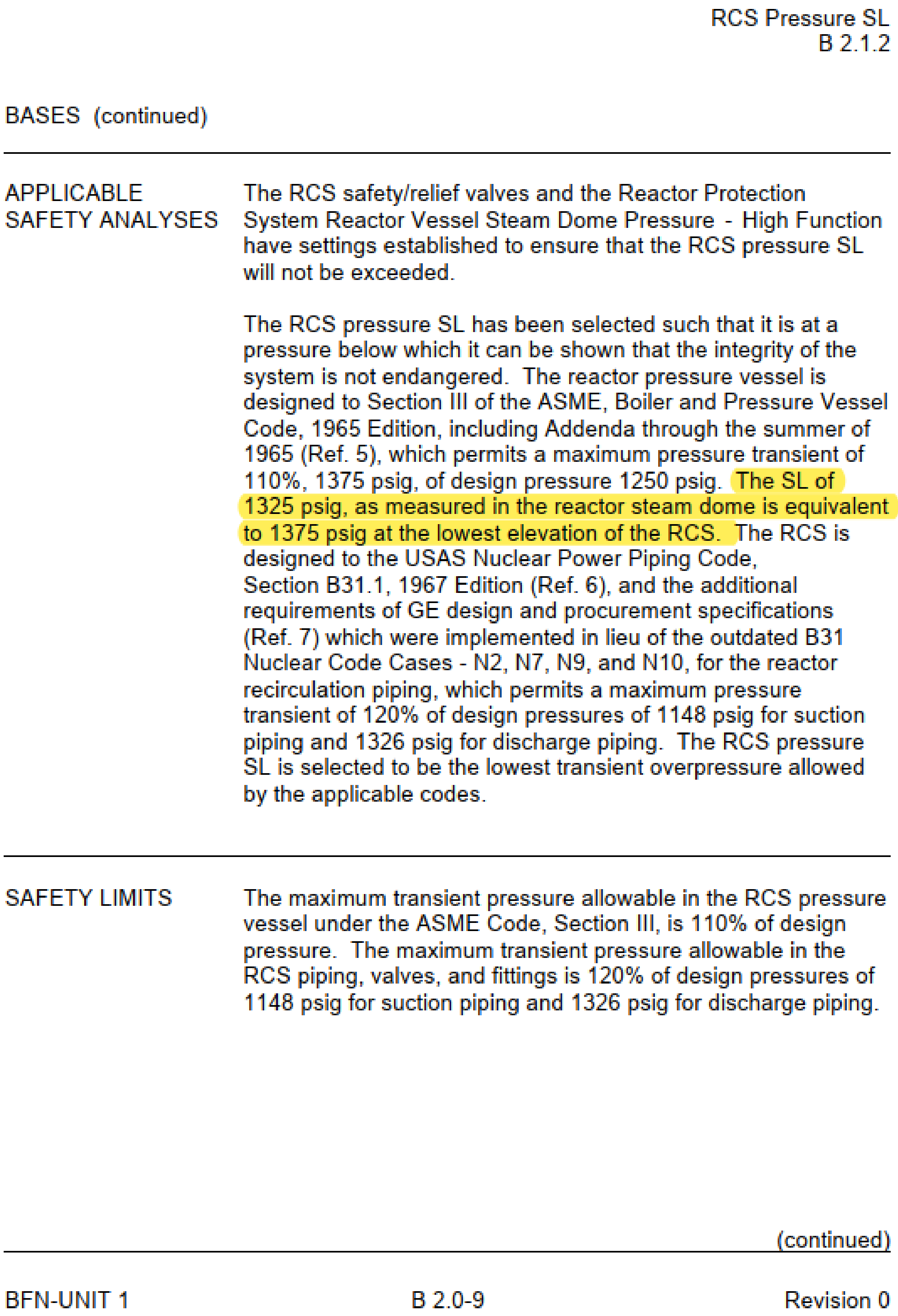
APPLICABILITY

SLs 2.1.1.1, 2.1.1.2 and 2.1.1.3 are applicable in all MODES.

(continued)

BFN-UNIT 1 B 2.0-6 Revision 0

Supports Distractors A(1), B(1)



RCS Pressure SL
B 2.1.2

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.

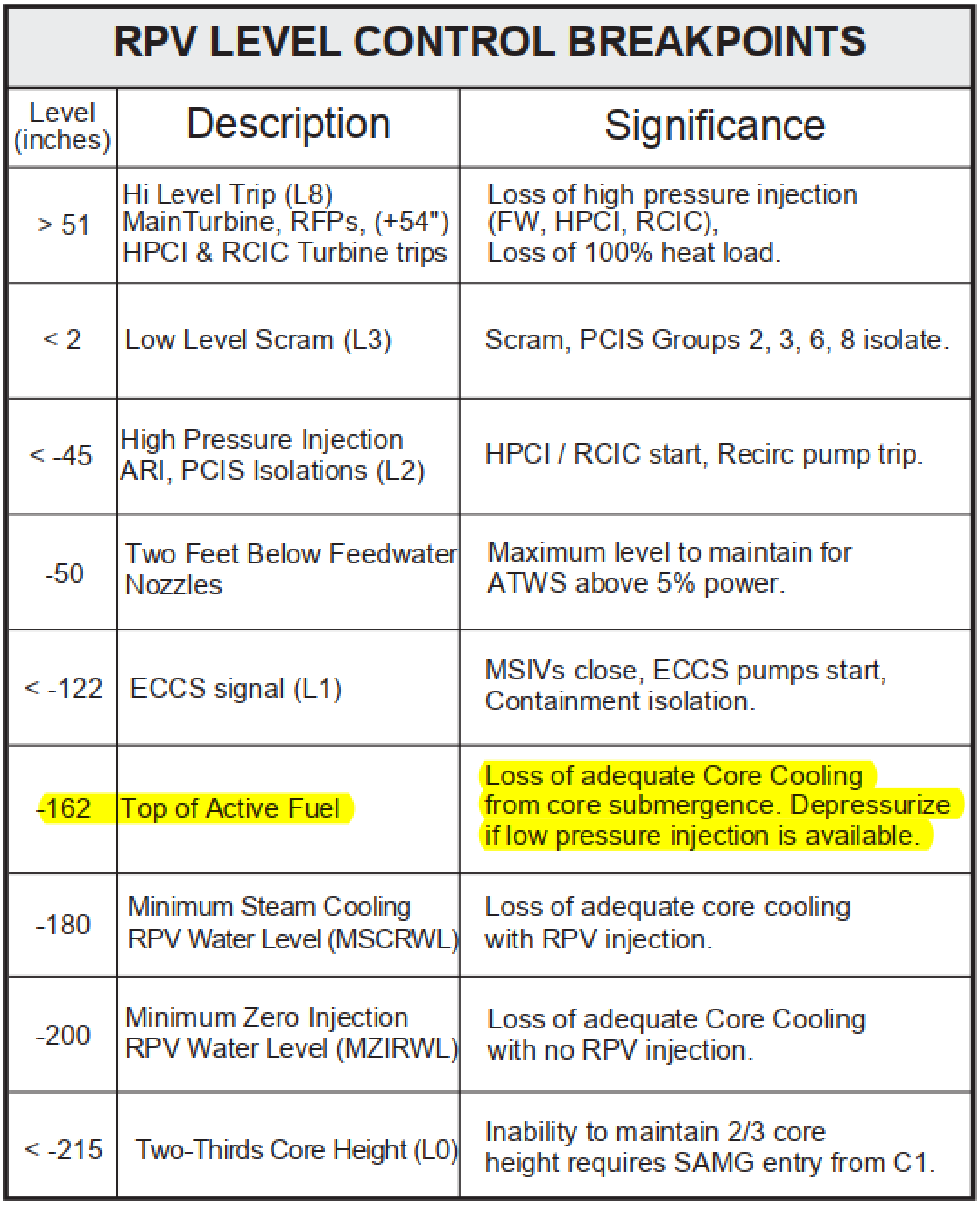
SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 120% of design pressures of 1148 psig for suction piping and 1326 psig for discharge piping.

(continued)

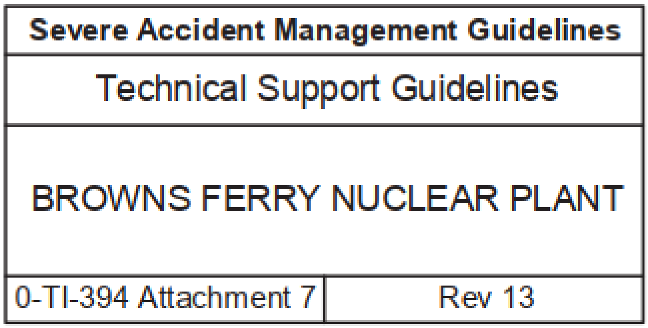
BFN-UNIT 1 B 2.0-9 Revision 0

Excerpt from 0-TI-394, Attachment 7: Illustrates that Reactor Water Level (-) 162 inches is Top of Active Fuel (TAF)



RPV LEVEL CONTROL BREAKPOINTS

Level (inches)	Description	Significance
> 51	Hi Level Trip (L8) MainTurbine, RFPs, (+54") HPCI & RCIC Turbine trips	Loss of high pressure injection (FW, HPCI, RCIC), Loss of 100% heat load.
< 2	Low Level Scram (L3)	Scram, PCIS Groups 2, 3, 6, 8 isolate.
< -45	High Pressure Injection ARI, PCIS Isolations (L2)	HPCI / RCIC start, Recirc pump trip.
-50	Two Feet Below Feedwater Nozzles	Maximum level to maintain for ATWS above 5% power.
< -122	ECCS signal (L1)	MSIVs close, ECCS pumps start, Containment isolation.
-162	Top of Active Fuel	Loss of adequate Core Cooling from core submergence. Depressurize if low pressure injection is available.
-180	Minimum Steam Cooling RPV Water Level (MSCRWL)	Loss of adequate core cooling with RPV injection.
-200	Minimum Zero Injection RPV Water Level (MZIRWL)	Loss of adequate Core Cooling with no RPV injection.
< -215	Two-Thirds Core Height (L0)	Inability to maintain 2/3 core height requires SAMG entry from C1.



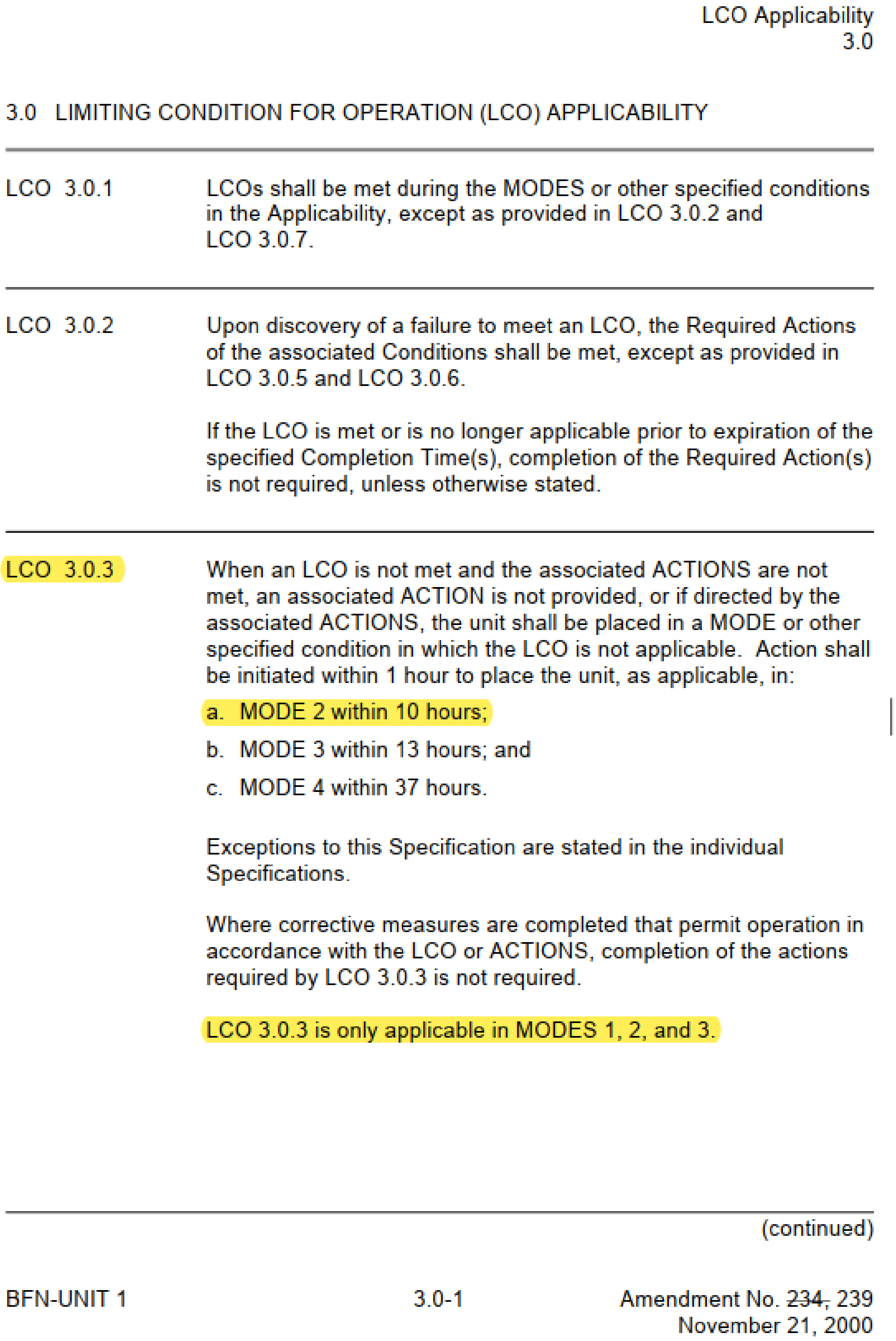
Severe Accident Management Guidelines

Technical Support Guidelines

BROWNS FERRY NUCLEAR PLANT

0-TI-394 Attachment 7	Rev 13

Excerpt from Tech Spec 3.0: Supports Distractors B(2), D(2)



LCO Applicability
3.0

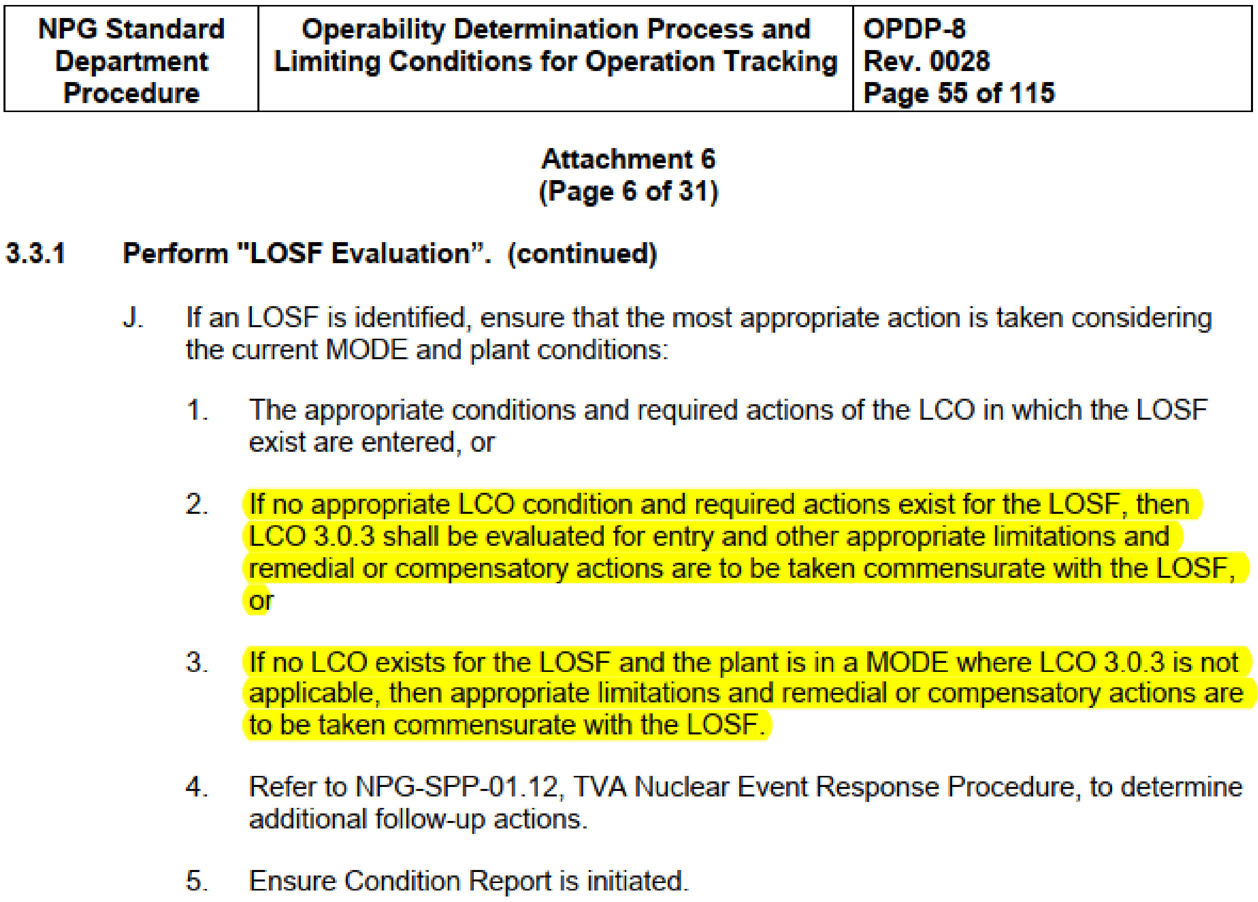
3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

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)

BFN-UNIT 1 3.0-1 Amendment No. ~~234,~~ 239
November 21, 2000

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. 0028** **Page 55 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:
295024 (EPE 1) High Drywell Pressure / 5
EK3.02 (10CFR 55.41.5)
Knowledge of the reasons for the following responses or actions as they apply to High Drywell Pressure:

- Suppression Pool Spray

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295024EK3.02	
Importance Rating	4.1	------

Proposed Question: **# 63**

In accordance with the PC Pressure leg of 2-EOI-2, Primary Containment Control and the associated Bases, Suppression Chamber Sprays are initiated to prevent ____**(1)**____ when Primary Containment Pressure cannot be maintained below ____**(2)**____.

A. (1) chugging in the downcomers
(2) 1.96 psig

B. (1) chugging in the downcomers
(2) 2.45 psig

C. (1) over-pressurizing the Suppression Chamber
(2) 1.96 psig

D. (1) over-pressurizing the Suppression Chamber
(2) 2.45 psig

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B)*. Second part is incorrect but plausible in that 1.96 psig is the interlock permissive Containment Pressure required for Primary Containment Sprays.

B **CORRECT:** *(See attached)* In accordance with EOIPM Section 0-V(G), EOI-2 Primary Containment Control Bases, states Suppression Chamber Sprays are initiated to prevent chugging in the downcomer openings as steam collapses. For second part, 2-EOI-2, Primary Containment Control, states to initiate Suppression Chamber Sprays when Primary Containment Pressure cannot be maintained below 2.45 psig.

C INCORRECT: First part is incorrect but plausible in that with a leak into Primary Containment, it can be mistaken that sprays are to condense steam in the air space to prevent over-pressurizing containment. 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 reasons and interlocks for initiating Suppression Chamber Sprays as it relates to High Drywell Pressure. This question is rated as Memory due to the fact that it requires the strict recall of facts.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): EOIPM 0-V(G), Rev. 1 (Attach if not previously provided)
2-EOI-2, Rev. 17
2-OI-2, Rev. 189

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.203 Obj. 7 (As available)

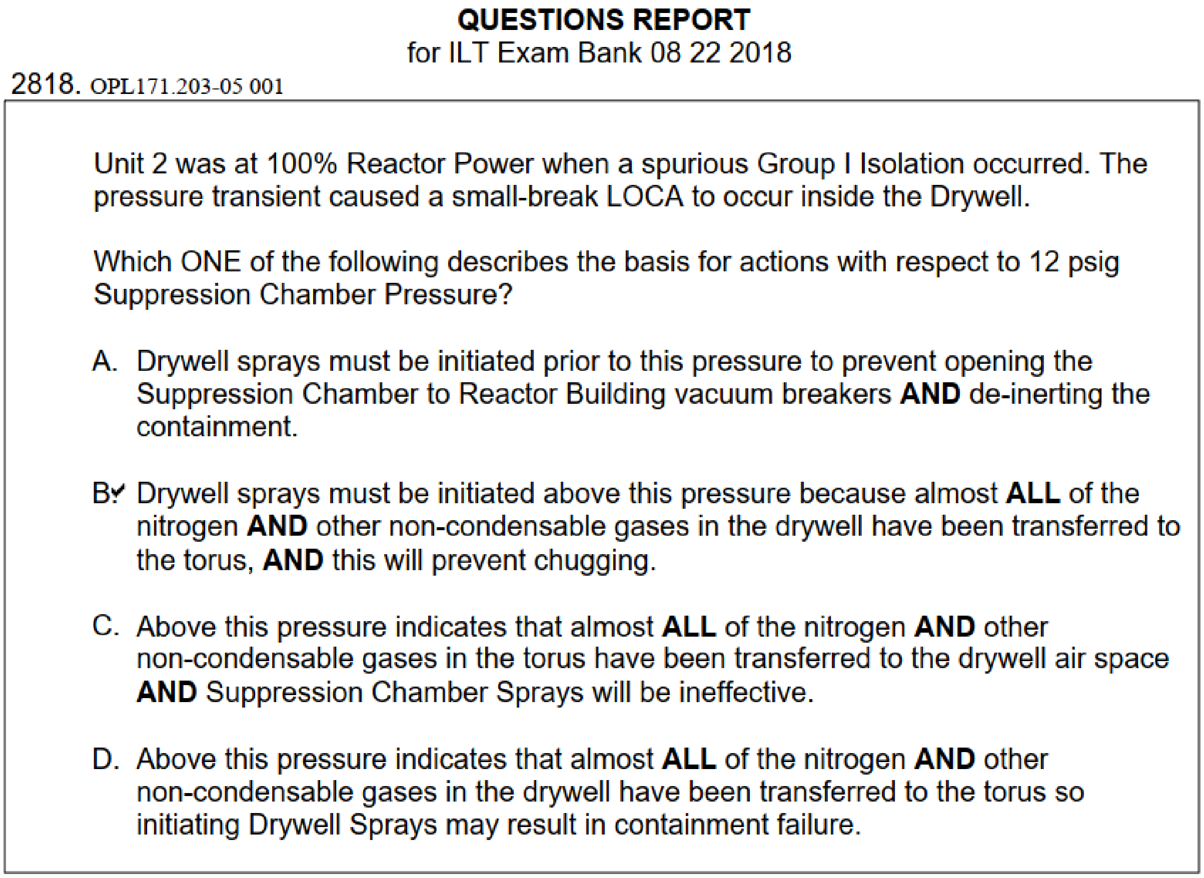
Question Source:	Bank #		
	Modified Bank #	OPL171.203-05 001 #2818	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

2818. OPL171.203-05 001

Unit 2 was at 100% Reactor Power when a spurious Group I Isolation occurred. The pressure transient caused a small-break LOCA to occur inside the Drywell.

Which ONE of the following describes the basis for actions with respect to 12 psig Suppression Chamber Pressure?

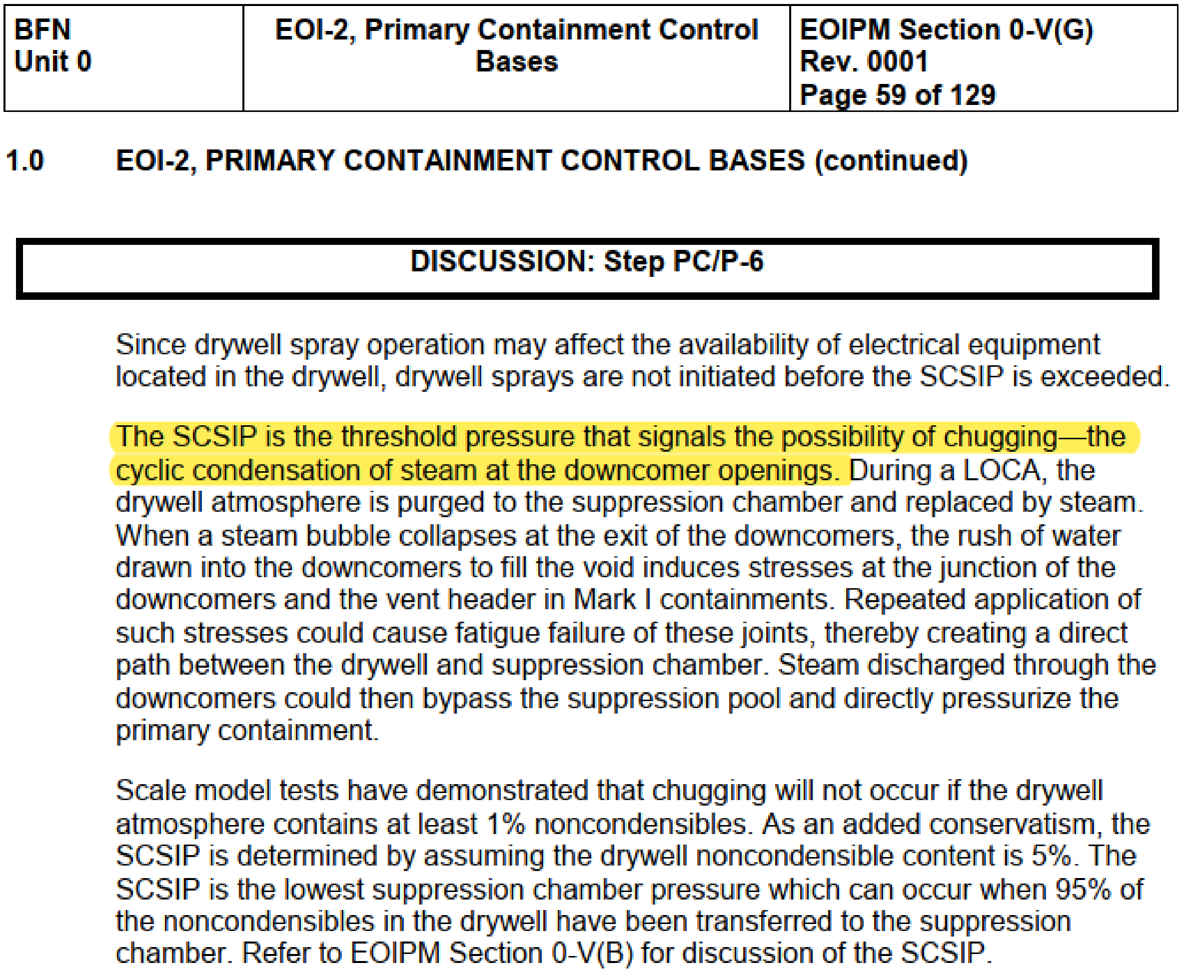
A. Drywell sprays must be initiated prior to this pressure to prevent opening the Suppression Chamber to Reactor Building vacuum breakers **AND** de-inerting the containment.

B✓ Drywell sprays must be initiated above this pressure because almost **ALL** of the nitrogen **AND** other non-condensable gases in the drywell have been transferred to the torus, **AND** this will prevent chugging.

C. Above this pressure indicates that almost **ALL** of the nitrogen **AND** other non-condensable gases in the torus have been transferred to the drywell air space **AND** Suppression Chamber Sprays will be ineffective.

D. Above this pressure indicates that almost **ALL** of the nitrogen **AND** other non-condensable gases in the drywell have been transferred to the torus so initiating Drywell Sprays may result in containment failure.

Excerpt from EOIPM Section 0-V(G):



BFN **Unit 0**	**EOI-2, Primary Containment Control Bases**	**EOIPM Section 0-V(G)** **Rev. 0001** **Page 59 of 129**

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

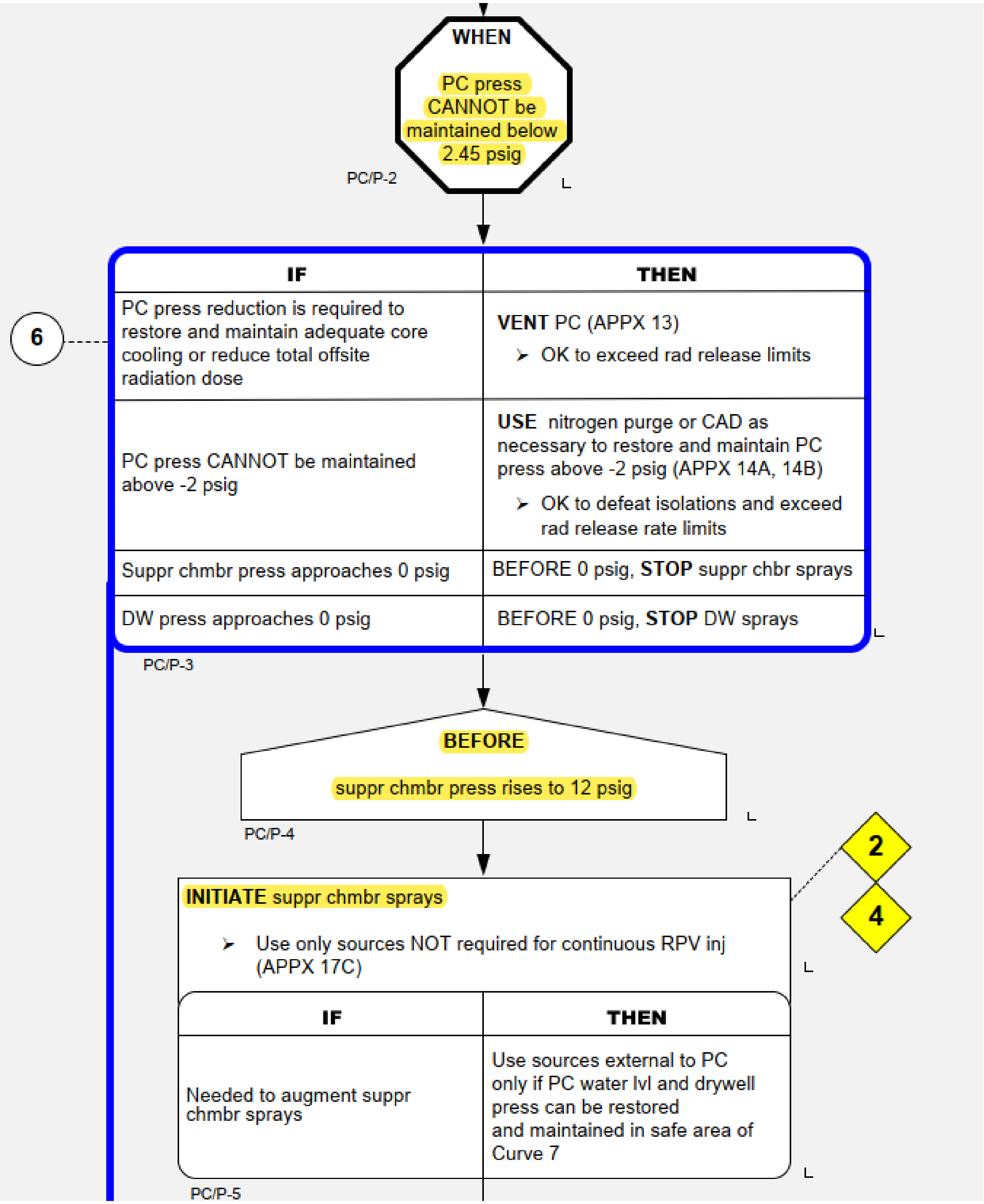
DISCUSSION: Step PC/P-6

Since drywell spray operation may affect the availability of electrical equipment located in the drywell, drywell sprays are not initiated before the SCSIP is exceeded.

The SCSIP is the threshold pressure that signals the possibility of chugging—the cyclic condensation of steam at the downcomer openings. During a LOCA, the drywell atmosphere is purged to the suppression chamber and replaced by steam. When a steam bubble collapses at the exit of the downcomers, the rush of water drawn into the downcomers to fill the void induces stresses at the junction of the downcomers and the vent header in Mark I containments. Repeated application of such stresses could cause fatigue failure of these joints, thereby creating a direct path between the drywell and suppression chamber. Steam discharged through the downcomers could then bypass the suppression pool and directly pressurize the primary containment.

Scale model tests have demonstrated that chugging will not occur if the drywell atmosphere contains at least 1% noncondensibles. As an added conservatism, the SCSIP is determined by assuming the drywell noncondensible content is 5%. The SCSIP is the lowest suppression chamber pressure which can occur when 95% of the noncondensibles in the drywell have been transferred to the suppression chamber. Refer to EOIPM Section 0-V(B) for discussion of the SCSIP.

Excerpt from 2-EOI-2:



WHEN

PC press CANNOT be maintained below 2.45 psig

IF	**THEN**
PC press reduction is required to restore and maintain adequate core cooling or reduce total offsite radiation dose	**VENT** PC (APPX 13) ➢ OK to exceed rad release limits
PC press CANNOT be maintained above -2 psig	**USE** nitrogen purge or CAD as necessary to restore and maintain PC press above -2 psig (APPX 14A, 14B) ➢ OK to defeat isolations and exceed rad release rate limits
Suppr chmbr press approaches 0 psig	BEFORE 0 psig, **STOP** suppr chbr sprays
DW press approaches 0 psig	BEFORE 0 psig, **STOP** DW sprays

BEFORE

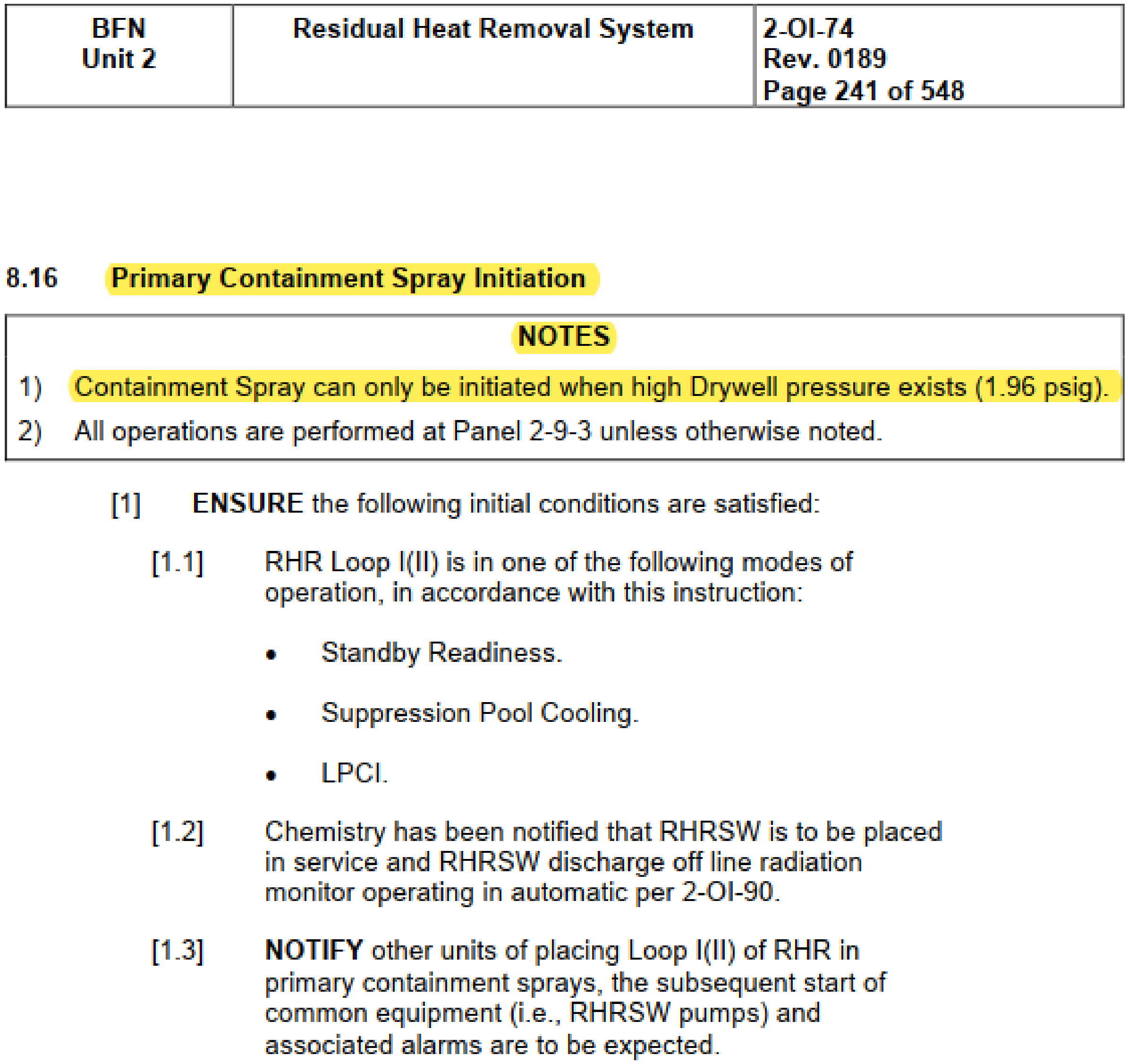
suppr chmbr press rises to 12 psig

INITIATE suppr chmbr sprays

- Use only sources NOT required for continuous RPV inj (APPX 17C)

IF	**THEN**
Needed to augment suppr chmbr sprays	Use sources external to PC only if PC water lvl and drywell press can be restored and maintained in safe area of Curve 7

Excerpt from 2-OI-74: Supports Distractors A(2), C(2)



BFN **Unit 2**	**Residual Heat Removal System**	**2-OI-74** **Rev. 0189** **Page 241 of 548**

8.16 Primary Containment Spray Initiation

NOTES

1) Containment Spray can only be initiated when high Drywell pressure exists (1.96 psig).

2) All operations are performed at Panel 2-9-3 unless otherwise noted.

[1] **ENSURE** the following initial conditions are satisfied:

[1.1] RHR Loop I(II) is in one of the following modes of operation, in accordance with this instruction:

- Standby Readiness.
- Suppression Pool Cooling.
- LPCI.

[1.2] Chemistry has been notified that RHRSW is to be placed in service and RHRSW discharge off line radiation monitor operating in automatic per 2-OI-90.

[1.3] **NOTIFY** other units of placing Loop I(II) of RHR in primary containment sprays, the subsequent start of common equipment (i.e., RHRSW pumps) and associated alarms are to be expected.

Examination Outline Cross-reference:	Level	RO	SRO
Radiation Control **G2.3.12 (**10CFR 55.41.12) Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters.	Tier #	3	------
	Group #	------	------
	K/A #	G2.3.12	
	Importance Rating	3.2	------

Proposed Question: **# 64**

Unit 1 was conducting a Refueling Outage when the following conditions occurred:

- Bridge Operator lowered a fuel assembly into the Reactor Core
- Unexpected criticality is observed in the Main Control Room

Given the above conditions, in accordance with 1-AOI-79-2, Inadvertent Criticality During Incore Fuel Movements, the Bridge Operator is required to **IMMEDIATELY** __________.

A. **ISOLATE** RWCU

B. **STOP** the CRD Pump

C. **START** an SLC Pump

D. **REMOVE** the fuel assembly from the Reactor Core

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that 1-AOI-79-2 states IF criticality is still evident, AND at the direction of the Unit Supervisor, PERFORM the given provided above ONLY as a Subsequent Action NOT as an Immediate Action.

B INCORRECT: Incorrect but plausible in that 1-AOI-79-2 states IF criticality is still evident, AND at the direction of the Unit Supervisor, PERFORM the given provided above ONLY as a Subsequent Action NOT as an Immediate Action.

C INCORRECT: Incorrect but plausible in that 1-AOI-79-2 states IF criticality is still evident, AND at the direction of the Unit Supervisor, PERFORM the given provided above ONLY as a Subsequent Action NOT as an Immediate Action.

D **CORRECT**: (*See attached*) In accordance with 1-AOI-79-2, Inadvertent Criticality During Incore Fuel Movements, Section 4.1 - Immediate Actions [3], **IF** unexpected criticality is observed following the insertion of a fuel assembly, **THEN IMMEDIATELY REMOVE** the fuel assembly from the Reactor Core.

RO Level Justification: Tests the candidate's knowledge of the required Immediate Operator Actions pertaining to abnormal plant conditions during fuel handling to ensure radiological safety. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): 1-AOI-79-2, Rev. 2 (Attach if not previously provided)

OPDP, Rev. 3

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.060, Obj. 3b (As available)

Question Source:	Bank #	BFN NRC 18-04 #68	
	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:

Proposed Question: **# 68**

Unit 2 was shutdown 8 days ago for a Refueling Outage with the following conditions:

- Reactor Cavity is flooded and Spent Fuel Pool Gates are removed
- Core Alterations are in progress

A new fuel assembly is inserted into the core. Unexpected criticality is observed as indicated by rising count rate on multiple SRM meters.

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

The correct **IMMEDIATE ACTION** in accordance with 2-AOI-79-2, Inadvertent Criticality During Incore Fuel Movements, is to ___________.

A. **VERIFY** all Control Rods are inserted

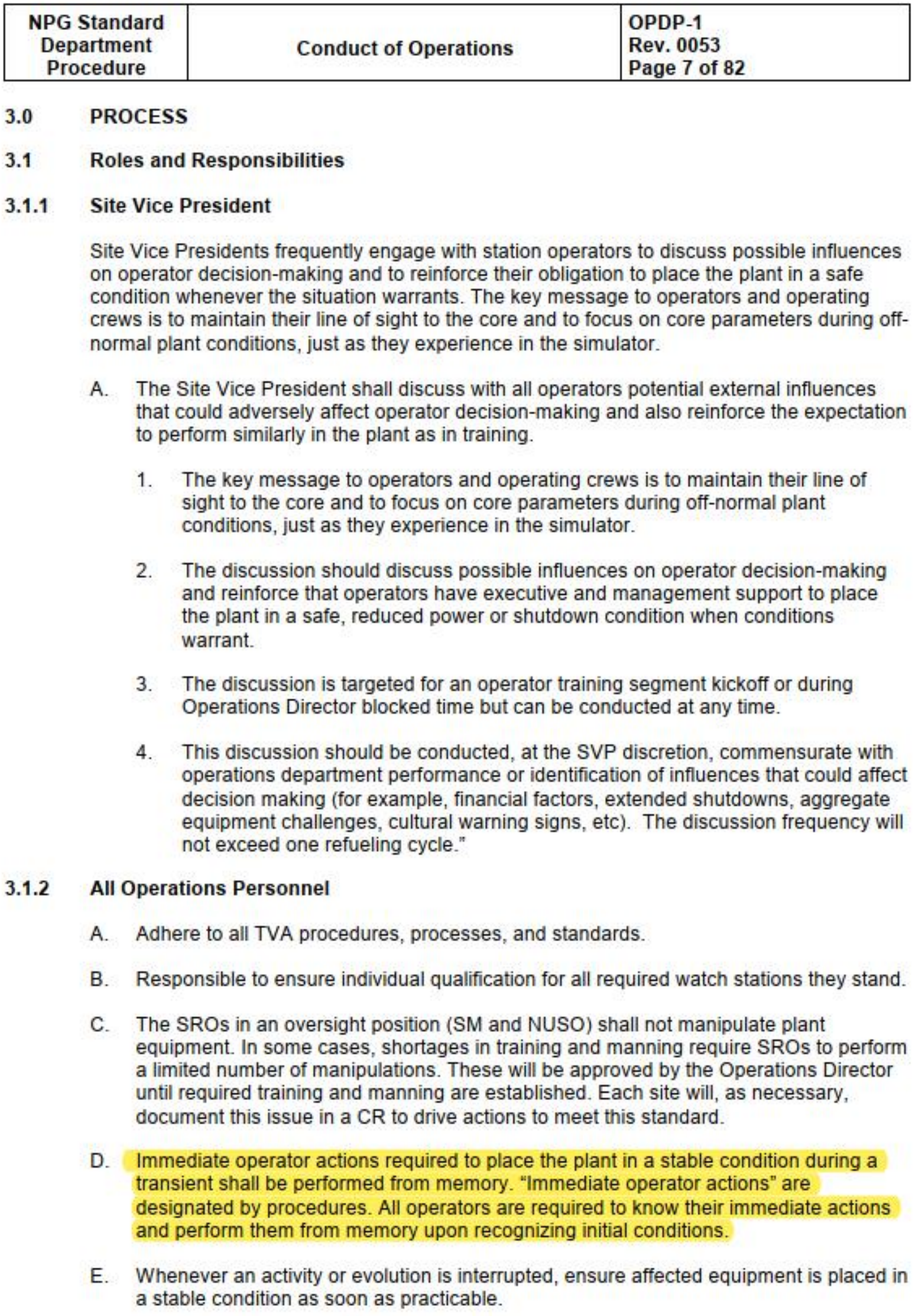
B. **UNLOCK** and **PLACE** 2-HS-063-0006A, SLC PUMP 2A/2B, control switch in START-A or START-B position

C. **TRAVERSE** the refueling bridge and fuel assembly away from the Reactor core, preferably to the area of the cattle chute

D. **VERIFY** fuel grapple latched onto the fuel assembly handle AND immediately **REMOVE** the fuel assembly from the Reactor core

Proposed Answer: **D**

Excerpts from OPDP 1:



NPG Standard Department Procedure	**Conduct of Operations**	**OPDP-1** **Rev. 0053** **Page 7 of 82**

3.0 PROCESS

3.1 Roles and Responsibilities

3.1.1 Site Vice President

Site Vice Presidents frequently engage with station operators to discuss possible influences on operator decision-making and to reinforce their obligation to place the plant in a safe condition whenever the situation warrants. 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.

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.

Excerpts from 1-AOI-79-2:

BFN **Unit 1**	**Inadvertent Criticality During Incore Fuel Movements**	**1-AOI-79-2** **Rev. 0002** **Page 6 of 9**

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** unexpected criticality is observed following control rod withdrawal, **THEN**

REINSERT the control rod.

[2] **IF** all control rods can NOT be fully inserted, **THEN**

MANUALLY SCRAM the Reactor.

[3] **IF** unexpected criticality is observed following the insertion of a fuel assembly, **THEN**

PERFORM the following:

[3.1] **ENSURE** fuel grapple latched onto the fuel assembly handle AND **IMMEDIATELY REMOVE** the fuel assembly from the Reactor core.

[3.2] **IF** the Reactor can be determined to be subcritical **AND** no radiological hazard is apparent, **THEN**

PLACE the fuel assembly in a spent fuel storage pool location with the least possible number of surrounding fuel assemblies and **LEAVE** the fuel grapple latched to the fuel assembly handle.

[3.3] **IF** the Reactor can NOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

TRAVERSE the Refueling Bridge and fuel assembly away from the Reactor core, preferably to the area of the cattle chute and **CONTINUE** at Step 4.1[4].

[4] **IF** the Reactor can NOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

EVACUATE the refuel floor.

BFN **Unit 1**	**Inadvertent Criticality During Incore Fuel Movements**	**1-AOI-79-2** **Rev. 0002** **Page 7 of 9**

4.2 Subsequent Actions

[1] **NOTIFY** the Shift Manager and Reactor Engineer.

[2] **IF** any EOI entry condition is met, **THEN**

ENTER the appropriate EOIs.

[3] **ENSURE** all control rods are inserted.

[4] **IF** criticality is still evident **AND** at the direction of the Unit SRO, **THEN**

PERFORM the following:

[4.1] **IF** the CRD pump is in operation, **THEN**

STOP the CRD pump.

[4.2] **IF** the RWCU system is in service, **THEN**

ISOLATE RWCU as follows:

[4.2.1] **CLOSE** 1-FCV-069-0001 using RWCU INBD SUCT ISOLATION VALVE, 1-HS-69-1.

[4.2.2] **CLOSE** 1-FCV-069-0002 using RWCU OUTBD SUCT ISOLATION VALVE, 1-HS-69-2A.

[4.3] **IF** SLC is operable, **THEN**

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

[5] **NOTIFY** RADCON to conduct surveys to determine radiation levels on Refuel Floor.

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.

[6] **NOTIFY** Chemistry to sample and analyze the Reactor water.

[7] **REFER TO** EPIP-1 for proper notifications.

Examination Outline Cross-reference:

Conduct of Operations

G2.1.29 (10CFR 55.41.10)

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.1.29	
Importance Rating	4.1	------

Proposed Question: **# 65**

In accordance with NPG-SPP-22.206, Verification Program, **(1)** verification is used for a locked throttle valve, and activities involving radiation exposure of greater than 10 mrem Total Effective Dose Equivalent (TEDE) **(2)** be exempted from verification requirements.

A. (1) concurrent
(2) can

B. (1) concurrent
(2) can NOT

C. (1) independent
(2) can

D. (1) independent
(2) can NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (*See Attached*) In accordance with NPG-SPP-22.206, Verification Program, concurrent verification is used for verification of locked throttle valve position. If a locked throttle valve was independently verified, the actions of the initial positioner or verification could be invalidated by the actions of the independent verifier. For second part, in accordance with NPG-22.206, activities involving significant exposure of more than 10 mrem TEDE are exempt from required verifications.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that clearance tag placement on valves is not exempt from verification when significant exposure is involved.

C INCORRECT: First part is incorrect but plausible in that independent verification is required to confirm that a non-throttle valve has been placed in the proper position such as during system alignments of safety-related or important equipment. 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 knowledge of the of conduct valve position verification requirements. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): NPG-SPP-22.206, Rev. 7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.110 Obj. 3, 4 (As available)

Question Source:	Bank #		
	Modified Bank #	OPL171.110-02.003 #2505	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

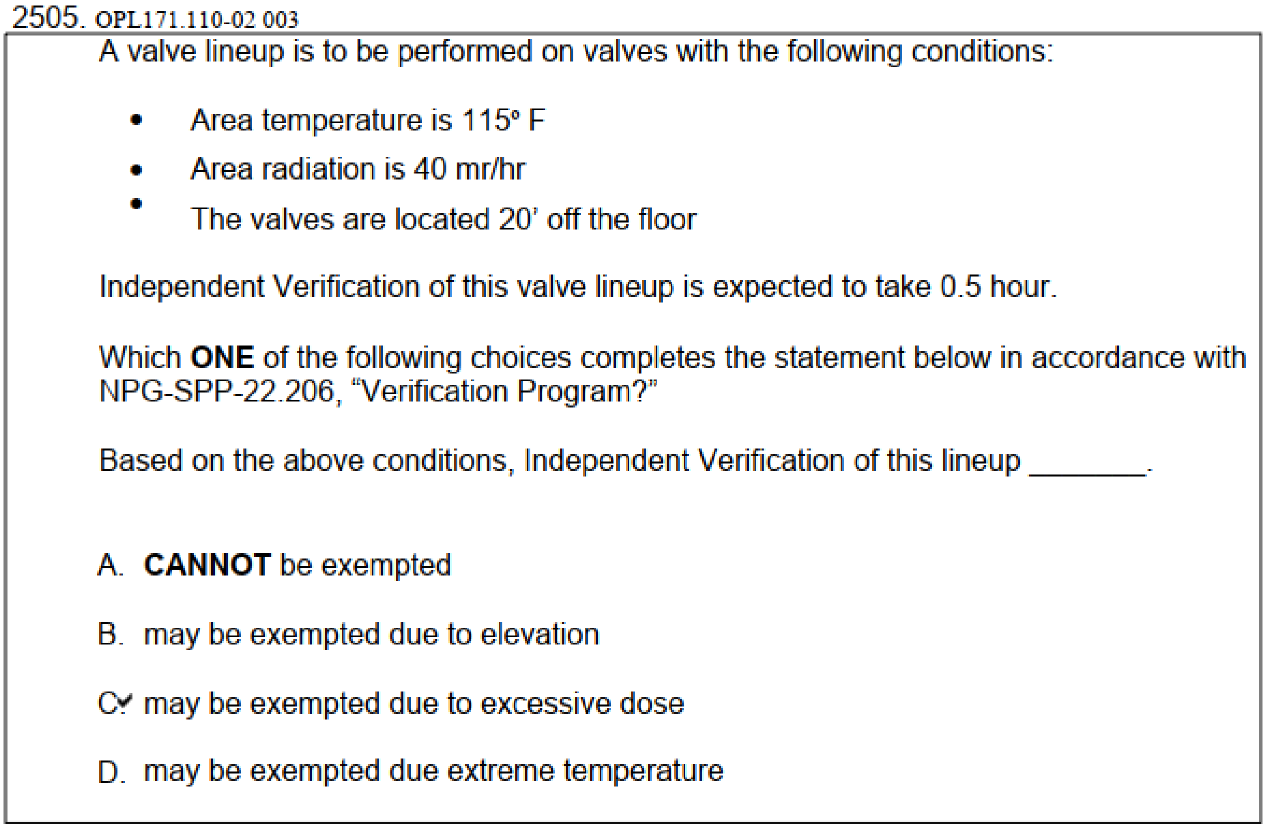
10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Copy of Bank Question:



2505. OPL171.110-02 003

A valve lineup is to be performed on valves with the following conditions:

- Area temperature is 115° F
- Area radiation is 40 mr/hr
- The valves are located 20’ off the floor

Independent Verification of this valve lineup is expected to take 0.5 hour.

Which **ONE** of the following choices completes the statement below in accordance with NPG-SPP-22.206, “Verification Program?”

Based on the above conditions, Independent Verification of this lineup _______.

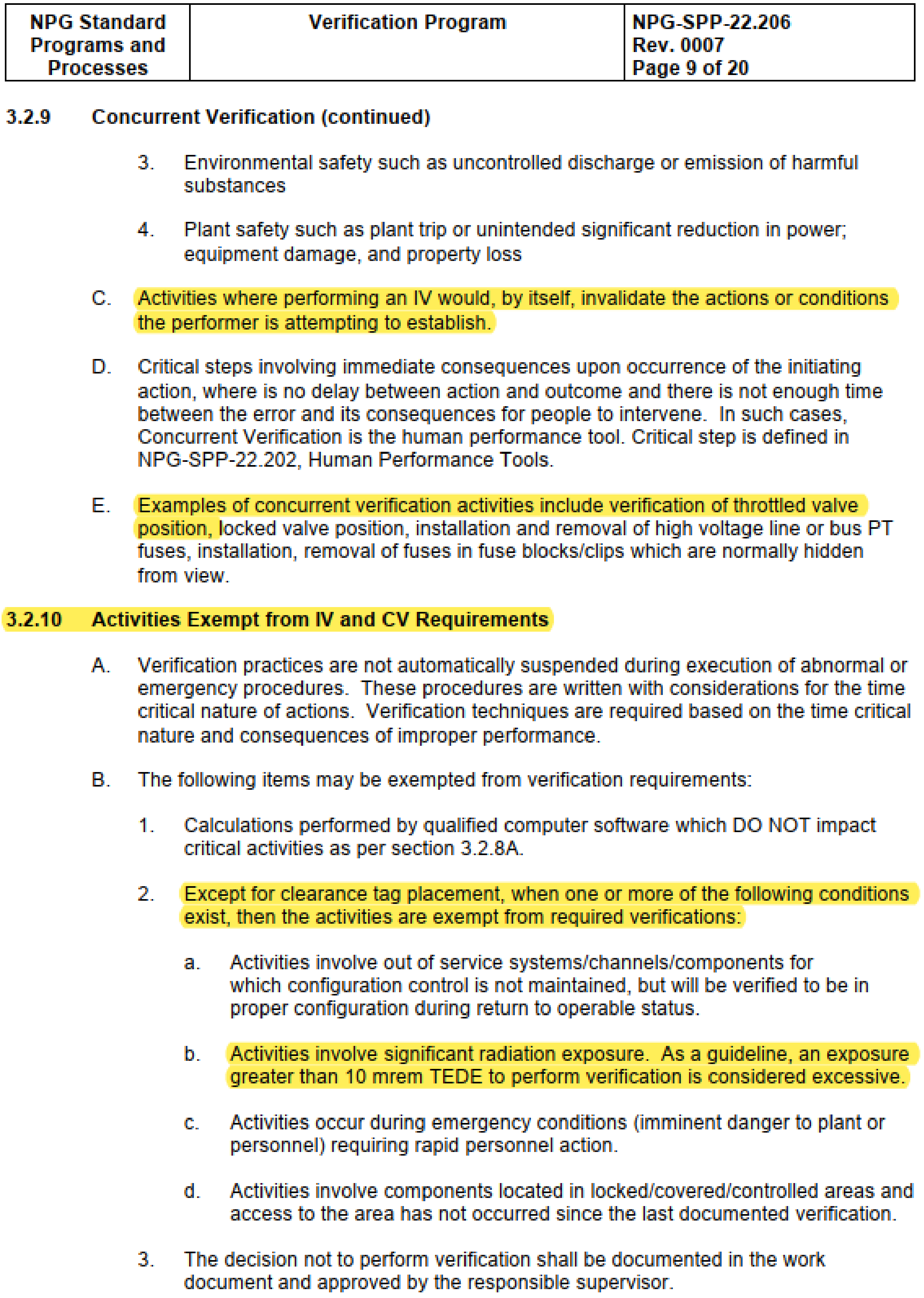
A. **CANNOT** be exempted

B. may be exempted due to elevation

C✓ may be exempted due to excessive dose

D. may be exempted due extreme temperature

Excerpts from NPG-SPP-22.206:



NPG Standard Programs and Processes	**Verification Program**	**NPG-SPP-22.206** **Rev. 0007** **Page 9 of 20**

3.2.9 Concurrent Verification (continued)

3. Environmental safety such as uncontrolled discharge or emission of harmful substances

4. Plant safety such as plant trip or unintended significant reduction in power; equipment damage, and property loss

C. Activities where performing an IV would, by itself, invalidate the actions or conditions the performer is attempting to establish.

D. Critical steps involving immediate consequences upon occurrence of the initiating action, where is no delay between action and outcome and there is not enough time between the error and its consequences for people to intervene. In such cases, Concurrent Verification is the human performance tool. Critical step is defined in NPG-SPP-22.202, Human Performance Tools.

E. Examples of concurrent verification activities include verification of throttled valve position, locked valve position, installation and removal of high voltage line or bus PT fuses, installation, removal of fuses in fuse blocks/clips which are normally hidden from view.

3.2.10 Activities Exempt from IV and CV Requirements

A. Verification practices are not automatically suspended during execution of abnormal or emergency procedures. These procedures are written with considerations for the time critical nature of actions. Verification techniques are required based on the time critical nature and consequences of improper performance.

B. The following items may be exempted from verification requirements:

1. Calculations performed by qualified computer software which DO NOT impact critical activities as per section 3.2.8A.

2. Except for clearance tag placement, when one or more of the following conditions exist, then the activities are exempt from required verifications:

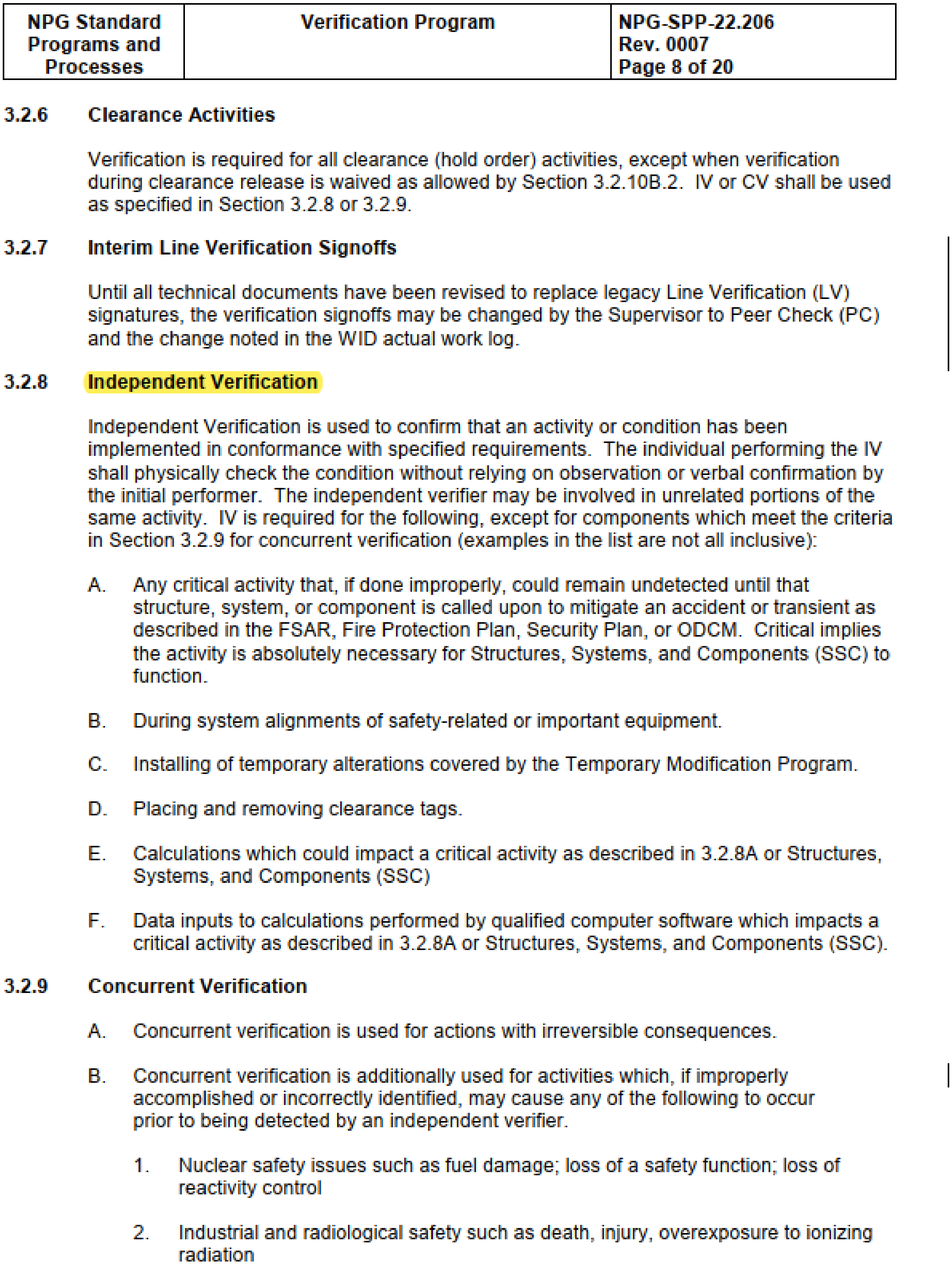
   a. Activities involve out of service systems/channels/components for which configuration control is not maintained, but will be verified to be in proper configuration during return to operable status.

   b. Activities involve significant radiation exposure. As a guideline, an exposure greater than 10 mrem TEDE to perform verification is considered excessive.

   c. Activities occur during emergency conditions (imminent danger to plant or personnel) requiring rapid personnel action.

   d. Activities involve components located in locked/covered/controlled areas and access to the area has not occurred since the last documented verification.

3. The decision not to perform verification shall be documented in the work document and approved by the responsible supervisor.



NPG Standard Programs and Processes	**Verification Program**	**NPG-SPP-22.206** **Rev. 0007** **Page 8 of 20**

3.2.6 Clearance Activities

Verification is required for all clearance (hold order) activities, except when verification during clearance release is waived as allowed by Section 3.2.10B.2. IV or CV shall be used as specified in Section 3.2.8 or 3.2.9.

3.2.7 Interim Line Verification Signoffs

Until all technical documents have been revised to replace legacy Line Verification (LV) signatures, the verification signoffs may be changed by the Supervisor to Peer Check (PC) and the change noted in the WID actual work log.

3.2.8 Independent Verification

Independent Verification is used to confirm that an activity or condition has been implemented in conformance with specified requirements. The individual performing the IV shall physically check the condition without relying on observation or verbal confirmation by the initial performer. The independent verifier may be involved in unrelated portions of the same activity. IV is required for the following, except for components which meet the criteria in Section 3.2.9 for concurrent verification (examples in the list are not all inclusive):

A. Any critical activity that, if done improperly, could remain undetected until that structure, system, or component is called upon to mitigate an accident or transient as described in the FSAR, Fire Protection Plan, Security Plan, or ODCM. Critical implies the activity is absolutely necessary for Structures, Systems, and Components (SSC) to function.

B. During system alignments of safety-related or important equipment.

C. Installing of temporary alterations covered by the Temporary Modification Program.

D. Placing and removing clearance tags.

E. Calculations which could impact a critical activity as described in 3.2.8A or Structures, Systems, and Components (SSC)

F. Data inputs to calculations performed by qualified computer software which impacts a critical activity as described in 3.2.8A or Structures, Systems, and Components (SSC).

3.2.9 Concurrent Verification

A. Concurrent verification is used for actions with irreversible consequences.

B. Concurrent verification is additionally used for activities which, if improperly accomplished or incorrectly identified, may cause any of the following to occur prior to being detected by an independent verifier.

  1. Nuclear safety issues such as fuel damage; loss of a safety function; loss of reactivity control

  2. Industrial and radiological safety such as death, injury, overexposure to ionizing radiation

Examination Outline Cross-reference:	Level	RO	SRO
293005 Thermodynamics - Cycles	Tier #	4	------
K1.06 (10CFR 55.41.14)	Group #	------	------
Describe how changes in system parameters affect thermodynamic efficiency.	K/A #	293005K1.06	
	Importance Rating	2.6	------

Proposed Question: **# 66**

Unit 1 is operating at 100% RTP.

Isolating Extraction Steam to the 1A1 Feedwater Heater will cause Reactor Core Thermal Power to **(1)** and steam cycle efficiency will **(2)** .

A. (1) lower
(2) rise

B. (1) lower
(2) lower

C. (1) rise
(2) rise

D. (1) rise
(2) lower

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the complexity of the Extraction Steam and Heater Drain System is often confusing for candidates. With a total of 15 high pressure and low pressure Feedwater heaters, 6 moisture separators, and 5 Extraction Steam supply locations from the high pressure and low pressure (various stages) Main Turbine, it is plausible to easily confuse system operation. Example, if Extraction Steam is isolated, generator output will actually lower slightly if the affected heater is not from the number one Feedwater heater (1A1, 1B1, or 1C1). Second part is incorrect but plausible if the candidate confuses the impact between Reactor Power and Reactor plant efficiency when Extraction Steam is isolated to high pressure heaters, where Reactor Power will rise.

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 1-OI-6, Feedwater Heating and Misc Drains System and 1-AOI-6, Feedwater Heater String/Extraction Steam Isolation, Reactor Power will rise when Extraction Steam is isolated to a Feedwater Heater. This occurs because without Extraction Steam, Feedwater entering the Reactor will be at a lower temperature, which causes more neutrons to be thermalized resulting in higher Reactor Power. For second part, Feedwater Temperature entering the Core lowers when Extraction Steam is lost, more Reactor Power must be used to heat the water in order to create steam for use by the Main Turbine. Therefore, plant thermal efficiency lowers.

RO Level Justification: Tests the candidate's knowledge of Generic Fundamentals, specifically the effect of a loss of Feedwater heating on Reactor Power and Reactor Plant efficiency. 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 must use the information provided concerning Extraction Steam and determine how the Reactor Plant will respond to the loss of Feedwater Heating Steam.

Technical Reference(s): 1-OI-6, Rev.41 (Attach if not previously provided)
1-AOI-6-1, Rev. 2
BWR GFES Thermodynamics Cycles, Ch. 5, Rev. 4

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.026, Obj. 7c (As available)

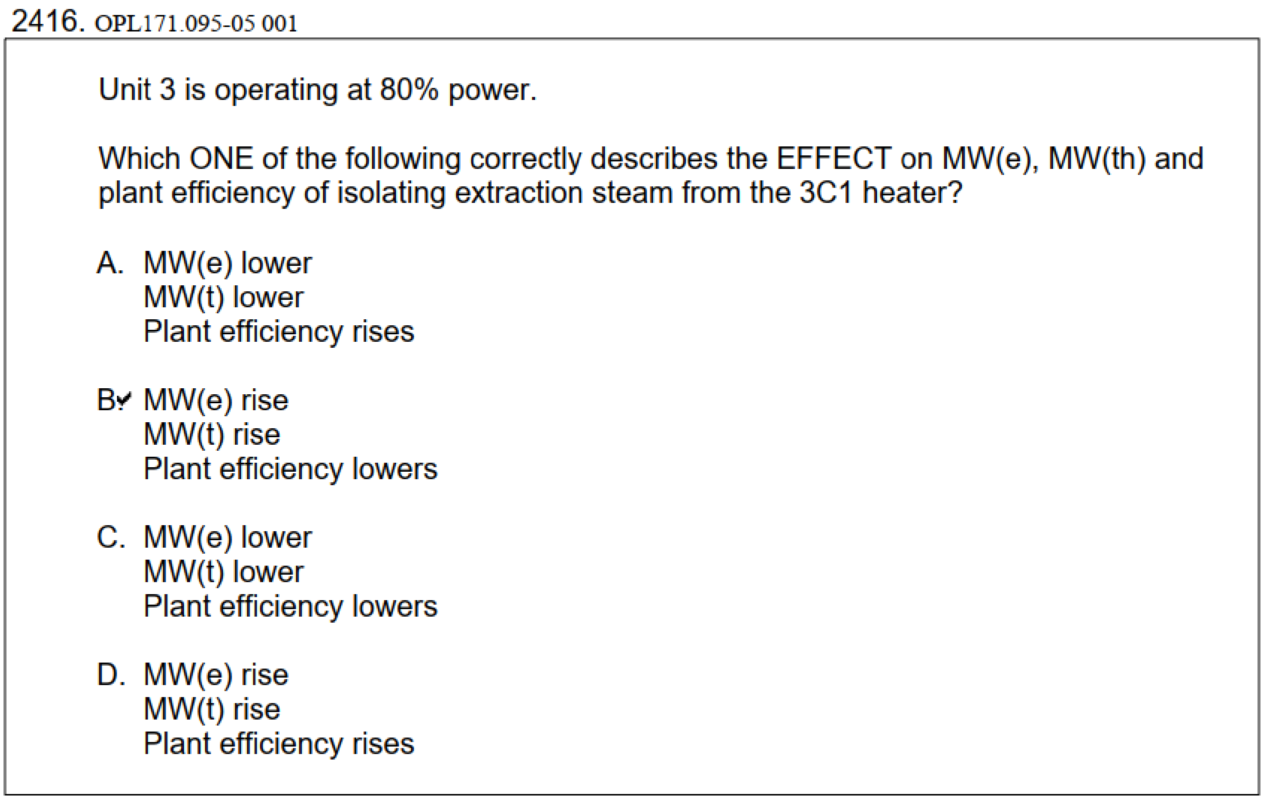
Question Source:	Bank #	OPL171.095-05 001 #2416	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

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

Comments:

Copy of Bank Question:



2416. OPL171.095-05 001

Unit 3 is operating at 80% power.

Which ONE of the following correctly describes the EFFECT on MW(e), MW(th) and plant efficiency of isolating extraction steam from the 3C1 heater?

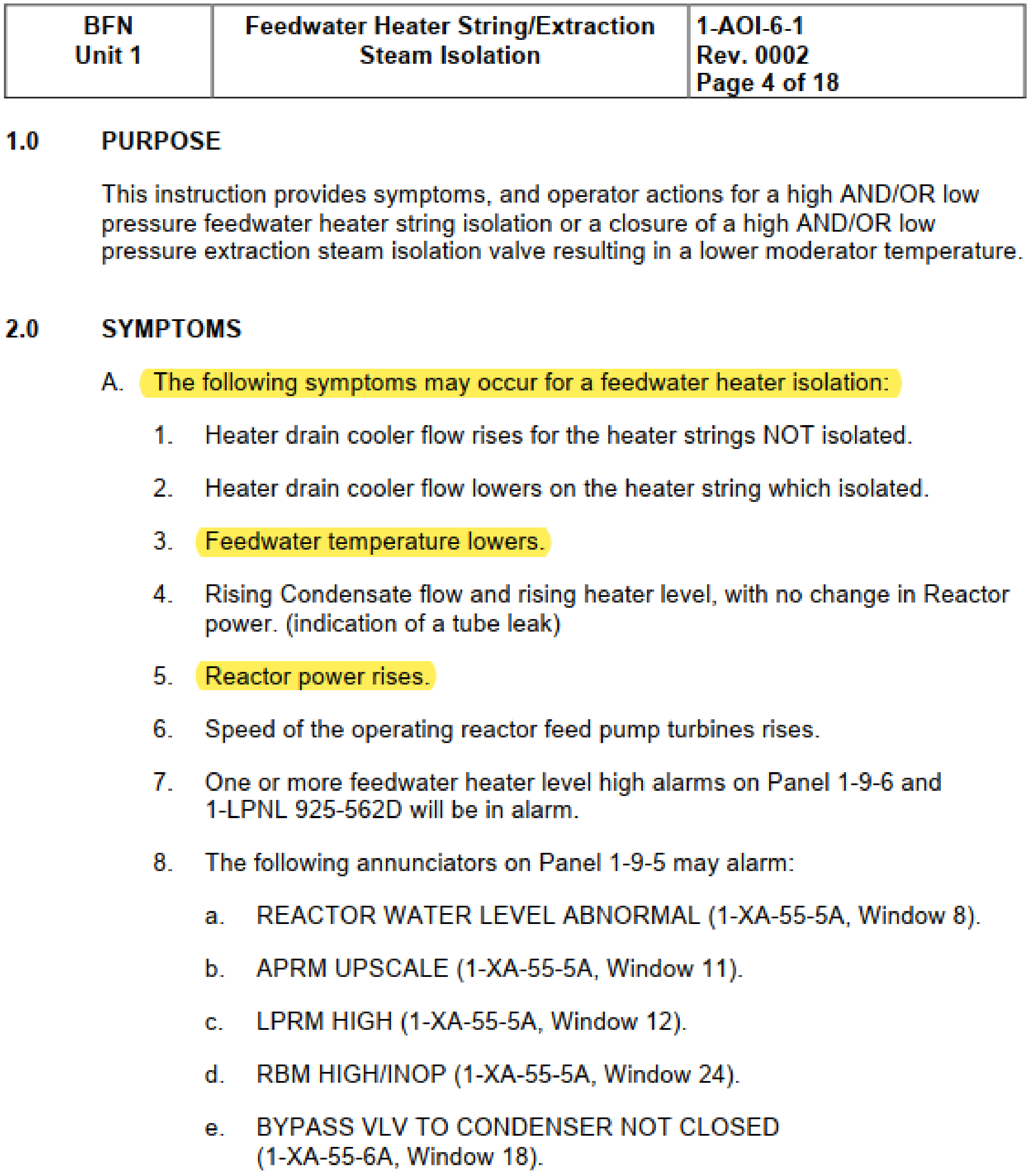
A. MW(e) lower
MW(t) lower
Plant efficiency rises

B✓ MW(e) rise
MW(t) rise
Plant efficiency lowers

C. MW(e) lower
MW(t) lower
Plant efficiency lowers

D. MW(e) rise
MW(t) rise
Plant efficiency rises

Excerpts from 1-AOI-6-1:



BFN Unit 1	**Feedwater Heater String/Extraction Steam Isolation**	**1-AOI-6-1 Rev. 0002 Page 4 of 18**

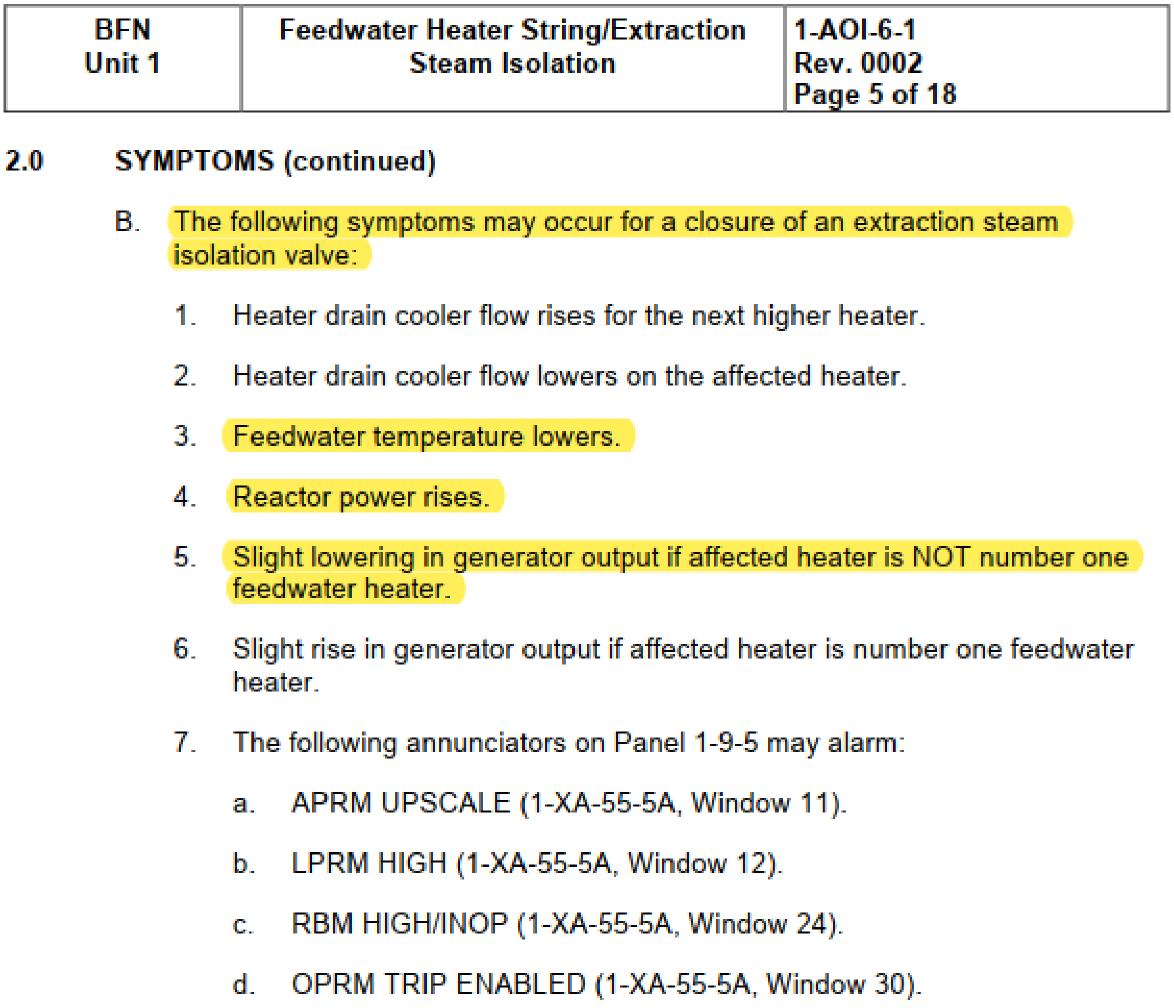
1.0 PURPOSE

This instruction provides symptoms, and operator actions for a high AND/OR low pressure feedwater heater string isolation or a closure of a high AND/OR low pressure extraction steam isolation valve resulting in a lower moderator temperature.

2.0 SYMPTOMS

A. The following symptoms may occur for a feedwater heater isolation:

1. Heater drain cooler flow rises for the heater strings NOT isolated.
2. Heater drain cooler flow lowers on the heater string which isolated.
3. Feedwater temperature lowers.
4. Rising Condensate flow and rising heater level, with no change in Reactor power. (indication of a tube leak)
5. Reactor power rises.
6. Speed of the operating reactor feed pump turbines rises.
7. One or more feedwater heater level high alarms on Panel 1-9-6 and 1-LPNL 925-562D will be in alarm.
8. The following annunciators on Panel 1-9-5 may alarm:
   a. REACTOR WATER LEVEL ABNORMAL (1-XA-55-5A, Window 8).
   b. APRM UPSCALE (1-XA-55-5A, Window 11).
   c. LPRM HIGH (1-XA-55-5A, Window 12).
   d. RBM HIGH/INOP (1-XA-55-5A, Window 24).
   e. BYPASS VLV TO CONDENSER NOT CLOSED (1-XA-55-6A, Window 18).

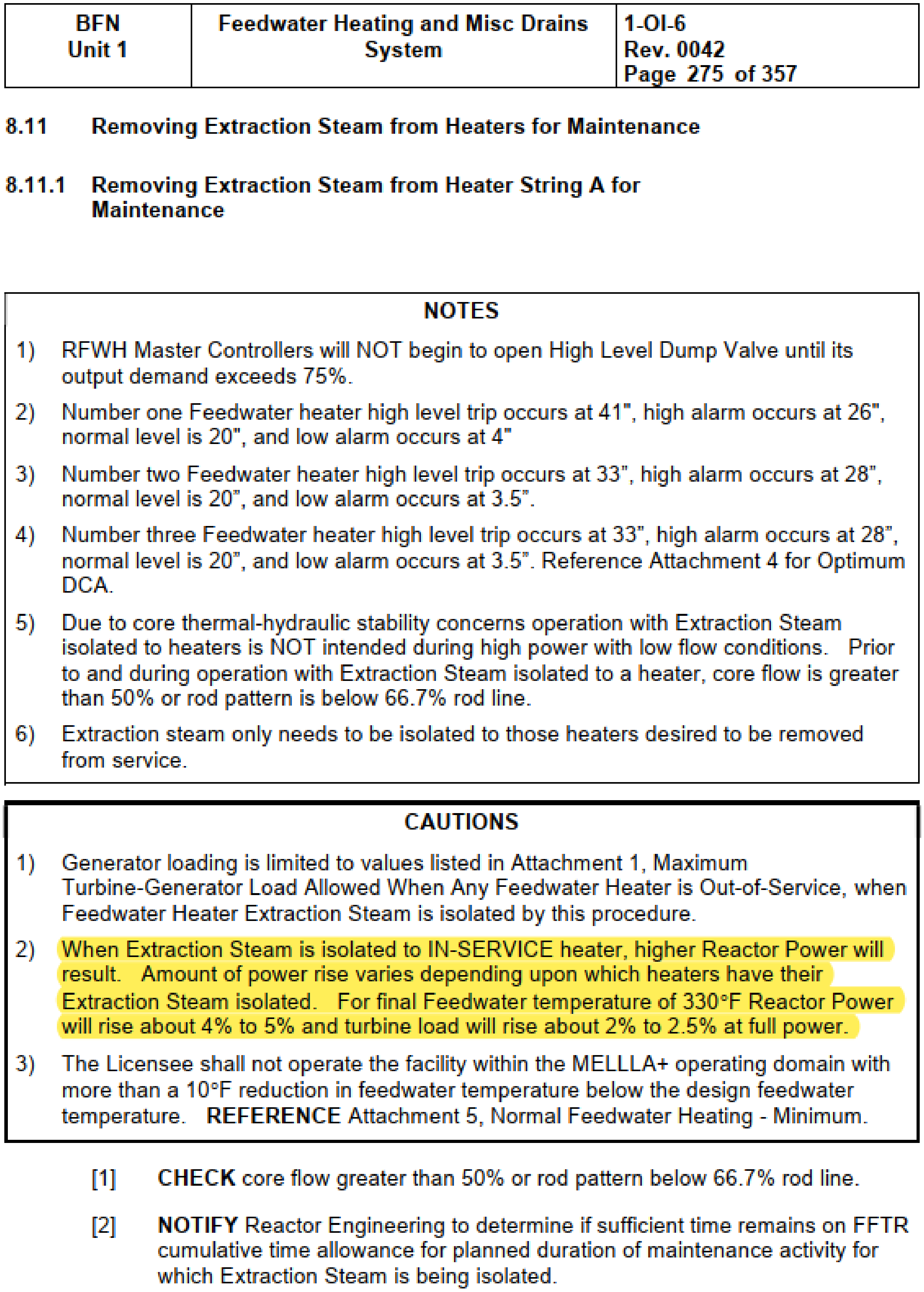


2.0 SYMPTOMS (continued)

B. The following symptoms may occur for a closure of an extraction steam isolation valve:

1. Heater drain cooler flow rises for the next higher heater.
2. Heater drain cooler flow lowers on the affected heater.
3. Feedwater temperature lowers.
4. Reactor power rises.
5. Slight lowering in generator output if affected heater is NOT number one feedwater heater.
6. Slight rise in generator output if affected heater is number one feedwater heater.
7. The following annunciators on Panel 1-9-5 may alarm:
   a. APRM UPSCALE (1-XA-55-5A, Window 11).
   b. LPRM HIGH (1-XA-55-5A, Window 12).
   c. RBM HIGH/INOP (1-XA-55-5A, Window 24).
   d. OPRM TRIP ENABLED (1-XA-55-5A, Window 30).

Excerpt from 1-OI-6:



BFN Unit 1	Feedwater Heating and Misc Drains System	1-OI-6 Rev. 0042 Page 275 of 357

8.11 Removing Extraction Steam from Heaters for Maintenance

8.11.1 Removing Extraction Steam from Heater String A for Maintenance

NOTES

1) RFWH Master Controllers will NOT begin to open High Level Dump Valve until its output demand exceeds 75%.

2) Number one Feedwater heater high level trip occurs at 41", high alarm occurs at 26", normal level is 20", and low alarm occurs at 4"

3) Number two Feedwater heater high level trip occurs at 33", high alarm occurs at 28", normal level is 20", and low alarm occurs at 3.5".

4) Number three Feedwater heater high level trip occurs at 33", high alarm occurs at 28", normal level is 20", and low alarm occurs at 3.5". Reference Attachment 4 for Optimum DCA.

5) Due to core thermal-hydraulic stability concerns operation with Extraction Steam isolated to heaters is NOT intended during high power with low flow conditions. Prior to and during operation with Extraction Steam isolated to a heater, core flow is greater than 50% or rod pattern is below 66.7% rod line.

6) Extraction steam only needs to be isolated to those heaters desired to be removed from service.

CAUTIONS

1) Generator loading is limited to values listed in Attachment 1, Maximum Turbine-Generator Load Allowed When Any Feedwater Heater is Out-of-Service, when Feedwater Heater Extraction Steam is isolated by this procedure.

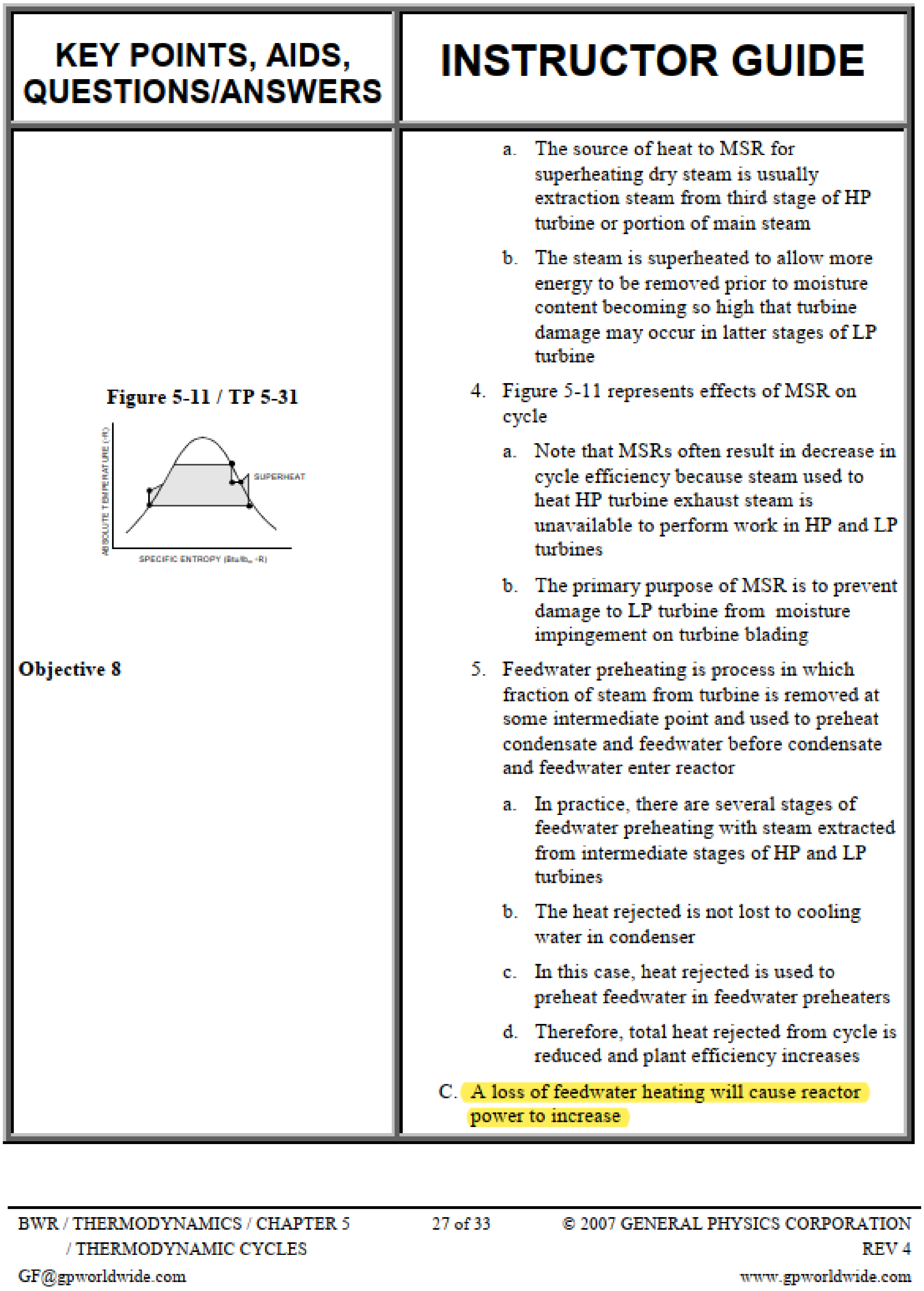
2) When Extraction Steam is isolated to IN-SERVICE heater, higher Reactor Power will result. Amount of power rise varies depending upon which heaters have their Extraction Steam isolated. For final Feedwater temperature of 330°F Reactor Power will rise about 4% to 5% and turbine load will rise about 2% to 2.5% at full power.

3) The Licensee shall not operate the facility within the MELLLA+ operating domain with more than a 10°F reduction in feedwater temperature below the design feedwater temperature. **REFERENCE** Attachment 5, Normal Feedwater Heating - Minimum.

[1] **CHECK** core flow greater than 50% or rod pattern below 66.7% rod line.

[2] **NOTIFY** Reactor Engineering to determine if sufficient time remains on FFTR cumulative time allowance for planned duration of maintenance activity for which Extraction Steam is being isolated.

Excerpts from BWR GFES Thermodynamics Cycles, Ch. 5:



KEY POINTS, AIDS, QUESTIONS/ANSWERS

INSTRUCTOR GUIDE

   a. The source of heat to MSR for superheating dry steam is usually extraction steam from third stage of HP turbine or portion of main steam

   b. The steam is superheated to allow more energy to be removed prior to moisture content becoming so high that turbine damage may occur in latter stages of LP turbine

Figure 5-11 / TP 5-31

4. Figure 5-11 represents effects of MSR on cycle

   a. Note that MSRs often result in decrease in cycle efficiency because steam used to heat HP turbine exhaust steam is unavailable to perform work in HP and LP turbines

   b. The primary purpose of MSR is to prevent damage to LP turbine from moisture impingement on turbine blading

Objective 8

5. Feedwater preheating is process in which fraction of steam from turbine is removed at some intermediate point and used to preheat condensate and feedwater before condensate and feedwater enter reactor

   a. In practice, there are several stages of feedwater preheating with steam extracted from intermediate stages of HP and LP turbines

   b. The heat rejected is not lost to cooling water in condenser

   c. In this case, heat rejected is used to preheat feedwater in feedwater preheaters

   d. Therefore, total heat rejected from cycle is reduced and plant efficiency increases

C. A loss of feedwater heating will cause reactor power to increase

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE

1. This occurs because steam that was formerly used to preheat feedwater is not preheating feedwater and now passes through turbine to condenser
2. The feedwater that is not warmed by steam enters reactor at lower temperature
3. The colder feedwater temperature causes reactor power to increase
4. Additionally, new plant efficiency will be lower than original plant efficiency because additional energy from fission process must be used to raise cooler feedwater to operating temperatures

Objective 9

D. Plant Parameter Changes Affecting Efficiency

1. Plant operators must be aware of how changing plant parameters will affect plant efficiency
    a. Any change in plant efficiency will cause change in power output from reactor
2. Plant parameter changes affecting plant thermodynamic efficiency include:
    a. Increasing condenser vacuum - lower pressure in main condenser, greater work done by turbine and, hence, greater overall plant efficiency
    b. Increasing circulating water system flow rate - higher circulating water system flow rate reduces condenser temperature, lowering pressure at exhaust of turbine
        1) Lower pressure at exhaust of turbine increases plant efficiency
    c. Lowering circulating water system inlet temperature - lower circulating water system inlet temperature increases heat transfer rate in condenser, resulting in lower condenser temperature and pressure

Examination Outline Cross-reference:

Equipment Control
G2.2.35 (10CFR 55.41.10)
Ability to determine Technical Specification Mode of Operation.

Level	RO	SRO
Tier #	3	------
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 3?

A. A Reactor SCRAM has just occurred. The MODE SWITCH is in SHUTDOWN. Moderator Temperature is 480 °F. The MSIVs are closed.

B. The Reactor is shutdown. The MODE SWITCH is in START&HOT STBY. Moderator Temperature is 180 °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 135 °F. All Control Rods are fully inserted.

D. The Reactor is shutdown. The MODE SWITCH is in REFUEL. Moderator Temperature is 140 °F. All Reactor Vessel Head Closure Bolts are **NOT** fully tensioned.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with Tech Spec 1.0, Table 1.1-1 MODES, the given plant conditions meet the Tech Spec definition of MODE 3 Hot Shutdown with the Reactor MODE Switch in the Shutdown position and moderator temperature is greater than 212 °F.

B INCORRECT: Incorrect but plausible in that with 5 different MODEs, any combination of MODE Switch positions and plant conditions is plausible with the numerical position not stated. The stated plant conditions will result in the Unit being in MODE 2.

C INCORRECT: *(See B and attached)* Incorrect but plausible in that the stated conditions result in the Unit being in MODE 4.

D INCORRECT: *(See B and attached)* Incorrect but plausible in that the stated conditions result in the Unit being in MODE 5.

RO Level Justification: Tests the candidate's ability to determine the plant MODE of operation 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**

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

Learning Objective: OPL171.087 Obj. 8 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN NRC 2104 #67	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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:

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.

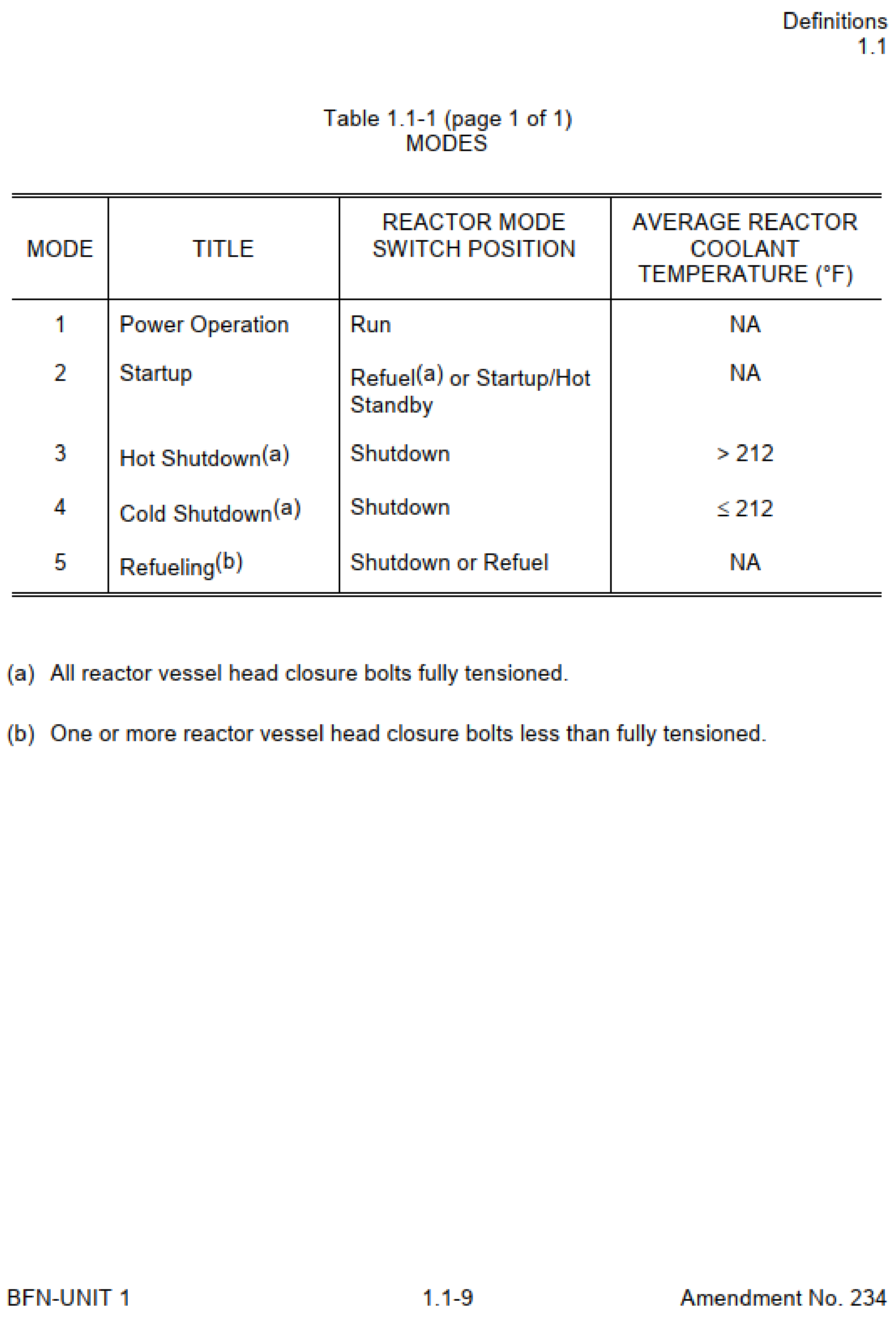
B. The Reactor is shutdown. The MODE SWITCH is in START&HOT STBY. Moderator Temperature is 180º F. All Control Rods are fully inserted.

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**

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.

BFN-UNIT 1 1.1-9 Amendment No. 234

Examination Outline Cross-reference:

293007 Thermodynamics – Heat Transfer

K1.03 (10CFR 55.41.14)

Heat Transfer, Explain the manner in which fluid films affect heat transfer

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	293007K1.03	
Importance Rating	2.8	------

Proposed Question: **# 68**

Unit 1 is operating at 100% RTP.

In an operating cooling water system, a higher stagnant fluid film thickness ___**(1)**___ the rate of heat transfer.

Stable film boiling is ___**(2)**___ efficient than nucleate boiling.

A. (1) raises
(2) more

B. (1) raises
(2) less

C. (1) lowers
(2) more

D. (1) lowers
(2) less

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that both the fluid film thickness and the fluid film conductivity directly affect the rate of heat transfer. Given the equation for the rate of heat transfer $\dot{Q}=hA\Delta T$, where h= $k_f/x_{f.,}$ if the candidate confuses the convection heat transfer coefficient (h) as x_f / k_f where x_f is the thickness of the fluid film and k_f is the thermal conductivity of the fluid film, then as fluid film thickness rises, heat transfer rate would rise. Second part is incorrect but plausible in that stable film boiling occurs when a steam blanket covers the heat transfer surface, which occurs after the departure from nucleate boiling. If the candidate confuses the regions on the Boiling Point Curve, then stable film boiling could be concluded to be more efficient than nucleate boiling.

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)* Given the equation for the rate of heat transfer $\dot{Q}=hA\Delta T$, where h= k_f/x_f., x_f is the thickness of the fluid film and k_f is the thermal conductivity of the fluid film, as fluid film thickness rises, heat transfer rate lowers. For second part, stable film boiling (also known as dryout) results in radiative heat transfer and may result in container failure. When bubbles form on a surface and collapse into the liquid, they disturb the stagnant liquid layer and transfer latent heat of vaporization to the liquid. This is known as nucleate boiling and is the most efficient form of heat transfer.

RO Level Justification: Tests the candidate's knowledge of Generic Fundamentals, specifically the various methods of heat transfer and how fluid film affects the heat transfer rate. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): BWR GFES Thermodynamics, Ch. 8, Rev. 4 (Attach if not previously provided)
BWR GFES Thermodynamics, Ch. 7, Rev. 4

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: GFES Thermodynamics, Ch. 7, Obj. 6 (As available)

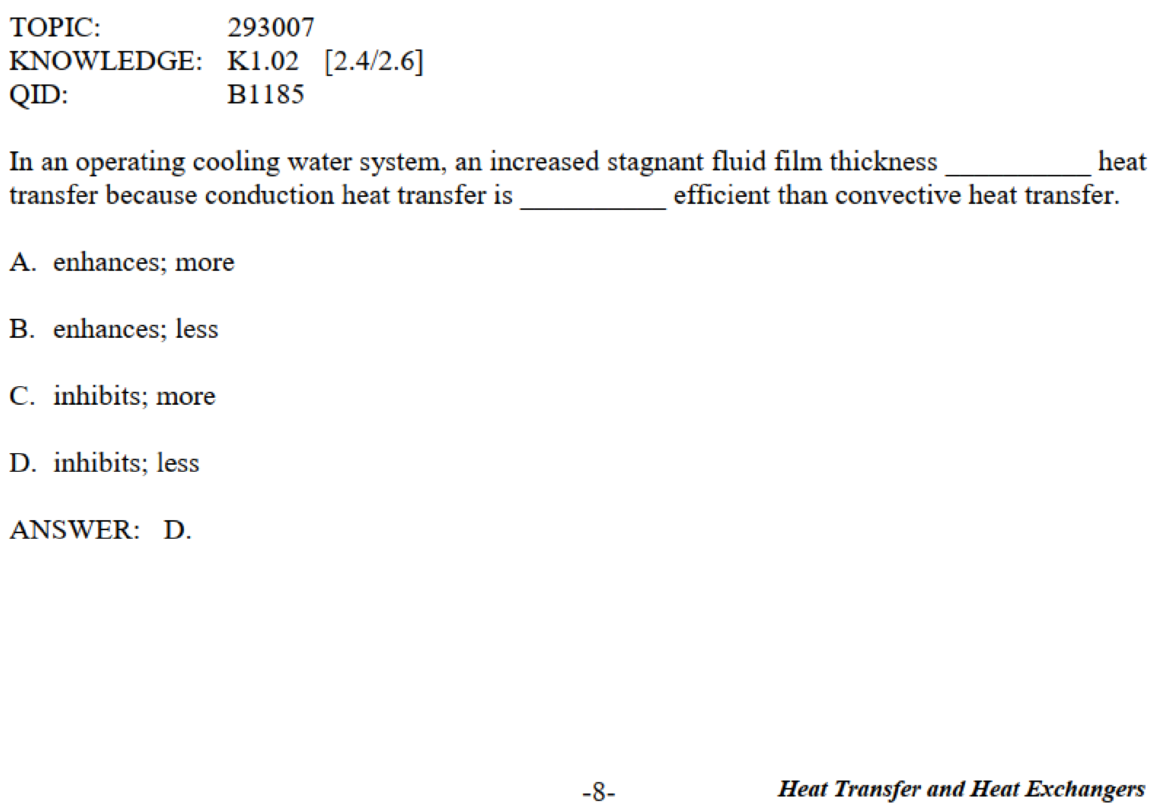
Question Source:

	Bank #	GFES Thermodynamics QID B1185	
	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

Copy of Bank Question:



TOPIC: 293007
KNOWLEDGE: K1.02 [2.4/2.6]
QID: B1185

In an operating cooling water system, an increased stagnant fluid film thickness ___________ heat transfer because conduction heat transfer is ___________ efficient than convective heat transfer.

A. enhances; more

B. enhances; less

C. inhibits; more

D. inhibits; less

ANSWER: D.

Excerpts from BWR GFES Thermodynamics, Ch. 7:

KEY POINTS, AIDS, QUESTIONS/ANSWERS

INSTRUCTOR GUIDE

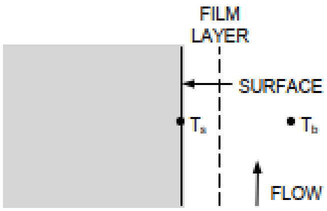
g. This is because no single property of heat transfer medium, such as thermal conductivity, can be defined to describe mechanism

h. Heat transfer by convection varies from situation to situation

i. It is frequently coupled with mode of fluid flow

j. In practice, heat transfer by convection is empirically treated.

Figure 7-3 / TP 7-24



2. Figure 7-3 shows generalized diagram of heat transfer by convection

   a. It involves transfer of heat between surface at temperature (T_s), and fluid at temperature (T_b), referred to as bulk temperature of fluid

   b. The exact definition of temperature of bulk fluid (T_b) varies depending on situation

   c. For flow adjacent to hot or cold surface, T_b is temperature far from surface

   d. For boiling or condensation, T_b is saturation temperature.

3. The basic relationship for heat transfer by convection has same form as that for heat transfer by conduction.

Equation 7-13 / TP 7-25

$$\dot{Q} = hA\Delta T$$

Where:

$\dot{Q}$ = heat transfer rate (Btu/hr)

h = convection heat transfer coefficient (Btu/hr ft^2 °F)

A = cross sectional area (ft^2)

ΔT = temperature difference (°F)

KEY POINTS, AIDS, QUESTIONS/ANSWERS

Equation 7-14 / TP 7-26

$$h = \frac{k_f}{x_f}$$

Where:

h = convection heat transfer coefficient (Btu/hr ft^2 °F)

k_f = stagnant film thermal conductivity (Btu/hr ft °F)

x_f = thickness of stagnant film layer (ft)

INSTRUCTOR GUIDE

4. The convection heat transfer coefficient (h) is derived in similar fashion as conduction heat transfer coefficient
   a. Generally, relationship between stagnant film thermal conductivity (k_f) and thickness of stagnant film layer (x_f) determines convection heat transfer coefficient.
5. This equation, however, is limited due to inability to predict convection heat transfer coefficient (h)
   a. The convection heat transfer coefficient widely varies with:
      1) Temperature
      2) Temperature drop through convection medium
      3) Flow rate
      4) Flow direction
      5) Surface irregularities
      6) Boiling
      7) Heat flux
6. Thus, it is often necessary to solve convection heat transfer processes empirically.
7. There are two types of convection heat transfer: natural convection and forced convection

KEY POINTS, AIDS, QUESTIONS/ANSWERS

$$Q'' = \frac{k_f}{x_f'}(T_s - T_{bulk})$$

Where:

Q'' = heat flux (Btu/hr ft^2)

h = convection heat transfer coefficient (Btu/hr ft^2 °F)

T_s = temperature at surface (°F)

T_{bulk} = bulk fluid temperature (°F)

k_f = stagnant film thermal conductivity (Btu/hr ft °F)

x_f = thickness of stagnant film layer (ft)

INSTRUCTOR GUIDE

7. Although this is oversimplified picture, it illustrates that as boundary layer becomes thinner, convection heat transfer coefficient becomes larger
8. Several factors affect heat transfer coefficient for fluid film
9. Most of these affect film thickness
10. Film thickness in conjunction with fluid thermal conductivity determines heat transfer coefficient
    a. Fluid velocity - The greater velocity of fluid stream, thinner fluid film.
        1) This causes heat transfer coefficient to increase
    b. Fluid thermal conductivity - An increase in thermal conductivity results in increase of heat transfer coefficient
    c. Fluid viscosity - The smaller viscosity, thinner film and larger heat transfer coefficient.

KEY POINTS, AIDS, QUESTIONS/ANSWERS	**INSTRUCTOR GUIDE**

d. Heat flux/nucleate boiling - If heat flux is sufficient to cause nucleate boiling, film thickness will effectively decrease (due to turbulence caused by bubbles)

1) This will increase heat transfer coefficient

11. To transfer large quantities of heat rapidly, one attempts to reduce boundary layer thickness as much as possible

a. This can be accomplished by increasing velocity and/or turbulence of fluid.

VIII. HEAT EXCHANGERS

A. Types of Heat Exchangers

1. Heat exchangers are devices that remove heat from, or add heat to, working fluid

a. They can be divided into two major operational categories, single-phase and two-phase

b. In single-phase heat exchangers, both cooling (heating) fluid and cooled (heated) fluid remain in their initial gaseous or liquid state (i.e., no phase change occurs)

c. Examples of single-phase heat exchanger are liquid-cooled oil cooler or air-cooled automobile radiator

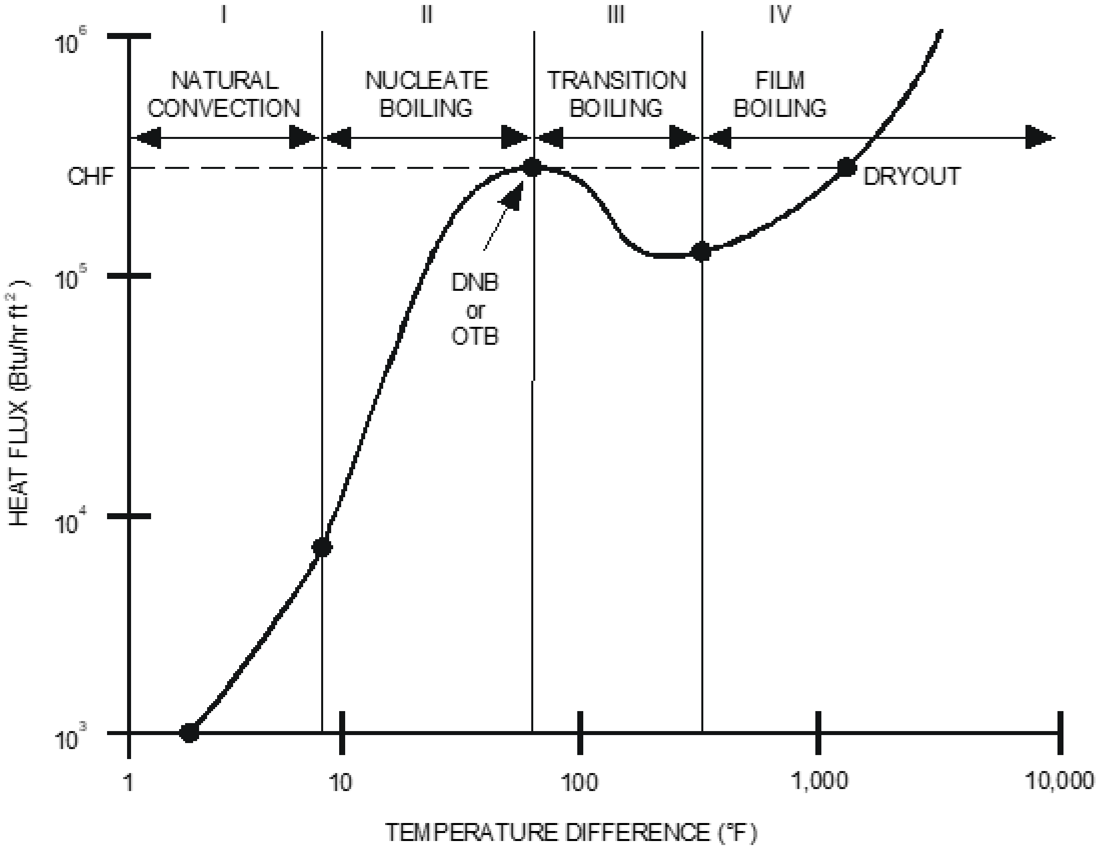
d. In two-phase heat exchangers, either cooling (heating) fluid or cooled (heated) fluid changes phases

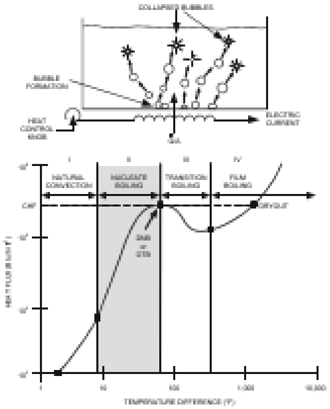
e. One important two-phase heat exchanger in BWR nuclear power plant is main condenser

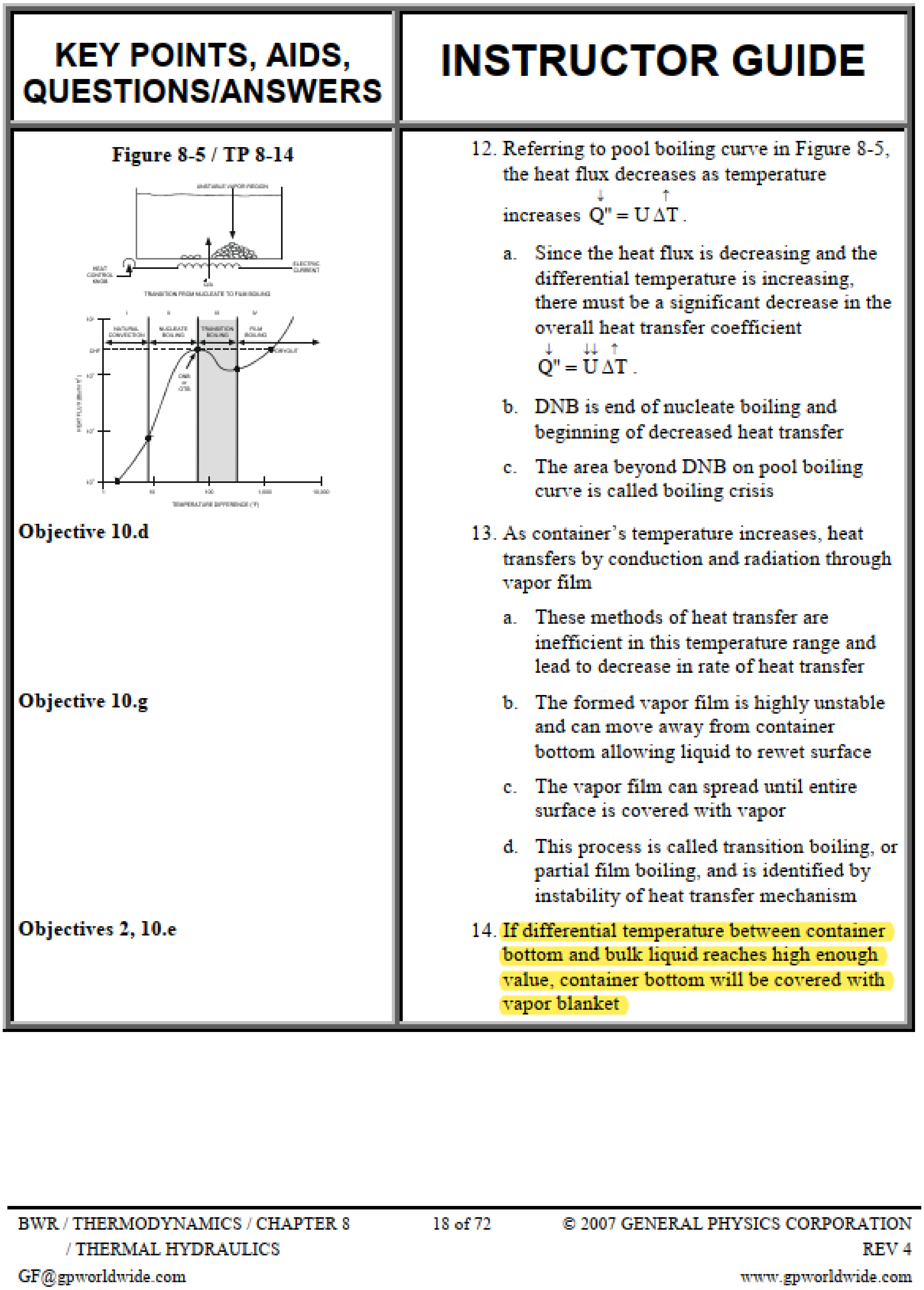
f. The gland exhaust condenser, air ejector condenser, and feedwater heaters are other two-phase heat exchangers

2. Single-phase heat exchangers and two-phase heat exchangers are similar in design

Excerpts from BWR GFES Thermodynamics, Ch. 8: The Boiling Point Curve/Regions



KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
	b. Also, each bubble carries energy of latent heat of vaporization absorbed during phase change from liquid to gas c. The enthalpy of bubbles is much greater than enthalpy of liquid heated by natural convection d. This process is called nucleate boiling because of mechanism of bubble formation
Objective 9 **Figure 8-3 / TP 8-12** 	8. Nucleate boiling allows large amount of heat to be transferred without extremely high surface temperatures a. The increased slope of pool boiling curve in Region II, highlighted in Figure 8-3, indicates greater heat transfer coefficient
Objective 11	9. The nucleate boiling discussed in previous example is called subcooled nucleate boiling because bulk temperature is below saturation temperature a. Formed bubbles collapse when they move into bulk liquid b. As more heat is applied to container, temperature of bulk liquid rises to saturation temperature



KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Figure 8-5 / TP 8-14	12. Referring to pool boiling curve in Figure 8-5, the heat flux decreases as temperature increases $\overset{\downarrow}{Q''} = U\,\overset{\uparrow}{\Delta T}$. a. Since the heat flux is decreasing and the differential temperature is increasing, there must be a significant decrease in the overall heat transfer coefficient $\overset{\downarrow}{Q''} = \overset{\downarrow\downarrow}{U}\,\overset{\uparrow}{\Delta T}$. b. DNB is end of nucleate boiling and beginning of decreased heat transfer c. The area beyond DNB on pool boiling curve is called boiling crisis
Objective 10.d	13. As container's temperature increases, heat transfers by conduction and radiation through vapor film a. These methods of heat transfer are inefficient in this temperature range and lead to decrease in rate of heat transfer
Objective 10.g	b. The formed vapor film is highly unstable and can move away from container bottom allowing liquid to rewet surface c. The vapor film can spread until entire surface is covered with vapor d. This process is called transition boiling, or partial film boiling, and is identified by instability of heat transfer mechanism
Objectives 2, 10.e	14. If differential temperature between container bottom and bulk liquid reaches high enough value, container bottom will be covered with vapor blanket

BWR / THERMODYNAMICS / CHAPTER 8 / THERMAL HYDRAULICS 18 of 72 © 2007 GENERAL PHYSICS CORPORATION REV 4
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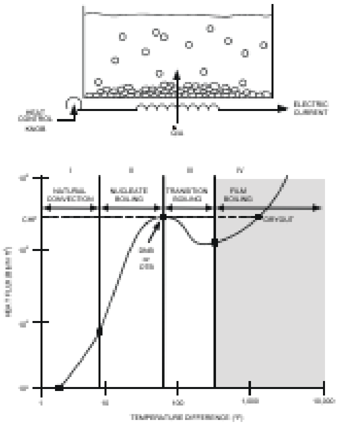
KEY POINTS, AIDS, QUESTIONS/ANSWERS

Objective 10.f

Figure 8-6 / TP 8-15



INSTRUCTOR GUIDE

a. This is represented on pool boiling curve (see Figure 8-6) as stable film boiling, or dryout

b. The major form of heat transfer in dryout is radiative heat transfer

c. If more heat is added in this region, container bottom will probably fail

d. This is known as burnout

e. Therefore, heat flux associated with failure of container is known as burnout heat flux

15. In pool boiling curve discussion, container's surface temperature was method of heat transfer control

a. For the purpose of this chapter, consider a nuclear reactor to be controlled by the heat transfer rate $(\dot{Q})$ or heat flux (Q'') instead of the clad temperature.

1) (Recall that heat flux is the heat transfer rate divided by the area.

2) If the area remains constant then any change in the heat transfer rate results in a corresponding change in the heat flux $\left(Q'' = \frac{\dot{Q}}{\overset{\leftrightarrow}{A}}\right)$.)

b. The following is a discussion of the change in heat transfer mechanisms when varying heat flux.

Examination Outline Cross-reference:	Level	RO	SRO
268000 (SF9 RW) Radwaste	Tier #	2	------
291007, K1.07 (10CFR 55.41.4)	Group #	2	------
Principles of demineralizer operation	K/A #	291007K1.07	
	Importance Rating	2.5	------

Proposed Question: **# 69**

A demineralizer works on the principle of mechanical **(1)** .

A result of proper Radwaste and Condensate demineralizer operation on water with ionic impurities is that the exiting water will always have a **(2)** conductivity.

A. (1) filtration **ONLY**
(2) lower

B. (1) filtration **ONLY**
(2) higher

C. (1) **AND** chemical filtration
(2) lower

D. (1) **AND** chemical filtration
(2) higher

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the Rad Waste and Condensate demineralizers are filled with very small resin beads and will mechanically filter the water that is passed through the resin. In addition, demineralizer differential pressure is continuously monitored, which is an indication of mechanical obstruction in the demineralizer due to mechanical filtration. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that as water passes through the resin, ions are exchanged, which may result in water conductivity being higher as ions are exchanged.

C **CORRECT:** *(See attached)* As water passes through resin, ion exchange (chemical filtration) takes place and mechanical filtration of suspended solids occurs. For second part, water with salts and impurities is passed through the resin, and the ions from the high conductivity water are exchanged resulting in a lower conductivity. Higher conductivity is a symptom of resin exhaustion.

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 Radwaste Demineralizer operation with respect to water filtration. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s):	BWR Components, Rev. 4	(Attach if not previously provided)
	OPL171.084, Rev. 10U4	
	OPL171.011A, Rev. 4	

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: BWR Demineralizers Guide, Obj. 8 (As available)

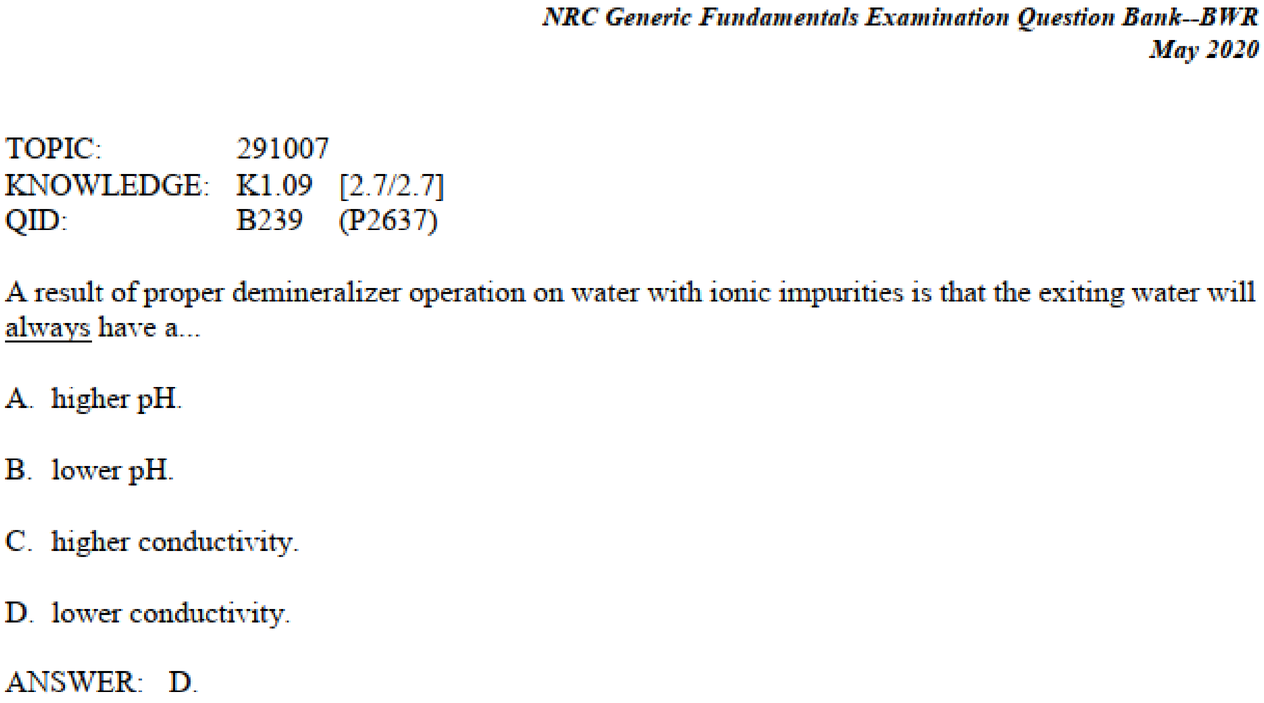
Question Source:	Bank #		
	Modified Bank #	GFE EXAM BANK B239	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	May 2020	

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:



NRC Generic Fundamentals Examination Question Bank--BWR
May 2020

TOPIC: 291007
KNOWLEDGE: K1.09 [2.7/2.7]
QID: B239 (P2637)

A result of proper demineralizer operation on water with ionic impurities is that the exiting water will always have a...

A. higher pH.

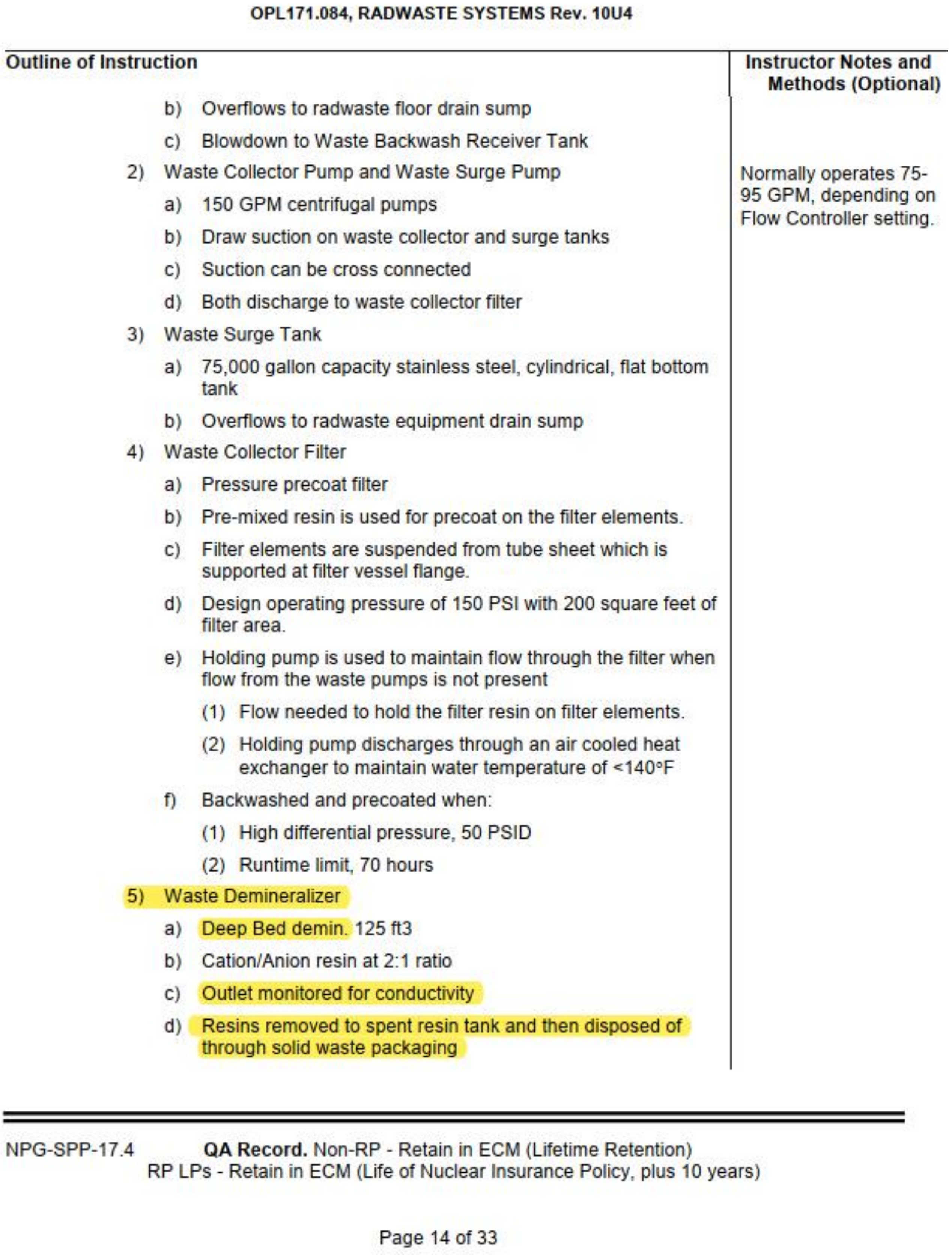
B. lower pH.

C. higher conductivity.

D. lower conductivity.

ANSWER: D.

Excerpt from OPL171.084:



OPL171.084, RADWASTE SYSTEMS Rev. 10U4

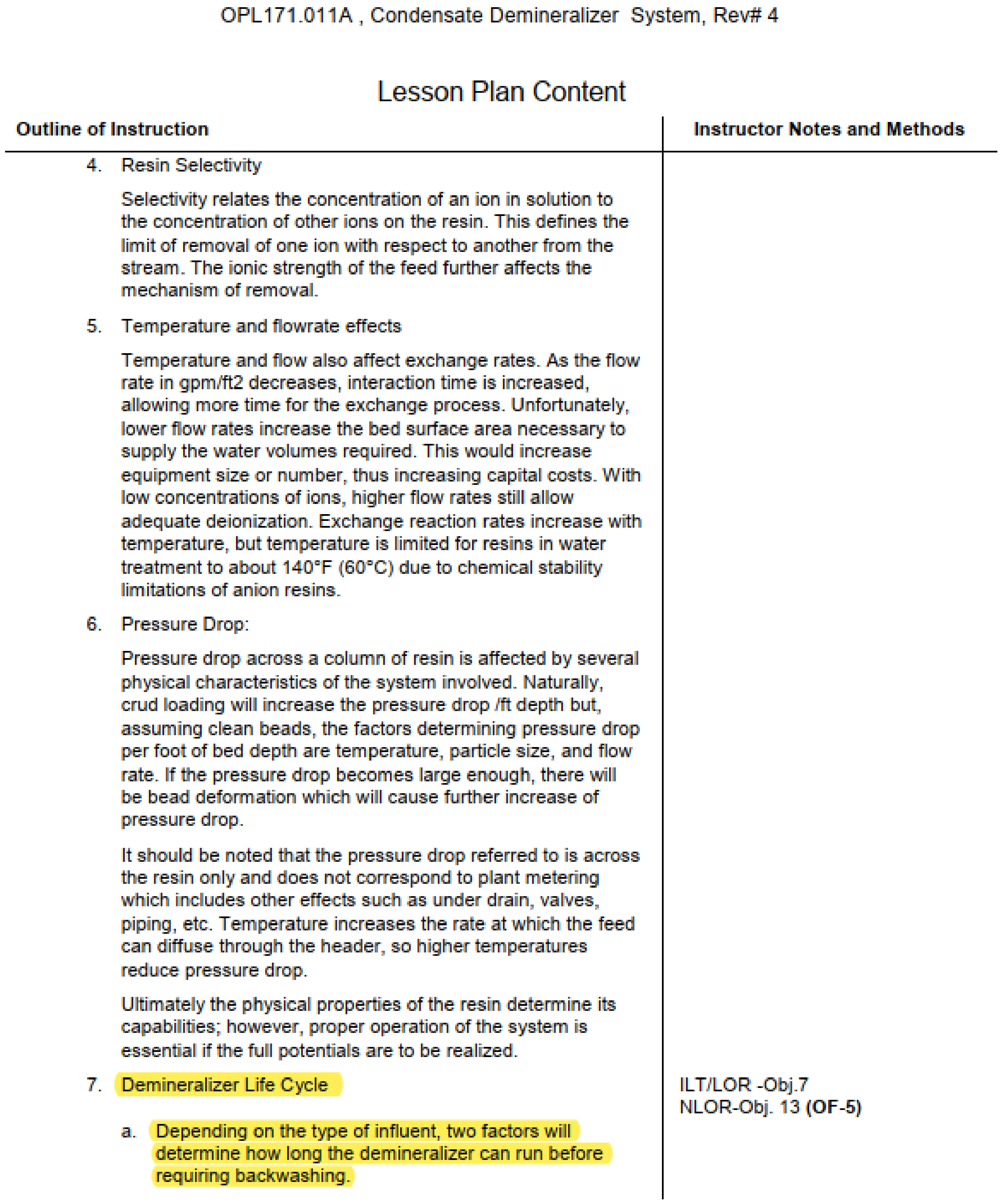
Outline of Instruction | **Instructor Notes and Methods (Optional)**

- b) Overflows to radwaste floor drain sump
- c) Blowdown to Waste Backwash Receiver Tank
- 2) Waste Collector Pump and Waste Surge Pump

  Normally operates 75-95 GPM, depending on Flow Controller setting.

  - a) 150 GPM centrifugal pumps
  - b) Draw suction on waste collector and surge tanks
  - c) Suction can be cross connected
  - d) Both discharge to waste collector filter
- 3) Waste Surge Tank
  - a) 75,000 gallon capacity stainless steel, cylindrical, flat bottom tank
  - b) Overflows to radwaste equipment drain sump
- 4) Waste Collector Filter
  - a) Pressure precoat filter
  - b) Pre-mixed resin is used for precoat on the filter elements.
  - c) Filter elements are suspended from tube sheet which is supported at filter vessel flange.
  - d) Design operating pressure of 150 PSI with 200 square feet of filter area.
  - e) Holding pump is used to maintain flow through the filter when flow from the waste pumps is not present
    - (1) Flow needed to hold the filter resin on filter elements.
    - (2) Holding pump discharges through an air cooled heat exchanger to maintain water temperature of <140°F
  - f) Backwashed and precoated when:
    - (1) High differential pressure, 50 PSID
    - (2) Runtime limit, 70 hours
- 5) Waste Demineralizer
  - a) Deep Bed demin. 125 ft3
  - b) Cation/Anion resin at 2:1 ratio
  - c) Outlet monitored for conductivity
  - d) Resins removed to spent resin tank and then disposed of through solid waste packaging

Excerpts from OPL171.011A:



OPL171.011A , Condensate Demineralizer System, Rev# 4

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
4. Resin Selectivity	
Selectivity relates the concentration of an ion in solution to the concentration of other ions on the resin. This defines the limit of removal of one ion with respect to another from the stream. The ionic strength of the feed further affects the mechanism of removal.	
5. Temperature and flowrate effects	
Temperature and flow also affect exchange rates. As the flow rate in gpm/ft2 decreases, interaction time is increased, allowing more time for the exchange process. Unfortunately, lower flow rates increase the bed surface area necessary to supply the water volumes required. This would increase equipment size or number, thus increasing capital costs. With low concentrations of ions, higher flow rates still allow adequate deionization. Exchange reaction rates increase with temperature, but temperature is limited for resins in water treatment to about 140°F (60°C) due to chemical stability limitations of anion resins.	
6. Pressure Drop:	
Pressure drop across a column of resin is affected by several physical characteristics of the system involved. Naturally, crud loading will increase the pressure drop /ft depth but, assuming clean beads, the factors determining pressure drop per foot of bed depth are temperature, particle size, and flow rate. If the pressure drop becomes large enough, there will be bead deformation which will cause further increase of pressure drop.	
It should be noted that the pressure drop referred to is across the resin only and does not correspond to plant metering which includes other effects such as under drain, valves, piping, etc. Temperature increases the rate at which the feed can diffuse through the header, so higher temperatures reduce pressure drop.	
Ultimately the physical properties of the resin determine its capabilities; however, proper operation of the system is essential if the full potentials are to be realized.	
7. Demineralizer Life Cycle	ILT/LOR -Obj.7 NLOR-Obj. 13 **(OF-5)**
a. Depending on the type of influent, two factors will determine how long the demineralizer can run before requiring backwashing.	

Lesson Plan Content

Outline of Instruction	**Instructor Notes and Methods**

b. Filtering of suspended solids will cause the resin to become clogged and increase the pressure drop across the filter. Too high of a pressure may cause effluent quality to decrease and resin breakthrough. The point at which the resin is near exhaustion or breakthrough is detected by analyzing for the most weakly held ion, which is the silicate radical (SiO_{3-2}).

c. As a unit reaches the end of its ion exchange capabilities the conductivity of the effluent will increase. This will be the limiting factor for cycle length where dissolved solids are the predominant impurity.

d. The effectiveness of ion exchange is monitored by checking some important parameter of the solution before and after it passes through the ion exchange. The parameter checked depends on the purpose of the demineralizer. Specific conductivity in mhos or mS, radioactivity in mci/ml, and concentration of impurities in ppm are common parameters. The demineralization or decontamination factor (DF) is defined as:

$$DF = \frac{\text{Inlet Conductivity, Radioactivity or Concentration}}{\text{Outlet Conductivity, Radioactivity or Concentration}}$$

e. Exhaustion of demineralizer resin is normally monitored using a concentration history curve which plots DF versus time.

8. To ensure proper demineralizer operation, the pressure drop across the demineralizer, the influent and the effluent conductivity should be monitored. Samples of the influent and effluent should be analyzed.

   a. The influent should be analyzed and conductivity checked for large increases in dissolved and suspended contaminants for the following reasons.

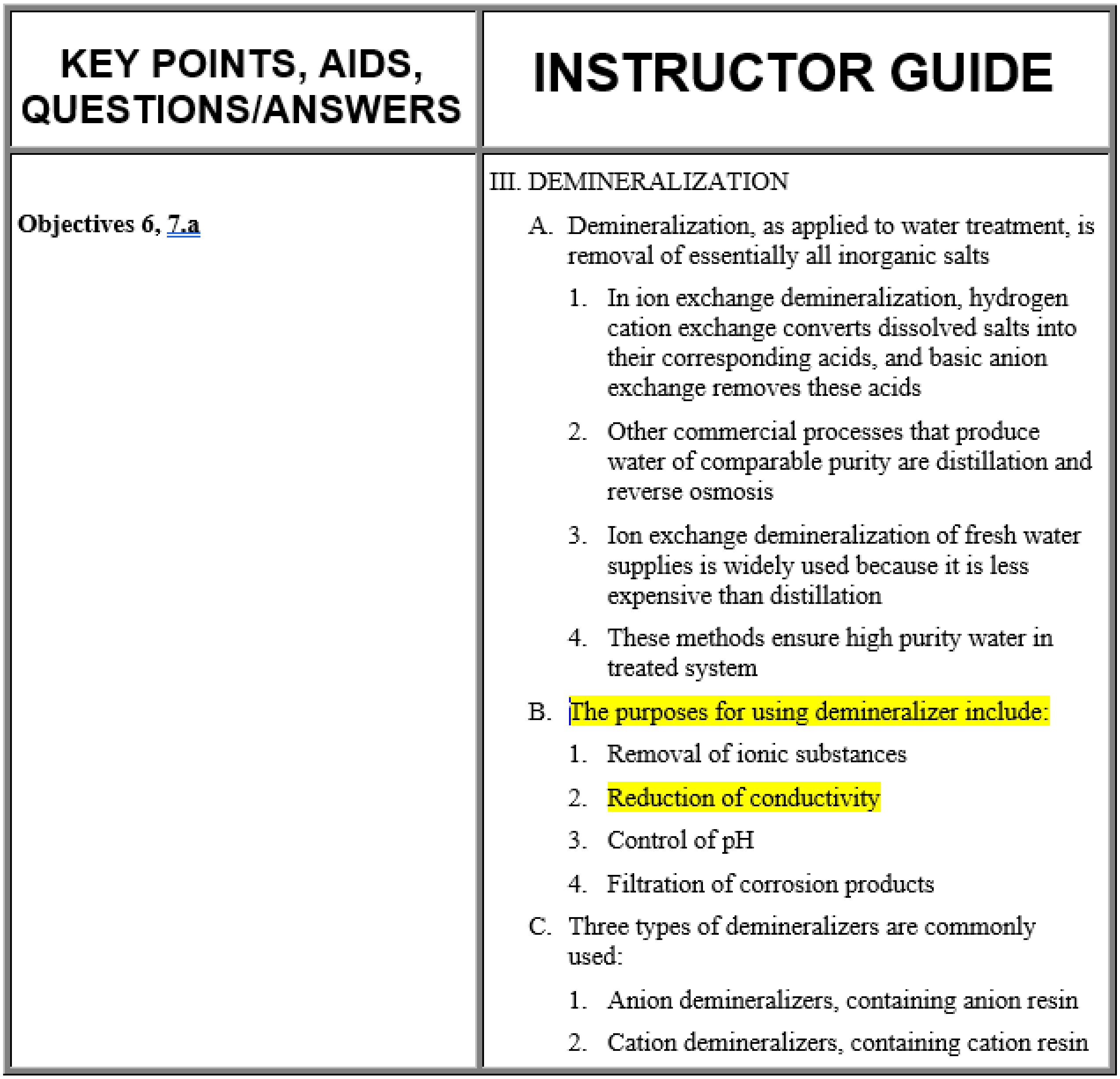
      (1) Condenser tube leakage, a new corrosion problem or a chemistry problem with makeup water may be detected.

      (2) The increased loading will shorten the cycle length of the demineralizer.

      (3) Problems can be detected and corrected much earlier than waiting for the demineralizer to become exhausted.

   b. Changes in Pressure drop may be used to indicate and identify many problems associated with demineralizers. Pressure drop should slowly increase over a cycle.

Excerpts from BWR GFES (Components) Instructor Guide:



KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Objectives 6, 7.a	III. DEMINERALIZATION A. Demineralization, as applied to water treatment, is removal of essentially all inorganic salts 1. In ion exchange demineralization, hydrogen cation exchange converts dissolved salts into their corresponding acids, and basic anion exchange removes these acids 2. Other commercial processes that produce water of comparable purity are distillation and reverse osmosis 3. Ion exchange demineralization of fresh water supplies is widely used because it is less expensive than distillation 4. These methods ensure high purity water in treated system B. The purposes for using demineralizer include: 1. Removal of ionic substances 2. Reduction of conductivity 3. Control of pH 4. Filtration of corrosion products C. Three types of demineralizers are commonly used: 1. Anion demineralizers, containing anion resin 2. Cation demineralizers, containing cation resin

KEY POINTS, AIDS, QUESTIONS/ANSWERS

Figure 4-1 / TP 4-5

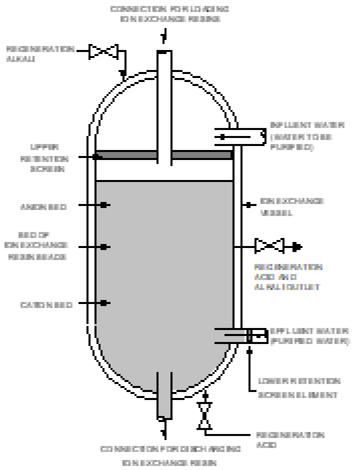
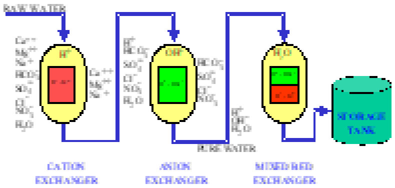


Figure 4-2 / TP 4-6



INSTRUCTOR GUIDE

3. Mixed bed demineralizers, containing both cation and anion resin (Figure 4-1)

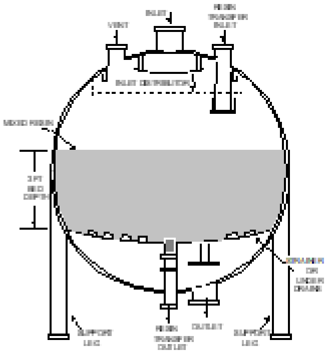
D. A demineralizer train usually uses all three ion exchangers (cation, anion, and mixed bed) consecutively (Figure 4-2)

   1. Mixed bed demineralizers are used as "polishing" demineralizers at end of water treatment chain
   2. Polishing demineralizers are used to continuously demineralize and filter condensate
   3. Because demineralizers are filled with very small resin beads, they also act as filters
   4. After raw water has been clarified and softened, it passes through demineralizer train to become pure condensate makeup water
   5. The makeup water is then stored in tanks which are called clearwells, condensate storage tanks, or, simply, makeup tanks

KEY POINTS, AIDS, QUESTIONS/ANSWERS

Objective 12, 13

Figure 4-6 / TP 4-12



INSTRUCTOR GUIDE

IV. DEMINERALIZER TYPES

A. There are two types of demineralizers: deep-bed and powdered resin filter

B. Deep-Bed Demineralizers

1. In deep-bed demineralizers, water enters at top, passes through mixed resin and strainers that do not pass resin beads, and exits at outlet (Figure 4-6)
    a. As water passes through resin mixture, ion exchange takes place and mechanical filtration of suspended solids occurs
    b. The effective length of time that batch of resin can be used is called operating cycle
2. The operating cycle is affected by two factors:
    a. The exhaustion of resin ion exchange capacity (i.e. majority of bead sites have already exchanged their original H^+ or OH^- ions), and
    b. An excessive pressure drop across resin due to buildup of suspended solids on resin beads
3. The main advantage of deep-bed demineralizer is large ionic capacity that allows time for orderly plant shutdown if significant condenser tube leak occurs
    a. Even for large amounts of depletion, only small percentage of leakage occurs
    b. Deep-bed demineralizers are normally used in plants that use seawater as their condenser cooling medium

Examination Outline Cross-reference:	Level	RO	SRO
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)	Tier #	2	------
291008, K1.06 (10CFR 55.41.7)	Group #	1	------
Interpreting one-line diagram of control circuitry.	K/A #	291008K1.06	
	Importance Rating	3.6	------

Proposed Question: **# 70**

In accordance with 0-OI-57C, 208V/120V AC Electrical System, Unit Preferred is operating in a normal lineup.

Using the 120VAC Distribution diagram, determine the correct answers below.

The attached diagram is specifically representative of ___**(1)**___ with Unit Preferred normally powered from the 480V RMOV Board, through the ___**(2)**___.

[REFERENCE PROVIDED]

A. (1) Unit 1
(2) rectifier/inverter, static switch **AND** breaker 1001

B. (1) Unit 1
(2) regulating transformer **AND** breaker 1002

C. (1) Unit 2
(2) rectifier/inverter, static switch **AND** breaker 1001

D. (1) Unit 2
(2) regulating transformer **AND** breaker 1002

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (See attached) In accordance with 0-OI-57C, 208V/120V AC Electrical System, Attachment 2 – Vital 120V AC Distribution, illustrates unique Unit differences especially between Unit 1 and Unit 2. Using a one-line diagram portion from Attachment 2, the candidate must determine that it specifically represents the Unit 1 portion of the Uninterruptable Power Supply 120V System. For second part, in accordance with 0-OI-57C, 208V/120V AC Electrical System, Unit 1 Unit Preferred normal power supply to Unit Preferred Battery Bus 1 Panel 11 is from 480V RMOV Board 1A, through the rectifier 1-INV-252-0001/inverter, static switch and breaker 1001.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that in accordance with 0-OI-57C, 208V/120V AC Electrical System, Unit 1 Unit Preferred alternate power supply to Unit Preferred Battery Bus 1 Panel 11 is through the Unit Preferred Regulating Transformer XFMR1 and breaker 1002.

C INCORRECT: First part is incorrect in that the Unit 2 Unit Preferred normal power supply to Unit Preferred Battery Bus 2 Panel 11 is through breaker 1001 from a motor-motor-generator (MMG) as is the case for Unit 3 respectively. The AC motor is powered from 480V Shutdown Board 2A and the DC motor is powered from 250V DC Battery Board 4. 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 interpret a one-line diagram of control circuity of Unit Preferred Power Supply (AC/DC) as it relates to 120VAC Distribution System. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. Given the complexity of the BFN Electrical Distribution System, the candidate must determine both the Unit differences and the specific AC and DC power supplies as it relates to specific breaker, rectifier, inverter and electrical board alignments. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-OI-57C, Rev. 135 (Attach if not previously provided)
OPL171.102, Rev. 11

Proposed references to be provided to applicants during examination: **Excerpt from 0-OI-57C, 208V/120V AC Elect Syst, Attachment 2 (Pg 1 of 1) Vital 120V AC Distribution diagram**

Learning Objective: OPL171.102 Obj 2f (As available)

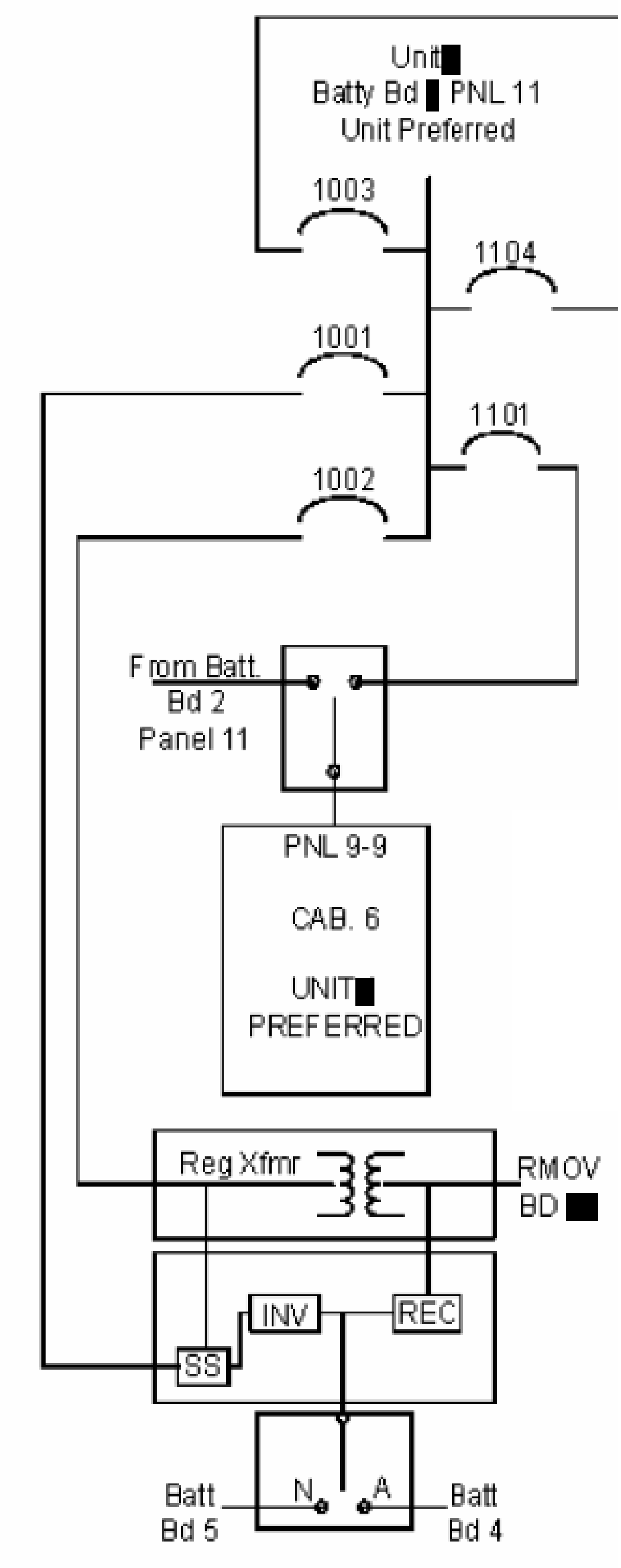
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

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

Comments:

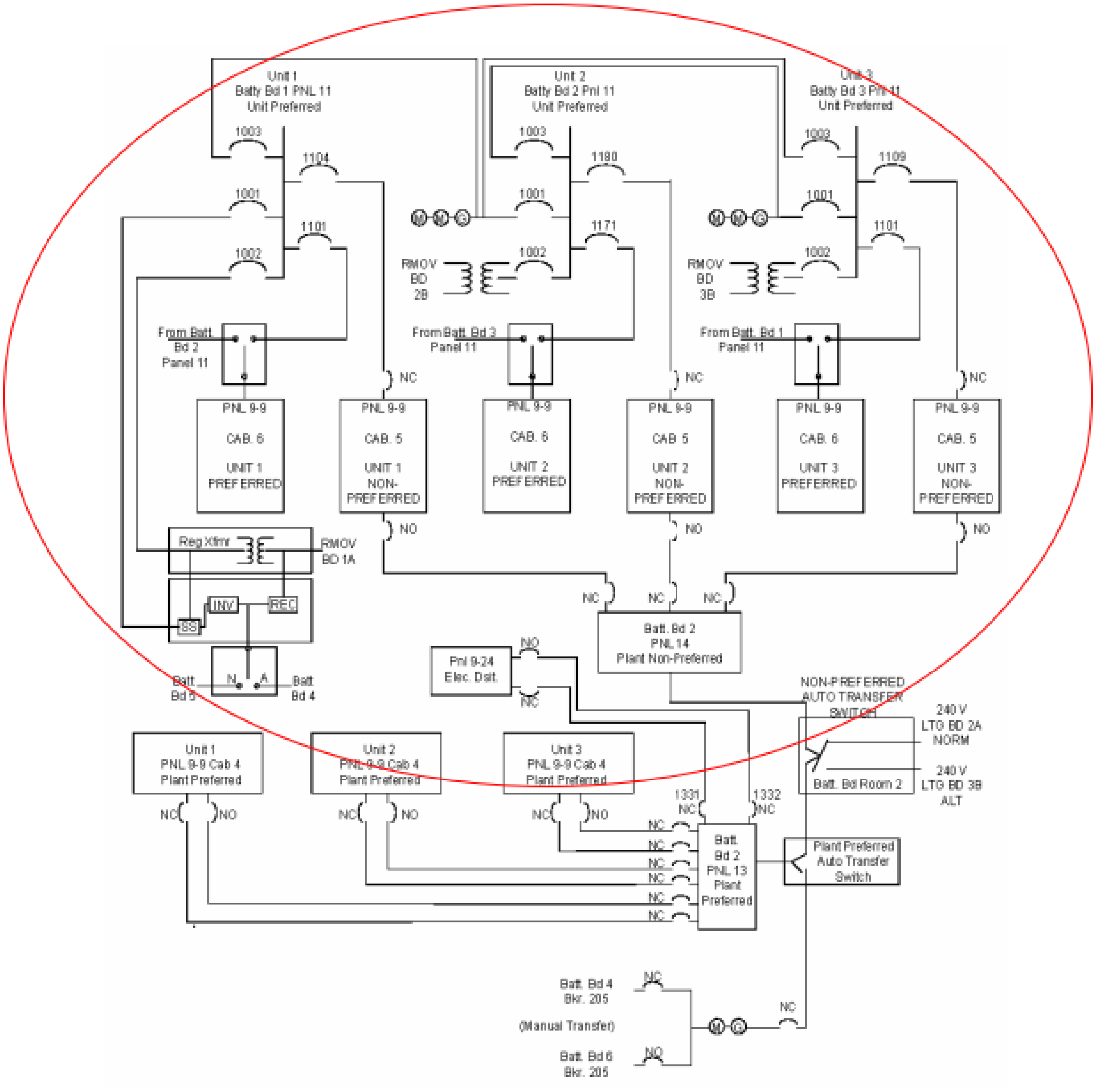
REFERENCE PROVIDED to candidate - 120 VAC Distribution diagram:

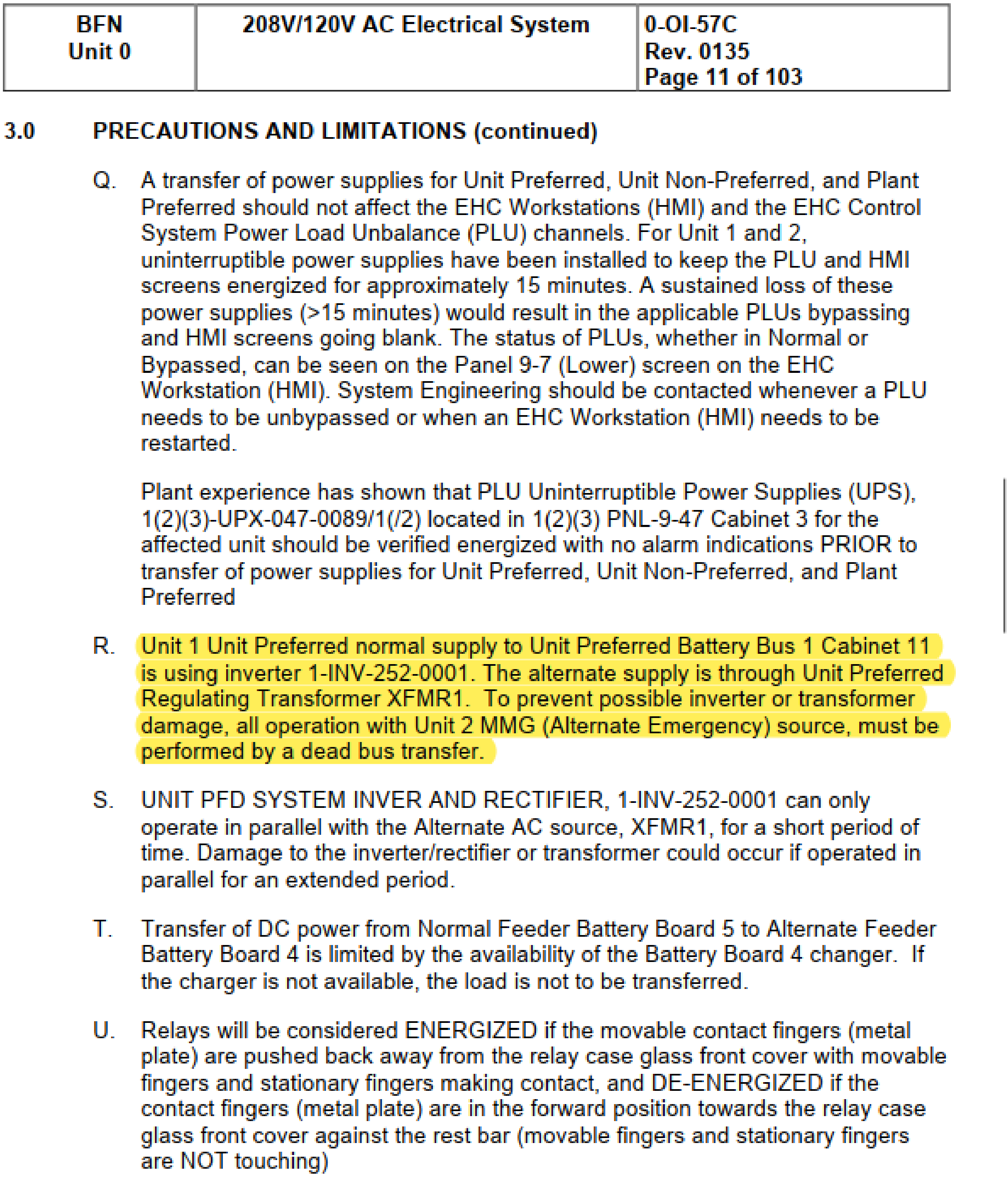


Excerpts from 0-OI-57C:

BFN Unit 0	**208V/120V AC Electrical System**	**0-OI-57C Rev. 0135 Page 100 of 103**

Attachment 2
(Page 1 of 1)
Vital 120 V AC Distribution





BFN **Unit 0**	**208V/120V AC Electrical System**	**0-OI-57C** **Rev. 0135** **Page 11 of 103**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Q. A transfer of power supplies for Unit Preferred, Unit Non-Preferred, and Plant Preferred should not affect the EHC Workstations (HMI) and the EHC Control System Power Load Unbalance (PLU) channels. For Unit 1 and 2, uninterruptible power supplies have been installed to keep the PLU and HMI screens energized for approximately 15 minutes. A sustained loss of these power supplies (>15 minutes) would result in the applicable PLUs bypassing and HMI screens going blank. The status of PLUs, whether in Normal or Bypassed, can be seen on the Panel 9-7 (Lower) screen on the EHC Workstation (HMI). System Engineering should be contacted whenever a PLU needs to be unbypassed or when an EHC Workstation (HMI) needs to be restarted.

Plant experience has shown that PLU Uninterruptible Power Supplies (UPS), 1(2)(3)-UPX-047-0089/1(/2) located in 1(2)(3) PNL-9-47 Cabinet 3 for the affected unit should be verified energized with no alarm indications PRIOR to transfer of power supplies for Unit Preferred, Unit Non-Preferred, and Plant Preferred

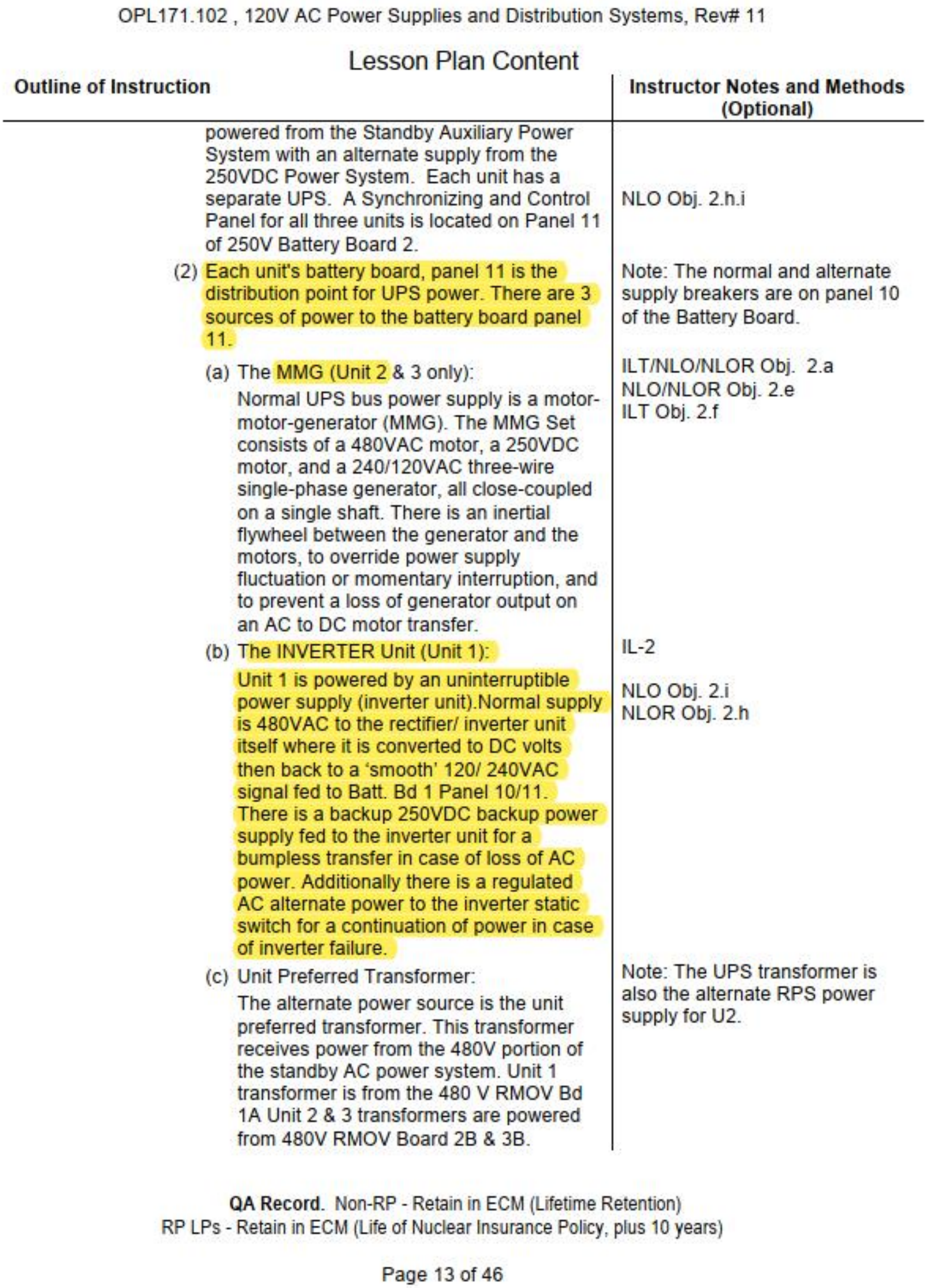
R. Unit 1 Unit Preferred normal supply to Unit Preferred Battery Bus 1 Cabinet 11 is using inverter 1-INV-252-0001. The alternate supply is through Unit Preferred Regulating Transformer XFMR1. To prevent possible inverter or transformer damage, all operation with Unit 2 MMG (Alternate Emergency) source, must be performed by a dead bus transfer.

S. UNIT PFD SYSTEM INVER AND RECTIFIER, 1-INV-252-0001 can only operate in parallel with the Alternate AC source, XFMR1, for a short period of time. Damage to the inverter/rectifier or transformer could occur if operated in parallel for an extended period.

T. Transfer of DC power from Normal Feeder Battery Board 5 to Alternate Feeder Battery Board 4 is limited by the availability of the Battery Board 4 changer. If the charger is not available, the load is not to be transferred.

U. Relays will be considered ENERGIZED if the movable contact fingers (metal plate) are pushed back away from the relay case glass front cover with movable fingers and stationary fingers making contact, and DE-ENERGIZED if the contact fingers (metal plate) are in the forward position towards the relay case glass front cover against the rest bar (movable fingers and stationary fingers are NOT touching)

Excerpts from OPL171.102 Lesson Plan:



OPL171.102 , 120V AC Power Supplies and Distribution Systems, Rev# 11

Lesson Plan Content

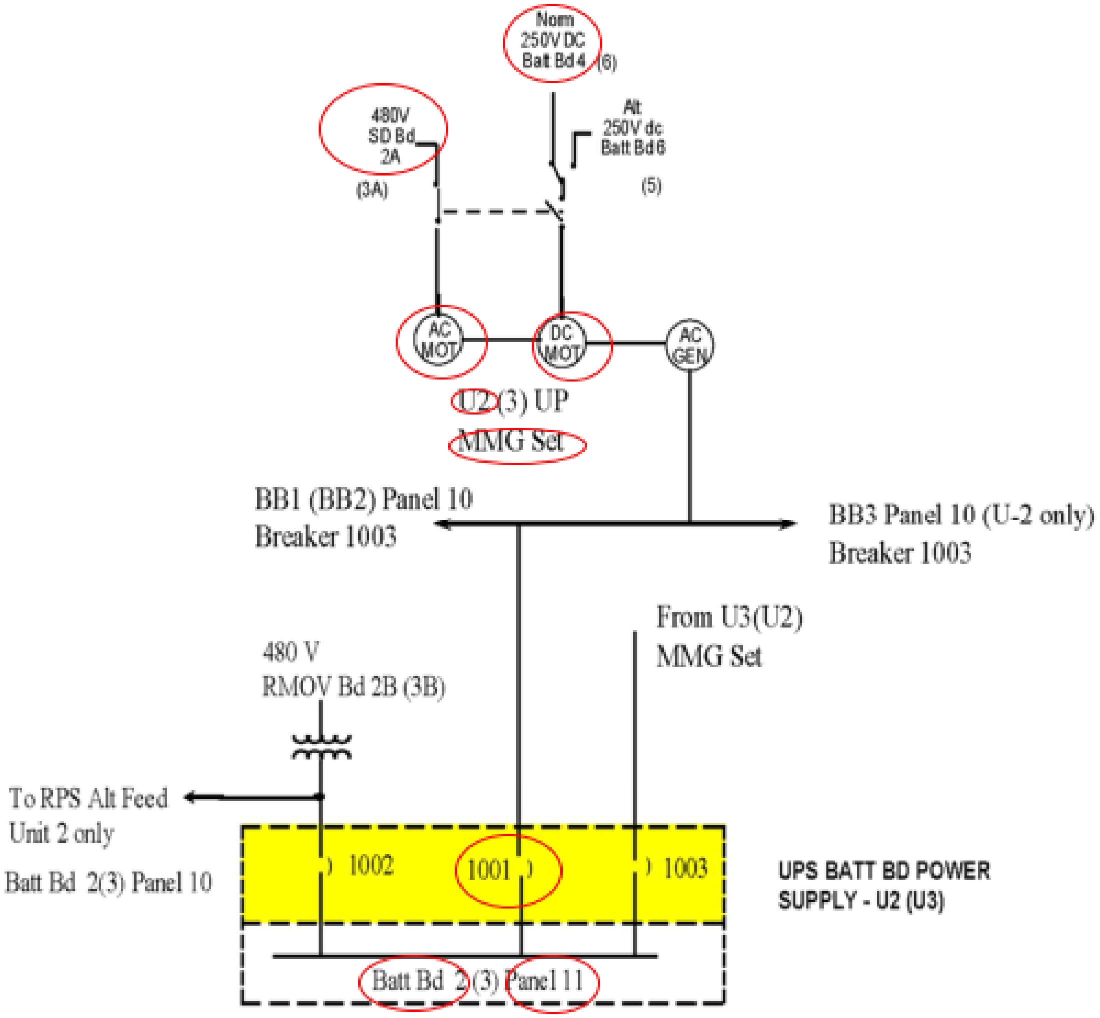
Outline of Instruction	**Instructor Notes and Methods (Optional)**
powered from the Standby Auxiliary Power System with an alternate supply from the 250VDC Power System. Each unit has a separate UPS. A Synchronizing and Control Panel for all three units is located on Panel 11 of 250V Battery Board 2.	NLO Obj. 2.h.i
(2) Each unit's battery board, panel 11 is the distribution point for UPS power. There are 3 sources of power to the battery board panel 11.	Note: The normal and alternate supply breakers are on panel 10 of the Battery Board.
(a) The MMG (Unit 2 & 3 only): Normal UPS bus power supply is a motor-motor-generator (MMG). The MMG Set consists of a 480VAC motor, a 250VDC motor, and a 240/120VAC three-wire single-phase generator, all close-coupled on a single shaft. There is an inertial flywheel between the generator and the motors, to override power supply fluctuation or momentary interruption, and to prevent a loss of generator output on an AC to DC motor transfer.	ILT/NLO/NLOR Obj. 2.a NLO/NLOR Obj. 2.e ILT Obj. 2.f
(b) The INVERTER Unit (Unit 1):	IL-2
Unit 1 is powered by an uninterruptible power supply (inverter unit).Normal supply is 480VAC to the rectifier/ inverter unit itself where it is converted to DC volts then back to a 'smooth' 120/ 240VAC signal fed to Batt. Bd 1 Panel 10/11. There is a backup 250VDC backup power supply fed to the inverter unit for a bumpless transfer in case of loss of AC power. Additionally there is a regulated AC alternate power to the inverter static switch for a continuation of power in case of inverter failure.	NLO Obj. 2.i NLOR Obj. 2.h
(c) Unit Preferred Transformer: The alternate power source is the unit preferred transformer. This transformer receives power from the 480V portion of the standby AC power system. Unit 1 transformer is from the 480 V RMOV Bd 1A Unit 2 & 3 transformers are powered from 480V RMOV Board 2B & 3B.	Note: The UPS transformer is also the alternate RPS power supply for U2.

QA Record. Non-RP - Retain in ECM (Lifetime Retention)
RP LPs - Retain in ECM (Life of Nuclear Insurance Policy, plus 10 years)

Page 13 of 46

OPL171.102 , 120V AC Power Supplies and Distribution Systems, Rev# 11

Appendix A: Illustrations
IL-11: Power Supply to UPS Batt Bd (U-2/3)



Examination Outline Cross-reference:

292001 Reactor Theory - Neutrons

K1.02 (10CFR 55.41.1)

Describe prompt and delayed neutrons.

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	292001K1.02	
Importance Rating	3.1	------

Proposed Question: **# 71**

In a comparison between a delayed neutron and a prompt neutron produced from the same fission event, the delayed neutron is more likely to _________.

A. leak out of the core

B. fission a U-238 nucleus

C. fission a Pu-240 nucleus

D. become a thermal neutron

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that delayed neutrons are born at a lower kinetic energy than prompt neutrons, and need to undergo fewer collisions and travel shorter distances before becoming thermal neutrons. Therefore, thermal neutrons are less likely to leak out while slowing down while prompt neutrons are more likely to leak out of the core.

B INCORRECT: Incorrect but plausible in that U-238 will fission if a neutron with a kinetic energy over 1.8 MeV is captured; however, delayed neutrons are born at a lower kinetic energy (~0.5MeV) than prompt neutrons, and cannot cause fast fission of U-238.

C INCORRECT: Incorrect but plausible in that Pu-240 will fission if a neutron with a kinetic energy over 1 MeV is captured; however, delayed neutrons are born at a lower kinetic energy (~0.5 MeV) than prompt neutrons, and cannot cause fast fission of Pu-240.

D **CORRECT:** *(See attached)* Delayed neutrons are born at 0.5 MeV and need to undergo fewer collisions and travel shorter distances before thermalizing. Prompt neutrons are born at higher kinetic energy and must undergo more collisions to give up energy to become a thermal neutron. This gives it a greater chance of leaking out of the core before becoming a thermal neutron. Therefore, delayed neutrons are less likely than prompt neutrons to leak out of the core and delayed neutrons are more likely to become a thermal neutron.

RO Level Justification: Tests the candidate’s knowledge of Generic Fundamentals with respect to the difference between prompt and delayed neutrons. This question is rated as Memory due to the fact that it requires the strict recall of facts.

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

Technical Reference(s): BWR Reactor Theory, Ch. 1, Rev. 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: GFES Reactor Theory, Ch.1, Obj. 20 (As available)

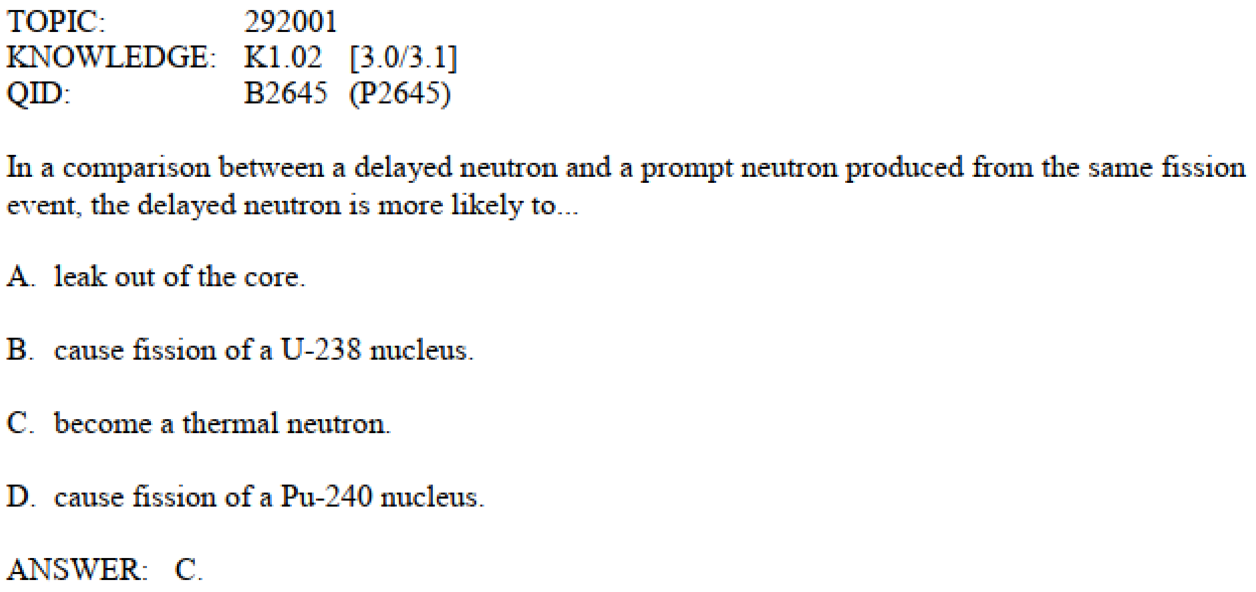
Question Source: Bank #	GFES Reactor Theory QID B2645	
Modified Bank #		(Note changes or attach parent)
New		
Question History: Last NRC Exam		

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:



TOPIC: 292001
KNOWLEDGE: K1.02 [3.0/3.1]
QID: B2645 (P2645)

In a comparison between a delayed neutron and a prompt neutron produced from the same fission event, the delayed neutron is more likely to...

A. leak out of the core.

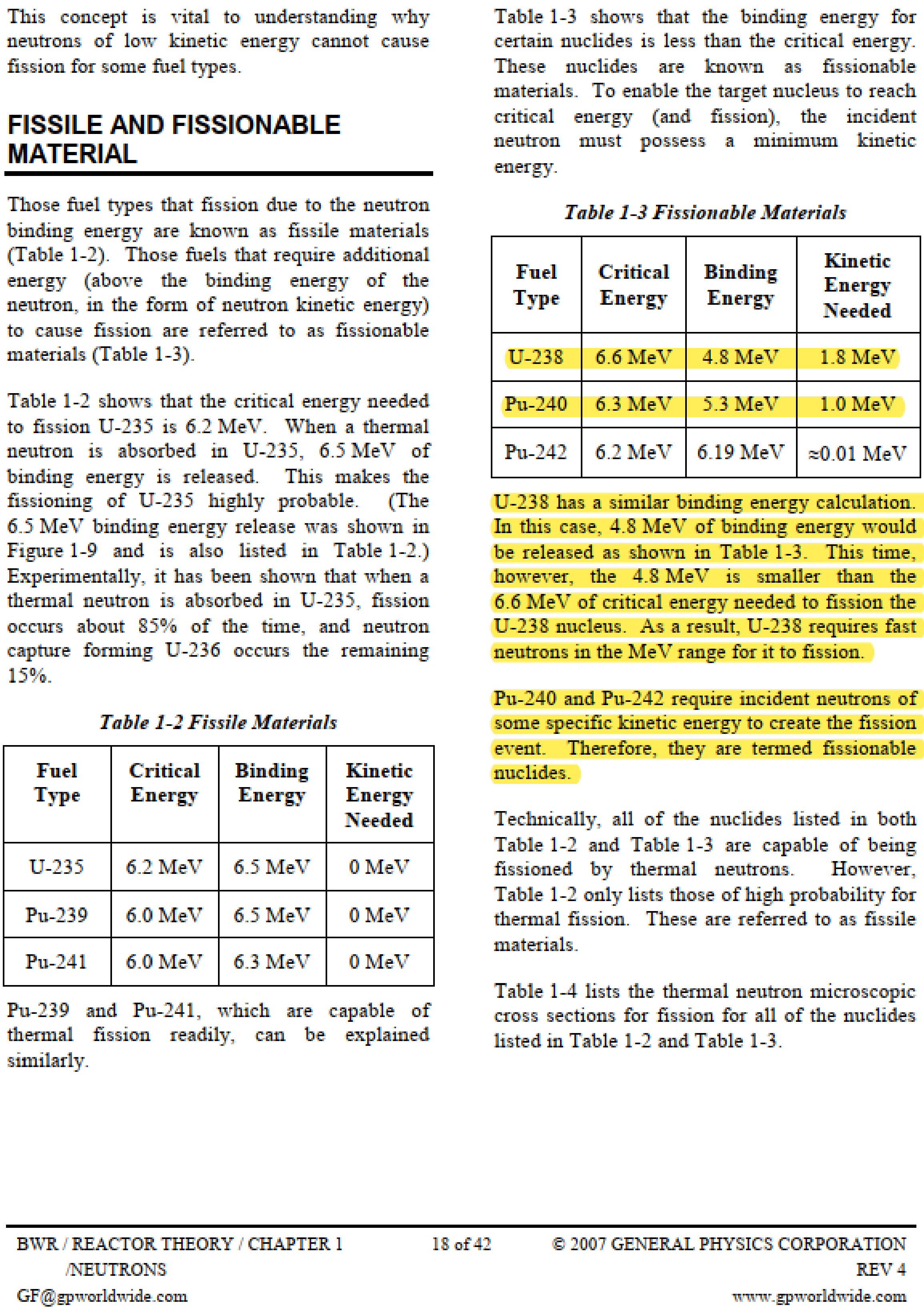
B. cause fission of a U-238 nucleus.

C. become a thermal neutron.

D. cause fission of a Pu-240 nucleus.

ANSWER: C.

Excerpts from BWR GFES Reactor Theory Guide:



This concept is vital to understanding why neutrons of low kinetic energy cannot cause fission for some fuel types.

FISSILE AND FISSIONABLE MATERIAL

Those fuel types that fission due to the neutron binding energy are known as fissile materials (Table 1-2). Those fuels that require additional energy (above the binding energy of the neutron, in the form of neutron kinetic energy) to cause fission are referred to as fissionable materials (Table 1-3).

Table 1-2 shows that the critical energy needed to fission U-235 is 6.2 MeV. When a thermal neutron is absorbed in U-235, 6.5 MeV of binding energy is released. This makes the fissioning of U-235 highly probable. (The 6.5 MeV binding energy release was shown in Figure 1-9 and is also listed in Table 1-2.) Experimentally, it has been shown that when a thermal neutron is absorbed in U-235, fission occurs about 85% of the time, and neutron capture forming U-236 occurs the remaining 15%.

Table 1-2 Fissile Materials

Fuel Type	Critical Energy	Binding Energy	Kinetic Energy Needed
U-235	6.2 MeV	6.5 MeV	0 MeV
Pu-239	6.0 MeV	6.5 MeV	0 MeV
Pu-241	6.0 MeV	6.3 MeV	0 MeV

Pu-239 and Pu-241, which are capable of thermal fission readily, can be explained similarly.

Table 1-3 shows that the binding energy for certain nuclides is less than the critical energy. These nuclides are known as fissionable materials. To enable the target nucleus to reach critical energy (and fission), the incident neutron must possess a minimum kinetic energy.

Table 1-3 Fissionable Materials

Fuel Type	Critical Energy	Binding Energy	Kinetic Energy Needed
U-238	6.6 MeV	4.8 MeV	1.8 MeV
Pu-240	6.3 MeV	5.3 MeV	1.0 MeV
Pu-242	6.2 MeV	6.19 MeV	≈0.01 MeV

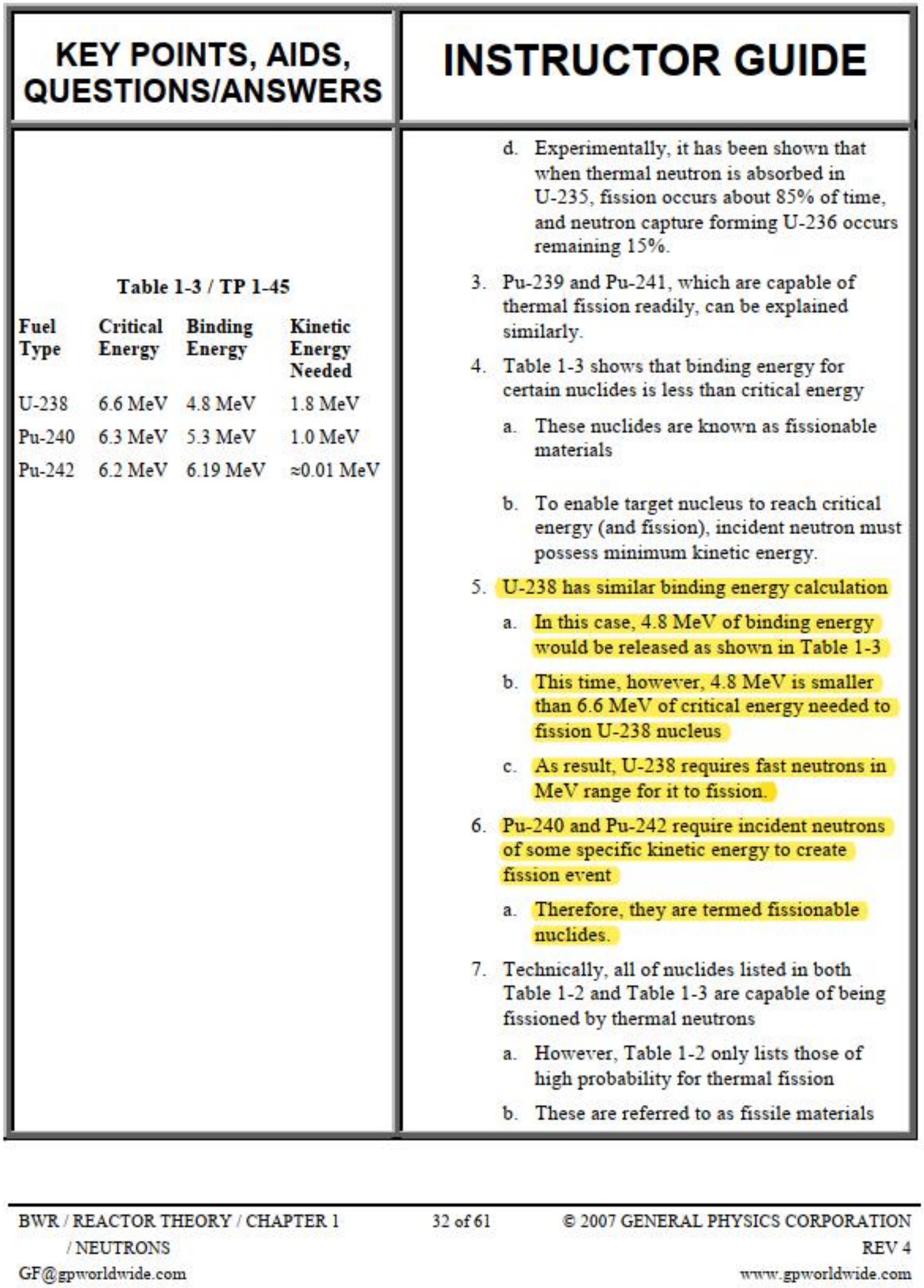
U-238 has a similar binding energy calculation. In this case, 4.8 MeV of binding energy would be released as shown in Table 1-3. This time, however, the 4.8 MeV is smaller than the 6.6 MeV of critical energy needed to fission the U-238 nucleus. As a result, U-238 requires fast neutrons in the MeV range for it to fission.

Pu-240 and Pu-242 require incident neutrons of some specific kinetic energy to create the fission event. Therefore, they are termed fissionable nuclides.

Technically, all of the nuclides listed in both Table 1-2 and Table 1-3 are capable of being fissioned by thermal neutrons. However, Table 1-2 only lists those of high probability for thermal fission. These are referred to as fissile materials.

Table 1-4 lists the thermal neutron microscopic cross sections for fission for all of the nuclides listed in Table 1-2 and Table 1-3.

Supports Distractor B, C



KEY POINTS, AIDS, QUESTIONS/ANSWERS

INSTRUCTOR GUIDE

Table 1-3 / TP 1-45

Fuel Type	Critical Energy	Binding Energy	Kinetic Energy Needed
U-238	6.6 MeV	4.8 MeV	1.8 MeV
Pu-240	6.3 MeV	5.3 MeV	1.0 MeV
Pu-242	6.2 MeV	6.19 MeV	≈0.01 MeV

d. Experimentally, it has been shown that when thermal neutron is absorbed in U-235, fission occurs about 85% of time, and neutron capture forming U-236 occurs remaining 15%.

3. Pu-239 and Pu-241, which are capable of thermal fission readily, can be explained similarly.

4. Table 1-3 shows that binding energy for certain nuclides is less than critical energy

   a. These nuclides are known as fissionable materials

   b. To enable target nucleus to reach critical energy (and fission), incident neutron must possess minimum kinetic energy.

5. U-238 has similar binding energy calculation

   a. In this case, 4.8 MeV of binding energy would be released as shown in Table 1-3

   b. This time, however, 4.8 MeV is smaller than 6.6 MeV of critical energy needed to fission U-238 nucleus

   c. As result, U-238 requires fast neutrons in MeV range for it to fission.

6. Pu-240 and Pu-242 require incident neutrons of some specific kinetic energy to create fission event

   a. Therefore, they are termed fissionable nuclides.

7. Technically, all of nuclides listed in both Table 1-2 and Table 1-3 are capable of being fissioned by thermal neutrons

   a. However, Table 1-2 only lists those of high probability for thermal fission

   b. These are referred to as fissile materials

BWR / REACTOR THEORY / CHAPTER 1 / NEUTRONS 32 of 61
REV 4
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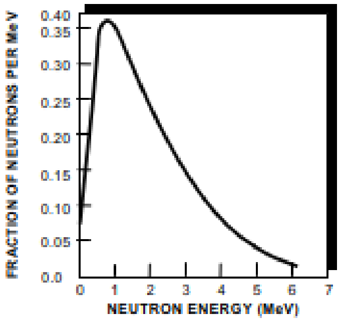
KEY POINTS, AIDS, QUESTIONS/ANSWERS

Objective 13.c, 20.a, 21, 22, and 23

Figure 1-24 / TP 1-53



INSTRUCTOR GUIDE

4. Neutrons in thermal equilibrium with their surroundings are defined as thermal neutrons. Depending on their surroundings, thermal neutrons can be fast, intermediate, or slow
   a. For conditions in commercial power plants, thermal neutron is in slow energy region
   b. Typical energies of thermal neutrons are 0.025 eV at 68°F and 0.049 eV at 550°F.

C. Origin Classifications

1. Neutrons that are produced from neutron-induced fission are defined as fission neutrons.
2. Neutrons that are produced independently of fission are defined as source neutrons
3. Source neutrons consist of neutrons produced by installed neutron sources and intrinsic neutron sources
4. Intrinsic neutron sources include neutrons produced by spontaneous fission and by alpha and gamma-neutron reactions
5. Source neutrons will be covered in greater detail in Chapter 3.

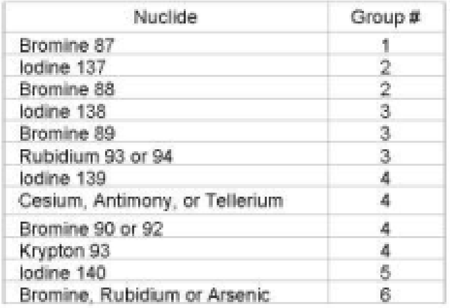
D. Production Time Classification

1. Neutrons emitted within 10^{-14} seconds of fission event that are direct result of fission process are defined as *prompt neutrons*
   a. The number of prompt neutrons emitted depends on type of fuel used
   b. For example, prompt neutrons account for 99.36% of all U-235 fission neutrons

KEY POINTS, AIDS, QUESTIONS/ANSWERS

Objectives 13.d, 20.b, 21, 22, and 23

Table 1-7 / TP 1-54



Nuclide	Group #
Bromine 87	1
Iodine 137	2
Bromine 88	2
Iodine 138	3
Bromine 89	3
Rubidium 93 or 94	3
Iodine 139	4
Cesium, Antimony, or Tellerium	4
Bromine 90 or 92	4
Krypton 93	4
Iodine 140	5
Bromine, Rubidium or Arsenic	6

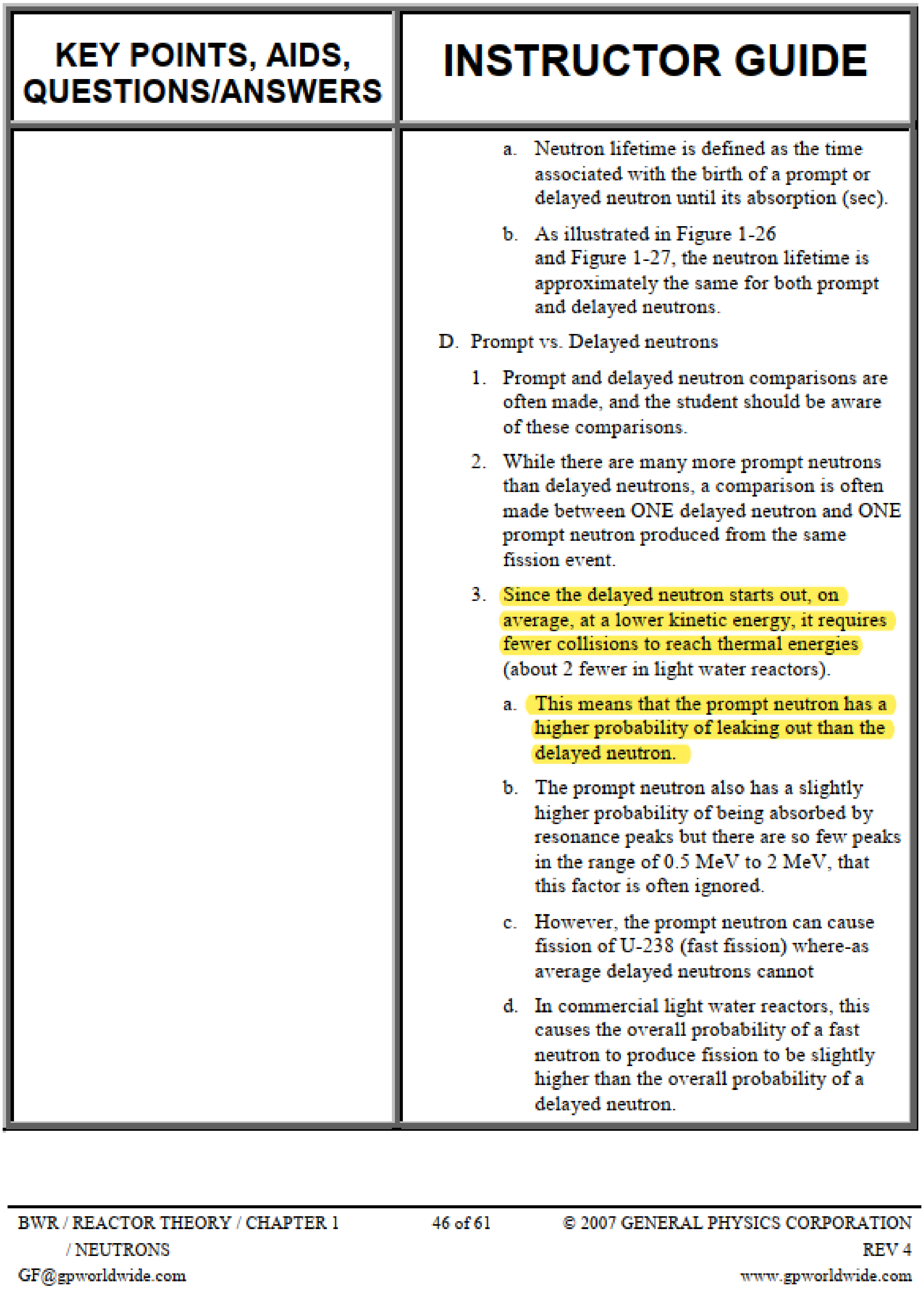
INSTRUCTOR GUIDE

   c. The most probable energy for prompt neutron is approximately 1 MeV, and average energy is approximately 2 MeV (Figure 1-24)

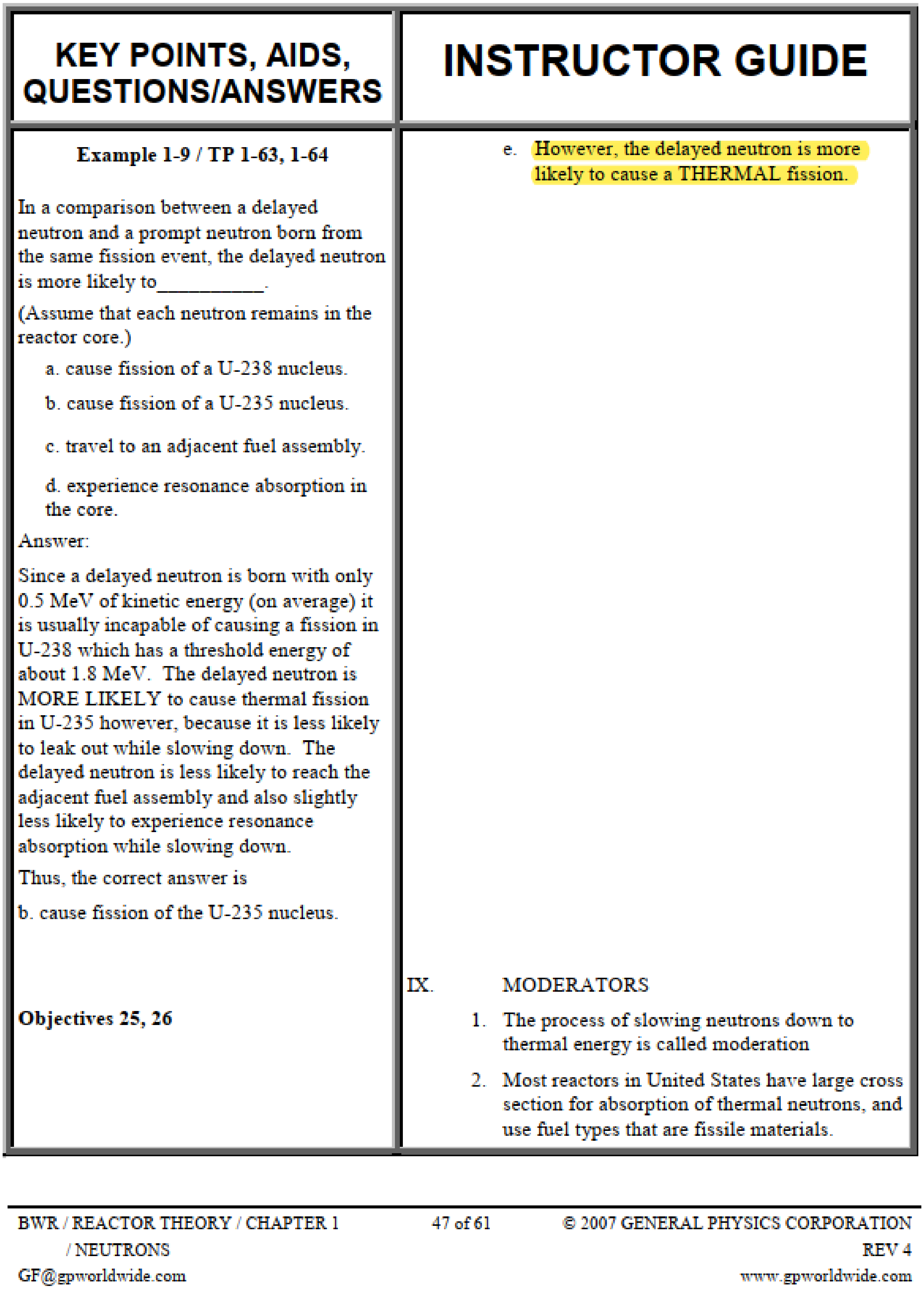
2. Neutrons born more than 10^{-14} seconds after fission event are defined as *delayed neutrons*
   a. On average, approximately 12.7 seconds after fission event, delayed neutron is emitted
   b. Delayed neutrons are born fast, but at lower energy level than prompt neutrons (approximately 0.5 MeV)
   c. They are named “delayed” neutrons because these neutrons are produced well after fission event.

3. As previously discussed, fission event produces two fission fragments
   a. These fission fragments are highly unstable and undergo decay to achieve stability
   b. Initially, beta minus decay occurs, and excited daughter product is result
   c. This daughter product also is excited and decays
   d. One possible method of decay of the excited daughter is that of neutron emission
      1) The resulting neutron is called a delayed neutron
   e. A small number of fission products go through this process.

Supports Distractor A



KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
	a. Neutron lifetime is defined as the time associated with the birth of a prompt or delayed neutron until its absorption (sec). b. As illustrated in Figure 1-26 and Figure 1-27, the neutron lifetime is approximately the same for both prompt and delayed neutrons. D. Prompt vs. Delayed neutrons 1. Prompt and delayed neutron comparisons are often made, and the student should be aware of these comparisons. 2. While there are many more prompt neutrons than delayed neutrons, a comparison is often made between ONE delayed neutron and ONE prompt neutron produced from the same fission event. 3. Since the delayed neutron starts out, on average, at a lower kinetic energy, it requires fewer collisions to reach thermal energies (about 2 fewer in light water reactors). a. This means that the prompt neutron has a higher probability of leaking out than the delayed neutron. b. The prompt neutron also has a slightly higher probability of being absorbed by resonance peaks but there are so few peaks in the range of 0.5 MeV to 2 MeV, that this factor is often ignored. c. However, the prompt neutron can cause fission of U-238 (fast fission) where-as average delayed neutrons cannot d. In commercial light water reactors, this causes the overall probability of a fast neutron to produce fission to be slightly higher than the overall probability of a delayed neutron.



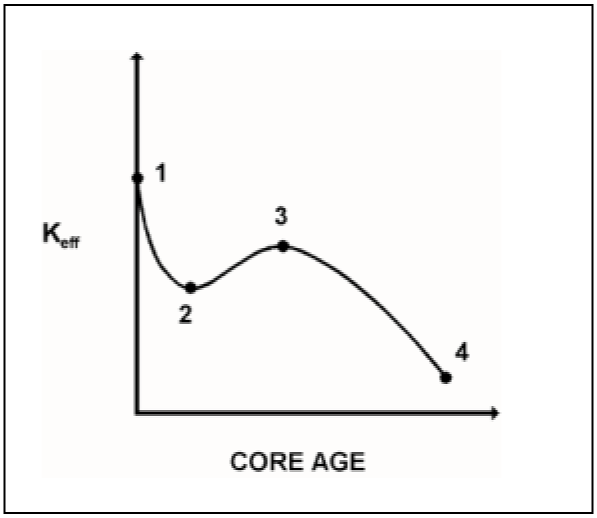
KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Example 1-9 / TP 1-63, 1-64 In a comparison between a delayed neutron and a prompt neutron born from the same fission event, the delayed neutron is more likely to__________. (Assume that each neutron remains in the reactor core.) a. cause fission of a U-238 nucleus. b. cause fission of a U-235 nucleus. c. travel to an adjacent fuel assembly. d. experience resonance absorption in the core. Answer: Since a delayed neutron is born with only 0.5 MeV of kinetic energy (on average) it is usually incapable of causing a fission in U-238 which has a threshold energy of about 1.8 MeV. The delayed neutron is MORE LIKELY to cause thermal fission in U-235 however, because it is less likely to leak out while slowing down. The delayed neutron is less likely to reach the adjacent fuel assembly and also slightly less likely to experience resonance absorption while slowing down. Thus, the correct answer is b. cause fission of the U-235 nucleus.	e. However, the delayed neutron is more likely to cause a THERMAL fission.
Objectives 25, 26	IX. MODERATORS 1. The process of slowing neutrons down to thermal energy is called moderation 2. Most reactors in United States have large cross section for absorption of thermal neutrons, and use fuel types that are fissile materials.

Examination Outline Cross-reference:	Level	RO	SRO
292007 Reactor Theory - Fuel Depletion and Burnable Poisons	Tier #	4	------
K1.03 (10CFR 55.41.1)	Group #	------	------
Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points.	K/A #	292007K1.03	
	Importance Rating	2.7	------

Proposed Question: **# 72**

Referring to the drawing of effective neutron multiplication factor (k_{eff}) versus Core Age, which **ONE** of the following completes the statement below?

The major cause for the change in k_{eff} from point 3 to point 4 is the _______.



A. depletion of fuel

B. depletion of Control Rods

C. burnout of burnable poisons

D. burnout of fission product poisons

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* From point 3 to 4, fuel depletion and fission product poison buildup (other than xenon and samarium) continuously insert negative reactivity. Excess reactivity (k_{excess}) lowers at a rising rate to the end of the fuel cycle.

B INCORRECT: Incorrect but plausible in that Control Rods may deplete over their service life but do not deplete at a rate that coincides with the curve between points 3 and 4.

C INCORRECT: Incorrect but plausible in that burnable poisons are loaded into the core; however, the burnout of the burnable poisons causes k_{excess} to increase from 25 days to about one-third to one-half of the fuel cycle (points 2 to 3). This occurs because the burnable poison depletes faster than the fuel.

D INCORRECT: Incorrect but plausible in that following a Reactor startup, during the first 25 days of the cycle (points 1 to 2), fission product poisons (samarium and xenon) build to equilibrium levels. Excess reactivity (k_{excess}) decreases until the equilibrium concentrations at point 2 are established.

RO Level Justification: Tests the candidate's knowledge of Generic Fundamentals, specifically why k-effective changes over core life. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): BWR Reactor Theory, Ch. 7, Rev. 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **k_{eff} vs. Core Age Graph**

Learning Objective: GFES Reactor Theory, Ch. 7, Obj. 3 (As available)

Question Source:

Bank #	GFES Reactor Theory QID B1563	
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 **X**

55.43

Comments:

Copy of Bank Question:

TOPIC: 292007
KNOWLEDGE: K1.03 [2.4/2.7]
QID: B1563

Refer to the drawing of K_{eff} versus core age (see figure below).

The major cause for the change in K_{eff} from point 3 to point 4 is the...

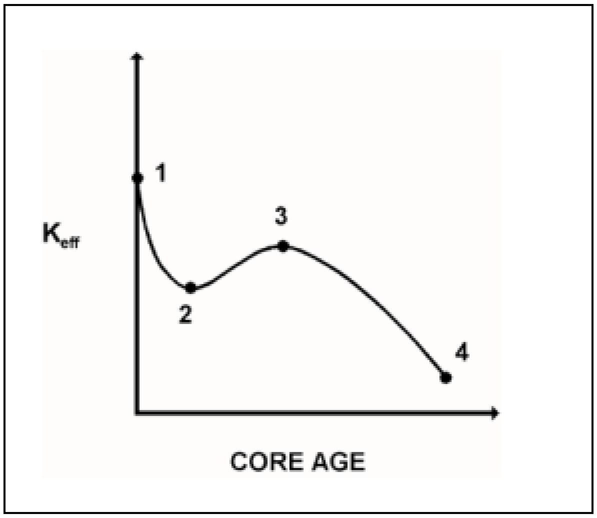
A. depletion of U-235.

B. depletion of U-238.

C. burnout of burnable poisons.

D. buildup of fission product poisons.

ANSWER: A.

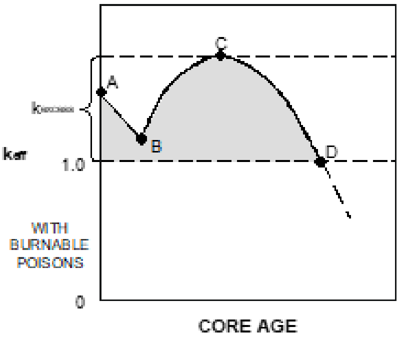


Excerpt from BWR GFES Reactor Theory, Ch. 7:

KEY POINTS, AIDS, QUESTIONS/ANSWERS

Objective 3

Figure 7-1 / TP 7-8



INSTRUCTOR GUIDE

B. Figure 7-1 illustrates total core k_{eff} over core life. The combination of control rods and burnable poisons controls positive reactivity in core due to excess fuel (k_{excess}).

C. For this discussion, assume that reactor is taken to 100% power at beginning of core life and maintained at that power level until EOC

  1. To maintain constant power, operator must change control rod density in response to negative and positive reactivity changes that occur in core during cycle.

D. During first 25 days of cycle (points A to B), fission product poisons (samarium and xenon) build to equilibrium levels

  1. Excess reactivity (k_{excess}) decreases until equilibrium concentrations at point B are established

E. The burnout of burnable poisons causes k_{excess} to increase from 25 days to about one-third to one-half of fuel cycle (points B to C)

  1. This occurs because burnable poison depletes faster than fuel. As core operation approaches point C, rate of fuel depletion and rate of poison depletion become same

  2. At this point, core has smallest value of shutdown margin.

F. From point C to D, fuel depletion and fission product poison buildup (other than xenon and samarium) continuously insert negative reactivity

Examination Outline Cross-reference:	Level	RO	SRO
293003 Thermodynamics - Steam	Tier #	4	------
K1.22 (10CFR 55.41.14)	Group #	------	------
Explain the usefulness of steam tables to the control room operator.	K/A #	293003 K1.22	
	Importance Rating	3.2	------

Proposed Question: **# 73**

Unit 1 has inserted a Reactor SCRAM with the following conditions:

- Reactor Pressure is 885 psig
- The NUSO has directed a cooldown using SRVs
- The BOP has opened one SRV

Given the conditions above, the tail pipe temperature for the open SRV will be approximately **(1)** .

In accordance with Tech Spec 3.4.9, the **MAXIMUM** cooldown rate allowed is **(2)** per hour.

[REFERENCE PROVIDED]

A. (1) 200 °F
(2) 90 °F

B. (1) 200 °F
(2) 100 °F

C. (1) 300 °F
(2) 90 °F

D. (1) 300 °F
(2) 100 °F

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate misapplies the Mollier diagram. The pressure in the Suppression Chamber is normally less than 1 psig which is about 14 - 14.7 psia. With about 10 feet of water above the SRV T-quenchers in the Suppression Pool, it would require approximately 20 psia to displace water from the SRV exhaust. Using the Mollier diagram, an approximate saturation temperature for 20 psia is between 200 - 220 °F. Second part is incorrect but plausible since 90 °F per hour is the cooldown limit in accordance with 1-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations.

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*) Using the Mollier diagram, when steam exhausts to lower pressure, enthalpy remains constant, so the SRV Tail Pipe Temperature is determined by following the initial enthalpy line straight from the intersection of 900 psia (885 psig) and the Saturation Curve line. From that intersection point to the right across the Mollier diagram, this results in an SRV tail pipe temperature of approximately 300 °F exhausting to the Suppression Chamber. For second part, each Unit's Tech Spec 3.4.9 has a limit for heat up or cooldown not to exceed 100 °F per hour using SR 3.4.9.1.

RO Level Justification: Tests the candidate's knowledge and ability to use steam tables and the Mollier diagram. 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): 2000 ASME Steam Tables (Attach if not previously provided)
U1 Tech Spec 3.4.9, Amend. 315
1-GOI-100-12A, Rev. 36

Proposed references to be provided to applicants during examination: **Steam Tables with Mollier Diagram**

Learning Objective: GFES Thermodynamics, Ch. 3, Obj. 7 (As available)

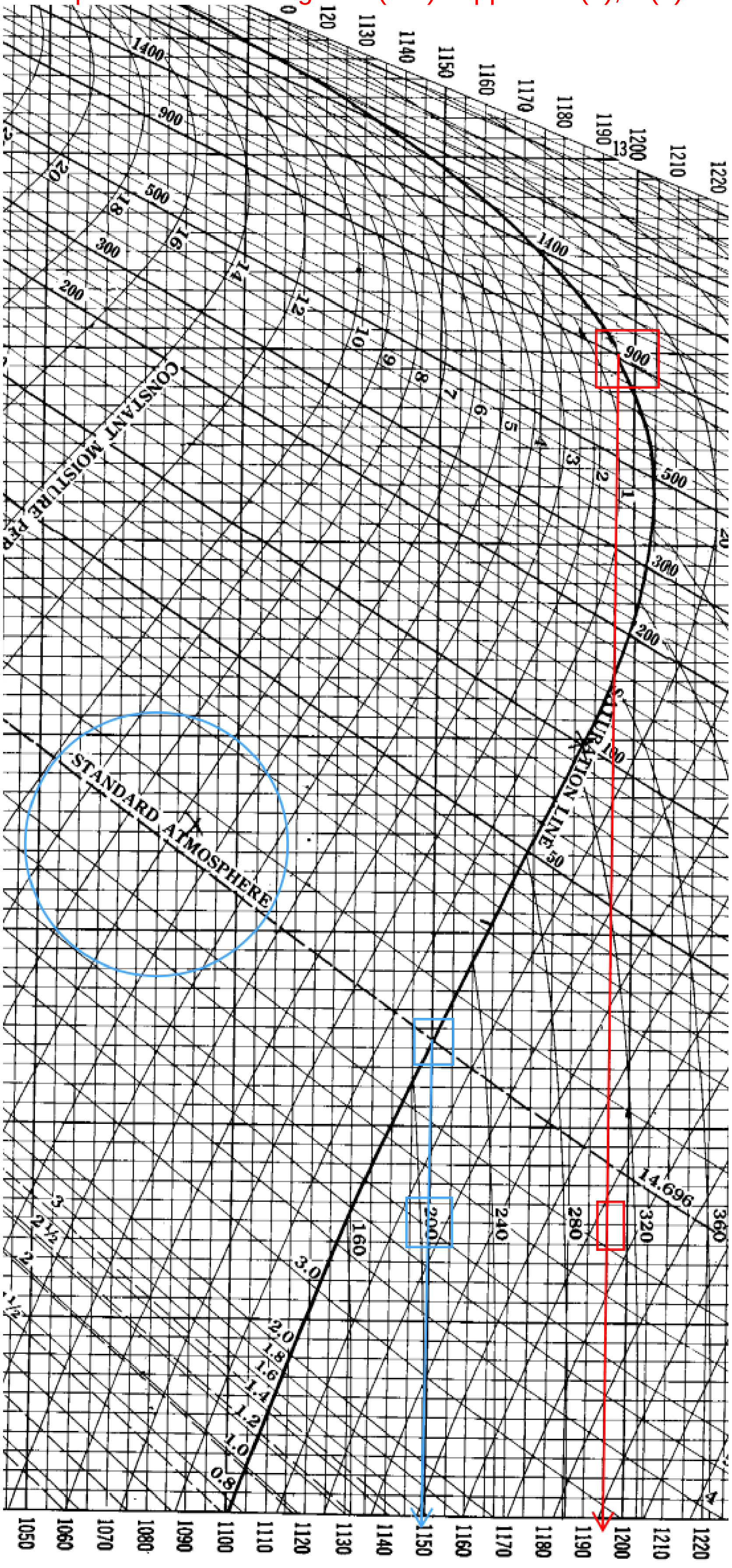
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

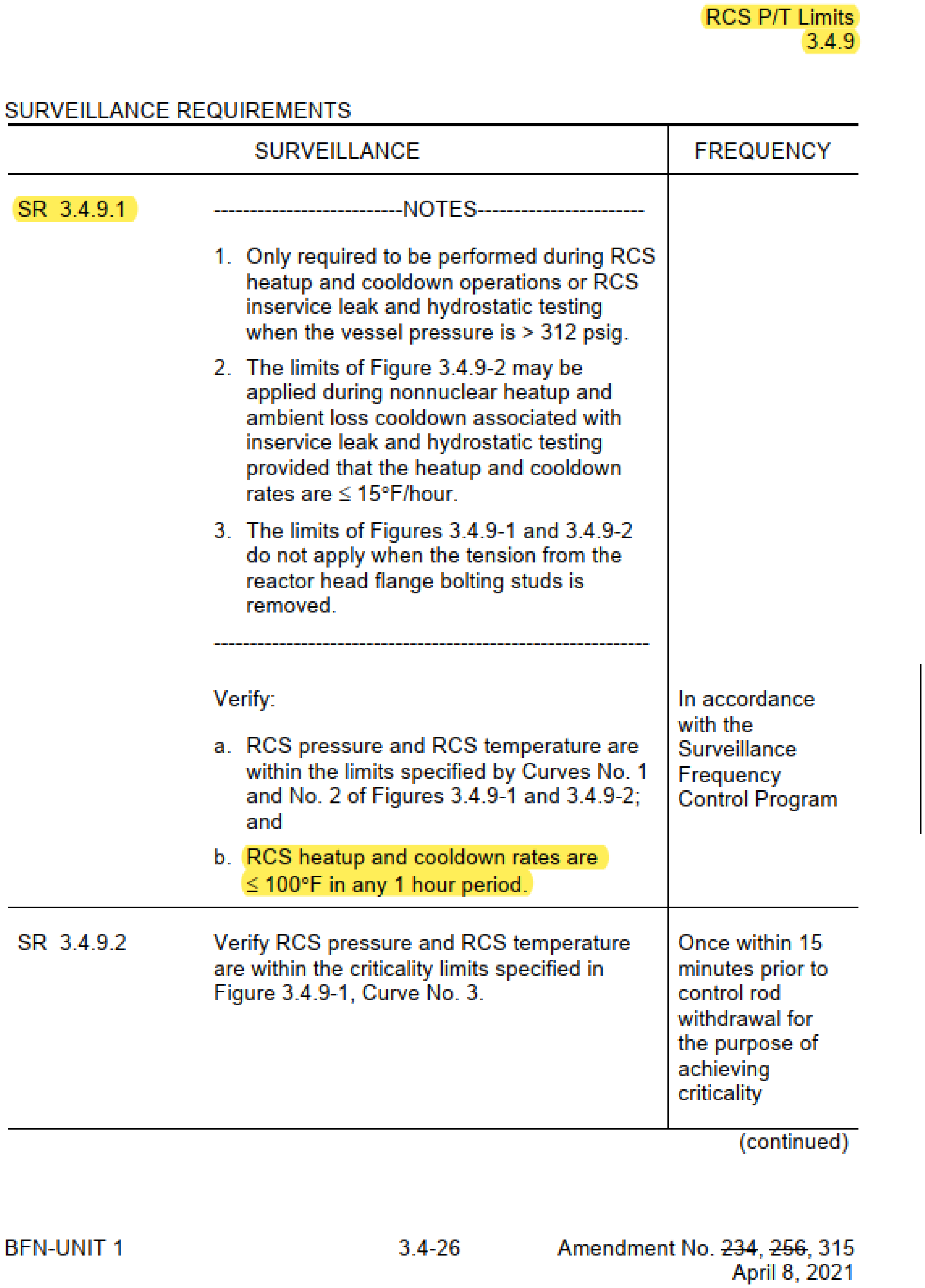
10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Excerpt from Mollier diagram: (red) supports C(1), D(1) and (blue) - Distractors A(1), B(1)



Excerpt from Tech Spec 3.4.9 Surveillance Requirements:



RCS P/T Limits
3.4.9

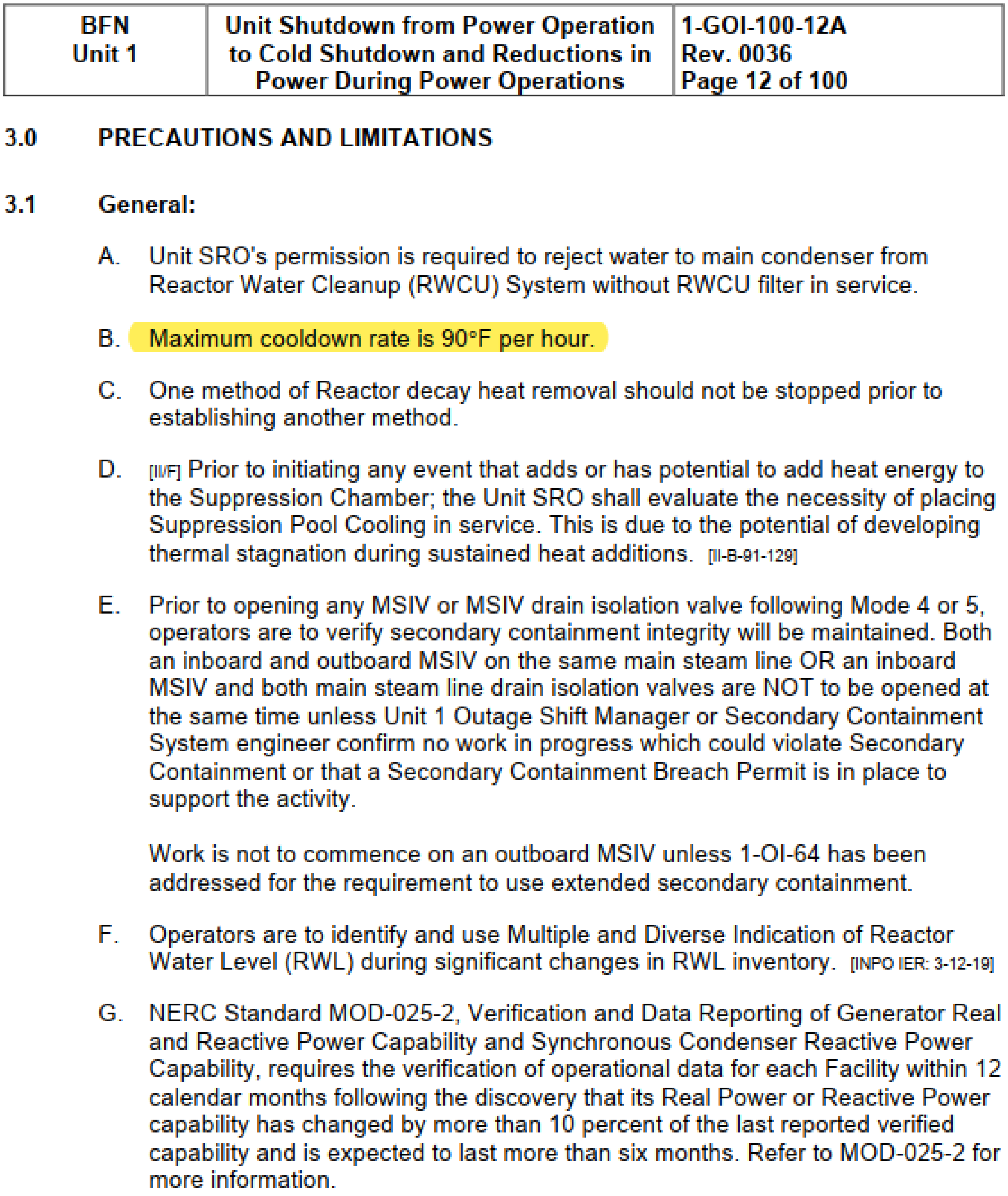
SURVEILLANCE REQUIREMENTS

	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.	In accordance with the Surveillance Frequency Control Program
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)

BFN-UNIT 1 3.4-26 Amendment No. ~~234~~, ~~256~~, 315
April 8, 2021

Excerpt from 1-GOI-100-12A: Supports Distractors A(2), C(2)



BFN Unit 1	Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations	1-GOI-100-12A Rev. 0036 Page 12 of 100

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General:

A. Unit SRO'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 SRO 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]

G. NERC Standard MOD-025-2, Verification and Data Reporting of Generator Real and Reactive Power Capability and Synchronous Condenser Reactive Power Capability, requires the verification of operational data for each Facility within 12 calendar months following the discovery that its Real Power or Reactive Power capability has changed by more than 10 percent of the last reported verified capability and is expected to last more than six months. Refer to MOD-025-2 for more information.

Examination Outline Cross-reference:

262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)

K6.02 (10CFR 55.41.7)

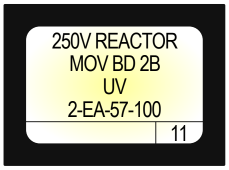
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Uninterruptible Power Supply (AC/DC):

- DC electrical distribution

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	262002K6.02	
Importance Rating	3.4	------

Proposed Question: **# 74**

Unit 2 is operating at 100% RTP with the following conditions:



- 250V REACTOR MOV BOARD 2B UNDERVOLTAGE (2-9-8C, Window 11) alarms
- **NO** Operator actions have been taken

Given the conditions above, 250V DC RMOV Board 2B provides _____**(1)**_____ to its respective ECCS Analog Trip Unit (ATU) **AND** _____**(2)**_____ ECCS ATU is de-energized.

A. (1) **ONLY** backup power supply
(2) Division I, Panel 2-9-81

B. (1) **ONLY** backup power supply
(2) Division II, Panel 2-9-82

C. (1) **BOTH** the normal **AND** backup power supply
(2) Division I, Panel 2-9-81

D. (1) **BOTH** the normal **AND** backup power supply
(2) Division II, Panel 2-9-82

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT**:** First part is incorrect but plausible in that the ECCS ATU power supply configuration is opposite of logic. Division I is powered by 'B' while Division II is powered by 'A' and as with many complex electrical systems at BFN, candidates could easily confuse the normal and backup power supply configurations. Second part is correct (*See C*).

B INCORRECT**:** First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that Division I Panel 2-9-81 is powered by 250V RMOV Board 2B while Division II Panel 2-9-82 is powered by 250V RMOV Board 2A and as with many complex electrical systems at BFN, candidates could easily confuse each Unit's Aux Instrument Room's power supply configurations.

C **CORRECT:** (*See attached*) In accordance with 2-AOI-57-11, Loss of Power to an ECCS ATU Panel/ECCS Inverter, Attachment 5 – Redundant Power Supplies, 250V DC RMOV Board 2B provides both the normal and backup power supplies to the Division I ECCS Inverter and 250/24 VDC Converter. For second part, 2-AOI-57-11, Division 1 Panel 2-9-81, ECCS ATU is de-energized located in the Unit 2 Aux Instrument Room.

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 effect of a DC power supply malfunction as it relates to AC/DC electrical distribution to critical ECCS components. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-ARP-9-8C, Rev. 19 (Attach if not previously provided)
2-AOI-57-11, Rev. 17

Proposed references to be provided to applicants during examination: **250V REACTOR MOV BOARD 2B UNDERVOLTAGE (2-9-8C, Window 11)**

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

Learning Objective: OPL171.037 Obj. 8 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 2104 #51	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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:

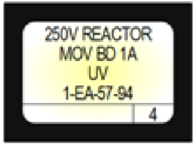
Examination Outline Cross-reference:
295004 (APE 4) Partial or Complete Loss of D.C. Power / 6
AK3.02 (10CFR 55.41.5)
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following:

- Ground isolation/fault determination

Level	RO	SRO
Tier #	1	------
Group #	1	------
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

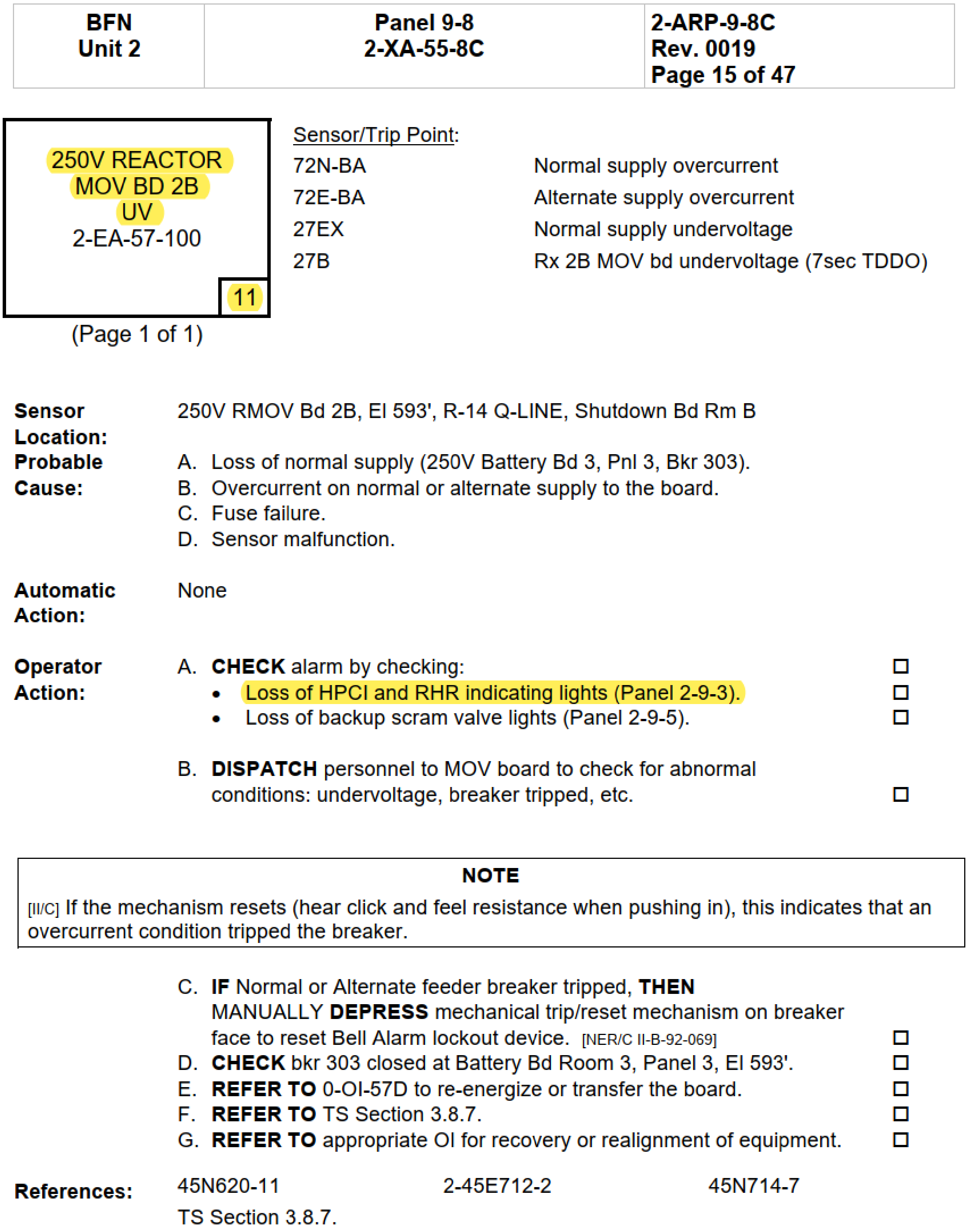
B. (1) **ONLY** one power source
(2) Division II, Panel 1-9-82

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**

Excerpt from 2-ARP-9-8C:



BFN **Unit 2**	**Panel 9-8** **2-XA-55-8C**	**2-ARP-9-8C** **Rev. 0019** **Page 15 of 47**

250V REACTOR
MOV BD 2B
UV
2-EA-57-100

11

(Page 1 of 1)

Sensor/Trip Point:

72N-BA	Normal supply overcurrent
72E-BA	Alternate supply overcurrent
27EX	Normal supply undervoltage
27B	Rx 2B MOV bd undervoltage (7sec TDDO)

Sensor Location: 250V RMOV Bd 2B, El 593', R-14 Q-LINE, Shutdown Bd Rm B

Probable Cause:

A. Loss of normal supply (250V Battery Bd 3, Pnl 3, Bkr 303).
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 checking: ☐
- Loss of HPCI and RHR indicating lights (Panel 2-9-3). ☐
- Loss of backup scram valve lights (Panel 2-9-5). ☐

B. **DISPATCH** personnel to MOV board 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 breaker face to reset Bell Alarm lockout device. [NER/C II-B-92-069] ☐
D. **CHECK** bkr 303 closed at Battery Bd Room 3, Panel 3, El 593'. ☐
E. **REFER TO** 0-OI-57D to re-energize or transfer the board. ☐
F. **REFER TO** TS Section 3.8.7. ☐
G. **REFER TO** appropriate OI for recovery or realignment of equipment. ☐

References: 45N620-11 2-45E712-2 45N714-7
TS Section 3.8.7.

Excerpts 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 4 of 31**

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 9-81 or 9-82 or loss of an ECCS inverter.

NOTES

1) Each Inverter provides electrical power to divisional logics plus one of the two redundant power supplies to its divisional ATU cabinet (**REFER TO** Attachment 5). The power supplies to ECCS ATU Panel 9-81 (Div. I), ECCS ATU Panel 9-82 (Div. II) and the ECCS inverters are as follows:

Panel 9-81

- Division I ECCS inverter	250V RMOV Board 2B, compartment 8A.
- Division I 250/24vdc converter	250V RMOV Board 2B, compartment 1B1.

Panel 9-82

- Division II ECCS inverter	250V RMOV Board 2A, compartment 11A1
- Division II 250/24vdc converter	250V RMOV Board 2A, compartment 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, or simultaneous loss of both redundant 24vdc power supplies in each ECCS ATU panel.

2) The total loss of power to the ECCS ATU panels results in power loss to all instrumentation on:

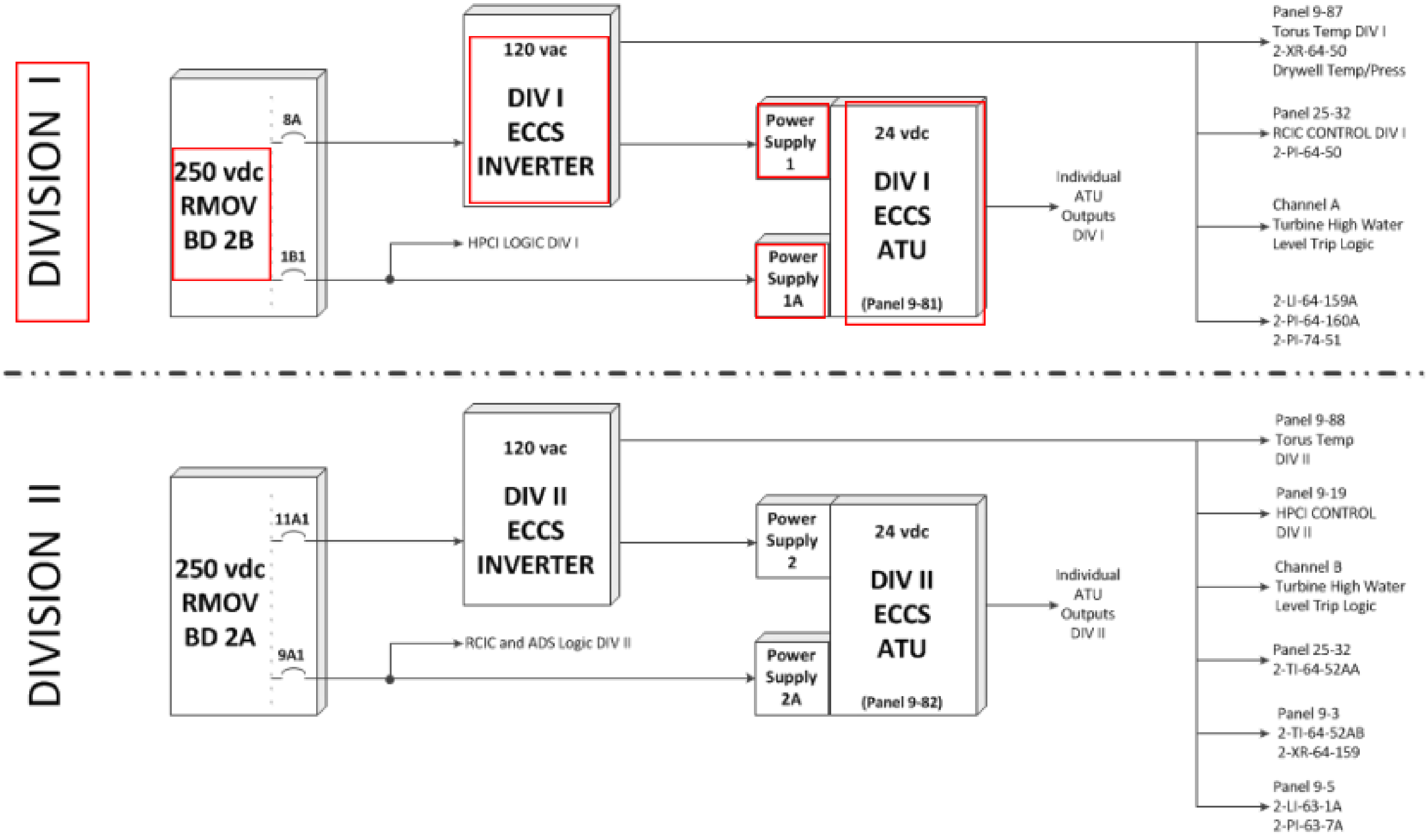
- Division I	Panel 9-81 (Aux Instrument Room)
- Division II	Panel 9-82 (Aux Instrument Room)

RHR system I(II) containment spray valve operation will require the use of the manual override switch 2-XS-74-122(130) upon a loss of Panel 9-81(82) due to a loss of the two-thirds core height level channel.

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



Examination Outline Cross-reference:

292005 Reactor Theory – Control Rods

K1.12 (10CFR 55.41.1)

Describe the effects of deep and shallow control rods on axial and radial flux distribution.

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	292005K1.12	
Importance Rating	2.9	------

Proposed Question: **# 75**

Control Rods that are located at position ___**(1)**___ in the Reactor Core are considered shallow.

The radial flux response to withdrawing a shallow Control Rod located at the **PERIPHERY** of the Core is based on a ___**(2)**___ fuel-moderator ratio.

A. (1) 10
(2) higher

B. (1) 10
(2) lower

C. (1) 40
(2) higher

D. (1) 40
(2) lower

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that deep Control Rods are Rods that are inserted to greater than 2/3 into the Core or ranging from notch positions 00 through 16. The lower the notch position number, the farther inserted into the Core the Control Rods are. Second part is incorrect but plausible in that the candidate could confuse axial and radial flux response as it relates to deep or shallow Control Rod location in the Core based on a higher fuel-moderator ratio. A shallow Control Rod has a large effect at the bottom of the channel, which in turn causes boiling to occur sooner and therefore voiding to go up. This will tend to push power production down in the channel toward the lower void percentage region and thereby have the greatest effect on axial profile.

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)* There are 48 notch positions relative to Control Rod and Core height. Shallow Control Rods are Rods that are inserted to less than 1/3 into the Core or ranging from notch positions 34 through 48. The higher the notch position number, the farther out of the Core or withdrawn the Control Rods are. For second part, the response of radial flux to shallow Control Rod movement in the periphery of the Core is relative to fuel-moderator ratio, which is lower in the Core periphery. Withdrawing shallow Control Rods have very little effect on average core Power (radial flux) while impacting axial flux more. This is especially true for Control Rods located at the outer portions of the Core where Reactor Power is generally less impacted during withdrawal. Deep Control Rods affect a lower flux level region compared to shallow Control Rods due to the presence of significant voiding around the tips of deep Rods. Flux density will be greater at the center of the Core than at the Core edges or periphery.

RO Level Justification: Tests the candidate's knowledge of the effect of deep and shallow Control Rods on axial and radial flux distribution in the Reactor Core. 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 for shallow or deep Control Rod positions in the Reactor Core as it relates to axial and radial flux distribution.

Technical Reference(s): BWR Reactor Theory, Ch. 5, Rev. 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: GFES Reactor Theory, Ch. 5, Obj. 20 (As available)

Question Source:	Bank #		
	Modified Bank #	GFES Reactor Theory QID B1757	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge

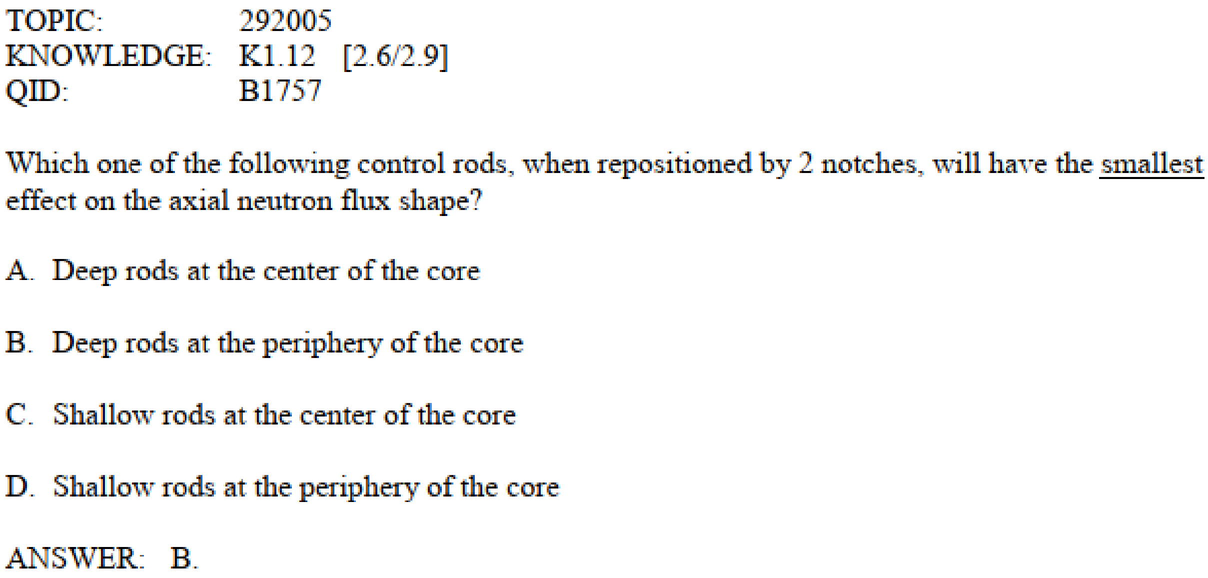
Comprehension or Analysis **X**

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

55.43

Comments:

Copy of Bank Question:



TOPIC: 292005
KNOWLEDGE: K1.12 [2.6/2.9]
QID: B1757

Which one of the following control rods, when repositioned by 2 notches, will have the smallest effect on the axial neutron flux shape?

A. Deep rods at the center of the core

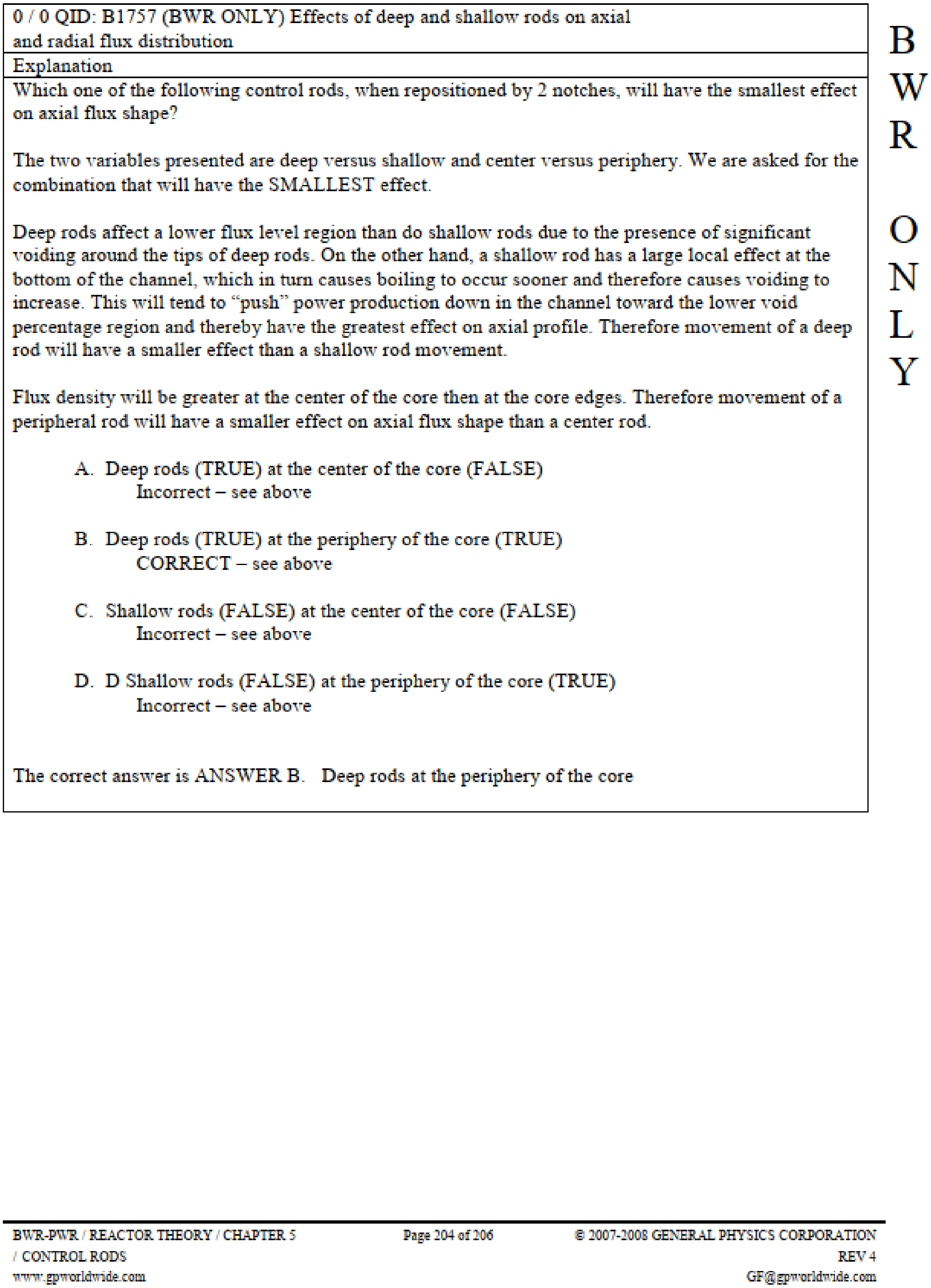
B. Deep rods at the periphery of the core

C. Shallow rods at the center of the core

D. Shallow rods at the periphery of the core

ANSWER: B.

Excerpt from BWR GFES Reactor Theory Exam Bank for question B1757:



0 / 0 QID: B1757 (BWR ONLY) Effects of deep and shallow rods on axial and radial flux distribution

Explanation

Which one of the following control rods, when repositioned by 2 notches, will have the smallest effect on axial flux shape?

The two variables presented are deep versus shallow and center versus periphery. We are asked for the combination that will have the SMALLEST effect.

Deep rods affect a lower flux level region than do shallow rods due to the presence of significant voiding around the tips of deep rods. On the other hand, a shallow rod has a large local effect at the bottom of the channel, which in turn causes boiling to occur sooner and therefore causes voiding to increase. This will tend to "push" power production down in the channel toward the lower void percentage region and thereby have the greatest effect on axial profile. Therefore movement of a deep rod will have a smaller effect than a shallow rod movement.

Flux density will be greater at the center of the core then at the core edges. Therefore movement of a peripheral rod will have a smaller effect on axial flux shape than a center rod.

A. Deep rods (TRUE) at the center of the core (FALSE)
   Incorrect – see above

B. Deep rods (TRUE) at the periphery of the core (TRUE)
   CORRECT – see above

C. Shallow rods (FALSE) at the center of the core (FALSE)
   Incorrect – see above

D. D Shallow rods (FALSE) at the periphery of the core (TRUE)
   Incorrect – see above

The correct answer is ANSWER B. Deep rods at the periphery of the core

B
W
R

O
N
L
Y

BWR-PWR / REACTOR THEORY / CHAPTER 5 / CONTROL RODS
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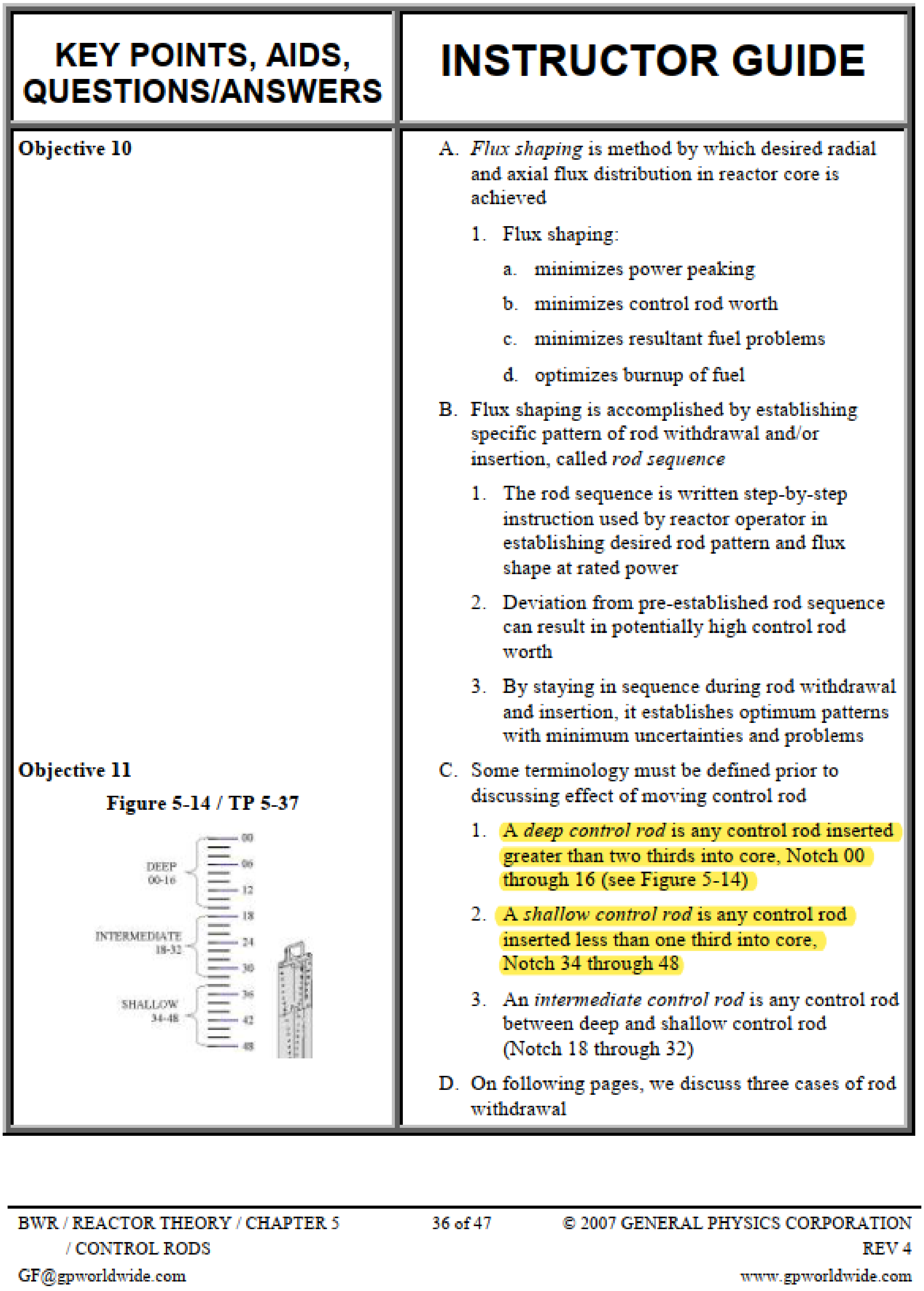
Excerpt from BWR GFES Reactor Theory, Ch. 5: Also supports Distractors A(1), B(1)



KEY POINTS, AIDS, QUESTIONS/ANSWERS

INSTRUCTOR GUIDE

Objective 10

A. *Flux shaping* is method by which desired radial and axial flux distribution in reactor core is achieved

1. Flux shaping:
   a. minimizes power peaking
   b. minimizes control rod worth
   c. minimizes resultant fuel problems
   d. optimizes burnup of fuel

B. Flux shaping is accomplished by establishing specific pattern of rod withdrawal and/or insertion, called *rod sequence*

1. The rod sequence is written step-by-step instruction used by reactor operator in establishing desired rod pattern and flux shape at rated power
2. Deviation from pre-established rod sequence can result in potentially high control rod worth
3. By staying in sequence during rod withdrawal and insertion, it establishes optimum patterns with minimum uncertainties and problems

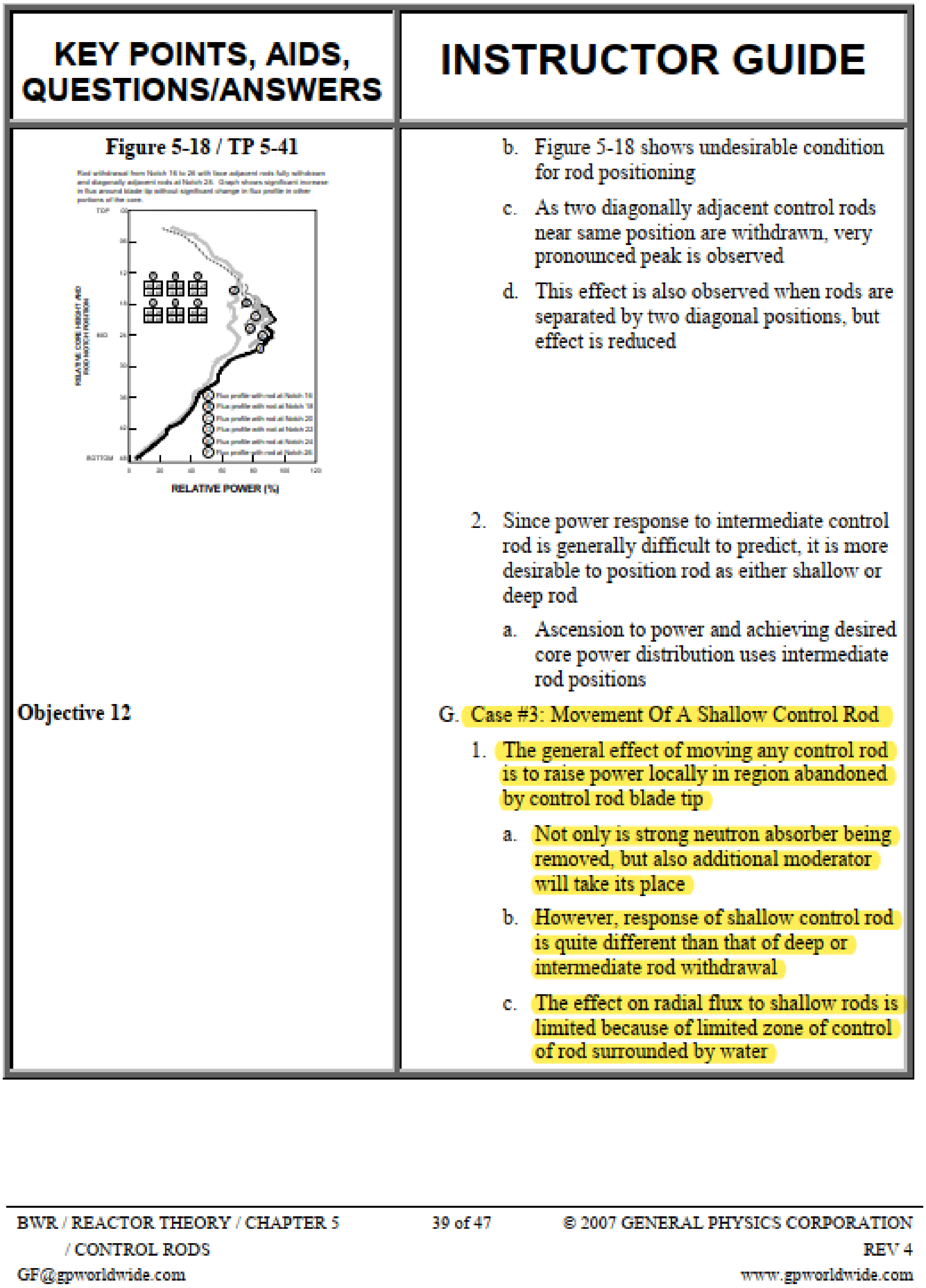
Objective 11

Figure 5-14 / TP 5-37

C. Some terminology must be defined prior to discussing effect of moving control rod

1. A *deep control rod* is any control rod inserted greater than two thirds into core, Notch 00 through 16 (see Figure 5-14)
2. A *shallow control rod* is any control rod inserted less than one third into core, Notch 34 through 48
3. An *intermediate control rod* is any control rod between deep and shallow control rod (Notch 18 through 32)

D. On following pages, we discuss three cases of rod withdrawal



KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Figure 5-18 / TP 5-41	b. Figure 5-18 shows undesirable condition for rod positioning c. As two diagonally adjacent control rods near same position are withdrawn, very pronounced peak is observed d. This effect is also observed when rods are separated by two diagonal positions, but effect is reduced
	2. Since power response to intermediate control rod is generally difficult to predict, it is more desirable to position rod as either shallow or deep rod a. Ascension to power and achieving desired core power distribution uses intermediate rod positions
Objective 12	G. Case #3: Movement Of A Shallow Control Rod 1. The general effect of moving any control rod is to raise power locally in region abandoned by control rod blade tip a. Not only is strong neutron absorber being removed, but also additional moderator will take its place b. However, response of shallow control rod is quite different than that of deep or intermediate rod withdrawal c. The effect on radial flux to shallow rods is limited because of limited zone of control of rod surrounded by water

KEY POINTS, AIDS, QUESTIONS/ANSWERS

Figure 5-19 / TP 5-42

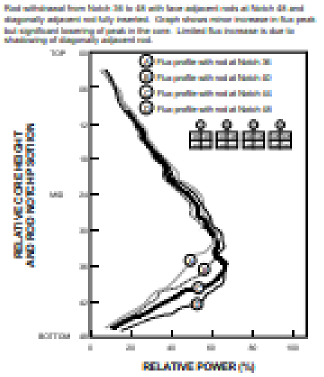
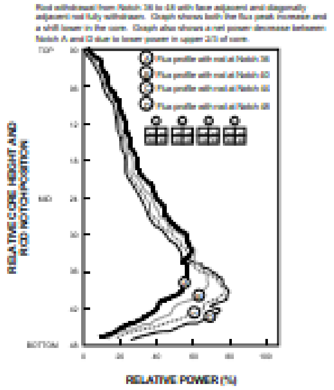


Figure 5-20 / TP 5-43



INSTRUCTOR GUIDE

d. This makes effect on neutron flux relatively large in localized area only

e. Therefore, shallow rods generally change power shape (axial flux) while affecting average core power (radial flux) very little

f. Another term for shallow control rods is *shaping rods*

2. In Figure 5-20, shallow rod exists in area where all adjacent rods are withdrawn

   a. When rod is withdrawn, it results in large increase in flux peak in conjunction with moving it lower in core

   b. The increased power from shallow rod movement will generate more steam bubbles (voids)

   c. These steam bubbles move upward through rest of fuel bundle

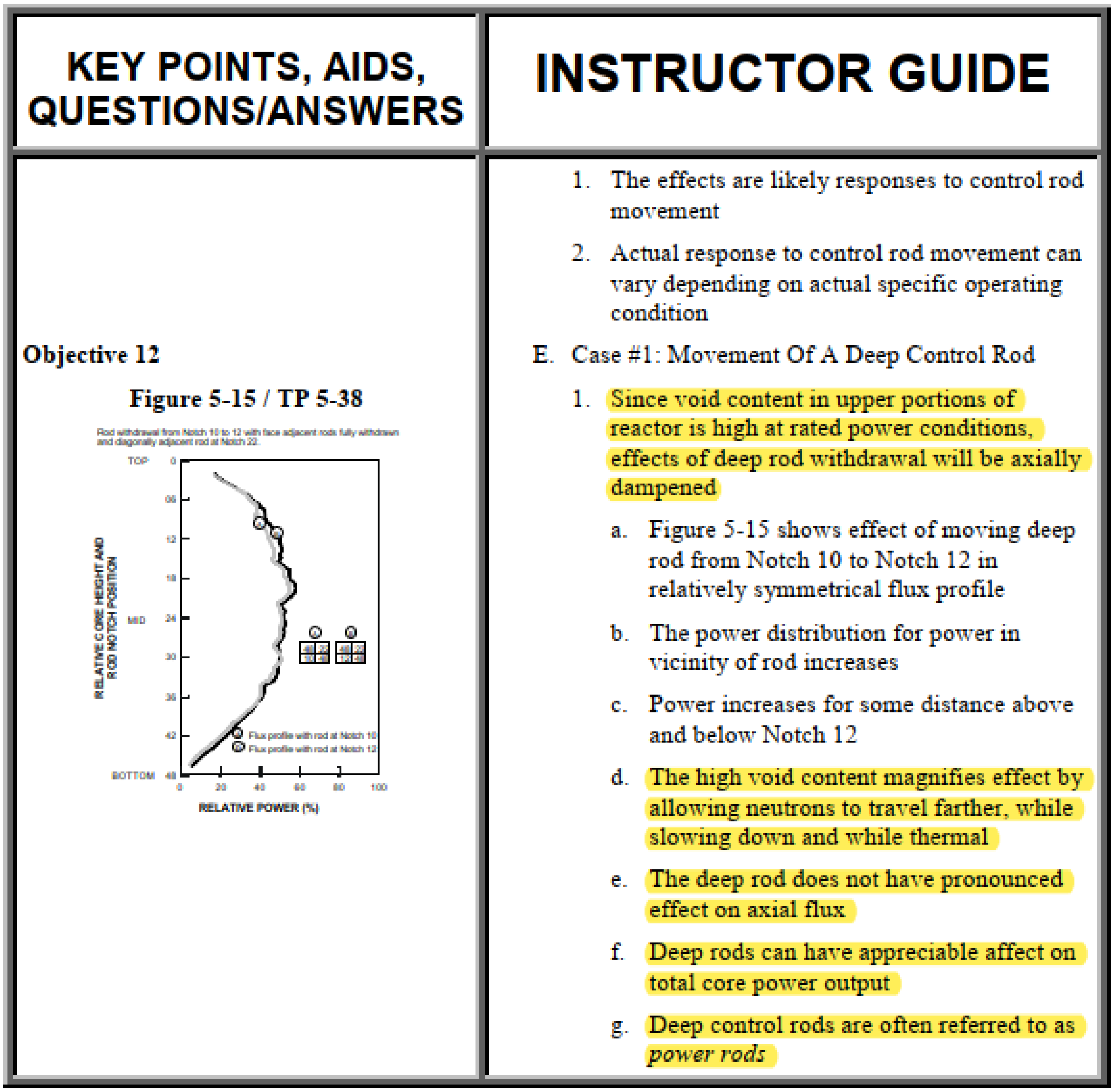
   d. The increase in void fraction in top of fuel bundle generally decreases power in that region

   e. Overall bundle power may increase slightly, remain same, or may even decrease, depending on relative magnitude of power increase in bottom of fuel bundle

   f. Total bundle power response is also dependent on position of nearby control rods and their shadowing effects

   g. It is not uncommon for one or two notch withdrawal of shallow rod to result in net power decrease, called *reverse power effect*

Supports Distractors A(2), C(2)



KEY POINTS, AIDS, QUESTIONS/ANSWERS

INSTRUCTOR GUIDE

Objective 12

Figure 5-15 / TP 5-38

1. The effects are likely responses to control rod movement
2. Actual response to control rod movement can vary depending on actual specific operating condition

E. Case #1: Movement Of A Deep Control Rod

1. Since void content in upper portions of reactor is high at rated power conditions, effects of deep rod withdrawal will be axially dampened
   a. Figure 5-15 shows effect of moving deep rod from Notch 10 to Notch 12 in relatively symmetrical flux profile
   b. The power distribution for power in vicinity of rod increases
   c. Power increases for some distance above and below Notch 12
   d. The high void content magnifies effect by allowing neutrons to travel farther, while slowing down and while thermal
   e. The deep rod does not have pronounced effect on axial flux
   f. Deep rods can have appreciable affect on total core power output
   g. Deep control rods are often referred to as *power rods*

Examination Outline Cross-reference:	Level	RO	SRO
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	Tier #	------	1
AA2.09** **(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 #	295001AA2.09	
- Reactor Pressure	Importance Rating	------	3.4

Proposed Question: **# 76**

Following a failure of a Jet Pump, Reactor Pressure will ____**(1)**____.

In accordance with Tech Spec Bases, the OPERABILITY of Jet Pumps is to allow reflooding of the core to ____**(2)**____ Reactor Water Level following a complete break of a Recirculation Pump suction pipe.

A. (1) lower
(2) (-) 162 inches

B. (1) lower
(2) (-) 215 inches

C. (1) rise
(2) (-) 162 inches

D. (1) rise
(2) (-) 215 inches

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). Second part is incorrect but plausible in that (-) 162 inches Reactor Water Level equals the Top of Active active fuel region in the core.

B **CORRECT**: (*See attached*) In accordance with 2-AOI-68-2, Jet Pump Failure, 2-FR-68-50, TOTAL CORE FLOW indication on Panel 2-9-5 will rise, however, MWE, Reactor Power and Reactor Pressure will lower. This is also illustrated on the Simulator Insight Trend table caption for both Reactor Power (APRM) and Reactor Pressure. For second part, in accordance with Tech Spec Bases 3.4.2, the OPERABILITY of all Jet Pumps is required since the structural failure of any Jet Pump could cause significant degradation in the ability of Jet Pumps to allow reflooding to the actual two-thirds (2/3) Core Height ((-) 215 inches Reactor Water Level) during a LOCA. With a no flow condition, 2/3 of the core will remain covered. Additionally, this supports core cooling by spray cooling with Core Spray following a DBA.

C INCORRECT: The first part is incorrect but plausible in that in accordance with 2-AOI-68-2, Jet Pump Failure, the ACTUAL Core Flow of coolant through the core will lower as calculated from core plate differential pressure. The flow will be diverted to the lower pressure area of the annulus where Jet Pump flow indication will rise due to these changing system dynamics. This concept can be confused by the candidate to assume that Reactor Power and Pressure will rise. 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*).

SRO Level Justification: Tests the candidate's ability to determine Reactor Power and Pressure as it relates to partial or complete loss of forced Core Flow circulation that would occur following a Jet Pump mechanical failure. Additionally, Technical Specification Bases knowledge is required from Memory. 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 both Tech Spec Bases and AOI 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 Tech Spec Bases 3.4.2, Rev. 0 (Attach if not previously provided)
2-AOI-68-2, Rev. 14
OPL171.003, Rev. 26
OPL171.202, Rev. 12U1

Proposed references to be provided to applicants during examination: **NONE**

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

Question Source:	Bank #		
	Modified Bank #	BFN 2104 #76	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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: **# 76**

Following a failure of a Jet Pump, **INDICATED** Core Flow will ____**(1)**____ in accordance with 2-AOI-68-2, Jet Pump Failure.

In accordance with Tech Spec Bases, the OPERABILITY of Jet Pumps is to allow reflooding of the core to ____**(2)**____ Reactor Water Level following a complete break of a Recirculation Pump suction pipe.

A. (1) rise
(2) (-) 162 inches

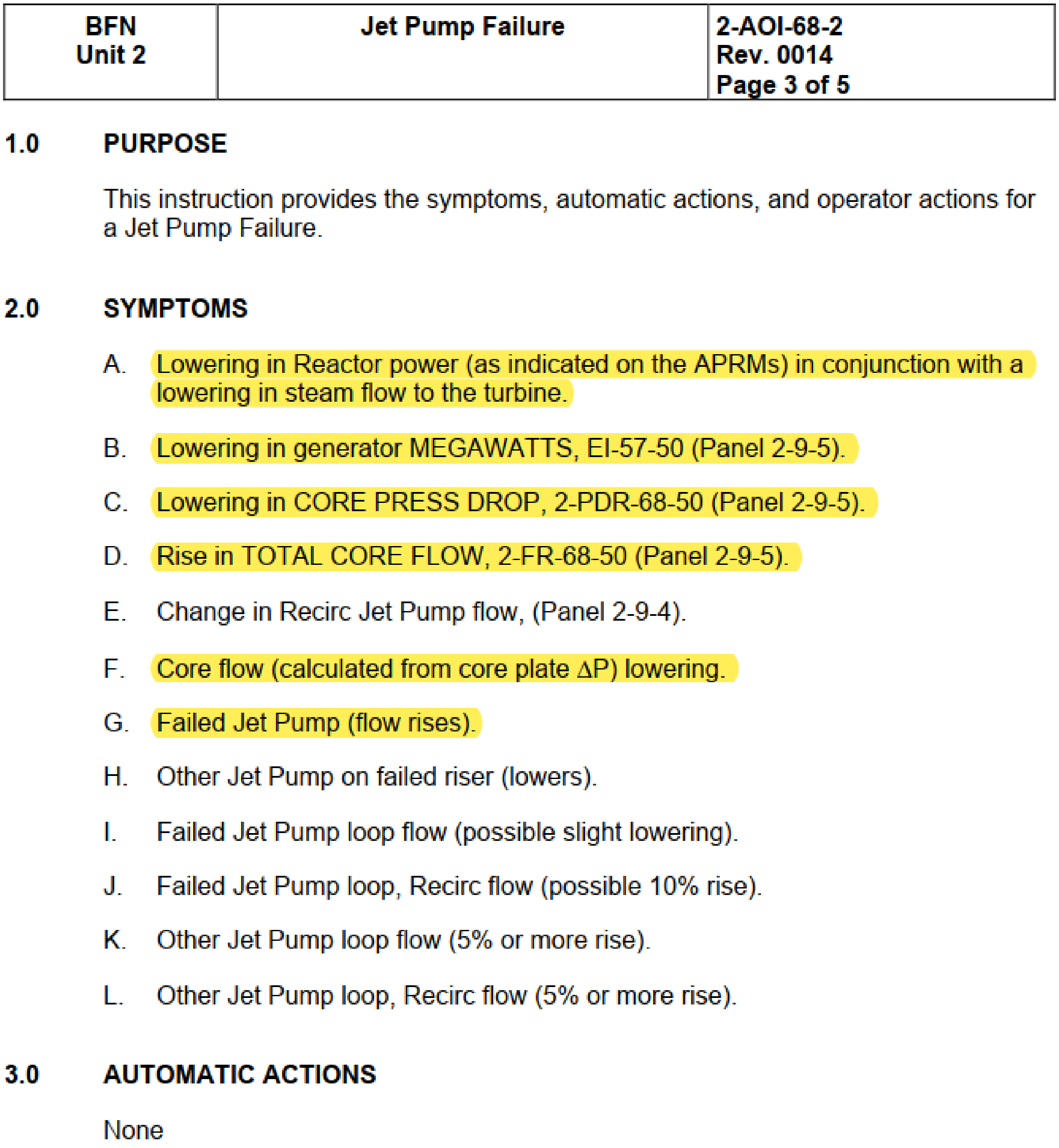
B. (1) rise
(2) (-) 215 inches

C. (1) lower
(2) (-) 162 inches

D. (1) lower
(2) (-) 215 inches

Proposed Answer: **B**

Excerpt from 2-AOI-68-2:



BFN Unit 2	**Jet Pump Failure**	**2-AOI-68-2 Rev. 0014 Page 3 of 5**

1.0 PURPOSE

This instruction provides the symptoms, automatic actions, and operator actions for a Jet Pump Failure.

2.0 SYMPTOMS

A. Lowering in Reactor power (as indicated on the APRMs) in conjunction with a lowering in steam flow to the turbine.

B. Lowering in generator MEGAWATTS, EI-57-50 (Panel 2-9-5).

C. Lowering in CORE PRESS DROP, 2-PDR-68-50 (Panel 2-9-5).

D. Rise in TOTAL CORE FLOW, 2-FR-68-50 (Panel 2-9-5).

E. Change in Recirc Jet Pump flow, (Panel 2-9-4).

F. Core flow (calculated from core plate ΔP) lowering.

G. Failed Jet Pump (flow rises).

H. Other Jet Pump on failed riser (lowers).

I. Failed Jet Pump loop flow (possible slight lowering).

J. Failed Jet Pump loop, Recirc flow (possible 10% rise).

K. Other Jet Pump loop flow (5% or more rise).

L. Other Jet Pump loop, Recirc flow (5% or more rise).

3.0 AUTOMATIC ACTIONS

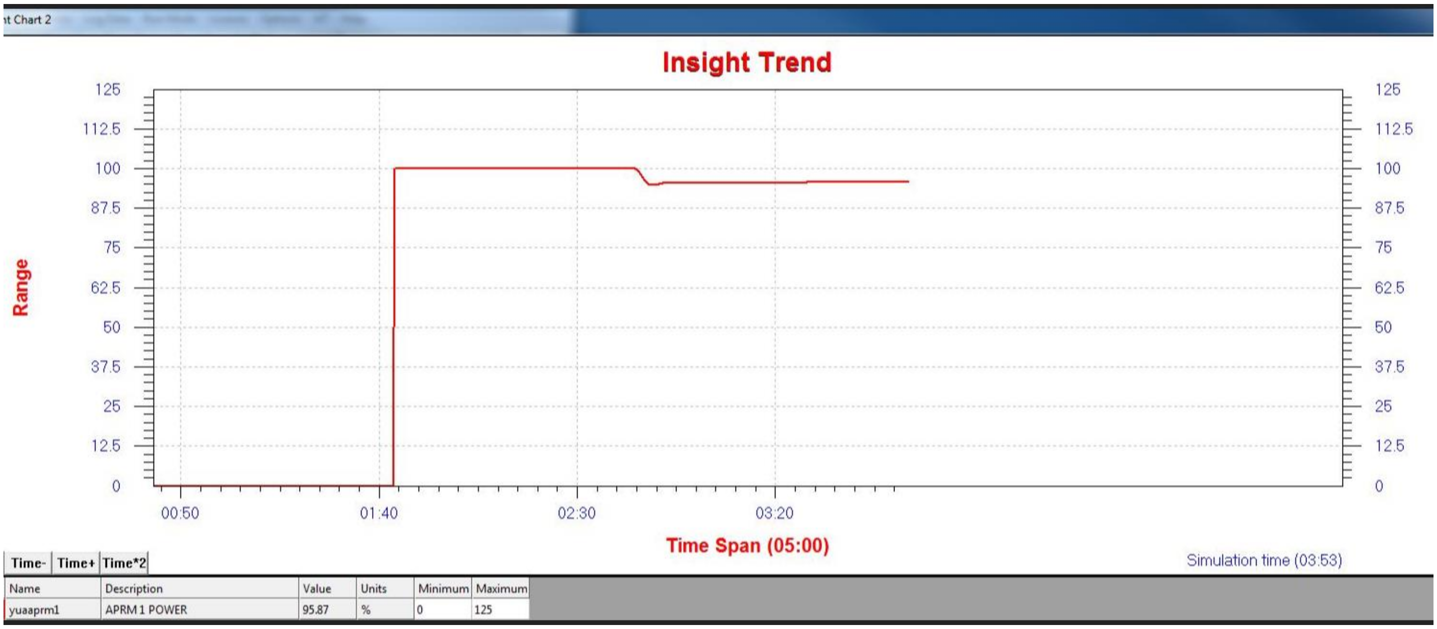
None

Excerpts from Simulator Insight Trend captions:

- Reactor Pressure

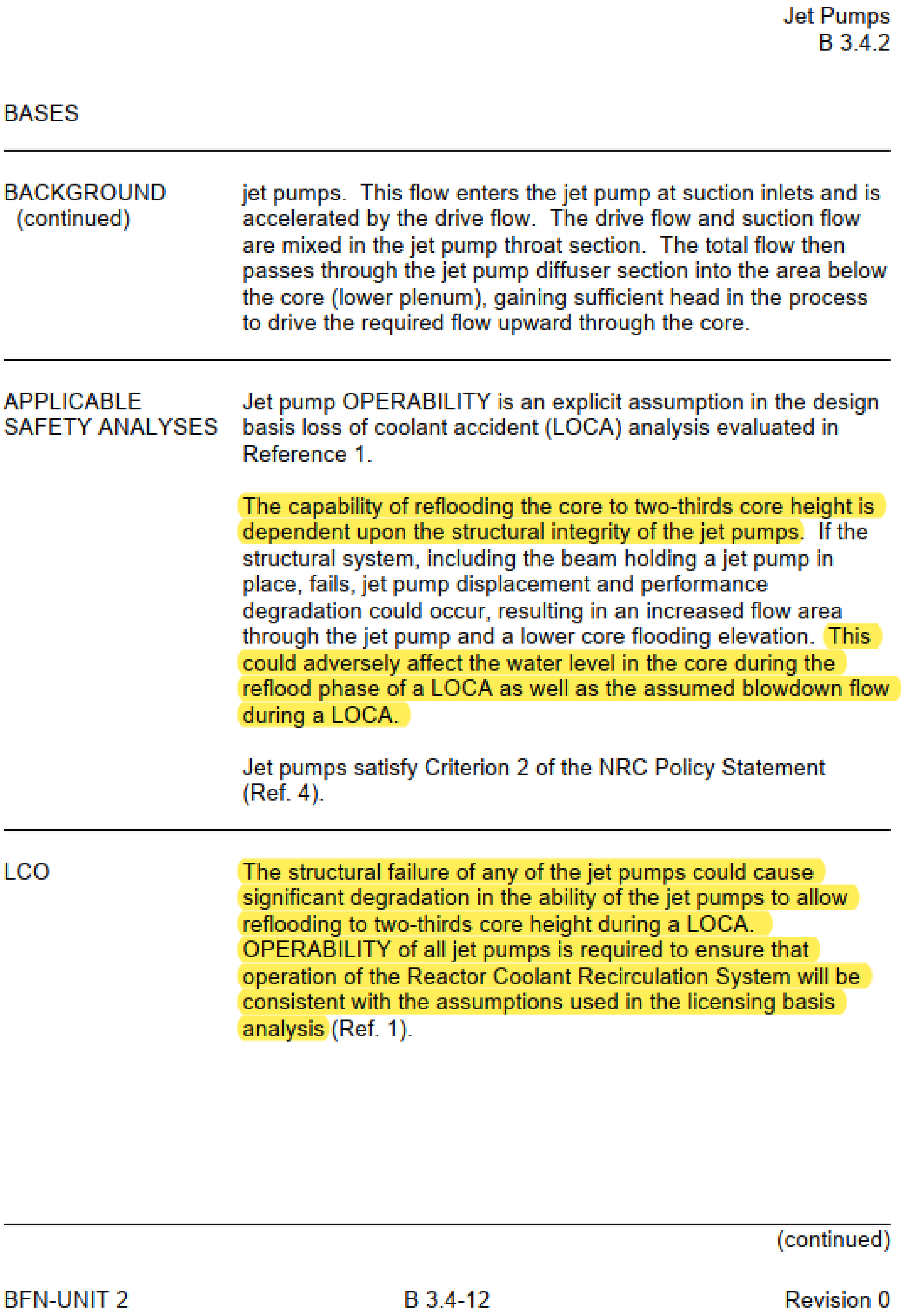
	Name	Description	Value	Units	Minimum	Maximum
🟥	zaonr0353r1	reactor pressure	1033	psig	1000	1045

- Reactor Power



Name	Description	Value	Units	Minimum	Maximum
yuaaprm1	APRM 1 POWER	95.87	%	0	125

Excerpt from Unit 2 Tech Spec Bases 3.4.2:



Jet Pumps
B 3.4.2

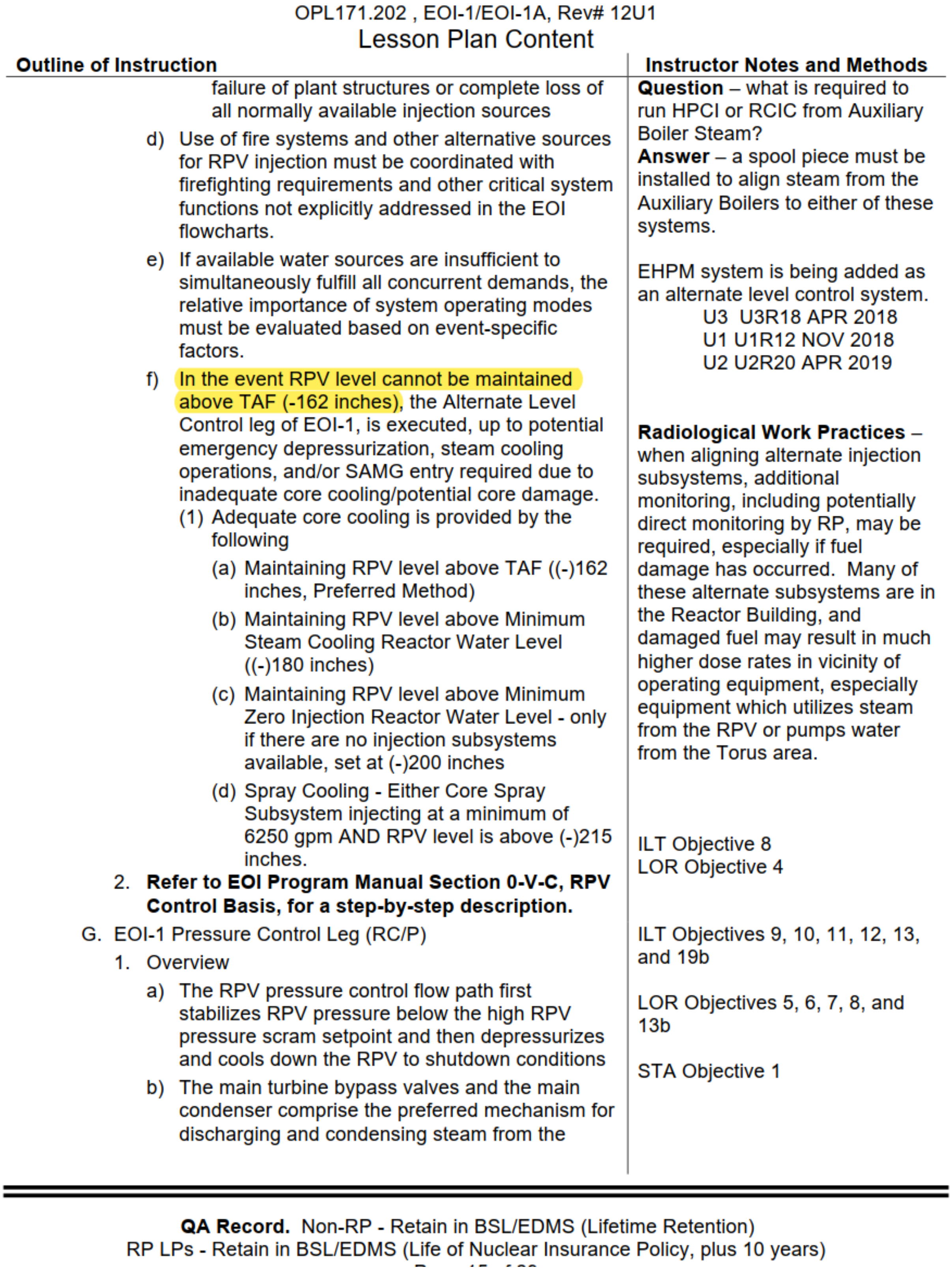
BASES

BACKGROUND (continued)	jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the drive flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.
APPLICABLE SAFETY ANALYSES	Jet pump OPERABILITY is an explicit assumption in the design basis loss of coolant accident (LOCA) analysis evaluated in Reference 1. The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If the structural system, including the beam holding a jet pump in place, fails, jet pump displacement and performance degradation could occur, resulting in an increased flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA. Jet pumps satisfy Criterion 2 of the NRC Policy Statement (Ref. 4).
LCO	The structural failure of any of the jet pumps could cause significant degradation in the ability of the jet pumps to allow reflooding to two-thirds core height during a LOCA. OPERABILITY of all jet pumps is required to ensure that operation of the Reactor Coolant Recirculation System will be consistent with the assumptions used in the licensing basis analysis (Ref. 1).

(continued)

BFN-UNIT 2 B 3.4-12 Revision 0

Excerpt from OPL171.202: Supports Distractors A(2), C(2)



OPL171.202 , EOI-1/EOI-1A, Rev# 12U1

Lesson Plan Content

Outline of Instruction

failure of plant structures or complete loss of all normally available injection sources

d) Use of fire systems and other alternative sources for RPV injection must be coordinated with firefighting requirements and other critical system functions not explicitly addressed in the EOI flowcharts.

e) If available water sources are insufficient to simultaneously fulfill all concurrent demands, the relative importance of system operating modes must be evaluated based on event-specific factors.

f) In the event RPV level cannot be maintained above TAF (-162 inches), the Alternate Level Control leg of EOI-1, is executed, up to potential emergency depressurization, steam cooling operations, and/or SAMG entry required due to inadequate core cooling/potential core damage.

  (1) Adequate core cooling is provided by the following

    (a) Maintaining RPV level above TAF ((-)162 inches, Preferred Method)

    (b) Maintaining RPV level above Minimum Steam Cooling Reactor Water Level ((-)180 inches)

    (c) Maintaining RPV level above Minimum Zero Injection Reactor Water Level - only if there are no injection subsystems available, set at (-)200 inches

    (d) Spray Cooling - Either Core Spray Subsystem injecting at a minimum of 6250 gpm AND RPV level is above (-)215 inches.

2. **Refer to EOI Program Manual Section 0-V-C, RPV Control Basis, for a step-by-step description.**

G. EOI-1 Pressure Control Leg (RC/P)

1. Overview

a) The RPV pressure control flow path first stabilizes RPV pressure below the high RPV pressure scram setpoint and then depressurizes and cools down the RPV to shutdown conditions

b) The main turbine bypass valves and the main condenser comprise the preferred mechanism for discharging and condensing steam from the

Instructor Notes and Methods

Question – what is required to run HPCI or RCIC from Auxiliary Boiler Steam?

Answer – a spool piece must be installed to align steam from the Auxiliary Boilers to either of these systems.

EHPM system is being added as an alternate level control system.

U3 U3R18 APR 2018
U1 U1R12 NOV 2018
U2 U2R20 APR 2019

Radiological Work Practices – when aligning alternate injection subsystems, additional monitoring, including potentially direct monitoring by RP, may be required, especially if fuel damage has occurred. Many of these alternate subsystems are in the Reactor Building, and damaged fuel may result in much higher dose rates in vicinity of operating equipment, especially equipment which utilizes steam from the RPV or pumps water from the Torus area.

ILT Objective 8
LOR Objective 4

ILT Objectives 9, 10, 11, 12, 13, and 19b

LOR Objectives 5, 6, 7, 8, and 13b

STA Objective 1

QA Record. Non-RP - Retain in BSL/EDMS (Lifetime Retention)
RP LPs - Retain in BSL/EDMS (Life of Nuclear Insurance Policy, plus 10 years)

Excerpt from OPL171.003:

OPL171.003 REACTOR VESSEL PROCESS INSTRUMENTATION REV 26

a) Set low enough to prevent spurious operation.

b) Set high enough to allow time to activate the low pressure ECCS so that no fuel melting will occur. Long-term cooling will be possible with no fuel meltdown.

3) Post Accident Flood Range Instrument (-268” to +32”)
   a) LT-3-52/LT-3-62
      1) Two-thirds core covered permissive interlock at - 183 inches Level 0

      2) Allows RHR system to be used for containment spray. Requires additional operator action if level is below -183".The level -215” is the actual 2/3 core covered level and -183” is the permissive setpoint.

g. Steam flow effect on reactor water level

Obj. ILT-8/ LOR-4

1) Steam flowing through the dryers is forced to change direction several times, resulting in a pressure drop across the dryers

2) At 100 percent steam flow, the pressure drop is 7 inches of water. On a reactor scram, this ΔP “goes away”, due to the void collapse, causing lower back pressure on the recirc pumps and jet pumps. Water from the annulus is relocated to the core area, causing sensed (and indicated) water level (in the annulus) to drop.

3) Therefore, at 100 percent steam flow, P1 is 7 inches of water less than P2.

4) The level outside the dryer skirt (down comer region) is 7 inches higher than inside the skirt.

5) Since the vessel level instruments compare the reference column height to the down comer (variable column) height, setpoints are adjusted to compensate for this error.

6) The water level inside the dryer skirt is slightly dome shaped.
   a) Moisture separator drains must flow to the outside (down comer) region to return to the core.

   b) In order for drains from the interior separators to flow outward, a hydraulic gradient is required.

   c) At 100 percent power, the "top" of the dome is 4 inches higher than the "outside," providing the hydraulic gradient. The degree of hydraulic gradient is variable, ranging from 0 inches at 0 percent power to 4 inches at 100 % power.

Examination Outline Cross-reference:	Level	RO	SRO
295021 (APE 21) Loss of Shutdown Cooling / 4	Tier #	------	1
G2.4.41 (10CFR 55.43.5 - SRO Only)	Group #	------	1
Knowledge of the emergency action level thresholds and Classifications	K/A #	295021G2.4.41	
	Importance Rating	------	4.6

Proposed Question: **# 77**

Unit 1 and 2 are in MODE 1, 100% RTP. Unit 3 is in MODE 4, when the following conditions occur on **Unit 3**:

At 0900:

- Shutdown Cooling isolates due to a fault

At 0925:

- Moderator temperature is 213 °F

At 1100:

- Maintenance reports the fault condition has been repaired and RHR can be placed in service
- Reactor Water Level maintained in the required band throughout the event

Given the conditions above and in accordance with EPIP-1, Emergency Classification Procedure, the **HIGHEST** required Emergency Classification to report will be a/an ___**(1)**___ and from the time of event declaration, NRC Notification must **NOT** exceed ___**(2)**___.

Note: Notification of Unusual Event (NOUE)

[REFERENCE PROVIDED]

A. (1) NOUE
(2) 15 minutes

B. (1) NOUE
(2) 60 minutes

C. (1) ALERT
(2) 15 minutes

D. (1) ALERT
(2) 60 minutes

Proposed Answer: **D**

Explanation (Optional):	A	INCORRECT: First part is incorrect but plausible in that a loss of Shutdown Cooling with an unplanned MODE change is a NOUE. There are other variables that are required to occur to upgrade to an ALERT. The candidate must first know the status of Secondary Containment. While it is not currently required for Unit 3, Secondary Containment at BFN is shared among the Units. Since Unit 1 and Unit 2 are given as in MODE 1, Secondary Containment is required to be OPERABLE, thereby intact. Additionally, the duration is also required for loss of Shutdown Cooling prior to reestablishing cooling. Second part is incorrect but plausible in that 15 minutes is the required time for both Classification of an event and the required time for notification to the State.
	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**: In accordance with EPIP-1, Attachment -1, Cold Initiating Conditions – MODE 4 -5 - Defueled, since an unplanned moderator temperature rose to greater than 212 °F, at least a NOUE was warranted. However, an ALERT is the HIGHEST required Emergency Classification since Shutdown Cooling was lost causing a heat-up duration for greater than 212 °F for greater than 60 minutes with Secondary Containment intact as indicated on Table C2. For second part, in accordance with EPIP-3, ALERT, notification of the NRC is required to completed as soon as possible, not to exceed 60 minutes from Classification Declaration.

SRO Level Justification: Tests the candidate's knowledge and ability to diagnose and implement Emergency Plan Implementing Procedures (EPIPs) related to a loss of Shutdown Cooling event. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(2) diagnosis that leads to selection of the procedures that should be used to respond to the evolution.

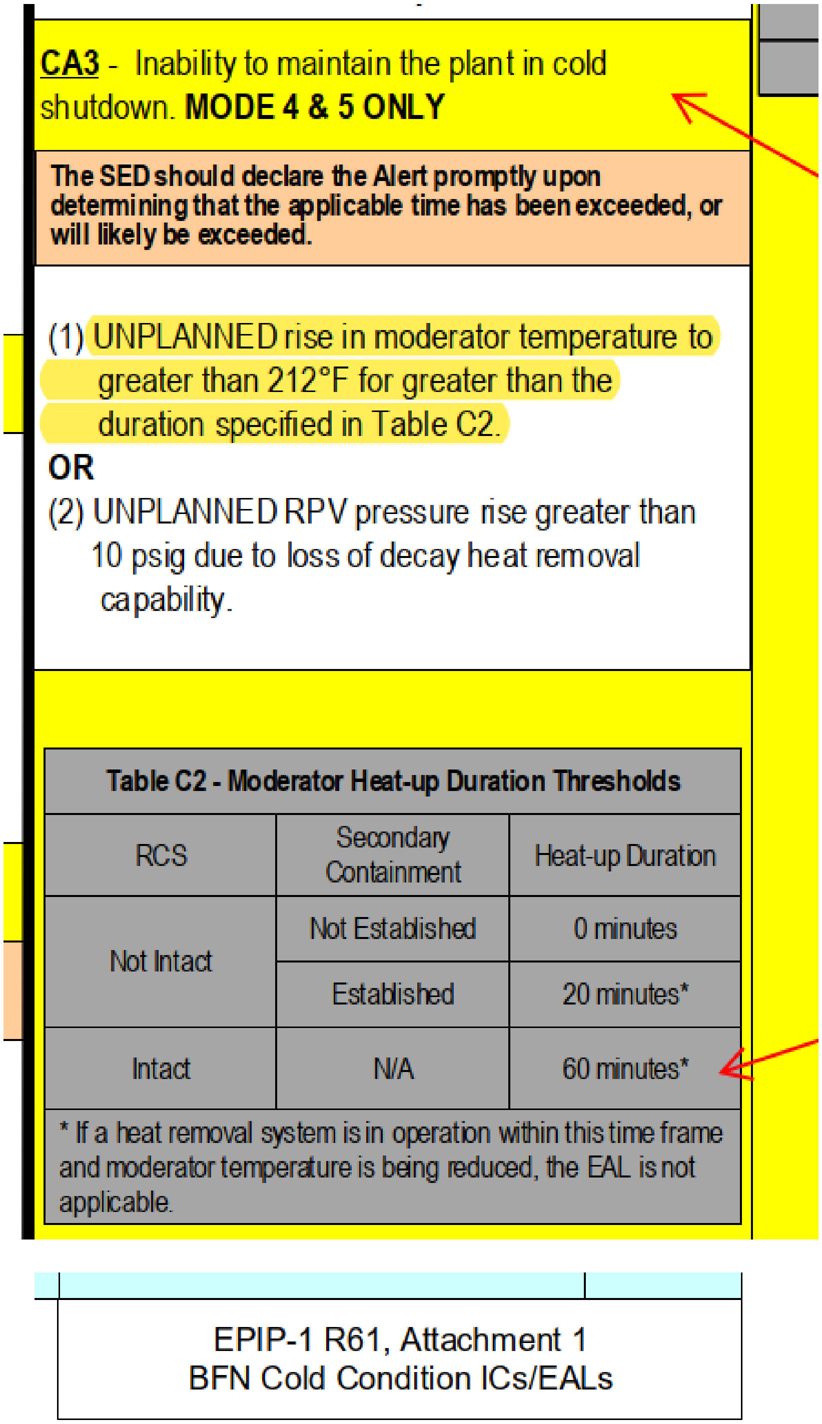
Technical Reference(s):	EPIP-1, Rev. 61	(Attach if not previously provided)
	EPIP-3, Rev. 44	

Proposed references to be provided to applicants during examination: **EPIP-1, Attachment -1, Cold Initiating Conditions – MODE 4 -5 - Defueled**

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, Attachment -1, Cold Initiating Conditions – MODE 4 -5 - Defueled:



CA3 - Inability to maintain the plant in cold shutdown. **MODE 4 & 5 ONLY**

The SED should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

(1) UNPLANNED rise in moderator temperature to greater than 212°F for greater than the duration specified in Table C2.

OR

(2) UNPLANNED RPV pressure rise greater than 10 psig due to loss of decay heat removal capability.

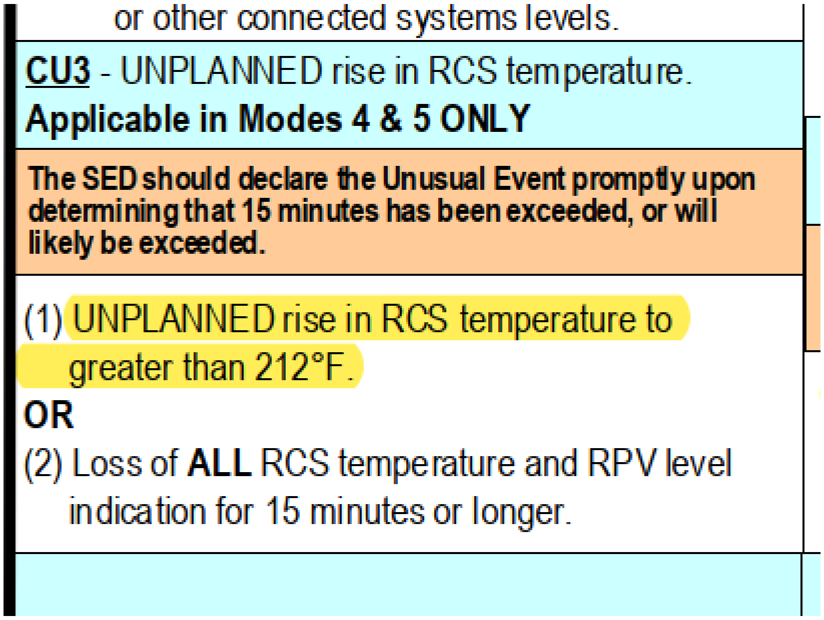
Table C2 - Moderator Heat-up Duration Thresholds

RCS	Secondary Containment	Heat-up Duration
Not Intact	Not Established	0 minutes
	Established	20 minutes*
Intact	N/A	60 minutes*

* If a heat removal system is in operation within this time frame and moderator temperature is being reduced, the EAL is not applicable.

EPIP-1 R61, Attachment 1
BFN Cold Condition ICs/EALs

Supports Distractors A(1), B(1):



or other connected systems levels.

CU3 - UNPLANNED rise in RCS temperature.

Applicable in Modes 4 & 5 ONLY

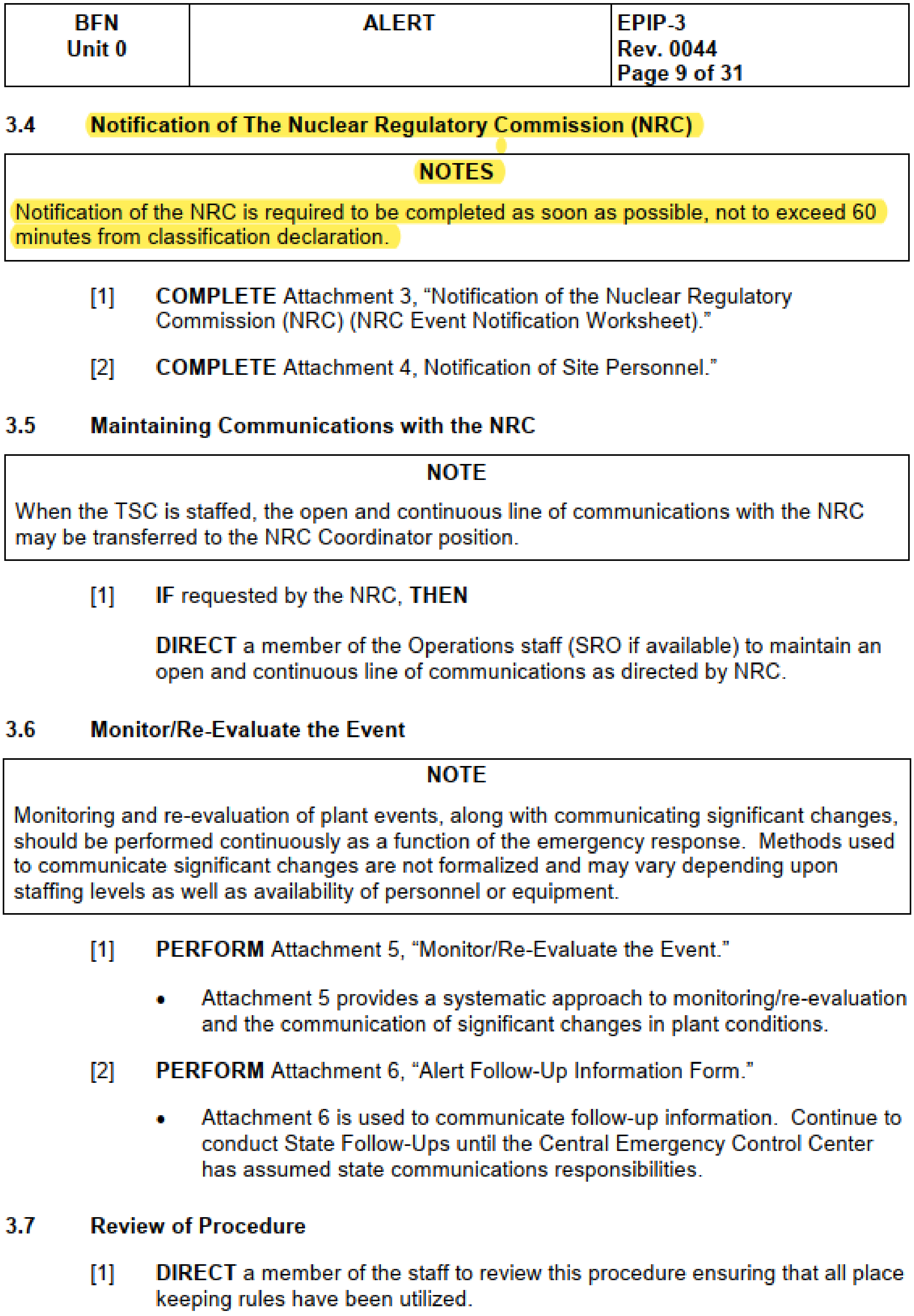
The SED should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) UNPLANNED rise in RCS temperature to greater than 212°F.

OR

(2) Loss of **ALL** RCS temperature and RPV level indication for 15 minutes or longer.

Excerpts from EPIP-3:



BFN **Unit 0**	**ALERT**	**EPIP-3** **Rev. 0044** **Page 9 of 31**

3.4 Notification of The Nuclear Regulatory Commission (NRC)

NOTES

Notification of the NRC is required to be completed as soon as possible, not to exceed 60 minutes from classification declaration.

[1] **COMPLETE** Attachment 3, “Notification of the Nuclear Regulatory Commission (NRC) (NRC Event Notification Worksheet).”

[2] **COMPLETE** Attachment 4, 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

NOTE

Monitoring and re-evaluation of plant events, along with communicating significant changes, should be performed continuously as a function of the emergency response. Methods used to communicate significant changes are not formalized and may vary depending upon staffing levels as well as availability of personnel or equipment.

[1] **PERFORM** Attachment 5, “Monitor/Re-Evaluate the Event.”

- Attachment 5 provides a systematic approach to monitoring/re-evaluation and the communication of significant changes in plant conditions.

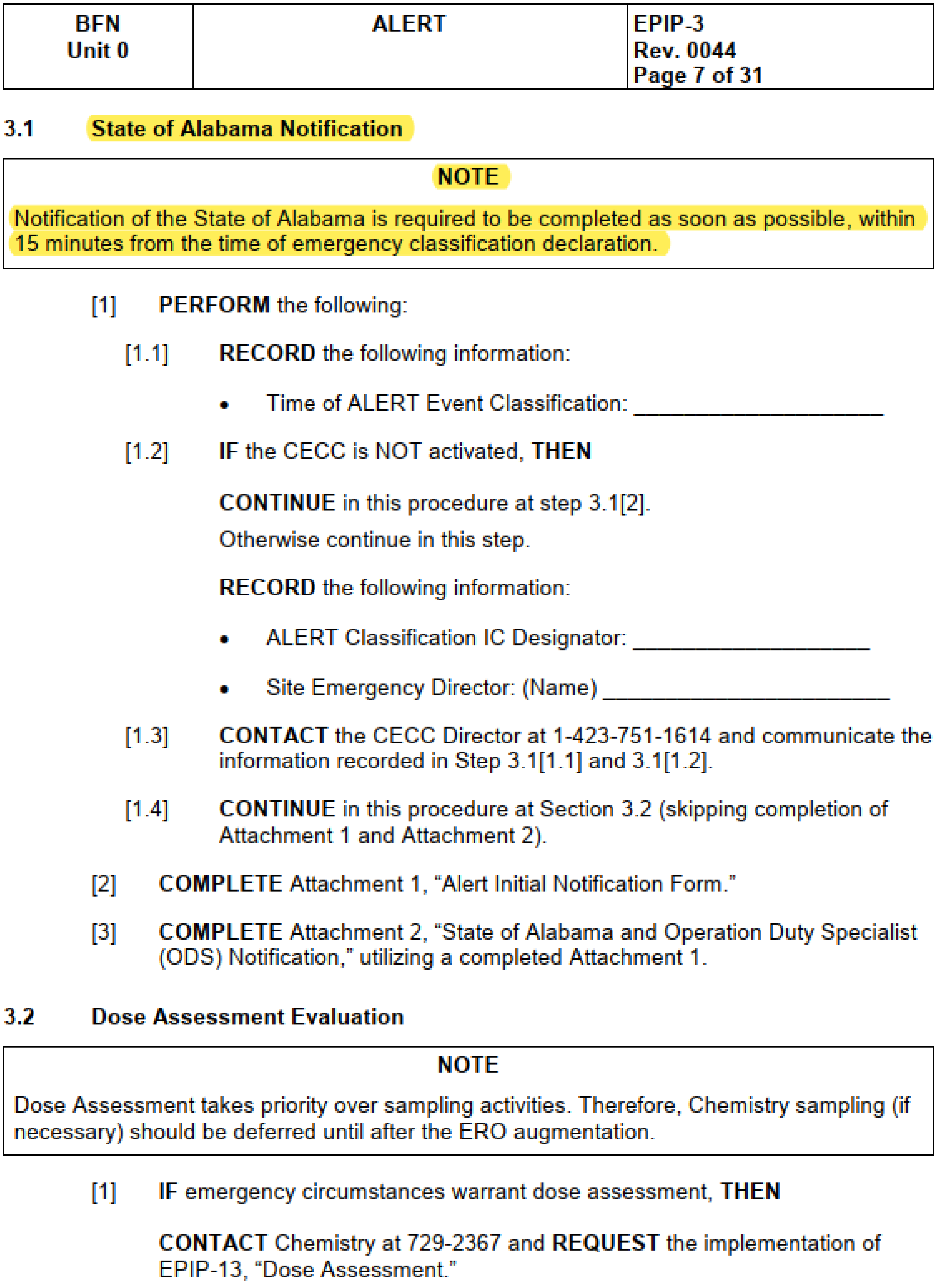
[2] **PERFORM** Attachment 6, “Alert Follow-Up Information Form.”

- Attachment 6 is used to communicate follow-up information. Continue to conduct State Follow-Ups until the Central Emergency Control Center has assumed state communications responsibilities.

3.7 Review of Procedure

[1] **DIRECT** a member of the staff to review this procedure ensuring that all place keeping rules have been utilized.

Supports Distractors A(2), C(2):



BFN **Unit 0**	**ALERT**	**EPIP-3** **Rev. 0044** **Page 7 of 31**

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 Attachment 1 and Attachment 2).

[2] **COMPLETE** Attachment 1, “Alert Initial Notification Form.”

[3] **COMPLETE** Attachment 2, “State of Alabama and Operation Duty Specialist (ODS) Notification,” utilizing a completed Attachment 1.

3.2 Dose Assessment Evaluation

NOTE

Dose Assessment takes priority over sampling activities. Therefore, Chemistry sampling (if necessary) should be deferred until after the ERO augmentation.

[1] **IF** emergency circumstances warrant dose assessment, **THEN**

CONTACT Chemistry at 729-2367 and **REQUEST** the implementation of EPIP-13, “Dose Assessment.”

Examination Outline Cross-reference:	Level	RO	SRO
295023 (APE 23) Refueling Accidents / 8	Tier #	------	1
AA2.03 (10CFR 55.43.5 – SRO Only)	Group #	------	1
Ability to determine and/or interpret the following as they apply to Refueling Accidents	K/A #	295023AA2.03	
- Airborne Contamination levels	Importance Rating	------	3.2

Proposed Question: **# 78**

Unit 2 was in MODE 5 during a refueling outage, when the following conditions occurred:

At 0905:

- Operators commence fuel movements
- The first fuel bundle is dropped due to grapple mechanical failure

At 0915:

- 2-RM-90-1A, Fuel Pool Floor, alarming and indicating 126 mr/hr
- 2-RM-90-250A, Reactor, Turbine, Refuel Exhaust, alarming and indicating 4.7E+6 µCi/s

At 0917:

- Shift Manager (as the SED), makes an Emergency Plan Event Declaration

Given the conditions above and in accordance with EPIP-1, Emergency Classification Procedure, the **HIGHEST** required Emergency Classification to report is a/an **___(1)___** and the State of Alabama is required to be notified **NO** later than **___(2)___**.

Note: SED judgement shall **NOT** be used as a basis for Classification

[REFERENCE PROVIDED]

A. (1) ALERT
(2) 0932

B. (1) ALERT
(2) 0945

C. (1) NOUE
(2) 0932

D. (1) NOUE
(2) 0945

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached*) In accordance with EPIP-1, Attachment -1, Cold Initiating Conditions – MODE 4 -5 – Defueled, RA2 would be the appropriate IC given the indication of damaged fuel with a dropped fuel bundle. The given radiological alarms within themselves would not merit an ALERT. Fuel damage with the radiological alarms warrant the IC for ALERT - RA2. For second part, in accordance with EPIP-3, ALERT, the Notification of the State of Alabama is required to be completed as soon as possible, and within 15 minutes from the time of the Emergency Classification*.* The event was classified at 0917, therefore adding 15 minutes yields a time of no later than 0932.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible in that 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. Therefore, 30 minutes + 0915 results in the incorrect time of 0945.

C INCORRECT: First part is incorrect but plausible in that 2-90-RM-250 alarming can be declared as a NOUE. The candidate may not correlate that the given alarms in conjunction with damage to an irradiated fuel bundle has occurred. Therefore, the candidate would select RU1 in accordance with EPIP-1. 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 the implications of a given airborne contamination levels during a refueling accident. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(2) diagnosis that leads to selection of the procedures that should be used to respond to the evolution and
(3) the progression of an event.

Technical Reference(s): EPIP-1, Rev. 61 (Attach if not previously provided)
EPIP-3, Rev. 44

Proposed references to be provided to applicants during examination: **EPIP-1, Attachment 1, COLD INITIATING CONDTIONS-MODES 4-5-DEFUELED**

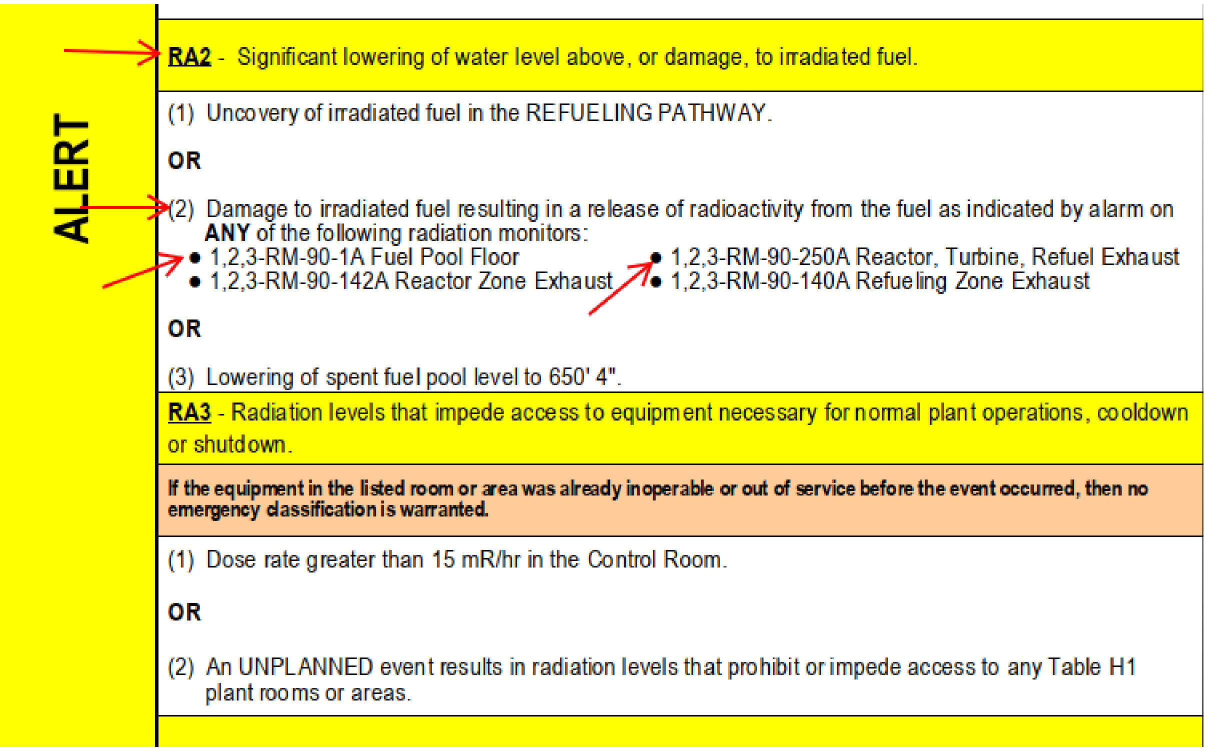
Learning Objective: OPL171.075 Obj. 2 (As available)

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**

Comments:

Excerpts from EPIP-1, Attachment -1, Cold Initiating Conditions – MODE 4 -5 - Defueled:



ALERT

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".

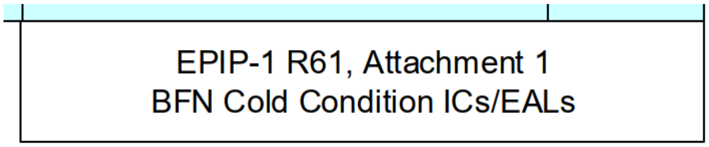
RA3 - Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.

If the equipment in the listed room or area was already inoperable or out of service before the event occurred, then no emergency classification is warranted.

(1) Dose rate greater than 15 mR/hr in the Control Room.

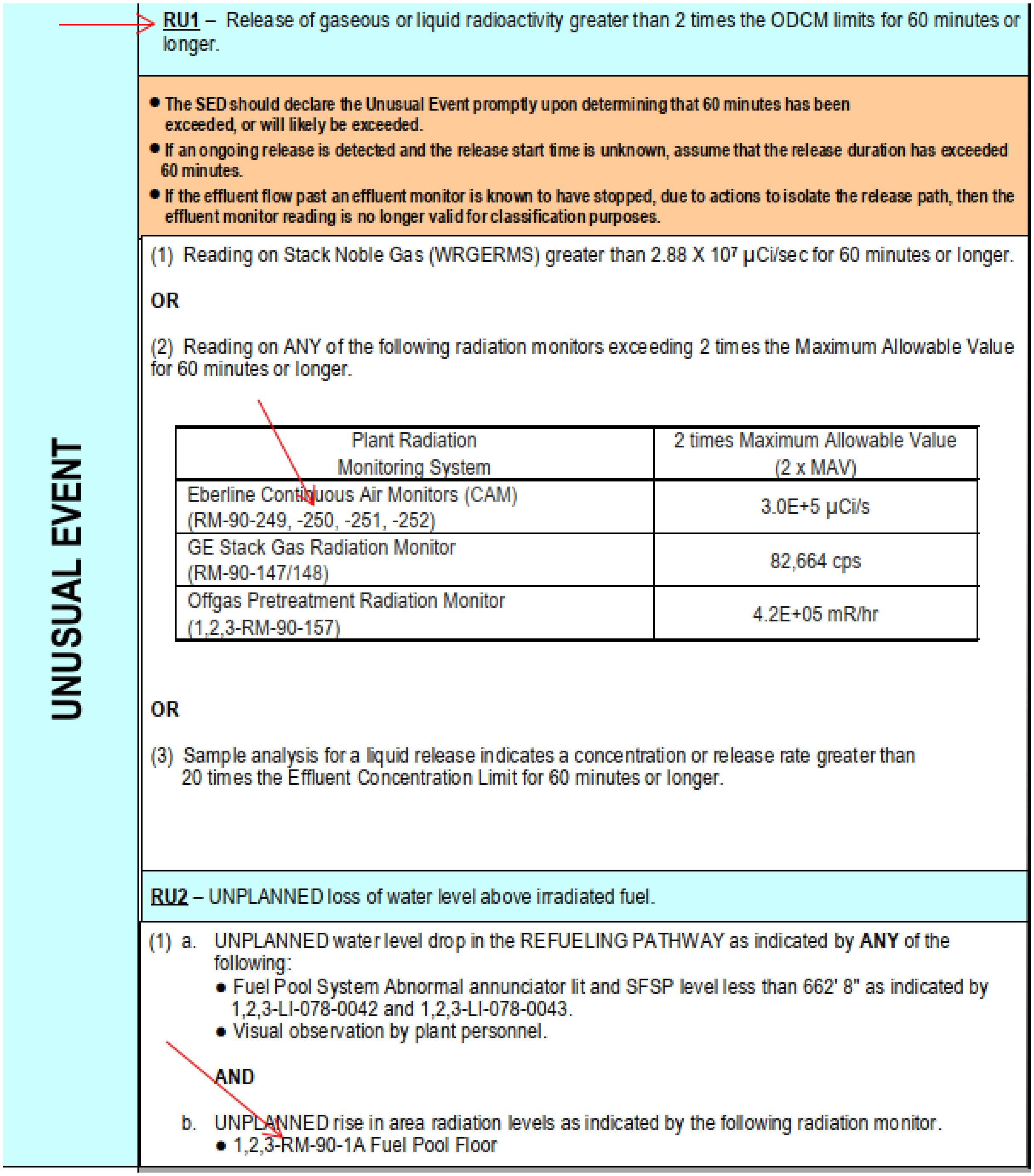
OR

(2) An UNPLANNED event results in radiation levels that prohibit or impede access to any Table H1 plant rooms or areas.



EPIP-1 R61, Attachment 1
BFN Cold Condition ICs/EALs

Supports Distractors C(1), D(1):



UNUSUAL EVENT

RU1 – Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

- **The SED should declare the Unusual Event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.**
- **If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes.**
- **If the effluent flow past an effluent monitor is known to have stopped, due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.**

(1) Reading on Stack Noble Gas (WRGERMS) greater than 2.88 X 10^7 µCi/sec for 60 minutes or longer.

OR

(2) Reading on ANY of the following radiation monitors exceeding 2 times the Maximum Allowable Value for 60 minutes or longer.

Plant Radiation Monitoring System	2 times Maximum Allowable Value (2 x MAV)
Eberline Continuous Air Monitors (CAM) (RM-90-249, -250, -251, -252)	3.0E+5 µCi/s
GE Stack Gas Radiation Monitor (RM-90-147/148)	82,664 cps
Offgas Pretreatment Radiation Monitor (1,2,3-RM-90-157)	4.2E+05 mR/hr

OR

(3) Sample analysis for a liquid release indicates a concentration or release rate greater than 20 times the Effluent Concentration Limit for 60 minutes or longer.

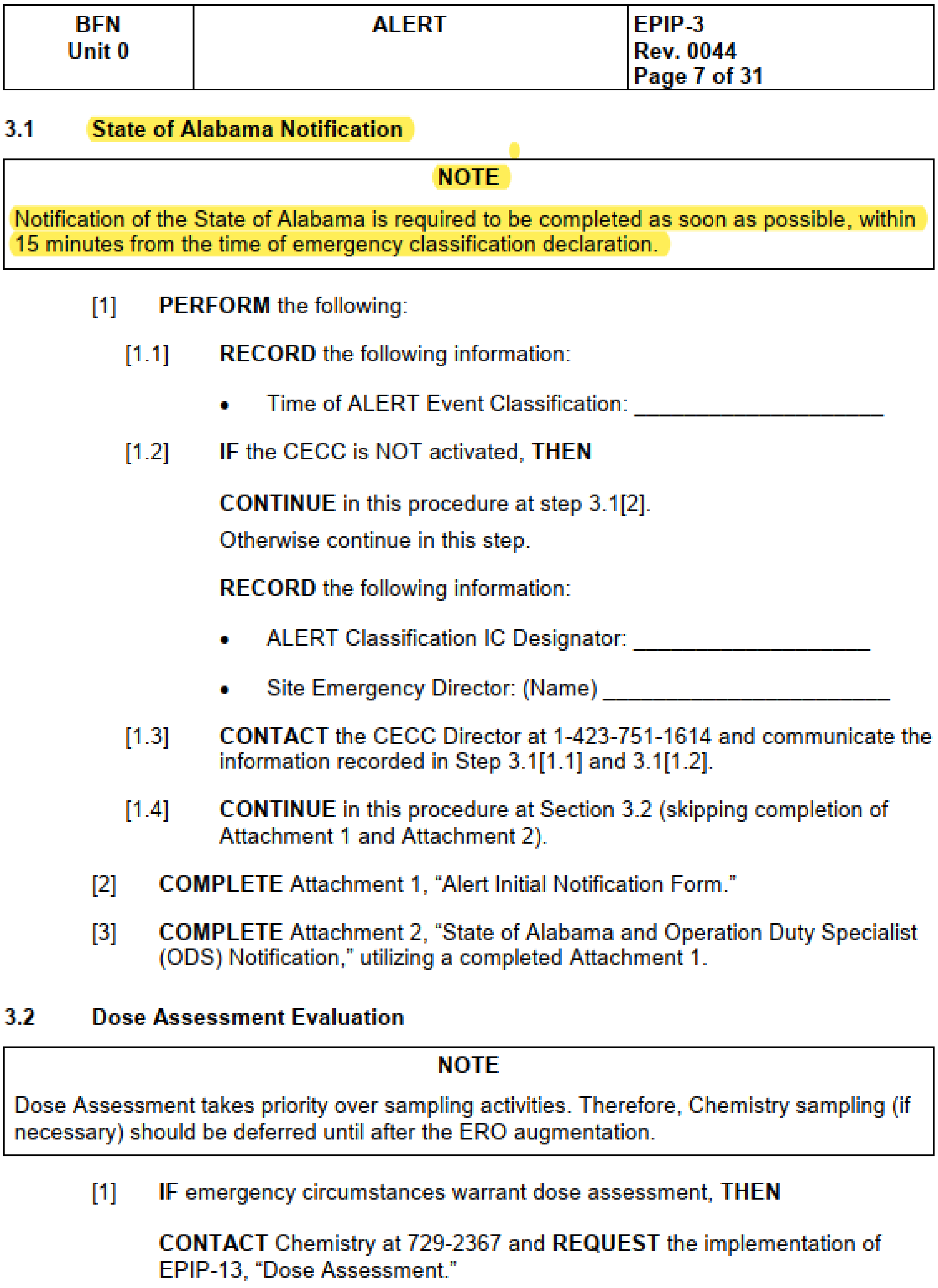
RU2 – UNPLANNED loss of water level above irradiated fuel.

(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:
- Fuel Pool System Abnormal annunciator lit and SFSP level less than 662' 8" as indicated by 1,2,3-LI-078-0042 and 1,2,3-LI-078-0043.
- Visual observation by plant personnel.

AND

b. UNPLANNED rise in area radiation levels as indicated by the following radiation monitor.
- 1,2,3-RM-90-1A Fuel Pool Floor

Excerpt from EPIP-3:



BFN **Unit 0**	**ALERT**	**EPIP-3** **Rev. 0044** **Page 7 of 31**

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 Attachment 1 and Attachment 2).

[2] **COMPLETE** Attachment 1, "Alert Initial Notification Form."

[3] **COMPLETE** Attachment 2, "State of Alabama and Operation Duty Specialist (ODS) Notification," utilizing a completed Attachment 1.

3.2 Dose Assessment Evaluation

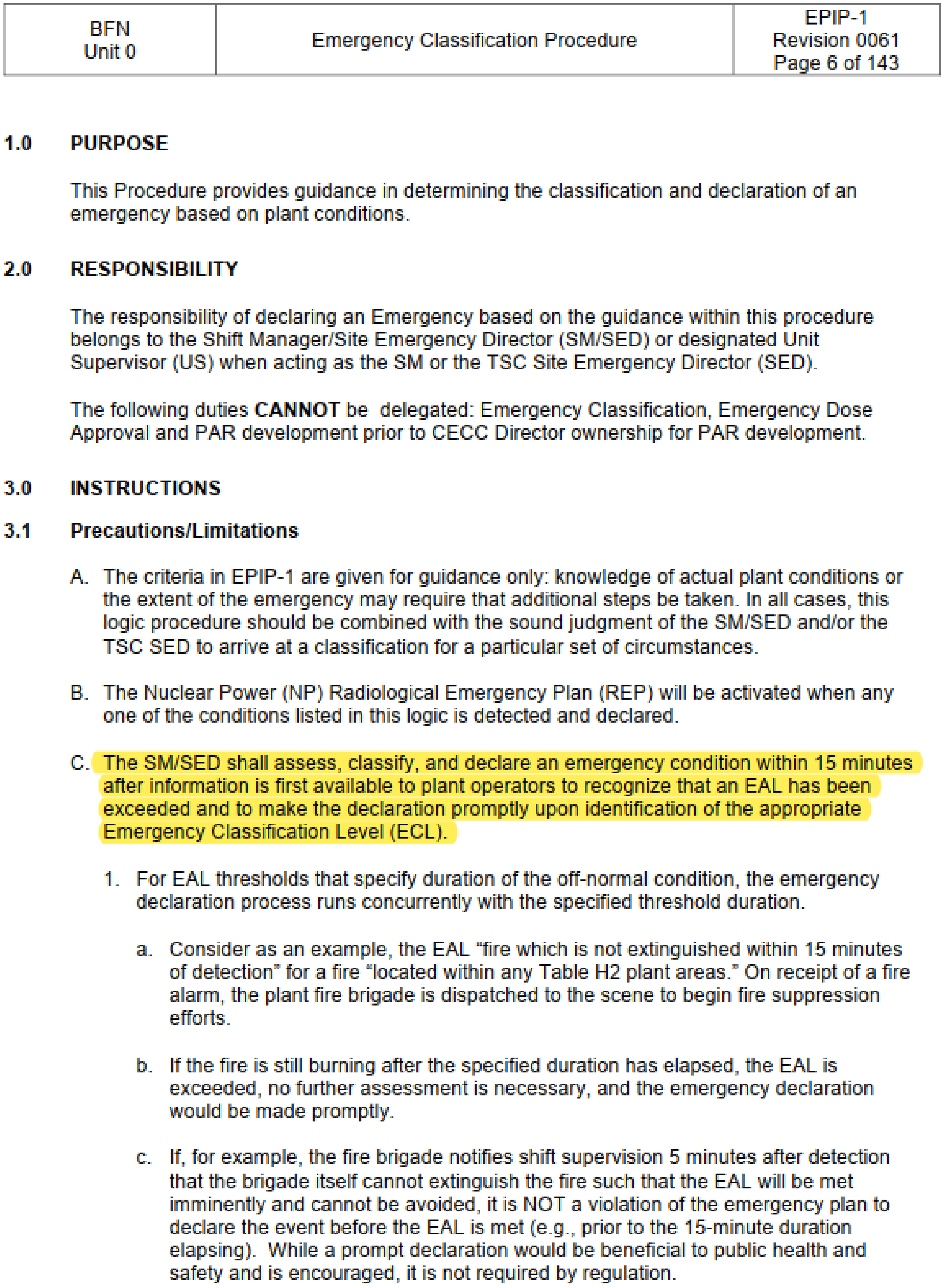
NOTE

Dose Assessment takes priority over sampling activities. Therefore, Chemistry sampling (if necessary) should be deferred until after the ERO augmentation.

[1] **IF** emergency circumstances warrant dose assessment, **THEN**

CONTACT Chemistry at 729-2367 and **REQUEST** the implementation of EPIP-13, "Dose Assessment."

Excerpt from EPIP-1: Supports Distractors B(2), D(2)



BFN Unit 0	Emergency Classification Procedure	EPIP-1 Revision 0061 Page 6 of 143

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).

   1. For EAL thresholds that specify duration of the off-normal condition, the emergency declaration process runs concurrently with the specified threshold duration.

      a. Consider as an example, the EAL “fire which is not extinguished within 15 minutes of detection” for a fire “located within any Table H2 plant areas.” On receipt of a fire alarm, the plant fire brigade is dispatched to the scene to begin fire suppression efforts.

      b. If the fire is still burning after the specified duration has elapsed, the EAL is exceeded, no further assessment is necessary, and the emergency declaration would be made promptly.

      c. If, for example, the fire brigade notifies shift supervision 5 minutes after detection that the brigade itself cannot extinguish the fire such that the EAL will be met imminently and cannot be avoided, it is NOT a violation of the emergency plan to declare the event before the EAL is met (e.g., prior to the 15-minute duration elapsing). While a prompt declaration would be beneficial to public health and safety and is encouraged, it is not required by regulation.

Examination Outline Cross-reference:	Level	RO	SRO
295024 (EPE 1) High Drywell Pressure / 5	Tier #	------	1
G2.4.2 (10CFR 55.43.5 - SRO Only)	Group #	------	1
Knowledge of system setpoints, interlocks and automatic actions associated with emergency and abnormal operating procedure entry conditions.	K/A #	295024G2.4.2	
	Importance Rating	------	4.6

Proposed Question: **# 79**

Unit 2 is operating at 8% RTP with the following conditions:

- Due to a temporary loss of Drywell cooling, Drywell Pressure **PEAKED** at 2.1 psig
- HPCI initiated automatically and subsequently was tripped and locked out
- Subsequently, Drywell cooling was restored with Drywell Pressure stable at 1.4 psig

Given the conditions above, the HPCI initiation was ___**(1)**___ and in accordance with Tech Spec Bases, the REACTOR MODE SWITCH ___**(2)**___ be placed in RUN.

A. (1) valid
(2) can

B. (1) valid
(2) can NOT

C. (1) invalid
(2) can

D. (1) invalid
(2) can NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT**:** First part is incorrect but plausible if the candidate misapplies the given Drywell Pressure of 2.1 psig as the initiation signal of 2.45 psig for HPCI. Normally, a trigger value is set for Drywell Pressure of 2.0 psig to initiate Core Flow reductions and then at 2.2 psig, to initiate a Reactor SCRAM given the trend. Additionally, 2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell, lists several alarms related to high Drywell Pressure ranging from 1.6 psig to 1.96 psig that also adds to complexity. Second part is incorrect but plausible if the candidate fails to remember (with no reference provided), that Tech Spec Bases for LCO 3.0.4.b does not apply to HPCI believing the performance of a Risk Assessment would allow the REACTOR MODE SWITCH to be placed into RUN for MODE 1 entry.

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-OI-73, High Pressure Coolant Injection System, HPCI automatically initiates when Drywell Pressure reaches 2.45 psig, therefore the inadvertent initiation was invalid. For second part, in accordance with Tech Spec Bases for LCO 3.0.4.b, entry can be allowed 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 HPCI's Tech Spec 3.5.1. With an inadvertent initiation of HPCI and subsequently locked out, HPCI is now INOPERABLE, therefore a MODE change is not allowed.

SRO Level Justification: Tests the candidate's knowledge of Technical Specifications as it relates to HPCI Operability and Drywell Pressure interlocks. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents and (4) the assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s):	U2 Tech Spec Bases 3.5.1, Amend. 286	(Attach if not previously provided)
	U2 Tech Spec Bases 3.0, Amend. 286	
	2-OI-73, Rev. 101	

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.042, Obj. 2 (As available)

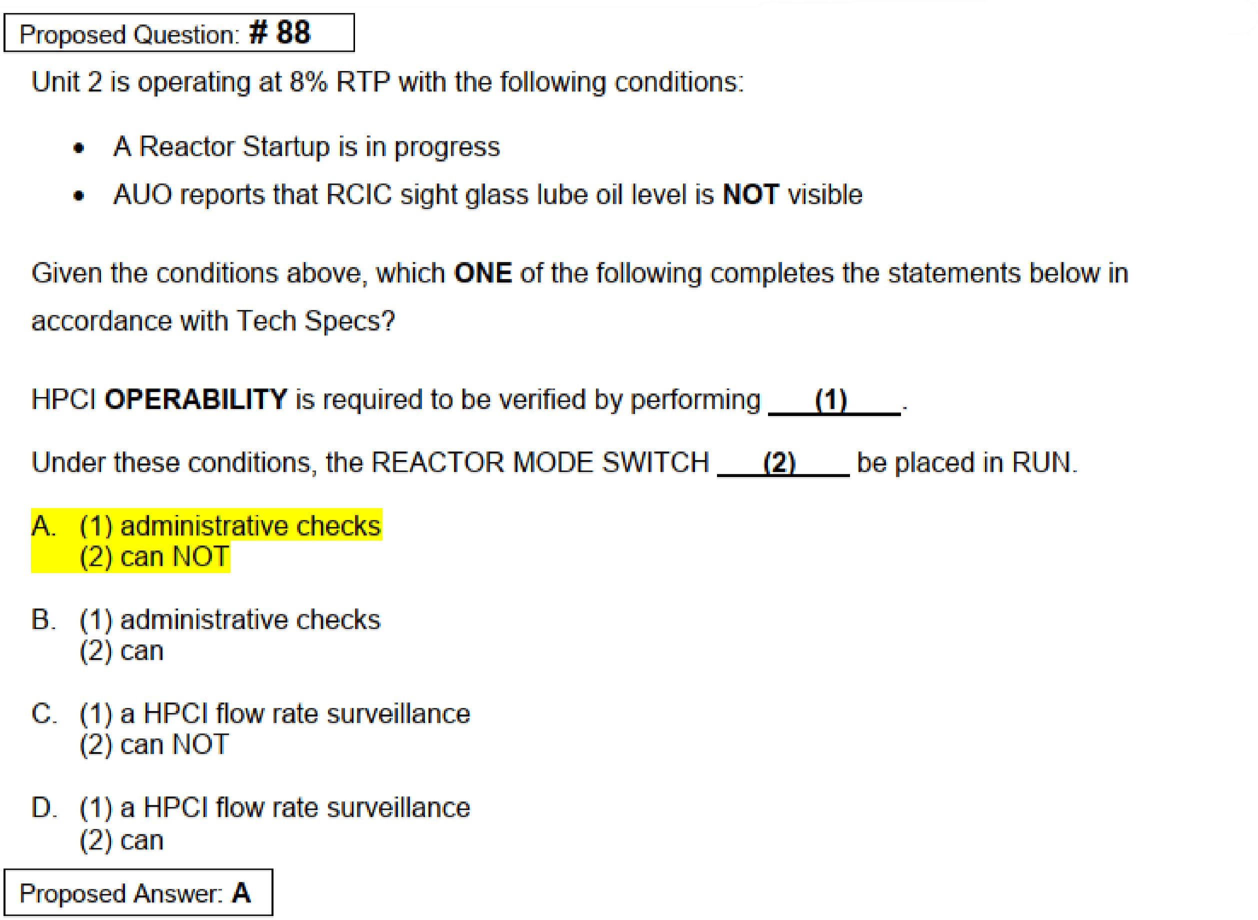
Question Source:	Bank #		
	Modified Bank #	BFN NRC 21-04 #88	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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: **# 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

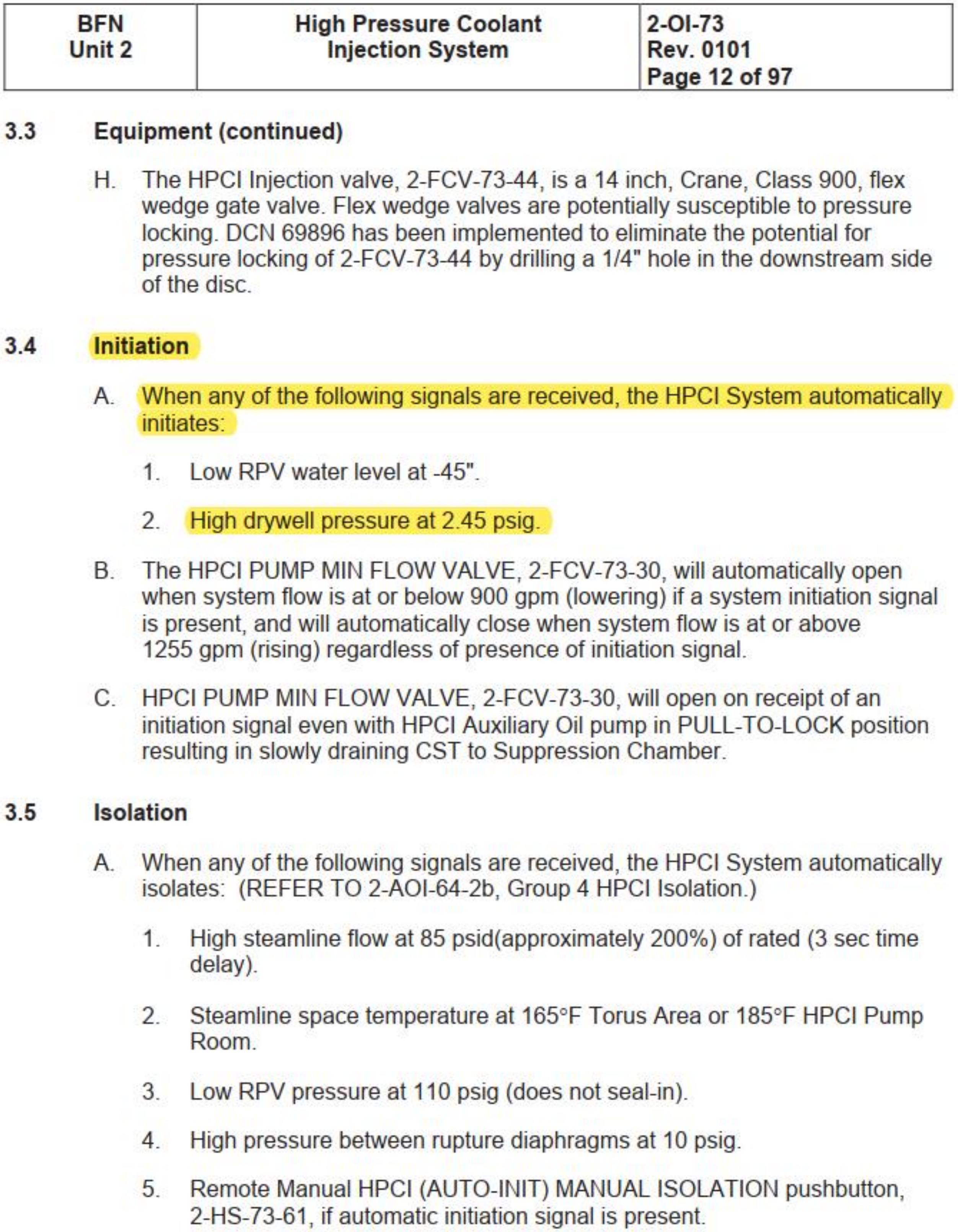
B. (1) administrative checks
(2) can

C. (1) a HPCI flow rate surveillance
(2) can NOT

D. (1) a HPCI flow rate surveillance
(2) can

Proposed Answer: **A**

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.

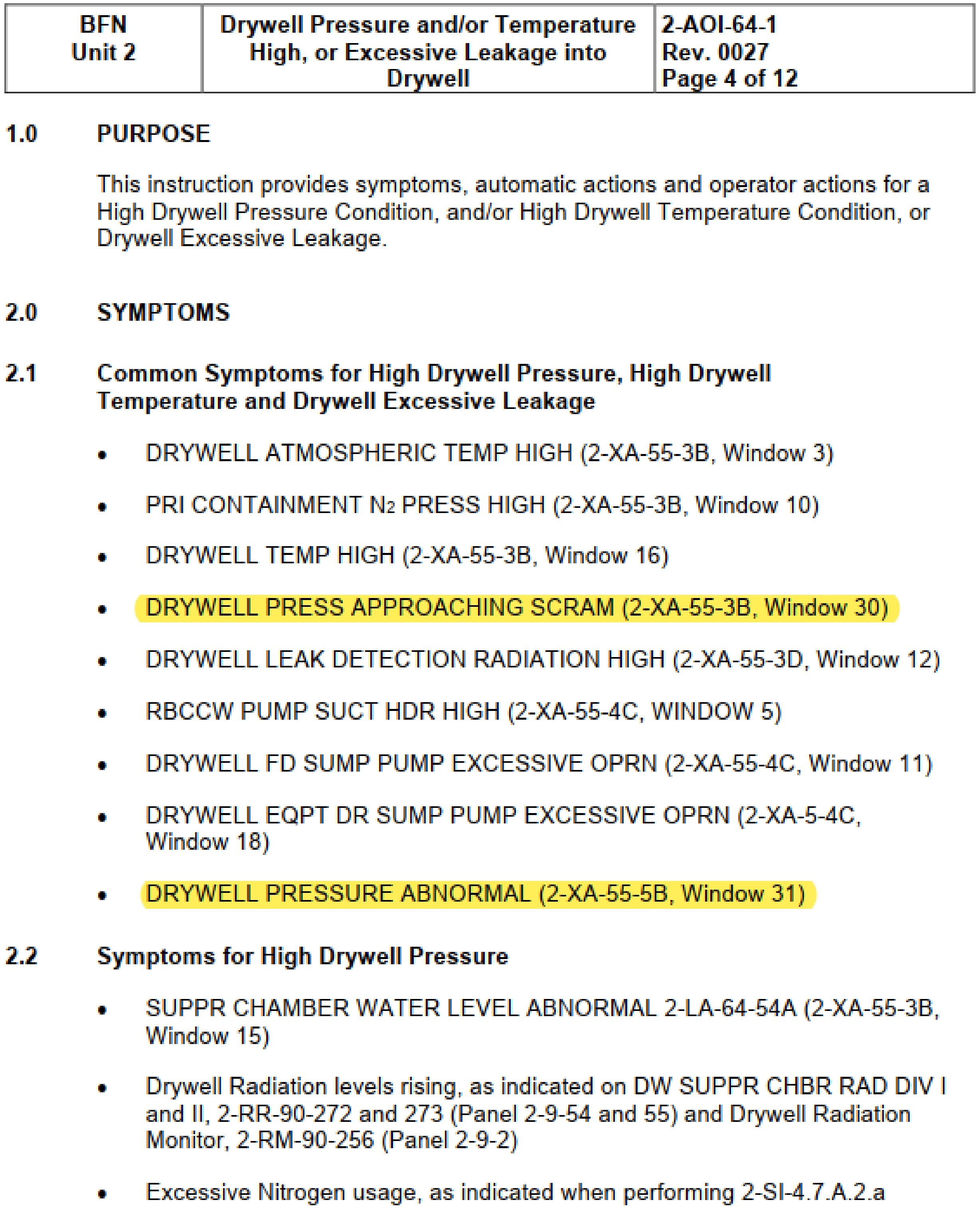
C. HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, will open on receipt of an initiation signal even with HPCI Auxiliary Oil pump in PULL-TO-LOCK position resulting in slowly draining CST to Suppression Chamber.

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.)

1. High steamline flow at 85 psid(approximately 200%) of rated (3 sec time delay).
2. Steamline space temperature at 165°F Torus Area or 185°F HPCI Pump Room.
3. Low RPV pressure at 110 psig (does not seal-in).
4. High pressure between rupture diaphragms at 10 psig.
5. Remote Manual HPCI (AUTO-INIT) MANUAL ISOLATION pushbutton, 2-HS-73-61, if automatic initiation signal is present.

Excerpt from 2-AOI-64-1: Supports Distractor A(1), B(1)



BFN Unit 2	**Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell**	**2-AOI-64-1 Rev. 0027 Page 4 of 12**

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a High Drywell Pressure Condition, and/or High Drywell Temperature Condition, or Drywell Excessive Leakage.

2.0 SYMPTOMS

2.1 Common Symptoms for High Drywell Pressure, High Drywell Temperature and Drywell Excessive Leakage

- DRYWELL ATMOSPHERIC TEMP HIGH (2-XA-55-3B, Window 3)
- PRI CONTAINMENT N_2 PRESS HIGH (2-XA-55-3B, Window 10)
- DRYWELL TEMP HIGH (2-XA-55-3B, Window 16)
- DRYWELL PRESS APPROACHING SCRAM (2-XA-55-3B, Window 30)
- DRYWELL LEAK DETECTION RADIATION HIGH (2-XA-55-3D, Window 12)
- RBCCW PUMP SUCT HDR HIGH (2-XA-55-4C, WINDOW 5)
- DRYWELL FD SUMP PUMP EXCESSIVE OPRN (2-XA-55-4C, Window 11)
- DRYWELL EQPT DR SUMP PUMP EXCESSIVE OPRN (2-XA-5-4C, Window 18)
- DRYWELL PRESSURE ABNORMAL (2-XA-55-5B, Window 31)

2.2 Symptoms for High Drywell Pressure

- SUPPR CHAMBER WATER LEVEL ABNORMAL 2-LA-64-54A (2-XA-55-3B, Window 15)
- Drywell Radiation levels rising, as indicated on DW SUPPR CHBR RAD DIV I and II, 2-RR-90-272 and 273 (Panel 2-9-54 and 55) and Drywell Radiation Monitor, 2-RM-90-256 (Panel 2-9-2)
- Excessive Nitrogen usage, as indicated when performing 2-SI-4.7.A.2.a

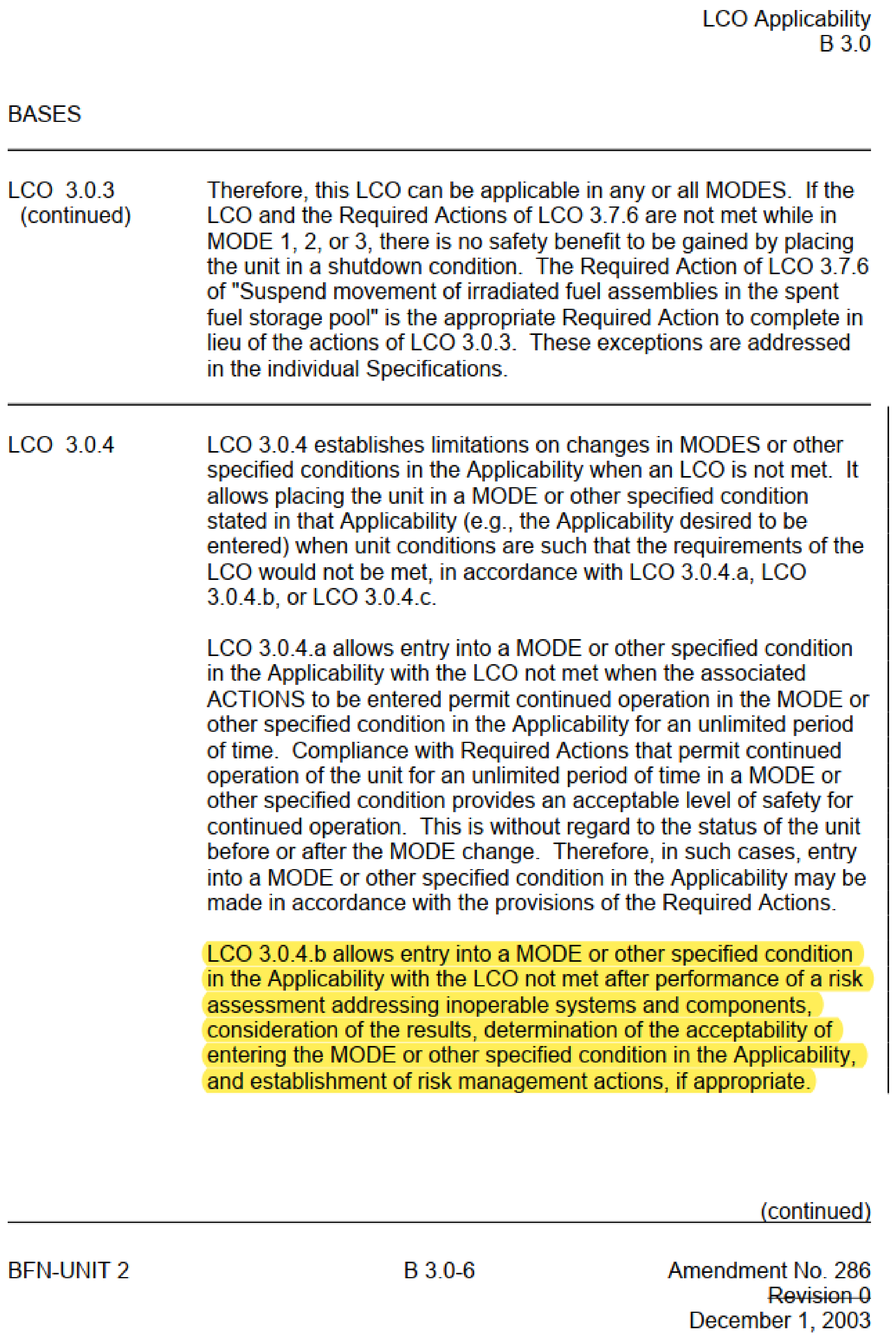
Excerpt from U2 Tech Spec Bases 3.5.1:

BASES

LCO (continued)	LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR low pressure permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.
APPLICABILITY	All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure is ≤ 150 psig, ADS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS - Shutdown."
ACTIONS	A Note prohibits the application of LCO 3.0.4 b to an inoperable HPCI subsystem. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable HPCI subsystem 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)

Excerpts from U2 Tech Spec Bases for LCO 3.0.4.b:



LCO Applicability
B 3.0

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)

BFN-UNIT 2 B 3.0-6 Amendment No. 286
~~Revision 0~~
December 1, 2003

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:	Level	RO	SRO
295028 (EPE 5) High Drywell Temperature / 5	Tier #	------	1
EA2.03 (10CFR 55.43.5 - SRO Only)	Group #	------	1
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: • Reactor Water Level	K/A #	295028EA2.03	
	Importance Rating	------	4.0

Proposed Question: **# 80**

Unit 2 was in MODE 4 with Reactor Water Level at 120 inches during vessel flood up when the following occurs:

- Loss of Shutdown Cooling
- Leak in Drywell
- Drywell Pressure was reported as 15 psig
- Drywell Temperature reported to be 230 °F
- 2-LI-3-55, FLOOD UP LEVEL INSTRUMENT, is indicating 46 inches

Given the conditions above and in accordance with the EOIs, 2-LI-3-55, FLOOD UP LEVEL INSTRUMENT ___**(1)**___ valid.

The NUSO will direct performance of ___**(2)**___.

Note: 2-EOI-APPENDIX-17B, RHR System Operation Drywell Sprays
2-EOI-APPENDIX-17C, RHR System Operation Suppression Chamber Sprays

[REFERENCE PROVIDED]

A. (1) is
(2) 2-EOI-APPENDIX-17C **ONLY**

B. (1) is
(2) 2-EOI-APPENDIX-17C **AND** 2-EOI-APPENDIX-17B

C. (1) is NOT
(2) 2-EOI-APPENDIX-17C **ONLY**

D. (1) is NOT
(2) 2-EOI-APPENDIX-17C **AND** 2-EOI-APPENDIX-17B

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** First part is correct *(See B).* Second part is incorrect but plausible in that the PC pressure leg of EOI-2 requires initiation of 2-EOI-APPENDIX-17B, Drywell Sprays when if in the Safe Area of Curve 5 and when Suppression Chamber pressure exceeds 12 psig, however 2-EOI-APPENDIX-17C, Suppression Chamber Sprays if also required.

B **CORRECT**: *(See attached)* In accordance with 2-EOI-5, CURVES AND CAUTIONS, Caution 1 Instrument Level Table, states that an RPV Water Level instrument may be used to determine or trend Level only when it reads above the Minimum indicated Level associated with the highest max Drywell or Secondary Containment run temperature. Given that 2-LI-3-55, FLOOD UP LEVEL INSTRUMENT (only located in the Drywell) is currently indicating (+) 46 inches with Drywell Temperature at 230 ºF, the Reactor Water Level can be considered good even down to (+) 30 inches. For second part, in accordance with 2-EOI-2, Primary Containment Control, the PC pressure leg states that WHEN PC pressure cannot be maintained below 2.45 psig and BEFORE Suppression Chamber Pressure rises to 12 psig to initiate 2-EOI-APPENDIX-17C, Suppression Chamber Sprays. Also, the PC pressure leg of EOI-2 requires initiation of 2-EOI-APPENDIX-17B, Drywell Sprays if in the Safe Area of Curve 5 and when Suppression Chamber pressure exceeds 12 psig. This must be extrapolated since it is not given from Drywell Pressure with no other failures. In the DW/T leg, Drywell can be sprayed WHEN Drywell Temperature cannot be maintained below 160 ºF, but is required before 280 ºF.

C INCORRECT: First part is incorrect but plausible if the candidate misapplies the Instrument Level Table from 2-EOI-5, CURVES AND CAUTIONS and Caution 1 for the given Reactor Water Level and Drywell Temperature and Table 6, Secondary Containment Instrument Runs. For the Reactor Water Level instrument to be considered invalid, Reactor Water Level would have to indicate less than 30 inches OR Drywell Temperature would have to be above 300 ºF. 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 determine and interpret the impact that High Drywell Pressure has on Reactor Water Level based on instrument locations. 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 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents and (4) the assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 2-EOI-2, Rev. 17 (Attach if not previously provided)
2-EOI-5, Rev. 3

Proposed references to be provided to applicants during examination: **2-EOI-5, CURVES AND CAUTIONS Caution 1 Instrument Level Table,**
Curve 5, Drywell Spray Initiation Limit

Learning Objective: OPL171.203, Obj. 8 (As available)

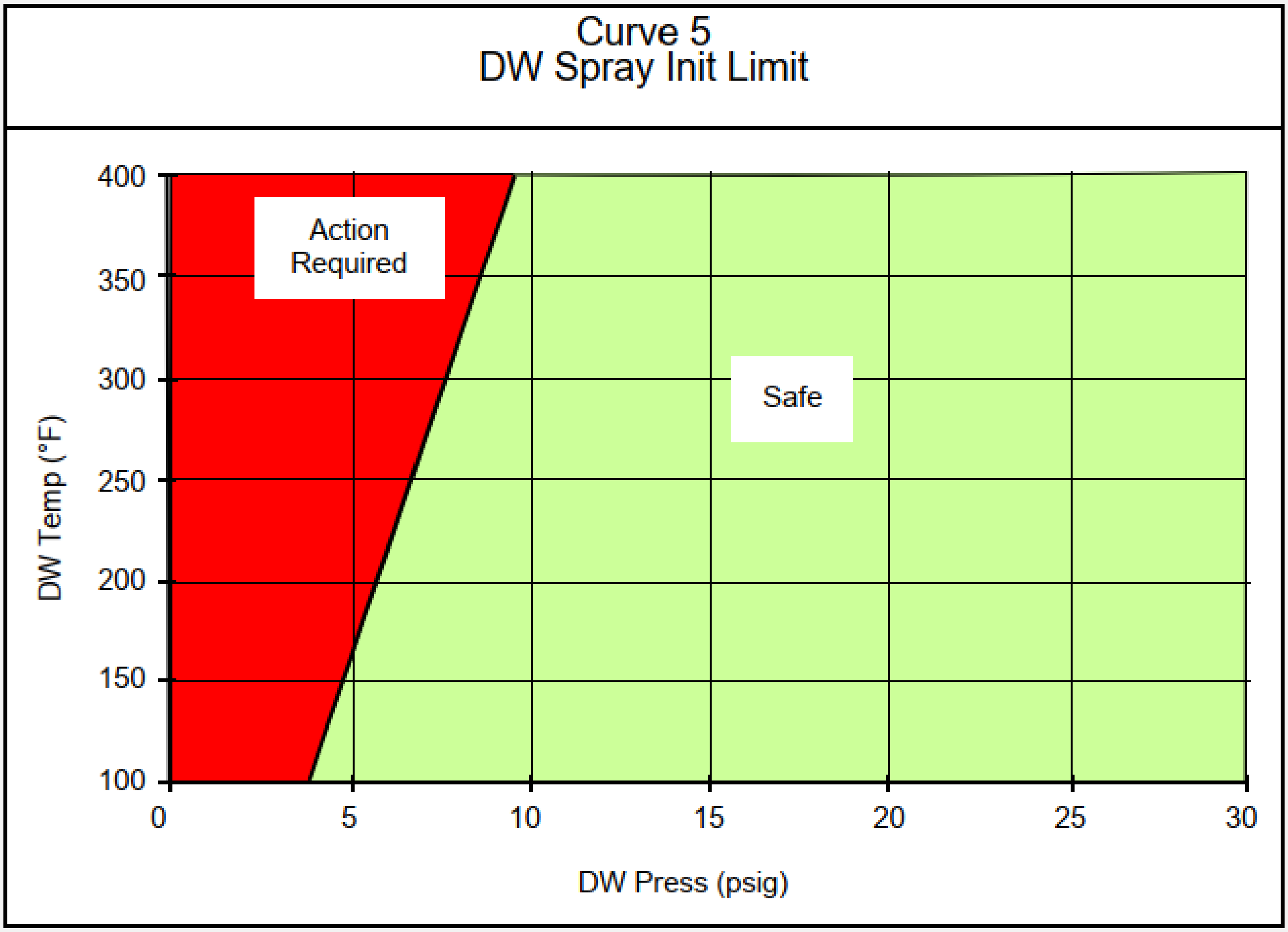
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

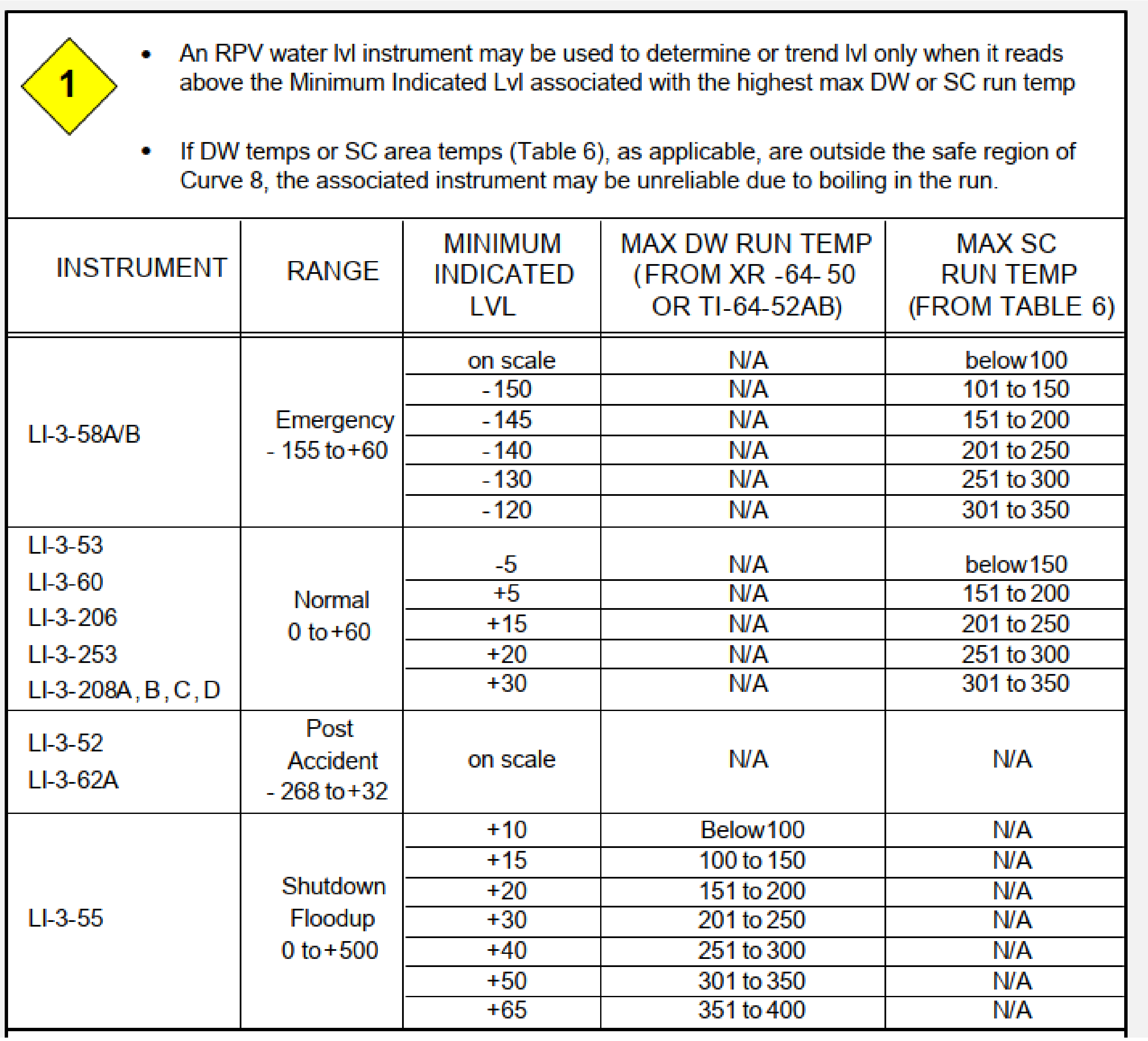
10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

2-EOI-5, Curve 5, Drywell Spray Initiation Limit Reference Provided to candidate:



2-EOI-5, Curve 5, Caution 1 Instrument Level Table Reference Provided to candidate:

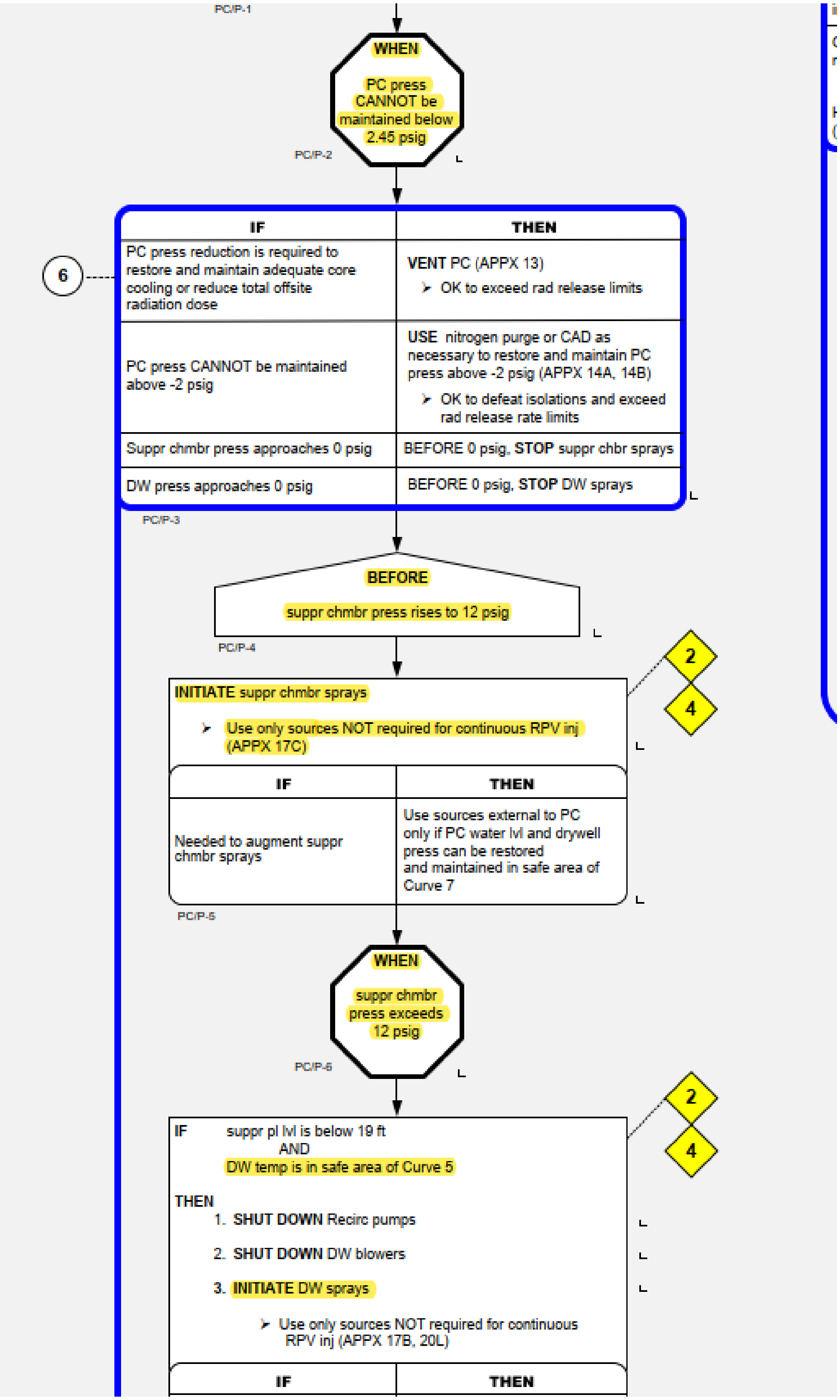


1

- 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	-5	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

Excerpt from 2-EOI-2 PC Pressure leg:



PC/P-1

WHEN

PC press CANNOT be maintained below 2.45 psig

PC/P-2

⑥

IF	THEN
PC press reduction is required to restore and maintain adequate core cooling or reduce total offsite radiation dose	**VENT** PC (APPX 13) ➤ OK to exceed rad release limits
PC press CANNOT be maintained above -2 psig	**USE** nitrogen purge or CAD as necessary to restore and maintain PC press above -2 psig (APPX 14A, 14B) ➤ OK to defeat isolations and exceed rad release rate limits
Suppr chmbr press approaches 0 psig	BEFORE 0 psig, **STOP** suppr chbr sprays
DW press approaches 0 psig	BEFORE 0 psig, **STOP** DW sprays

PC/P-3

BEFORE

suppr chmbr press rises to 12 psig

PC/P-4

INITIATE suppr chmbr sprays

➤ Use only sources NOT required for continuous RPV inj (APPX 17C)

IF	THEN
Needed to augment suppr chmbr sprays	Use sources external to PC only if PC water lvl and drywell press can be restored and maintained in safe area of Curve 7

2

4

PC/P-5

WHEN

suppr chmbr press exceeds 12 psig

PC/P-6

IF suppr pl lvl is below 19 ft
AND
DW temp is in safe area of Curve 5

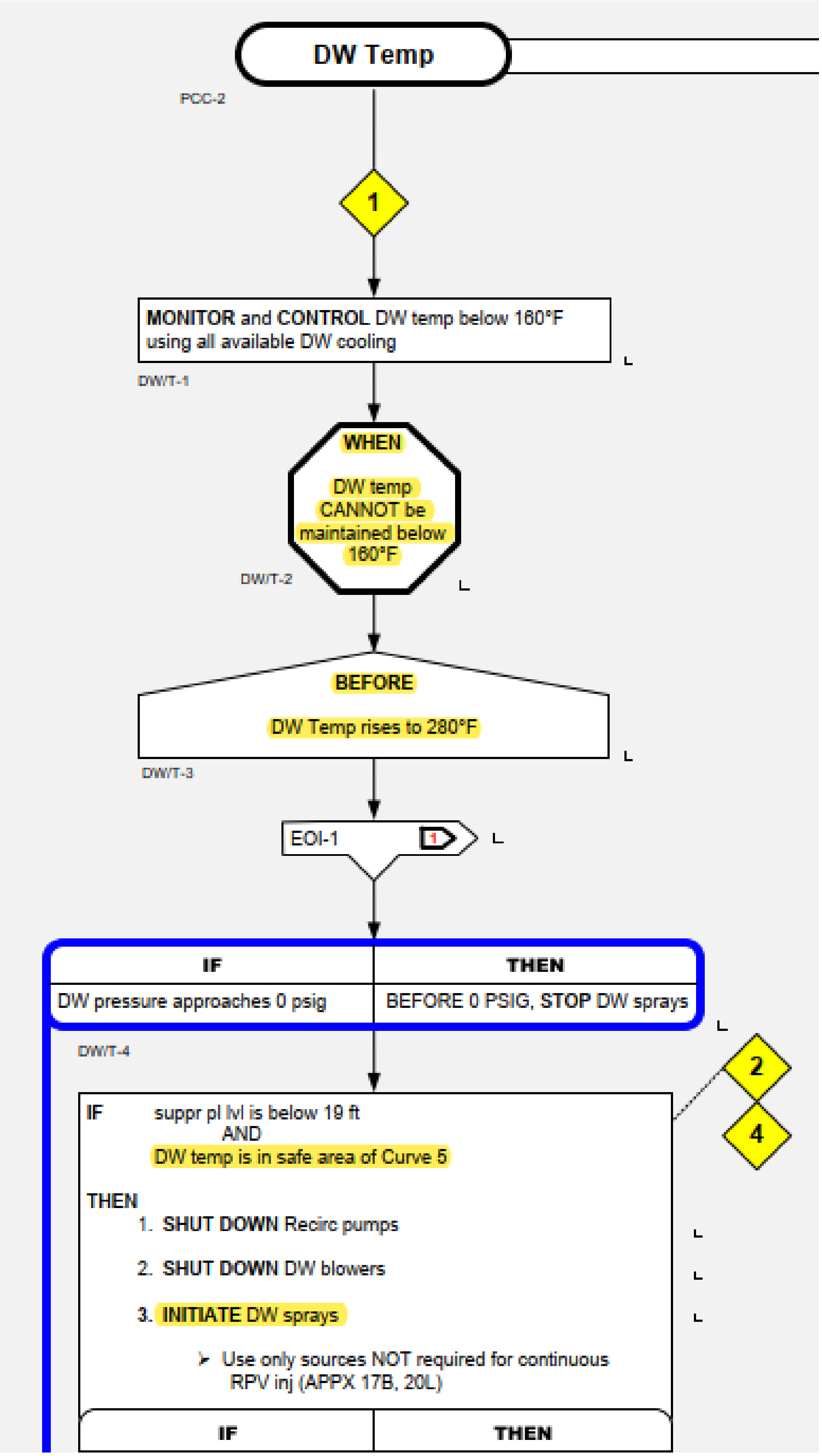
THEN
1. **SHUT DOWN** Recirc pumps
2. **SHUT DOWN** DW blowers
3. **INITIATE** DW sprays
   - ➤ Use only sources NOT required for continuous RPV inj (APPX 17B, 20L)

2

4

IF	THEN

Excerpt from 2-EOI-2 DW Temperature leg: Supports Distractors A(2), C(2):



Examination Outline Cross-reference:

295037 (EPE 14) SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1

G2.1.32 (10CFR 55.43.2 - SRO Only)

Ability to explain and apply system precautions, limitations, notes, or cautions

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295037G2.1.32	
Importance Rating	------	4.0

Proposed Question: **# 81**

An ATWS has occurred on Unit 3 resulting in the following conditions:

- Reactor Power is 20%
- Reactor Water Level is 0 inches

3-EOI-1A, ATWS RPV Control, ___**(1)**___ direct Operators to **STOP** and **PREVENT** injection from EHPM and RCIC.

In accordance with the EOI Program Manual Bases, the reason Reactor Water Level is lowered to (-) 50 inches is to uncover the ___**(2)**___ spargers.

A. (1) does
(2) Feedwater

B. (1) does
(2) Core Spray

C. (1) does NOT
(2) Feedwater

D. (1) does NOT
(2) Core Spray

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the Reactor Water level leg of 3-EOI-1A states that level is lowered if greater than 5% RTP and above (-) 50 inches Reactor Water Level by using 3-EOI-APPENDIX-4, Prevention of Injection. Appendix-4 allows the low injection flow rates by using SLC at 60 gpm and CRD at 180 gpm post SCRAM, whereas RCIC is 620 gpm and the EHPM can inject over 1000 gpm. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that Core Spray spargers are inside the core shroud positioned above Top of Active Fuel (TAF) which is (-) 162 inches to spray the upper one third of the core. Feedwater nozzles are much higher in the vessel and are outside the shroud.

C **CORRECT**: (*See attached)* In accordance with 3-EOI-1A, ATWS RPV Control, ARC/L-9 states to **STOP** and **PREVENT** all injection except into the RPV to deliberately lower RPV Water Level except from EHPM, RCIC, CRD, and SLC. Although the EHPM and RCIC have significantly higher flow rates, they are allowed to continue to inject as long as Reactor Water Level is still lowered. For second part, in accordance with EOIPM Section 0-V-D, lowering Reactor Water Level to (-) 50 inches, which equates to 24 inches below the feedwater spargers, results in a reduction in core inlet subcooling thus lowering Reactor Power.

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

SRO Level Justification: Test the candidate’s knowledge of the EOI Bases as it relates to lowering Reactor Water Level during ATWS SCRAM conditions. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site’s procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 3-EOI-1A, Rev. 3 (Attach if not previously provided)
EOIPM 0-V(D), Rev. 1
OPL171.002, Rev. 14

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.202 Obj. 23 (As available)

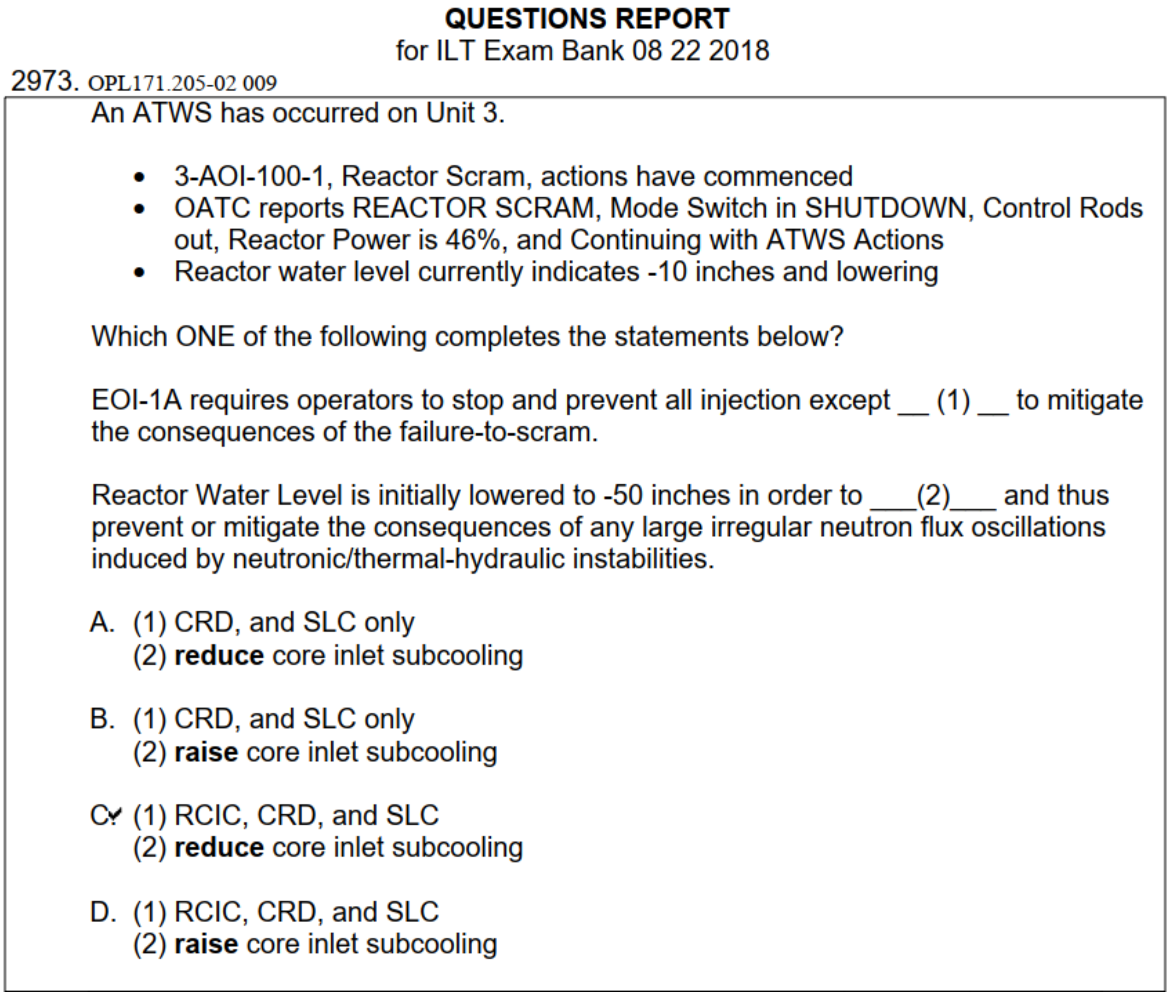
Question Source:	Bank #		
	Modified Bank #	OPL171.205-02 009 #2973	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

2973. OPL171.205-02 009

An ATWS has occurred on Unit 3.

- 3-AOI-100-1, Reactor Scram, actions have commenced
- OATC reports REACTOR SCRAM, Mode Switch in SHUTDOWN, Control Rods out, Reactor Power is 46%, and Continuing with ATWS Actions
- Reactor water level currently indicates -10 inches and lowering

Which ONE of the following completes the statements below?

EOI-1A requires operators to stop and prevent all injection except __ (1) __ to mitigate the consequences of the failure-to-scram.

Reactor Water Level is initially lowered to -50 inches in order to ___(2)___ and thus prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities.

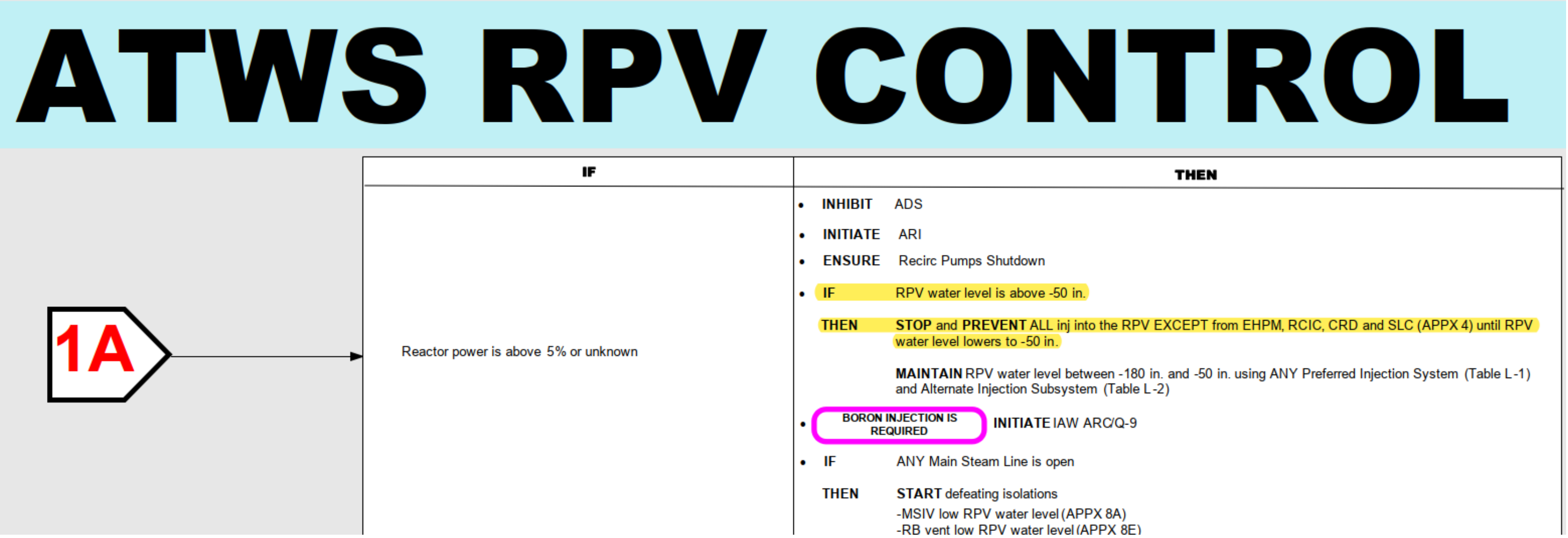
A. (1) CRD, and SLC only
(2) **reduce** core inlet subcooling

B. (1) CRD, and SLC only
(2) **raise** core inlet subcooling

C✓ (1) RCIC, CRD, and SLC
(2) **reduce** core inlet subcooling

D. (1) RCIC, CRD, and SLC
(2) **raise** core inlet subcooling

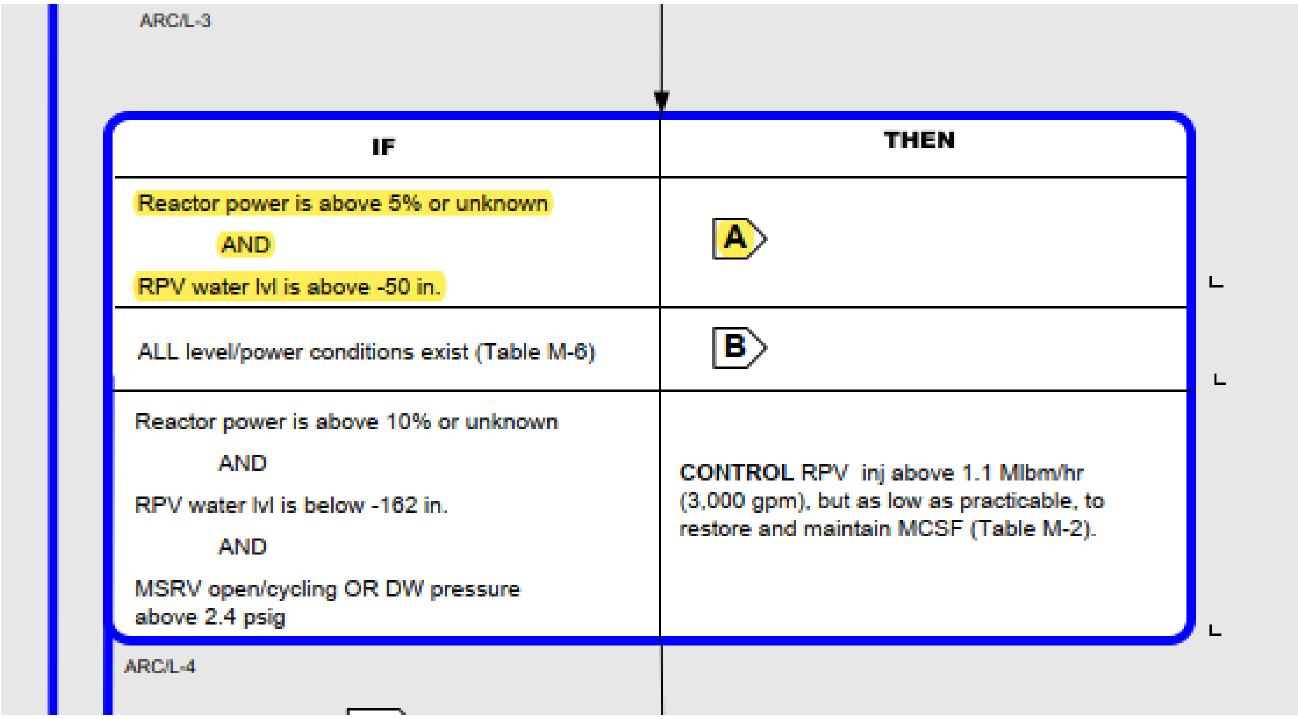
Excerpts from 3-EOI-1A:



ATWS RPV CONTROL

1A

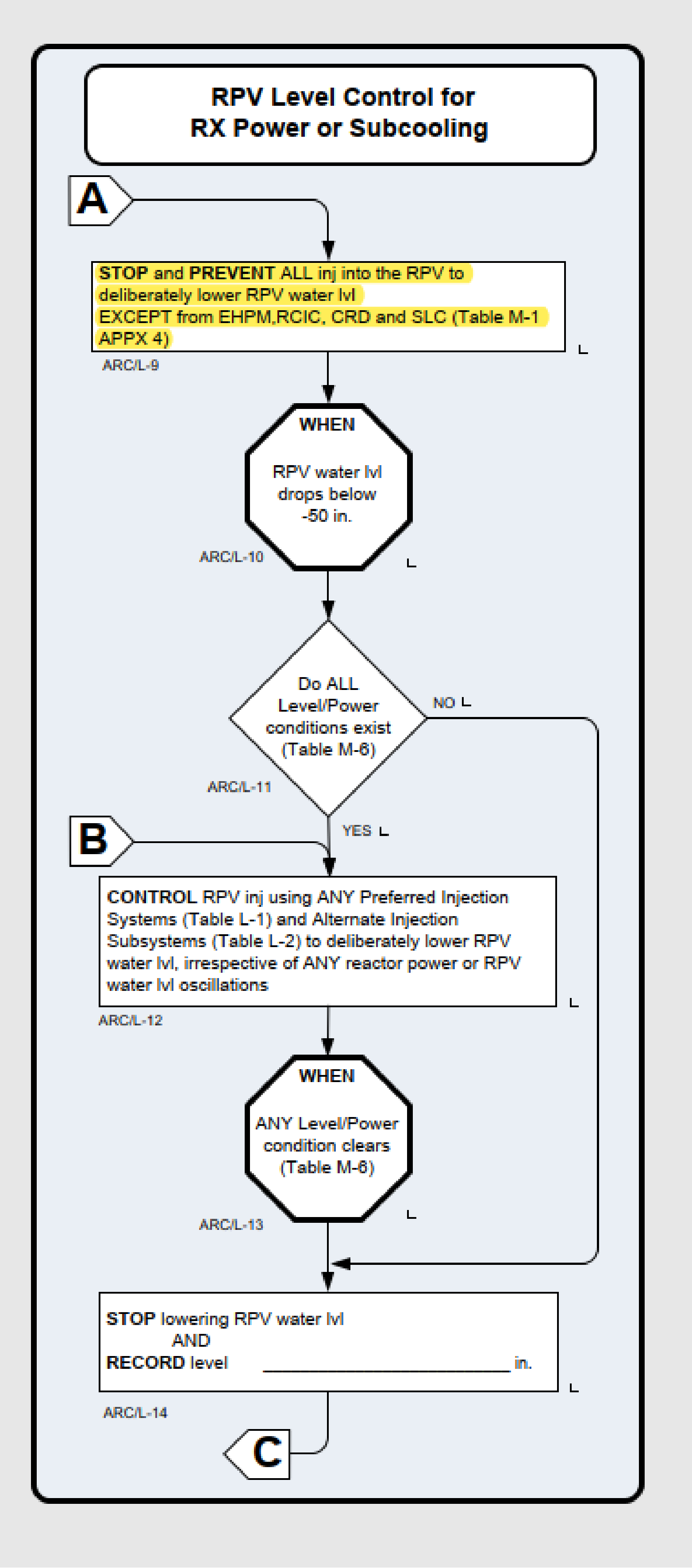
IF	THEN
Reactor power is above 5% or unknown	• **INHIBIT** ADS • **INITIATE** ARI • **ENSURE** Recirc Pumps Shutdown • **IF** RPV water level is above -50 in. **THEN** **STOP** and **PREVENT** ALL inj into the RPV EXCEPT from EHPM, RCIC, CRD and SLC (APPX 4) until RPV water level lowers to -50 in. **MAINTAIN** RPV water level between -180 in. and -50 in. using ANY Preferred Injection System (Table L-1) and Alternate Injection Subsystem (Table L-2) • **BORON INJECTION IS REQUIRED** **INITIATE** IAW ARC/Q-9 • **IF** ANY Main Steam Line is open **THEN** **START** defeating isolations -MSIV low RPV water level (APPX 8A) -RB vent low RPV water level (APPX 8E)



ARC/L-3

IF	THEN
Reactor power is above 5% or unknown AND RPV water lvl is above -50 in.	A
ALL level/power conditions exist (Table M-6)	B
Reactor power is above 10% or unknown AND RPV water lvl is below -162 in. AND MSRV open/cycling OR DW pressure above 2.4 psig	**CONTROL** RPV inj above 1.1 Mlbm/hr (3,000 gpm), but as low as practicable, to restore and maintain MCSF (Table M-2).

ARC/L-4



RPV Level Control for RX Power or Subcooling

STOP and **PREVENT** ALL inj into the RPV to deliberately lower RPV water lvl EXCEPT from EHPM,RCIC, CRD and SLC (Table M-1 APPX 4)

WHEN RPV water lvl drops below -50 in.

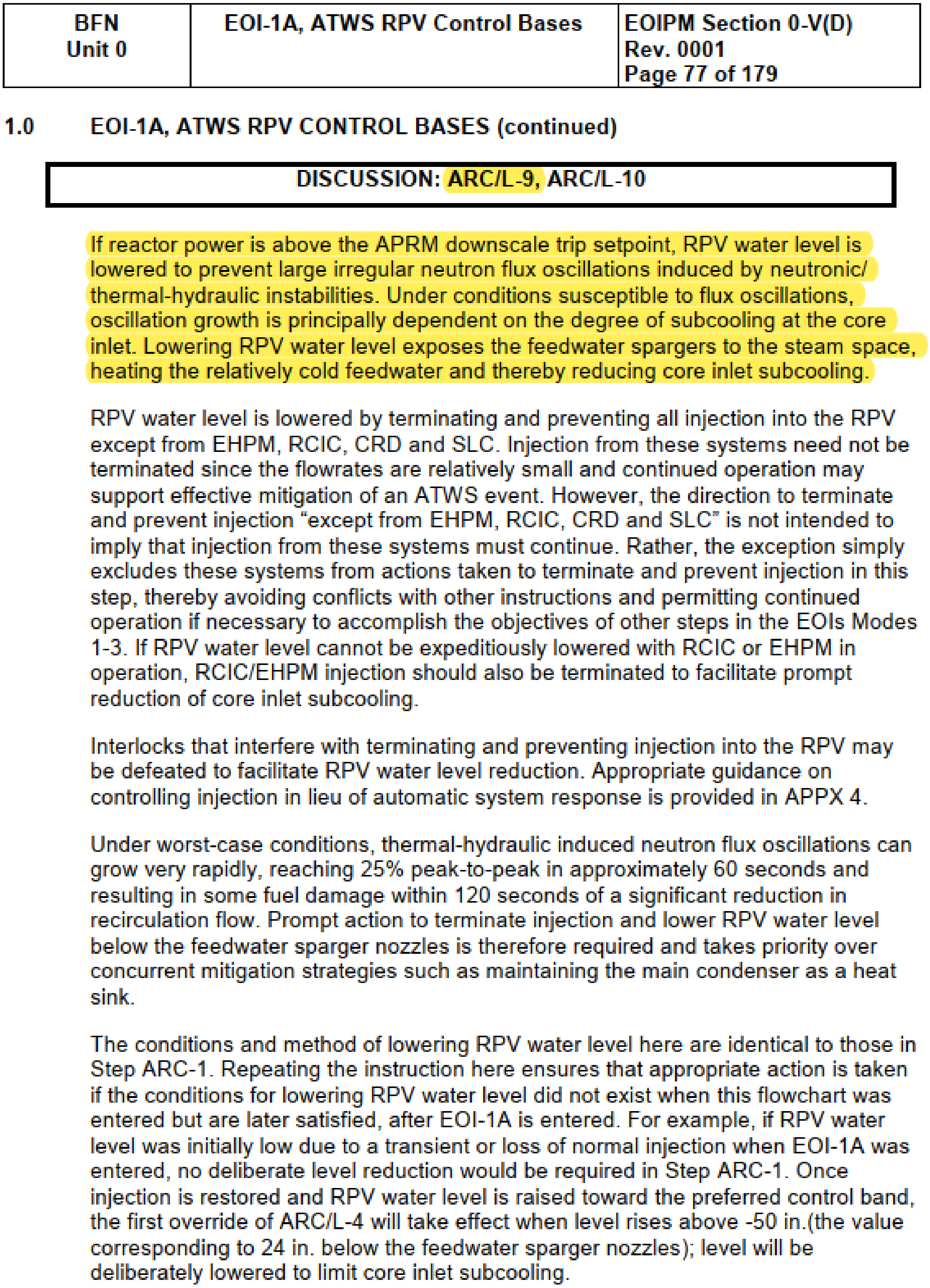
Do ALL Level/Power conditions exist (Table M-6)

CONTROL RPV inj using ANY Preferred Injection Systems (Table L-1) and Alternate Injection Subsystems (Table L-2) to deliberately lower RPV water lvl, irrespective of ANY reactor power or RPV water lvl oscillations

WHEN ANY Level/Power condition clears (Table M-6)

STOP lowering RPV water lvl
AND
RECORD level ______________________ in.

Excerpt from EOIPM 0-V(D):



BFN **Unit 0**	**EOI-1A, ATWS RPV Control Bases**	**EOIPM Section 0-V(D)** **Rev. 0001** **Page 77 of 179**

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/L-9, ARC/L-10

If reactor power is above the APRM downscale trip setpoint, RPV water level is lowered to prevent large irregular neutron flux oscillations induced by neutronic/ thermal-hydraulic instabilities. Under conditions susceptible to flux oscillations, oscillation growth is principally dependent on the degree of subcooling at the core inlet. Lowering RPV water level exposes the feedwater spargers to the steam space, heating the relatively cold feedwater and thereby reducing core inlet subcooling.

RPV water level is lowered by terminating and preventing all injection into the RPV except from EHPM, RCIC, CRD and SLC. Injection from these systems need not be terminated since the flowrates are relatively small and continued operation may support effective mitigation of an ATWS event. However, the direction to terminate and prevent injection "except from EHPM, RCIC, CRD and SLC" is not intended to imply that injection from these systems must continue. Rather, the exception simply excludes these systems from actions taken to terminate and prevent injection in this step, thereby avoiding conflicts with other instructions and permitting continued operation if necessary to accomplish the objectives of other steps in the EOIs Modes 1-3. If RPV water level cannot be expeditiously lowered with RCIC or EHPM in operation, RCIC/EHPM injection should also be terminated to facilitate prompt reduction of core inlet subcooling.

Interlocks that interfere with terminating and preventing injection into the RPV may be defeated to facilitate RPV water level reduction. Appropriate guidance on controlling injection in lieu of automatic system response is provided in APPX 4.

Under worst-case conditions, thermal-hydraulic induced neutron flux oscillations can grow very rapidly, reaching 25% peak-to-peak in approximately 60 seconds and resulting in some fuel damage within 120 seconds of a significant reduction in recirculation flow. Prompt action to terminate injection and lower RPV water level below the feedwater sparger nozzles is therefore required and takes priority over concurrent mitigation strategies such as maintaining the main condenser as a heat sink.

The conditions and method of lowering RPV water level here are identical to those in Step ARC-1. Repeating the instruction here ensures that appropriate action is taken if the conditions for lowering RPV water level did not exist when this flowchart was entered but are later satisfied, after EOI-1A is entered. For example, if RPV water level was initially low due to a transient or loss of normal injection when EOI-1A was entered, no deliberate level reduction would be required in Step ARC-1. Once injection is restored and RPV water level is raised toward the preferred control band, the first override of ARC/L-4 will take effect when level rises above -50 in.(the value corresponding to 24 in. below the feedwater sparger nozzles); level will be deliberately lowered to limit core inlet subcooling.

Excerpt from OPL171.002 Lesson Plan: Supports Distractors B(2), D(2)

OPL171.002, REACTOR PRESSURE VESSEL AND INTERNALS, REV 14

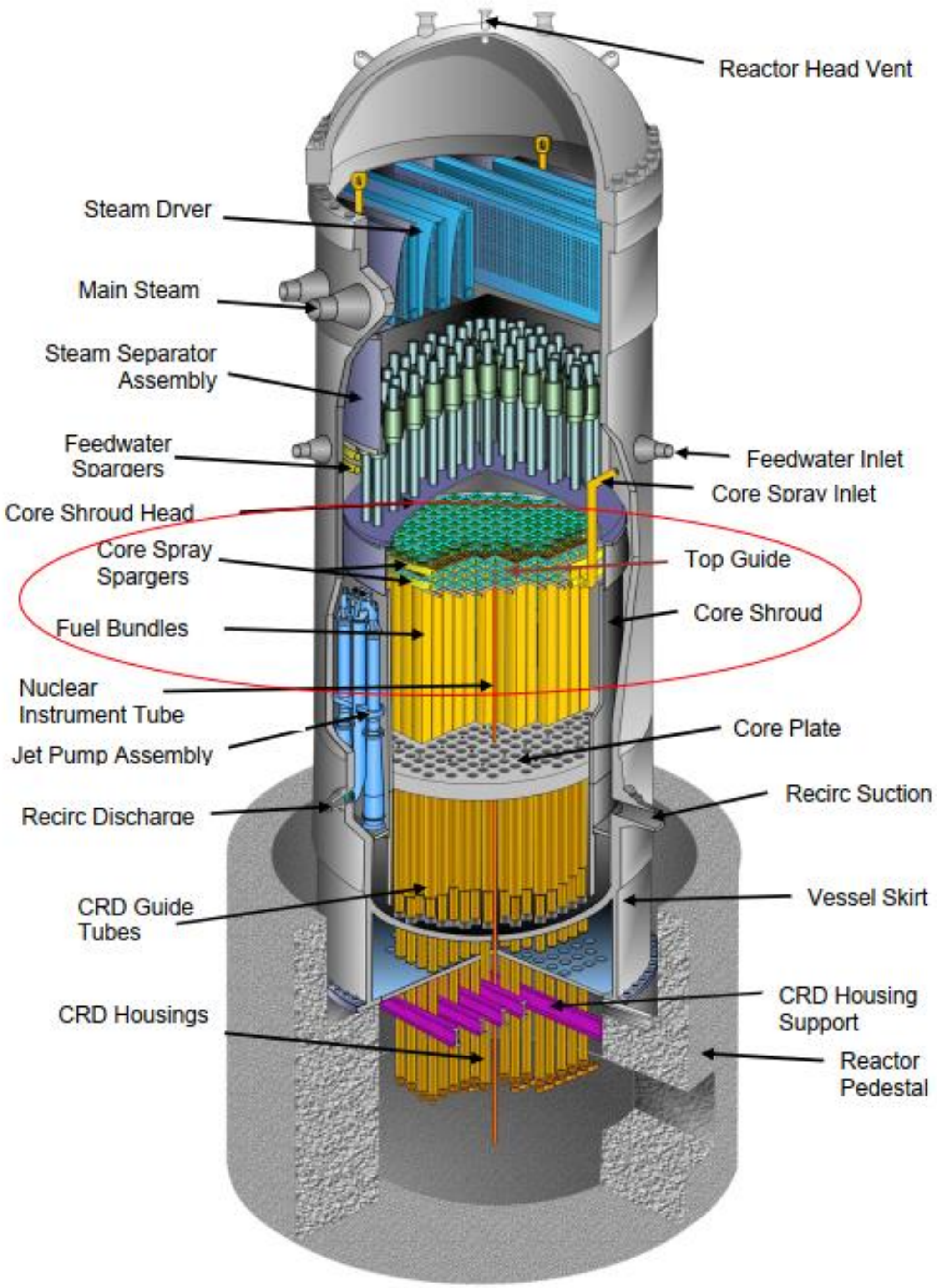


Figure-1 Reactor Vessel Assembly

QA Record. Non-RP - Retain in ECM (Lifetime Retention)
RP LPs - Retain in ECM (Life of Nuclear Insurance Policy, plus 10 years)
Page 38 of 62

Examination Outline Cross-reference:

600000 (APE 24) Plant Fire On Site / 8

AA2.18 (10CFR 55.43.5 - SRO Only)

Ability to determine and/or interpret the following as they apply to Plant Fire on Site:

- Assessment of control room habitability

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	600000AA2.18	
Importance Rating	------	3.6

Proposed Question: **# 82**

Unit 3 is operating at 100% RTP when the following conditions occur:

At 1000

- A fire is reported in the Unit 3 Main Control Room (MCR)

At 1010

- The Unit 3 MCR is abandoned

At 1025

- RCIC is controlling Reactor Water Level and MSRVs are controlling Reactor Pressure from 3-Panel-25-32, BACKUP CONTROL PANEL

Given the conditions above and in accordance with 3-AOI-100-2, Control Room Abandonment, the responsibility of assessing MCR habitability is performed by the ___**(1)**___.

In accordance with EPIP-1, Emergency Classification Procedure, the **HIGHEST** required Emergency Classification to report is a/an ___**(2)**___.

[REFERENCE PROVIDED]

A. (1) Shift Manager
(2) ALERT

B. (1) Shift Manager
(2) SITE AREA EMERGENCY

C. (1) Incident Commander
(2) ALERT

D. (1) Incident Commander
(2) SITE AREA EMERGENCY

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with 3-AOI-100-2, Control Room Abandon states that the Shift Manager/Unit SRO have the primary responsibility for implementing this procedure. For ALL situations, they make the decisions regarding actions to preclude evacuation, the evacuation itself, assignment of people, and returning to the control room. For second part, in accordance with EPIP-1, Emergency Classification Procedure, the **HIGHEST** required Emergency Classification to report is an ALERT - IC HA6 since control of parameters has been established within the 20 minutes window.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible if Backup Control of Reactor Water Level and Reactor Pressure (RCS heat removal) cannot be established within 20 minutes, a Site Area Emergency – IC HS6 is warranted.

C INCORRECT: First part is incorrect but plausible in that in accordance with EPIP-17, Fire Emergency Procedure, the Incident Commander stays in communication with the Shift Manager. There is an extensive list of responsibilities for the Incident Commander including plant alignments for combating the fire. Although the Incident Commander works with the fire brigade and is an SRO, the individual in that position does not have the authority to make a final decision regarding the Main Control Room habitability, only recommendations. Additionally, in accordance with OPDP-1, Conduct of Operations, the Incident Commander will be a shift SRO not assigned to a Unit or the STA role. 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 knowledge of the assessment of Main Control Room habitability during a plant fire and apply applicable emergency procedures. 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 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 3-AOI-100-2, Rev. 27 (Attach if not previously provided)
EPIP-1, Rev. 61
EPIP-17, Rev. 36
OPDP-1, Rev. 53

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)

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

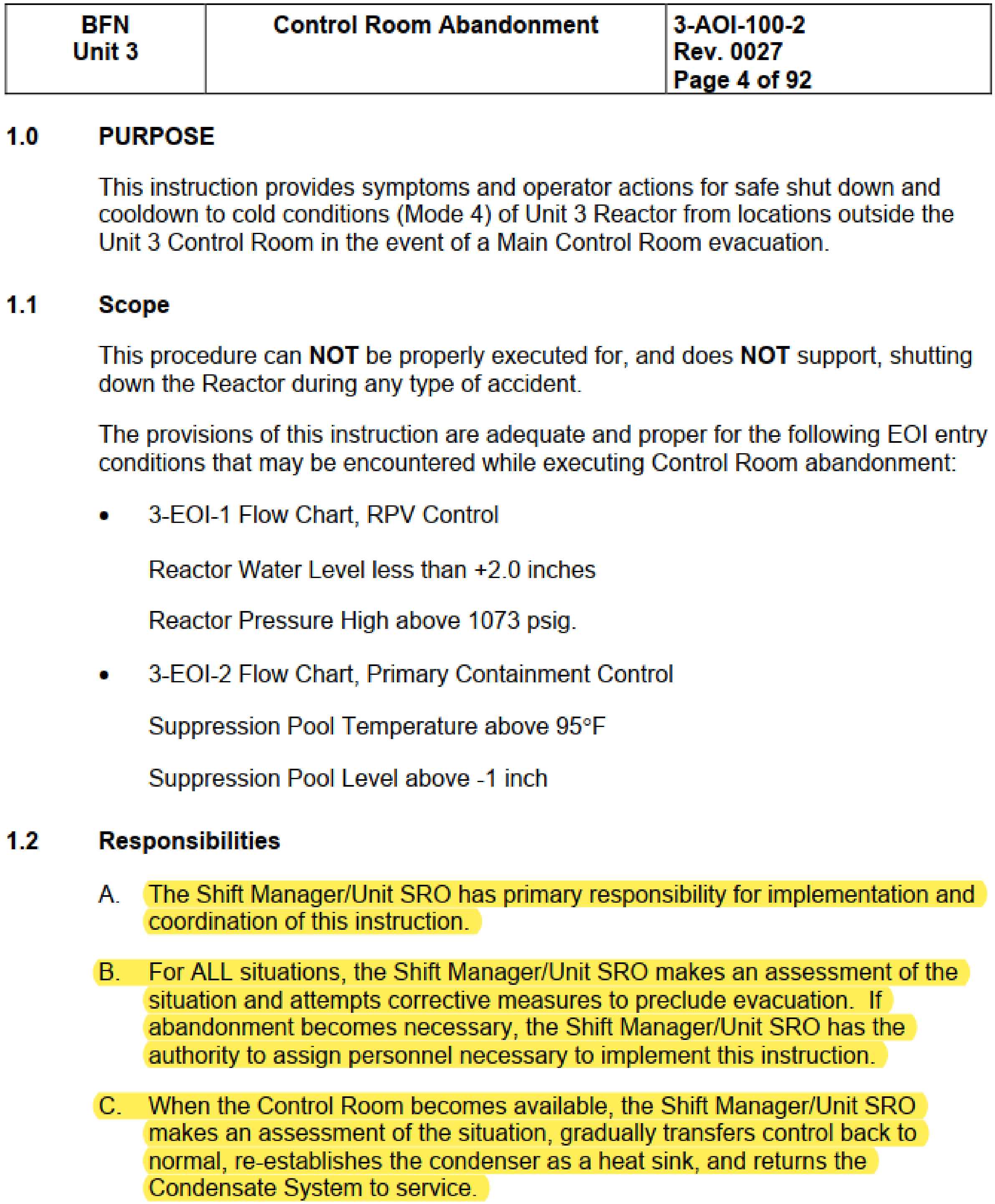
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

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

Comments:

Excerpt from 3-AOI-100-2:



BFN **Unit 3**	**Control Room Abandonment**	**3-AOI-100-2** **Rev. 0027** **Page 4 of 92**

1.0 PURPOSE

This instruction provides symptoms and operator actions for safe shut down and cooldown to cold conditions (Mode 4) of Unit 3 Reactor from locations outside the Unit 3 Control Room in the event of a Main Control Room evacuation.

1.1 Scope

This procedure can **NOT** be properly executed for, and does **NOT** support, shutting down the Reactor during any type of accident.

The provisions of this instruction are adequate and proper for the following EOI entry conditions that may be encountered while executing Control Room abandonment:

- 3-EOI-1 Flow Chart, RPV Control

  Reactor Water Level less than +2.0 inches

  Reactor Pressure High above 1073 psig.

- 3-EOI-2 Flow Chart, Primary Containment Control

  Suppression Pool Temperature above 95°F

  Suppression Pool Level above -1 inch

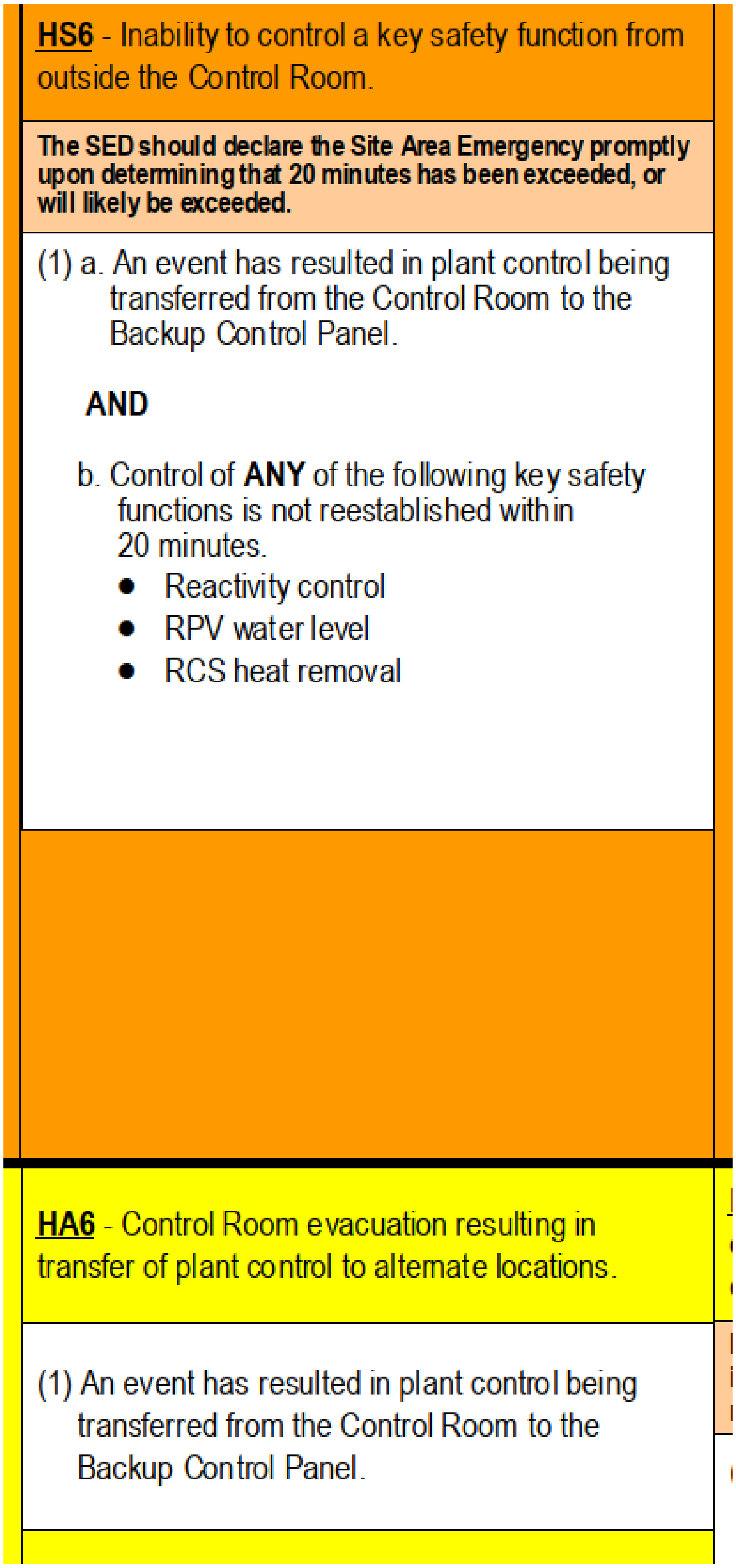
1.2 Responsibilities

A. The Shift Manager/Unit SRO has primary responsibility for implementation and coordination of this instruction.

B. For ALL situations, the Shift Manager/Unit SRO makes an assessment of the situation and attempts corrective measures to preclude evacuation. If abandonment becomes necessary, the Shift Manager/Unit SRO has the authority to assign personnel necessary to implement this instruction.

C. When the Control Room becomes available, the Shift Manager/Unit SRO makes an assessment of the situation, gradually transfers control back to normal, re-establishes the condenser as a heat sink, and returns the Condensate System to service.

Excerpt from EPIP-1, HOT INITIATING CONDTIONS-MODES 1-2-3:



HS6 - Inability to control a key safety function from outside the Control Room.

The SED should declare the Site Area Emergency promptly upon determining that 20 minutes has been exceeded, or will likely be exceeded.

(1) a. An event has resulted in plant control being transferred from the Control Room to the Backup Control Panel.

AND

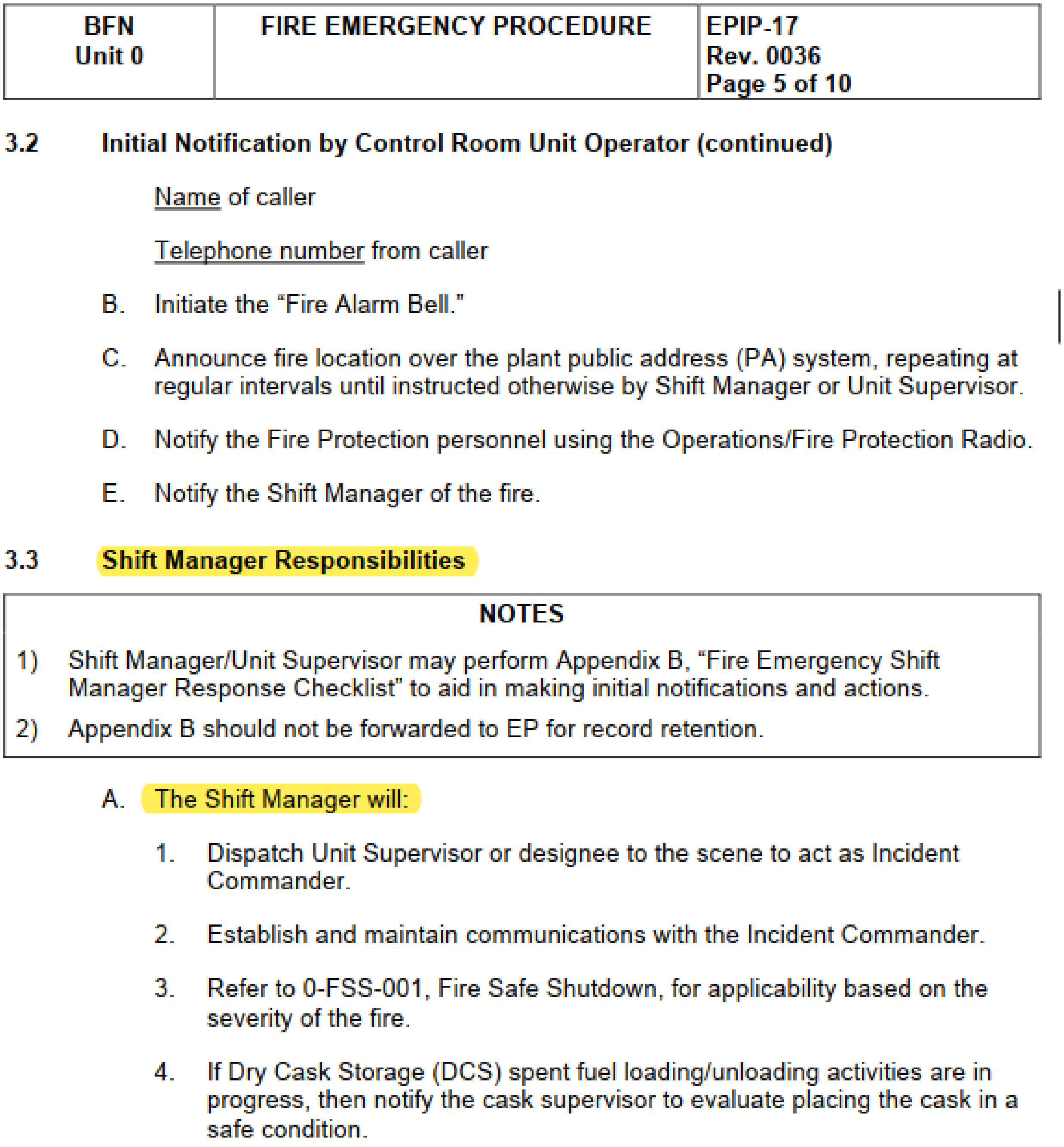
b. Control of **ANY** of the following key safety functions is not reestablished within 20 minutes.

- Reactivity control
- RPV water level
- RCS heat removal

HA6 - Control Room evacuation resulting in transfer of plant control to alternate locations.

(1) An event has resulted in plant control being transferred from the Control Room to the Backup Control Panel.

Excerpt from EPIP-17: Also supports Distractors C(1), D(1)



BFN **Unit 0**	**FIRE EMERGENCY PROCEDURE**	**EPIP-17** **Rev. 0036** **Page 5 of 10**

3.2 Initial Notification by Control Room Unit Operator (continued)

Name of caller

Telephone number from caller

B. Initiate the "Fire Alarm Bell."

C. Announce fire location over the plant public address (PA) system, repeating at regular intervals until instructed otherwise by Shift Manager or Unit Supervisor.

D. Notify the Fire Protection personnel using the Operations/Fire Protection Radio.

E. Notify the Shift Manager of the fire.

3.3 Shift Manager Responsibilities

NOTES

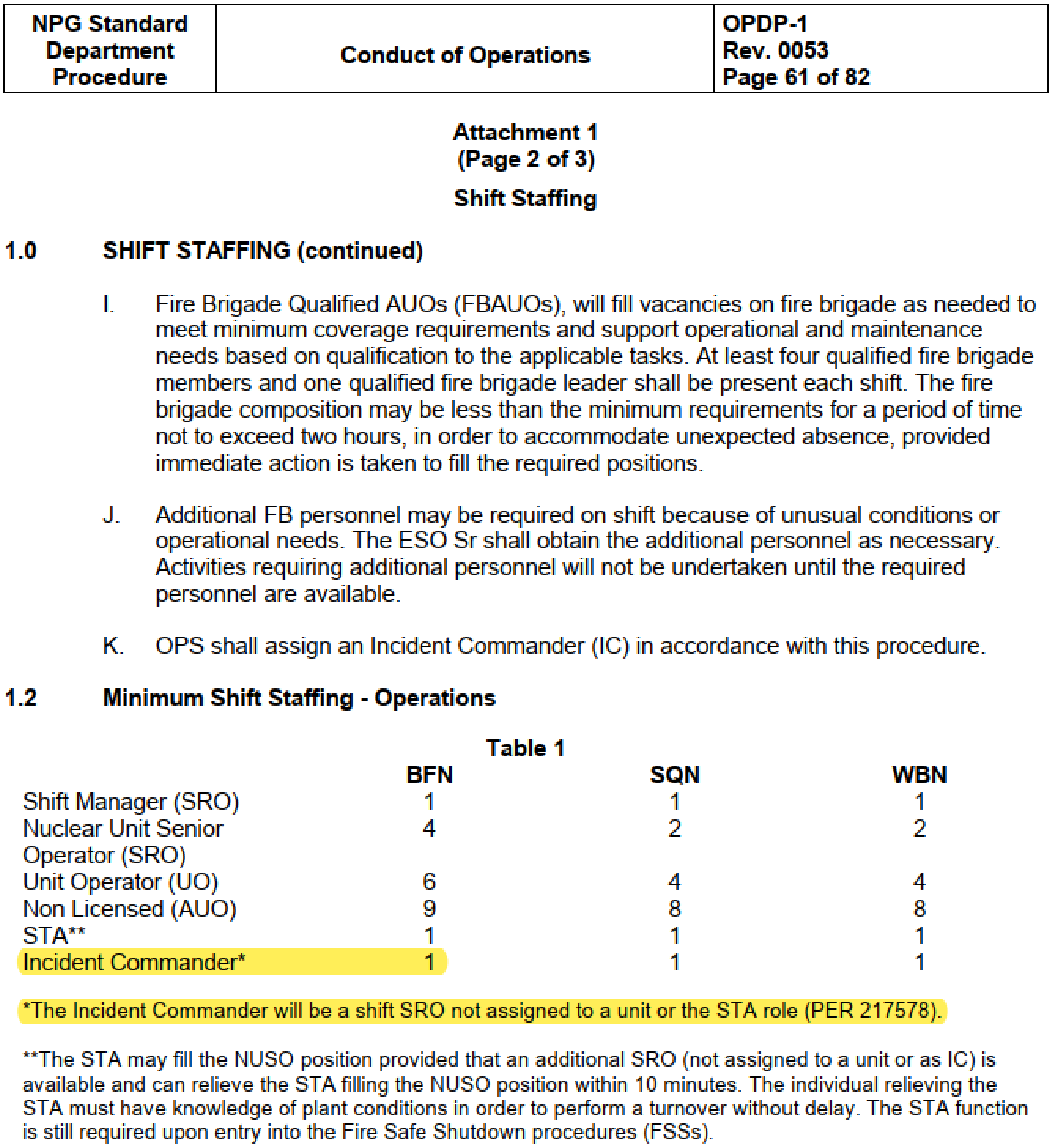
1) Shift Manager/Unit Supervisor may perform Appendix B, "Fire Emergency Shift Manager Response Checklist" to aid in making initial notifications and actions.

2) Appendix B should not be forwarded to EP for record retention.

A. The Shift Manager will:

1. Dispatch Unit Supervisor or designee to the scene to act as Incident Commander.
2. Establish and maintain communications with the Incident Commander.
3. Refer to 0-FSS-001, Fire Safe Shutdown, for applicability based on the severity of the fire.
4. If Dry Cask Storage (DCS) spent fuel loading/unloading activities are in progress, then notify the cask supervisor to evaluate placing the cask in a safe condition.

Excerpt from OPDP-1: Supports Distractors C(1), D(1)



NPG Standard Department Procedure	**Conduct of Operations**	**OPDP-1** **Rev. 0053** **Page 61 of 82**

Attachment 1
(Page 2 of 3)

Shift Staffing

1.0 SHIFT STAFFING (continued)

I. Fire Brigade Qualified AUOs (FBAUOs), will fill vacancies on fire brigade as needed to meet minimum coverage requirements and support operational and maintenance needs based on qualification to the applicable tasks. At least four qualified fire brigade members and one qualified fire brigade leader shall be present each shift. The fire brigade composition may be less than the minimum requirements for a period of time not to exceed two hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

J. Additional FB personnel may be required on shift because of unusual conditions or operational needs. The ESO Sr shall obtain the additional personnel as necessary. Activities requiring additional personnel will not be undertaken until the required personnel are available.

K. OPS shall assign an Incident Commander (IC) in accordance with this procedure.

1.2 Minimum Shift Staffing - Operations

Table 1

	BFN	**SQN**	**WBN**
Shift Manager (SRO)	1	1	1
Nuclear Unit Senior Operator (SRO)	4	2	2
Unit Operator (UO)	6	4	4
Non Licensed (AUO)	9	8	8
STA**	1	1	1
Incident Commander*	1	1	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).

Examination Outline Cross-reference:

295008 (APE 8) High Reactor Water Level / 2

AA2.01 (10CFR 55.43.5 – SRO Only)

Ability to determine and/or interpret the following as they apply to High Reactor Water Level

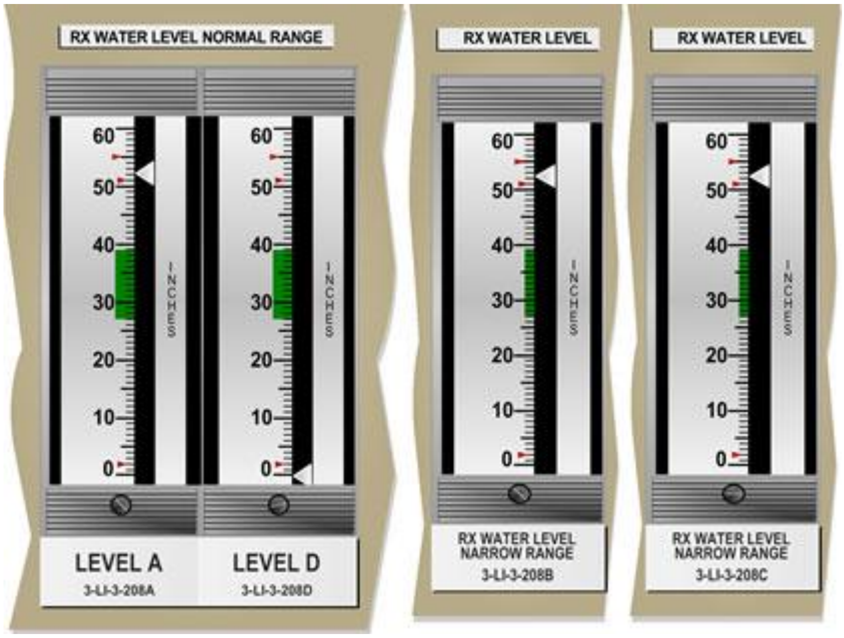
- Reactor Water Level

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295008AA2.01	
Importance Rating	------	4.4

Proposed Question: **# 83**

Unit 3 is operating at 100% RTP when a Feedwater Control System malfunction occurs with the following conditions:

- **3-LI-3-208D**, RX WATER LEVEL NARROW RANGE, fails **DOWNSCALE**
- Reactor Water Level peaked as indicated
- Tech Spec 3.3.2.2, Condition A has been entered



Given the conditions above, the Unit 3 Main Turbine is **CURRENTLY ___(1)___**.

A few minutes later, **3-LI-3-208C**, RX WATER LEVEL NARROW RANGE, fails **DOWNSCALE** in the next minute, LCO 3.3.2.2 entry into Condition B **___(2)___** required.

[REFERENCE PROVIDED]

A. (1) tripped
(2) is

B. (1) tripped
(2) is **NOT**

C. (1) operating
(2) is

D. (1) operating
(2) is **NOT**

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The first part is incorrect but plausible in that the indicated Reactor Water Level is above one of the two red turbine 'trip' setpoint indicators (as given). This could easily be confused with HPCI or RCIC High Reactor Water Level trip setpoints at (+) 51 inches. 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 if the candidate confuses the High Reactor Water Level trip logic to think that both Channels (3-LI-3-208D and 3-LI-3-208C) are in the same trip logic system, versus two separate trip logic systems. The logic is arranged in two Channels; Channel A is fed from 3-LI-3-208A and 3-LI-3-208C while Channel B is fed from 3-LI-3-208B and 3-LI-3-208D.

C **CORRECT**: (*See attached*) In accordance with 3-OI-47, Turbine-Generator System, High Reactor Water Level Trip logic for the Main Turbine at (+) 55 inches is taken from Narrow Range Level instruments 3-LI-3-208A/B/C/D. The logic is arranged in two channels; Channel A is fed from 3-LI-3-208A and 3-LI-3-208C and Channel B is fed from 3-LI-3-208B and 3-LI-3-208D. A trip of the Main Turbine and RFPTs will occur if **both** instruments in Channel A **OR** Channel B sense Reactor Water Level at ≥ (+) 55 inches. There are two red turbine 'trip' setpoint indicators on these four instruments (as given), but the three given OPERABLE instruments currently only indicate above (+) 51 inches. This will ONLY result in a HPCI and RCIC Turbine High Level trip; NOT a Main Turbine trip. Therefore, the Main Turbine is still operating. For second part, in accordance with Tech Spec 3.3.2.2, as soon as **any** one channel is INOPERABLE, Condition A is entered. As long as any one channel is INOPERABLE, Condition A won't be exited. With 3-LI-3-208D (Channel B trip logic) being the first INOPERABLE, when 3-LI-3-208C (Channel A trip logic) fails downscale, this results in one failure in **each** trip system, requiring entry into Condition B as well.

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

SRO Level Justification: Test the candidate's ability to monitor and interpret High Reactor Water Level instrument indications as it applies to Main Turbine trip logic. 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 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. The candidate must read the multiple indications correctly, integrate them together, and recall the specific facts associated with each to reach a conclusion.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s):	3-OI-47, Rev. 13	(Attach if not previously provided)
	3-OI-71, Rev. 58	
	3-OI-73, Rev. 63	
	U3 Tech Spec 3.3.2.2, Amend. 283	
	U3 Tech Spec Bases 3.3.2.2, Amend. 213	

Proposed references to be provided to applicants during examination: **Unit 3 Tech Spec 3.3.2.2 (No Bases), 3-LI-3-208A, B, C, D – REACTOR WATER LEVEL NORMAL NARROW RANGE**

Learning Objective: OPL171.010 Obj. 12 (As available)

Question Source:	Bank #	BFN ILT 1804 #78	
	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	
	55.43	X

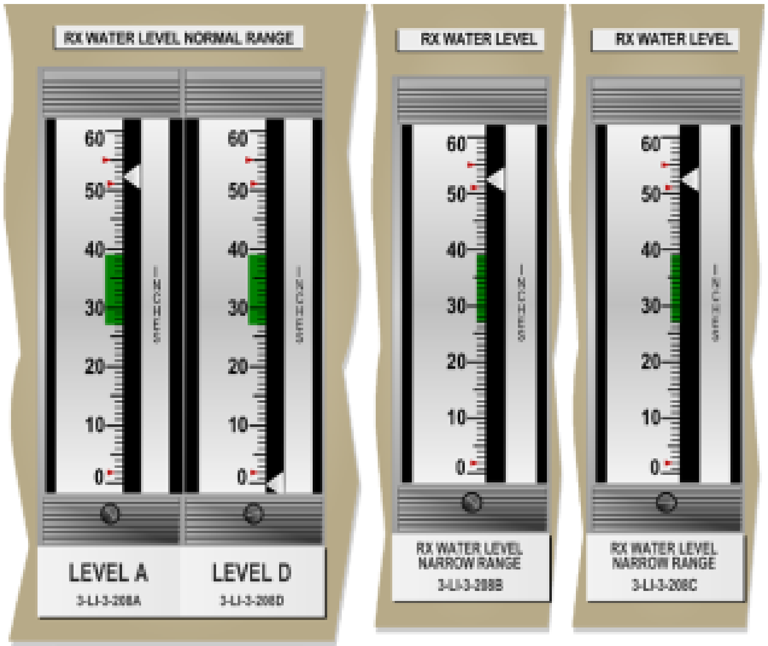
Comments:

Copy of Bank Question:

ES-401 **Sample Written Examination Question Worksheet** **Form ES-401-5**

Unit 3 is operating at 100% RTP when a Feedwater Control System malfunction occurs with plant conditions as follows:

- **3-LI-3-208D**, RX WATER LEVEL NARROW RANGE, fails **downscale**
- Reactor Water Level peaked as indicated
- Operators are taking action to restore Reactor Water Level
- Tech Spec 3.3.2.2, Condition A has been entered



Based upon the indications provided, which ONE of the following completes the statements below?

The Unit 3 Main Turbine is **CURRENTLY** ___**(1)**___.

IF a second instrument **3-LI-3-208C**, RX WATER LEVEL NARROW RANGE, fails **downscale** in the next minute, LCO 3.3.2.2 entry into Condition B ___**(2)**___required.

[REFERENCE PROVIDED]

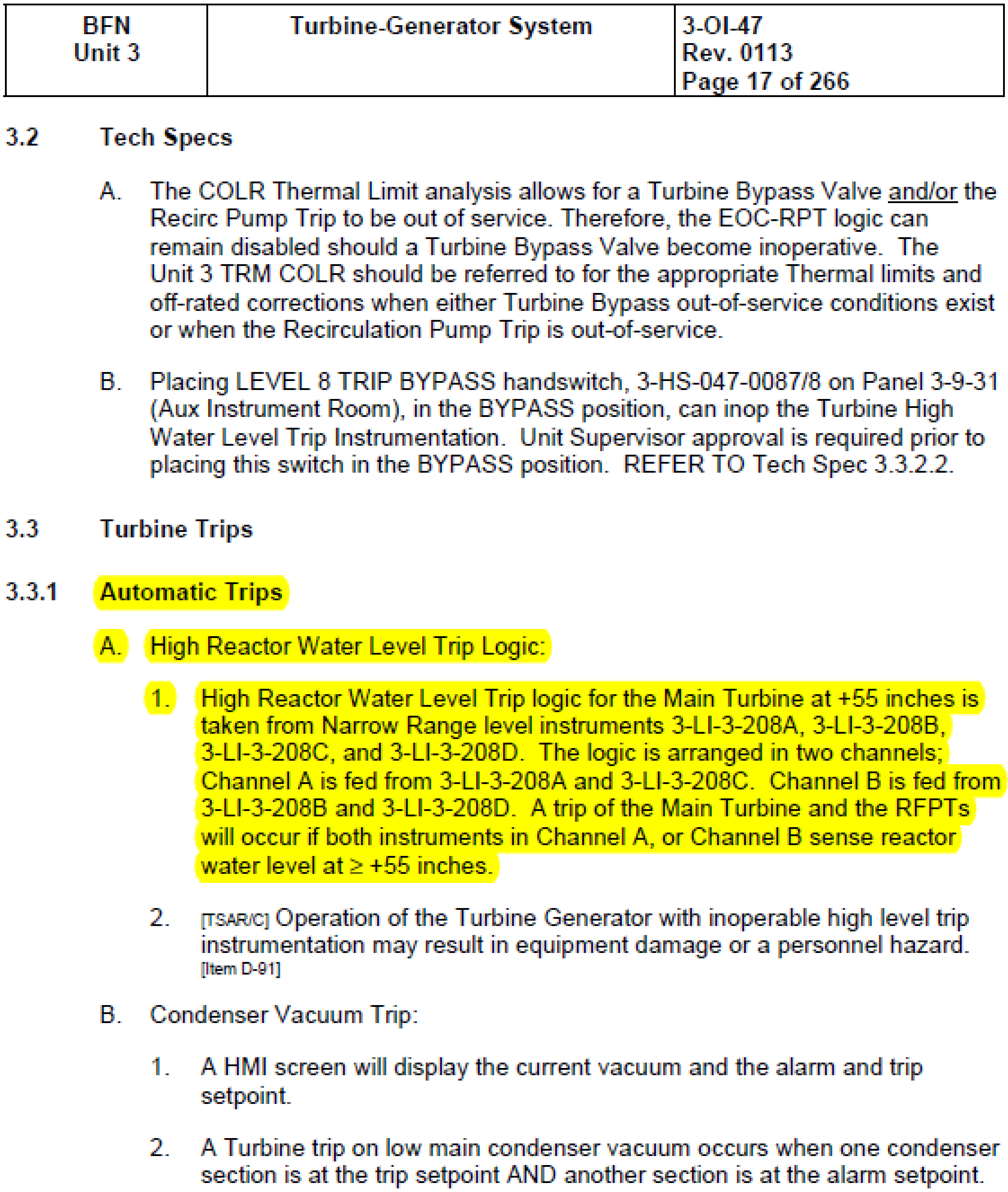
A. **(1)** tripped
(2) is NOT

B. **(1)** tripped
(2) is

C. **(1)** operating
(2) is NOT

D. **(1)** operating
(2) is

Excerpt from 3-OI-47:



BFN **Unit 3**	**Turbine-Generator System**	**3-OI-47** **Rev. 0113** **Page 17 of 266**

3.2 Tech Specs

A. The COLR Thermal Limit analysis allows for a Turbine Bypass Valve and/or the Recirc Pump Trip to be out of service. Therefore, the EOC-RPT logic can remain disabled should a Turbine Bypass Valve become inoperative. The Unit 3 TRM COLR should be referred to for the appropriate Thermal limits and off-rated corrections when either Turbine Bypass out-of-service conditions exist or when the Recirculation Pump Trip is out-of-service.

B. Placing LEVEL 8 TRIP BYPASS handswitch, 3-HS-047-0087/8 on Panel 3-9-31 (Aux Instrument Room), in the BYPASS position, can inop the Turbine High Water Level Trip Instrumentation. Unit Supervisor approval is required prior to placing this switch in the BYPASS position. REFER TO Tech Spec 3.3.2.2.

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 3-LI-3-208A, 3-LI-3-208B, 3-LI-3-208C, and 3-LI-3-208D. The logic is arranged in two channels; Channel A is fed from 3-LI-3-208A and 3-LI-3-208C. Channel B is fed from 3-LI-3-208B and 3-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.

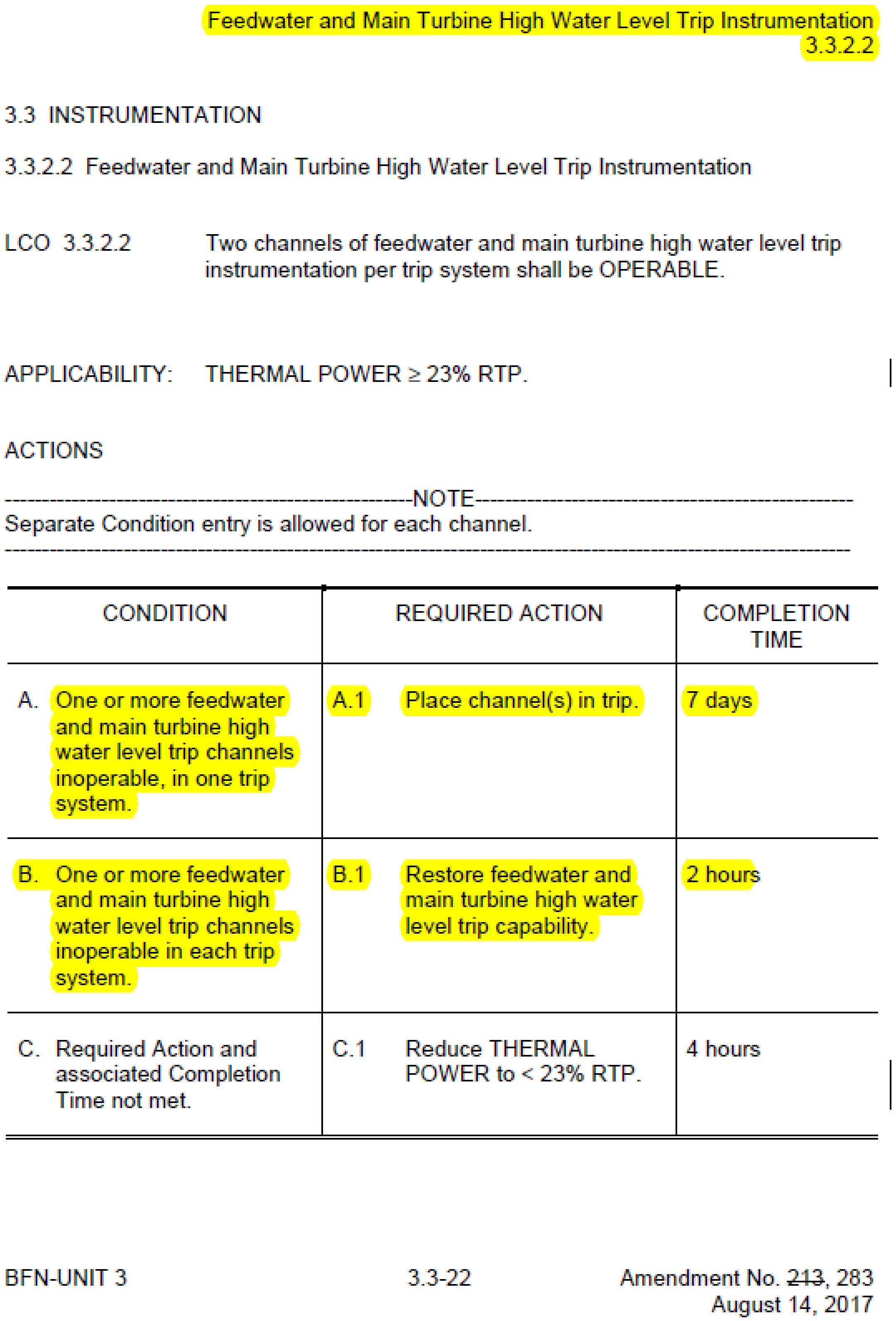
2. [TSAR/C] Operation of the Turbine Generator with inoperable high level trip instrumentation may result in equipment damage or a personnel hazard. [Item D-91]

B. Condenser Vacuum Trip:

1. A HMI screen 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 Unit 3 Tech Spec 3.3.2.2:



Feedwater and Main Turbine High Water Level Trip Instrumentation
3.3.2.2

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Two channels of feedwater and main turbine high water level trip instrumentation per trip system shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more feedwater and main turbine high water level trip channels inoperable, in one trip system.	A.1 Place channel(s) in trip.	7 days
B. One or more feedwater and main turbine high water level trip channels inoperable in each trip system.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

BFN-UNIT 3 3.3-22 Amendment No. ~~213~~, 283
August 14, 2017

Excerpt from Unit 3 Tech Spec Bases 3.3.2.2:

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

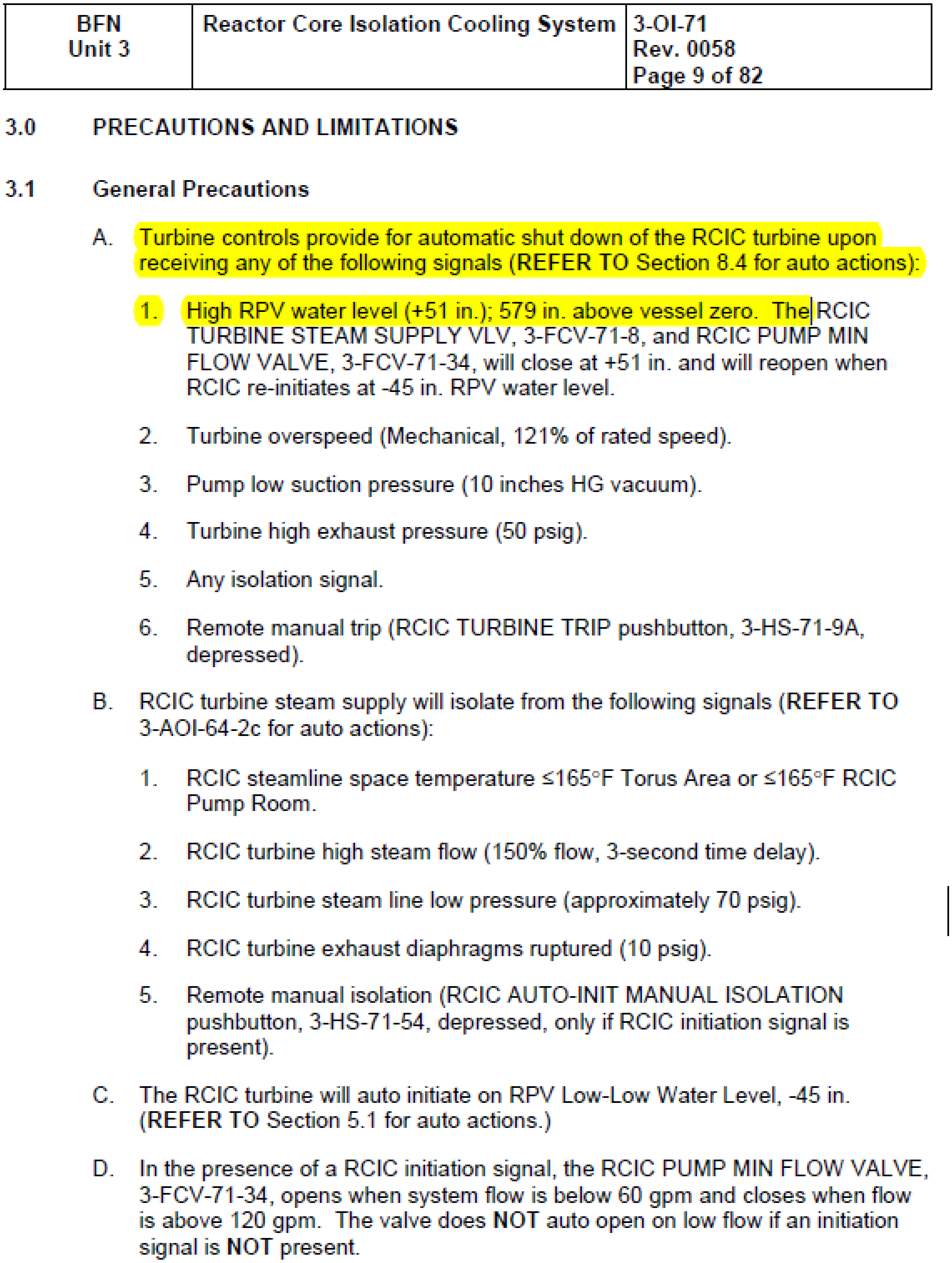
With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level reference point, causing the trip of the three feedwater pump turbines and the main turbine.

Reactor Vessel Water Level - High signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Two channels of Reactor Vessel Water Level - High instrumentation per trip system are provided as input to a two-out-of-two initiation logic that trips the three feedwater pump turbines and the main turbine. There are two trip systems, either of which will initiate a trip. The channels include electronic equipment, LS-3-208A, LS-3-208B, LS-3-208C, and LS-3-208D (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a main feedwater and turbine trip signal to the trip logic.

A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

(continued)

Excerpt from 3-OI-71: Supports Distractors A(1), B(1)



BFN Unit 3	Reactor Core Isolation Cooling System	3-OI-71 Rev. 0058 Page 9 of 82

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions

A. Turbine controls provide for automatic shut down 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, 3-FCV-71-8, and RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34, will close at +51 in. and will reopen when RCIC re-initiates at -45 in. RPV water level.
2. Turbine overspeed (Mechanical, 121% 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 pushbutton, 3-HS-71-9A, depressed).

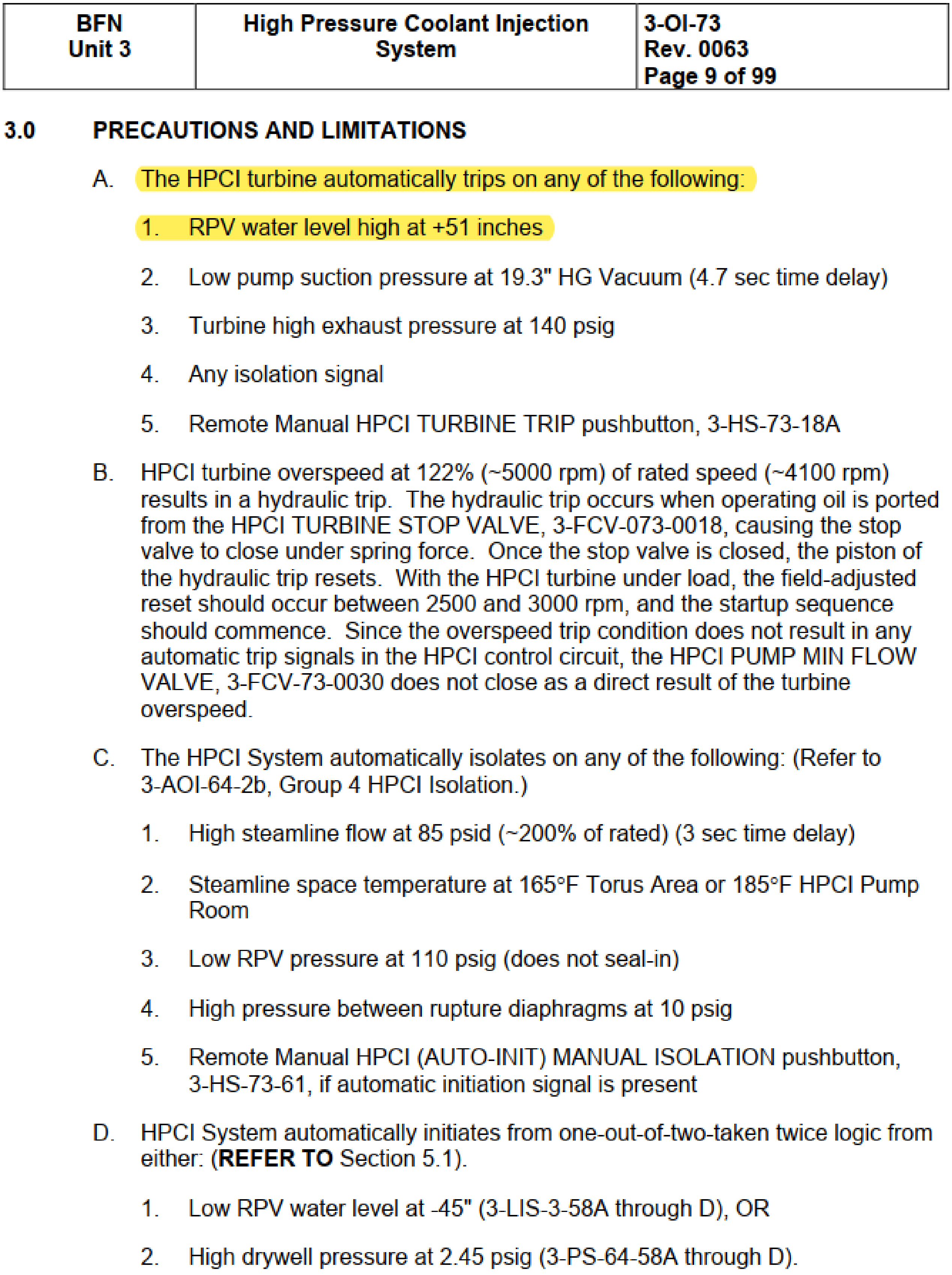
B. RCIC turbine steam supply will isolate from the following signals (**REFER TO** 3-AOI-64-2c for auto actions):

1. RCIC steamline space temperature ≤165°F Torus Area or ≤165°F RCIC Pump Room.
2. RCIC turbine high steam flow (150% flow, 3-second time delay).
3. RCIC turbine steam line low pressure (approximately 70 psig).
4. RCIC turbine exhaust diaphragms ruptured (10 psig).
5. Remote manual isolation (RCIC AUTO-INIT MANUAL ISOLATION pushbutton, 3-HS-71-54, depressed, only if RCIC initiation signal is present).

C. The RCIC turbine will auto initiate on RPV Low-Low Water Level, -45 in. (**REFER TO** Section 5.1 for auto actions.)

D. In the presence of a RCIC initiation signal, the RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34, opens when system flow is below 60 gpm and closes when flow is above 120 gpm. The valve does **NOT** auto open on low flow if an initiation signal is **NOT** present.

Excerpt from 3-OI-73: Supports Distractors A(1), B(1)



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.

C. The HPCI System automatically isolates on any of the following: (Refer to 3-AOI-64-2b, Group 4 HPCI Isolation.)

1. High steamline flow at 85 psid (~200% of rated) (3 sec time delay)
2. Steamline space temperature at 165°F Torus Area or 185°F HPCI Pump Room
3. Low RPV pressure at 110 psig (does not seal-in)
4. High pressure between rupture diaphragms at 10 psig
5. Remote Manual HPCI (AUTO-INIT) MANUAL ISOLATION pushbutton, 3-HS-73-61, if automatic initiation signal is present

D. HPCI System automatically initiates from one-out-of-two-taken twice logic from either: (**REFER TO** Section 5.1).

1. Low RPV water level at -45" (3-LIS-3-58A through D), OR
2. High drywell pressure at 2.45 psig (3-PS-64-58A through D).

Examination Outline Cross-reference:

295009 (APE 9) Low Reactor Water Level / 2

G2.4.18 (10CFR 55.43.1 - SRO Only)

Knowledge of the specific bases for emergency and abnormal operating procedures

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295009G2.4.18	
Importance Rating	------	4.0

Proposed Question: **# 84**

Unit 1 is operating in MODE 1 when the following conditions occur:

- Reactor SCRAM, all Control Rods inserted
- **NO** high **OR** low pressure make-up sources are available

In accordance with the EOIs, Emergency Depressurization is required **BEFORE** Reactor Water Level lowers to **(1)** .

In accordance with the EOIPM SECTION 0-V(L), C1, ALTERNATE LEVEL/PRESSURE CONTROL BASES, Minimum Zero Injection RPV Water Level (MZIRWL) lower limit ensures fuel clad temperatures will **NOT** exceed **(2)** .

A. (1) (-) 235 inches
(2) 1500 °F

B. (1) (-) 235 inches
(2) 1800 °F

C. (1) (-) 162 inches
(2) 1500 °F

D. (1) (-) 162 inches
(2) 1800 °F

Proposed Answer: **B**

Explanation (Optional):	A	INCORRECT: First part is correct *(See B)*. Second part is incorrect but plausible in that in accordance with EOIPM Section 0-V(C), EOI-1, RPV Control Modes 1-3 Bases, Minimum Steam Cooling RPV Water Level (MSCRWL) is the lowest Reactor Water Level WITH injection at which the covered portion of the Reactor core can be cooled to ensure peak clad temperature will not exceed 1500 °F.

B **CORRECT:** (*See attached*) In accordance with 1-EOI-1, RPV CONTROL MODES 1 – 3, Step RC-3 – It is anticipated that available Injection Subsystems (Table L-3) alone CANNOT assure adequate core cooling, the NUSO must transition to 1-C-1, Alternate Level/Pressure Control, since no high or low pressure injection sources are available. Additionally, as Reactor Water Level lowers below (-) 162 inches (Top of Active Fuel – TAF), STEAM COOLING would be required. Emergency Depressurization is required when Reactor Water Level drops to (-) 235 inches. For second part, in accordance with EOIPM Section 0-V(L), C1 Alternate Level/Control Bases, Minimum Zero-Injection RPV Water Level (MZIRWL) is where the steam produced by decay heat in the covered portion of the core provides some cooling to the uncovered portion through convective heat transfer without any available injection. Given this, adequate core cooling for this condition is defined to exist as long as the peak clad temperature remains below 1800 ºF which is expected to be a higher temperature than with some injection while Reactor Water Level is allowed to be lower prior to Emergency Depressurization.

C INCORRECT: First part is incorrect but plausible in that in accordance with 1-EOI-1, with injection sources available and Reactor Water Level could not be maintained above (-) 162 inches, an Emergency Depressurization can be performed. Below that level, the fuel has is uncovered. With Reactor Water Level below TAF, It, is plausible from the candidate to think action is required up to the point of Emergency Depressurization. Since there are no sources of injection, C1 Alternate Level/Control entry is required and Emergency Depressurizations is not allowed at TAF. Steam Cooling entry would be required with Emergency depressurization being required before (-) 235 inches Reactor Water Level. 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 knowledge of specific Emergency Operating Instruction Program Bases as it relates to Low Reactor Water Level. SRO only because of the link to 10CFR55.43 (1): Conditions and limitations in the facility license. 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 accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 1-EOI-1, Rev. 7 (Attach if not previously provided)
1-C-1, Rev. 7
EOIPM Section 0-V(C), Rev. 1
EOIPM Section 0-V(L) Rev.1

Proposed references to be provided to applicants during examination: **NONE**

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

Learning Objective: OPL171.205 Obj. 1 (As available)

Question Source:	Bank #		
	Modified Bank #	OPL171.205-02 002 #2966	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:

Memory or Fundamental Knowledge

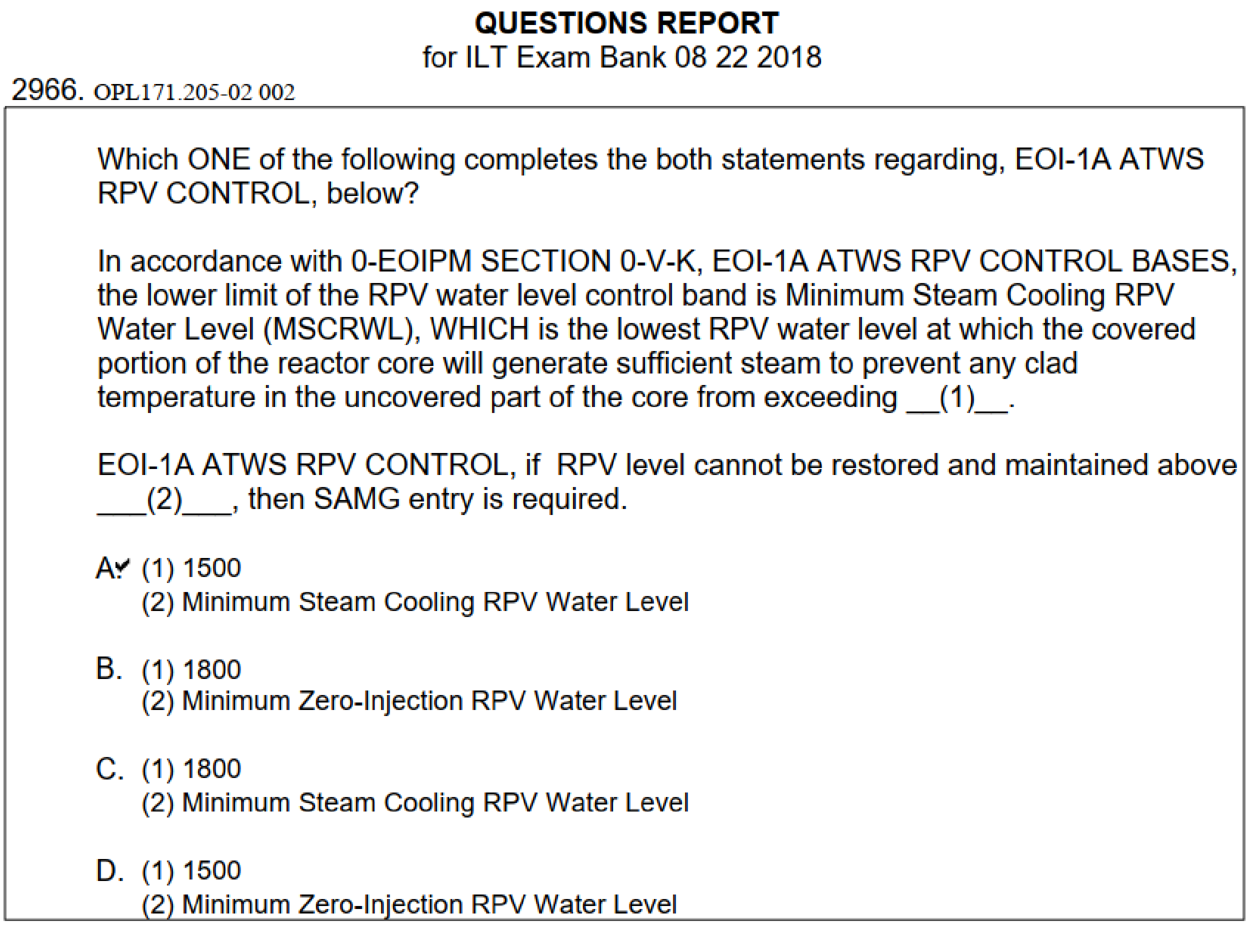
Comprehension or Analysis **X**

10 CFR Part 55 Content:

55.41

55.43 **X**

Copy of Bank Question:



QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

2966. OPL171.205-02 002

Which ONE of the following completes the both statements regarding, EOI-1A ATWS RPV CONTROL, below?

In accordance with 0-EOIPM SECTION 0-V-K, EOI-1A ATWS RPV CONTROL BASES, the lower limit of the RPV water level control band is Minimum Steam Cooling RPV Water Level (MSCRWL), WHICH is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding __(1)__.

EOI-1A ATWS RPV CONTROL, if RPV level cannot be restored and maintained above ___(2)___, then SAMG entry is required.

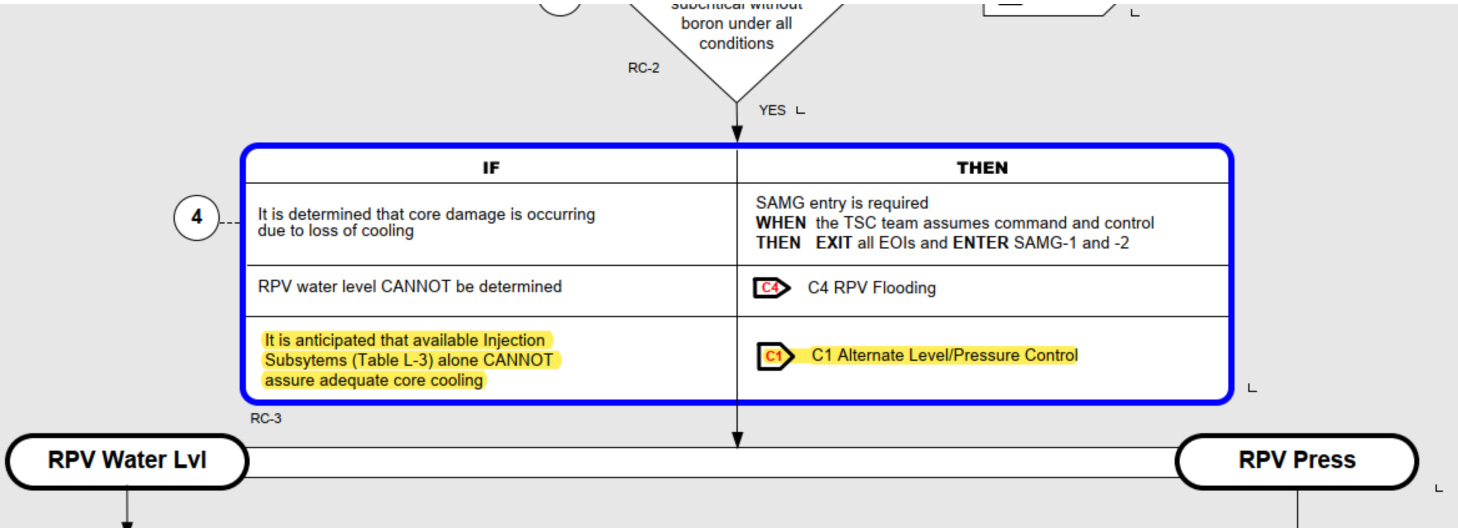
A✓ (1) 1500
(2) Minimum Steam Cooling RPV Water Level

B. (1) 1800
(2) Minimum Zero-Injection RPV Water Level

C. (1) 1800
(2) Minimum Steam Cooling RPV Water Level

D. (1) 1500
(2) Minimum Zero-Injection RPV Water Level

Excerpts from 1-EOI-1:



IF	THEN
It is determined that core damage is occurring due to loss of cooling	SAMG entry is required **WHEN** the TSC team assumes command and control **THEN** **EXIT** all EOIs and **ENTER** SAMG-1 and -2
RPV water level CANNOT be determined	C4 C4 RPV Flooding
It is anticipated that available Injection Subsytems (Table L-3) alone CANNOT assure adequate core cooling	C1 C1 Alternate Level/Pressure Control

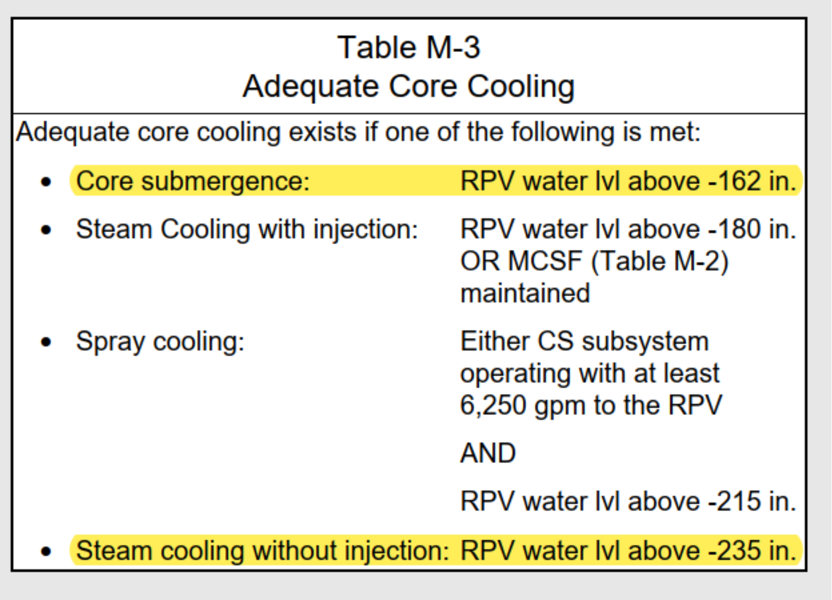
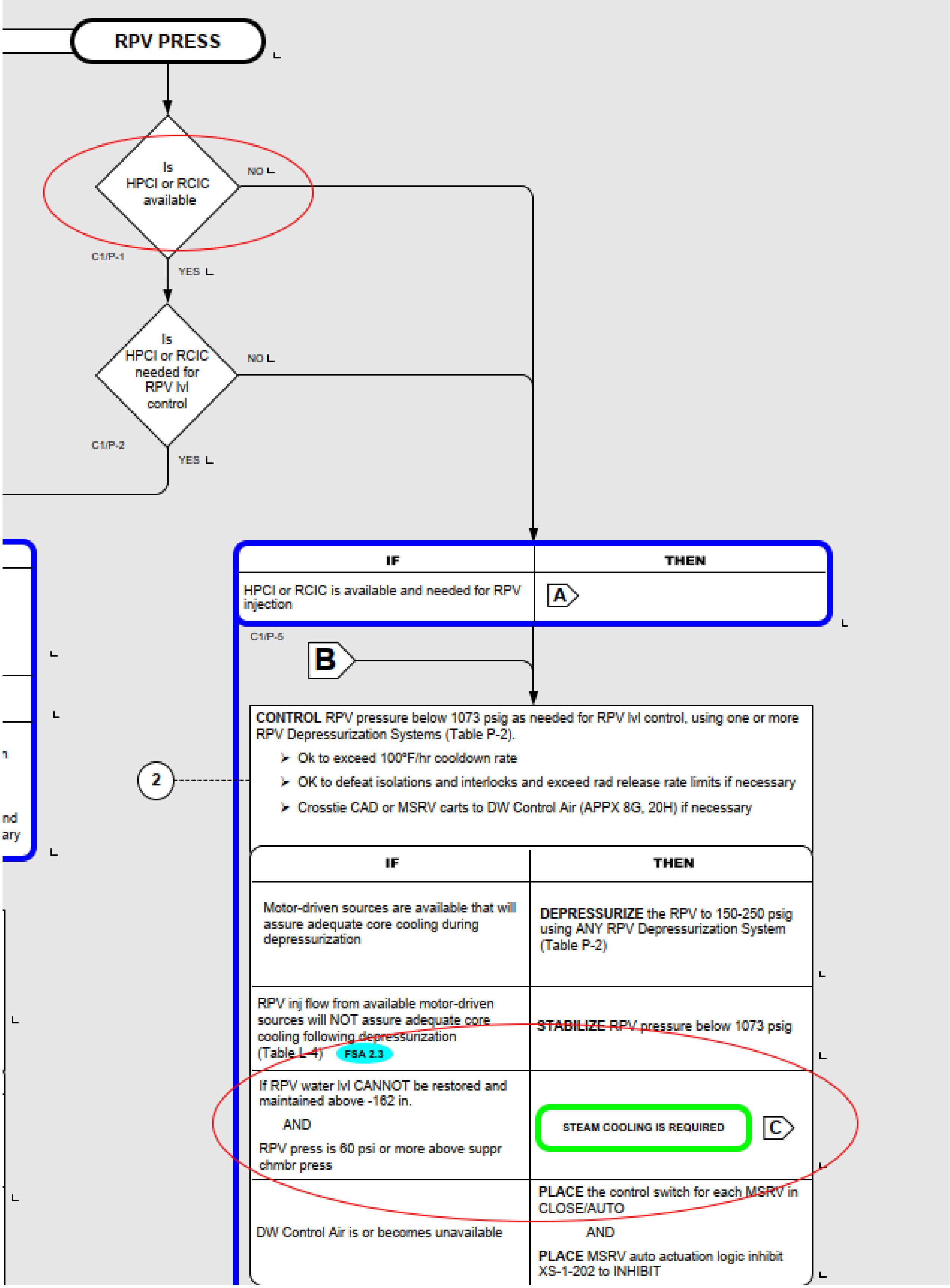


Table M-3
Adequate Core Cooling

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. OR MCSF (Table M-2) maintained
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 lvl above -235 in.

Excerpts from 1-C-1:

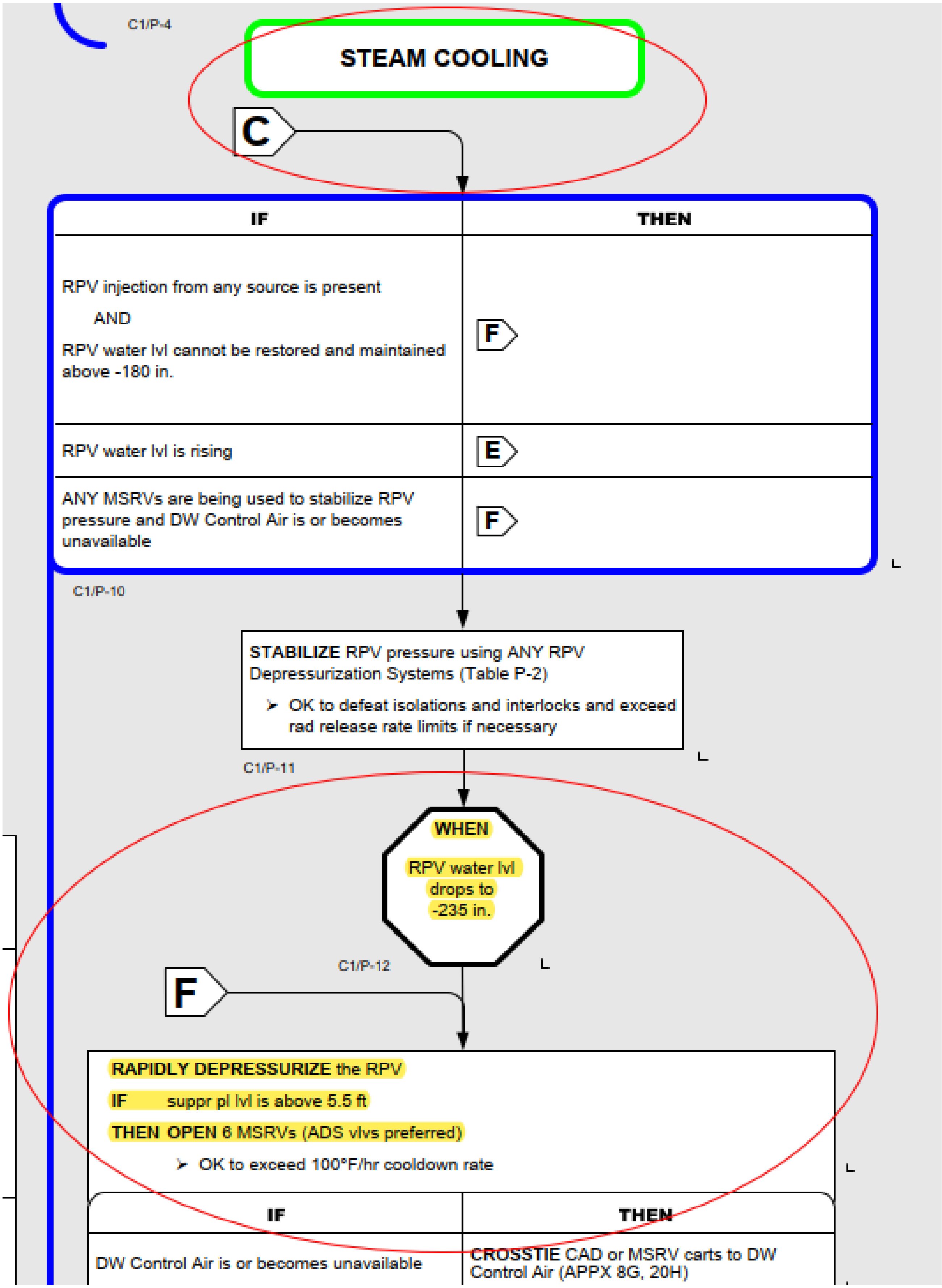


IF	THEN
HPCI or RCIC is available and needed for RPV injection	A

CONTROL RPV pressure below 1073 psig as needed for RPV lvl control, using one or more RPV Depressurization Systems (Table P-2).

- Ok to exceed 100°F/hr cooldown rate
- OK to defeat isolations and interlocks and exceed rad release rate limits if necessary
- Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary

IF	THEN
Motor-driven sources are available that will assure adequate core cooling during depressurization	**DEPRESSURIZE** the RPV to 150-250 psig using ANY RPV Depressurization System (Table P-2)
RPV inj flow from available motor-driven sources will NOT assure adequate core cooling following depressurization (Table L-4)	**STABILIZE** RPV pressure below 1073 psig
If RPV water lvl CANNOT be restored and maintained above -162 in. AND RPV press is 60 psi or more above suppr chmbr press	STEAM COOLING IS REQUIRED C
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



C1/P-4

STEAM COOLING

C

IF	THEN
RPV injection from any source is present AND RPV water lvl cannot be restored and maintained above -180 in.	F
RPV water lvl is rising	E
ANY MSRVs are being used to stabilize RPV pressure and DW Control Air is or becomes unavailable	F

C1/P-10

STABILIZE RPV pressure using ANY RPV Depressurization Systems (Table P-2)

- OK to defeat isolations and interlocks and exceed rad release rate limits if necessary

C1/P-11

WHEN

RPV water lvl drops to -235 in.

C1/P-12

F

RAPIDLY DEPRESSURIZE the RPV

IF suppr pl lvl is above 5.5 ft

THEN OPEN 6 MSRVs (ADS vlvs preferred)

- OK to exceed 100°F/hr cooldown rate

IF	THEN
DW Control Air is or becomes unavailable	**CROSSTIE** CAD or MSRV carts to DW Control Air (APPX 8G, 20H)

Excerpts from EOIPM Section 0-V(L):

BFN **Unit 0**	**C-1, Alternate Level/Pressure Control**	**EOIPM Section 0-V(L)** **Rev. 0001** **Page 75 of 93**

1.0 C1, ALTERNATE LEVEL/PRESSURE CONTROL BASES (continued)

DISCUSSION: Flowpath C, Steam Cooling Overview

Steam Cooling in accordance with Flowpath C is performed to prolong the time that adequate core cooling is assured when RPV water level cannot be restored and maintained above the top of the active fuel and either no RPV injection source is available or RPV pressure is above the maximum injection pressure of available injection sources. Adequate core cooling for this condition is defined to exist as long as the peak clad temperature remains below 1800°F, the threshold for significant metal-water reaction.

A conceptual plot of fuel peak clad temperature (PCT) vs. time during steam cooling is illustrated in Figure C1-1. At (1) the core begins to uncover and PCT begins to increase. As boiloff continues between (1) and (2), the steam produced by decay heat in the covered portion of the core provides some cooling to the uncovered portion through convective heat transfer. During this time, fuel temperatures in the uncovered region rise to an equilibrium value at which heat transfer to the steam is sufficient to remove the decay heat generated. As water level decreases, the equilibrium fuel temperature must increase since (a) less steam is produced in the covered region and (b) more heat is added to the steam in the uncovered region. At (2) the steam generated by the covered portion of the core is exactly sufficient to remove the heat generated in the uncovered region with PCT at 1800°F. The level at which this correspondence occurs is designated the Minimum Zero-Injection RPV Water Level (MZIRWL).

When RPV water level drops below the MZIRWL at (2), steam cooling may no longer be sufficient to preclude the peak clad temperature from exceeding 1800°F. MSRVs are then opened to increase steam flow and rapidly depressurize the RPV. Unless the RPV is already fully depressurized, it is expected that the resulting swell will be sufficient to quench the uncovered portion of the fuel and reduce PCT almost to the value that would exist if the core were submerged. As the swell subsides and steam flow through the open MSRVs decreases at (3) however, PCT turns and again rises. If no injection source can be made available, PCT will eventually exceed 1800°F. The steam cooling strategy defined in Flowpath C thus delays the emergency depressurization as long as possible.

1.0 C1, ALTERNATE LEVEL/PRESSURE CONTROL BASES (continued)

DISCUSSION: Flowpath C, Steam Cooling Overview (Cont'd)

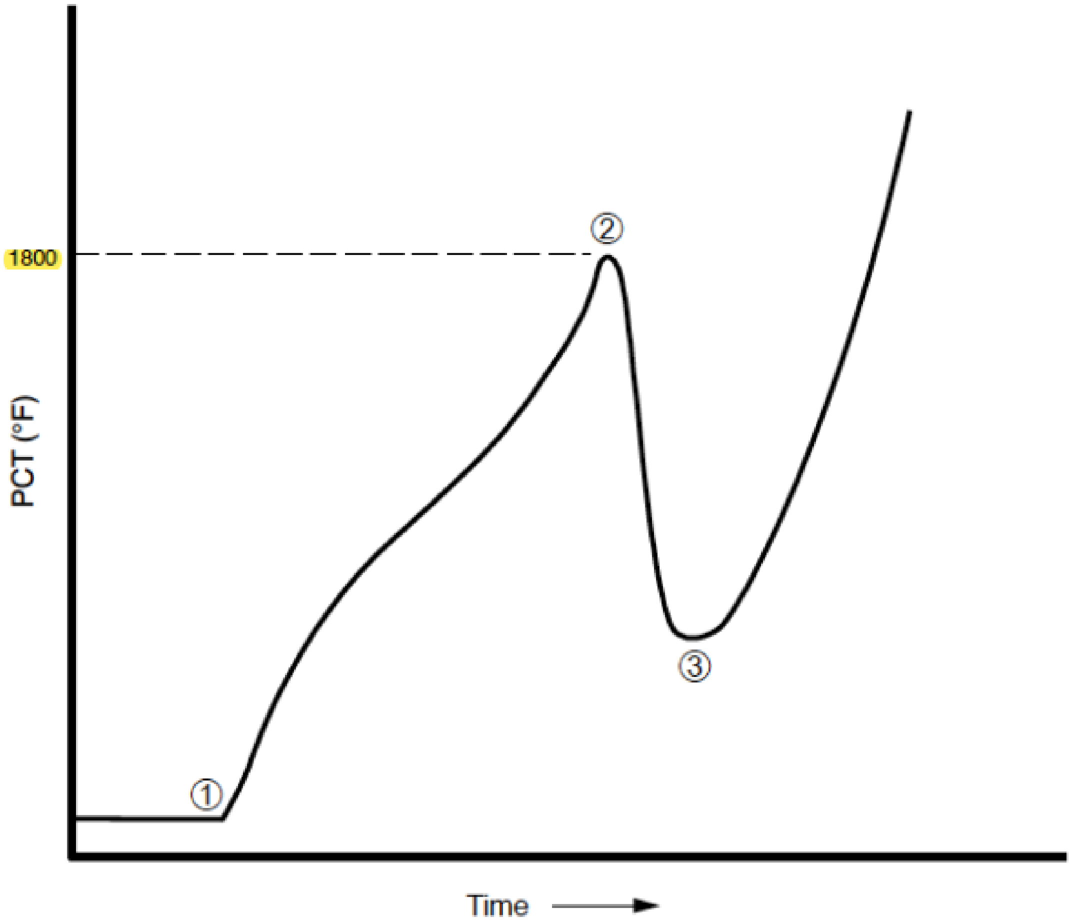


Figure C1-1: Conceptual PCT Profile for Steam Cooling

Excerpts from EOIPM Section 0-V(C): Supports Distractors A(2), C(2)

BFN **Unit 0**	**EOI-1, RPV Control Modes 1-3 Bases**	**EOIPM Section 0-V(C)** **Rev. 0001** **Page 53 of 102**

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/L-11 Provisional Action

Adequate core cooling (Table M-3) is ensured following emergency RPV depressurization if:

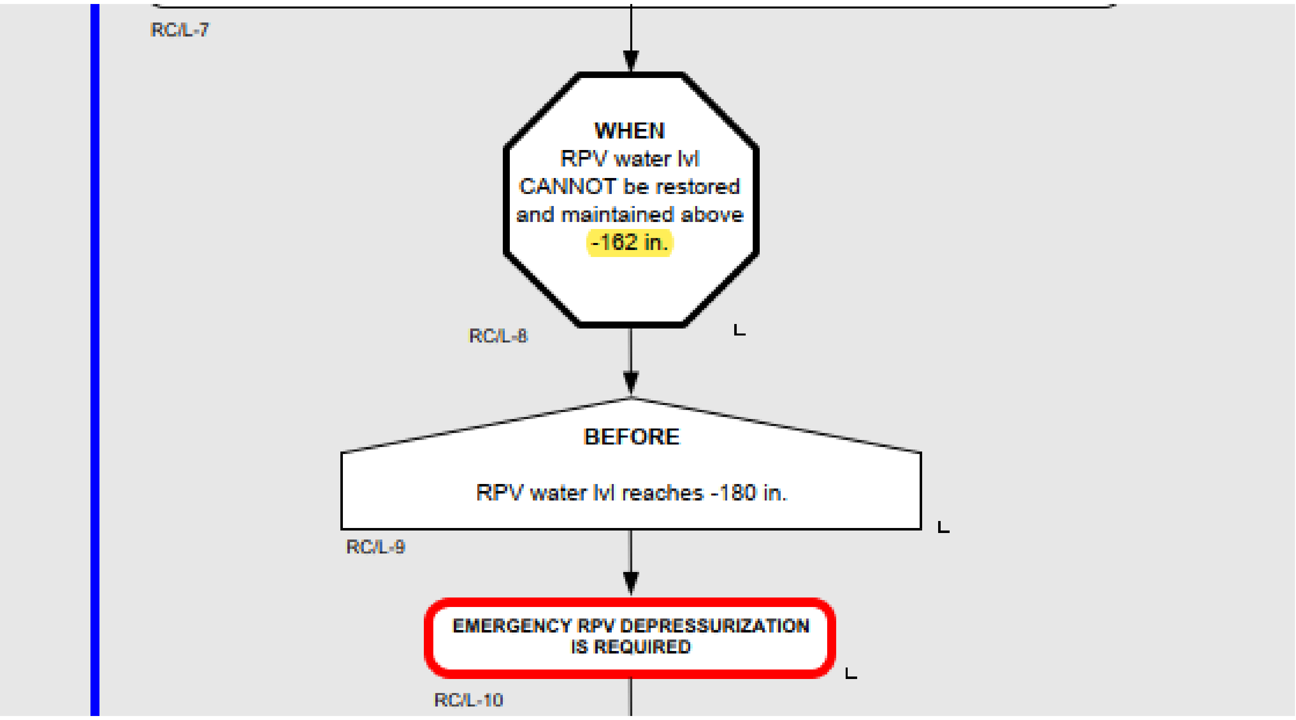
- RPV water level can be restored and maintained above the Minimum Steam Cooling RPV Water Level (MSCRWL). The core is then cooled by a combination of submergence and steam cooling.
- Design core spray flow requirements are satisfied and RPV water level can be restored and maintained at or above the jet pump suctions.

The MSCRWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F assuming the most limiting top-peaked power shape prior to reactor shutdown. Refer to EOIPM Section 0-V(B) for discussion of the MSCRWL.

Following a large recirculation line break, it may not be possible to restore and maintain RPV water level above the MSCRWL if the available injection capacity is insufficient to overcome flow through the break. The core may still be considered adequately cooled, however, if design basis core spray flow requirements are satisfied. Following a design basis recirculation line break, RPV water level is expected to stabilize at two-thirds core height which is slightly below the MSCRWL. The covered portion of the core is then cooled by submergence while the uncovered portion is cooled by spray flow.

If RPV water level can be restored and maintained above the MSCRWL or spray cooling conditions can be established, the core will remain adequately cooled and no further action need be taken. Efforts to restore RPV water level above the top of the active fuel should continue, but at some point it may become necessary to throttle ECCS flow to observe NPSH restrictions or divert flow to containment cooling functions in accordance with design basis assumptions. The system operating details and cautions listed in Tables L-1 and L-2 continue to apply; NPSH and vortex limits should be observed if possible, but may be exceeded if the situation warrants.

Excerpt from 1-EOI-1: Supports Distractors C(1), D(1)



Examination Outline Cross-reference:

295035 (EPE 12) Secondary Containment High Differential Pressure / 5

EA2.01 (10CFR 55.43.5 - SRO Only)

Ability to determine and/or interpret the following as they apply to Secondary Containment High Differential Pressure:

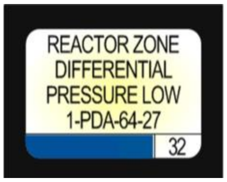
- Secondary Containment Pressure

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295035EA2.01	
Importance Rating	------	3.9

Proposed Question: **# 85**

Unit 1 is operating at 100% RTP when the following conditions occur:

- REACTOR BUILDING VENTILATION ABNORMAL (1-9-3D, Window 3) alarms
- REACTOR ZONE DIFFERENTIAL PRESSURE LOW (1-9-3D, Window 32) alarms
- AUO reports REACTOR ZONE DIFFERENTIAL PRESSURE is (+) .5 inches of water locally
- Assume **NO** Operator action has been taken



Given the conditions above, Standby Gas Treatment __**(1)**__ automatically started.

To mitigate this condition, the NUSO __**(2)**__ direct entry into 1-AOI-30B-1, Reactor Building Ventilation Failure.

A. **(1)** has
(2) will

B. **(1)** has
(2) will NOT

C. **(1)** has NOT
(2) will

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

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that a PCIS Group 6 Isolation signal of the Reactor or Refuel Zone Ventilation also would automatically start and align Standby Gas Train, however the given conditions do not cause any PCIS isolations. This is a Secondary Containment Static Pressure Isolation Lockout due solely to Reactor Building differential pressure. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that since the Reactor Building Ventilation System has tripped due to a differential pressure static isolation, the candidate may think that 1-AOI-64-2D, Group 6 Ventilation System Isolation, is appropriate. However, 1-AOI-64-2D addresses PCIS isolation and does not provide guidance for restoring from a static isolation.

C **CORRECT:** (*See attached*) In accordance with the given alarms, probable causes are trip of any Reactor Building supply and/or exhaust fan or Reactor Zone high differential pressure (-) 1.0 to (+) 0.5 inches of water. The given AUO report supports the alarms which indicate a Reactor Building Static Pressure Lockout has occurred. This causes a trip of the associated supply and exhaust fans and a closure of the fan dampers. However, in accordance with 0-OI-65, Standby Gas Treatment (SGT) System, this Lockout will NOT cause an automatic start of the SGT System. For second part, in accordance with 1-AOI-30B-1, Reactor Building Ventilation Failure, the NUSO will direct the restoration of ventilation for both the Reactor Zone and the Refuel Zones following a static isolation of high and low D/P. It also provides guidance for other system malfunctions not related to a PCIS isolation.

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

SRO Level Justification: Tests the candidate’s ability to determine the impact on the Standby Gas Treatment System from a Secondary Containment High Differential Pressure causing Reactor Building Ventilation Isolation and select the respective procedures for the given set of plant 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. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site’s procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents, and (2) diagnosis that leads to selection of the procedures that should be used to respond to the evolution.

Technical Reference(s): 0-OI-65, Rev. 55 (Attach if not previously provided)
1-ARP-9-3D, Rev. 32
1-AOI-64-2D, Rev. 21
1-AOI-30B-1, Rev. 14

Proposed references to be provided to applicants during examination: **REACTOR BUILDING VENTILATION ABNORMAL (1-9-3D, Window 3) REACTOR ZONE DIFFERENTIAL PRESSURE LOW (1-9-3D, Window 32)**

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

Learning Objective: OPL171.018, Obj. 4 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 19-09 #50	(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:

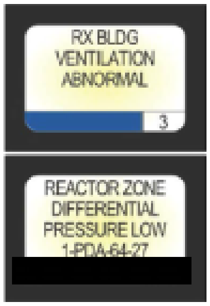
55.41	
55.43	**X**

Comments:

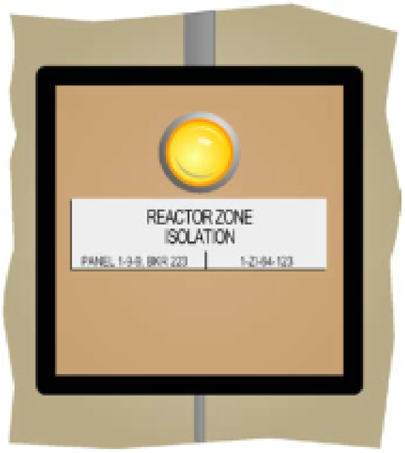
Copy of Bank Question:

Proposed Question: **# 50**

Unit 1 is operating at 100% RTP when the following conditions occur:



- REACTOR BUILDING VENTILATION ABNORMAL (1-9-3D, Window 3) alarms
- REACTOR ZONE DIFFERENTIAL PRESSURE LOW (1-9-3D, Window 32) alarms
- AUO reports REACTOR ZONE DIFFERENTIAL PRESSURE is (+) .5 inches of water locally
- Panel 1-9-25, amber light illuminates for REACTOR ZONE ISOLATION
- Assume **NO** Operator action has been taken



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

Given the conditions above, Standby Gas Treatment (SGT) ___**(1)**___ automatically started.

1-EOI-3, Secondary Containment Control, entry ___**(2)**___ required.

A. **(1)** has
(2) is

B. **(1)** has
(2) is NOT

C. **(1)** has NOT
(2) is

D. **(1)** has NOT
(2) is NOT

Proposed Answer: **C**

Excerpts from 1-ARP-9-3D:

BFN **Unit 1**	**Panel 9-3** **XA-55-3D**	**1-ARP-9-3D** **Rev. 0032** **Page 6 of 42**

RX BLDG
VENTILATION
ABNORMAL

3

(Page 1 of 1)

Sensor/Trip Point:

1-74-064-0011AC	5 sec TDPU, relay energized on loss of any supply or exhaust fan in Rx Bldg, or Refuel Floor.

Sensor Location:
Rx Building Vent Bd 1B behind Compt 4B
Rx Bldg, El 565', R-4 U-Line

Probable Cause:

A. Trip of any Rx Bldg or Refuel Floor Supply and/or Exhaust Fan.
B. PCIS Group 6 isolation, Rx and Refuel Zone Isolation.
C. Loss of 480V AC Rx Bldg Vent Board 1A or 1B.
D. Rx Zone or Refuel Zone high ∆P (-1.0 to +0.5 inch H_2O).

Automatic Action:

A. Fan Trip
   1. Closes associated discharge damper.
   2. Auto Starts standby fan.

B. PCIS Group 6 Isolation.
   1. All fans trip and discharge dampers close.
   2. SBGT auto starts.
   3. Control Bay Emergency Pressurization Units auto start.

Operator Action:

A. **IF** PCIS Group 6 isolation exists, **THEN**
   REFER TO 1-AOI-64-2d.
B. **NOTIFY** Unit 2 and Unit 3 Unit SRO.
C. **CHECK** standby fans start.
D. **DISPATCH** personnel to check Bldg ∆P (1-PDIC-064-0002, El 639 Rx Bldg).
E. **DISPATCH** personnel to check 480V AC Rx Bldg Vent Bd 1A and 1B.
F. **IF** unable to restore ventilation, **THEN**
   REFER TO 1-AOI-30B-1.

References:

1-45E620-2-2	1-45E614-5	1-45E755-1 and -2
1-45E777-3		

BFN **Unit 1**	**Panel 9-3** **XA-55-3D**	**1-ARP-9-3D** **Rev. 0032** **Page 38 of 42**

REACTOR ZONE
DIFFERENTIAL
PRESSURE LOW
1-PDA-64-27
32

(Page 1 of 1)

Sensor/Trip Point:

1-PDS-064-0027 -0.17 in of water

Sensor Location: 1-LPNL-925-0213
Reactor Bldg El. 639

Probable Cause:

A. Securing/Alternating Refuel Zone Fans.
B. Trip of any Rx Bldg Zone Exh. Fan.
C. PCIS Group 6 Isolation.
D. Differential Pressure switches fail closed.
E. Rapidly changing barometric pressure or high winds.
F. Normal ventilation in service with Standby Gas Treatment System running at the same time.
G. High energy line break in Secondary Containment.

Automatic Action: Annunciation only

Operator Action:

A. **IF** the alarm is intermittent, **THEN**
CHECK for high wind conditions (ex., > 20 mph) on ICS.
B. **IF** high wind conditions **CANNOT** be confirmed, **THEN**
REQUEST personnel check Bldg ΔP locally.
C. **REQUEST** personnel check fans locally for any apparent problems.
D. **REFER TO** 1-OI-30B and **PLACE** standby fan in service to restore normal ΔP.

References: 1-45E620-2-2 1-47E610-64-1

Excerpts from 1-AOI-30B-1:

BFN **Unit 1**	**Reactor Building Ventilation Failure**	**1-AOI-30B-1** **Rev. 0014** **Page 4 of 10**

1.0 PURPOSE

This Abnormal Operating Instruction provides symptoms, automatic action and operator action for the degradation or loss of Reactor Building or Refuel Zone ventilation for causes other than Group 6 Isolation. The Group 6 Isolation is addressed in another abnormal operating instruction.

2.0 SYMPTOMS

A. One or a combination of reactor or refuel zone supply or exhaust fans indicate shutdown on Panel 1-9-25.

B. One or more ventilation dampers in the ventilation flow path indicate closed on Panel 1-9-25.

C. Annunciator REAC BLDG VENTILATION ABNORMAL, XA-55-3D, Window 3, is in alarm.

D. Annunciator REACTOR ZONE DIFFERENTIAL PRESSURE LOW , XA-55-3D, Window 32, is in alarm.

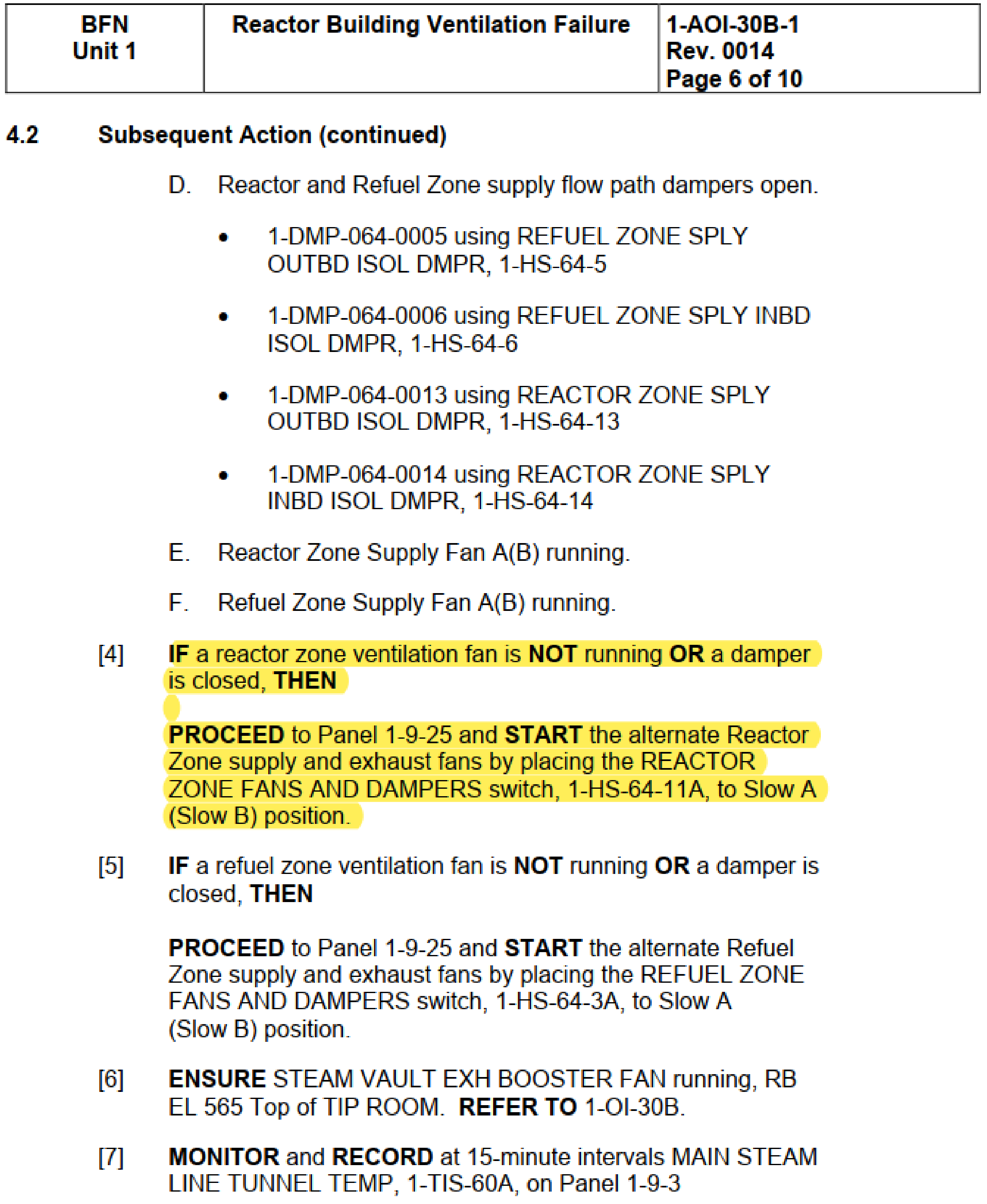
E. REF ZONE STATIC DIFF PRESS CONT, 1-PDIC-064-0002, on Panel 25-213 indicates Reactor Building pressure is NOT within -0.25 to -0.40 inch H_2O.

F. REF ZONE STATIC DIFF PRESS CONT, 1-PDIC-064-0001, on Panel 25-219 indicates Refuel Zone pressure is NOT within -0.25 to -0.40 inch H_2O.

3.0 AUTOMATIC ACTION

A. If Reactor Building pressure has risen to +0.5 inch H_2O or lowered to -1.0 inch H_2O, that Units Reactor ventilation system isolates.

B. If Refuel Zone pressure has risen to +0.5 inch H_2O or lowered to -1.0 inch H_2O, all Units Refuel Zone ventilation systems isolate.



4.2 Subsequent Action (continued)

D. Reactor and Refuel Zone supply flow path dampers open.

- 1-DMP-064-0005 using REFUEL ZONE SPLY OUTBD ISOL DMPR, 1-HS-64-5
- 1-DMP-064-0006 using REFUEL ZONE SPLY INBD ISOL DMPR, 1-HS-64-6
- 1-DMP-064-0013 using REACTOR ZONE SPLY OUTBD ISOL DMPR, 1-HS-64-13
- 1-DMP-064-0014 using REACTOR ZONE SPLY INBD ISOL DMPR, 1-HS-64-14

E. Reactor Zone Supply Fan A(B) running.

F. Refuel Zone Supply Fan A(B) running.

[4] **IF** a reactor zone ventilation fan is **NOT** running **OR** a damper is closed, **THEN**

PROCEED to Panel 1-9-25 and **START** the alternate Reactor Zone supply and exhaust fans by placing the REACTOR ZONE FANS AND DAMPERS switch, 1-HS-64-11A, to Slow A (Slow B) position.

[5] **IF** a refuel zone ventilation fan is **NOT** running **OR** a damper is closed, **THEN**

PROCEED to Panel 1-9-25 and **START** the alternate Refuel Zone supply and exhaust fans by placing the REFUEL ZONE FANS AND DAMPERS switch, 1-HS-64-3A, to Slow A (Slow B) position.

[6] **ENSURE** STEAM VAULT EXH BOOSTER FAN running, RB EL 565 Top of TIP ROOM. **REFER TO** 1-OI-30B.

[7] **MONITOR** and **RECORD** at 15-minute intervals MAIN STEAM LINE TUNNEL TEMP, 1-TIS-60A, on Panel 1-9-3

BFN **Unit 1**	**Reactor Building Ventilation Failure**	**1-AOI-30B-1** **Rev. 0014** **Page 7 of 10**

4.2 Subsequent Action (continued)

CAUTION

Main Steam isolation valves (MSIVs) will automatically isolate at 189 F (Group 1). In accordance with the Atwood & Morrill MSIV Instruction Manual, the MSIV solenoid pilot valves will operate properly at 200 F maximum continuous service, thus they should not be allowed to exceed 200 F.

[8] **NOTIFY** Unit SRO if MAIN STEAM LINE TUNNEL TEMP, 1-TIS-1-60A reaches 150°F.

[9] **IF** building pressure is more negative than -0.40 inch H_2O,
THEN

CHECK all ventilation supply fan suction dampers open locally.

[10] **IF** building pressure is more positive than -0.25 inch H_2O,
THEN

CHECK all ventilation exhaust fan discharge dampers open locally.

[11] **CHECK** locally all reactor and refuel supply and exhaust fans for normal operation:

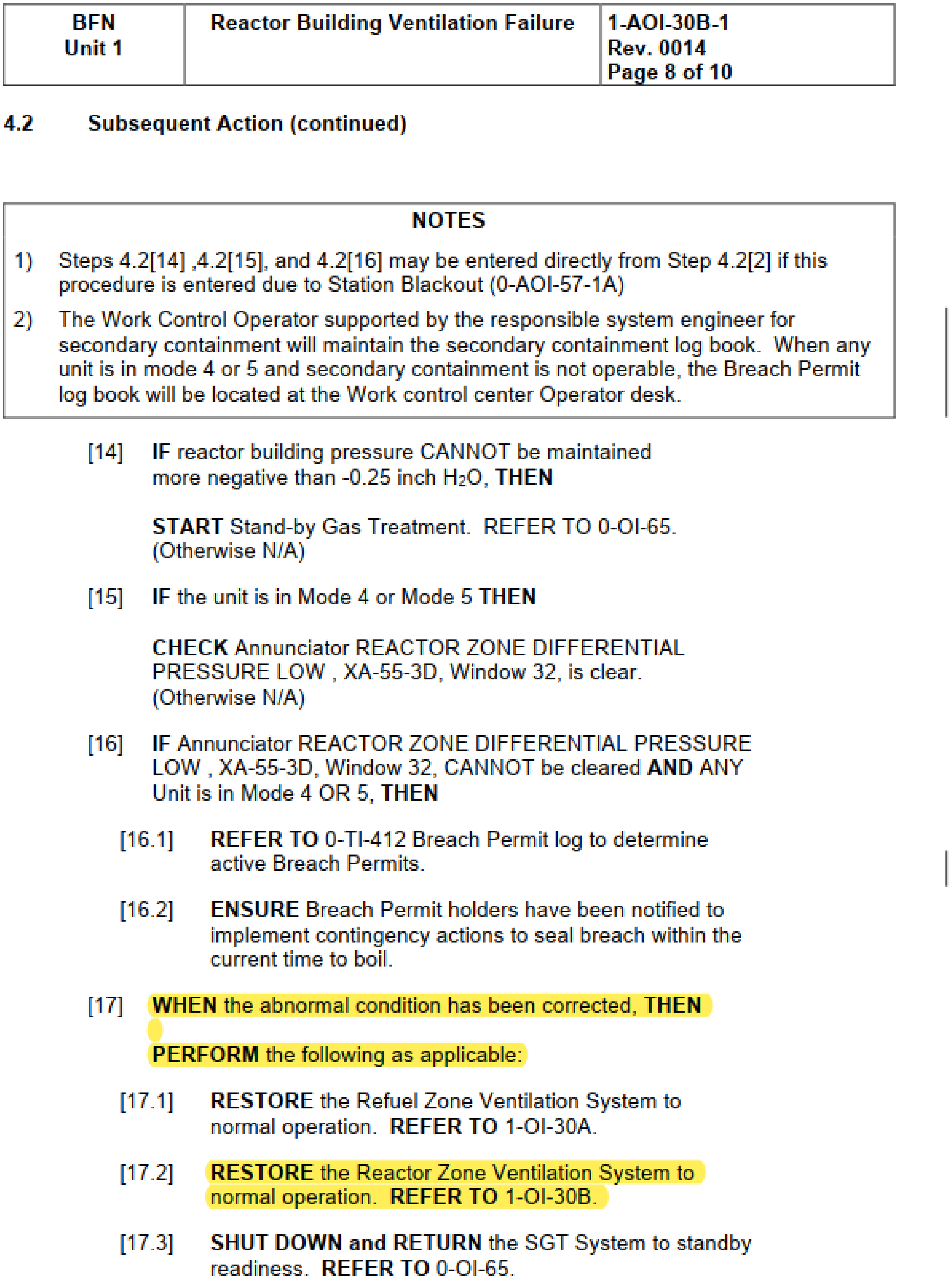
A. Motors, drive belts, fans are working.

B. Dampers in operating ventilation system flow path are open.

C. Dampers in non-operating ventilation system flow path are closed.

[12] **SHUT DOWN** the malfunctioning ventilation equipment in accordance with the appropriate operating instruction.

[13] **START** the alternate ventilation equipment in accordance with the appropriate operating instruction.



BFN **Unit 1**	**Reactor Building Ventilation Failure**	**1-AOI-30B-1** **Rev. 0014** **Page 8 of 10**

4.2 Subsequent Action (continued)

NOTES

1) Steps 4.2[14] ,4.2[15], and 4.2[16] may be entered directly from Step 4.2[2] if this procedure is entered due to Station Blackout (0-AOI-57-1A)

2) The Work Control Operator supported by the responsible system engineer for secondary containment will maintain the secondary containment log book. When any unit is in mode 4 or 5 and secondary containment is not operable, the Breach Permit log book will be located at the Work control center Operator desk.

[14] **IF** reactor building pressure CANNOT be maintained more negative than -0.25 inch H_2O, **THEN**

START Stand-by Gas Treatment. REFER TO 0-OI-65. (Otherwise N/A)

[15] **IF** the unit is in Mode 4 or Mode 5 **THEN**

CHECK Annunciator REACTOR ZONE DIFFERENTIAL PRESSURE LOW , XA-55-3D, Window 32, is clear. (Otherwise N/A)

[16] **IF** Annunciator REACTOR ZONE DIFFERENTIAL PRESSURE LOW , XA-55-3D, Window 32, CANNOT be cleared **AND** ANY Unit is in Mode 4 OR 5, **THEN**

[16.1] **REFER TO** 0-TI-412 Breach Permit log to determine active Breach Permits.

[16.2] **ENSURE** Breach Permit holders have been notified to implement contingency actions to seal breach within the current time to boil.

[17] **WHEN** the abnormal condition has been corrected, **THEN**

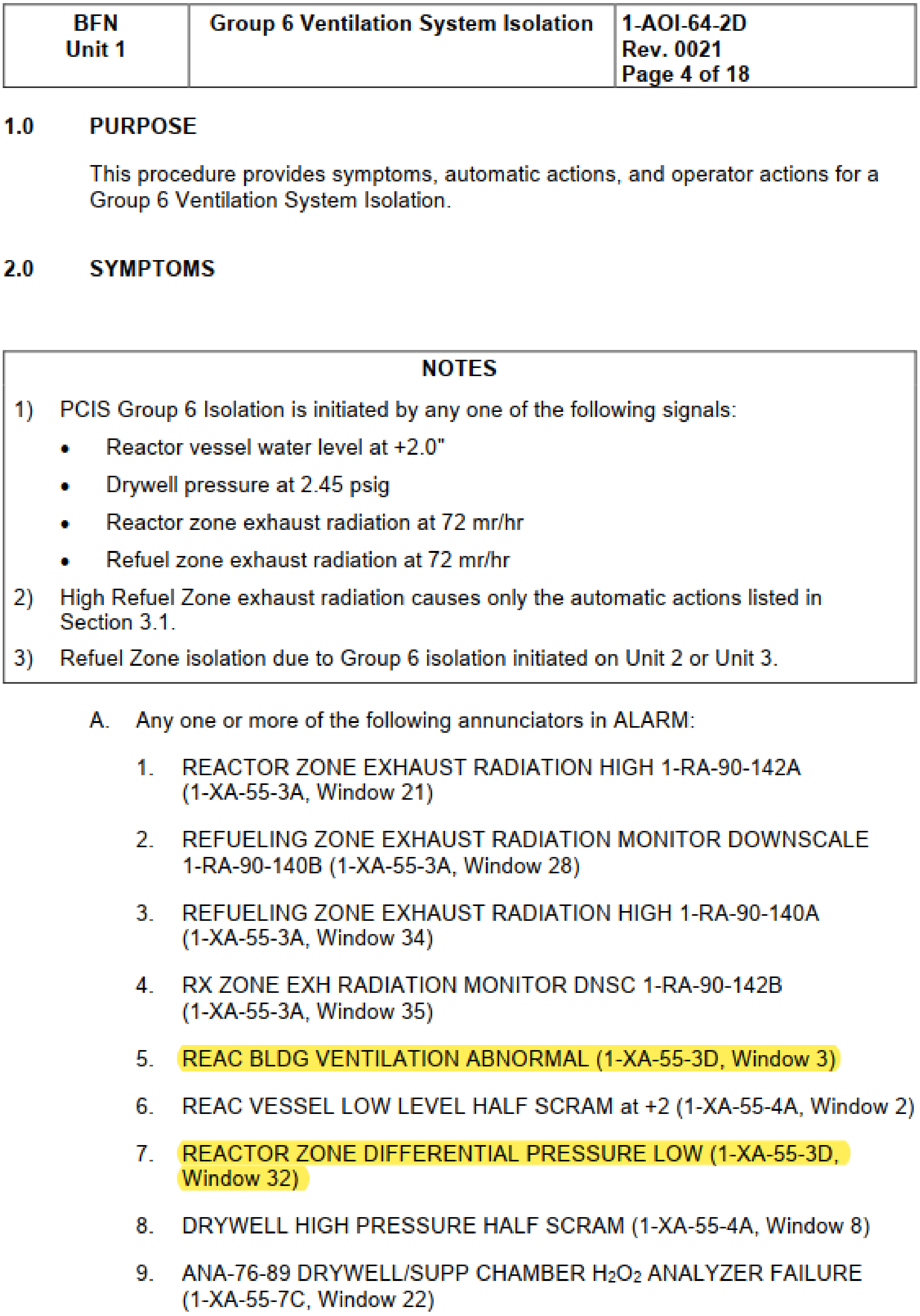
PERFORM the following as applicable:

[17.1] **RESTORE** the Refuel Zone Ventilation System to normal operation. **REFER TO** 1-OI-30A.

[17.2] **RESTORE** the Reactor Zone Ventilation System to normal operation. **REFER TO** 1-OI-30B.

[17.3] **SHUT DOWN and RETURN** the SGT System to standby readiness. **REFER TO** 0-OI-65.

Excerpts from 1-AOI-64-2D: Supports Distractors B(2), D(2)



BFN Unit 1	**Group 6 Ventilation System Isolation**	**1-AOI-64-2D Rev. 0021 Page 4 of 18**

1.0 PURPOSE

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

2.0 SYMPTOMS

NOTES

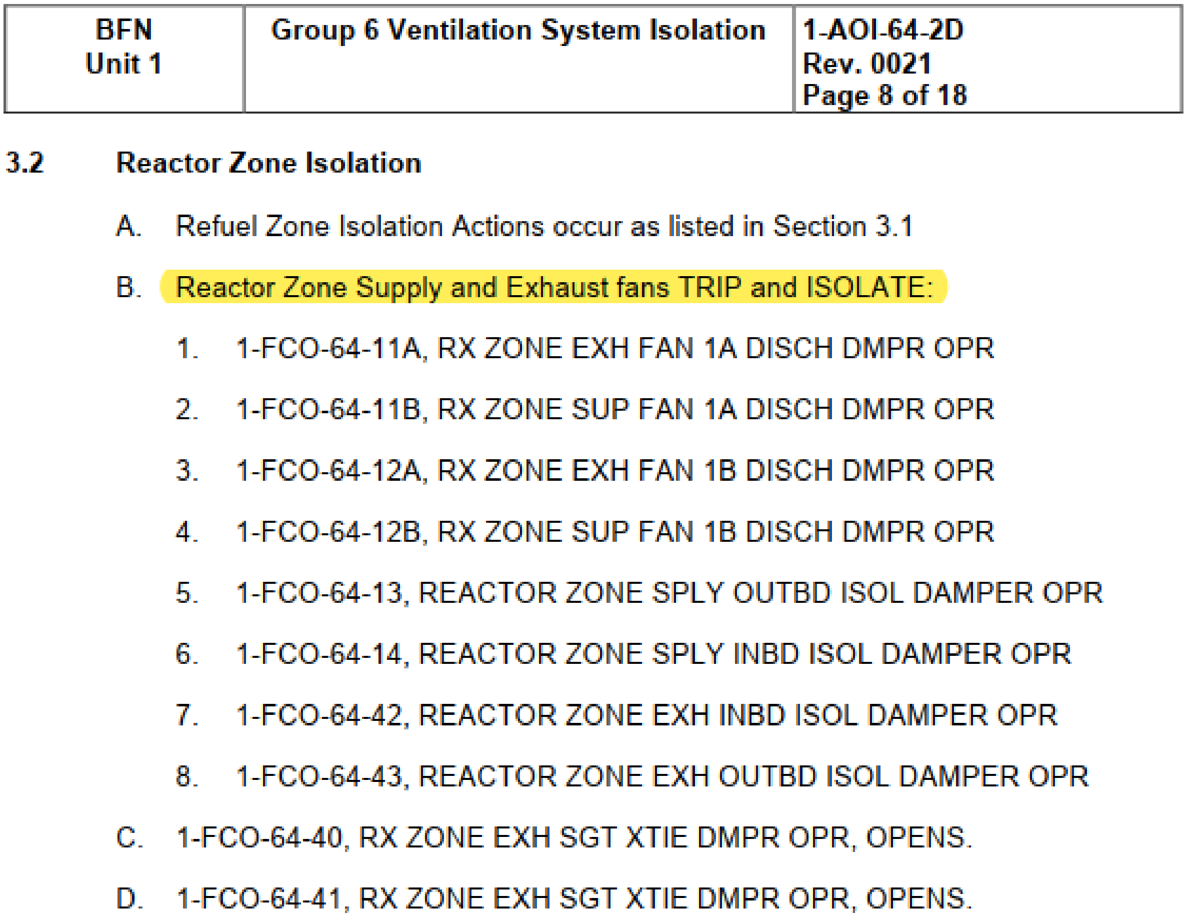
1) PCIS Group 6 Isolation is initiated by any one of the following signals:
   - Reactor vessel water level at +2.0"
   - Drywell pressure at 2.45 psig
   - Reactor zone exhaust radiation at 72 mr/hr
   - Refuel zone exhaust radiation at 72 mr/hr

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. 0021 Page 8 of 18**

3.2 Reactor Zone Isolation

A. Refuel Zone Isolation Actions occur as listed in Section 3.1

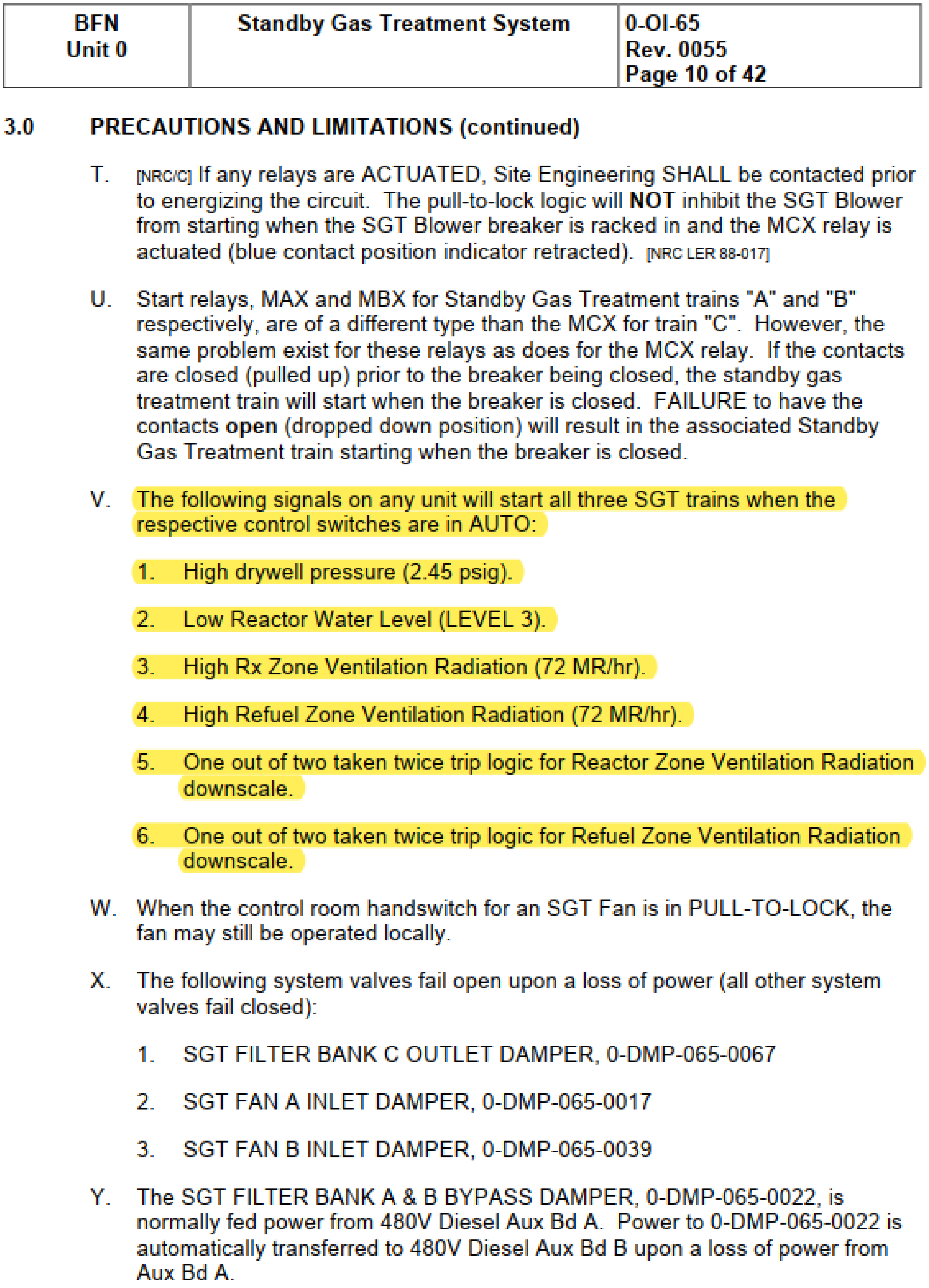
B. Reactor Zone Supply and Exhaust fans TRIP and ISOLATE:

1. 1-FCO-64-11A, RX ZONE EXH FAN 1A DISCH DMPR OPR
2. 1-FCO-64-11B, RX ZONE SUP FAN 1A DISCH DMPR OPR
3. 1-FCO-64-12A, RX ZONE EXH FAN 1B DISCH DMPR OPR
4. 1-FCO-64-12B, RX ZONE SUP FAN 1B DISCH DMPR OPR
5. 1-FCO-64-13, REACTOR ZONE SPLY OUTBD ISOL DAMPER OPR
6. 1-FCO-64-14, REACTOR ZONE SPLY INBD ISOL DAMPER OPR
7. 1-FCO-64-42, REACTOR ZONE EXH INBD ISOL DAMPER OPR
8. 1-FCO-64-43, REACTOR ZONE EXH OUTBD ISOL DAMPER OPR

C. 1-FCO-64-40, RX ZONE EXH SGT XTIE DMPR OPR, OPENS.

D. 1-FCO-64-41, RX ZONE EXH SGT XTIE DMPR OPR, OPENS.

Excerpt from 0-OI-65: Supports Distractors A(1), B(1)



BFN **Unit 0**	**Standby Gas Treatment System**	**0-OI-65** **Rev. 0055** **Page 10 of 42**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

T. [NRC/C] If any relays are ACTUATED, Site Engineering SHALL be contacted prior to energizing the circuit. The pull-to-lock logic will **NOT** inhibit the SGT Blower from starting when the SGT Blower breaker is racked in and the MCX relay is actuated (blue contact position indicator retracted). [NRC LER 88-017]

U. Start relays, MAX and MBX for Standby Gas Treatment trains "A" and "B" respectively, are of a different type than the MCX for train "C". However, the same problem exist for these relays as does for the MCX relay. If the contacts are closed (pulled up) prior to the breaker being closed, the standby gas treatment train will start when the breaker is closed. FAILURE to have the contacts **open** (dropped down position) will result in the associated Standby Gas Treatment train starting when the breaker is closed.

V. The following signals on any unit will start all three SGT trains when the respective control switches are in AUTO:

1. High drywell pressure (2.45 psig).
2. Low Reactor Water Level (LEVEL 3).
3. High Rx Zone Ventilation Radiation (72 MR/hr).
4. High Refuel Zone Ventilation Radiation (72 MR/hr).
5. One out of two taken twice trip logic for Reactor Zone Ventilation Radiation downscale.
6. One out of two taken twice trip logic for Refuel Zone Ventilation Radiation downscale.

W. When the control room handswitch for an SGT Fan is in PULL-TO-LOCK, the fan may still be operated locally.

X. The following system valves fail open upon a loss of power (all other system valves fail closed):

1. SGT FILTER BANK C OUTLET DAMPER, 0-DMP-065-0067
2. SGT FAN A INLET DAMPER, 0-DMP-065-0017
3. SGT FAN B INLET DAMPER, 0-DMP-065-0039

Y. The SGT FILTER BANK A & B BYPASS DAMPER, 0-DMP-065-0022, is normally fed power from 480V Diesel Aux Bd A. Power to 0-DMP-065-0022 is automatically transferred to 480V Diesel Aux Bd B upon a loss of power from Aux Bd A.

Examination Outline Cross-reference:	Level	RO	SRO
212000 (SF7 RPS) Reactor Protection	Tier #	------	2
G2.1.20 **(10CFR 55.43.5 - SRO Only)**	Group #	------	1
Ability to interpret and execute procedure steps	K/A #	212000G2.1.20	
	Importance Rating	------	4.6

Proposed Question: **# 86**

Unit 2 Operators have entered 2-EOI-1A, ATWS RPV CONTROL, with the following conditions:

- Reactor Power indicates 10%
- **ALL** SCRAM inlet/outlet blue lights are **EXTINGUISHED**

Given the conditions above and in accordance with 2-EOI-1A, ATWS RPV CONTROL, the Recirc Pumps ___**(1)**___ required to be tripped.

To mitigate this condition, the NUSO will direct ___**(2)**___.

Note: 2-EOI Appendix-1A, Removal and Replacement of RPS SCRAM Solenoid Fuses
2-EOI Appendix-1F, Manual SCRAM

A. (1) are
(2) 2-EOI Appendix-1A

B. (1) are
(2) 2-EOI Appendix-1F

C. (1) are NOT
(2) 2-EOI Appendix-1A

D. (1) are NOT
(2) 2-EOI Appendix-1F

Proposed Answer: **A**

Explanation (Optional):

A **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. Upon the automatic SCRAM, all 185 SCRAM inlet/outlet valves should have opened with their associated position switches illuminating all 185 blue Control Rod SCRAM lights on the Rod Status Display on Panel 2-9-5. However, this was not the case as given in the stem, therefore 2-AOI-100-1, Reactor SCRAM Immediate Actions for ATWS conditions where Reactor Power is above 5%, require both Recirc Pumps to be tripped. For second part, in accordance with 2-EOI-1A, since all 185 SCRAM inlet/outlet valves failed to open as indicated by the blue lights being extinguished (electrical ATWS), the NUSO must direct 2-EOI Appendix-1A, Removal and Replacement of RPS SCRAM Solenoid Fuses. Appendix-1A allows the RPS A and B SCRAM fuses to be removed from Aux Instrument Panels to de-energize the SCRAM solenoids in order to get all 185 SCRAM inlet/outlet valves to open.

B INCORRECT: The first part is correct *(See A).* 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 listed from Table Q-2 without a reference makes any of them plausible choices. Additionally, 2-EOI Appendix-1F, Manual SCRAM, would be used if the SCRAM Discharge Volume was full as in a hydraulic ATWS.

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 and/or 2-AOI-100-1, Reactor SCRAM Immediate Actions for ATWS conditions where Reactor Power is above 5%. This would lead a candidate to conclude that Recirc Pumps are not required to be tripped. 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 interpret Control Rod Positions following an Electrical ATWS and procedurally mitigate the failure of Control Rods to insert using the Reactor Protection System. 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.

Technical Reference(s):	2-EOI-1A, Rev. 3	(Attach if not previously provided)
	2-EOI Appendix-1A, Rev. 7	
	2-EOI Appendix-1F, Rev. 6	
	2-AOI-100-1, Rev. 118	
	OPL171.005, Rev. 24U2	

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.202, Obj. 21 (As available)

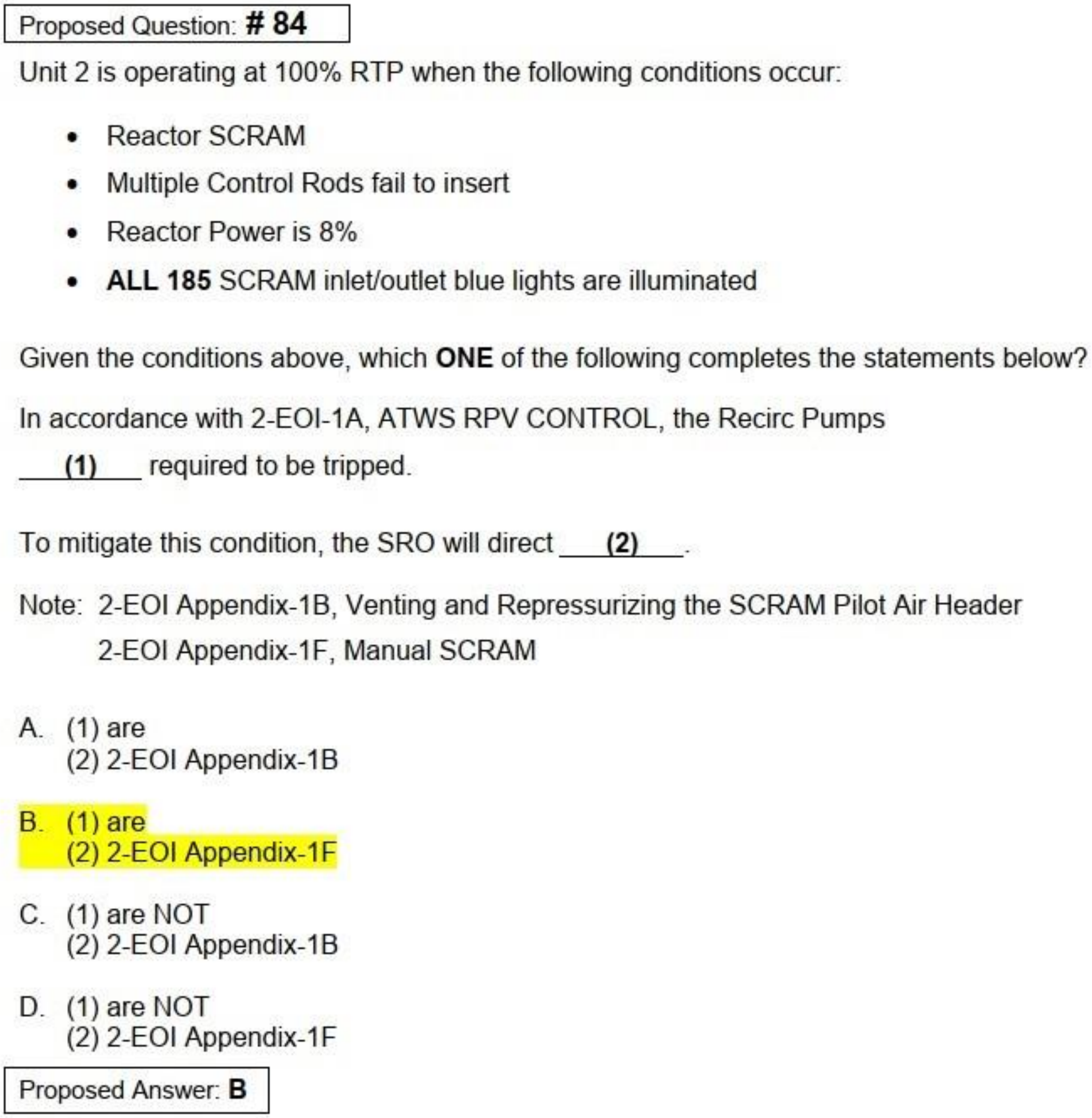
Form 4.2-1 **Written Examination Question Worksheet**

Question Source:	Bank #		
	Modified Bank #	BFN 21-04 #84	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	
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: **# 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

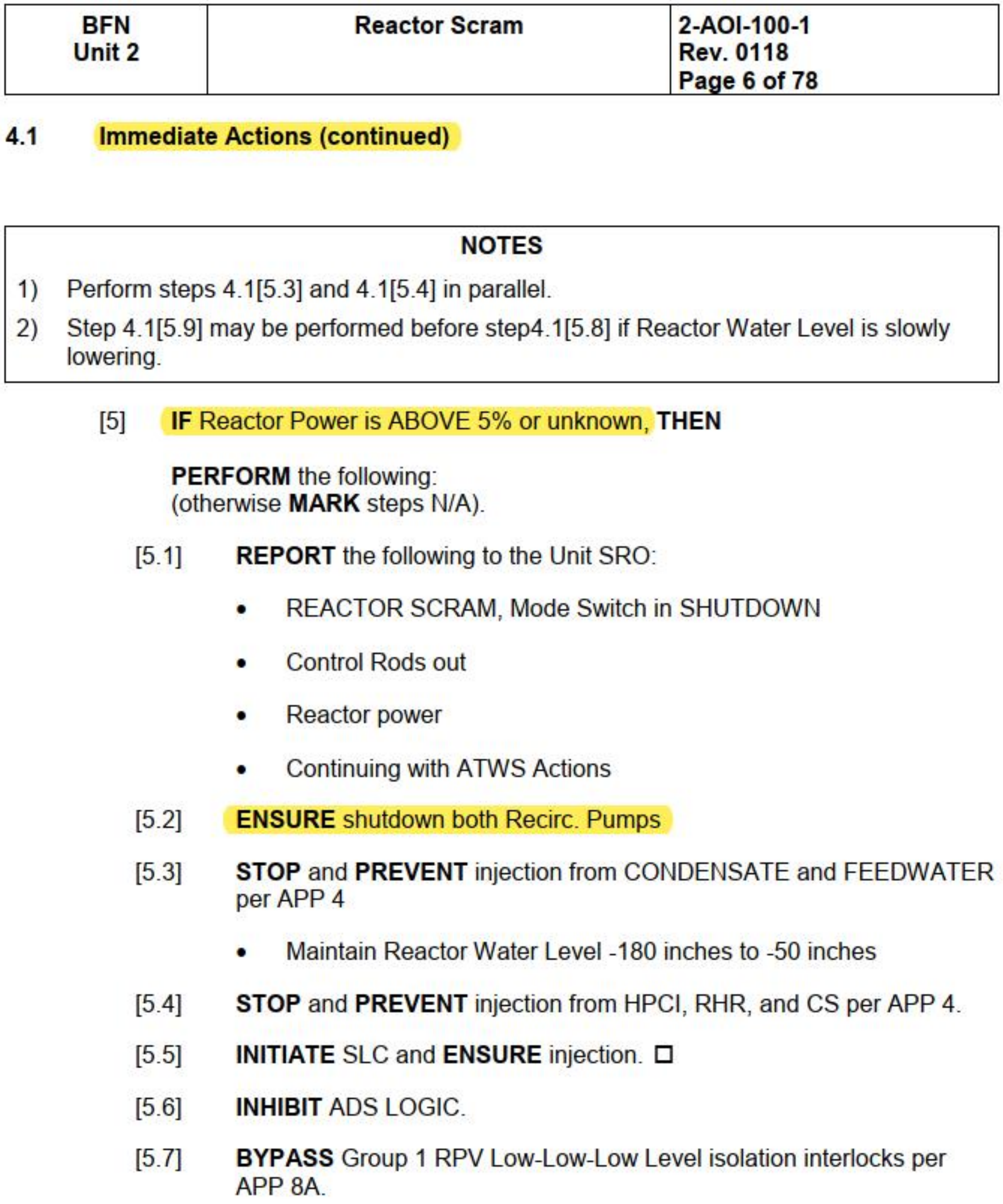
B. (1) are
(2) 2-EOI Appendix-1F

C. (1) are NOT
(2) 2-EOI Appendix-1B

D. (1) are NOT
(2) 2-EOI Appendix-1F

Proposed Answer: **B**

Excerpt from 2-AOI-100-1:



BFN **Unit 2**	**Reactor Scram**	**2-AOI-100-1** **Rev. 0118** **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[5.9] may be performed before step4.1[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

- Maintain Reactor Water Level -180 inches to -50 inches

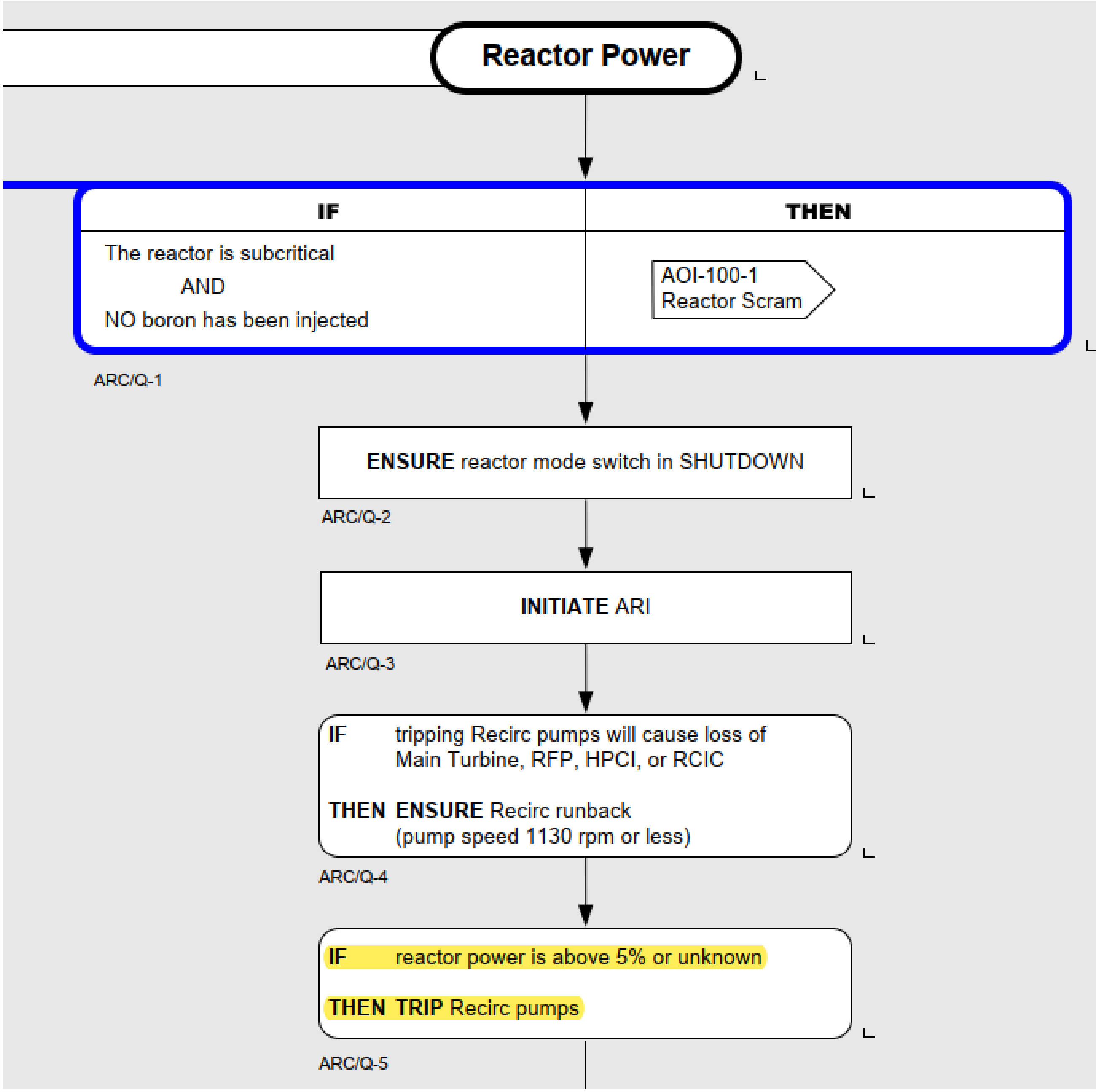
[5.4] **STOP** and **PREVENT** injection from HPCI, RHR, and CS per APP 4.

[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.

Excerpts from 2-EOI-1A:



Reactor Power

IF	**THEN**
The reactor is subcritical AND NO boron has been injected	AOI-100-1 Reactor Scram

ARC/Q-1

ENSURE reactor mode switch in SHUTDOWN

ARC/Q-2

INITIATE ARI

ARC/Q-3

IF tripping Recirc pumps will cause loss of Main Turbine, RFP, HPCI, or RCIC

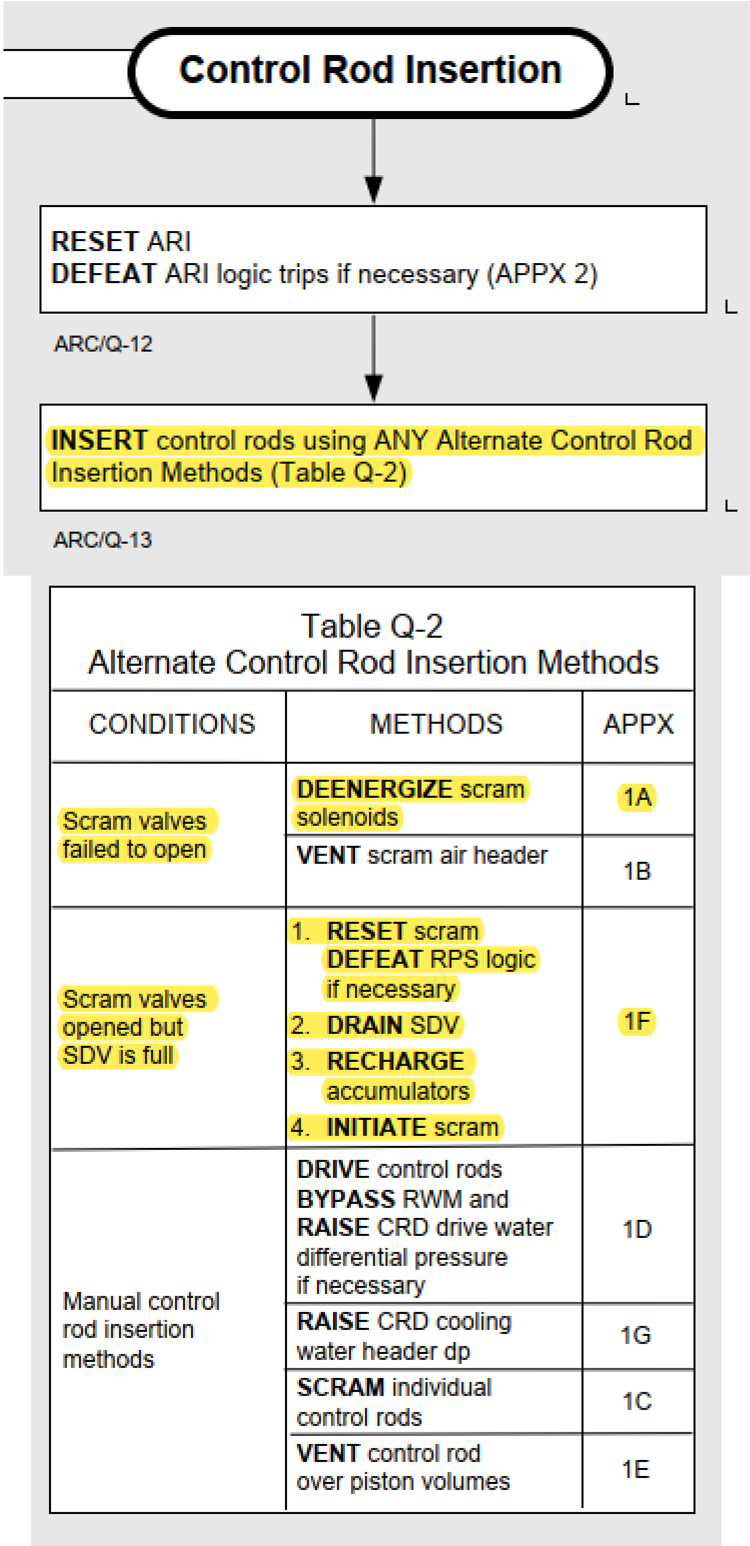
THEN **ENSURE** Recirc runback (pump speed 1130 rpm or less)

ARC/Q-4

IF reactor power is above 5% or unknown

THEN **TRIP** Recirc pumps

ARC/Q-5



Control Rod Insertion

RESET ARI
DEFEAT ARI logic trips if necessary (APPX 2)

ARC/Q-12

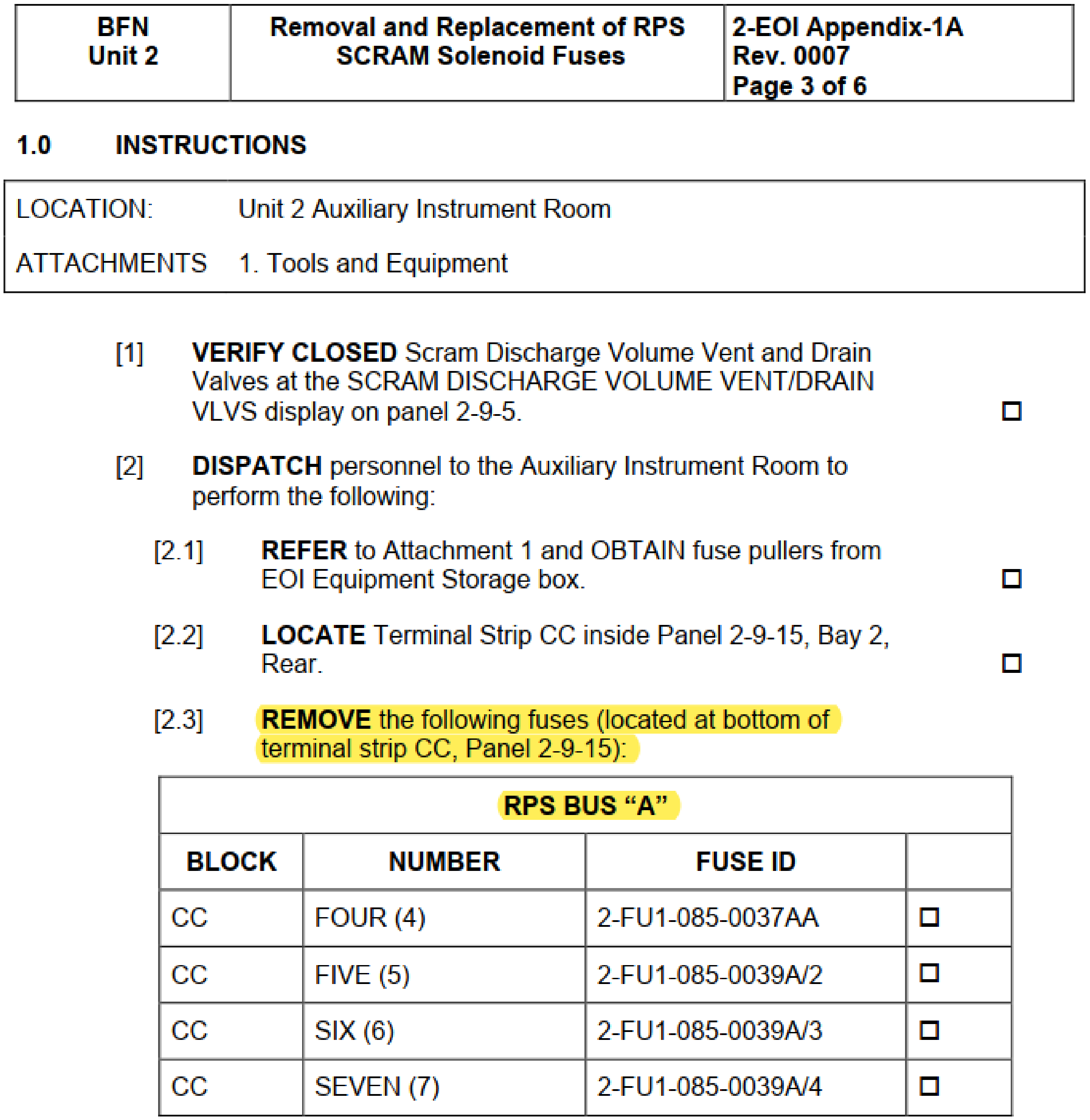
INSERT control rods using ANY Alternate Control Rod Insertion Methods (Table Q-2)

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

Excerpts from 2-EOI Appendix-1A:



BFN **Unit 2**	**Removal and Replacement of RPS SCRAM Solenoid Fuses**	**2-EOI Appendix-1A** **Rev. 0007** **Page 3 of 6**

1.0 INSTRUCTIONS

LOCATION:	Unit 2 Auxiliary Instrument Room
ATTACHMENTS	1. Tools and Equipment

[1] **VERIFY CLOSED** Scram Discharge Volume Vent and Drain Valves at the SCRAM DISCHARGE VOLUME VENT/DRAIN VLVS display on panel 2-9-5. ☐

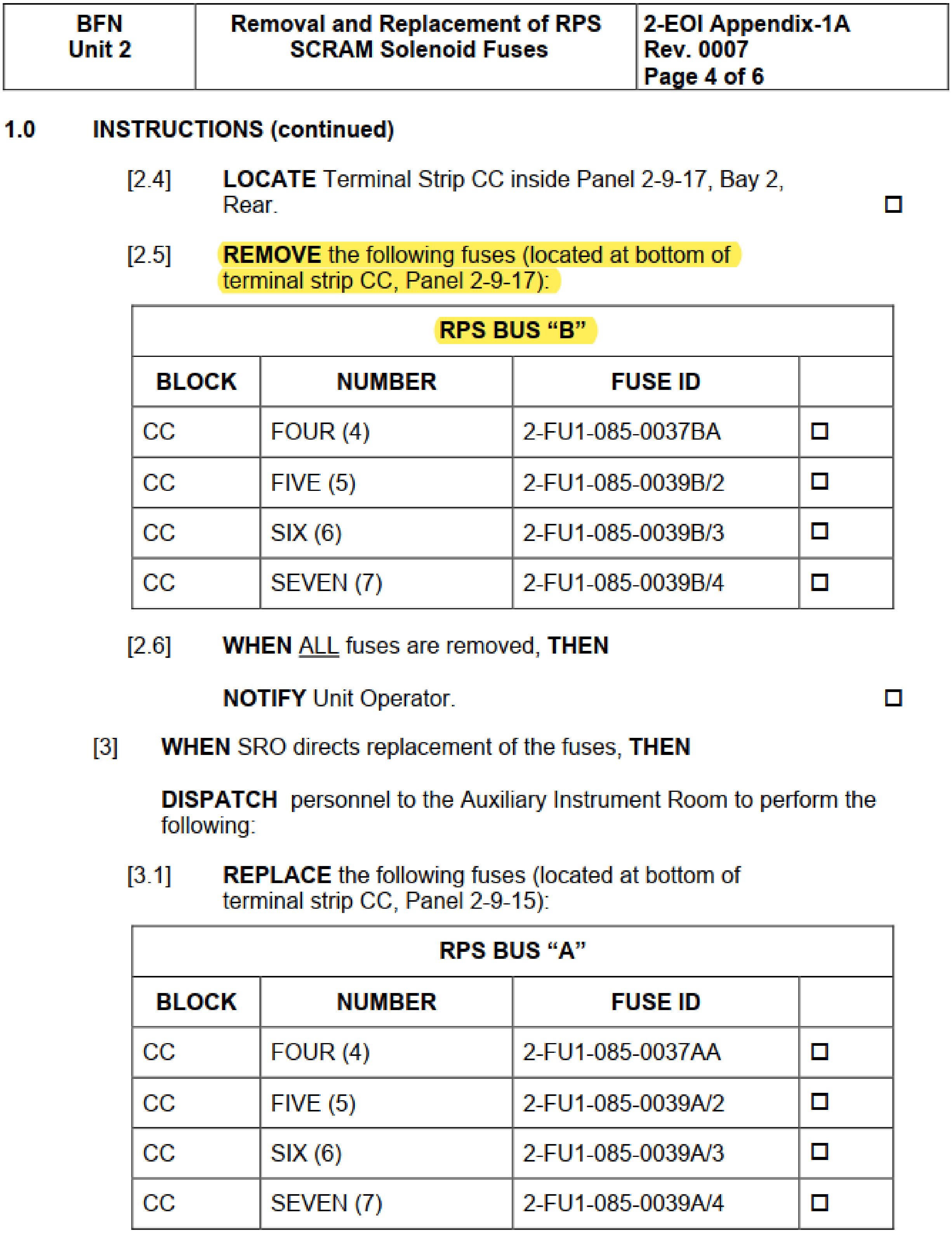
[2] **DISPATCH** personnel to the Auxiliary Instrument Room to perform the following:

[2.1] **REFER** to Attachment 1 and OBTAIN fuse pullers from EOI Equipment Storage box. ☐

[2.2] **LOCATE** Terminal Strip CC inside Panel 2-9-15, Bay 2, Rear. ☐

[2.3] **REMOVE** the following fuses (located at bottom of terminal strip CC, Panel 2-9-15):

RPS BUS “A”			
BLOCK	**NUMBER**	**FUSE ID**	
CC	FOUR (4)	2-FU1-085-0037AA	☐
CC	FIVE (5)	2-FU1-085-0039A/2	☐
CC	SIX (6)	2-FU1-085-0039A/3	☐
CC	SEVEN (7)	2-FU1-085-0039A/4	☐



BFN **Unit 2**	**Removal and Replacement of RPS SCRAM Solenoid Fuses**	**2-EOI Appendix-1A** **Rev. 0007** **Page 4 of 6**

1.0 INSTRUCTIONS (continued)

[2.4] **LOCATE** Terminal Strip CC inside Panel 2-9-17, Bay 2, Rear. ☐

[2.5] **REMOVE** the following fuses (located at bottom of terminal strip CC, Panel 2-9-17):

RPS BUS "B"			
BLOCK	**NUMBER**	**FUSE ID**	
CC	FOUR (4)	2-FU1-085-0037BA	☐
CC	FIVE (5)	2-FU1-085-0039B/2	☐
CC	SIX (6)	2-FU1-085-0039B/3	☐
CC	SEVEN (7)	2-FU1-085-0039B/4	☐

[2.6] **WHEN** ALL fuses are removed, **THEN**

NOTIFY Unit Operator. ☐

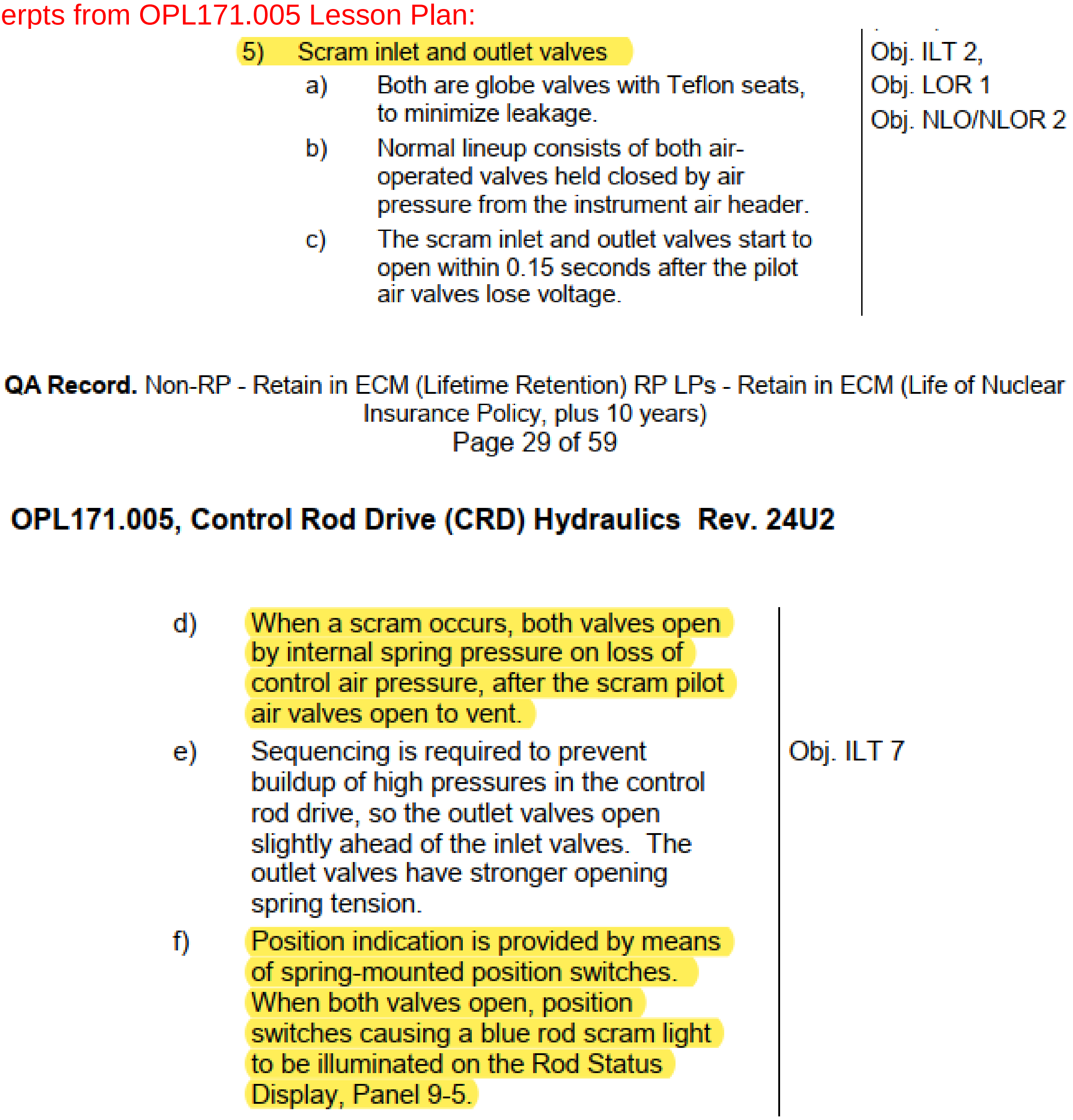
[3] **WHEN** SRO directs replacement of the fuses, **THEN**

DISPATCH personnel to the Auxiliary Instrument Room to perform the following:

[3.1] **REPLACE** the following fuses (located at bottom of terminal strip CC, Panel 2-9-15):

RPS BUS "A"			
BLOCK	**NUMBER**	**FUSE ID**	
CC	FOUR (4)	2-FU1-085-0037AA	☐
CC	FIVE (5)	2-FU1-085-0039A/2	☐
CC	SIX (6)	2-FU1-085-0039A/3	☐
CC	SEVEN (7)	2-FU1-085-0039A/4	☐

Excerpts from OPL171.005 Lesson Plan:



5) Scram inlet and outlet valves

Obj. ILT 2,
Obj. LOR 1
Obj. NLO/NLOR 2

   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.

   c) The scram inlet and outlet valves start to open within 0.15 seconds after the pilot air valves lose voltage.

QA Record. Non-RP - Retain in ECM (Lifetime Retention) RP LPs - Retain in ECM (Life of Nuclear Insurance Policy, plus 10 years)
Page 29 of 59

OPL171.005, Control Rod Drive (CRD) Hydraulics Rev. 24U2

   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

   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.

Excerpt from 2-EOI Appendix-1F: Supports Distractors B(2), D(2)

BFN **Unit 2**	**Manual Scram**	**2-EOI Appendix-1F** **Rev. 0006** **Page 4 of 9**

1.0 INSTRUCTIONS (continued)

[4] **DRAIN** SDV UNTIL the following annunciators clear:

- WEST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 2-9-4, 2-XA-55-4A, Window 1). ☐
- EAST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 2-9-4, 2-XA-55-4A, Window 29). ☐

[5] **DISPATCH** personnel to **VERIFY OPEN** 2-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. ☐

[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

- SRO directs otherwise. ☐

END OF TEXT

Examination Outline Cross-reference:

217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling

A2.14 (10CFR 55.43.5 - SRO Only)

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:

- Rupture disc failure: exhaust-diaphragm

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	217000A2.14	
Importance Rating	------	3.6

Proposed Question: **# 87**

Unit 2 was operating at 100% RTP while performing the 2-SR-3.5.3.3, RCIC Comprehensive Pump Test surveillance when the following occurred:

- A manual SCRAM was inserted, **ALL** Control Rods are in
- RCIC exhaust line disc rupture occurs
- RCIC failed to automatically isolate and attempts to manually isolate RCIC are unsuccessful

Steam Leak Detection panel indications (Panel 2-9-21) are as follows:

- 2-TI-71-41A at 205 °F and rising
- 2-TI-71-41B at 243 °F and rising
- 2-TI-71-41C at 256 °F and rising

No other temperature indicators are in alarm.

In accordance with EOIs, the NUSO will direct the crew to ________.

[REFERENCE PROVIDED]

A. **SHUT DOWN** the Reactor using 2-GOI-100-12A

B. **EMERGENCY DEPRESSURIZE** using 6 ADS valves

C. **DELIBERATELY LOWER** Reactor Pressure to 500 – 600 psig

D. **RAPIDLY DEPRESSURIZE** using Main Turbine Bypass Valves

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible if the candidate believes that at least 2 areas have reached their Max Safe values in EOI-3 Table SC-1 with a Primary System discharging into Secondary Containment. If there is no leak with multiple areas at Max Safe, this would be the correct answer. Although a SCRAM has been inserted, the Reactor is still in MODE 3 which is applicable for this action in EOI-3.

B **CORRECT:** *(See attached)* In accordance with 2-EOI-3, Secondary Containment Control, Step SC-7 - WHEN a primary system is discharging into Secondary Containment (due to the failed rupture disc), Table SC-1 must be referenced. Given that both 2-TI-41A and 2-TI-71-41C are above Max Safe, this meets the '2 or more areas for the same parameter' criteria to perform Emergency Depressurization. 2-EOI-1, RPV Control MODES 1-3 entry has already been completed since the stem states that all Control Rods are in due to the manual SCRAM. Therefore, in accordance 2-EOI-1 and 2-EOI-3, Emergency Depressurization using 6 MSRVs (ADS valves preferred) is conducted.

C INCORRECT: Incorrect but plausible in that EOI-1A, ATWS RPV Control would direct the NUSO to deliberately lower Reactor Pressure until all Control Rods are in or cold shutdown boron weight was injected. Once that is completed, Emergency Depressurization by rapidly depressurization would be required.

D INCORRECT: Incorrect but plausible if the decision is anticipated Emergency Depressurization, then RAPIDLY DEPRESSURIZE using Main Turbine Bypass Valves can only be made before Emergency Depressurization is required. Once 2 areas have reached Max Safe, Emergency Depressurization must be with SRVs.

SRO Level Justification: Tests the candidate's ability to recognize and predict the impact of a failure of the RCIC rupture exhaust disc and the plant procedures that are required to mitigate the plant 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. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-EOI-1, Rev. 19 (Attach if not previously provided)

2-EOI-1A, Rev. 3

2-EOI-3, Rev. 18

Proposed references to be provided to applicants during examination: **2-EOI-3, Table SC-1**

Learning Objective: OPL171.204 Obj. 9 (As available)

Question Source:	Bank #	OPL171.204-05 017 #2930	
	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**

Comments:

2-EOI-3, Table 2C-1 Reference Provided to candidate:

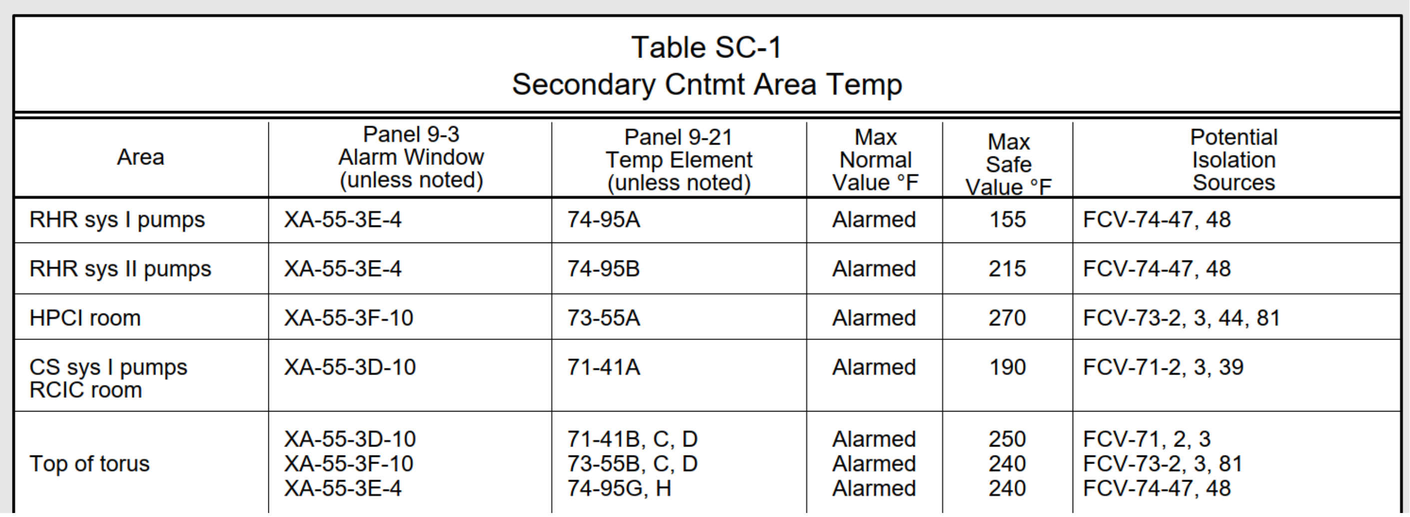
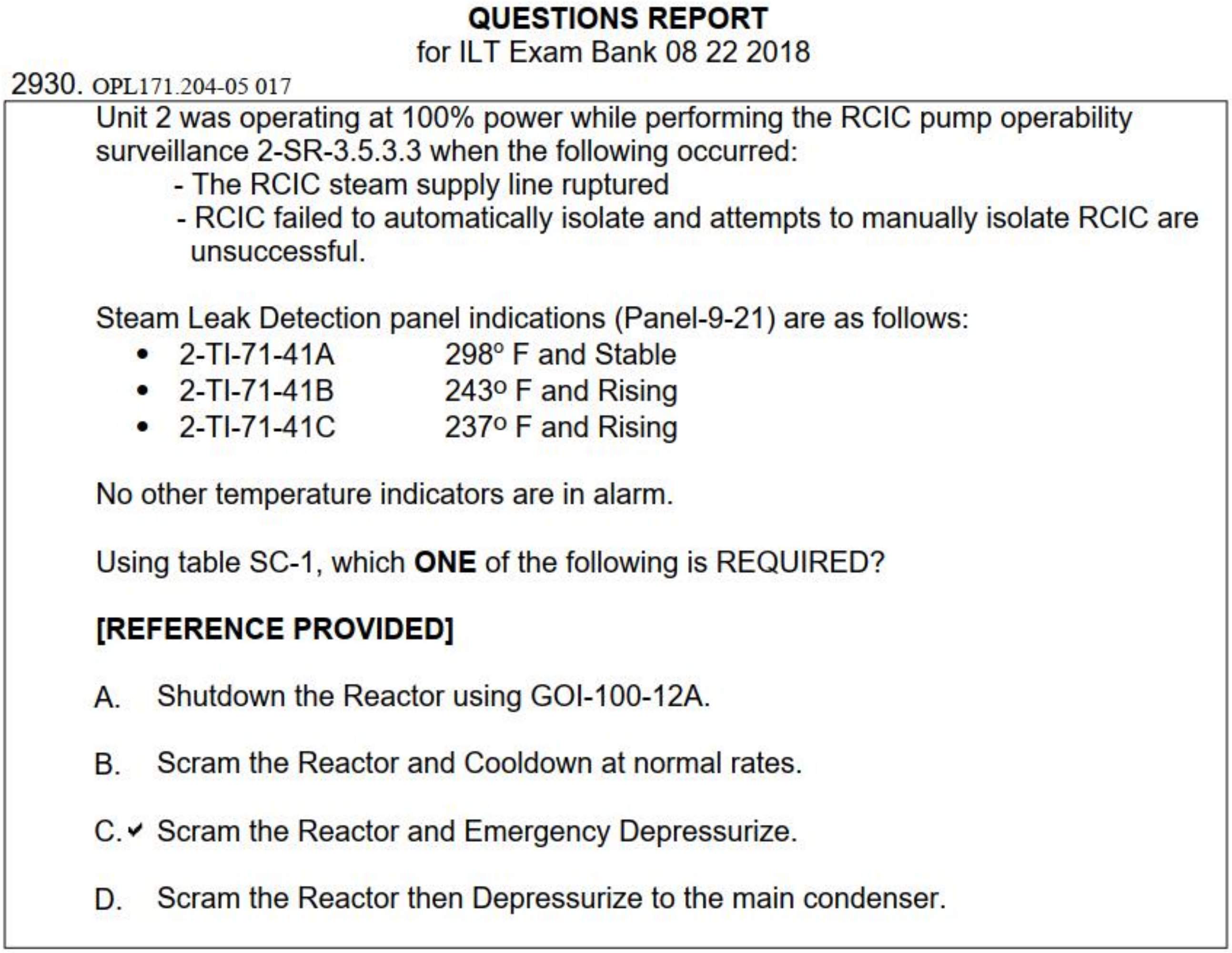


Table SC-1
Secondary Cntmt Area Temp

Area	Panel 9-3 Alarm Window (unless noted)	Panel 9-21 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	155	FCV-74-47, 48
RHR sys II pumps	XA-55-3E-4	74-95B	Alarmed	215	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	250 240 240	FCV-71, 2, 3 FCV-73-2, 3, 81 FCV-74-47, 48

Copy of Bank Question:



QUESTIONS REPORT

for ILT Exam Bank 08 22 2018

2930. OPL171.204-05 017

Unit 2 was operating at 100% power while performing the RCIC pump operability surveillance 2-SR-3.5.3.3 when the following occurred:

- The RCIC steam supply line ruptured
- RCIC failed to automatically isolate and attempts to manually isolate RCIC are unsuccessful.

Steam Leak Detection panel indications (Panel-9-21) are as follows:

2-TI-71-41A	298° F and Stable
2-TI-71-41B	243° F and Rising
2-TI-71-41C	237° F and Rising

No other temperature indicators are in alarm.

Using table SC-1, which **ONE** of the following is REQUIRED?

[REFERENCE PROVIDED]

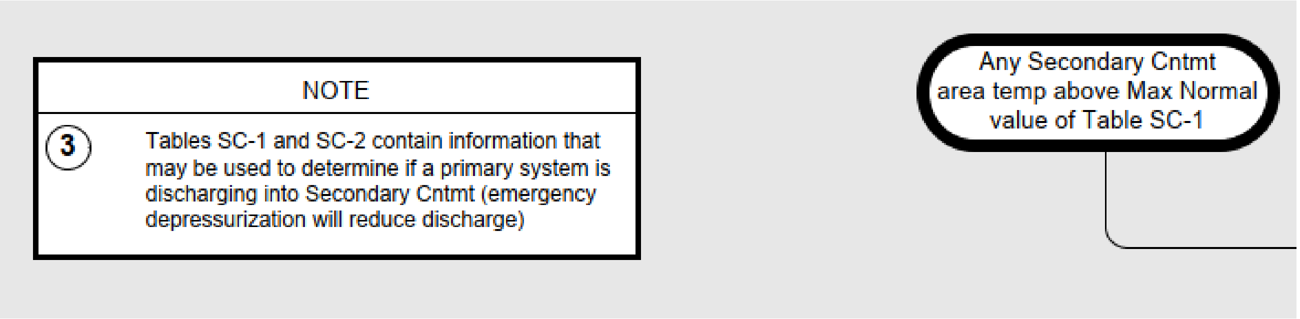
A. Shutdown the Reactor using GOI-100-12A.

B. Scram the Reactor and Cooldown at normal rates.

C.✓ Scram the Reactor and Emergency Depressurize.

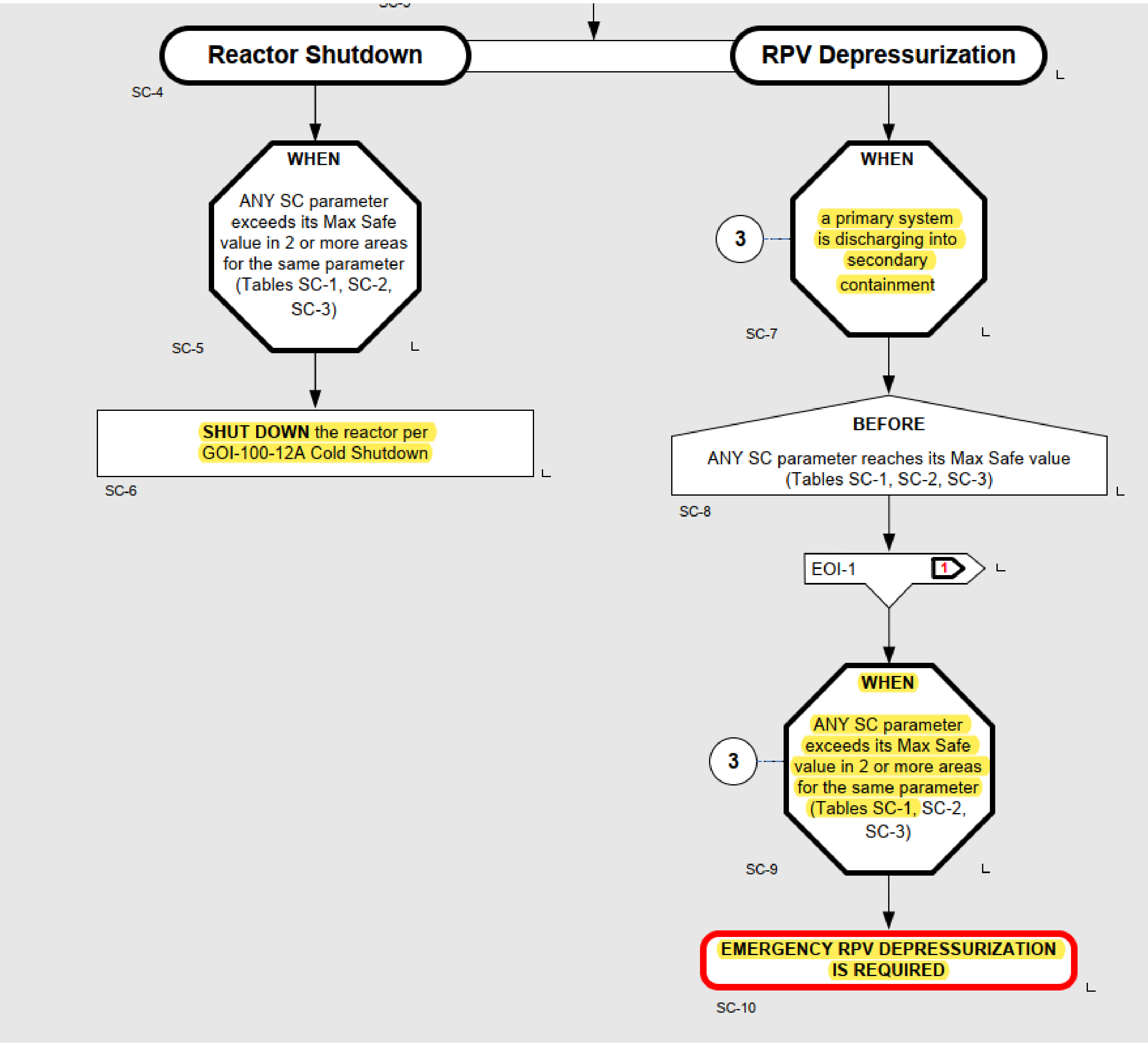
D. Scram the Reactor then Depressurize to the main condenser.

Excerpts from 2-EOI-3:



NOTE

③ Tables SC-1 and SC-2 contain information that may be used to determine if a primary system is discharging into Secondary Cntmt (emergency depressurization will reduce discharge)



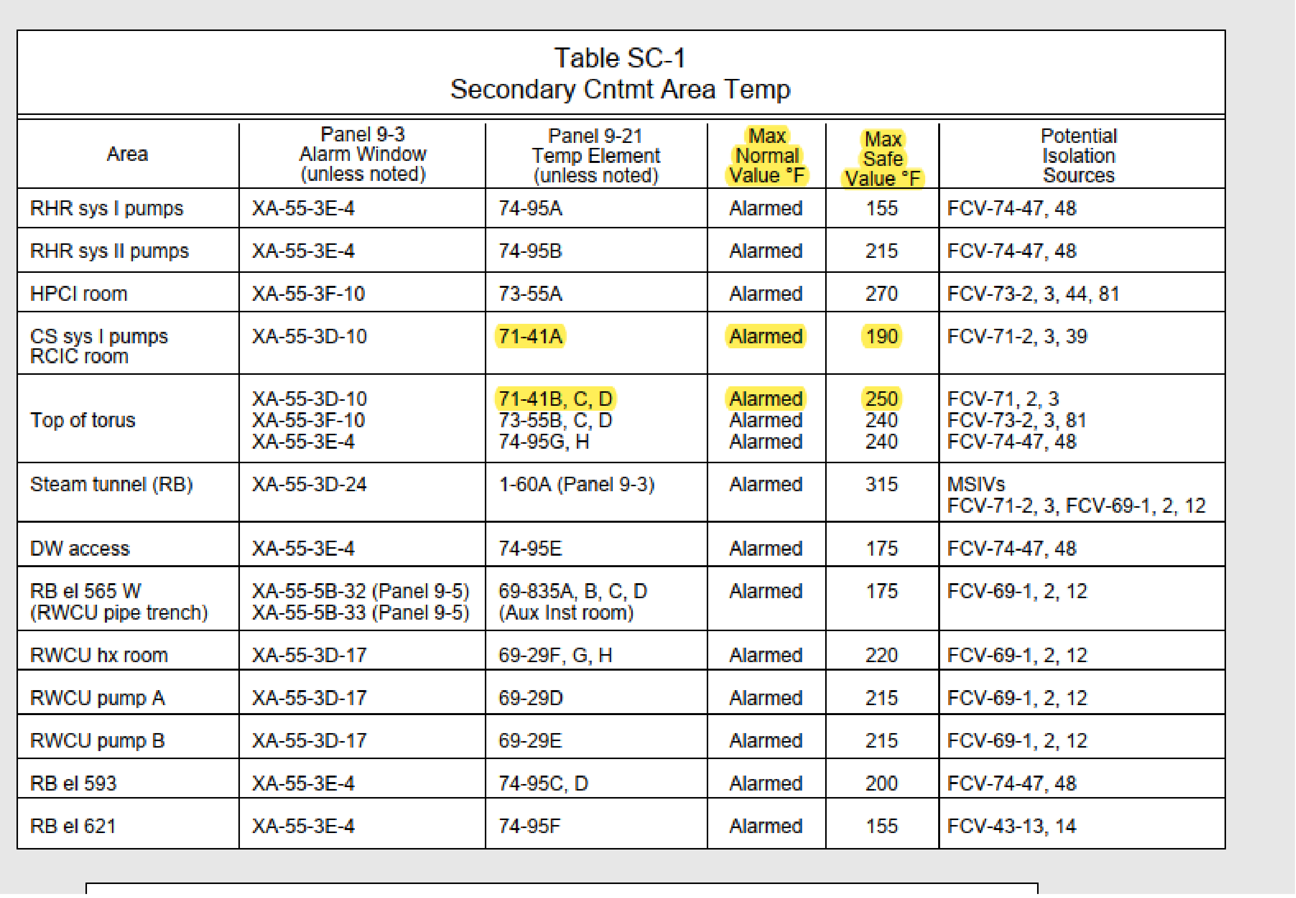
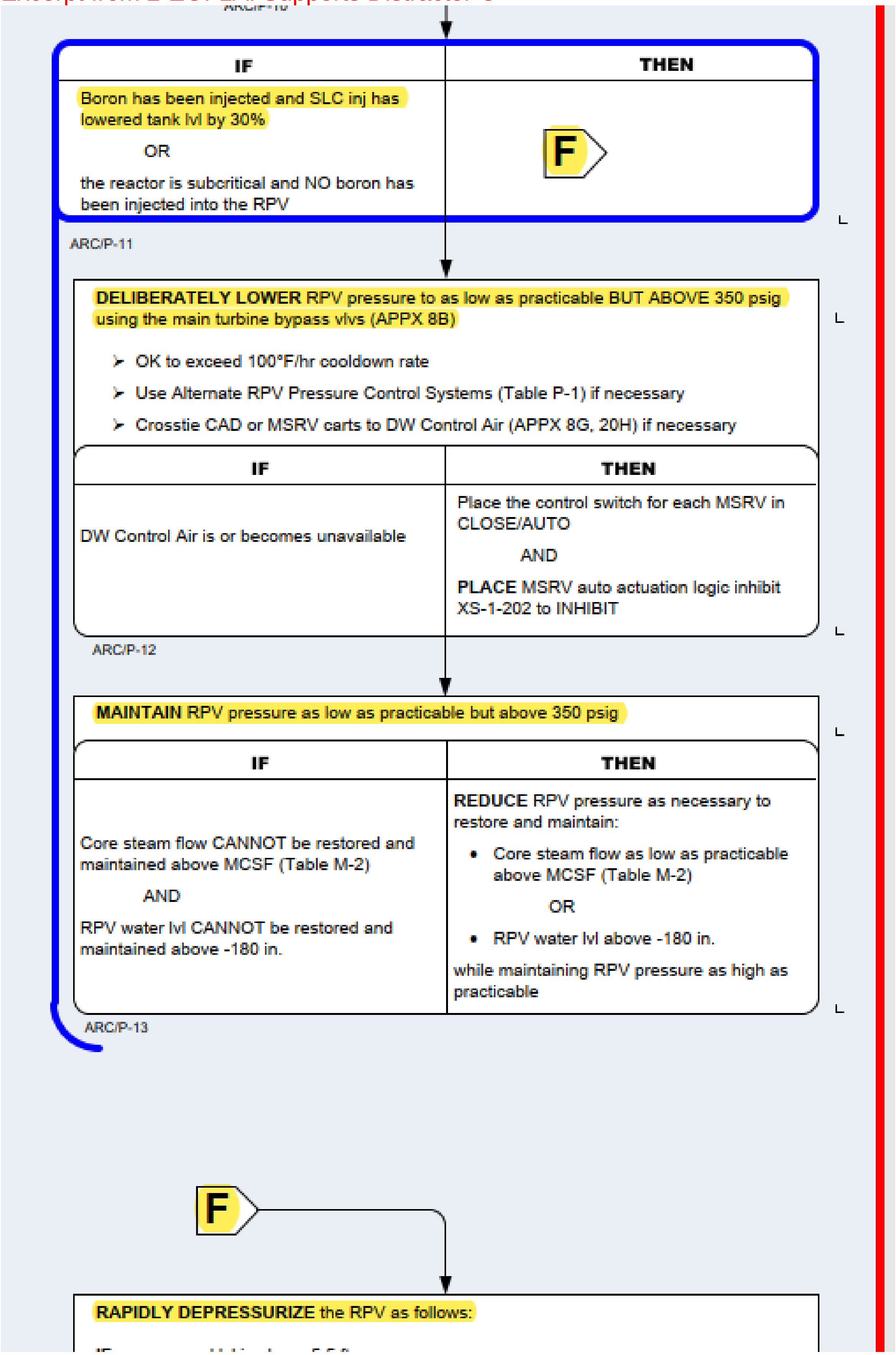


Table SC-1
Secondary Cntmt Area Temp

Area	Panel 9-3 Alarm Window (unless noted)	Panel 9-21 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	155	FCV-74-47, 48
RHR sys II pumps	XA-55-3E-4	74-95B	Alarmed	215	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	250 240 240	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	175	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	175	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	200	FCV-74-47, 48
RB el 621	XA-55-3E-4	74-95F	Alarmed	155	FCV-43-13, 14

Excerpt from 2-EOI-1A: Supports Distractor C



IF	THEN
Boron has been injected and SLC inj has lowered tank lvl by 30% OR the reactor is subcritical and NO boron has been injected into the RPV	F

ARC/P-11

DELIBERATELY LOWER RPV pressure to as low as practicable BUT ABOVE 350 psig using the main turbine bypass vlvs (APPX 8B)

- ➢ OK to exceed 100°F/hr cooldown rate
- ➢ 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

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

ARC/P-12

MAINTAIN RPV pressure as low as practicable but above 350 psig

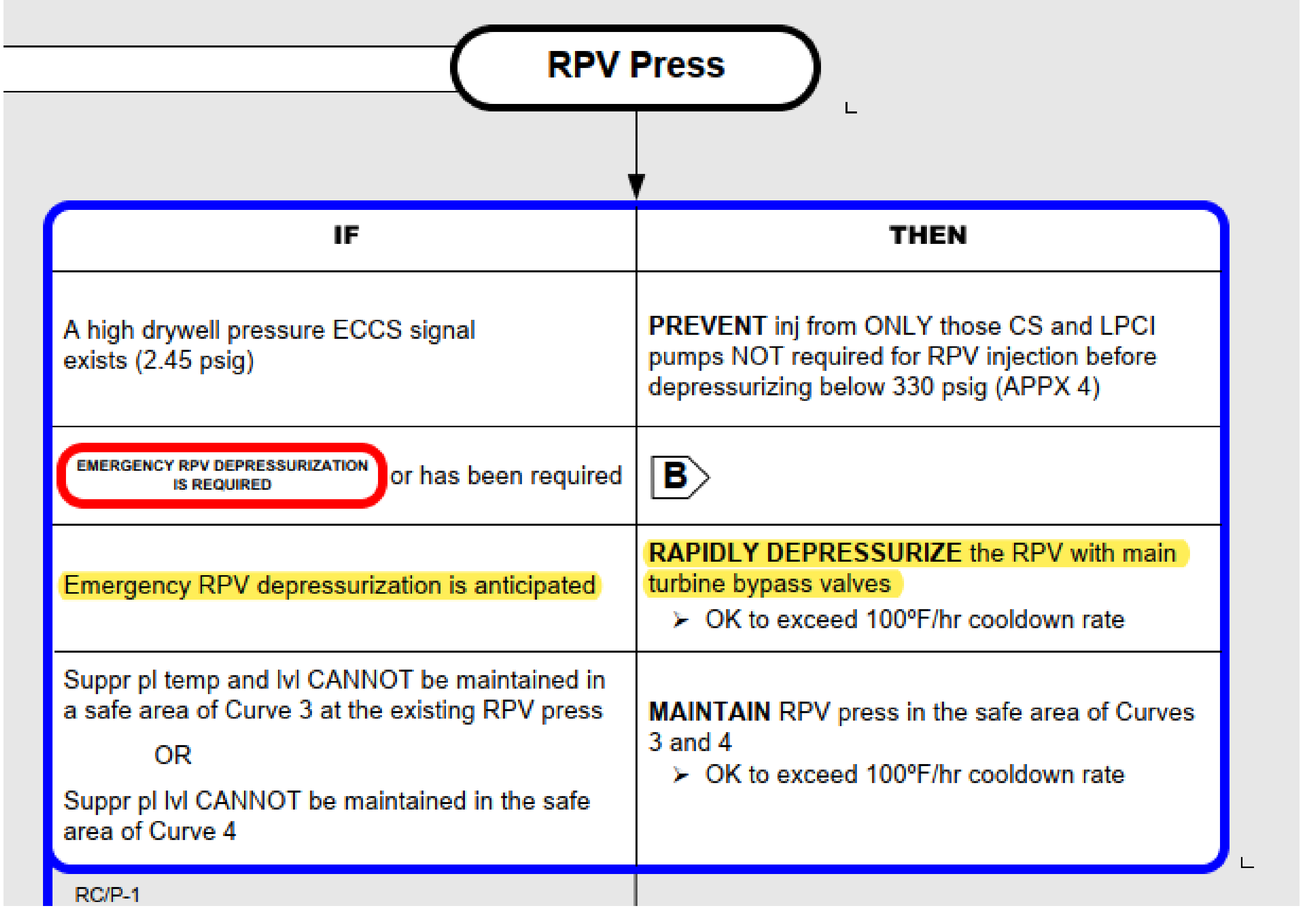
IF	THEN
Core steam flow CANNOT be restored and maintained above MCSF (Table M-2) AND RPV water lvl CANNOT be restored and maintained above -180 in.	**REDUCE** RPV pressure as necessary to restore and maintain: • Core steam flow as low as practicable above MCSF (Table M-2) OR • RPV water lvl above -180 in. while maintaining RPV pressure as high as practicable

ARC/P-13

F

RAPIDLY DEPRESSURIZE the RPV as follows:

Excerpt from 2-EOI-1: Supports Distractor D



RPV Press

IF	THEN
A high drywell pressure ECCS signal exists (2.45 psig)	**PREVENT** inj from ONLY those CS and LPCI pumps NOT required for RPV injection before depressurizing below 330 psig (APPX 4)
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED or has been required	**B**
Emergency RPV depressurization is anticipated	**RAPIDLY DEPRESSURIZE** the RPV with main turbine bypass valves ➢ OK to exceed 100°F/hr cooldown rate
Suppr pl temp and lvl CANNOT be maintained in a safe area of Curve 3 at the existing RPV press OR Suppr pl lvl CANNOT be maintained in the safe area of Curve 4	**MAINTAIN** RPV press in the safe area of Curves 3 and 4 ➢ OK to exceed 100°F/hr cooldown rate

RC/P-1

Examination Outline Cross-reference:

Conduct of Operations

G2.1.36 (10CFR 55.43.6 – SRO Only)

Knowledge of procedures and limitations involved in core alterations.

Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.1.36	
Importance Rating	------	4.1

Proposed Question: **# 88**

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**

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 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 AN ADJACENT** core quadrant 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, 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 (A in this case) to be OPERABLE in the quadrant (affected) where CORE ALTERATIONS are being performed and the other SRM (B or D 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 (6): Procedures and limitations involved in initial core loading, alterations in core configuration, Control Rod programming, and determination of various internal and external effects of core reactivity. 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. 99	
	2-GOI-100-1A, Rev. 183	

Proposed references to be provided to applicants during examination: **0-GOI-100-3C, Attachment 14 (Page 1 of 1) Core Quadrants**

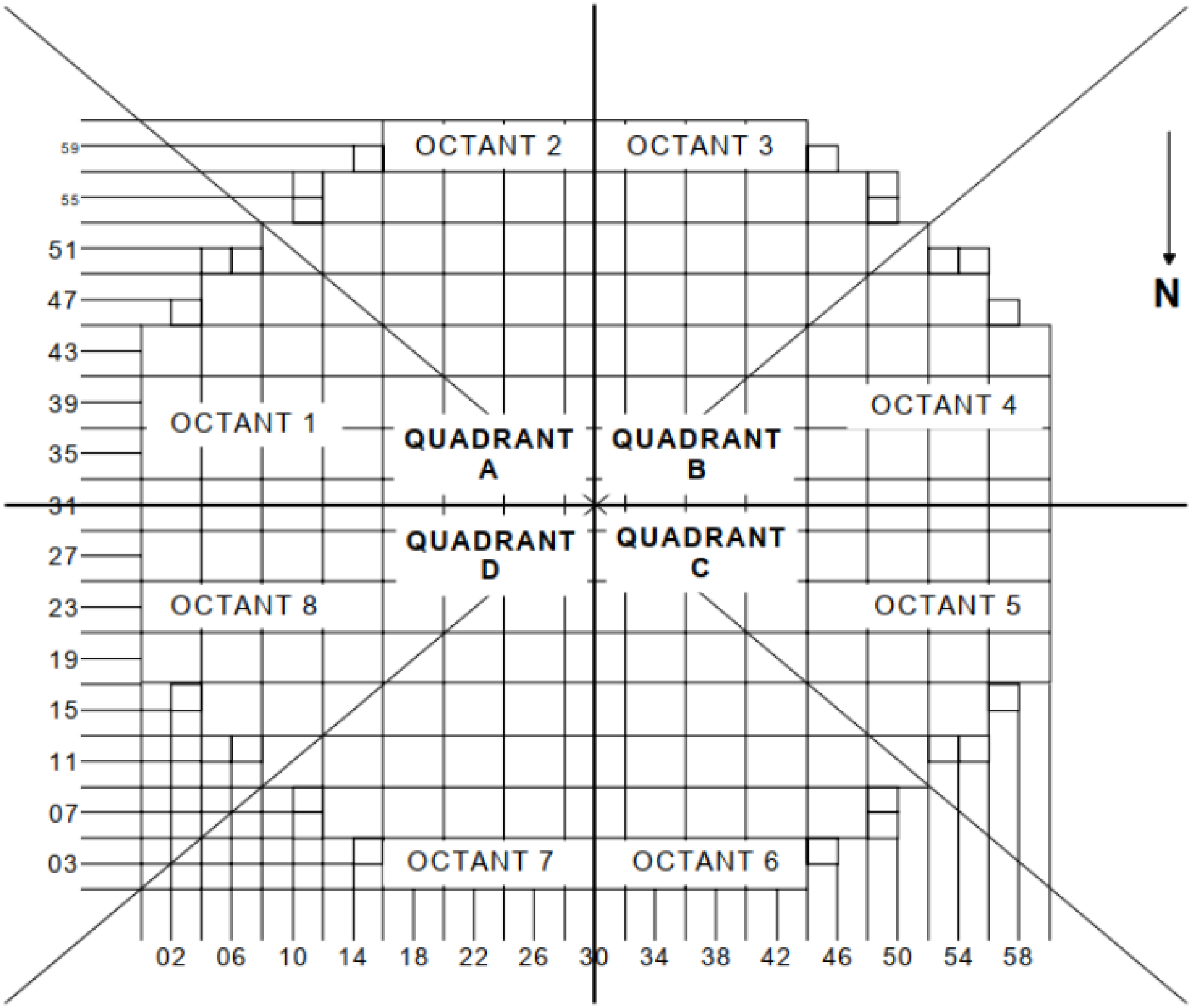
Learning Objective: OPL171.053 Obj. 9 (As available)

Question Source:	Bank #	BFN 2104 #95	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	X

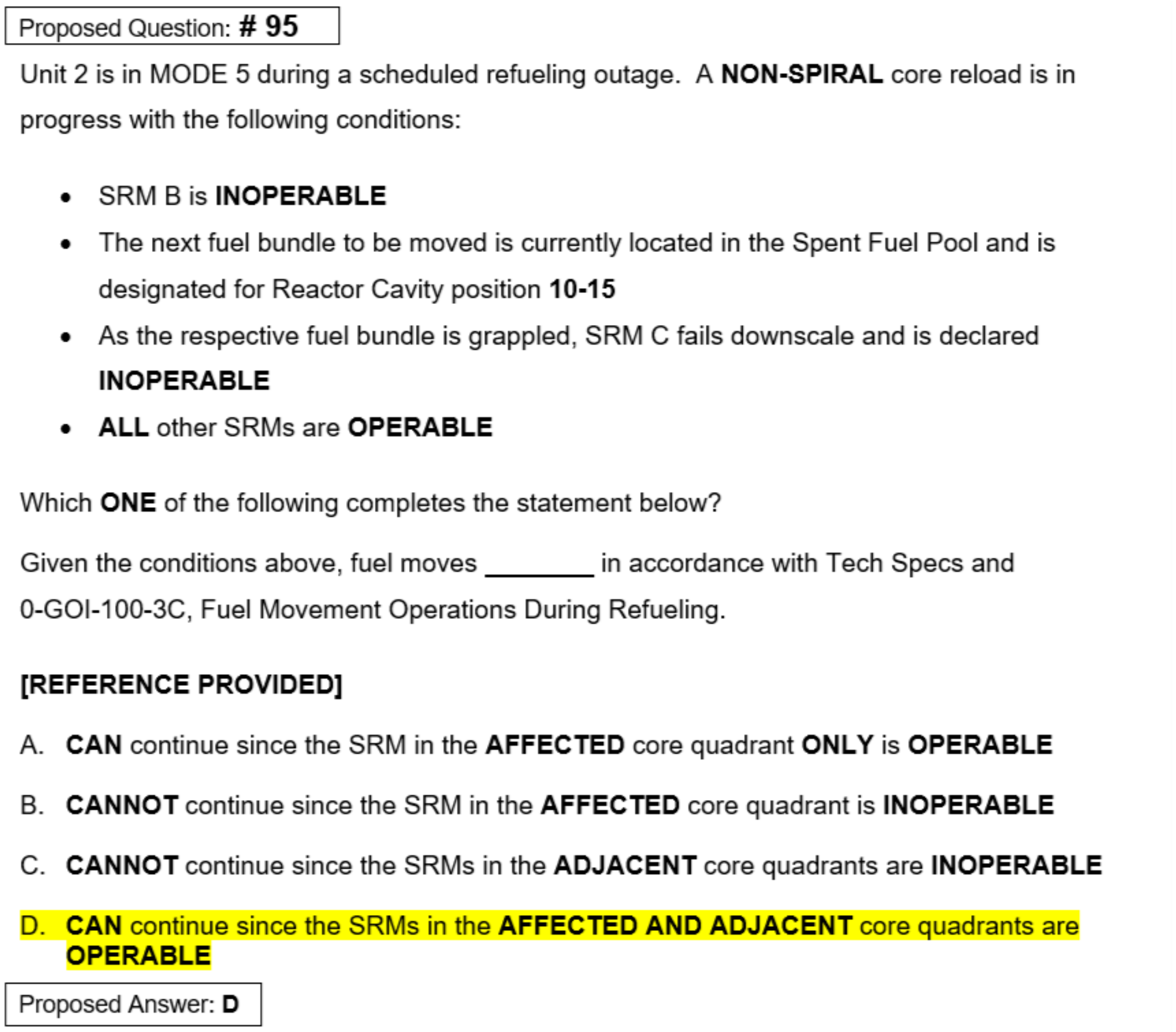
REFERENCE PROVIDED (to candidate):

Core Quadrants/Octants



CORE QUADRANTS / OCTANTS

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 **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**

Excerpts from Tech Spec 3.3.1.2: (Not provided to candidate)

3.3 INSTRUMENTATION

3.3.1.2 Source Range Monitor (SRM) Instrumentation

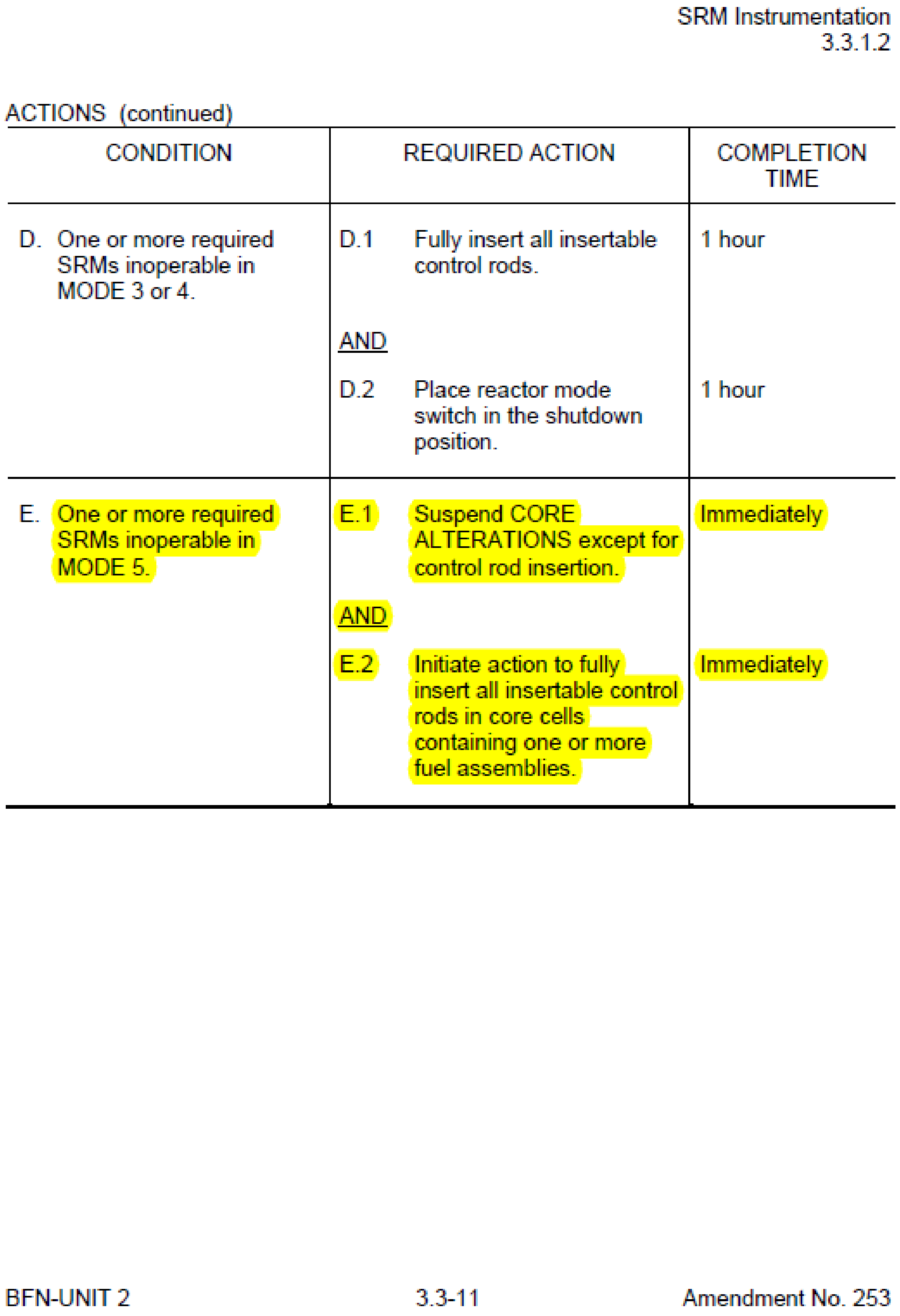
LCO 3.3.1.2 The SRM instrumentation in Table 3.3.1.2-1 shall be OPERABLE.

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)

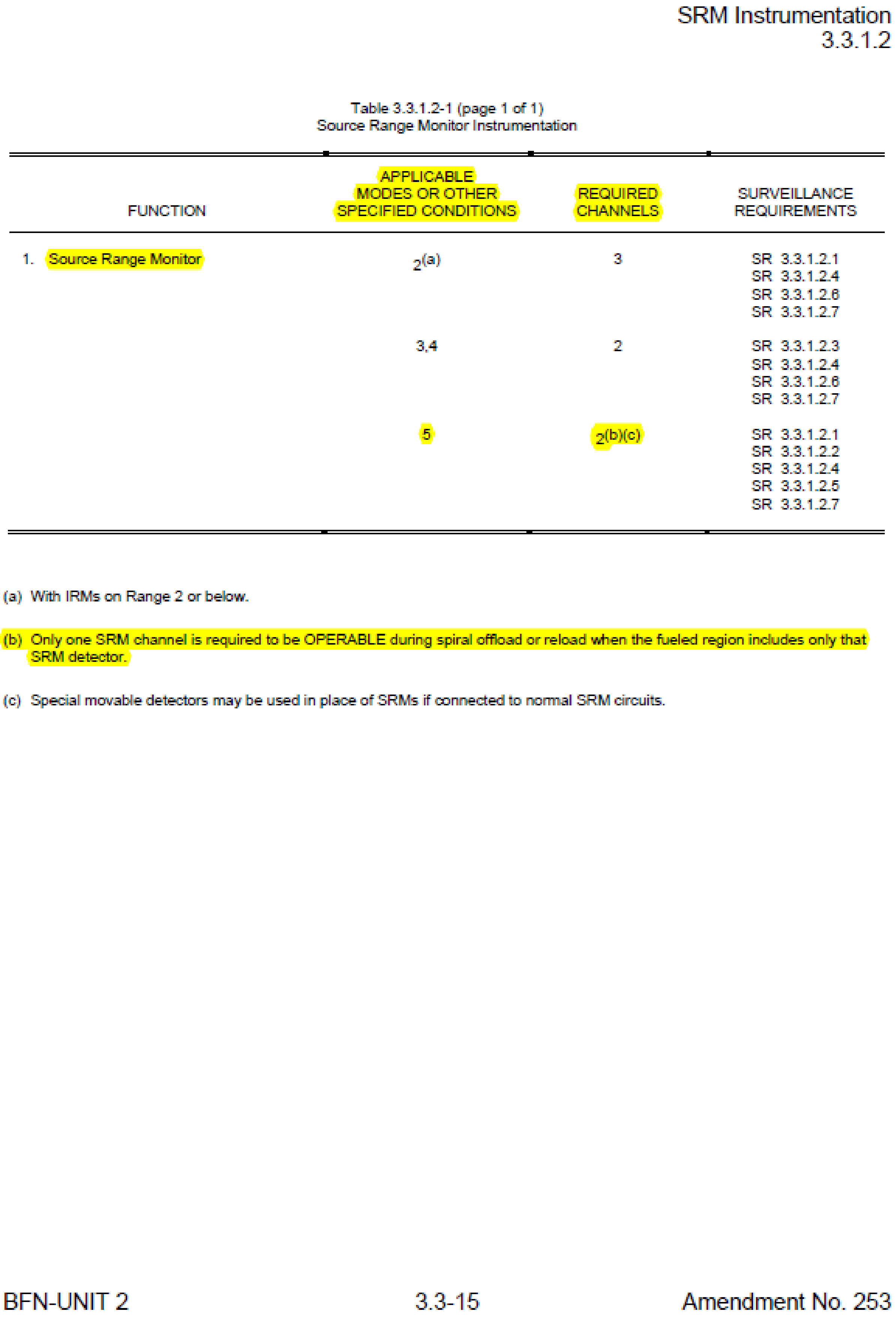


SRM Instrumentation
3.3.1.2

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

BFN-UNIT 2 3.3-11 Amendment No. 253



SRM Instrumentation
3.3.1.2

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

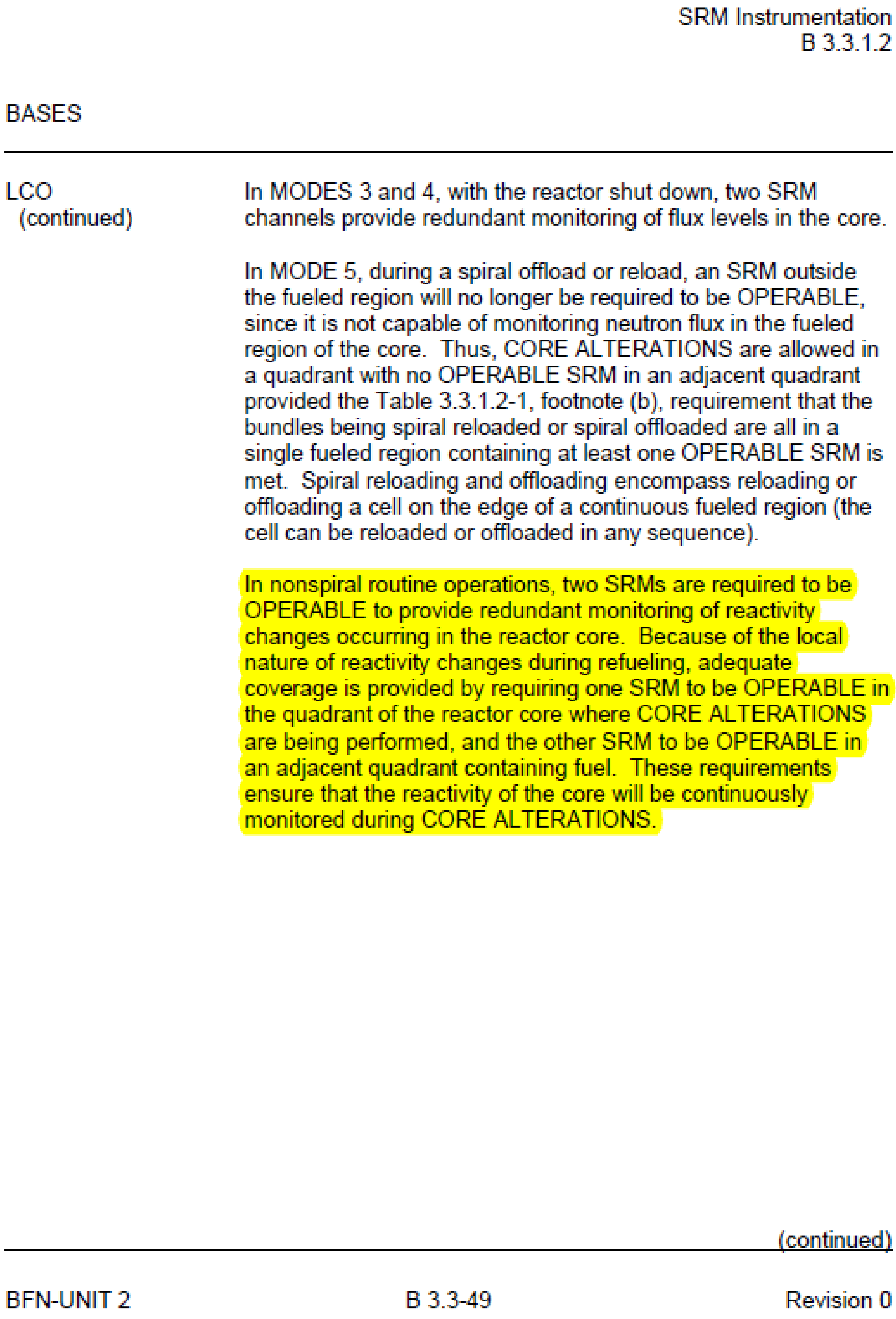
(a) With IRMs on Range 2 or below.

(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.

BFN-UNIT 2 3.3-15 Amendment No. 253

Excerpts from Tech Spec Bases 3.3.1.2:



BASES

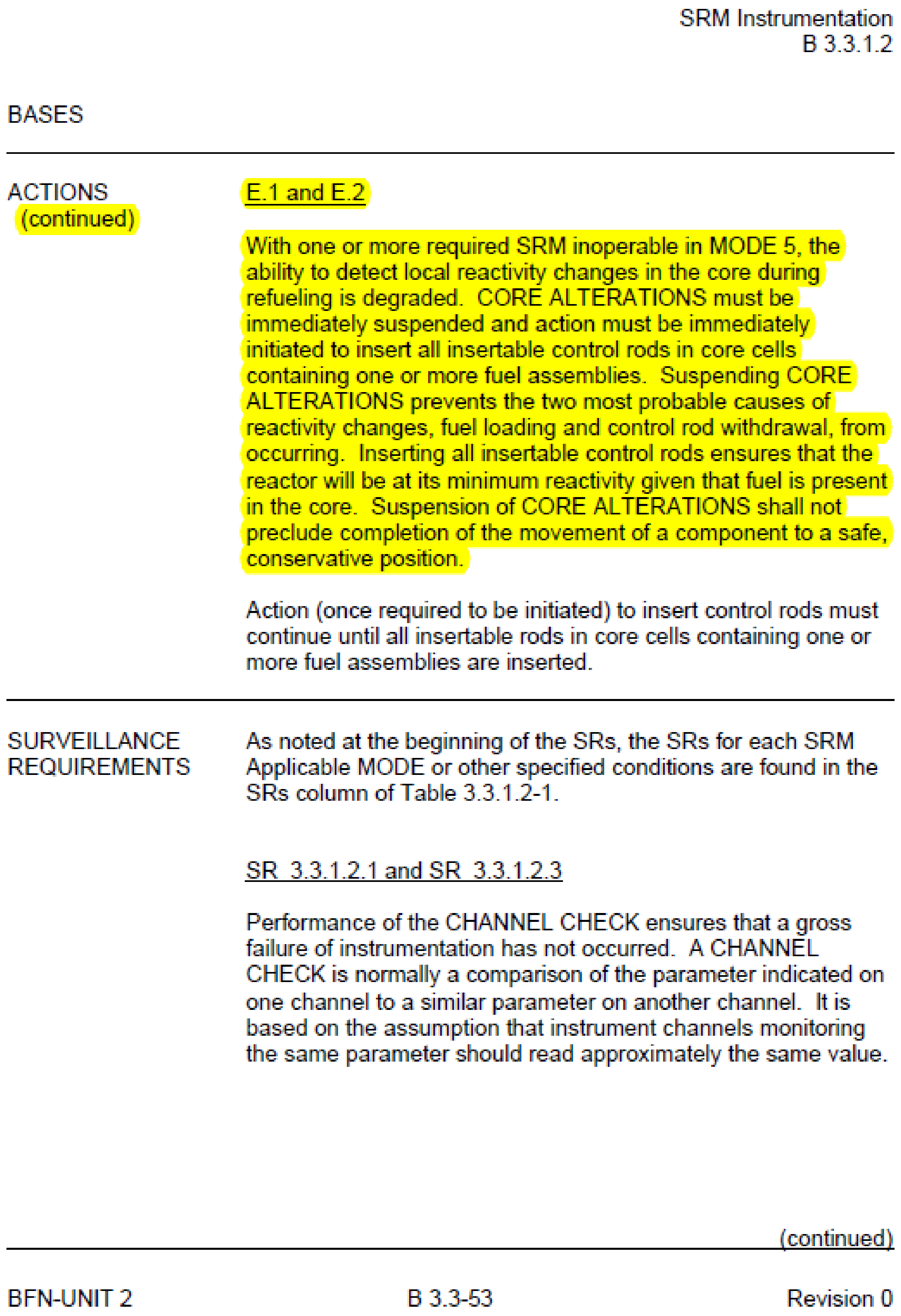
LCO (continued)

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

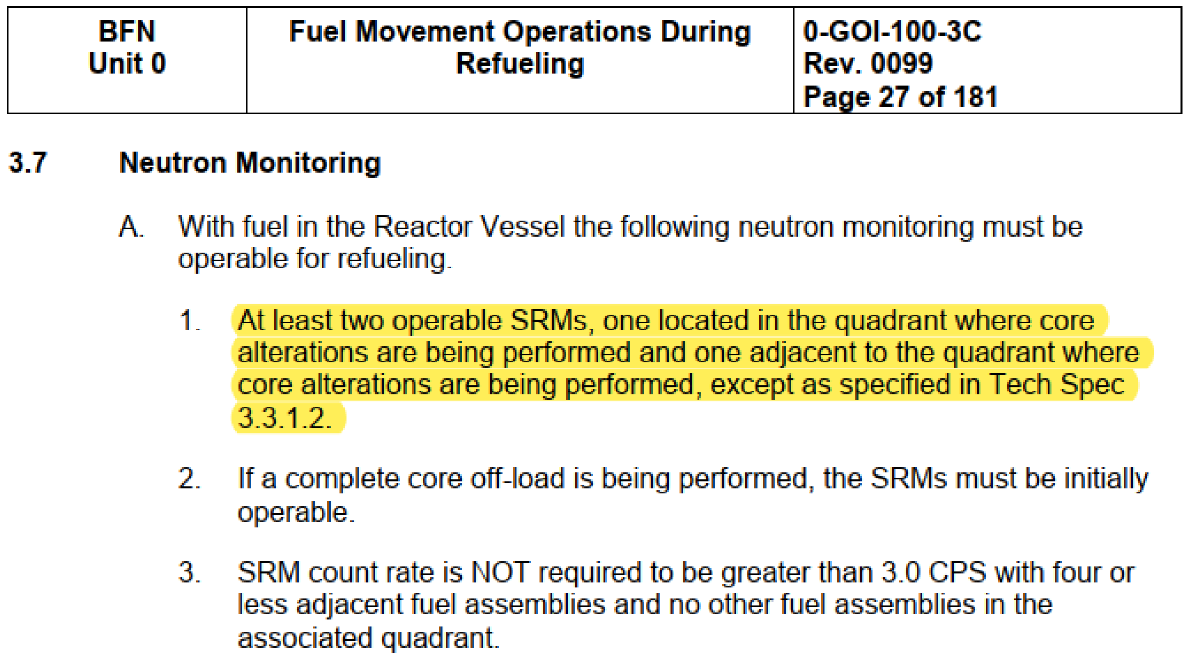
As noted at the beginning of the SRs, the SRs for each SRM Applicable MODE or other specified conditions are found in the SRs column of Table 3.3.1.2-1.

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:



BFN Unit 0	**Fuel Movement Operations During Refueling**	**0-GOI-100-3C Rev. 0099 Page 27 of 181**

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.

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.

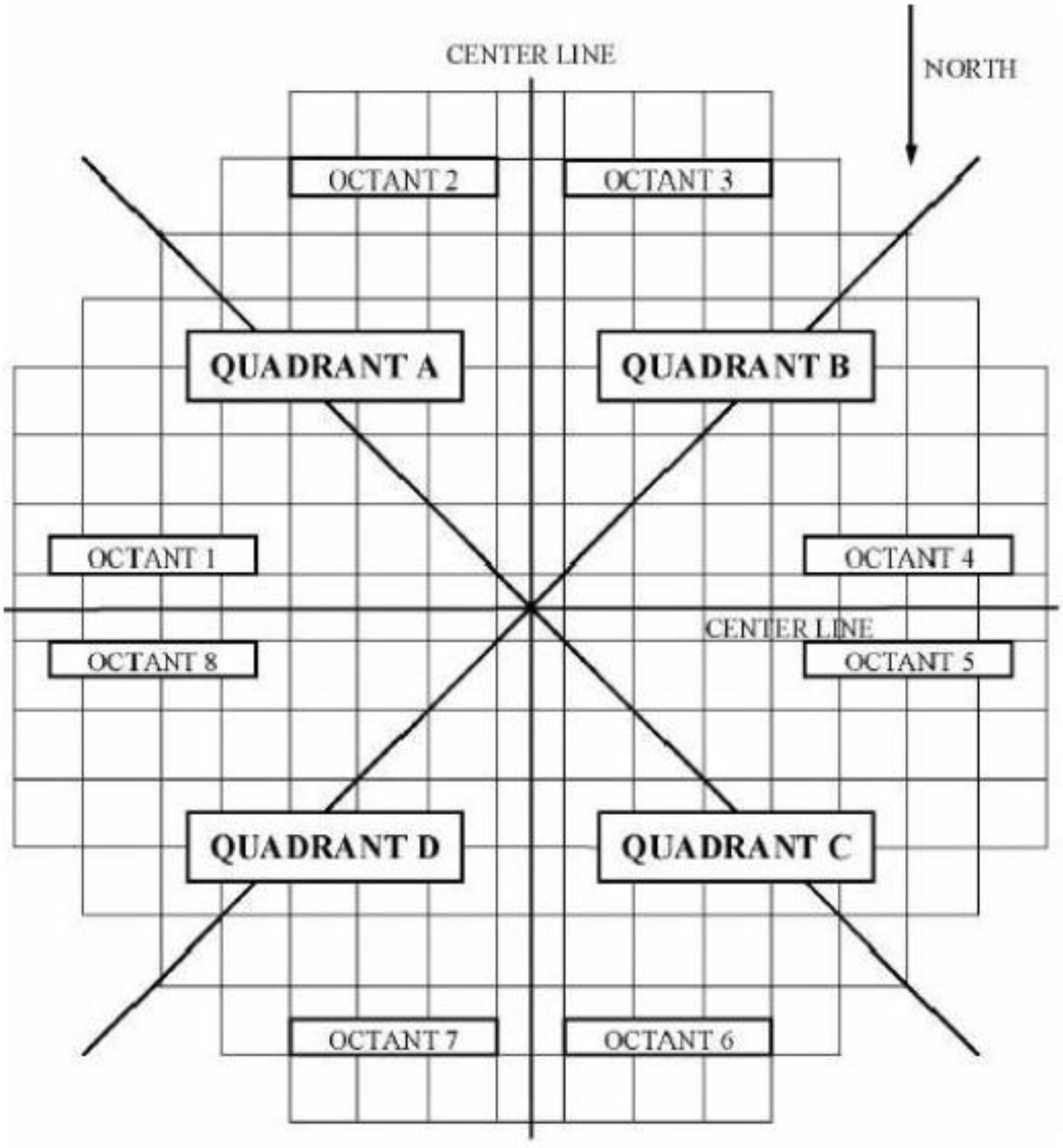
Fuel Moved in Quadrant	Required Operable SRM/FLC Quadrant Locations
A	A&B or A&D
B	A&B or B&C
C	B&C or C&D
D	C&D or A&D

(8)

1st Initials	Date	Time
IV Initials	Date	Time

BFN **Unit 0**	**Fuel Movement Operations During Refueling**	**0-GOI-100-3C** **Rev. 0099** **Page 175 of 181**

Attachment 14
(Page 1 of 1)
Core Quadrants



Examination Outline Cross-reference:	Level	RO	SRO
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG	Tier #	------	2
A2.10 (10CFR 55.43.5 – SRO Only)	Group #	------	1
Ability to (a) predict the impacts of the following on the Emergency Generators; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	264000A2.10	
- LOCA	Importance Rating	------	4.4

Proposed Question: **# 89**

Unit 3 is operating at 100% RTP when an event occurs with the following conditions:

- A transformer fault results in a Loss of Offsite Power to Unit 3
- Drywell Pressure **PEAKED** at 15 psig

Given the conditions above, ___**(1)**___ EDGs have started.

Subsequently, the following conditions are observed:

- ‘3A’ EDG is manually tripped due to an oil leak

To restore power to 3A 4KV Shutdown Board, the NUSO will direct re-energizing the board by aligning ___**(2)**___.

Note: 0-OI-82, Standby Diesel Generator System
0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV)/Station Blackout

A. (1) **ALL**
(2) ‘A’ EDG in accordance with 0-OI-82

B. (1) **ALL**
(2) ‘3B’ EDG in accordance with 0-AOI-57-1A

C. (1) **ONLY** Unit 3
(2) ‘A’ EDG in accordance with 0-OI-82

D. (1) **ONLY** Unit 3
(2) ‘3B’ EDG in accordance with 0-AOI-57-1A

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible in that there are numerous EDG alignments that can be achieved in accordance 0-OI-82, but re-energizing the 3A 4KV Shutdown Board is not one of them. If the board is de-energized, 0-AOI-57-1A has the direction to re-energize the board.

B **CORRECT:** *(See attached)* In accordance with 3-OI-82, Standby Diesel Generator System, a Pre-Accident Signal (High Drywell Pressure of 2.45 psig) on Unit 1, 2, or 3 will start all eight EDGs. A Loss of Offsite Power was given, at BFN, Unit 3 could experience a Loss of Offsite Power without the entire site losing power. For second part, in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV)/Station Blackout, Attachment 10 - Energizing 4KV SD BD 3EA or 3EC during Station Blackout, 1.0 [2] gives the required procedure steps to use '3B' EDG through 4KV SD BD 3EB to re-energize 4KV SD BD 3EA given that '3A' EDG is manually tripped.

C INCORRECT: First part is incorrect but is plausible if the candidate does not recognize the Pre-Accident signal and concludes that only Unit 3 EDGs would start due to the Loss of Offsite Power to Unit 3. Second part is incorrect but plausible *(See A).*

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

SRO Level Justification: Tests the candidate's ability to predict the impact of a Loss of Coolant Accident (LOCA) and Loss of Offsite Power on Emergency Diesel Generators and select procedures to mitigate plant 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. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-OI-82, Rev. 147 (Attach if not previously provided)
0-AOI-57-1A, Rev. 114

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.038 Obj. 4 (As available)

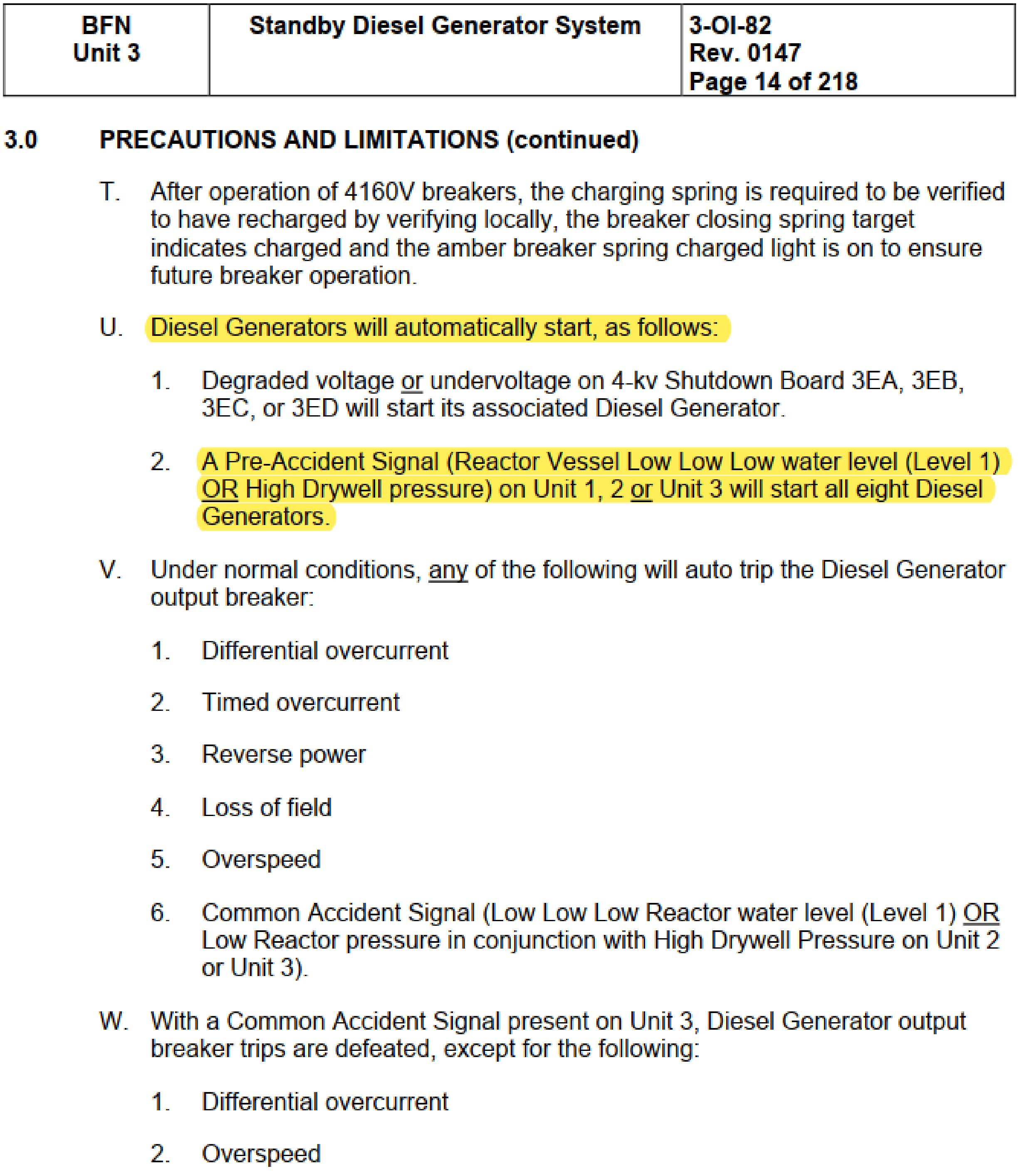
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

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

Comments:

Excerpt from 3-OI-82:



BFN Unit 3	**Standby Diesel Generator System**	**3-OI-82 Rev. 0147 Page 14 of 218**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

T. 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.

U. Diesel Generators will automatically start, as follows:

1. Degraded voltage or undervoltage on 4-kv Shutdown Board 3EA, 3EB, 3EC, or 3ED will start its associated Diesel Generator.
2. A Pre-Accident Signal (Reactor Vessel Low Low Low water level (Level 1) OR High Drywell pressure) on Unit 1, 2 or Unit 3 will start all eight Diesel Generators.

V. 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 (Level 1) OR Low Reactor pressure in conjunction with High Drywell Pressure on Unit 2 or Unit 3).

W. With a Common Accident Signal present on Unit 3, Diesel Generator output breaker trips are defeated, except for the following:

1. Differential overcurrent
2. Overspeed

Excerpt from 0-AOI-57-1A:

BFN **Unit 0**	**Loss of Offsite Power (161 and 500 KV)/Station Blackout**	**0-AOI-57-1A** **Rev. 0114** **Page 82 of 119**

Attachment 10
(Page 1 of 3)
ENERGIZING 4KV SD BD 3EA or 3EC during Station Blackout

NOTES

1) This attachment is used to energize 4kV Shutdown Board 3EA using Shutdown Board 3EB or to energize 4Kv Shutdown Board 3EC using 4Kv Shutdown Board 3ED.

2) The use of 3EB-3EA and 3ED -3EC cross-ties are required to mitigate Unit 3 Station Blackout (SBO) scenarios (i.e. only 3B or 3D Diesel Generators available).

1.0 BOTH 4KV SHUTDOWN BOARDS 3EA AND 3EC ARE DE-ENERGIZED

[1] **IF** both 4Kv Shutdown Boards 3EA and 3EC are de-energized, **THEN**

PERFORM the following:

[1.1] **DETERMINE** which 4Kv Shutdown Board is energized (3EB or 3ED).

[1.2] **RE-ENERGIZE** the desired shutdown board (3EA or 3EC) using the available board (3EB or 3ED) using Step 1.0[2] or STEP 1.0[3] as applicable.

[2] **IF** desired to re-energize 4KV Shutdown Board 3EA using cross-tie from 4KV Shutdown Board 3EB, **THEN**

PERFORM the following (otherwise N/A):

[2.1] **ENSURE** Diesel Generator 3B is supplying 4Kv Shutdown Board 3EB.

[2.2] **ENSURE** 4Kv Shutdown Board 3EA is de-energized.

[2.3] **CHECK OPEN** 4Kv Bus Tie Board breaker 1732 (3-IL-210-1/4B).

[2.4] **ENSURE** 4KV SD BD 3EA AUTO/LOCKOUT RESET switch, 3-43-211-3EA, is tripped to MANUAL.

[2.5] **ENSURE** 4KV SD BD 3EB AUTO/LOCKOUT RESET switch, 3-43-211-3EB, is tripped to MANUAL.

Excerpts from 3-OI-82: Supports Distractors A(2), C(2)

BFN Unit 3	**Standby Diesel Generator System**	**3-OI-82 Rev. 0147 Page 4 of 218**

Table of Contents (continued)

Table of Contents (continued)

Examination Outline Cross-reference:

400000 (SF8 CCS) Component Cooling Water

G2.2.42 (10CFR 55.43.2 - SRO Only)

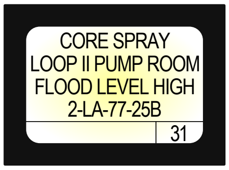
Ability to recognize system parameters that are entry-level conditions for technical specifications.

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	400000G2.2.42	
Importance Rating	------	4.6

Proposed Question: **# 90**

A leak from the Unit 2 Loop II Core Spray Room Cooler has resulted in the following:

- CORE SPRAY LOOP II PUMP ROOM FLOOD LEVEL HIGH, (2-9-4C, Window 31), is in alarm
- EECW to the Loop II Core Spray Room Cooler was determined to be the source of the leak and has been isolated



Given the conditions above, Loop II Core Spray ___**(1)**___ OPERABLE and entry into 2-EOI-3, Secondary Containment Control ___**(2)**___ required.

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: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the Core Spray System will still operate with the Core Spray Room Cooler out of service, however it is not Tech Spec OPERABLE. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A)*. Second part is incorrect but plausible if NUSO candidate fails to recognize that the given valid alarm directs EOI-3 entry.

C **CORRECT:** *(See attached)* In accordance with TR 3.5.3, Equipment Area Coolers, the Cooler associated with each set of Core Spray Pumps (A and C or B and D) must be OPERABLE when that Core Spray System is considered to be OPERABLE. With the Equipment Area Cooler INOPERABLE, TR 3.5.3 Condition A requires the pumps served by the Cooler to be declared INOPERABLE immediately (See Tech Spec 3.5.1 for Core Spray). For second part, in accordance with the given 1-ARP-9-4C, Window 31, the alarm setpoint is ≥ 2 inches of water from the Core Spray Pump room floor due to the given leak, which requires the NUSO to direct the Operator to enter 2-EOI-3, Secondary Containment Control, to take mitigating actions.

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

SRO Level Justification: Tests the candidate's ability to recognize the impact Secondary Containment leakage has on the Core Spray Room Cooler and Pump operations as it relates to Emergency Operating Instructions entry conditions. Core Spray Tech Spec Operability is impacted based on the inability for the Room Coolers to maintain the Core Spray Pump heat load within a required operational band. This question is 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-EOI-3, Rev. 18 (Attach if not previously provided)
2-ARP-9-4C, Rev. 36
U2 TR 3.5.3, Rev. 0
U2 Tech Spec 3.5.1 Amend. 294

Proposed references to be provided to applicants during examination: **CORE SPRAY LOOP II PUMP ROOM FLOOD LEVEL HIGH, (2-9-4C, Window 31)**

Learning Objective: OPL171.204 Obj. 2 (As available)

Question Source:	Bank #	BFN 18-04 #90	
	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
55.43 **X**

Comments:

Copy of Bank Question:

Proposed Question: **# 90**

A leak from the Unit 1 Loop II Core Spray Room Cooler has resulted in the following:

- CORE SPRAY LOOP II PUMP ROOM FLOOD LEVEL HIGH, (1-9-4C, Window 31), is in alarm
- Loop II Core Spray (CS) Room Cooler has tripped **AND** will **NOT** reset

Which ONE of the following completes the statements below?

Loop II Core Spray ___**(1)**___OPERABLE.
Entry into 1-EOI-3, Secondary Containment Control ___**(2)**___ required.

A. **(1)** is
(2) is

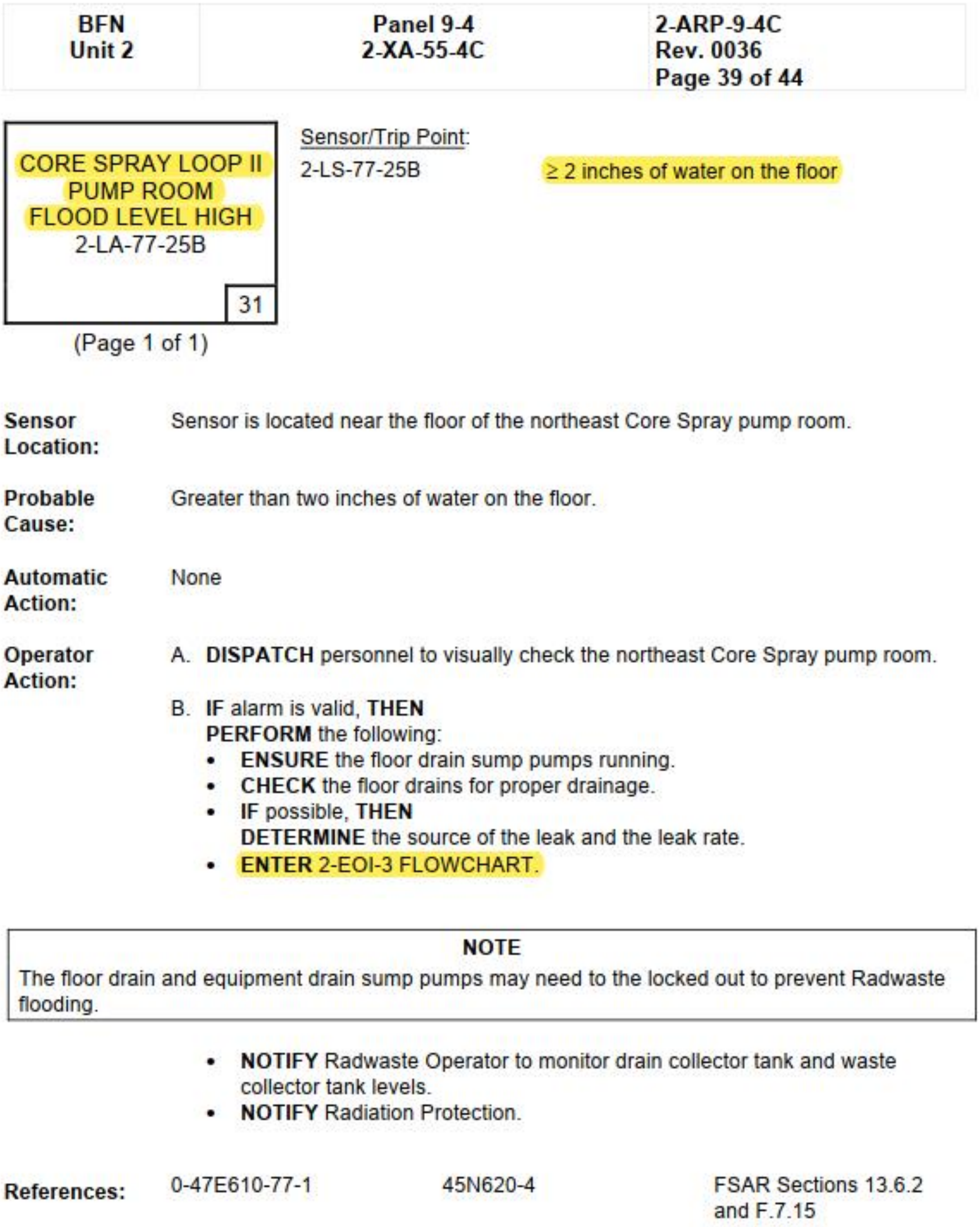
B. **(1)** is
(2) is NOT

C. **(1)** is NOT
(2) is

D. **(1)** is NOT
(2) is NOT

Proposed Answer: C

Excerpt from 2-ARP-9-4C:



BFN **Unit 2**	**Panel 9-4** **2-XA-55-4C**	**2-ARP-9-4C** **Rev. 0036** **Page 39 of 44**

CORE SPRAY LOOP II
PUMP ROOM
FLOOD LEVEL HIGH
2-LA-77-25B
31

(Page 1 of 1)

Sensor/Trip Point:

2-LS-77-25B	≥ 2 inches of water on the floor

Sensor Location:	Sensor is located near the floor of the northeast Core Spray pump room.
Probable Cause:	Greater than two inches of water on the floor.
Automatic Action:	None

Operator Action:

A. **DISPATCH** personnel to visually check the northeast Core Spray pump room.

B. **IF** alarm is valid, **THEN**
PERFORM the following:
- **ENSURE** 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.

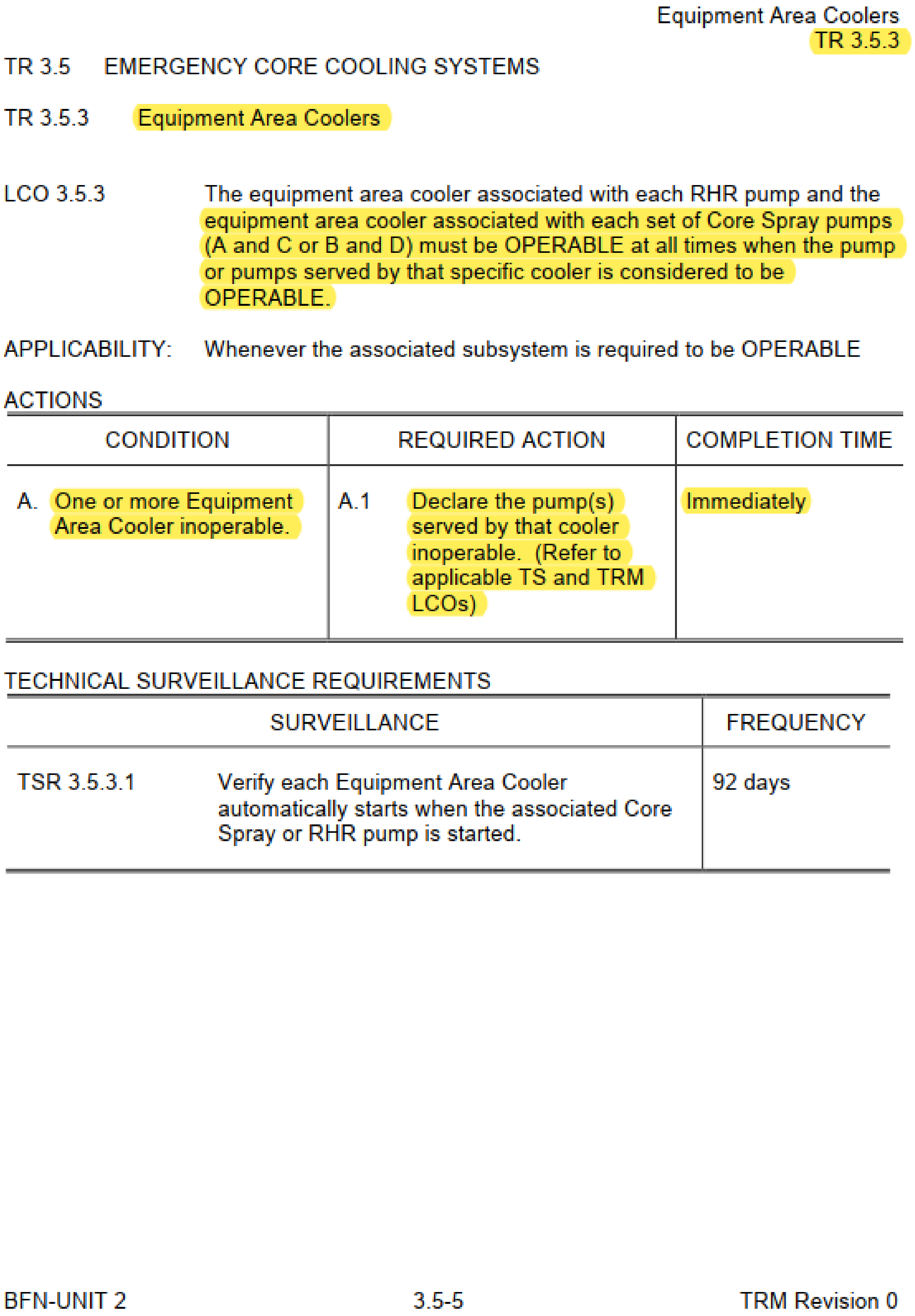
NOTE

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

Excerpt from U2 TR 3.5.3:



Equipment Area Coolers
TR 3.5.3

TR 3.5 EMERGENCY CORE COOLING SYSTEMS

TR 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

ACTIONS

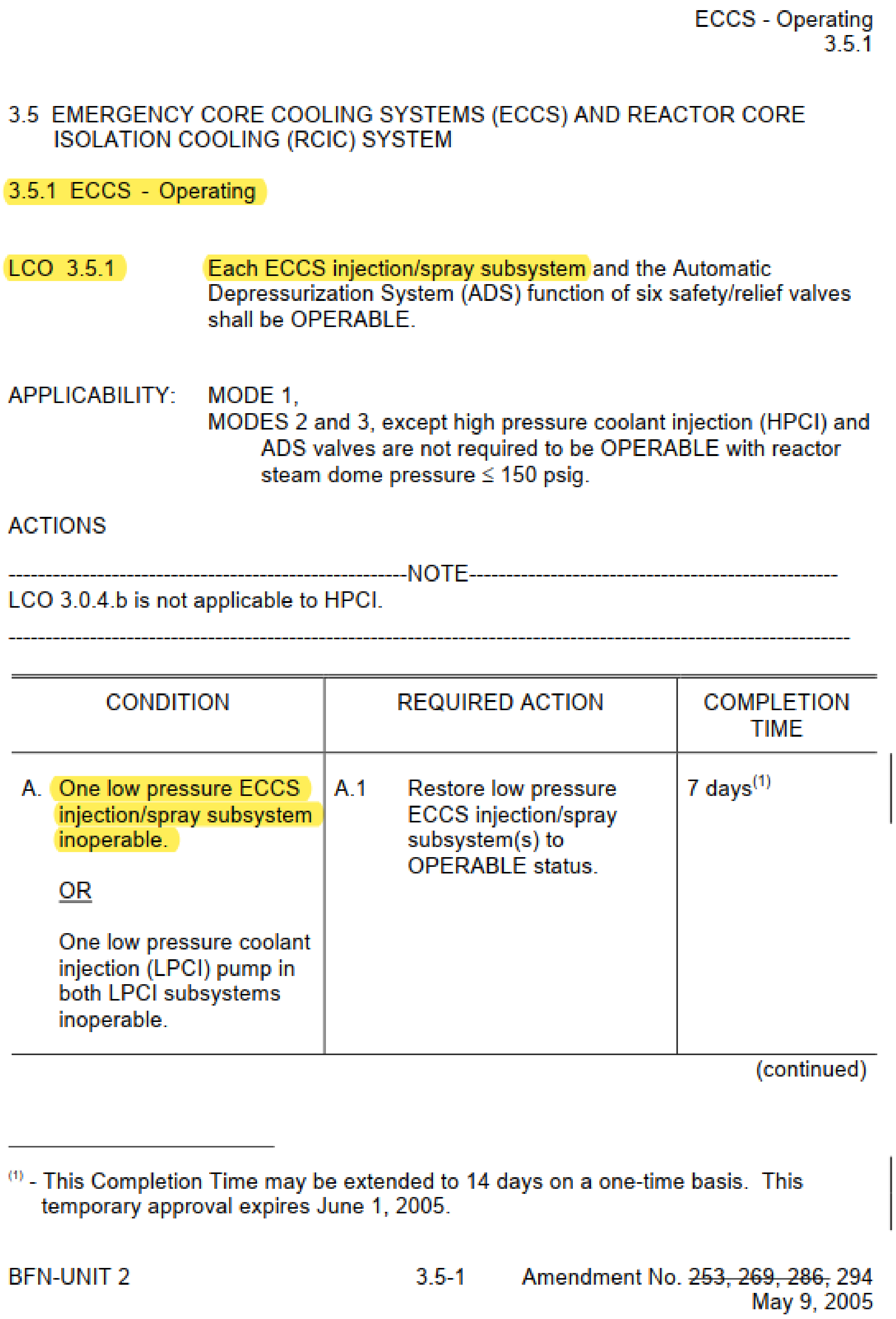
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Equipment Area Cooler inoperable.	A.1 Declare the pump(s) served by that cooler inoperable. (Refer to applicable TS and TRM LCOs)	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.5.3.1	Verify each Equipment Area Cooler automatically starts when the associated Core Spray or RHR pump is started.	92 days

BFN-UNIT 2 3.5-5 TRM Revision 0

Excerpt from U2 Tech Spec 3.5.1 referenced from U2 TR 3.5.3:



ECCS - Operating
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)

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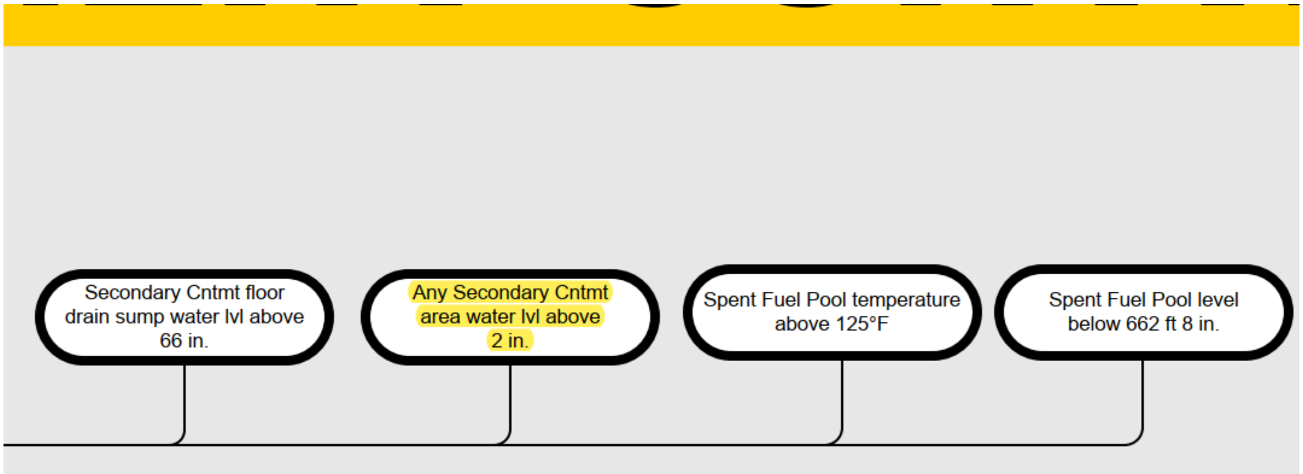
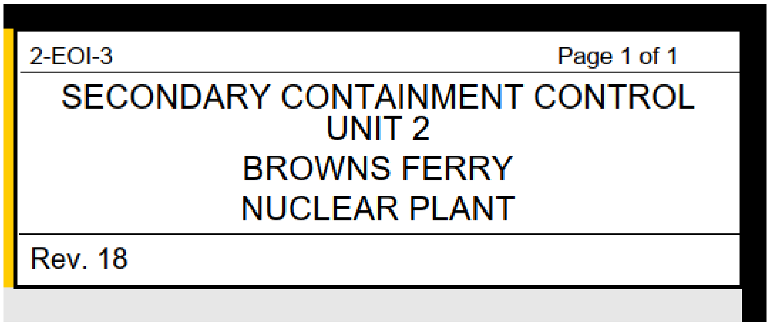
(1) - This Completion Time may be extended to 14 days on a one-time basis. This temporary approval expires June 1, 2005.

BFN-UNIT 2 3.5-1 Amendment No. ~~253, 269, 286,~~ 294
May 9, 2005

Excerpt from 2-EOI-3:




2-EOI-3 Page 1 of 1

SECONDARY CONTAINMENT CONTROL
UNIT 2
BROWNS FERRY
NUCLEAR PLANT

Rev. 18

Examination Outline Cross-reference:	Level	RO	SRO
204000 (SF2 RWCU) Reactor Water Cleanup	Tier #	------	2
A2.05 (10CFR 55.43.5 - SRO Only)	Group #	------	2
Ability to (a) predict the impacts of the following on the Reactor Water Cleanup System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: • Abnormal Valve Position	K/A #	204000A2.05	
	Importance Rating	------	3.0

Proposed Question: **# 91**

Unit 3 is operating at 100% RTP when the following conditions occur:

At 0600 on 5/16/22:

- RPS 3A trips
- 3-FCV-69-1, RWCU INBOARD SUCTION ISOLATION VALVE indicates closed
- 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION VALVE indicates open
- 3-FCV-69-12, RWCU RETURN ISOLATION VALVE indicates open

At 0615 on 5/16/22:

- 3A RPS Bus is energized on its alternate supply

The NUSO will enter Tech Spec 3.6.1.3 CONDITION ___**(1)**___ and Chemistry must obtain the first Reactor Coolant sample by ___**(2)**___.

[REFERENCE PROVIDED]

A. (1) A **ONLY**
(2) 1000

B. (1) A **ONLY**
(2) 1015

C. (1) A **AND** B
(2) 1000

D. (1) A **AND** B
(2) 1015

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* Given that only 3A RPS Bus initially trips, 3-FCV-69-1, 2, and 3 valves all receive a PCIS Group 3 Isolation signal to CLOSE in accordance with 3-OI-99, Reactor Protection System Attachment 1 – RPS Bus A or B Power Transfer. In accordance with 3-AOI-64-2a, Group 3 RWCU Isolation, both RWCU Pumps tripped subsequently from suction valve position (only 3-FCV-69-1 closing in this case). However, since 2-FCV-69-2 failed to close on the loss of 3A RPS as given in the stem, it is considered INOPERABLE in accordance with Tech Spec 3.6.1.3. As far as 3-FCV-69-12, which isolates exactly like a Primary Containment Isolation Valve (PCIV, 3-FCV-69-1 and 2), however it is not a PCIV. The candidate must be aware of this to realize only 1 PCIV (2-FCV-69-2) is INOPERABLE, thus the NUSO will enter Tech Spec 3.6.1.3 CONDITION A ONLY. For second part, in accordance with TRM 3.4.1, Coolant Chemistry with the given initial event start time of 0600, sampling is required to occur within 4 hours of losing continuous sampling. Therefore, Chemistry must obtain the first Reactor Coolant sample by 1000. The application of start times for LCO's are being tested given the valves (3-FCV-69-1, 2, and 3) position can be verified as soon as the system isolates. The candidate must first know the valve power supplies and then apply the start time to the initial event, not to when power is restored and all positions verified.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible if the candidate misapplies the point of discovery to when power is restored at 0615, given some power losses remove the ability to determine valve position. This would lead them to believe that Chemistry is required to obtain the first sample by 1015. However, the RWCU valves given in the stem will retain light indication despite the power loss, thus valve position can be verified from the initiating event at 0600.

C INCORRECT: First part is incorrect but plausible in that 3-FCV-69-12 repositions from signals generated in PCIS Group 3 logic, however it is not a PCIV. The fact that it does not close does not qualify for entering Tech Spec 3.6.1.3 Condition B for 2 valves being INOPERABLE. It is plausible if the candidate confuses the shared logic versus what is required to occur by Tech Specs. 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 predict the impact of loss of power and valve failures in the Reactor Water Cleanup System as it relates to the Reactor Protection System and Primary Containment Isolation System and them apply the correct Technical Specification requirements from the given plant conditions. 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 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): 3-OI-69, Rev. 110 (Attach if not previously provided)

3-OI-99, Rev. 67

3-AOI-64-2a, Rev. 12

U3 TR 3.4.1, Rev. 21

U3 Tech Spec 3.6.1.3, Amend. 212

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

Proposed references to be provided to applicants during examination: **U3 TR 3.4.1 (No Bases)**
U3 Tech Spec 3.6.1.3
(No Bases)

Learning Objective: OPL171.204 Obj. 2 (As available)

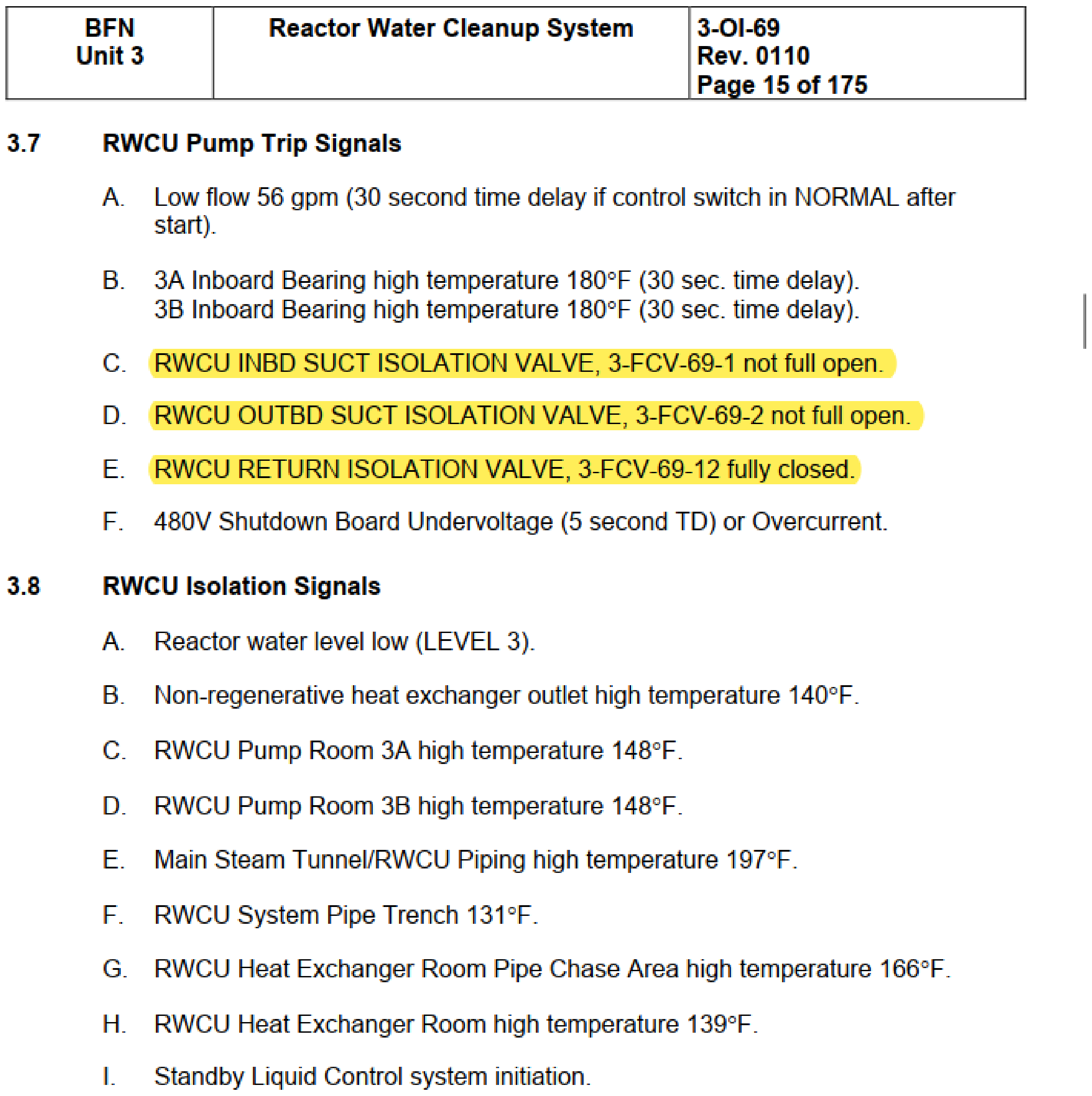
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

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

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

Comments:

Excerpt from 3-OI-69:



BFN Unit 3	**Reactor Water Cleanup System**	**3-OI-69 Rev. 0110 Page 15 of 175**

3.7 RWCU Pump Trip Signals

A. Low flow 56 gpm (30 second time delay if control switch in NORMAL after start).

B. 3A Inboard Bearing high temperature 180°F (30 sec. time delay).
3B Inboard Bearing high temperature 180°F (30 sec. time delay).

C. RWCU INBD SUCT ISOLATION VALVE, 3-FCV-69-1 not full open.

D. RWCU OUTBD SUCT ISOLATION VALVE, 3-FCV-69-2 not full open.

E. RWCU RETURN ISOLATION VALVE, 3-FCV-69-12 fully closed.

F. 480V Shutdown Board Undervoltage (5 second TD) or Overcurrent.

3.8 RWCU Isolation Signals

A. Reactor water level low (LEVEL 3).

B. Non-regenerative heat exchanger outlet high temperature 140°F.

C. RWCU Pump Room 3A high temperature 148°F.

D. RWCU Pump Room 3B high temperature 148°F.

E. Main Steam Tunnel/RWCU Piping high temperature 197°F.

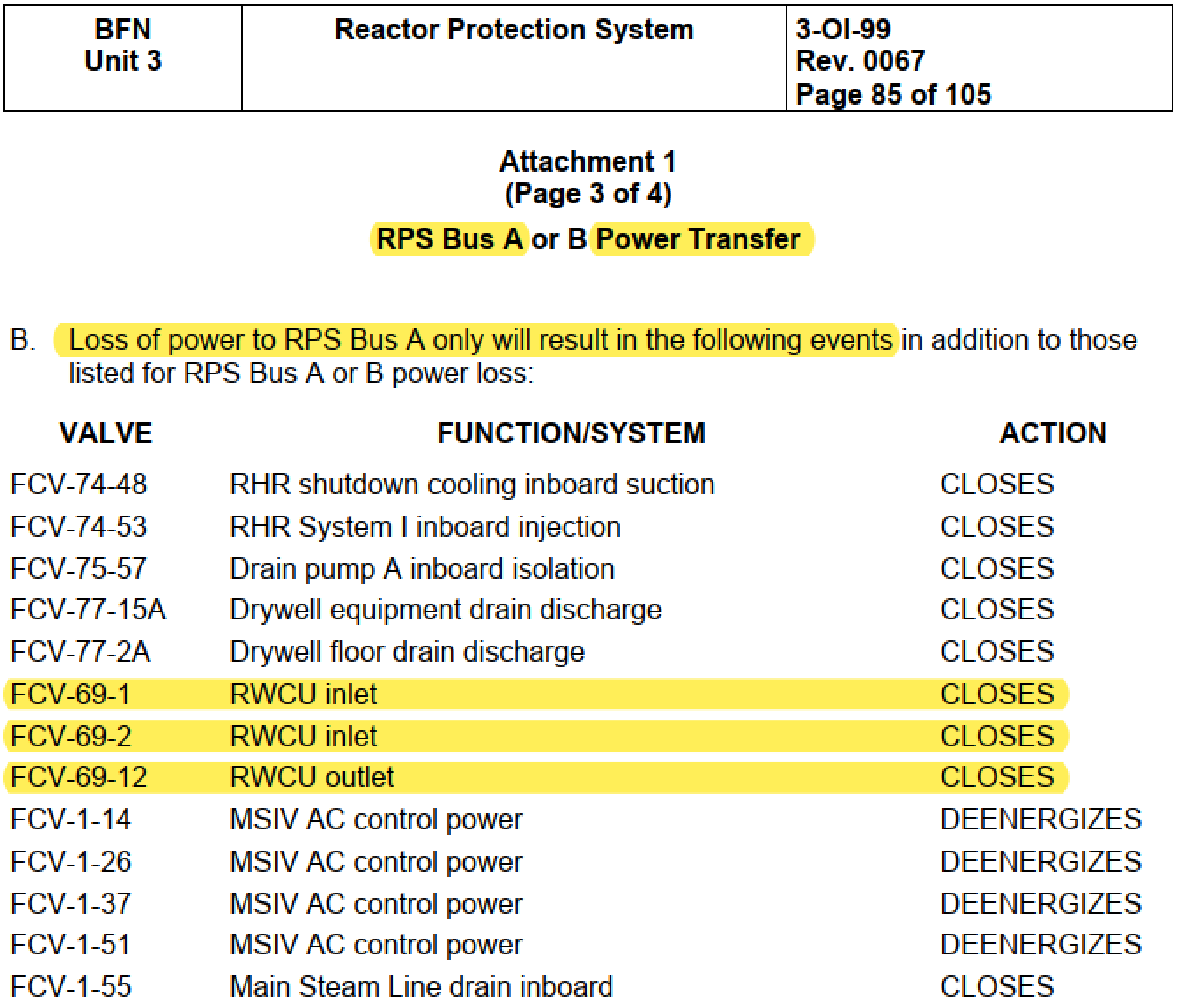
F. RWCU System Pipe Trench 131°F.

G. RWCU Heat Exchanger Room Pipe Chase Area high temperature 166°F.

H. RWCU Heat Exchanger Room high temperature 139°F.

I. Standby Liquid Control system initiation.

Excerpt from 3-OI-99:



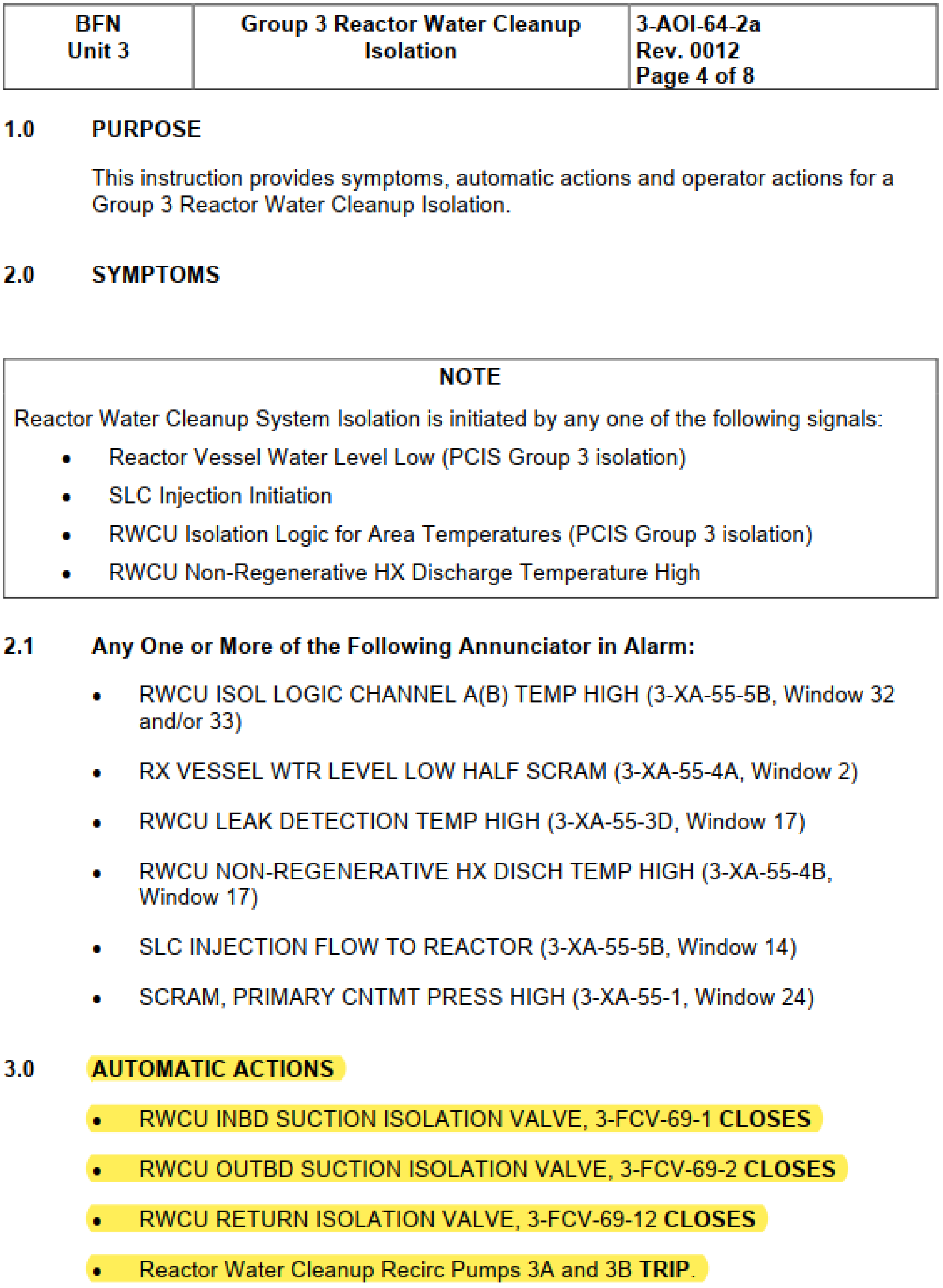
BFN **Unit 3**	**Reactor Protection System**	**3-OI-99** **Rev. 0067** **Page 85 of 105**

Attachment 1
(Page 3 of 4)
RPS Bus A or B Power Transfer

B. Loss of power to RPS Bus A only will result in the following events in addition to those listed for RPS Bus A or B power loss:

VALVE	FUNCTION/SYSTEM	ACTION
FCV-74-48	RHR shutdown cooling inboard suction	CLOSES
FCV-74-53	RHR System I inboard injection	CLOSES
FCV-75-57	Drain pump A inboard isolation	CLOSES
FCV-77-15A	Drywell equipment drain discharge	CLOSES
FCV-77-2A	Drywell floor drain discharge	CLOSES
FCV-69-1	RWCU inlet	CLOSES
FCV-69-2	RWCU inlet	CLOSES
FCV-69-12	RWCU outlet	CLOSES
FCV-1-14	MSIV AC control power	DEENERGIZES
FCV-1-26	MSIV AC control power	DEENERGIZES
FCV-1-37	MSIV AC control power	DEENERGIZES
FCV-1-51	MSIV AC control power	DEENERGIZES
FCV-1-55	Main Steam Line drain inboard	CLOSES

Excerpt from 3-AOI-64-2a:



BFN **Unit 3**	**Group 3 Reactor Water Cleanup Isolation**	**3-AOI-64-2a** **Rev. 0012** **Page 4 of 8**

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)
- SLC Injection Initiation
- RWCU Isolation Logic for Area Temperatures (PCIS Group 3 isolation)
- RWCU Non-Regenerative HX Discharge Temperature High

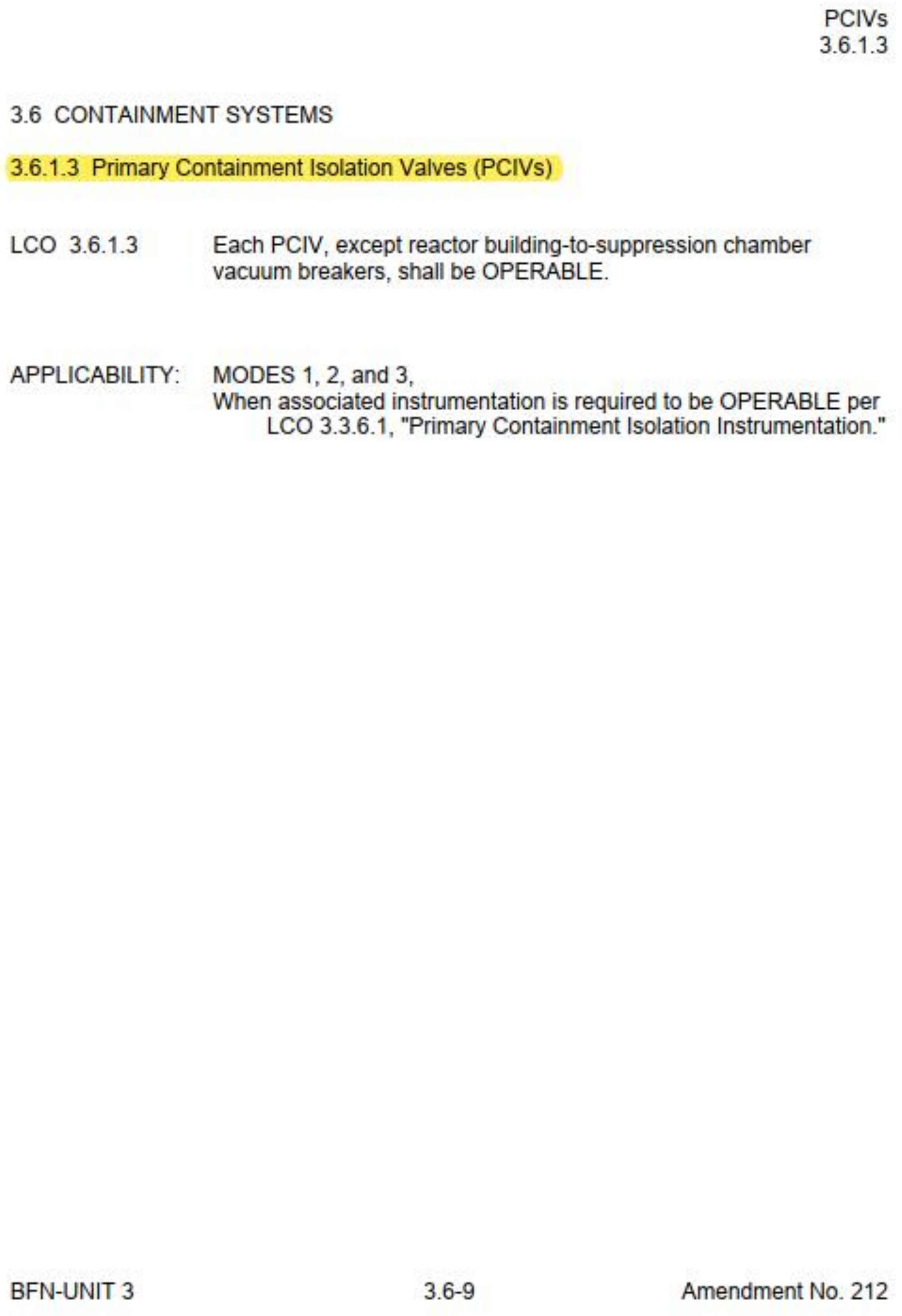
2.1 Any One or More of the Following Annunciator in Alarm:

- RWCU ISOL LOGIC CHANNEL A(B) TEMP HIGH (3-XA-55-5B, Window 32 and/or 33)
- RX VESSEL WTR LEVEL LOW HALF SCRAM (3-XA-55-4A, Window 2)
- RWCU LEAK DETECTION TEMP HIGH (3-XA-55-3D, Window 17)
- RWCU NON-REGENERATIVE HX DISCH TEMP HIGH (3-XA-55-4B, Window 17)
- SLC INJECTION FLOW TO REACTOR (3-XA-55-5B, Window 14)
- SCRAM, PRIMARY CNTMT PRESS HIGH (3-XA-55-1, Window 24)

3.0 AUTOMATIC ACTIONS

- RWCU INBD SUCTION ISOLATION VALVE, 3-FCV-69-1 **CLOSES**
- RWCU OUTBD SUCTION ISOLATION VALVE, 3-FCV-69-2 **CLOSES**
- RWCU RETURN ISOLATION VALVE, 3-FCV-69-12 **CLOSES**
- Reactor Water Cleanup Recirc Pumps 3A and 3B **TRIP**.

Excerpts from U3 Tech Spec 3.6.1.3:

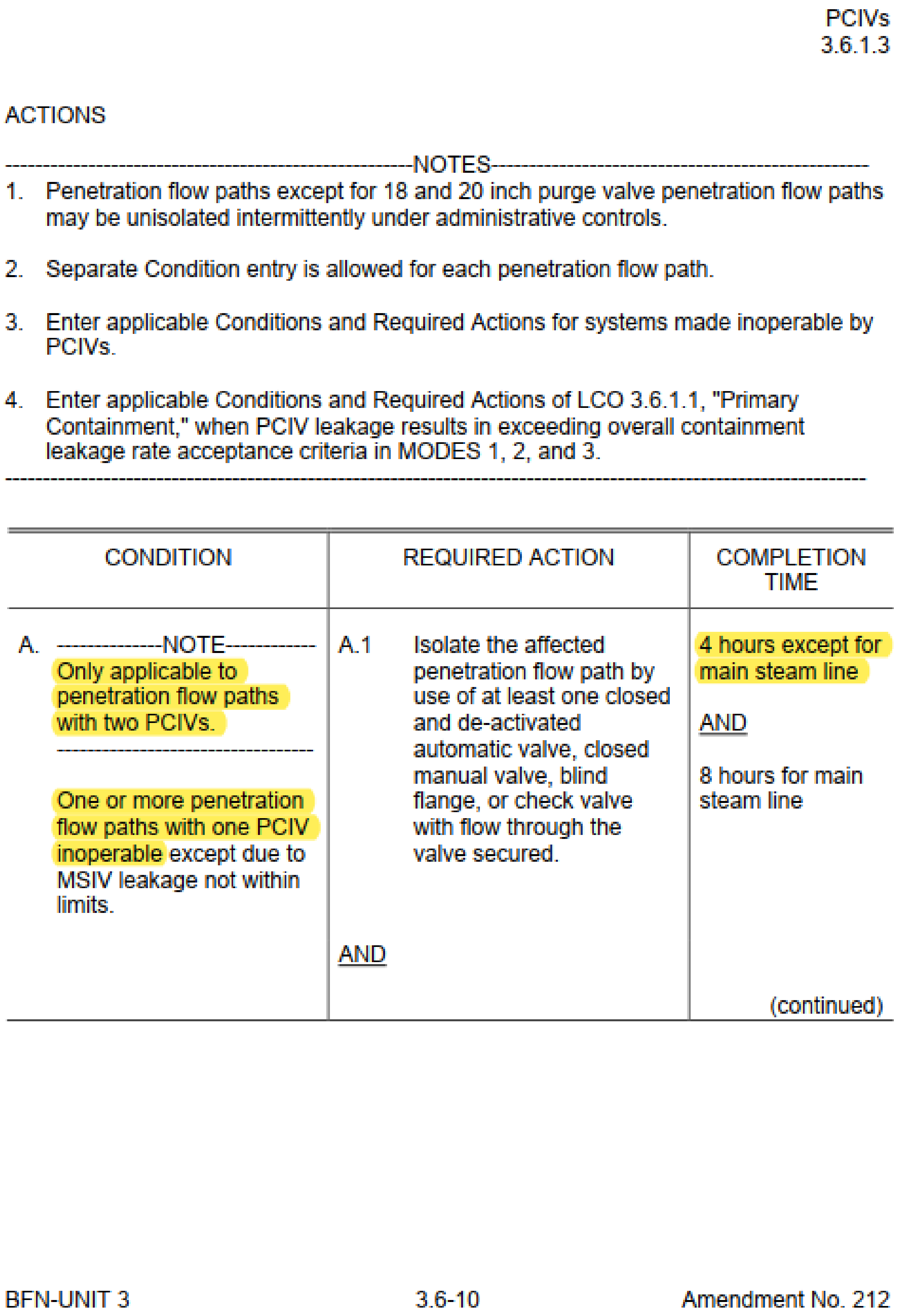


3.6 CONTAINMENT SYSTEMS

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."



PCIVs
3.6.1.3

ACTIONS

---NOTES---

1. Penetration flow paths except for 18 and 20 inch purge valve penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

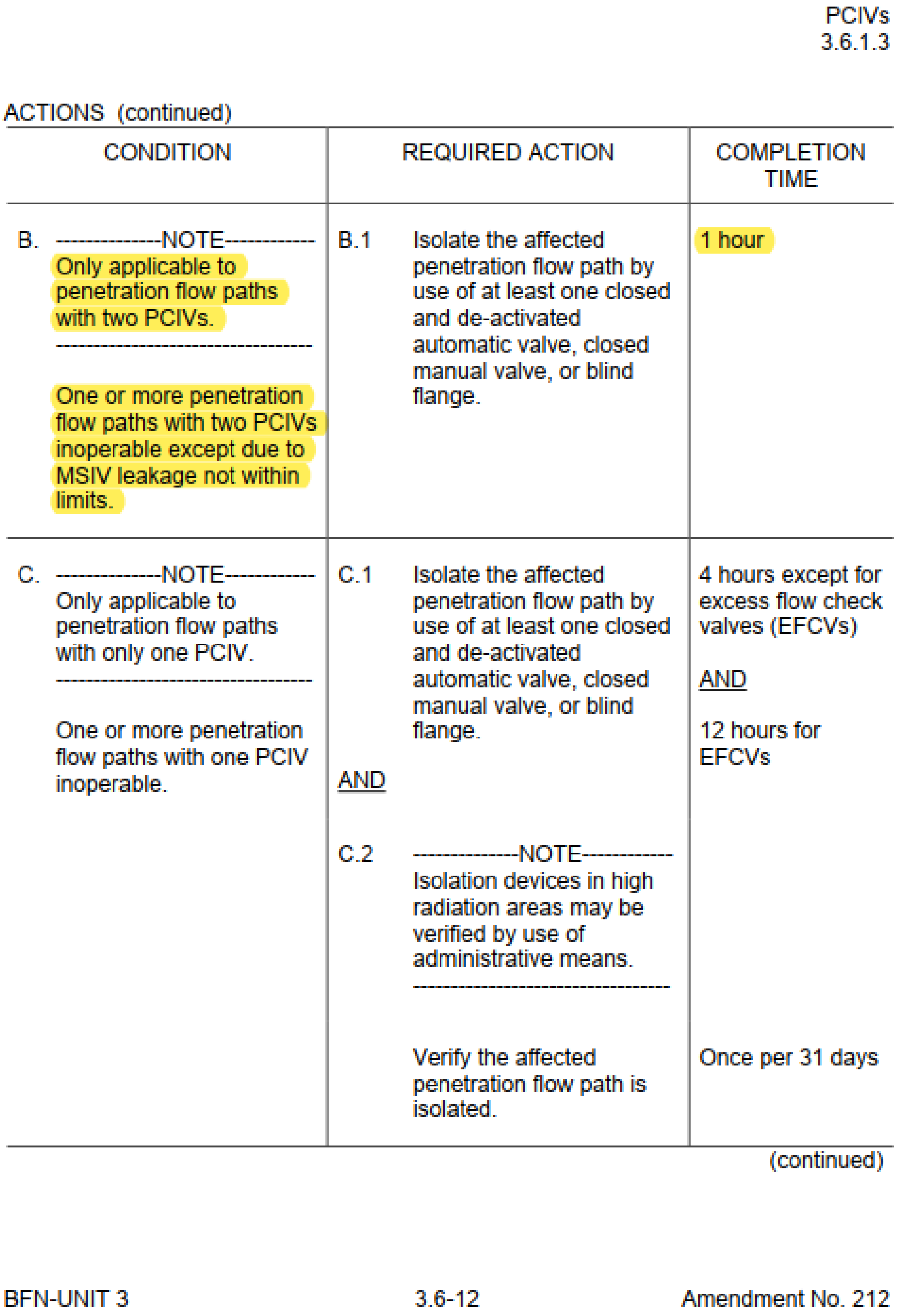
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.

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	REQUIRED ACTION		COMPLETION TIME
A. --------------NOTE------------ Only applicable to penetration flow paths with two PCIVs. ------------------------------------ One or more penetration flow paths with one PCIV inoperable except due to MSIV leakage not within limits.	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.	4 hours except for main steam line <u>AND</u> 8 hours for main steam line
	<u>AND</u>		
			(continued)

BFN-UNIT 3 3.6-10 Amendment No. 212

Supports Distractor C(1), D(1):



PCIVs
3.6.1.3

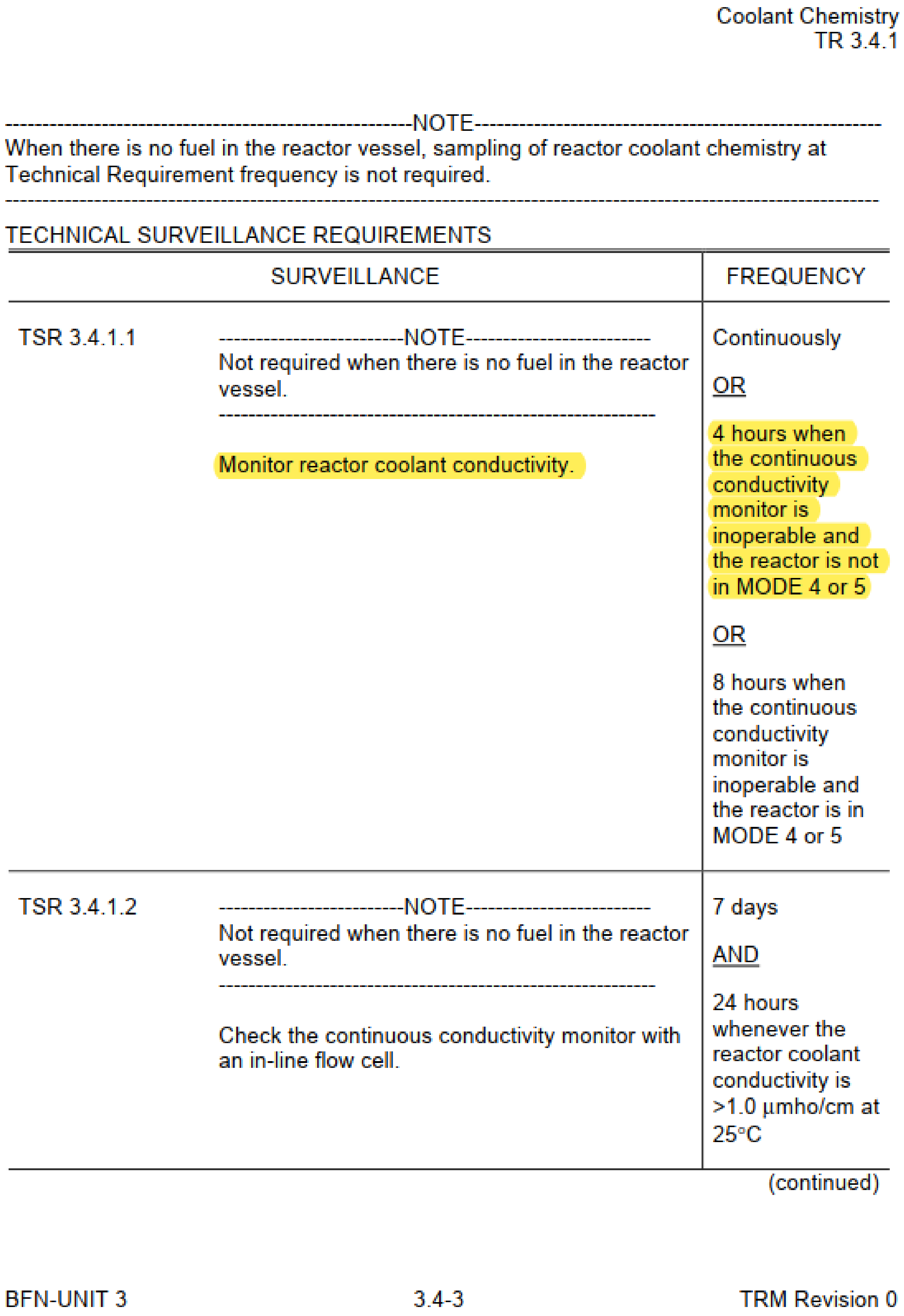
ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------------NOTE------------ Only applicable to penetration flow paths with two PCIVs. ---------------------------------- One or more penetration flow paths with two PCIVs inoperable except due to MSIV leakage not within limits.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. --------------NOTE------------ Only applicable to penetration flow paths with only one PCIV. ---------------------------------- One or more penetration flow paths with one PCIV inoperable.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours except for excess flow check valves (EFCVs) <u>AND</u> 12 hours for EFCVs
	<u>AND</u>		
	C.2	--------------NOTE------------ Isolation devices in high radiation areas may be verified by use of administrative means. ---------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days

(continued)

BFN-UNIT 3 3.6-12 Amendment No. 212

Excerpts from U3 TR 3.4.1:



Coolant Chemistry
TR 3.4.1

----------NOTE----------
When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at Technical Requirement frequency is not required.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.1.1	----------NOTE---------- Not required when there is no fuel in the reactor vessel. ---------- Monitor reactor coolant conductivity.	Continuously OR 4 hours when the continuous conductivity monitor is inoperable and the reactor is not in MODE 4 or 5 OR 8 hours when the continuous conductivity monitor is inoperable and the reactor is in MODE 4 or 5
TSR 3.4.1.2	----------NOTE---------- Not required when there is no fuel in the reactor vessel. ---------- Check the continuous conductivity monitor with an in-line flow cell.	7 days AND 24 hours whenever the reactor coolant conductivity is >1.0 µmho/cm at 25°C

(continued)

BFN-UNIT 3 3.4-3 TRM Revision 0

Examination Outline Cross-reference:

241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating

G2.4.30 (10CFR 55.43.5 - SRO Only)

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

Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	241000G2.4.30	
Importance Rating	------	4.1

Proposed Question: **# 92**

Unit 2 is operating at 50% RTP when the following conditions occur:

- All Turbine Bypass Valves have failed open
- Reactor Steam Dome Pressure is 850 psig and lowering
- Prior to Operators performing any manual actions, the Reactor SCRAMs

The SCRAM occurred directly from a ____**(1)**____ signal.

A ____**(2)**____ report to the NRC is required in accordance with NPG-SPP-3.5, Regulatory Reporting Requirements.

[REFERENCE PROVIDED]

A. (1) Reactor Pressure
(2) 4 Hour

B. (1) Reactor Pressure
(2) 8 Hour

C. (1) MSIV position
(2) 4 Hour

D. (1) MSIV position
(2) 8 Hour

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that High Reactor Pressure will directly SCRAM the Reactor, but low Reactor Pressure does not. Low Reactor Pressure initiates a PCIS Group 1 MSIV isolation while in MODE 1 (RUN) with the subsequent MSIV positions generating the SCRAM signal. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that in accordance with NPG-SPP-3.5, Reporting Requirements, Section 3.1.D 3.a.(1), an 8-hour Notification is required for a Reactor SCRAM (MODE 4 or 5).

C **CORRECT:** *(See attached)* In accordance with 2-OI-1, Main Steam System, when the REACTOR MODE SWITCH in RUN, a PCIS Group 1 Isolation occurs when Reactor Pressure lowers to 852 psig causing the MSIVs to close. In accordance with 2-OI-99, RPS, once the MSIVs are less than 90% open with the REACTOR MODE SWITCH in RUN, an automatic Reactor SCRAM signal is generated from the RPS logic. While the specific MODE is not given in the stem, it is inferred with Reactor Power at 50%. For second part, in accordance with NPG-SPP-3.5, Reporting Requirements, Section 3.1.C.3, the following criteria require 4-hour Notification; Any event or condition that results in actuation of the Reactor Protection System (RPS) when the Reactor is critical.

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

SRO Level Justification: Tests the candidate’s knowledge to determine and interpret plant conditions relative to automatic SCRAM signals caused from a failure of the Turbine Pressure Regulating System and the appropriate NRC Reporting 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 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-1, Rev. 55 (Attach if not previously provided)
2-OI-99, Rev. 96
NPG-SPP 3.5, Rev. 17

Proposed references to be provided to applicants during examination: **NPG-SPP-3.5 Attachment 1 (Page 1 through 18)**

Learning Objective: OPL171.017 Obj. 2a (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:

Excerpt from 2-OI-1:

BFN **Unit 2**	**Main Steam System**	**2-OI-1** **Rev. 0055** **Page 9 of 68**

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 a 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 a 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.

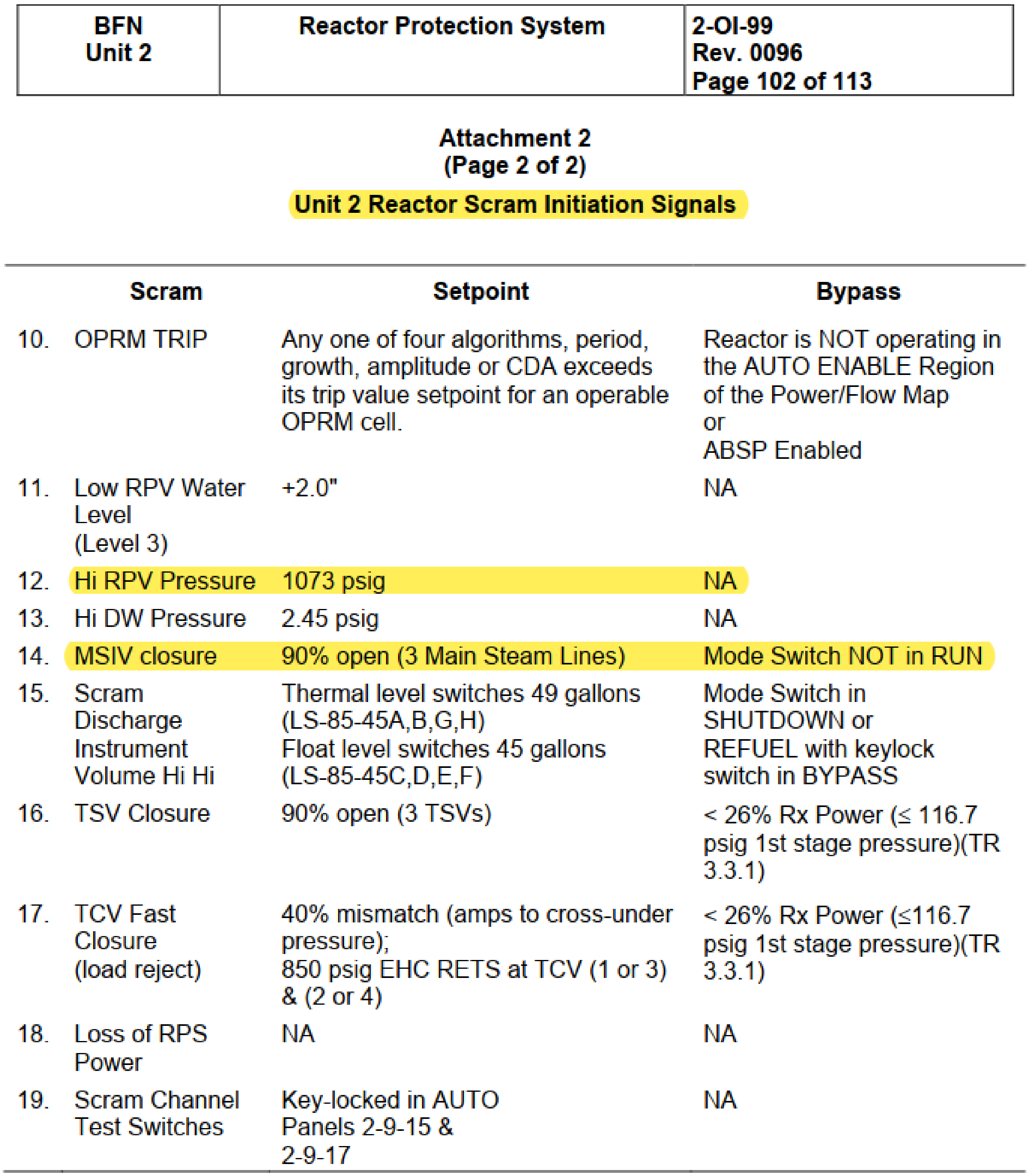
B. Whenever reactor pressure is reduced to 852 psig and the reactor mode switch is in RUN position, the MSIVs will close.

C. The MSIVs will close if 250 Vdc and 120 Vac power to the MSIV control logic is de-energized.

D. Reactor power should be less than or equal to 75% prior to closing an MSIV greater than 15 percent during closure testing. This should prevent a high steam line flow MSIV closure and subsequent reactor scram.

E. All MSIV Handswitches need to be placed in the Close Position to allow the PCIS group one trip logic to be reset, and prevent rapid re-opening upon reset of any one not in close. Placing all INBOARD MSIV Handswitches in the Close Position allows the PCIS group one A1/B1 Division I trip logic to be reset and placing all OUTBOARD MSIV Handswitches in the Close Position allows the PCIS group one A2/B2 Division II trip logic to be reset. (PER 658229-012)

Excerpt from 2-OI-99:



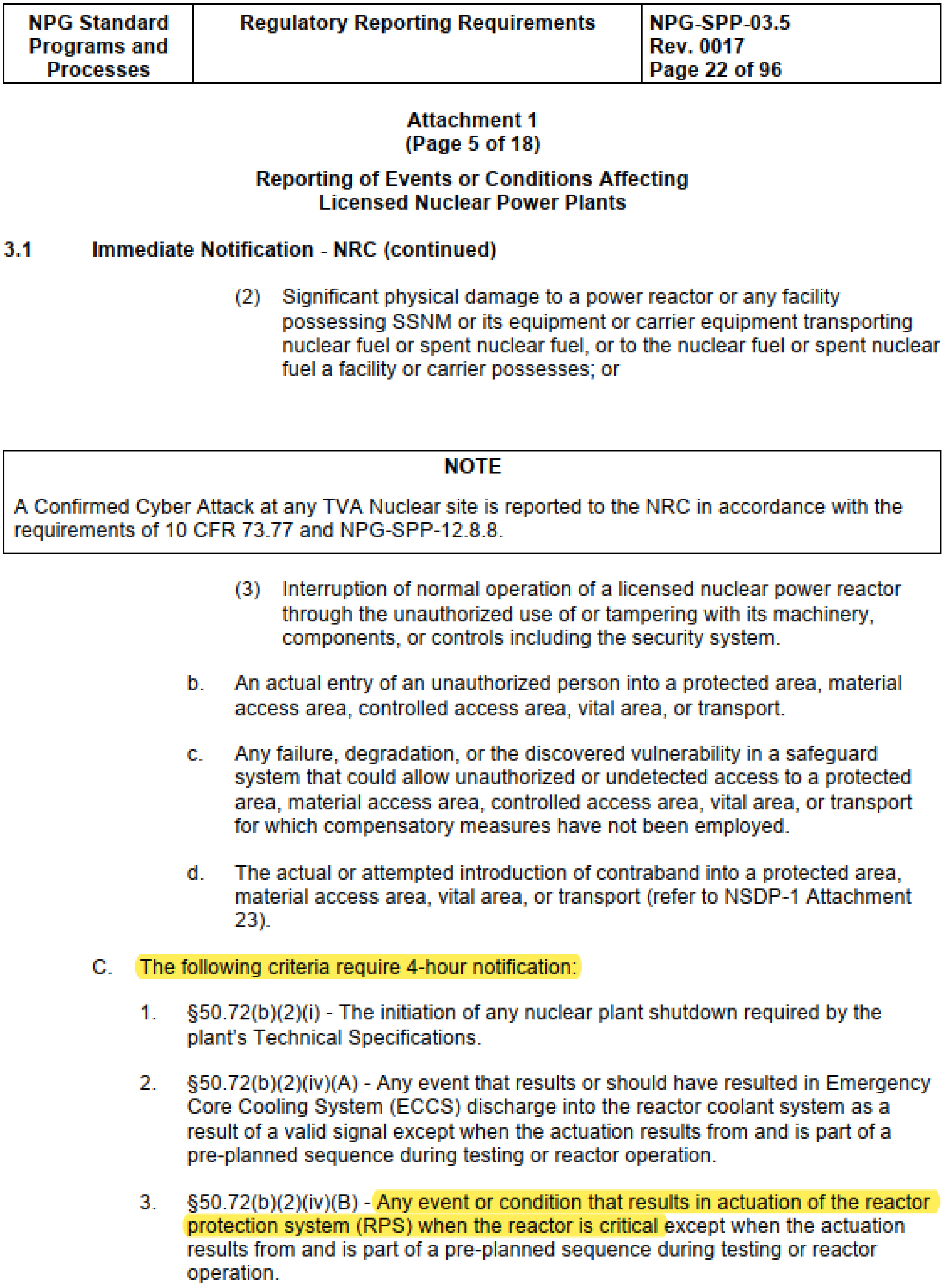
BFN Unit 2	**Reactor Protection System**	**2-OI-99 Rev. 0096 Page 102 of 113**

Attachment 2
(Page 2 of 2)

Unit 2 Reactor Scram Initiation Signals

	Scram	Setpoint	Bypass
10.	OPRM TRIP	Any one of four algorithms, period, growth, amplitude or CDA exceeds its trip value setpoint for an operable OPRM cell.	Reactor is NOT operating in the AUTO ENABLE Region of the Power/Flow Map or ABSP Enabled
11.	Low RPV Water Level (Level 3)	+2.0"	NA
12.	Hi RPV Pressure	1073 psig	NA
13.	Hi DW Pressure	2.45 psig	NA
14.	MSIV closure	90% open (3 Main Steam Lines)	Mode Switch NOT in RUN
15.	Scram Discharge Instrument Volume Hi Hi	Thermal level switches 49 gallons (LS-85-45A,B,G,H) Float level switches 45 gallons (LS-85-45C,D,E,F)	Mode Switch in SHUTDOWN or REFUEL with keylock switch in BYPASS
16.	TSV Closure	90% open (3 TSVs)	< 26% Rx Power (≤ 116.7 psig 1st stage pressure)(TR 3.3.1)
17.	TCV Fast Closure (load reject)	40% mismatch (amps to cross-under pressure); 850 psig EHC RETS at TCV (1 or 3) & (2 or 4)	< 26% Rx Power (≤116.7 psig 1st stage pressure)(TR 3.3.1)
18.	Loss of RPS Power	NA	NA
19.	Scram Channel Test Switches	Key-locked in AUTO Panels 2-9-15 & 2-9-17	NA

Excerpts from NPG-SPP-03.5:



NPG Standard Programs and Processes	**Regulatory Reporting Requirements**	**NPG-SPP-03.5** **Rev. 0017** **Page 22 of 96**

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).

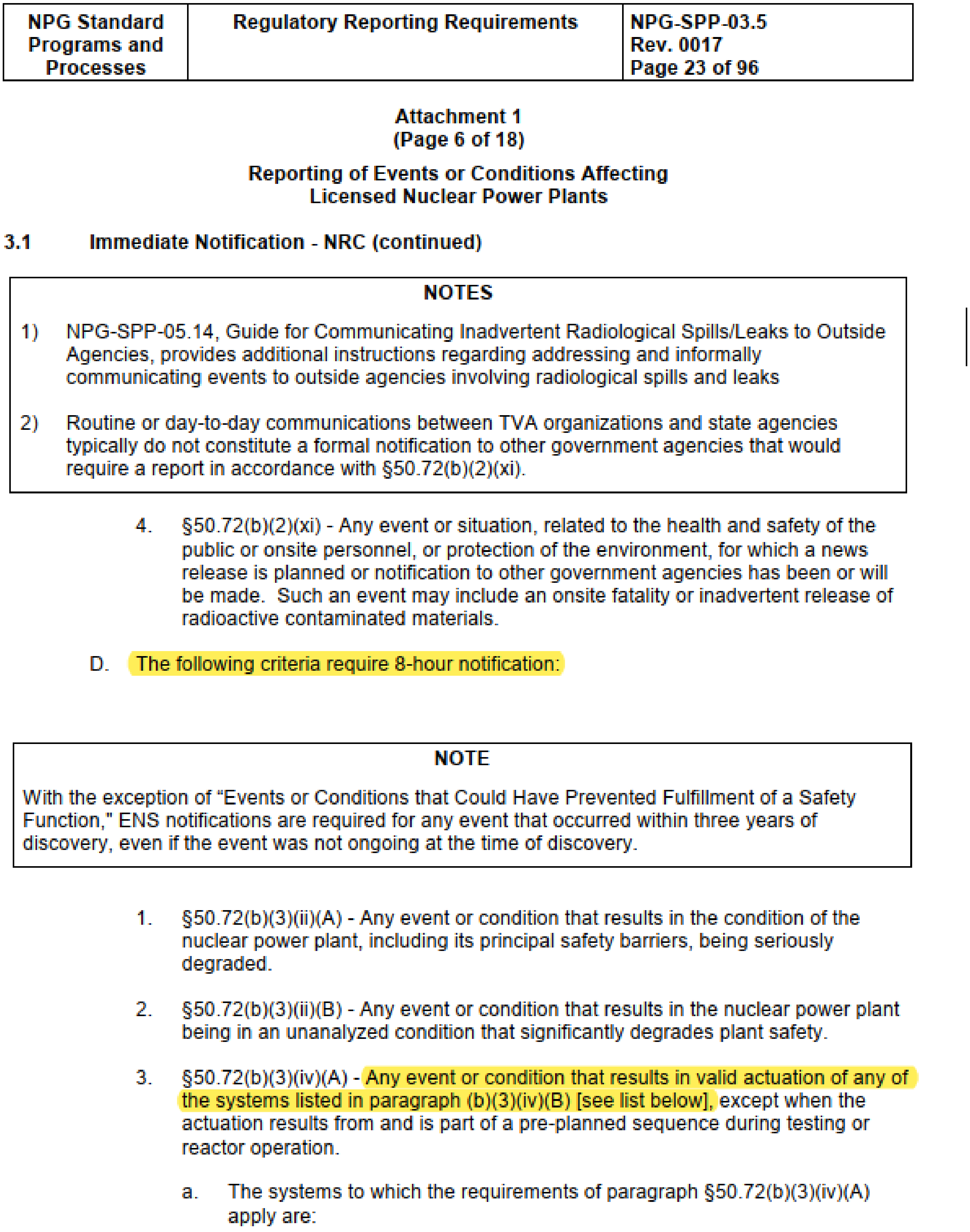
C. The following criteria require 4-hour notification:

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.

Supports Distractors B(2), D(2):



NPG Standard Programs and Processes	**Regulatory Reporting Requirements**	**NPG-SPP-03.5** **Rev. 0017** **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, Guide for Communicating Inadvertent Radiological Spills/Leaks to Outside Agencies, 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

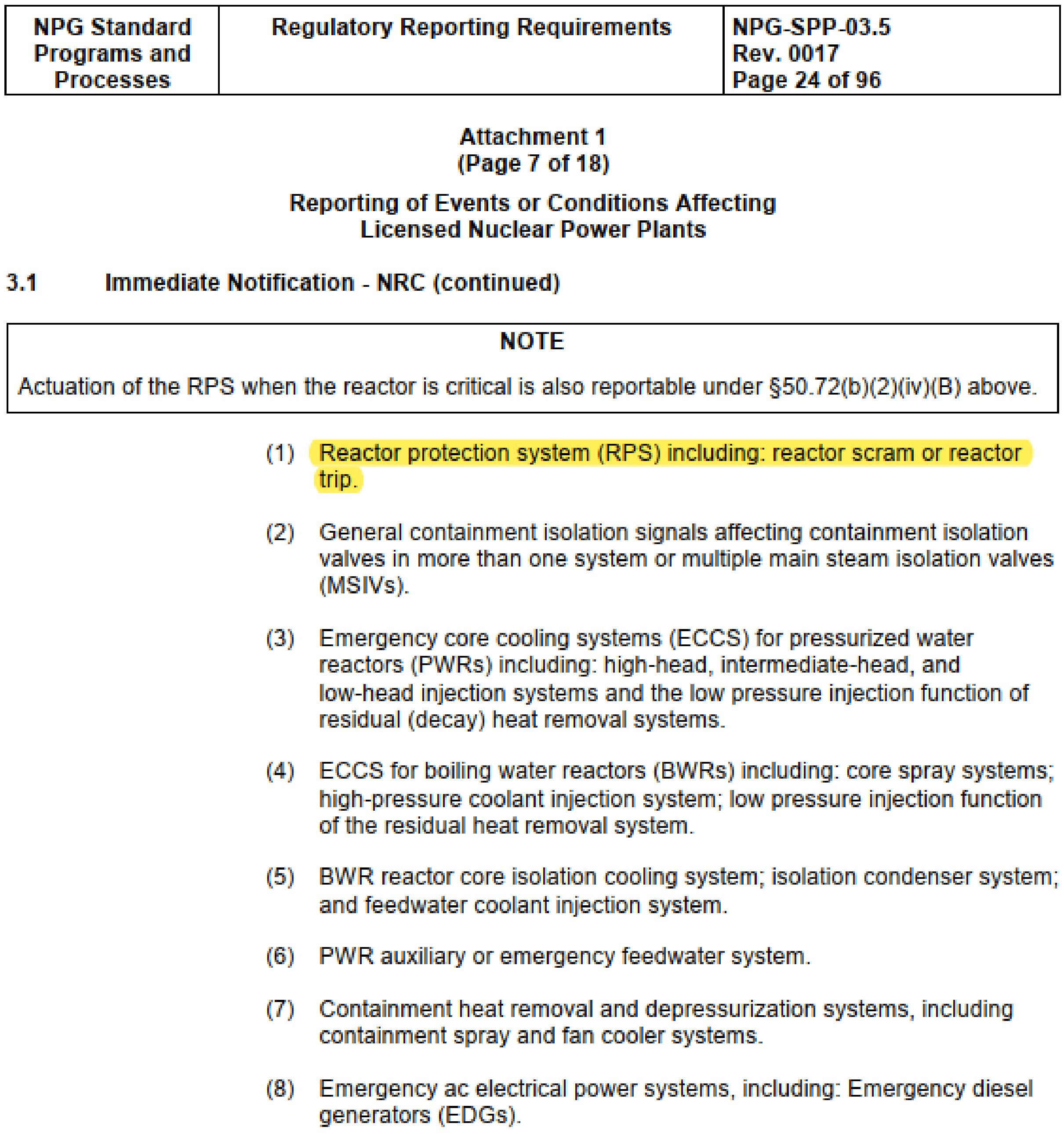
With the exception of "Events or Conditions that Could Have Prevented Fulfillment of a Safety Function," ENS notifications are required for any event that occurred within three years of discovery, even if the event was not ongoing at the time of discovery.

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. 0017** **Page 24 of 96**

Attachment 1
(Page 7 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTE

Actuation of the RPS when the reactor is critical is also reportable under §50.72(b)(2)(iv)(B) above.

(1) Reactor protection system (RPS) including: reactor scram or reactor trip.

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(4) ECCS for boiling water reactors (BWRs) including: core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

(6) PWR auxiliary or emergency feedwater system.

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency ac electrical power systems, including: Emergency diesel generators (EDGs).

Examination Outline Cross-reference:

288000 (SF9, PVS) Plant Ventilation
A2.05 (10CFR 55.43.5 – SRO Only)
Ability to (a) predict the impacts of the following on the Plant Ventilation SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

- Extreme outside weather conditions

Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	288000A2.05	
Importance Rating	------	2.9

Proposed Question: **# 93**

All 3 Units are operating at 100% RTP when the following conditions occur:

At 0845:

- BFN entered 0-AOI-100-7, Severe Weather
- Core Spray Pump 1B and 1D are tagged for scheduled maintenance

An AUO is dispatched to start ___**(1)**___ EDG Exhaust Fan(s) for each EDG room

At 0900:

- The AUO reports that for the ‘A’ EDG, neither fan can be placed in service.

In accordance with Tech Specs 3.8.1, Unit 1 entry in LCO 3.0.3 is required at ___**(2)**___.

[REFERENCE PROVIDED]

A. (1) **BOTH**
(2) 0900

B. (1) **BOTH**
(2) 1300

C. (1) **ONLY** one
(2) 0900

D. (1) **ONLY** one
(2) 1300

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect, but plausible if the candidate recalls that 0-AOI-100-7, Severe Weather, does contain procedure sections sequentially to start both EDG Exhaust Fans. However, procedurally, 'A' fan is started first, then 'B' fan is only started if 'A' fan fails to start. Second part is incorrect but plausible due to testing the application of redundant equipment required by Tech Specs in regards to EDG supplies. It is plausible for the candidate to conclude that since the Safety Function is not maintained, Core Spray Pumps REQUIRED ACTIONS must be Immediately entered when the INOPERABILITY of 'A' EDG occurs at 0900.

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)* For the first part 0-AOI-100-7, Severe Weather states that only one EDG Exhaust Fan for each EDG room is to be started and the other fan will only be started if the previous fan fails to start. For second part, given that Loop II (1B and 1D Pumps) are tagged and INOPERABLE, Loop I (1A and 1C Pumps) is the redundant required equipment to Loop II. Once neither Exhaust Fan for 'A' EDG fails to start, 'A' EDG in INOPERABLE thereby making Loop I Core Spray INOPERABLE. In accordance with Tech Spec 3.5.1 Condition H, with two or more low pressure ECCS injection/spray subsystems INOPERBLE, LCO 3.0.3 is required to be entered Immediately. However, in accordance with Tech Spec 3.8.1 Condition B, once one required Unit 1 and 2 EDG is INOPERABLE, the REQUIRED ACTION B.3 states to declare required feature(s), supported by the INOPERABLE Unit 1 and 2 EDG ('A' in this case) when the redundant required feature(s) are INOPERABLE. Therefore, the COMPLETION TIME is 4 hours from discovery (0900) of Condition B concurrent with INOPERABILITY of redundant required feature(s). Once 4 hours expires in accordance with Tech Spec 3.8.1, the since Safety Function is lost by Tech Spec 3.5.1, all Core Spray Pumps are INOPERABLE and LCO 3.0.3 entry is required at 1300.

SRO Level Justification: Tests the candidate's ability to analyze plant conditions during extreme weather as it relates to Plant Ventilation and determine appropriate actions in accordance with plant procedures and Technical Specifications. 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.

Technical Reference(s): (Attach if not previously provided)

U1 Tech Spec 3.5.1, Amend. 240

U1 Tech Spec 3.8.1, Amend. 249

0-AOI-100-7, Rev. 46

OPL171.045, Rev. 22

1-OI-75, Rev. 42

Proposed references to be provided to applicants during examination: **U1 Tech Spec 3.5.1 and 3.8.1 (No Bases)**

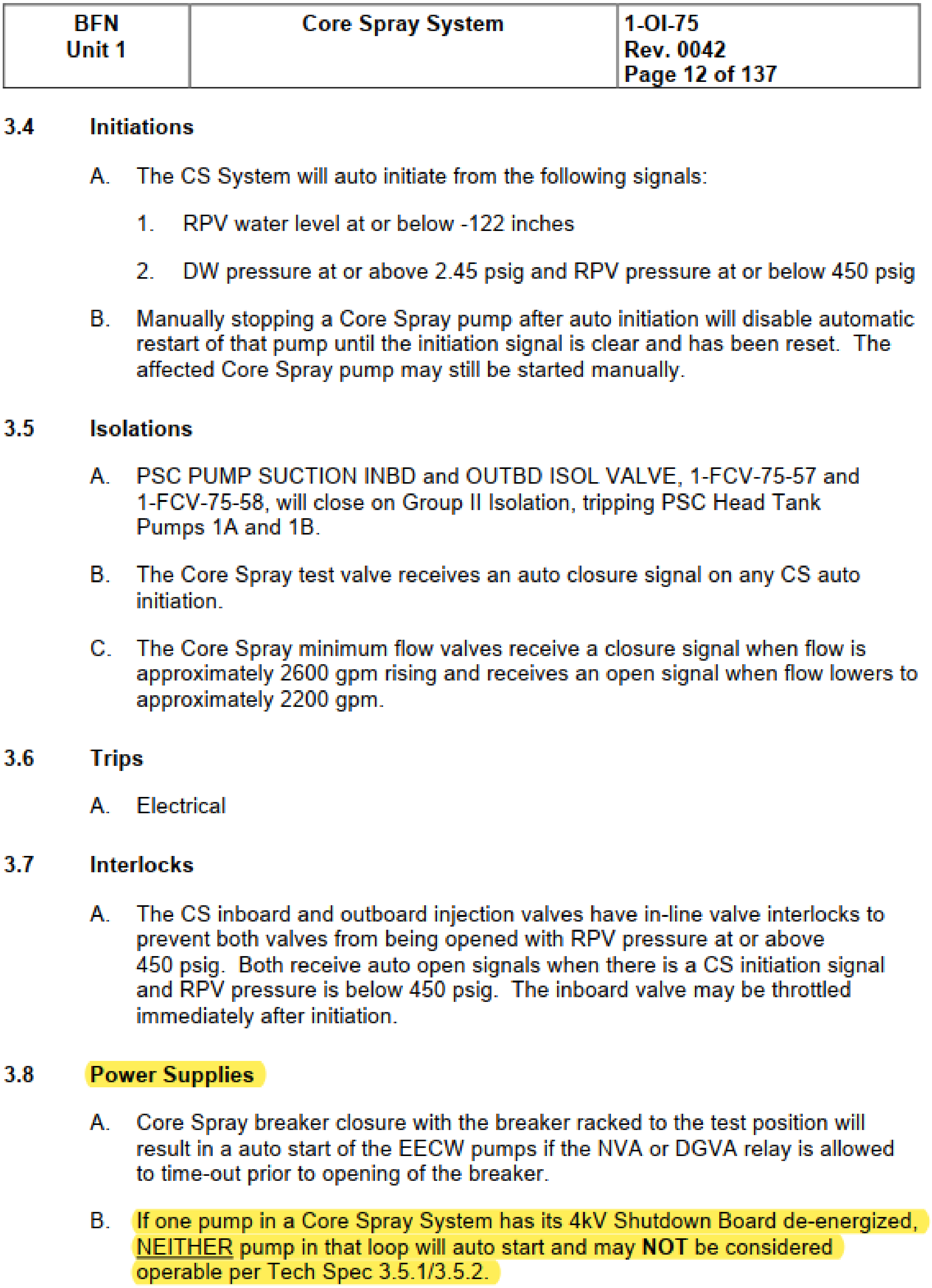
Form 4.2-1 **Written Examination Question Worksheet**

Learning Objective: OPL171.074 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

Excerpt from 1-OI-75:



BFN **Unit 1**	**Core Spray System**	**1-OI-75** **Rev. 0042** **Page 12 of 137**

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 Lesson Plan:

OPL171.045, Core Spray System, Rev.22

d) Each loop delivers at least 6,250 gpm against a system head corresponding to a 105 psi differential pressure between the reactor vessel and the primary containment.

e) Power supplies for all Core Spray System pump motors are shown on the following chart for Units 1, 2, and 3:

Obj. ILT 3.a
Obj. LOR 2.a
Obj. NLO.12.a
Obj. NLOR 10.a
Obj. ILT 3.d
Obj. LOR 2.d

PUMP	1A	2A	3A	1B	2B	3B	1C	2C	3C	1D	2D	3D
LOOP (SYS)	I	I	I	II	II	II	I	I	I	II	II	II
ELECT. DIV.	I	I	I	II	II	II	I	I	I	II	II	II
S/D BD.	A	A	3EA	C	C	3EC	B	B	3EB	D	D	3ED
D/G	A	A	3A	C	C	3C	B	B	3B	D	D	3D

f) Pumps are located in the basement of the Reactor Building (NE, NW Quads-Elev. 519). The head of water from torus provides NPSH requirements.

Obj. ILT 2.f
Obj. LOR 1.e

3. Spargers

a. Two 360° spray spargers located within core shroud.

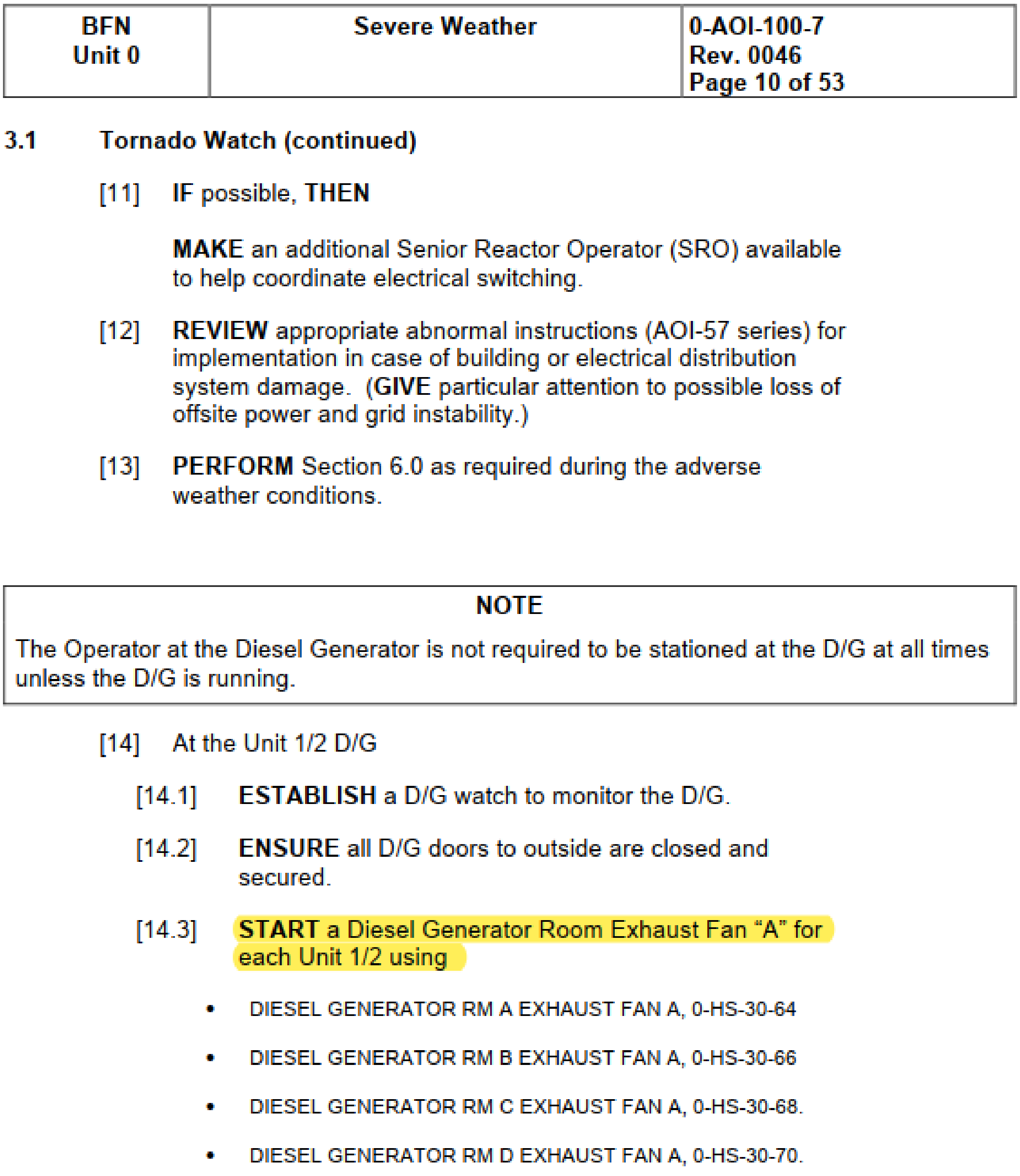
Obj. ILT 1
Obj. NLOR 1
Obj. NLO 3
Obj. NLO 2

b. Each sparger is split into two 180° segments.

c. Spargers are separate, each receiving flow from one of the two loops provided.

d. Nozzles direct the Core Spray water toward the vertical centerline of the fuel.

Excerpts from 0-AOI-100-7:



BFN **Unit 0**	**Severe Weather**	**0-AOI-100-7** **Rev. 0046** **Page 10 of 53**

3.1 Tornado Watch (continued)

[11] **IF** possible, **THEN**

MAKE an additional Senior Reactor Operator (SRO) available to help coordinate electrical switching.

[12] **REVIEW** appropriate abnormal instructions (AOI-57 series) for implementation in case of building or electrical distribution system damage. (**GIVE** particular attention to possible loss of offsite power and grid instability.)

[13] **PERFORM** Section 6.0 as required during the adverse weather conditions.

NOTE

The Operator at the Diesel Generator is not required to be stationed at the D/G at all times unless the D/G is running.

[14] At the Unit 1/2 D/G

[14.1] **ESTABLISH** a D/G watch to monitor the D/G.

[14.2] **ENSURE** all D/G doors to outside are closed and secured.

[14.3] **START** a Diesel Generator Room Exhaust Fan “A” for each Unit 1/2 using

- DIESEL GENERATOR RM A EXHAUST FAN A, 0-HS-30-64
- DIESEL GENERATOR RM B EXHAUST FAN A, 0-HS-30-66
- DIESEL GENERATOR RM C EXHAUST FAN A, 0-HS-30-68.
- DIESEL GENERATOR RM D EXHAUST FAN A, 0-HS-30-70.

BFN Unit 0	**Severe Weather**	**0-AOI-100-7 Rev. 0046 Page 11 of 53**

3.1 Tornado Watch (continued)

[14.4] **IF** the Diesel Generator Room Exhaust Fan "A" can not be started for a Diesel Generator Room, **THEN**

START the Diesel Generator Room Exhaust Fan "B" for the associate Diesel Generator Room using: (Otherwise N/A)

- DIESEL GENERATOR RM AEXHAUST FAN B, 0-HS-30-65
- DIESEL GENERATOR RM B EXHAUST FAN B, 0-HS-30-67
- DIESEL GENERATOR RM C EXHAUST FAN B, 0-HS-30-69.
- DIESEL GENERATOR RM D EXHAUST FAN B, 0-HS-30-71.

NOTE

The Operator at the Diesel Generator is not required to be stationed at the D/G at all times unless the D/G is running.

[15] At Unit 3 D/G

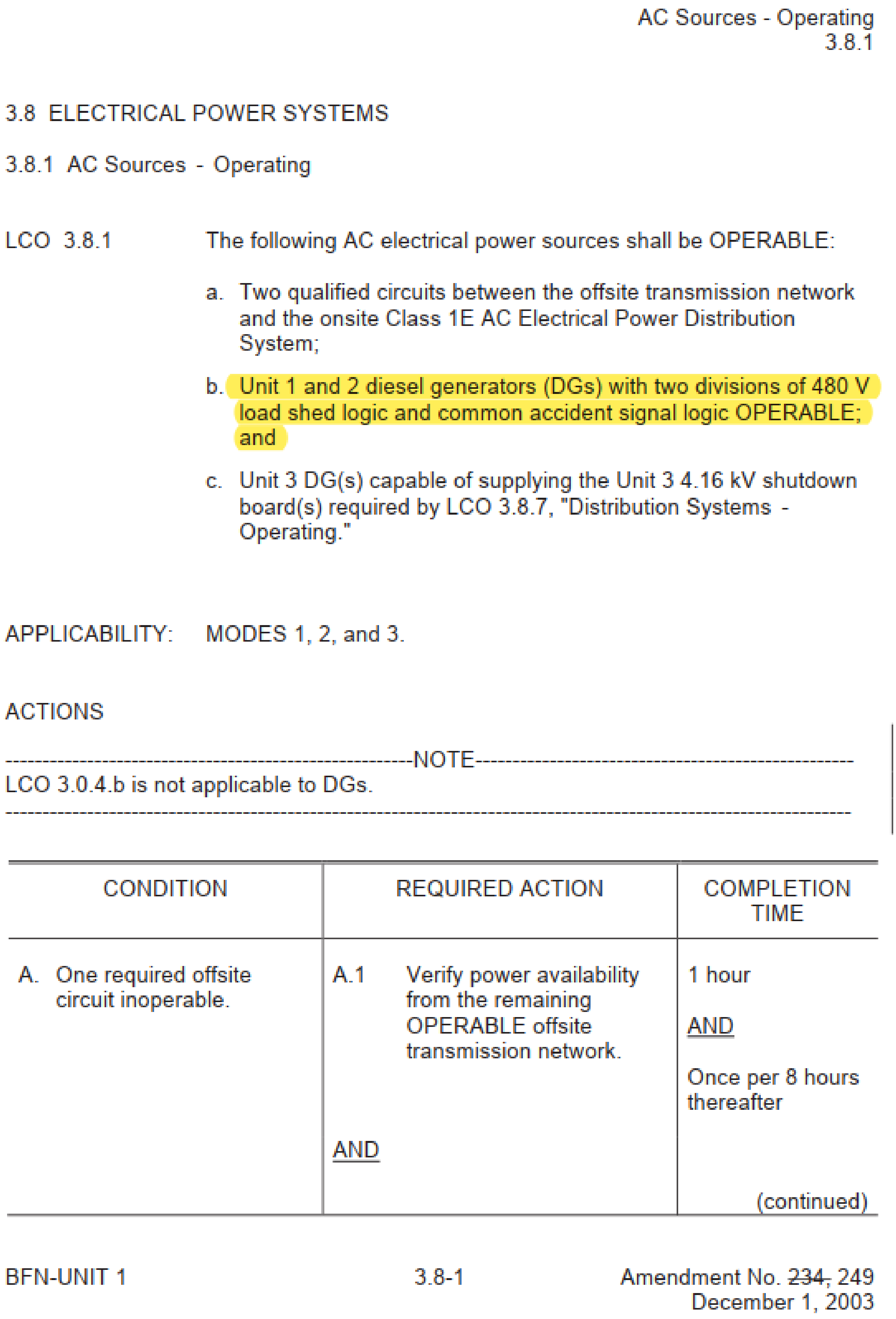
[15.1] **ESTABLISH** a D/G watch to monitor the D/G.

[15.2] **ENSURE** all D/G doors to outside are closed and secured.

[15.3] **START** a Diesel Generator Room Exhaust Fan "A" for each Unit 3 Diesel Generator Room using:

- DIESEL GENERATOR RM 3A EXHAUST FAN A, 3-HS-030-0230
- DIESEL GENERATOR RM 3B EXHAUST FAN A, 3-HS-030-0232
- DIESEL GENERATOR RM 3C EXHAUST FAN A, 3-HS-030-0234.
- DIESEL GENERATOR RM 3D EXHAUST FAN A, 3-HS-030-0236.

Excerpts from U1 Tech Spec 3.8.1:



AC Sources - Operating
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.

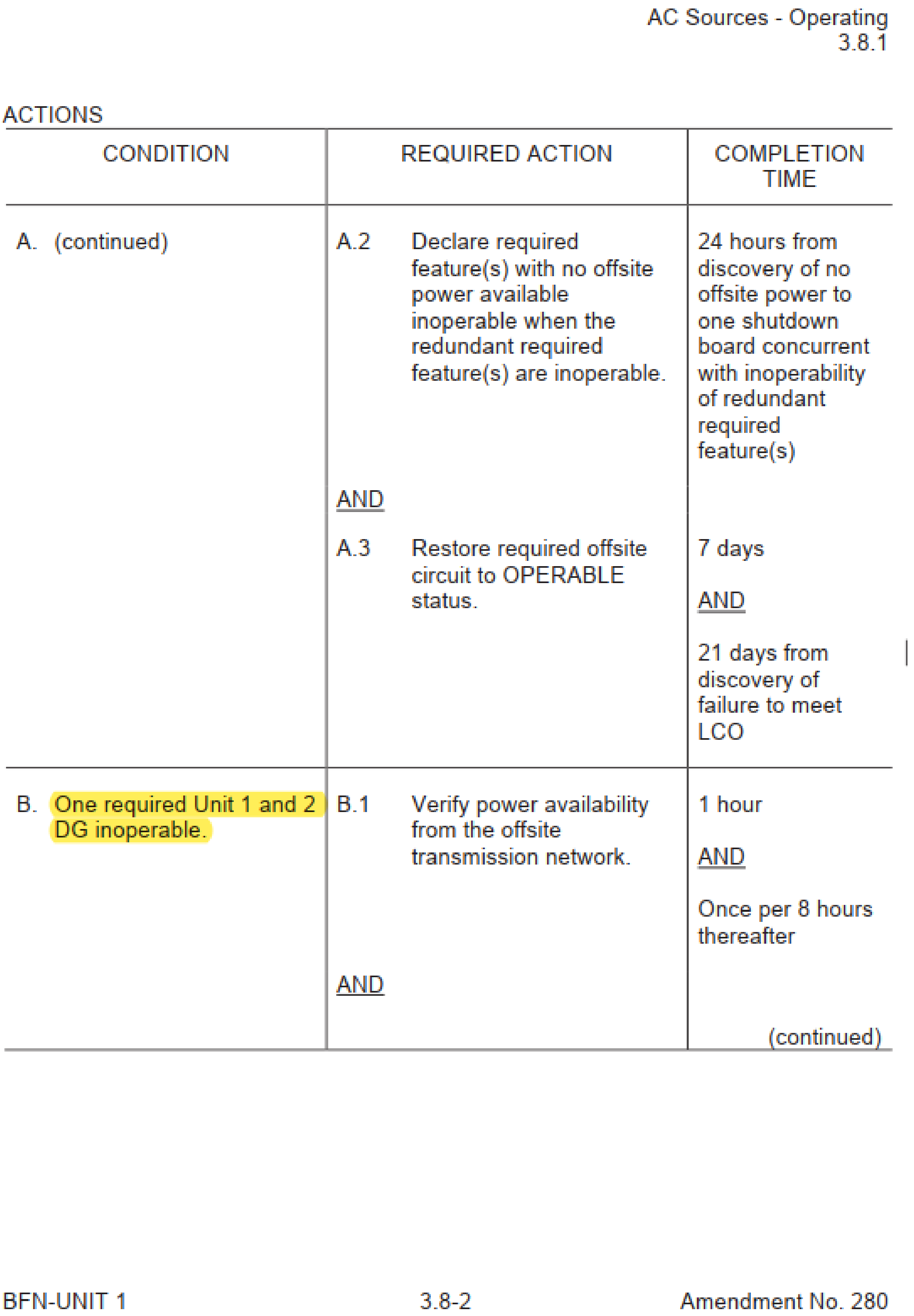
ACTIONS

--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.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	
		(continued)

BFN-UNIT 1 3.8-1 Amendment No. ~~234,~~ 249
December 1, 2003

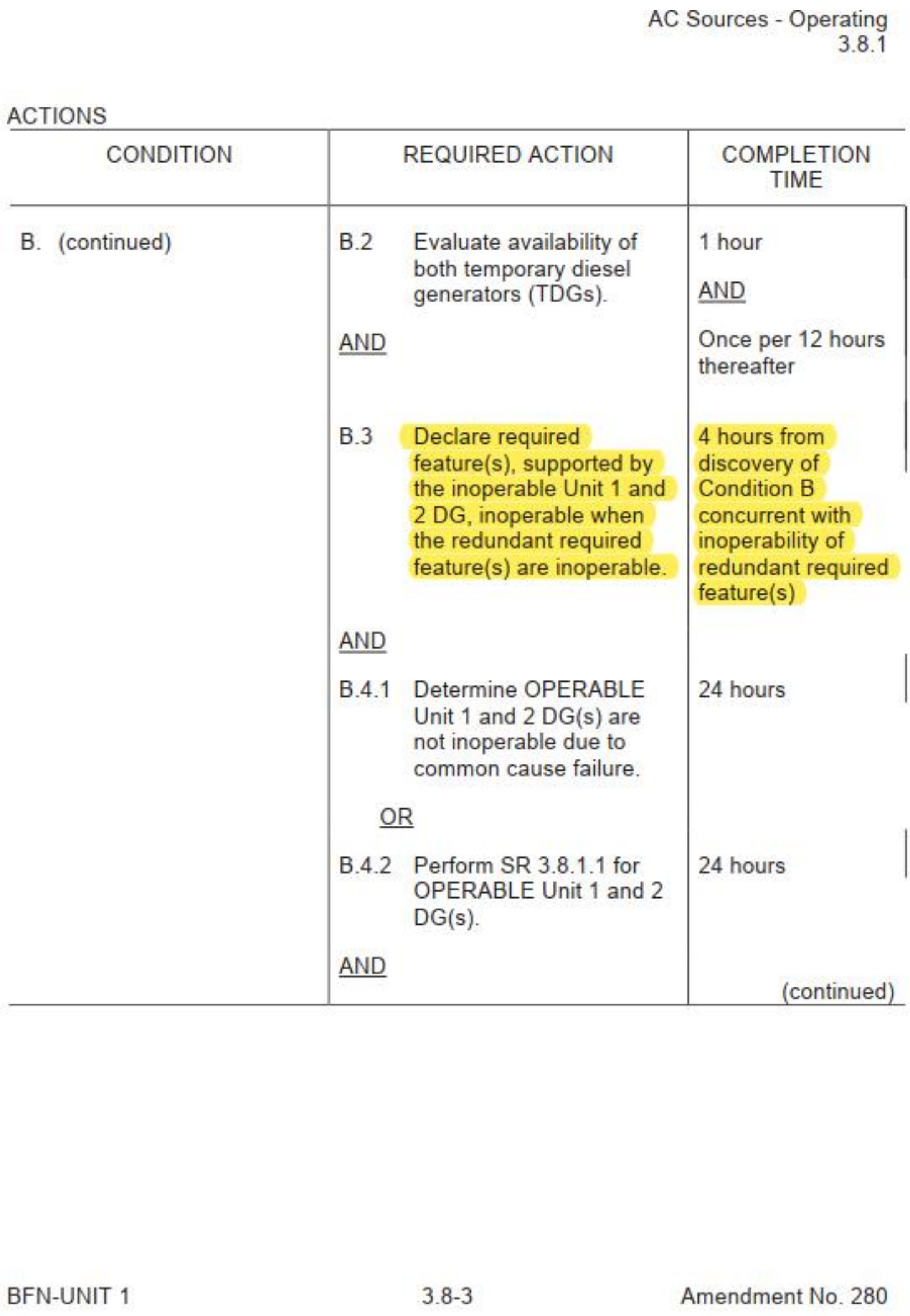


AC Sources - Operating
3.8.1

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)
	<u>AND</u>		
	A.3	Restore required offsite circuit to OPERABLE status.	7 days <u>AND</u> 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 <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
			(continued)

BFN-UNIT 1 3.8-2 Amendment No. 280



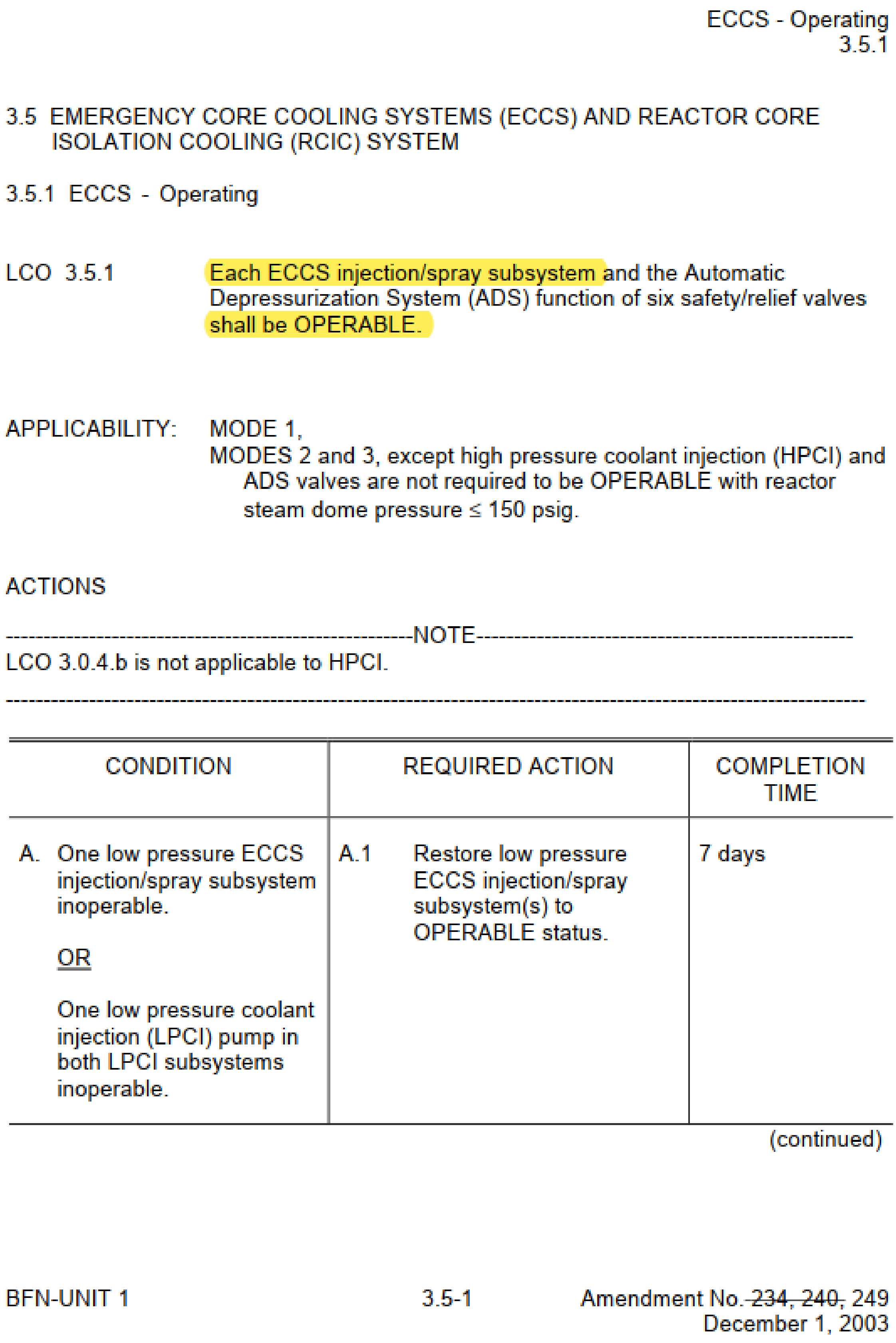
AC Sources - Operating
3.8.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Evaluate availability of both temporary diesel generators (TDGs).	1 hour AND Once per 12 hours thereafter
	AND	
	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)
	AND	
	B.4.1 Determine OPERABLE Unit 1 and 2 DG(s) are not inoperable due to common cause failure.	24 hours
	OR	
	B.4.2 Perform SR 3.8.1.1 for OPERABLE Unit 1 and 2 DG(s).	24 hours
	AND	(continued)

BFN-UNIT 1 3.8-3 Amendment No. 280

Excerpts from U1 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.

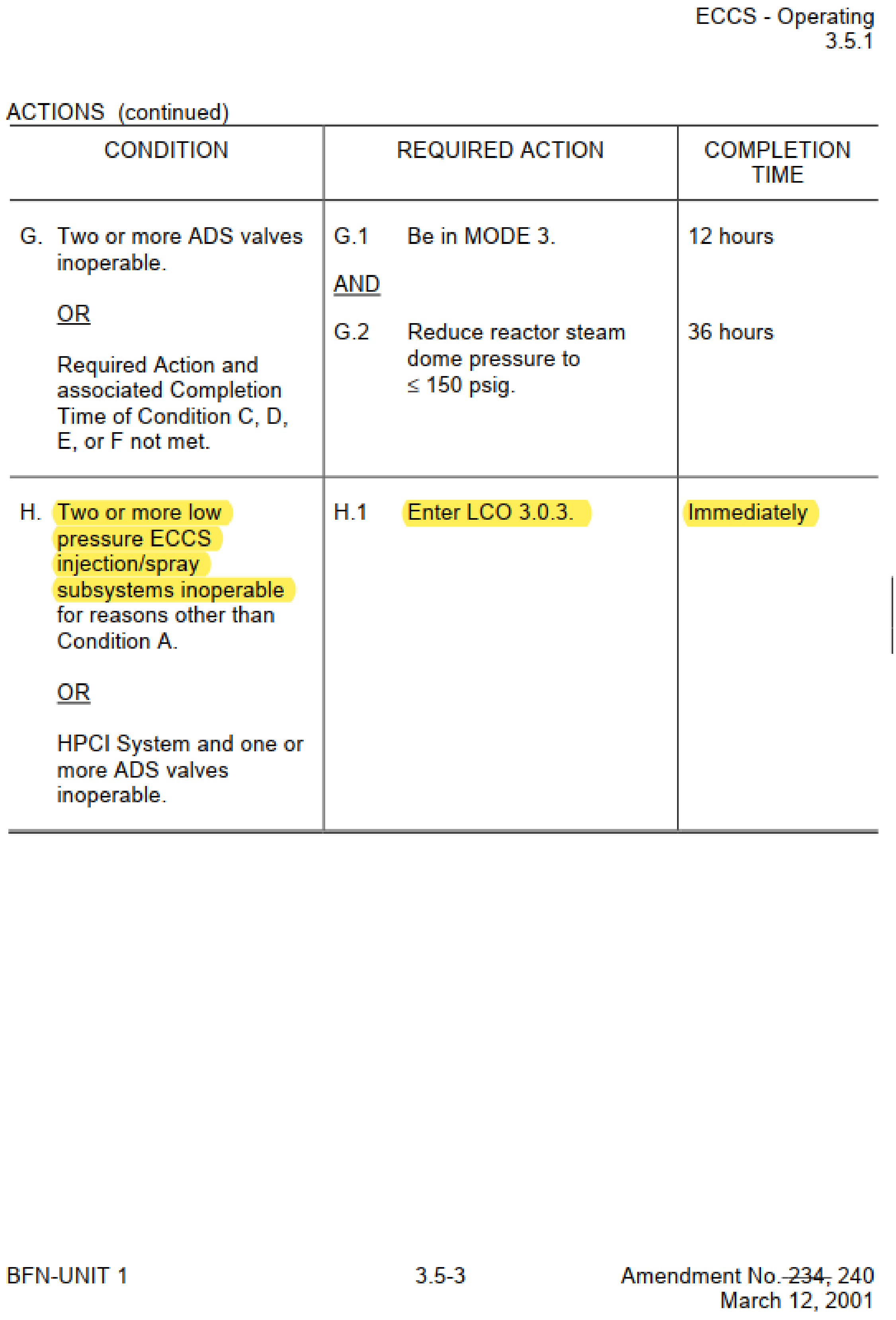
ACTIONS

-------------------------------------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

(continued)



ECCS - Operating
3.5.1

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. AND G.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 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

BFN-UNIT 1 3.5-3 Amendment No. ~~234,~~ 240
March 12, 2001

Examination Outline Cross-reference:

223002 (SF5, PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff

G2.4.31 (10CFR 55.43.5 - SRO Only)

Knowledge of annunciation alarms, indications, or response procedures.

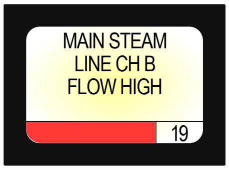
Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	223002G2.4.31	
Importance Rating	------	4.1

Proposed Question: **# 94**

Unit 3 is in MODE 1 when the following conditions occur:

At 1000 on 5/01/22:

- MAIN STEAM LINE CH B FLOW HIGH (3-9-5B, Window 19) alarms due to a failed pressure transmitter



The setpoint for the above alarm is ___**(1)**___ of rated steam flow.

In accordance with Tech Spec 3.3.6.1, the REQUIRED ACTION is to enter MODE 3 at 2200 on ___**(2)**___.

[REFERENCE PROVIDED]

A. (1) 135%
(2) 5/01/22

B. (1) 135%
(2) 5/02/22

C. (1) 200%
(2) 5/01/22

D. (1) 200%
(2) 5/02/22

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** First part is correct *(See B).* Second part is incorrect but plausible in that Tech Spec 3.3.6.1 Condition A has 2 different REQUIRED ACTION times, 12 hours and 24 hours. It states 12 hours for Functions 2.a, 2.b, 5.h, 6.b, and 6.c. Additionally, if the candidate misapplies the concept that Condition A must expire prior to entering the Condition stated in the table, then this concept will again be confused as it often is for each specific function for each PCIS Group having different REQUIRED ACTION times.

B **CORRECT:** *(See attached)* In accordance with the given alarm, MAIN STEAM LINE CH B FLOW HIGH (3-9-5B, Window 19), the setpoint is 135% of rated Main Steam line flow. In accordance with 3-OI-1, Main Steam System, this also results in a PCIS Group 1 Isolation of MSIVs due to high Main Steam line flow. For second part, Tech Spec 3.3.6.1, Condition A has 2 different REQUIRED ACTION times, 12 hours and 24 hours. It states 12 hours for Functions 2.a, 2.b, 5.h, 6.b, and 6.c, Using Table 3.3.6.1-1, Primary Containment Isolation Instrumentation, the Main Steam High Flow Function is listed as 1.c so it's REQUIRED ACTION time must be 24 hours. Condition C states to enter the Condition referenced in the table immediately. From the table, Function 1.c, Condition D is entered which requires MODE 3 entry within 12 hours. This results in 24 hours plus 12 hours equals 36 hours. Therefore, 36 hours from 1000 on 5/01/22 will be 5/02/22 at 2200.

C INCORRECT: First part is incorrect but plausible since 200% is the PCIS Group 5 Isolation setpoint for HPCI steam flow. Additionally, HPCI STEAM LINE FLOW EXCESSIVE (3-9-9F, Window 18) alarm setpoint is 200% of rated flow. 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 identify PCIS isolation setpoints and the appropriate Tech Spec Required Actions if an instrument failed given indications and alarms. 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): 3-OI-1, Rev. 48 (Attach if not previously provided)

3-ARP-9-3F, Rev. 37

3-ARP-9-5B. Rev. 33

U3 Tech Spec 3.3.6.1, Amend. 213

Proposed references to be provided to applicants during examination: **MAIN STEAM LINE CH B FLOW HIGH (3-9-5B, Window 19), Unit 3 Tech Spec 3.3.6.1 (No Bases)**

Learning Objective: OPL 171.017 Obj. 4 (As available)

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

Question Source:	Bank #		
	Modified Bank #	OPL171.017-02 009 #605	(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:

605. OPL171.017-02 009

A plant startup is in progress on Unit 3 and the following conditions exist:

- The Reactor Mode Switch is in STARTUP.
- Two Turbine Bypass Valves are open.
- Reactor pressure is 940 psig and steady.

Subsequently,

- MAIN STEAM LINE CH B FLOW HIGH (Panel 3-9-5B, Window 19) is received due to differential pressure transmitter 3-PDIS-001-0050B failing high.

Which ONE of the following completes the statements below?

The PCIS group one isolation setpoint for a Main Steam Line (MSL) high flow actuation is __(1)__ steam flow.

Based on the 3-PDIS-001-0050B failing high, the current PCIS group one isolation status at Panel 3-9-4 is __(2)__.

A. (1) 110%
(2) Picture 1

B✓ (1) 135%
(2) Picture 1

C. (1) 110%
(2) Picture 2

D. (1) 135%
(2) Picture 2

Excerpt from 3-ARP-9-5B:

BFN **Unit 3**	**Panel 9-5** **3-XA-55-5B**	**3-ARP-9-5B** **Rev. 0033** **Page 23 of 44**

MAIN STEAM
LINE CH B
FLOW HIGH

19

(Page 1 of 1)

Sensor/Trip Point:

Relays	16A-K3B	3-PdIS-1-13B, 25B 3-PdIS-1-36B, 50B	164.7 psid (Flow of 135%) 5.55 Mlbm/hr
	16A-K3D	3-PdIS-1-13D, 25D 3-PdIS-1-36D, 50D	

Sensor Location:	Relays, Panel 3-9-17, Aux Instrument Room. B switches - Panel 3-9-84, D switches - Panel 3-9-86
Probable Cause:	A. Indicates possible break outside containment. B. SI (or SR) in progress. C. Sensor malfunction.
Automatic Action:	PCIS Group 1 half-isolation.

Operator Action:

A. **CONFIRM** alarm by checking main steam flow indicators.

B. **IF** alarm is valid on any steam line, **THEN** **MANUALLY SCRAM** Reactor and **PLACE** Rx Mode Sw. in Shutdown and **CLOSE** MSIVs. **REFER TO** 3-AOI-100-1.

C. **IF** any flow indicators are low, **THEN** **CHECK OPEN** all MSIVs.

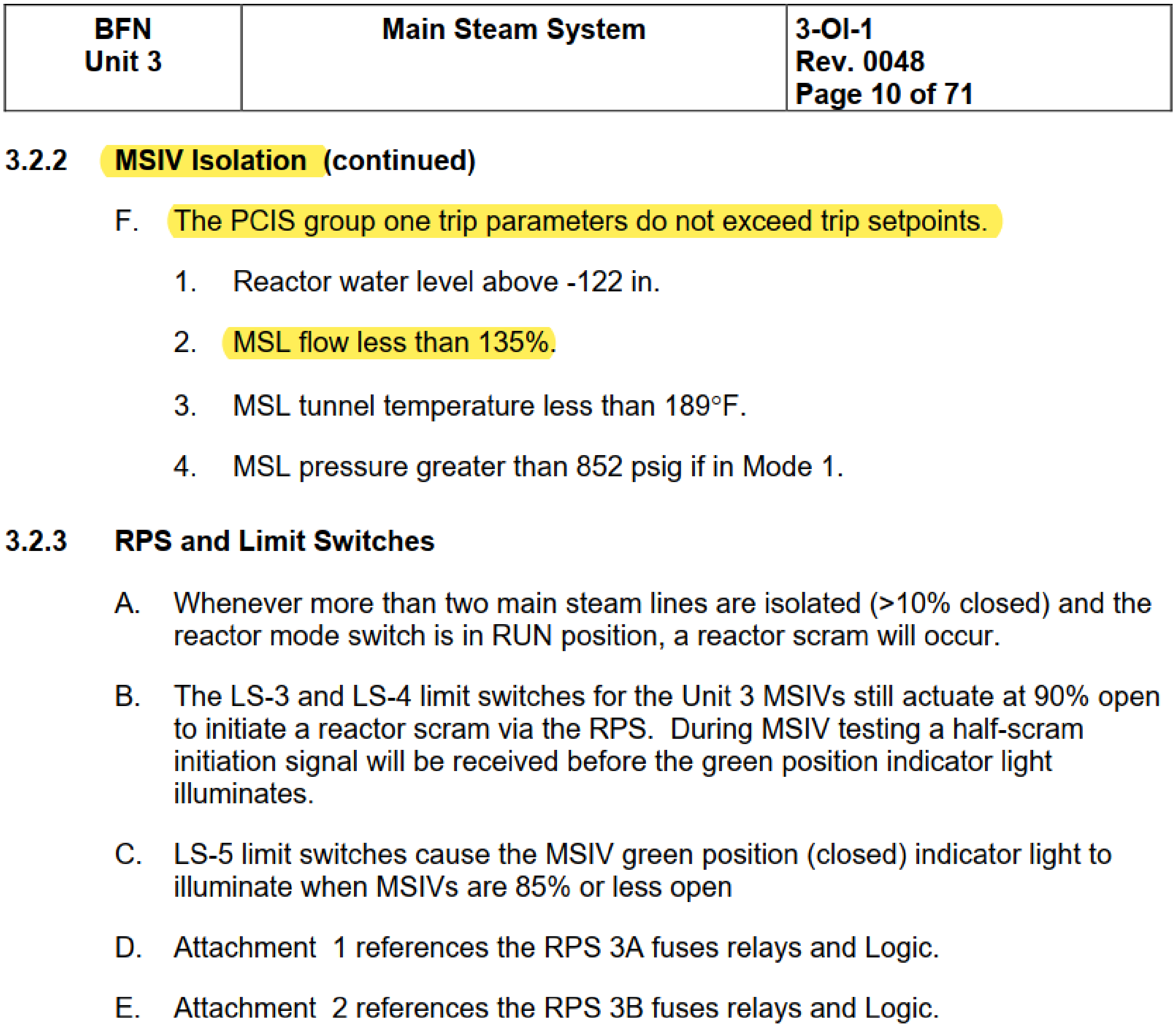
D. **REFER TO** 3-AOI-1-3.

E. **REFER TO** Tech Spec Table 3.3.6.1-1.

References:

3-45E620-6	3-47E610-1-1	3-730E927-7
3-AOI-100-1	3-AOI-1-3	Technical Specifications

Excerpt from 3-OI-1:



BFN **Unit 3**	**Main Steam System**	**3-OI-1** **Rev. 0048** **Page 10 of 71**

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.

3.2.3 RPS and Limit Switches

A. Whenever more than two main steam lines are isolated (>10% closed) and the reactor mode switch is in RUN position, a reactor scram will occur.

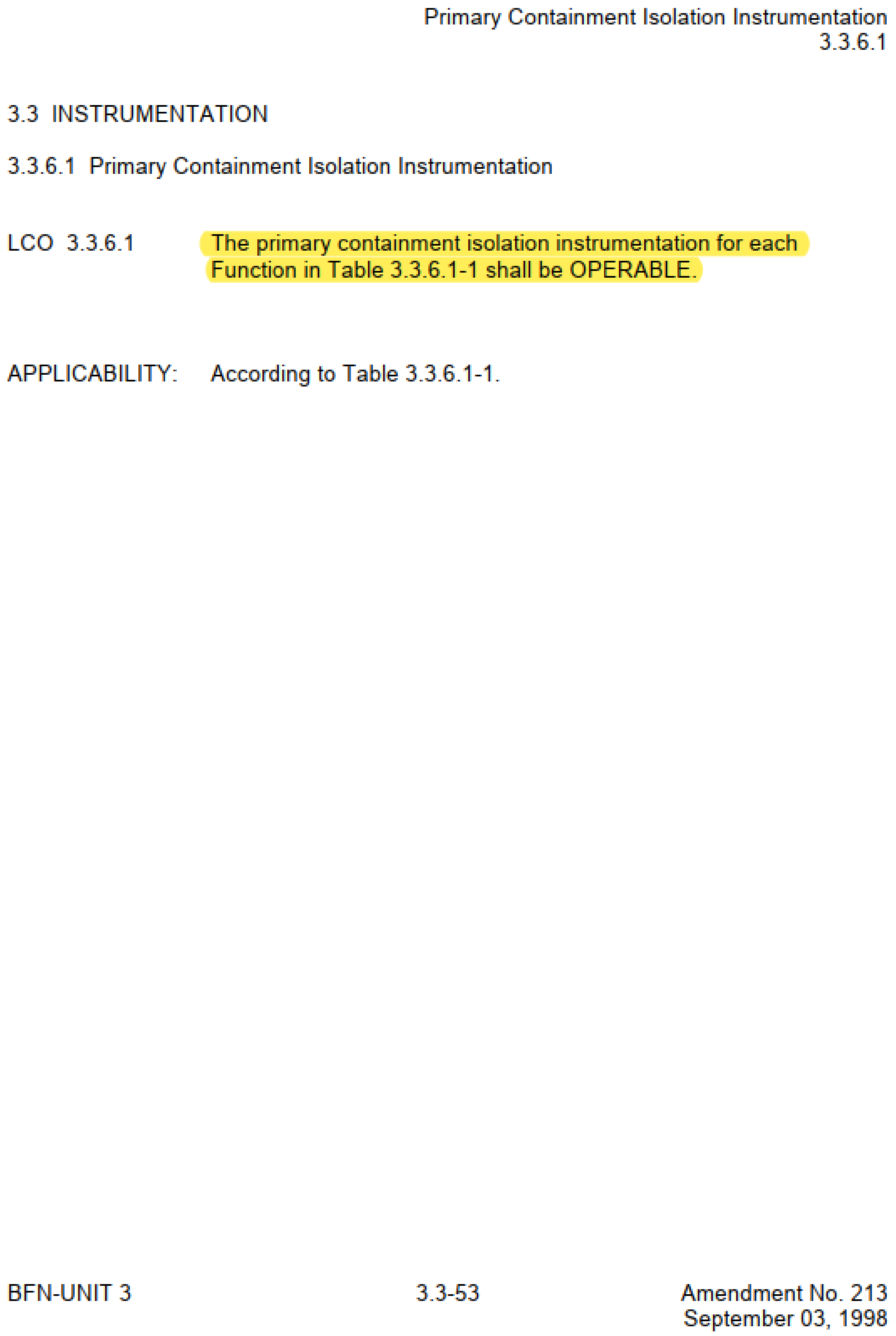
B. The LS-3 and LS-4 limit switches for the Unit 3 MSIVs still actuate at 90% open to initiate a reactor scram via the RPS. During MSIV testing a half-scram initiation signal will be received before the green position indicator light illuminates.

C. LS-5 limit switches cause the MSIV green position (closed) indicator light to illuminate when MSIVs are 85% or less open

D. Attachment 1 references the RPS 3A fuses relays and Logic.

E. Attachment 2 references the RPS 3B fuses relays and Logic.

Excerpts from Tech Spec 3.3.6.1:



Primary Containment Isolation Instrumentation
3.3.6.1

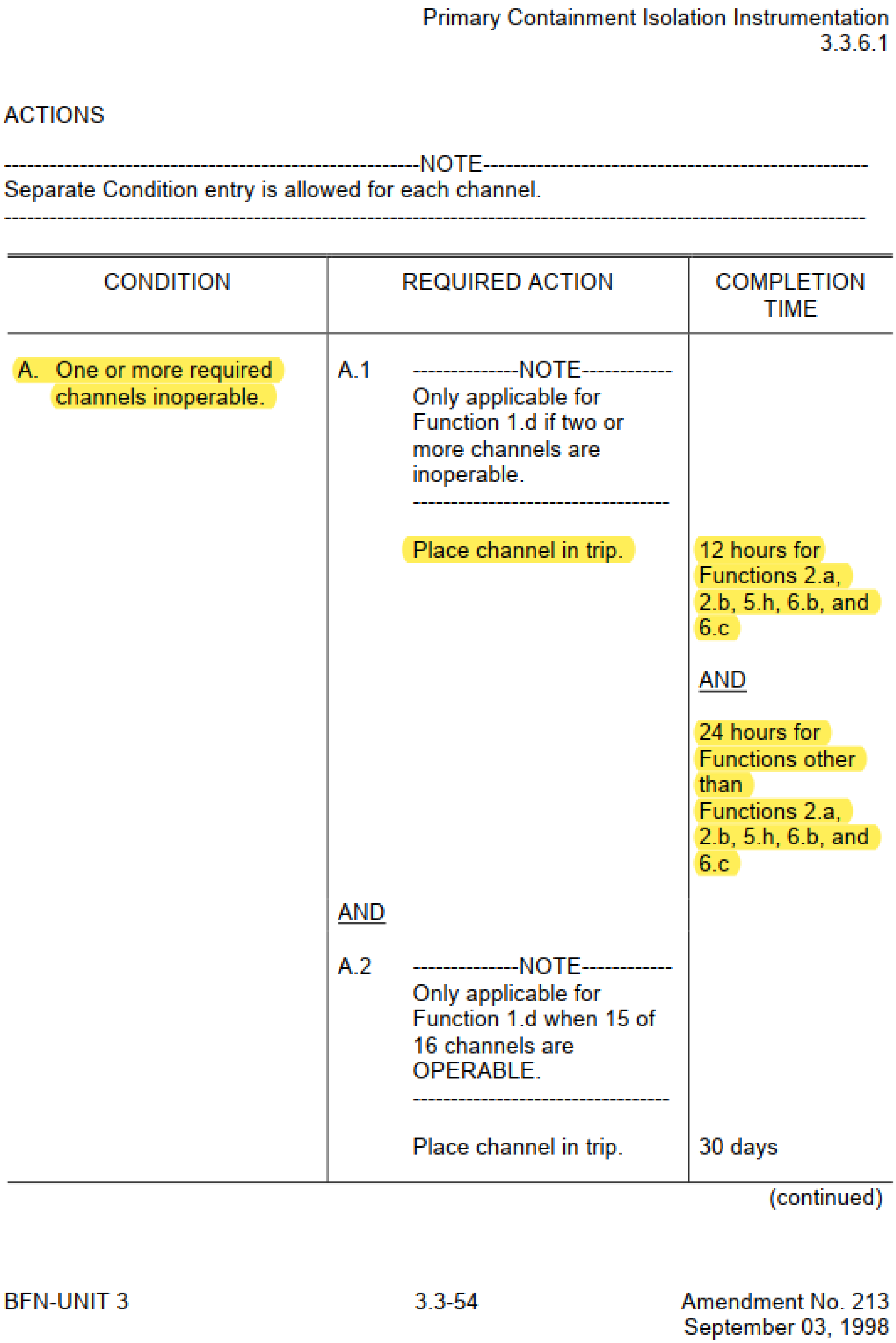
3.3 INSTRUMENTATION

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.

BFN-UNIT 3 3.3-53 Amendment No. 213
September 03, 1998



ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

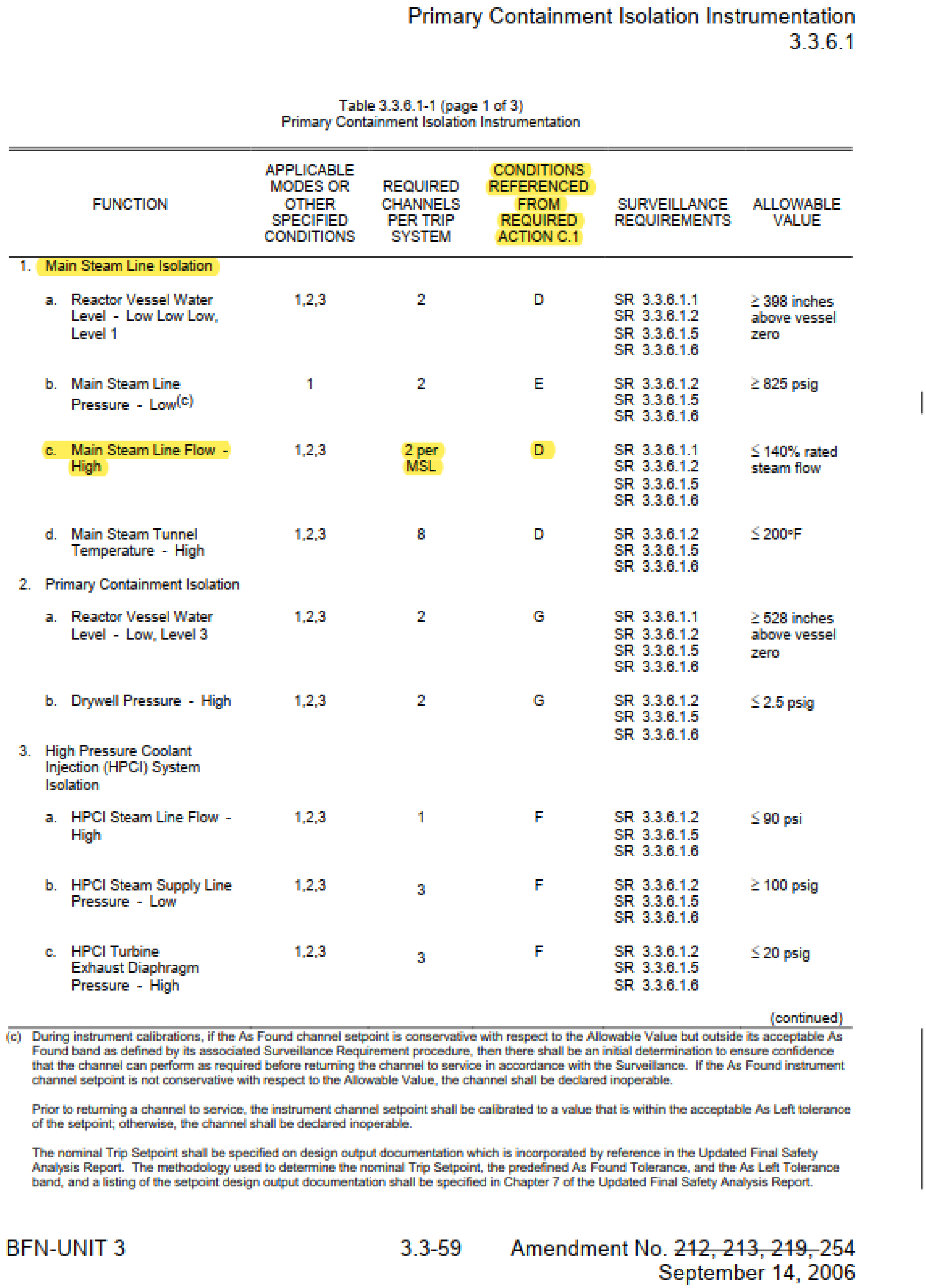
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 ---------------NOTE-------------- Only applicable for Function 1.d if two or more channels are inoperable. ----------------------------------- Place channel in trip.	12 hours for Functions 2.a, 2.b, 5.h, 6.b, and 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)

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).	12 hours
	OR	
	D.2.1 Be in MODE 3.	12 hours
	AND	
	D.2.2 Be in MODE 4.	36 hours

(continued)



Primary Containment Isolation Instrumentation
3.3.6.1

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 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure - Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

(continued)

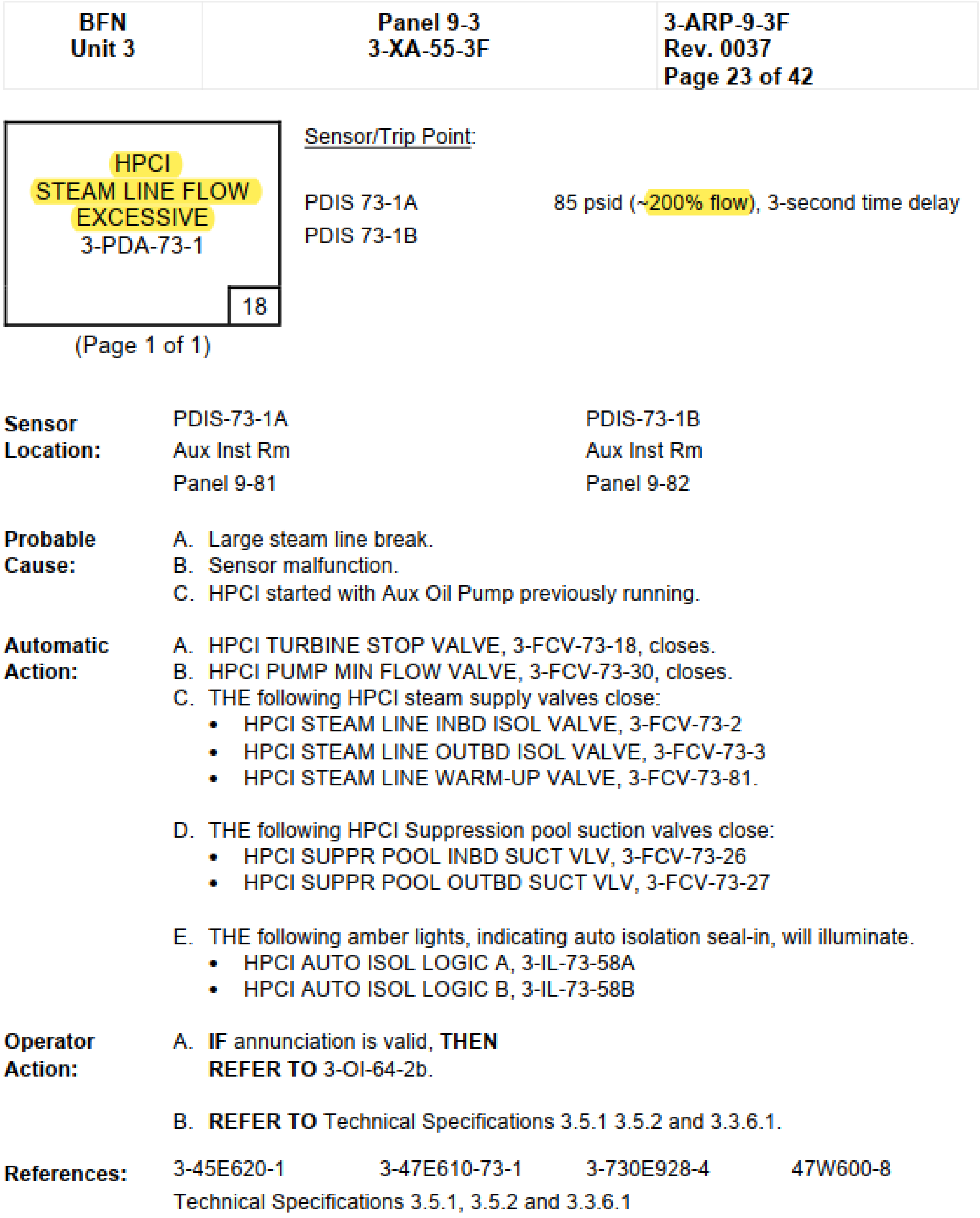
(c) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

BFN-UNIT 3 3.3-59 Amendment No. ~~212, 213, 219,~~ 254
September 14, 2006

Excerpt from 3-ARP-9-3F: Supports Distractors C(1), D(1)



BFN **Unit 3**	**Panel 9-3** **3-XA-55-3F**	**3-ARP-9-3F** **Rev. 0037** **Page 23 of 42**

HPCI
STEAM LINE FLOW
EXCESSIVE
3-PDA-73-1
18

(Page 1 of 1)

Sensor/Trip Point:

PDIS 73-1A 85 psid (~200% flow), 3-second time delay
PDIS 73-1B

Sensor Location:	PDIS-73-1A Aux Inst Rm Panel 9-81	PDIS-73-1B Aux Inst Rm Panel 9-82

Probable Cause:

A. Large steam line break.
B. Sensor malfunction.
C. HPCI started with Aux Oil Pump previously running.

Automatic Action:

A. HPCI TURBINE STOP VALVE, 3-FCV-73-18, closes.
B. HPCI PUMP MIN FLOW VALVE, 3-FCV-73-30, closes.
C. THE following HPCI steam supply valves close:
- HPCI STEAM LINE INBD ISOL VALVE, 3-FCV-73-2
- HPCI STEAM LINE OUTBD ISOL VALVE, 3-FCV-73-3
- HPCI STEAM LINE WARM-UP VALVE, 3-FCV-73-81.

D. THE following HPCI Suppression pool suction valves close:
- HPCI SUPPR POOL INBD SUCT VLV, 3-FCV-73-26
- HPCI SUPPR POOL OUTBD SUCT VLV, 3-FCV-73-27

E. THE following amber lights, indicating auto isolation seal-in, will illuminate.
- HPCI AUTO ISOL LOGIC A, 3-IL-73-58A
- HPCI AUTO ISOL LOGIC B, 3-IL-73-58B

Operator Action:

A. **IF** annunciation is valid, **THEN**
REFER TO 3-OI-64-2b.

B. **REFER TO** Technical Specifications 3.5.1 3.5.2 and 3.3.6.1.

References: 3-45E620-1 3-47E610-73-1 3-730E928-4 47W600-8
Technical Specifications 3.5.1, 3.5.2 and 3.3.6.1

Examination Outline Cross-reference:	Level	RO	SRO
Equipment Control	Tier #	------	3
G2.2.4 **(10CFR 55.43.2 – SRO Only)**	Group #	------	------
(Multi-unit license) Ability to explain the variations in control room layouts, systems, instrumentation, or procedural actions taken between units at a facility.	K/A #	G2.2.4	
	Importance Rating	------	3.6

Proposed Question: **# 95**

In accordance with 0-OI-65, Standby Gas Treatment System (SGT), it is **PREFERRED** to **START** SGT from the ____**(1)**____ Main Control Room(s).

In accordance with Tech Spec Bases, following a Design Basis Accident, a **MINIMUM** of ____**(2)**____ SGT train(s) are required to maintain a (-) 0.25 inches water while providing the design flow of 12000 cfm.

A. (1) Unit 3
(2) 1

B. (1) Unit 3
(2) 2

C. (1) Unit 1 **AND** Unit 2
(2) 1

D. (1) Unit 1 **AND** Unit 2
(2) 2

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that all 3 SGT trains can be started from Unit 3, but also from Unit 1 and 2 Main Control Rooms. Second part is incorrect but plausible in that the crews do not have a procedure that operates 2 SGT trains normally. When venting the Drywell in accordance with 0-OI-65, 1 SGT train is used. One train can be started if Reactor Building or Refuel ventilation is out of service. During the operation of HPCI, only one train is required to be in service. All 3 SGT trains are placed in service for RPS transfers or during outage activities that would initiate SGT, but that is not an option.

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-OI-65, Standby Gas Treatment System (SGT), it is recommended that the trains be started from the Units 1 and 2 Main Control Rooms due to the availability of instrumentation and shutdown capability. All 3 SGT trains can be started from Unit 3 and from Unit 1 and 2 Main Control Rooms. 'C' SGT train can be started/stopped from Unit 2 and 'A and B' SGT trains can be started/stopped from Unit 1 side of the common Unit 1 and 2 Main Control Rooms. Since Unit 3 can ONLY start 'A, B, C' SGT trains, it is not preferred to operate SGT from Unit 3. For second part, in accordance with Tech Spec Bases for SR 3.6.4.1.3 and SR 3.6.4.1.4, verifies that two SGT trains will draw at least 0.25 inches of vacuum water gauge in no more than 120 seconds at a stable flow rate of no more than 12000 cfm. Tech Spec Bases 3.6.4.3 states SGT is designed to perform this function in an accident and that any two trains are required to be able to perform this design function.

SRO Level Justification: Tests the candidate's ability to utilize Unit differences in controls, indications, and procedures related to the Standby Gas Treatment System and the applicable Technical Specification Bases as it relates to design bases. 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 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 T.S. Bases 3.6.4.1, Amend. 123 (Attach if not previously provided)

U2 T.S. Bases 3.6.4.3, Amend. 29

0-OI-65, Rev. 55

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.018 Obj. 11 (As available)

Question Source:	Bank #		
	Modified Bank #	OPL171.018-12 001 #667	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

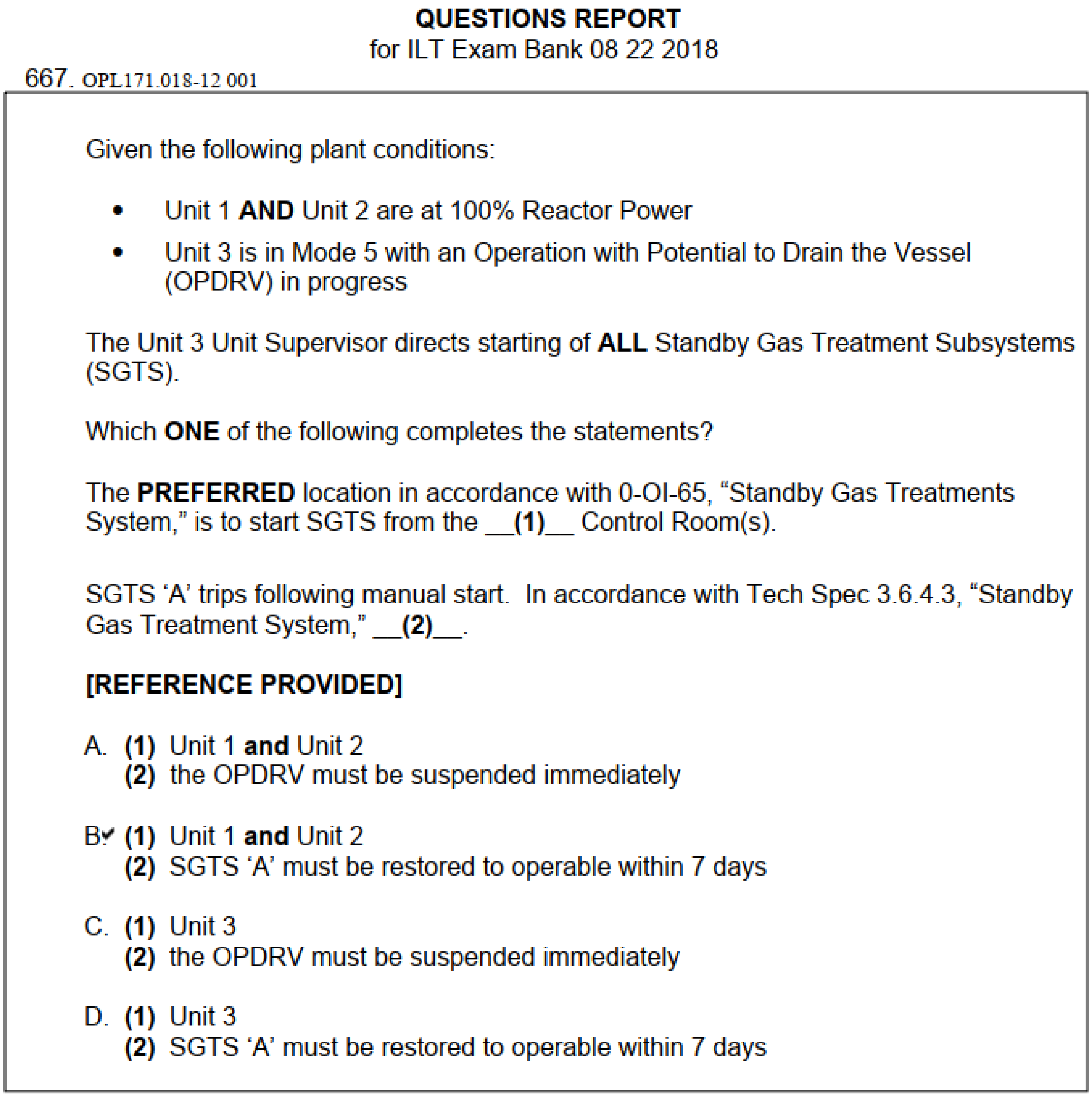
Question Cognitive Level: Memory or Fundamental Knowledge

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

667. OPL171.018-12 001

Given the following plant conditions:

- Unit 1 **AND** Unit 2 are at 100% Reactor Power
- Unit 3 is in Mode 5 with an Operation with Potential to Drain the Vessel (OPDRV) in progress

The Unit 3 Unit Supervisor directs starting of **ALL** Standby Gas Treatment Subsystems (SGTS).

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

The **PREFERRED** location in accordance with 0-OI-65, "Standby Gas Treatments System," is to start SGTS from the __**(1)**__ Control Room(s).

SGTS 'A' trips following manual start. In accordance with Tech Spec 3.6.4.3, "Standby Gas Treatment System," __**(2)**__.

[REFERENCE PROVIDED]

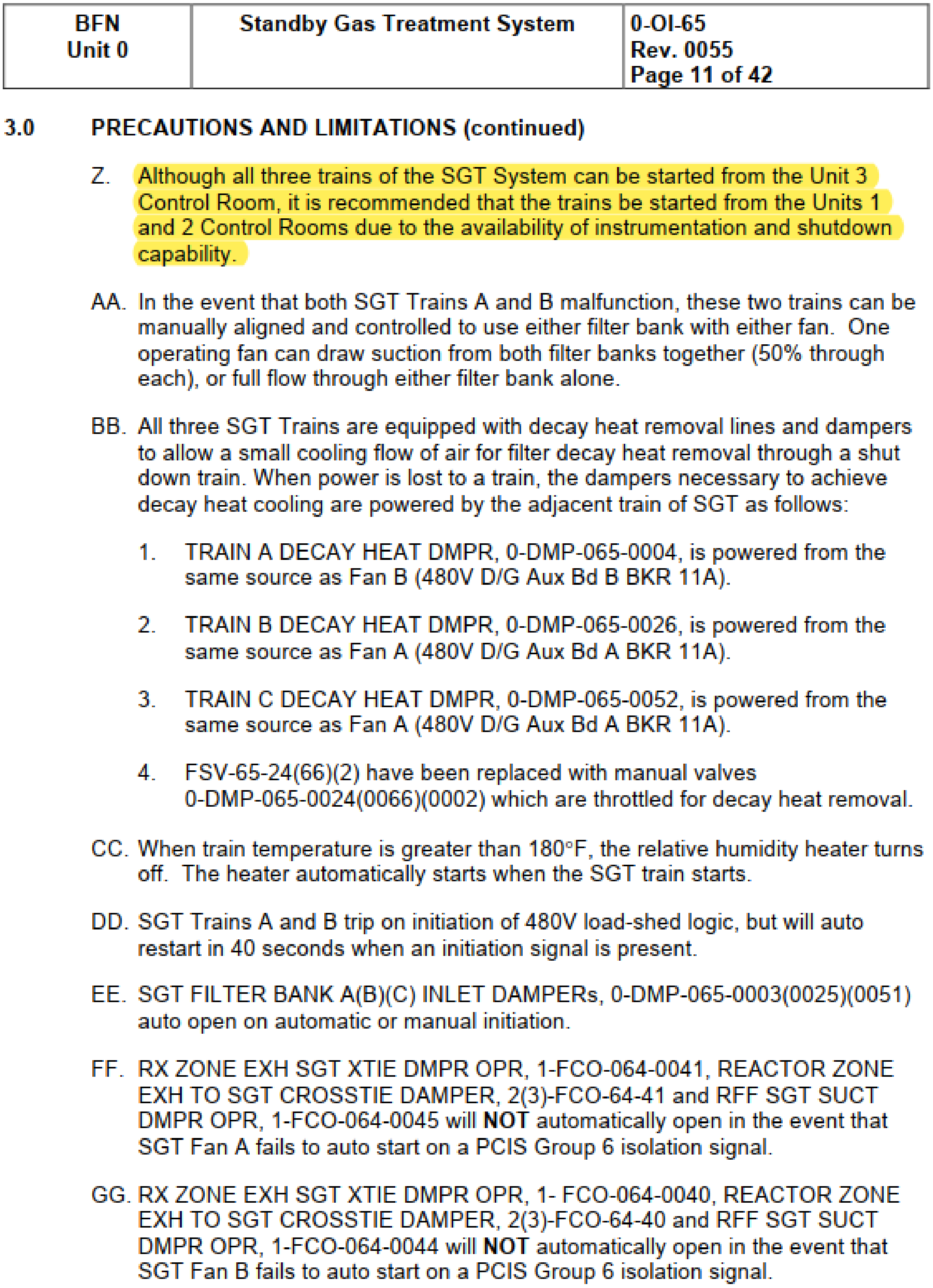
A. **(1)** Unit 1 **and** Unit 2
(2) the OPDRV must be suspended immediately

B✓ **(1)** Unit 1 **and** Unit 2
(2) SGTS 'A' must be restored to operable within 7 days

C. **(1)** Unit 3
(2) the OPDRV must be suspended immediately

D. **(1)** Unit 3
(2) SGTS 'A' must be restored to operable within 7 days

Excerpts from 0-OI-65:



BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0055 Page 11 of 42

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Z. Although all three trains of the SGT System can be started from the Unit 3 Control Room, it is recommended that the trains be started from the Units 1 and 2 Control Rooms due to the availability of instrumentation and shutdown capability.

AA. In the event that both SGT Trains A and B malfunction, these two trains can be manually aligned and controlled to use either filter bank with either fan. One operating fan can draw suction from both filter banks together (50% through each), or full flow through either filter bank alone.

BB. All three SGT Trains are equipped with decay heat removal lines and dampers to allow a small cooling flow of air for filter decay heat removal through a shut down train. When power is lost to a train, the dampers necessary to achieve decay heat cooling are powered by the adjacent train of SGT as follows:

1. TRAIN A DECAY HEAT DMPR, 0-DMP-065-0004, is powered from the same source as Fan B (480V D/G Aux Bd B BKR 11A).
2. TRAIN B DECAY HEAT DMPR, 0-DMP-065-0026, is powered from the same source as Fan A (480V D/G Aux Bd A BKR 11A).
3. TRAIN C DECAY HEAT DMPR, 0-DMP-065-0052, is powered from the same source as Fan A (480V D/G Aux Bd A BKR 11A).
4. FSV-65-24(66)(2) have been replaced with manual valves 0-DMP-065-0024(0066)(0002) which are throttled for decay heat removal.

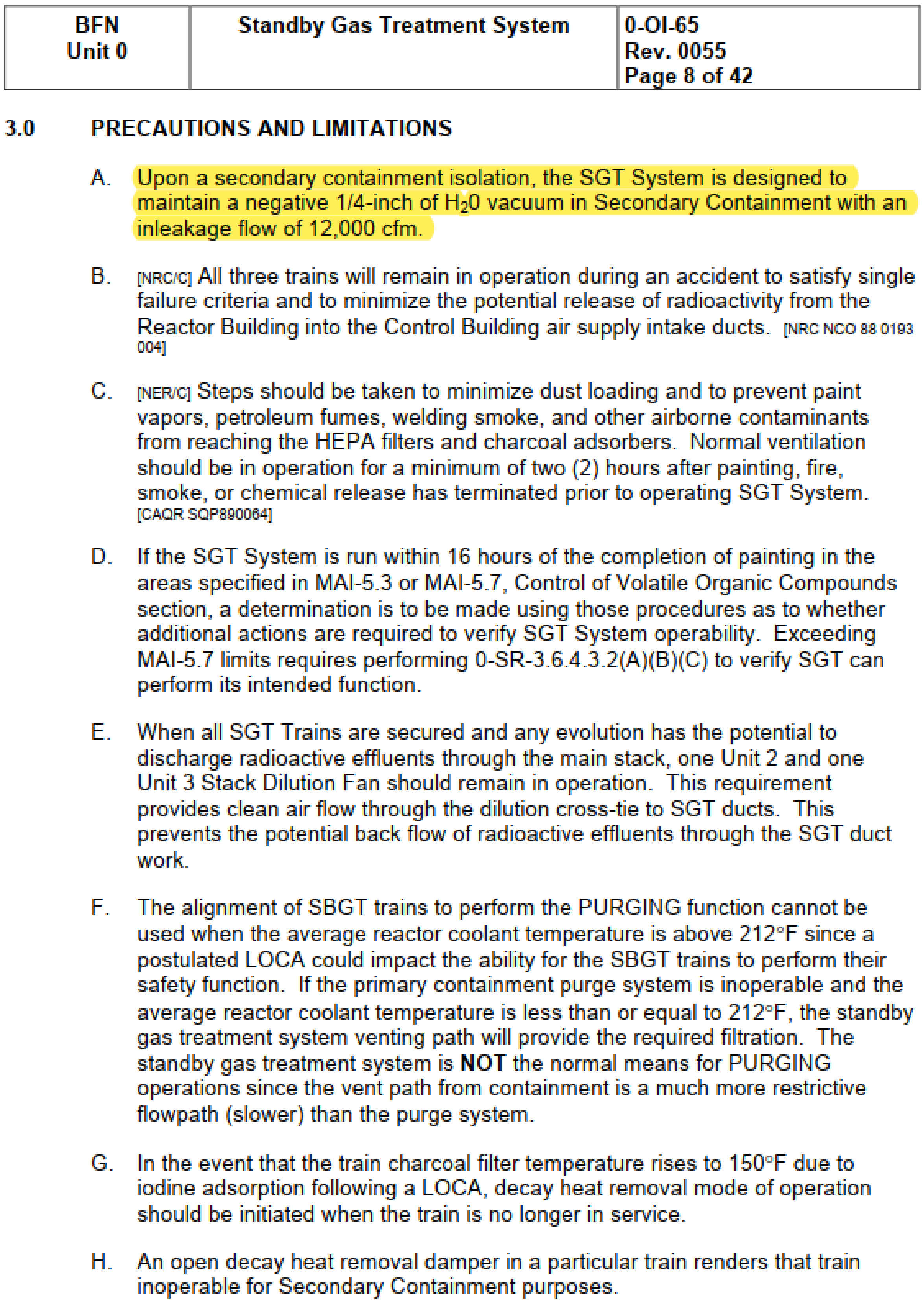
CC. When train temperature is greater than 180°F, the relative humidity heater turns off. The heater automatically starts when the SGT train starts.

DD. SGT Trains A and B trip on initiation of 480V load-shed logic, but will auto restart in 40 seconds when an initiation signal is present.

EE. SGT FILTER BANK A(B)(C) INLET DAMPERs, 0-DMP-065-0003(0025)(0051) auto open on automatic or manual initiation.

FF. RX ZONE EXH SGT XTIE DMPR OPR, 1-FCO-064-0041, REACTOR ZONE EXH TO SGT CROSSTIE DAMPER, 2(3)-FCO-64-41 and RFF SGT SUCT DMPR OPR, 1-FCO-064-0045 will **NOT** automatically open in the event that SGT Fan A fails to auto start on a PCIS Group 6 isolation signal.

GG. RX ZONE EXH SGT XTIE DMPR OPR, 1- FCO-064-0040, REACTOR ZONE EXH TO SGT CROSSTIE DAMPER, 2(3)-FCO-64-40 and RFF SGT SUCT DMPR OPR, 1-FCO-064-0044 will **NOT** automatically open in the event that SGT Fan B fails to auto start on a PCIS Group 6 isolation signal.



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.

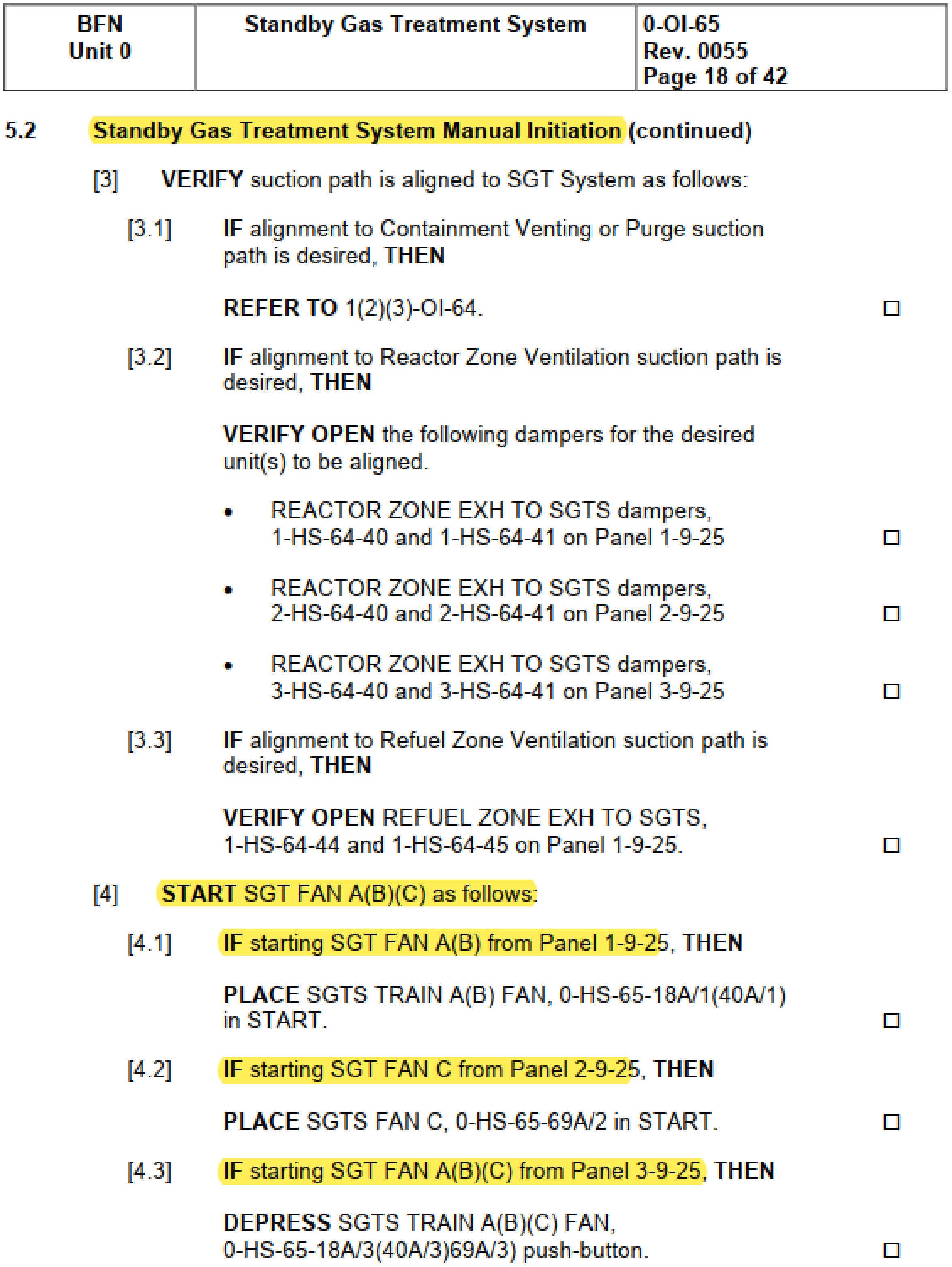
E. When all SGT Trains are secured and any evolution has the potential to discharge radioactive effluents through the main stack, one Unit 2 and one Unit 3 Stack Dilution Fan should remain in operation. This requirement provides clean air flow through the dilution cross-tie to SGT ducts. This prevents 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.

Illustrates the manual starting capability of SGT A(B)(C) trains and locations respectively:



BFN **Unit 0**	**Standby Gas Treatment System**	**0-OI-65** **Rev. 0055** **Page 18 of 42**

5.2 **Standby Gas Treatment System Manual Initiation (continued)**

[3] **VERIFY** suction path is aligned to SGT System as follows:

[3.1] **IF** alignment to Containment Venting or Purge suction path is desired, **THEN**

REFER TO 1(2)(3)-OI-64. ☐

[3.2] **IF** alignment to Reactor Zone Ventilation suction path is desired, **THEN**

VERIFY OPEN the following dampers for the desired unit(s) to be aligned.

- REACTOR ZONE EXH TO SGTS dampers, 1-HS-64-40 and 1-HS-64-41 on Panel 1-9-25 ☐
- REACTOR ZONE EXH TO SGTS dampers, 2-HS-64-40 and 2-HS-64-41 on Panel 2-9-25 ☐
- REACTOR ZONE EXH TO SGTS dampers, 3-HS-64-40 and 3-HS-64-41 on Panel 3-9-25 ☐

[3.3] **IF** alignment to Refuel Zone Ventilation suction path is desired, **THEN**

VERIFY OPEN REFUEL ZONE EXH TO SGTS, 1-HS-64-44 and 1-HS-64-45 on Panel 1-9-25. ☐

[4] **START** SGT FAN A(B)(C) as follows:

[4.1] **IF** starting SGT FAN A(B) from Panel 1-9-25, **THEN**

PLACE SGTS TRAIN A(B) FAN, 0-HS-65-18A/1(40A/1) in START. ☐

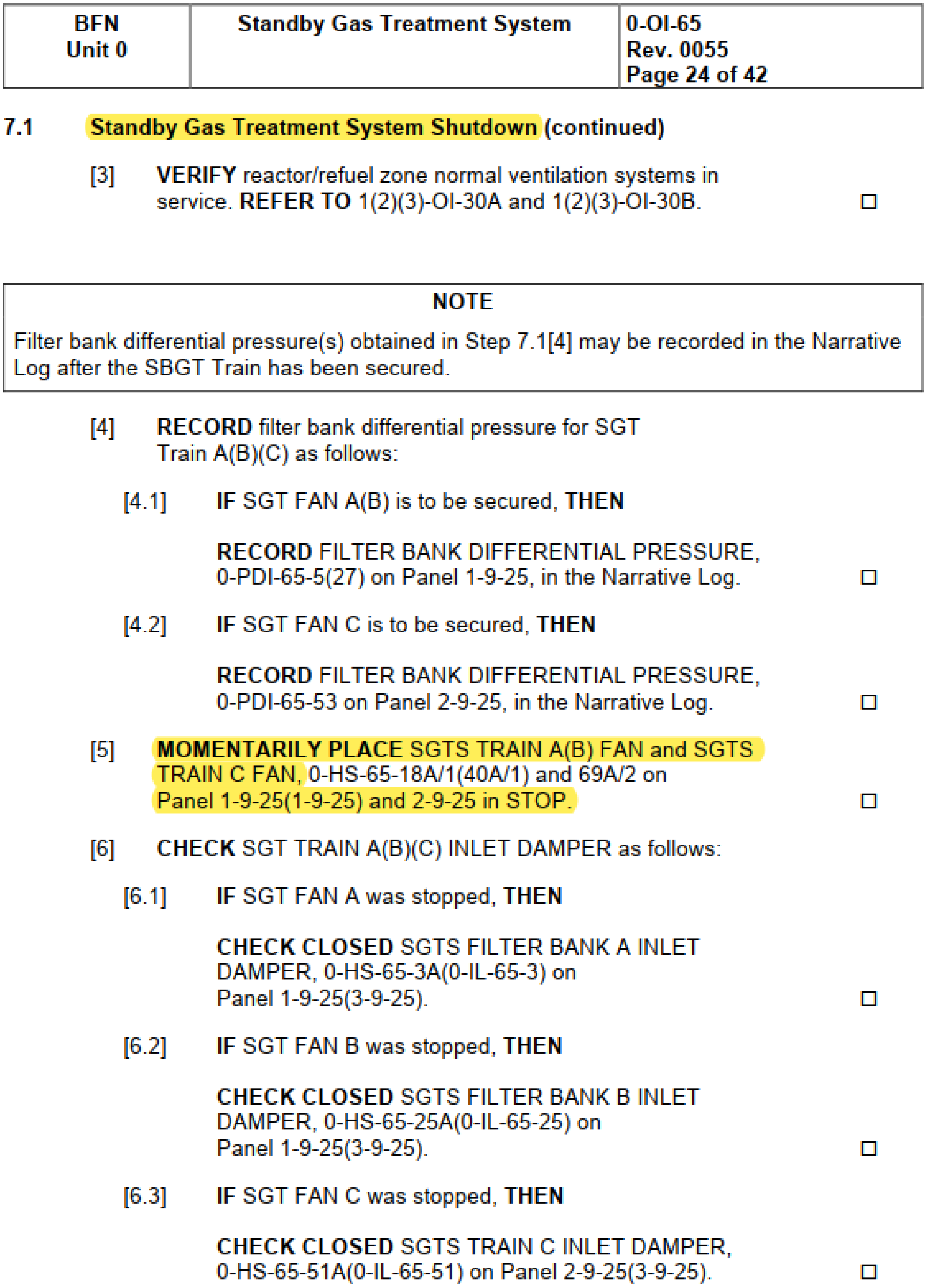
[4.2] **IF** starting SGT FAN C from Panel 2-9-25, **THEN**

PLACE SGTS FAN C, 0-HS-65-69A/2 in START. ☐

[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. ☐

Illustrates the shutdown capability of SGT A(B)(C) trains and locations respectively:



BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0055 Page 24 of 42

7.1 Standby Gas Treatment System Shutdown (continued)

[3] **VERIFY** reactor/refuel zone normal ventilation systems in service. **REFER TO** 1(2)(3)-OI-30A and 1(2)(3)-OI-30B. ☐

NOTE

Filter bank differential pressure(s) obtained in Step 7.1[4] may be recorded in the Narrative Log after the SBGT Train has been secured.

[4] **RECORD** filter bank differential pressure for SGT Train A(B)(C) as follows:

[4.1] **IF** SGT FAN A(B) is to be secured, **THEN**

RECORD FILTER BANK DIFFERENTIAL PRESSURE, 0-PDI-65-5(27) on Panel 1-9-25, in the Narrative Log. ☐

[4.2] **IF** SGT FAN C is to be secured, **THEN**

RECORD FILTER BANK DIFFERENTIAL PRESSURE, 0-PDI-65-53 on Panel 2-9-25, in the Narrative Log. ☐

[5] **MOMENTARILY PLACE** SGTS TRAIN A(B) FAN and SGTS TRAIN C FAN, 0-HS-65-18A/1(40A/1) and 69A/2 on Panel 1-9-25(1-9-25) and 2-9-25 in STOP. ☐

[6] **CHECK** SGT TRAIN A(B)(C) INLET DAMPER as follows:

[6.1] **IF** SGT FAN A was stopped, **THEN**

CHECK CLOSED SGTS FILTER BANK A INLET DAMPER, 0-HS-65-3A(0-IL-65-3) on Panel 1-9-25(3-9-25). ☐

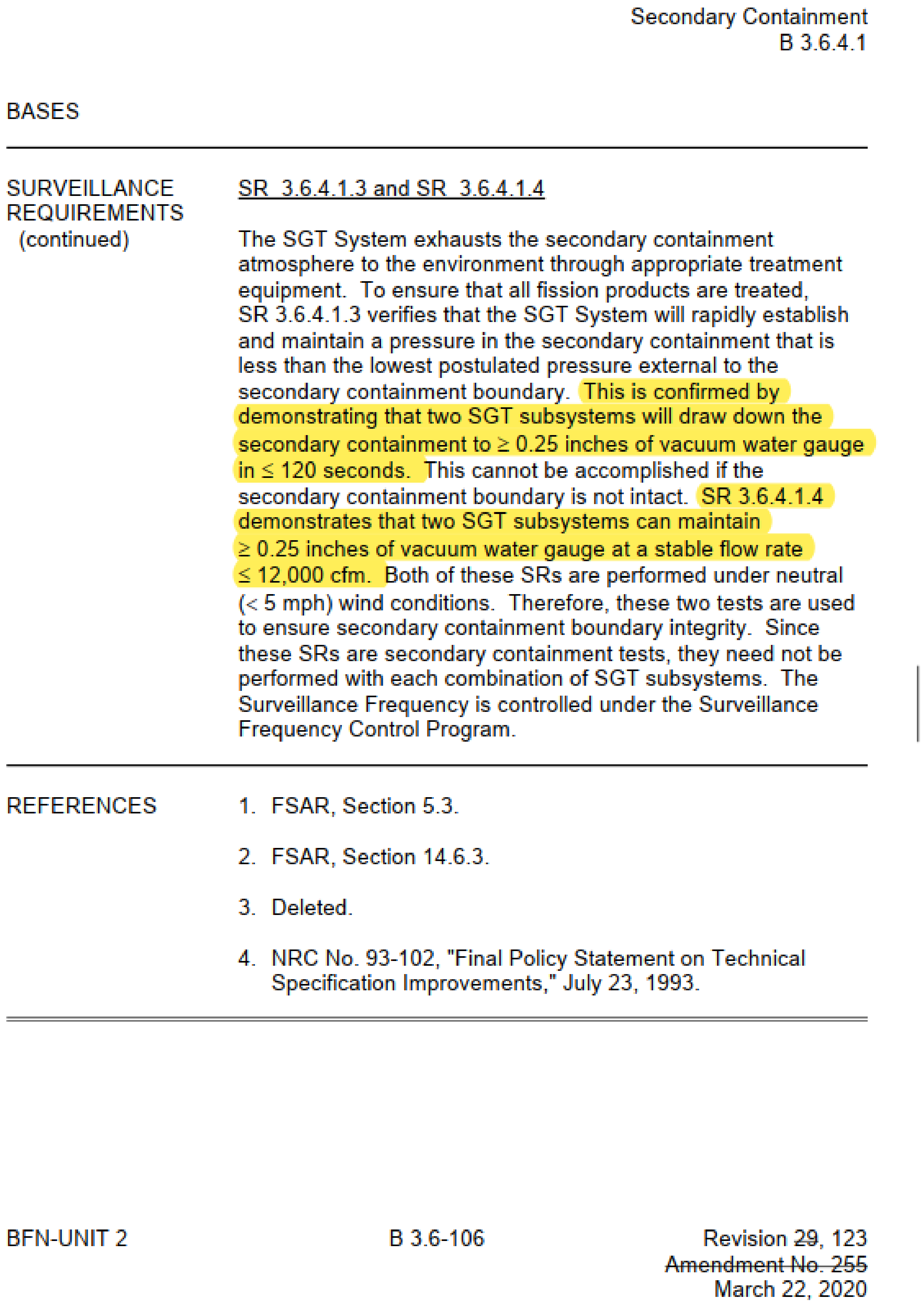
[6.2] **IF** SGT FAN B was stopped, **THEN**

CHECK CLOSED SGTS FILTER BANK B INLET DAMPER, 0-HS-65-25A(0-IL-65-25) on Panel 1-9-25(3-9-25). ☐

[6.3] **IF** SGT FAN C was stopped, **THEN**

CHECK CLOSED SGTS TRAIN C INLET DAMPER, 0-HS-65-51A(0-IL-65-51) on Panel 2-9-25(3-9-25). ☐

Excerpt from Unit 2 Tech Spec Bases 3.6.4.1: (Applicable for all 3 Units since Standby Gas is for common use for all Units)



Secondary Containment
B 3.6.4.1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that two SGT subsystems will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ 120 seconds. This cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.4 demonstrates that two SGT subsystems can maintain ≥ 0.25 inches of vacuum water gauge at a stable flow rate ≤ 12,000 cfm. Both of these SRs are performed under neutral (< 5 mph) wind conditions. Therefore, these two tests are used to ensure secondary containment boundary integrity. Since these SRs are secondary containment tests, they need not be performed with each combination of SGT subsystems. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 5.3.
2. FSAR, Section 14.6.3.
3. Deleted.
4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

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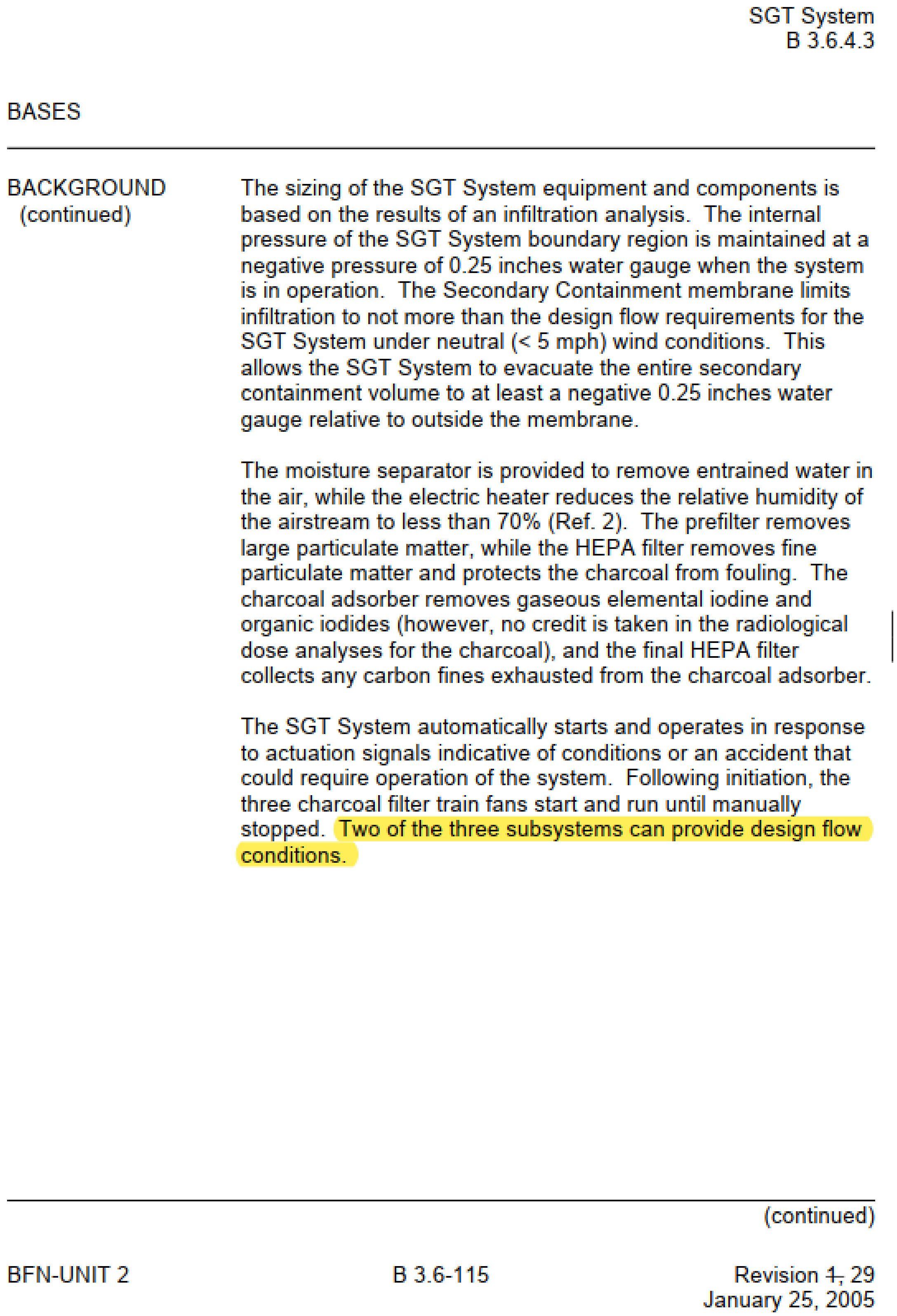
~~Amendment No. 255~~
March 22, 2020

Excerpts from Unit 2 Tech Spec Bases 3.6.4.3: (Applicable for all 3 Units since Standby Gas is for common use for all Units):



SGT System
B 3.6.4.3

BASES

BACKGROUND (continued)

The sizing of the SGT System equipment and components is based on the results of an infiltration analysis. The internal pressure of the SGT System boundary region is maintained at a negative pressure of 0.25 inches water gauge when the system is in operation. The Secondary Containment membrane limits infiltration to not more than the design flow requirements for the SGT System under neutral (< 5 mph) wind conditions. This allows the SGT System to evacuate the entire secondary containment volume to at least a negative 0.25 inches water gauge relative to outside the membrane.

The moisture separator is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides (however, no credit is taken in the radiological dose analyses for the charcoal), and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, the three charcoal filter train fans start and run until manually stopped. Two of the three subsystems can provide design flow conditions.

(continued)

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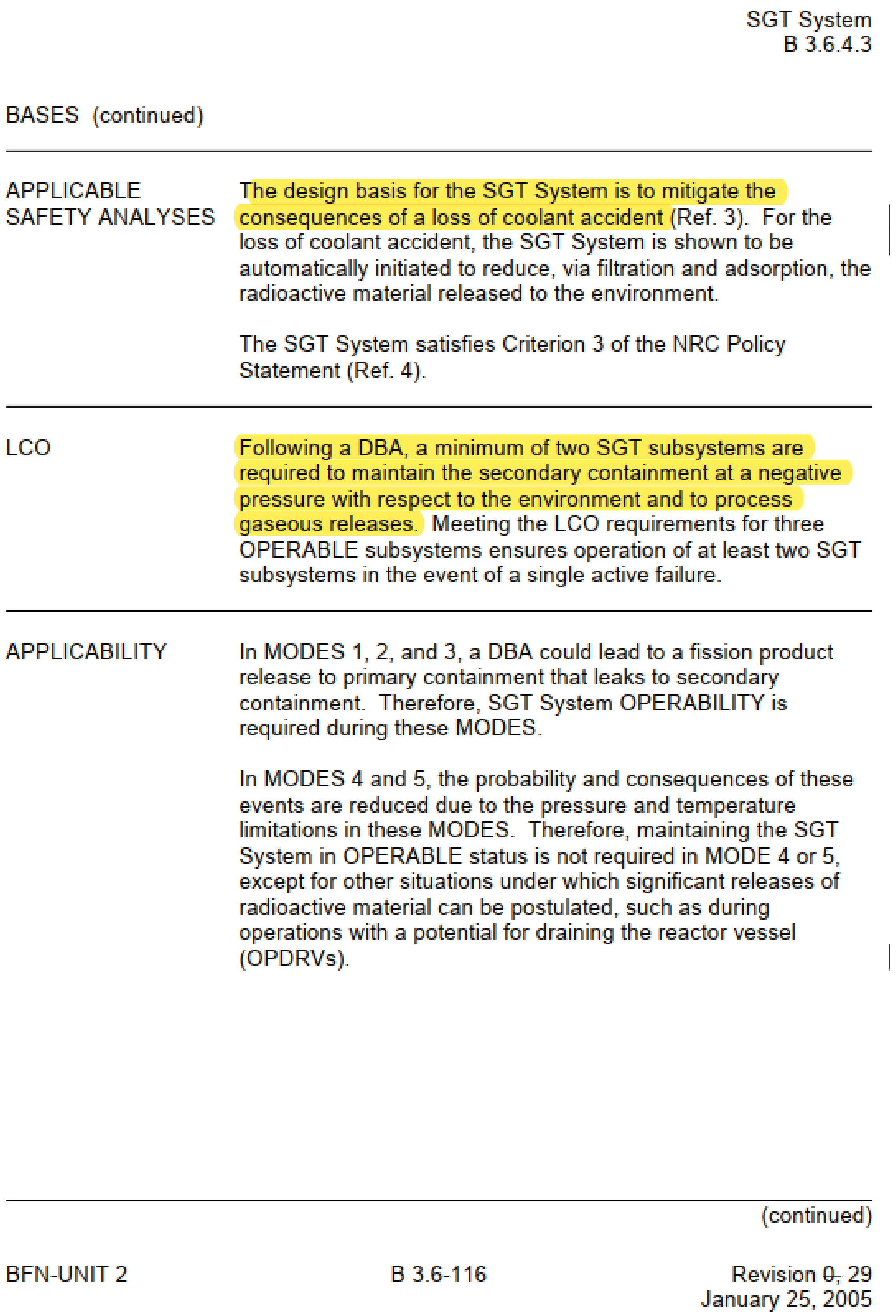
January 25, 2005



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).
LCO	Following a DBA, a minimum of two SGT subsystems are required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for three OPERABLE subsystems ensures operation of at least two SGT subsystems in the event of a single active failure.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs).

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
Equipment Control **G2.2.19 (10CFR 55.43.5 – SRO Only)** Knowledge of maintenance work order requirements.	Tier #	------	3
	Group #	------	------
	K/A #	G2.2.19	
	Importance Rating	------	3.4

Proposed Question: **# 96**

In accordance with NPG-SPP-07.3, Work Activity Risk Management Process, Operations ____**(1)**____ **REQUIRED** to review Attachment 4, BWR Operational Risk Review – RED SHEET for a normally scheduled performance of 0-SR-3.8.1.1(A) – Diesel Generator 'A' Monthly Operability Test.

In accordance with NPG-SPP 6.1, Work Order Process, the____**(2)**____ has the responsibility of assigning a task as a Priority 1 emergent work order.

A. (1) is NOT
(2) Shift Manager

B. (1) is NOT
(2) Work Week Manager

C. (1) is
(2) Shift Manager

D. (1) is
(2) Work Week Manager

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with NPG-SPP-07.3, Work Activity Risk Management Process, Section 3.6, periodic instructions performed more than once per refueling cycle requires the completion of the High Risk Management Plan (Form 41218) at least once per cycle to assess risk. The RED SHEET, ATTACHMENT 4, states that it is completed for activities that are not normally scheduled activities. For second part, in accordance with NPG-SPP-06.1, Work Order Process, the Shift Manager has the responsibility of assigning the Priority to a Work Order.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible in that the Work Week Manager (WWM) owns the process and ensuring the work is risk reviewed for the responsible WWM's week.
In accordance with NPG-SPP-07.3, Work Activity Risk Management Process, the WWM responsibilities include monitoring/managing all risk for his work week, ensures all emergent work has risk reviews, schedules critical evolutions are scheduled and completed, and tracking high risk for week.

C INCORRECT: First part is incorrect but plausible in that in accordance with NPG-SPP-07.3, the RED SHEET is required for risk evaluation typically when work must be completed, but has not been through the normal risk screening process. It can be used at the discretion of Operations when an activity is performed as additional RISK identification methods, but not required. Since the test involves EDGs and the grid, the risk involved could be assumed to be at a point where the candidate thinks the RED SHEET is required. 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 work order development and the risk review process. 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 strictly recalling facts related to work order development.

Technical Reference(s): NPG-SPP-07.3, Rev.36 (Attach if not previously provided)

NPG-SPP-06.1, Rev. 11

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.239 Obj. 3_ (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**

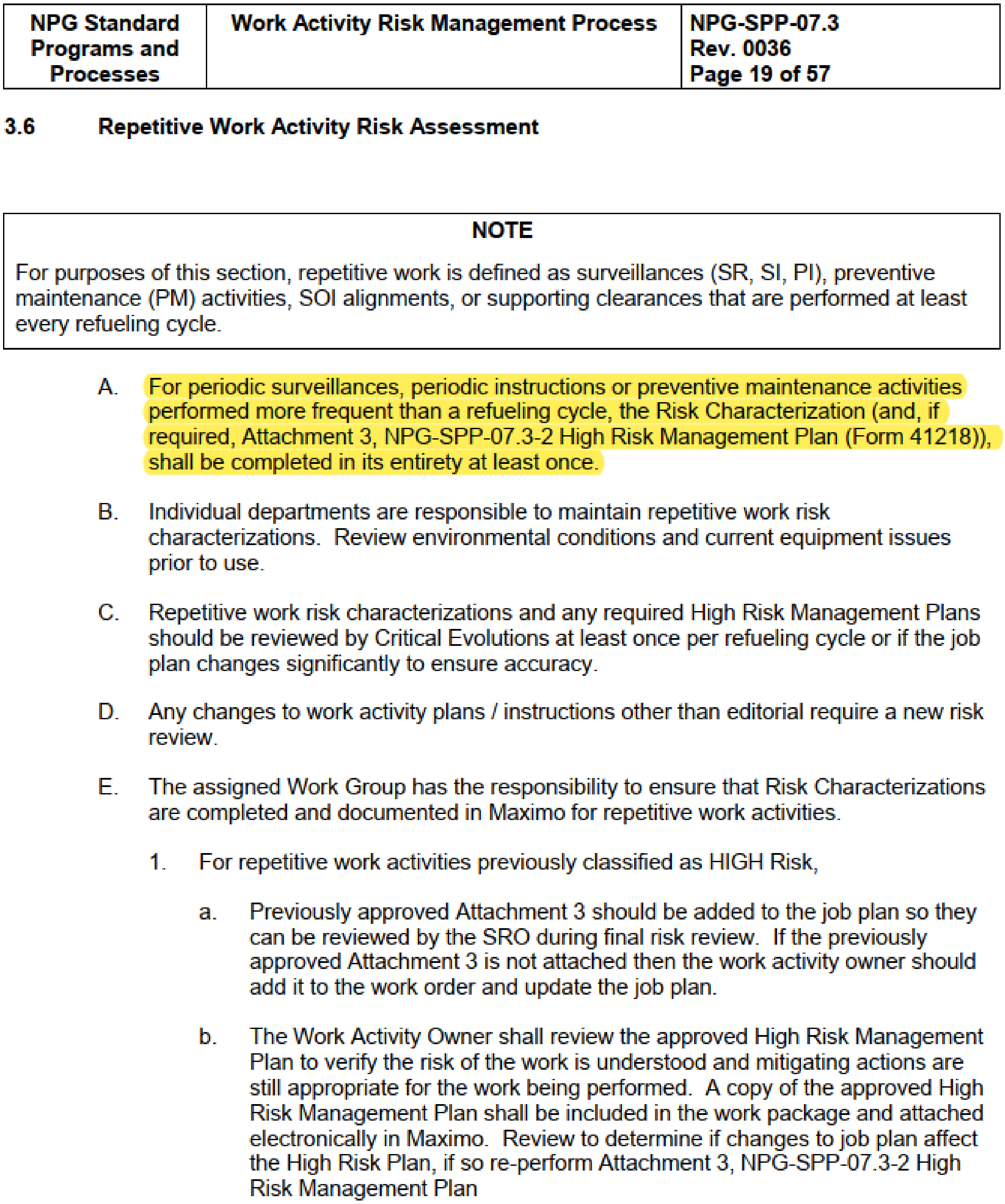
Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Excerpts from NPG-SPP-07.3:



NPG Standard Programs and Processes	**Work Activity Risk Management Process**	**NPG-SPP-07.3** **Rev. 0036** **Page 19 of 57**

3.6 Repetitive Work Activity Risk Assessment

NOTE

For purposes of this section, repetitive work is defined as surveillances (SR, SI, PI), preventive maintenance (PM) activities, SOI alignments, or supporting clearances that are performed at least every refueling cycle.

A. For periodic surveillances, periodic instructions or preventive maintenance activities performed more frequent than a refueling cycle, the Risk Characterization (and, if required, Attachment 3, NPG-SPP-07.3-2 High Risk Management Plan (Form 41218)), shall be completed in its entirety at least once.

B. Individual departments are responsible to maintain repetitive work risk characterizations. Review environmental conditions and current equipment issues prior to use.

C. Repetitive work risk characterizations and any required High Risk Management Plans should be reviewed by Critical Evolutions at least once per refueling cycle or if the job plan changes significantly to ensure accuracy.

D. Any changes to work activity plans / instructions other than editorial require a new risk review.

E. The assigned Work Group has the responsibility to ensure that Risk Characterizations are completed and documented in Maximo for repetitive work activities.

  1. For repetitive work activities previously classified as HIGH Risk,

    a. Previously approved Attachment 3 should be added to the job plan so they can be reviewed by the SRO during final risk review. If the previously approved Attachment 3 is not attached then the work activity owner should add it to the work order and update the job plan.

    b. The Work Activity Owner shall review the approved High Risk Management Plan to verify the risk of the work is understood and mitigating actions are still appropriate for the work being performed. A copy of the approved High Risk Management Plan shall be included in the work package and attached electronically in Maximo. Review to determine if changes to job plan affect the High Risk Plan, if so re-perform Attachment 3, NPG-SPP-07.3-2 High Risk Management Plan

3.1.7 Manager - Online Work Management (continued)

D. Monitors work week development for low risk activities to bring to Critical Evolutions. This is intended to be an opportunity for the team to better understand routine low risk work or as a development tool for the site by exercising the process. This selection practice is discretionary and not required by this procedure.

E. Ensures all scheduled high risk activities are presented to Critical Evolutions.

3.1.8 Operations SRO

A. Performs Operational Impact review for planned work and documents any Operational Impacts on the Operations tab in Maximo.

B. Review the Operational impacts tab in Maximo for any component failure modes that the planner may have annotated under plant response.

C. Responsible for documenting the Final Risk Characterization for the work activity in the RISK ASSESSMENT field in Maximo using Attachment 6, Use of Electronic Documentation. This Final Risk Characterization and Operational Impact review should be performed by the SRO advancing the Work Order status. The SRO is the final authority for all risk classifications. Final risk shall be based on using a consequence bias by setting probabiliy to 1 (Risk = Probability x Consequence).

D. Performs review for pre-authorization of work using Attachment 9, Work Activities that are Eligible for Pre-Authorization and documents required level of Operations review of work during the execution week in the work order.

E. Evaluates combination of LOW risk activities planned by one or more workgroups that may create a HIGH risk for the site (use of discretionary High risk classification on Site Aggregate Risk Assessment) due to the nature of the work or system interrelationships. This is not an objective evaluation, but a subjective evaluation using the skills and knowledge inherent in the qualifications of an SRO.

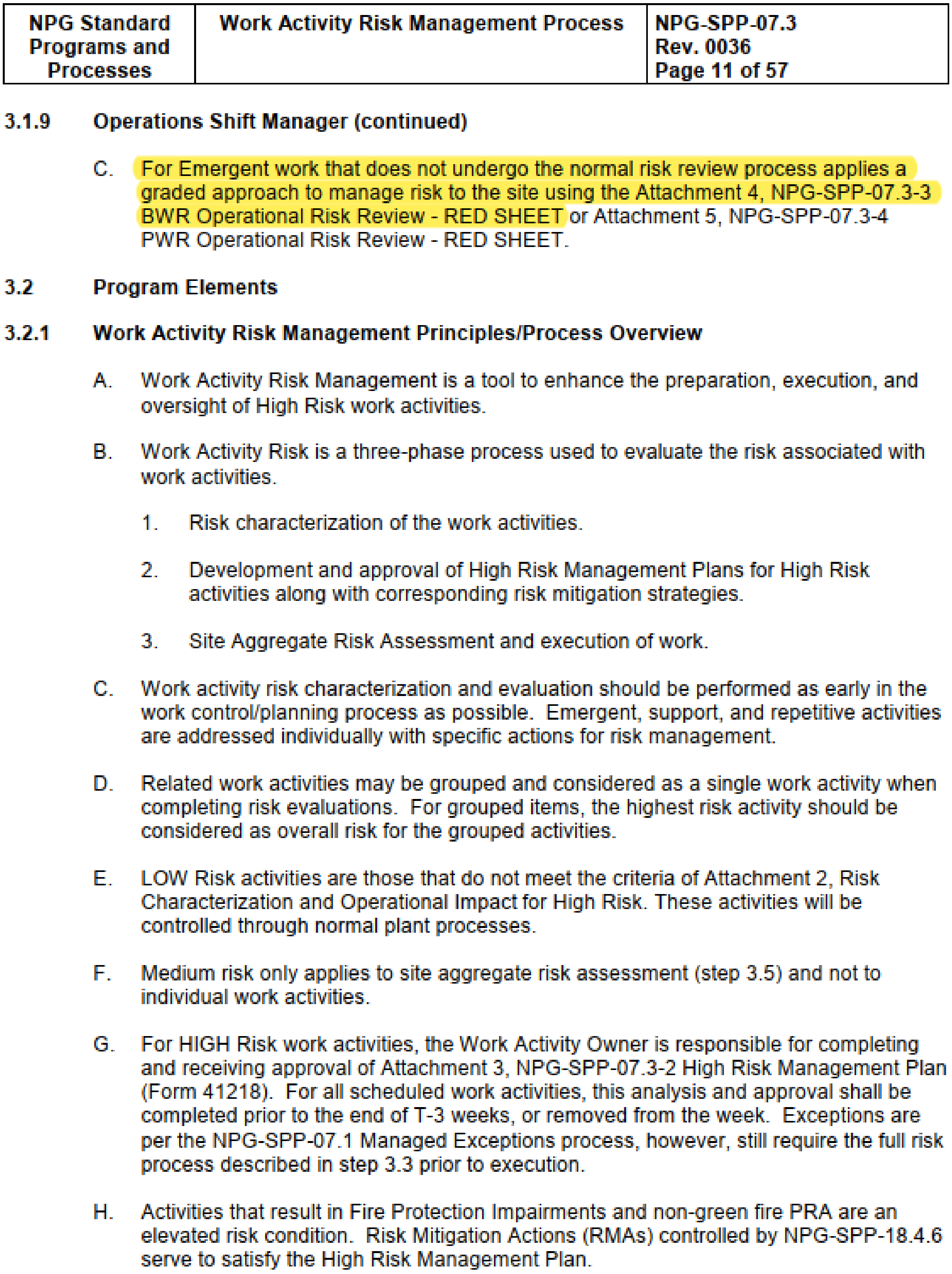
F. Assists with maintaining Attachment 8, Site Aggregate Risk Assessment with the help of Work Week Manager.

G. When required by step 3.7 or the Shift Manager, completes an Operational risk review for Emergent, High priority work using Attachment 4, NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET / Attachment 5, NPG-SPP-07.3-4 PWR Operational Risk Review - RED SHEET.

3.1.9 Operations Shift Manager

A. Responsible for implementation of the site schedule and ensuring the Site Aggregate Risk Assessment is current based upon plant conditions and Emergent Work.

B. Responsible for managing risk changes based on plant conditions, schedule changes, and Emergent Work.



NPG Standard Programs and Processes	**Work Activity Risk Management Process**	**NPG-SPP-07.3** **Rev. 0036** **Page 11 of 57**

3.1.9 Operations Shift Manager (continued)

C. For Emergent work that does not undergo the normal risk review process applies a graded approach to manage risk to the site using the Attachment 4, NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET or Attachment 5, NPG-SPP-07.3-4 PWR Operational Risk Review - RED SHEET.

3.2 Program Elements

3.2.1 Work Activity Risk Management Principles/Process Overview

A. Work Activity Risk Management is a tool to enhance the preparation, execution, and oversight of High Risk work activities.

B. Work Activity Risk is a three-phase process used to evaluate the risk associated with work activities.

1. Risk characterization of the work activities.
2. Development and approval of High Risk Management Plans for High Risk activities along with corresponding risk mitigation strategies.
3. Site Aggregate Risk Assessment and execution of work.

C. Work activity risk characterization and evaluation should be performed as early in the work control/planning process as possible. Emergent, support, and repetitive activities are addressed individually with specific actions for risk management.

D. Related work activities may be grouped and considered as a single work activity when completing risk evaluations. For grouped items, the highest risk activity should be considered as overall risk for the grouped activities.

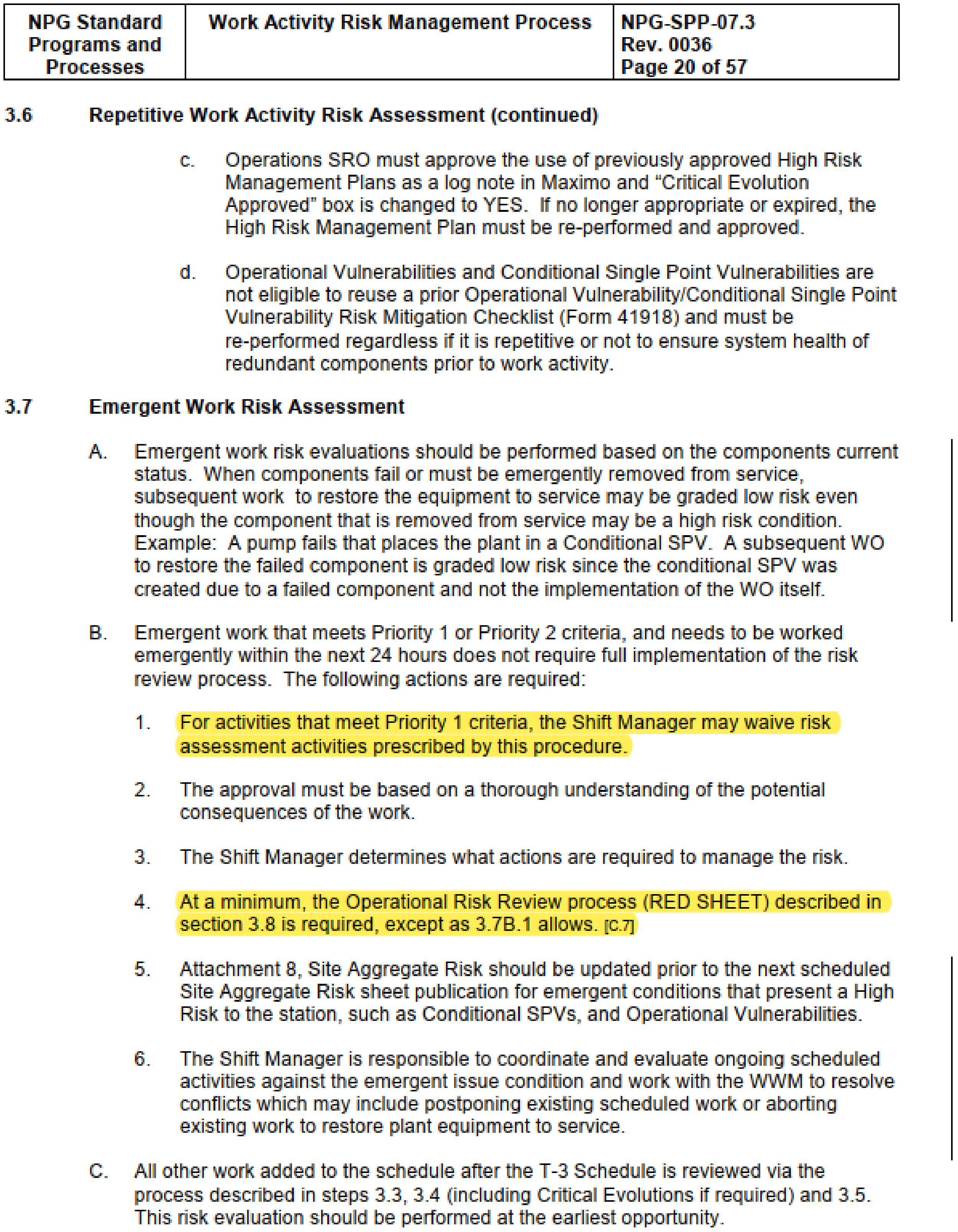
E. LOW Risk activities are those that do not meet the criteria of Attachment 2, Risk Characterization and Operational Impact for High Risk. These activities will be controlled through normal plant processes.

F. Medium risk only applies to site aggregate risk assessment (step 3.5) and not to individual work activities.

G. For HIGH Risk work activities, the Work Activity Owner is responsible for completing and receiving approval of Attachment 3, NPG-SPP-07.3-2 High Risk Management Plan (Form 41218). For all scheduled work activities, this analysis and approval shall be completed prior to the end of T-3 weeks, or removed from the week. Exceptions are per the NPG-SPP-07.1 Managed Exceptions process, however, still require the full risk process described in step 3.3 prior to execution.

H. Activities that result in Fire Protection Impairments and non-green fire PRA are an elevated risk condition. Risk Mitigation Actions (RMAs) controlled by NPG-SPP-18.4.6 serve to satisfy the High Risk Management Plan.

Supports Distractors C(1), D(1):



NPG Standard Programs and Processes	**Work Activity Risk Management Process**	**NPG-SPP-07.3** **Rev. 0036** **Page 20 of 57**

3.6 Repetitive Work Activity Risk Assessment (continued)

c. Operations SRO must approve the use of previously approved High Risk Management Plans as a log note in Maximo and “Critical Evolution Approved” box is changed to YES. If no longer appropriate or expired, the High Risk Management Plan must be re-performed and approved.

d. Operational Vulnerabilities and Conditional Single Point Vulnerabilities are not eligible to reuse a prior Operational Vulnerability/Conditional Single Point Vulnerability Risk Mitigation Checklist (Form 41918) and must be re-performed regardless if it is repetitive or not to ensure system health of redundant components prior to work activity.

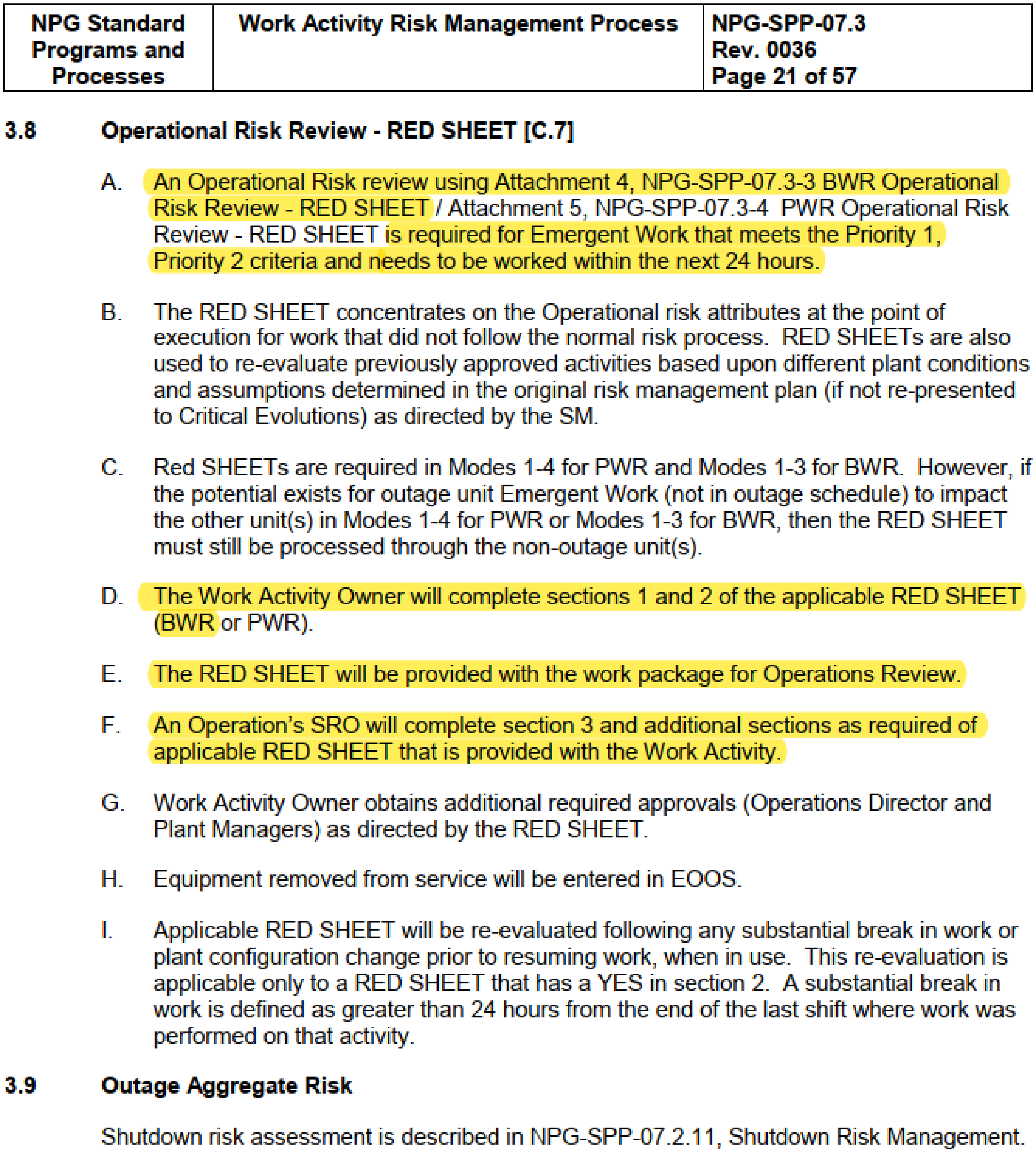
3.7 Emergent Work Risk Assessment

A. Emergent work risk evaluations should be performed based on the components current status. When components fail or must be emergently removed from service, subsequent work to restore the equipment to service may be graded low risk even though the component that is removed from service may be a high risk condition. Example: A pump fails that places the plant in a Conditional SPV. A subsequent WO to restore the failed component is graded low risk since the conditional SPV was created due to a failed component and not the implementation of the WO itself.

B. Emergent work that meets Priority 1 or Priority 2 criteria, and needs to be worked emergently within the next 24 hours does not require full implementation of the risk review process. The following actions are required:

1. For activities that meet Priority 1 criteria, the Shift Manager may waive risk assessment activities prescribed by this procedure.
2. The approval must be based on a thorough understanding of the potential consequences of the work.
3. The Shift Manager determines what actions are required to manage the risk.
4. At a minimum, the Operational Risk Review process (RED SHEET) described in section 3.8 is required, except as 3.7B.1 allows. [C.7]
5. Attachment 8, Site Aggregate Risk should be updated prior to the next scheduled Site Aggregate Risk sheet publication for emergent conditions that present a High Risk to the station, such as Conditional SPVs, and Operational Vulnerabilities.
6. The Shift Manager is responsible to coordinate and evaluate ongoing scheduled activities against the emergent issue condition and work with the WWM to resolve conflicts which may include postponing existing scheduled work or aborting existing work to restore plant equipment to service.

C. All other work added to the schedule after the T-3 Schedule is reviewed via the process described in steps 3.3, 3.4 (including Critical Evolutions if required) and 3.5. This risk evaluation should be performed at the earliest opportunity.



NPG Standard Programs and Processes	**Work Activity Risk Management Process**	**NPG-SPP-07.3** **Rev. 0036** **Page 21 of 57**

3.8 Operational Risk Review - RED SHEET [C.7]

A. An Operational Risk review using Attachment 4, NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET / Attachment 5, NPG-SPP-07.3-4 PWR Operational Risk Review - RED SHEET is required for Emergent Work that meets the Priority 1, Priority 2 criteria and needs to be worked within the next 24 hours.

B. The RED SHEET concentrates on the Operational risk attributes at the point of execution for work that did not follow the normal risk process. RED SHEETs are also used to re-evaluate previously approved activities based upon different plant conditions and assumptions determined in the original risk management plan (if not re-presented to Critical Evolutions) as directed by the SM.

C. Red SHEETs are required in Modes 1-4 for PWR and Modes 1-3 for BWR. However, if the potential exists for outage unit Emergent Work (not in outage schedule) to impact the other unit(s) in Modes 1-4 for PWR or Modes 1-3 for BWR, then the RED SHEET must still be processed through the non-outage unit(s).

D. The Work Activity Owner will complete sections 1 and 2 of the applicable RED SHEET (BWR or PWR).

E. The RED SHEET will be provided with the work package for Operations Review.

F. An Operation's SRO will complete section 3 and additional sections as required of applicable RED SHEET that is provided with the Work Activity.

G. Work Activity Owner obtains additional required approvals (Operations Director and Plant Managers) as directed by the RED SHEET.

H. Equipment removed from service will be entered in EOOS.

I. Applicable RED SHEET will be re-evaluated following any substantial break in work or plant configuration change prior to resuming work, when in use. This re-evaluation is applicable only to a RED SHEET that has a YES in section 2. A substantial break in work is defined as greater than 24 hours from the end of the last shift where work was performed on that activity.

3.9 Outage Aggregate Risk

Shutdown risk assessment is described in NPG-SPP-07.2.11, Shutdown Risk Management.

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Attachment 4
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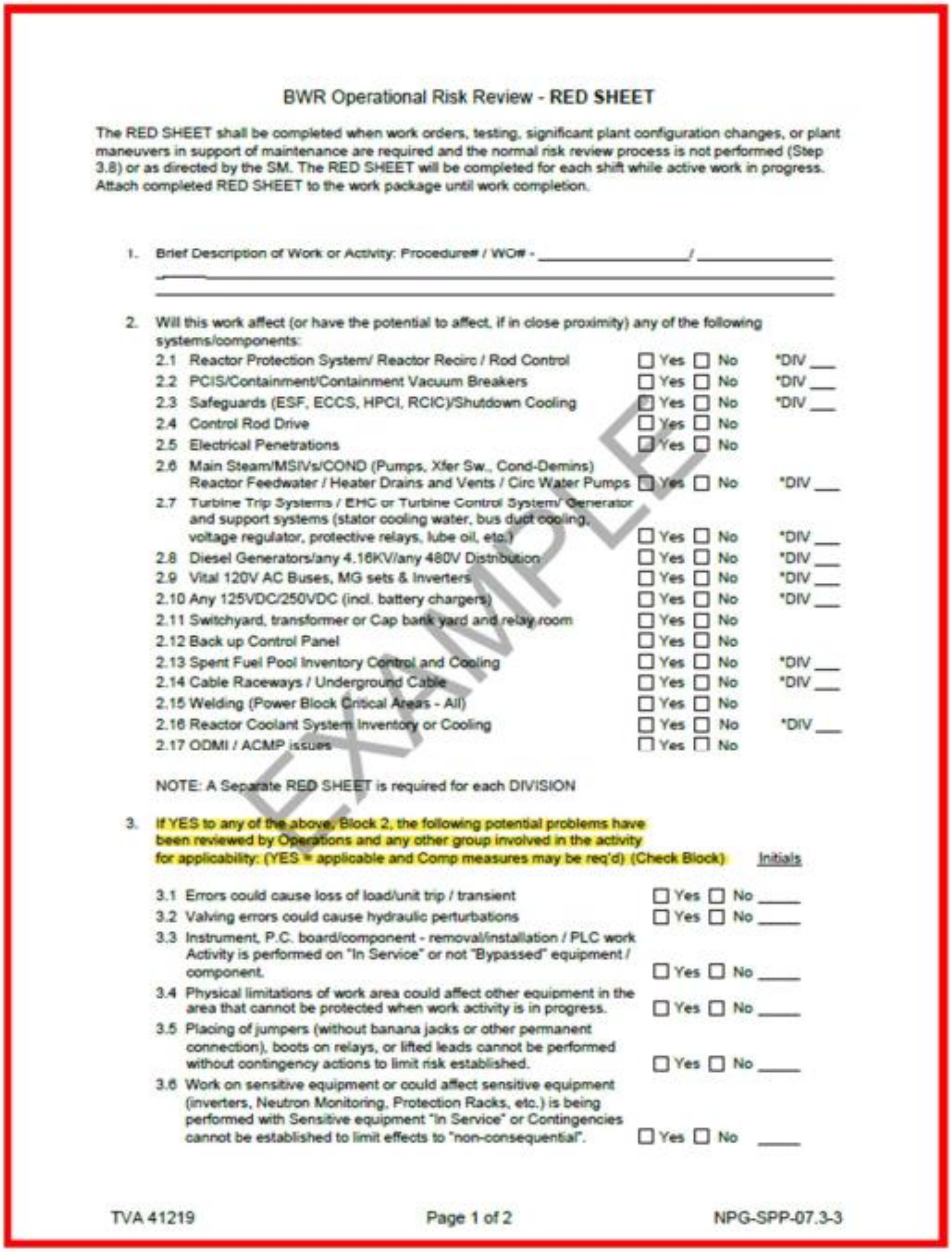
NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET



BWR Operational Risk Review - **RED SHEET**

The RED SHEET shall be completed when work orders, testing, significant plant configuration changes, or plant maneuvers in support of maintenance are required and the normal risk review process is not performed (Step 3.8) or as directed by the SM. The RED SHEET will be completed for each shift while active work in progress. Attach completed RED SHEET to the work package until work completion.

1. Brief Description of Work or Activity: Procedure# / WO# - ________________/ ______________

2. Will this work affect (or have the potential to affect, if in close proximity) any of the following systems/components:
   - 2.1 Reactor Protection System/ Reactor Recirc / Rod Control ☐ Yes ☐ No *DIV ___
   - 2.2 PCIS/Containment/Containment Vacuum Breakers ☐ Yes ☐ No *DIV ___
   - 2.3 Safeguards (ESF, ECCS, HPCI, RCIC)/Shutdown Cooling ☐ Yes ☐ No *DIV ___
   - 2.4 Control Rod Drive ☐ Yes ☐ No
   - 2.5 Electrical Penetrations ☐ Yes ☐ No
   - 2.6 Main Steam/MSIVs/COND (Pumps, Xfer Sw., Cond-Demins) Reactor Feedwater / Heater Drains and Vents / Circ Water Pumps ☐ Yes ☐ No *DIV ___
   - 2.7 Turbine Trip Systems / EHC or Turbine Control System/ Generator and support systems (stator cooling water, bus duct cooling, voltage regulator, protective relays, lube oil, etc.) ☐ Yes ☐ No *DIV ___
   - 2.8 Diesel Generators/any 4.16KV/any 480V Distribution ☐ Yes ☐ No *DIV ___
   - 2.9 Vital 120V AC Buses, MG sets & Inverters ☐ Yes ☐ No *DIV ___
   - 2.10 Any 125VDC/250VDC (incl. battery chargers) ☐ Yes ☐ No *DIV ___
   - 2.11 Switchyard, transformer or Cap bank yard and relay room ☐ Yes ☐ No
   - 2.12 Back up Control Panel ☐ Yes ☐ No
   - 2.13 Spent Fuel Pool Inventory Control and Cooling ☐ Yes ☐ No *DIV ___
   - 2.14 Cable Raceways / Underground Cable ☐ Yes ☐ No *DIV ___
   - 2.15 Welding (Power Block Critical Areas - All) ☐ Yes ☐ No
   - 2.16 Reactor Coolant System Inventory or Cooling ☐ Yes ☐ No *DIV ___
   - 2.17 ODMI / ACMP issues ☐ Yes ☐ No

   NOTE: A Separate RED SHEET is required for each DIVISION

3. If YES to any of the above, Block 2, the following potential problems have been reviewed by Operations and any other group involved in the activity for applicability: (YES = applicable and Comp measures may be req'd) (Check Block) Initials
   - 3.1 Errors could cause loss of load/unit trip / transient ☐ Yes ☐ No _____
   - 3.2 Valving errors could cause hydraulic perturbations ☐ Yes ☐ No _____
   - 3.3 Instrument, P.C. board/component - removal/installation / PLC work Activity is performed on "In Service" or not "Bypassed" equipment / component. ☐ Yes ☐ No _____
   - 3.4 Physical limitations of work area could affect other equipment in the area that cannot be protected when work activity is in progress. ☐ Yes ☐ No _____
   - 3.5 Placing of jumpers (without banana jacks or other permanent connection), boots on relays, or lifted leads cannot be performed without contingency actions to limit risk established. ☐ Yes ☐ No _____
   - 3.6 Work on sensitive equipment or could affect sensitive equipment (inverters, Neutron Monitoring, Protection Racks, etc.) is being performed with Sensitive equipment "In Service" or Contingencies cannot be established to limit effects to "non-consequential". ☐ Yes ☐ No _____

TVA 41219 Page 1 of 2 NPG-SPP-07.3-3

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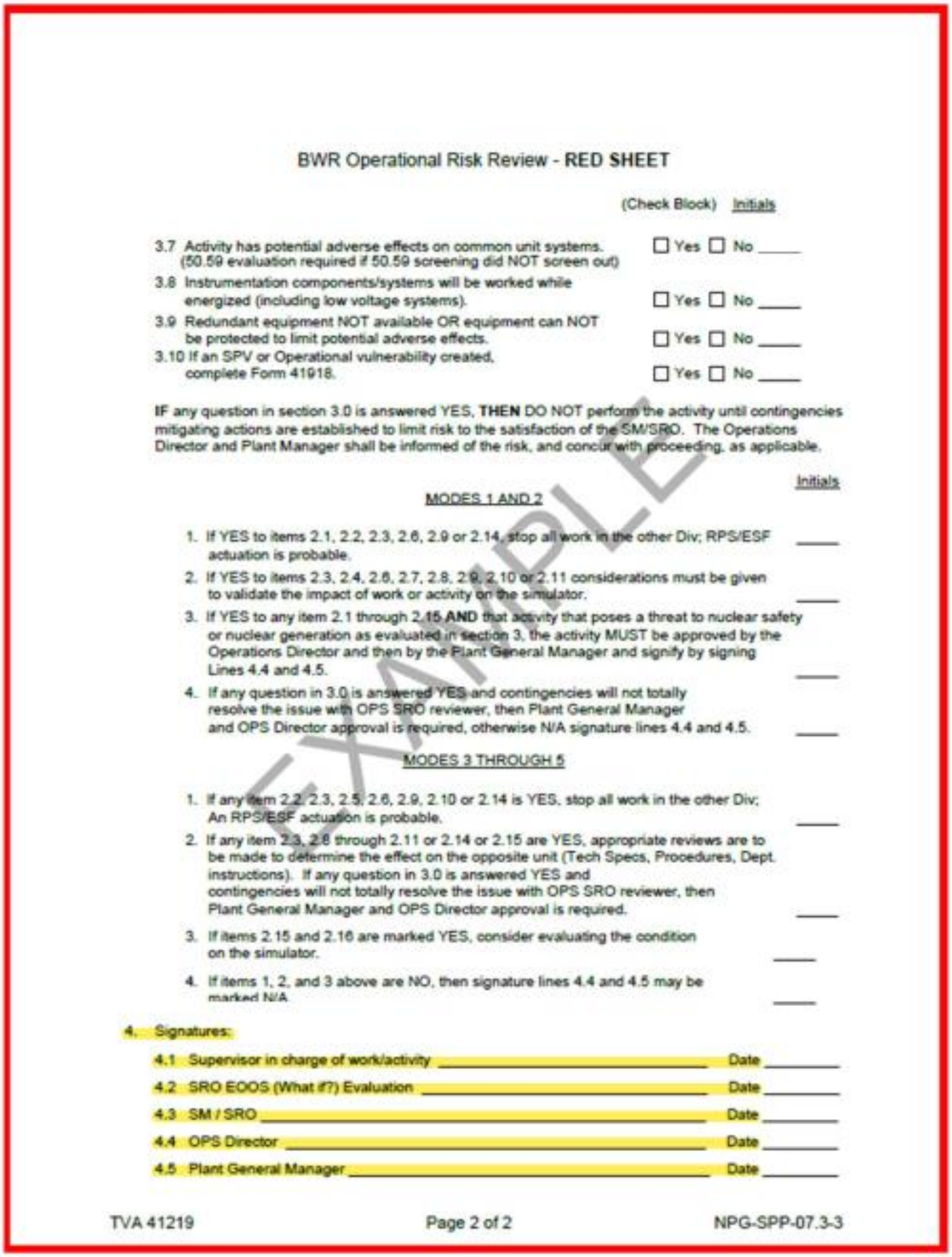
Attachment 4
(Page 2 of 2)

NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET



BWR Operational Risk Review - **RED SHEET**

(Check Block) Initials

3.7 Activity has potential adverse effects on common unit systems. (50.59 evaluation required if 50.59 screening did NOT screen out) ☐ Yes ☐ No _____

3.8 Instrumentation components/systems will be worked while energized (including low voltage systems). ☐ Yes ☐ No _____

3.9 Redundant equipment NOT available OR equipment can NOT be protected to limit potential adverse effects. ☐ Yes ☐ No _____

3.10 If an SPV or Operational vulnerability created, complete Form 41918. ☐ Yes ☐ No _____

IF any question in section 3.0 is answered YES, **THEN** DO NOT perform the activity until contingencies mitigating actions are established to limit risk to the satisfaction of the SM/SRO. The Operations Director and Plant Manager shall be informed of the risk, and concur with proceeding, as applicable.

Initials

MODES 1 AND 2

1. If YES to items 2.1, 2.2, 2.3, 2.6, 2.9 or 2.14, stop all work in the other Div; RPS/ESF actuation is probable. _____
2. If YES to items 2.3, 2.4, 2.6, 2.7, 2.8, 2.9, 2.10 or 2.11 considerations must be given to validate the impact of work or activity on the simulator. _____
3. If YES to any item 2.1 through 2.15 **AND** that activity that poses a threat to nuclear safety or nuclear generation as evaluated in section 3, the activity MUST be approved by the Operations Director and then by the Plant General Manager and signify by signing Lines 4.4 and 4.5. _____
4. If any question in 3.0 is answered YES and contingencies will not totally resolve the issue with OPS SRO reviewer, then Plant General Manager and OPS Director approval is required, otherwise N/A signature lines 4.4 and 4.5. _____

MODES 3 THROUGH 5

1. If any item 2.2, 2.3, 2.5, 2.6, 2.9, 2.10 or 2.14 is YES, stop all work in the other Div; An RPS/ESF actuation is probable. _____
2. If any item 2.3, 2.8 through 2.11 or 2.14 or 2.15 are YES, appropriate reviews are to be made to determine the effect on the opposite unit (Tech Specs, Procedures, Dept. instructions). If any question in 3.0 is answered YES and contingencies will not totally resolve the issue with OPS SRO reviewer, then Plant General Manager and OPS Director approval is required. _____
3. If items 2.15 and 2.16 are marked YES, consider evaluating the condition on the simulator. _____
4. If items 1, 2, and 3 above are NO, then signature lines 4.4 and 4.5 may be marked N/A _____

4. Signatures:

4.1 Supervisor in charge of work/activity ____________________ Date ________

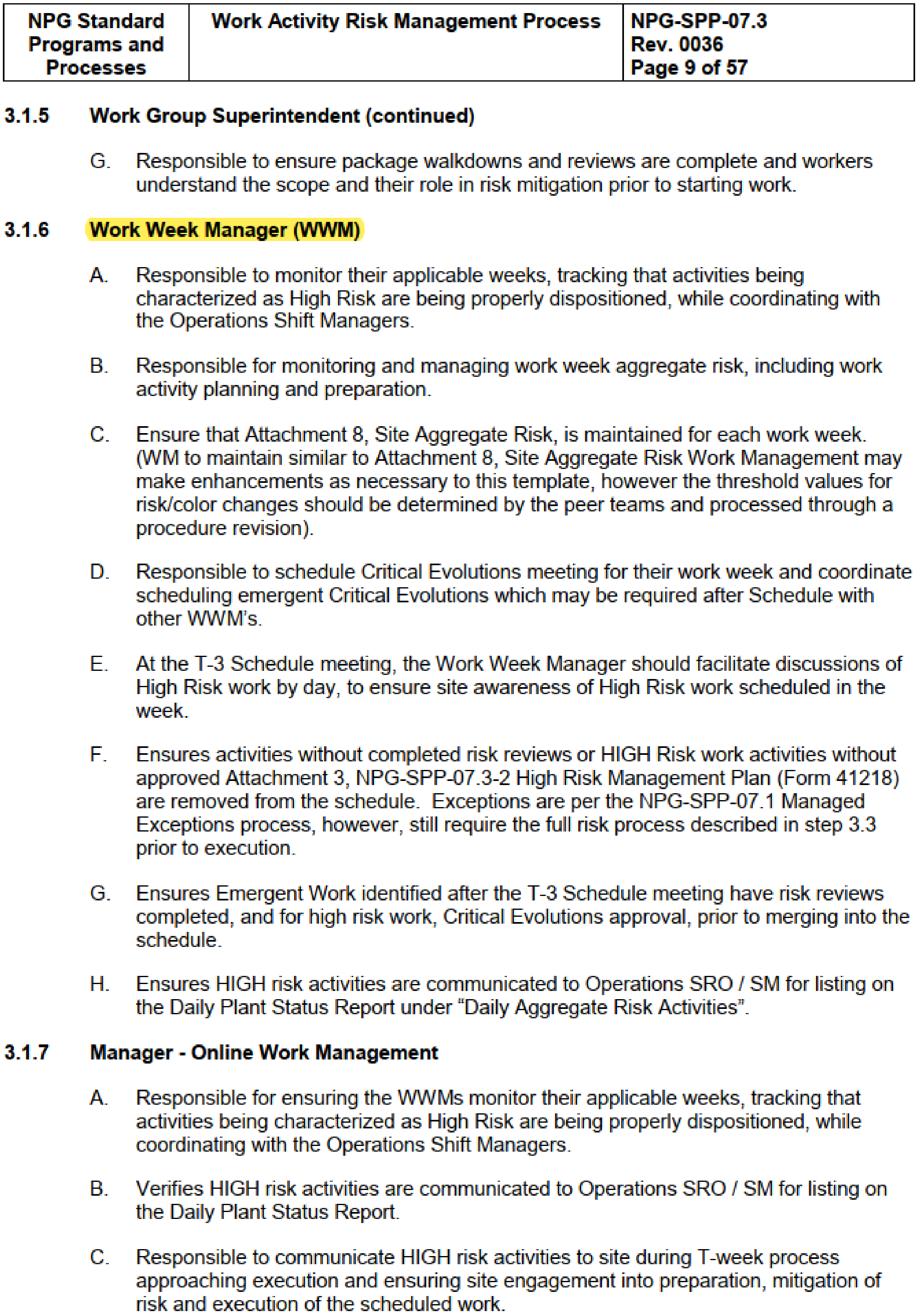
4.2 SRO EOOS (What if?) Evaluation ____________________ Date ________

4.3 SM / SRO ____________________ Date ________

4.4 OPS Director ____________________ Date ________

4.5 Plant General Manager ____________________ Date ________

Supports Distractors B(2), D(2):



NPG Standard Programs and Processes	**Work Activity Risk Management Process**	**NPG-SPP-07.3** **Rev. 0036** **Page 9 of 57**

3.1.5 Work Group Superintendent (continued)

G. Responsible to ensure package walkdowns and reviews are complete and workers understand the scope and their role in risk mitigation prior to starting work.

3.1.6 Work Week Manager (WWM)

A. Responsible to monitor their applicable weeks, tracking that activities being characterized as High Risk are being properly dispositioned, while coordinating with the Operations Shift Managers.

B. Responsible for monitoring and managing work week aggregate risk, including work activity planning and preparation.

C. Ensure that Attachment 8, Site Aggregate Risk, is maintained for each work week. (WM to maintain similar to Attachment 8, Site Aggregate Risk Work Management may make enhancements as necessary to this template, however the threshold values for risk/color changes should be determined by the peer teams and processed through a procedure revision).

D. Responsible to schedule Critical Evolutions meeting for their work week and coordinate scheduling emergent Critical Evolutions which may be required after Schedule with other WWM's.

E. At the T-3 Schedule meeting, the Work Week Manager should facilitate discussions of High Risk work by day, to ensure site awareness of High Risk work scheduled in the week.

F. Ensures activities without completed risk reviews or HIGH Risk work activities without approved Attachment 3, NPG-SPP-07.3-2 High Risk Management Plan (Form 41218) are removed from the schedule. Exceptions are per the NPG-SPP-07.1 Managed Exceptions process, however, still require the full risk process described in step 3.3 prior to execution.

G. Ensures Emergent Work identified after the T-3 Schedule meeting have risk reviews completed, and for high risk work, Critical Evolutions approval, prior to merging into the schedule.

H. Ensures HIGH risk activities are communicated to Operations SRO / SM for listing on the Daily Plant Status Report under "Daily Aggregate Risk Activities".

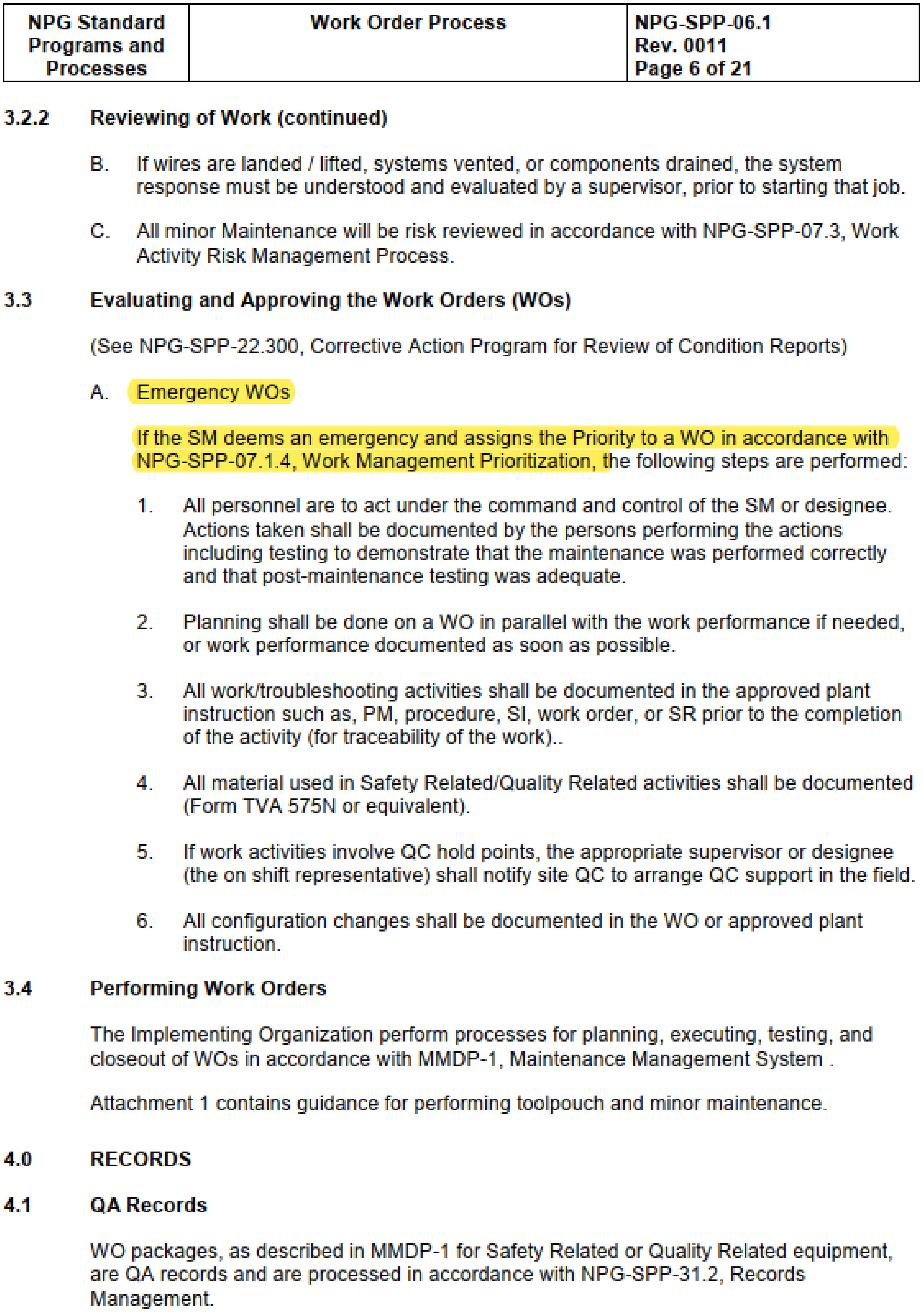
3.1.7 Manager - Online Work Management

A. Responsible for ensuring the WWMs monitor their applicable weeks, tracking that activities being characterized as High Risk are being properly dispositioned, while coordinating with the Operations Shift Managers.

B. Verifies HIGH risk activities are communicated to Operations SRO / SM for listing on the Daily Plant Status Report.

C. Responsible to communicate HIGH risk activities to site during T-week process approaching execution and ensuring site engagement into preparation, mitigation of risk and execution of the scheduled work.

Excerpt from NPG-SPP-06.1:



NPG Standard Programs and Processes	Work Order Process	NPG-SPP-06.1 Rev. 0011 Page 6 of 21

3.2.2 Reviewing of Work (continued)

B. If wires are landed / lifted, systems vented, or components drained, the system response must be understood and evaluated by a supervisor, prior to starting that job.

C. All minor Maintenance will be risk reviewed in accordance with NPG-SPP-07.3, Work Activity Risk Management Process.

3.3 Evaluating and Approving the Work Orders (WOs)

(See NPG-SPP-22.300, Corrective Action Program for Review of Condition Reports)

A. Emergency WOs

If the SM deems an emergency and assigns the Priority to a WO in accordance with NPG-SPP-07.1.4, Work Management Prioritization, the following steps are performed:

1. All personnel are to act under the command and control of the SM or designee. Actions taken shall be documented by the persons performing the actions including testing to demonstrate that the maintenance was performed correctly and that post-maintenance testing was adequate.
2. Planning shall be done on a WO in parallel with the work performance if needed, or work performance documented as soon as possible.
3. All work/troubleshooting activities shall be documented in the approved plant instruction such as, PM, procedure, SI, work order, or SR prior to the completion of the activity (for traceability of the work)..
4. All material used in Safety Related/Quality Related activities shall be documented (Form TVA 575N or equivalent).
5. If work activities involve QC hold points, the appropriate supervisor or designee (the on shift representative) shall notify site QC to arrange QC support in the field.
6. All configuration changes shall be documented in the WO or approved plant instruction.

3.4 Performing Work Orders

The Implementing Organization perform processes for planning, executing, testing, and closeout of WOs in accordance with MMDP-1, Maintenance Management System .

Attachment 1 contains guidance for performing toolpouch and minor maintenance.

4.0 RECORDS

4.1 QA Records

WO packages, as described in MMDP-1 for Safety Related or Quality Related equipment, are QA records and are processed in accordance with NPG-SPP-31.2, Records Management.

Examination Outline Cross-reference:

Radiation Control

G2.3.6 **(10CFR 55.43.4 – SRO Only)**

Ability to approve liquid or gaseous release permits

Level	RO	SRO
Tier #	----	3
Group #	----	----
K/A #	G2.3.6	
Importance Rating	----	3.8

Proposed Question: **# 97**

In accordance with 2-OI-27, Condenser Circulating Water (CCW) System, 2-FCV-77-61, RADWASTE DISCHARGE VALVE to Unit 2 CCW discharge tunnel cannot be opened without a minimum of 2 CCW pumps operating on ___**(1)**___.

In accordance with 0-SI-4.8.A.1-1, Liquid Release Permit, if 0-RM-90-130, RADWASTE EFFLUENT RADIATION MONITOR, is found **NONFUNCTIONAL** during a release, then the release ___**(2)**___.

A. (1) Unit 2
(2) is no longer valid and a new release permit is required

B. (1) Unit 2
(2) may continue if an independent verification of the valve lineup and release calculation is immediately performed

C. (1) Any Unit
(2) is no longer valid and a new release permit is required

D. (1) Any Unit
(2) may continue if an independent verification of the valve lineup and release calculation is immediately performed

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 2-OI-27, Condenser Circulating Water (CCW) System, 2-FCV-77-61, RADWASTE DISCHARGE VALVE automatically closes or is prevented from opening if less than two Unit 2 CCW Pumps (any 2) are running or Gate 1A is less than full open. This is specific to a Unit 2 release in this case, which requires the previous mentioned two interlocks be met. This helps to ensure adequate mixing for proper dilution and with Gate 1A not full open, ensures flow exiting the plant is directed in the correct direction. For second part, in accordance with 0-SI-4.8.A.1-1, Liquid Release Permit, if 0-RM-90-130, RADWASTE EFFLUENT RADIATION MONITOR, is NONFUNCTIONAL per ODCM criteria, during a release, the release is no longer valid and a new procedure is required. Additionally, functional status from a least one Unit SRO must be authorized for the release pathway as a prerequisite action.

B INCORRECT: First part is correct *(See A).* Second part is incorrect, but plausible in that if 0-RM-90-130 is NONFUNCTIONAL per ODCM criteria, releases can continue with isolation logic bypassed provided that two qualified Chemistry personnel independently check release rate calculations and two qualified Operations personnel perform independent verification of valve line-ups.

C INCORRECT: First part is incorrect but plausible in that 2 CCW pumps are required. Candidates often confuse this concept. All 3 Units CCW system take suction from a common source. Much of the possible discharge paths are common as well. Since the purpose of requiring 2 pumps in service is for dilution purposes, it is reasonable to conclude that any 2 pumps would allow the release. 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 a SRO's ability and responsibility in approving radiological releases. 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 to strictly recalling facts related to approval requirements for liquid releases.

Technical Reference(s): 0-SI-4.8.A.1-1, Rev. 102 (Attach if not previously provided)
2-OI-27 Rev. 99

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.050, Obj. 9 (As available)

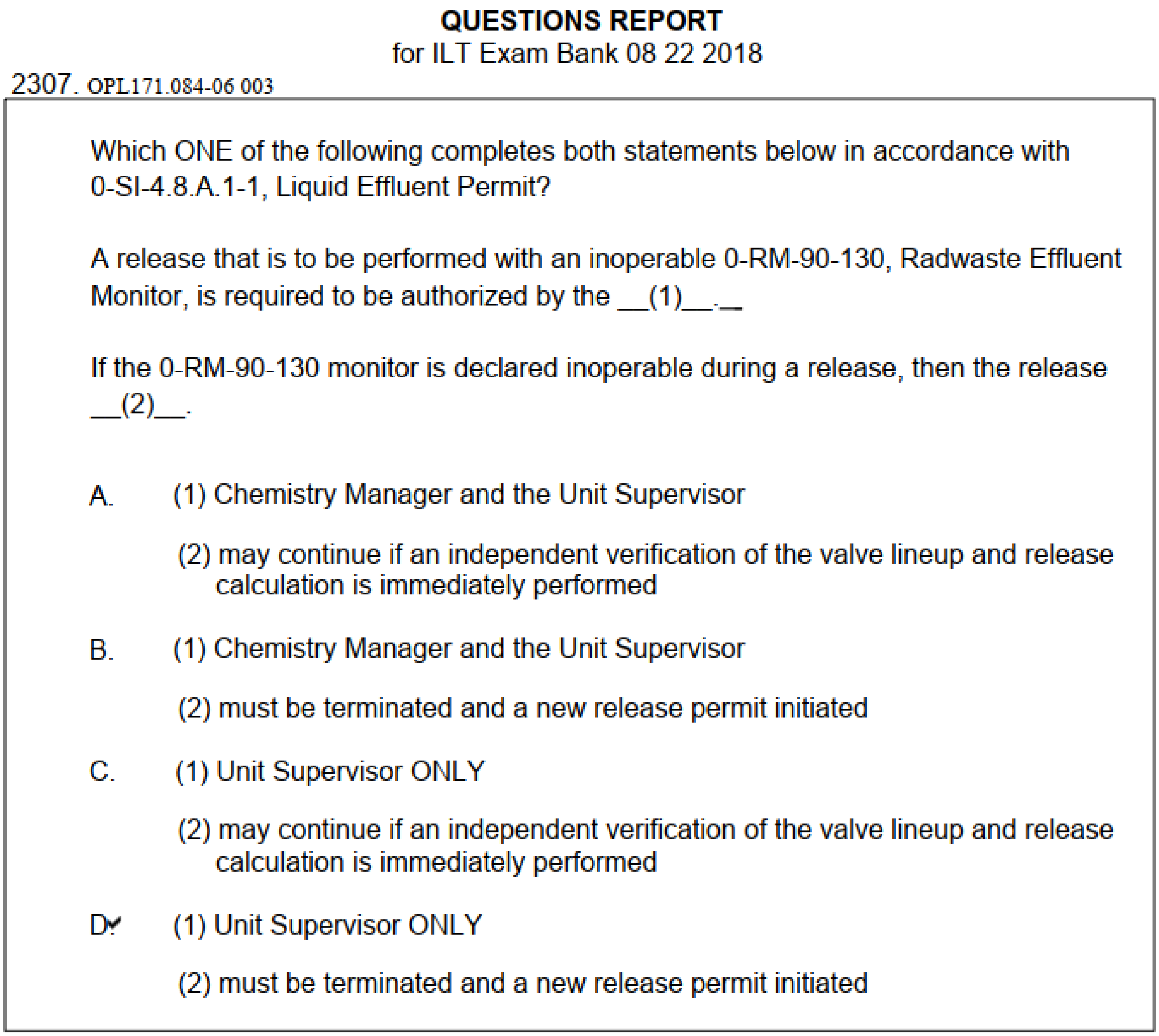
Question Source:	Bank #		
	Modified Bank #	OPL171.084-06 003 #2307	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

2307. OPL171.084-06 003

Which ONE of the following completes both statements below in accordance with 0-SI-4.8.A.1-1, Liquid Effluent Permit?

A release that is to be performed with an inoperable 0-RM-90-130, Radwaste Effluent Monitor, is required to be authorized by the __(1)__._

If the 0-RM-90-130 monitor is declared inoperable during a release, then the release __(2)__.

A. (1) Chemistry Manager and the Unit Supervisor

(2) may continue if an independent verification of the valve lineup and release calculation is immediately performed

B. (1) Chemistry Manager and the Unit Supervisor

(2) must be terminated and a new release permit initiated

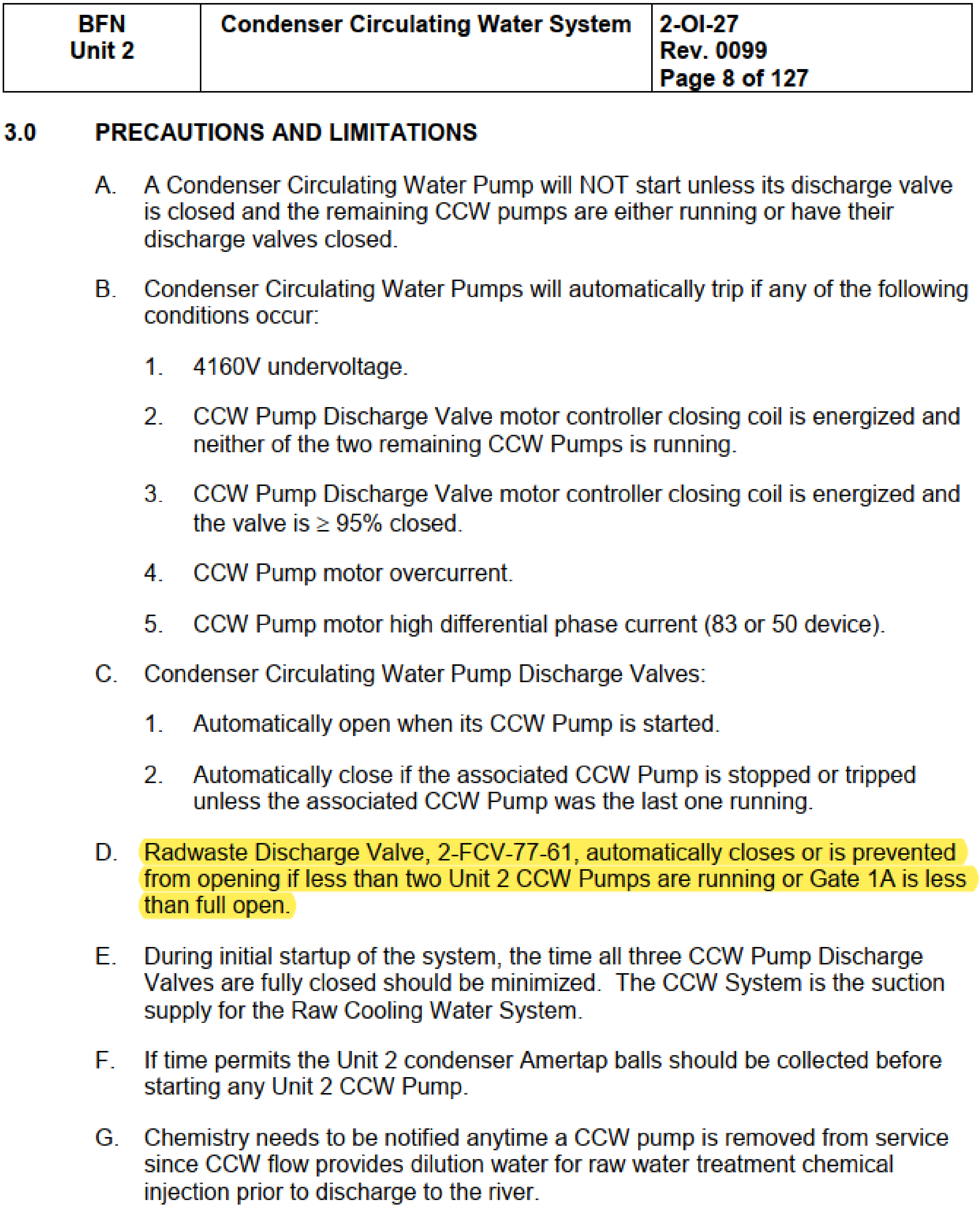
C. (1) Unit Supervisor ONLY

(2) may continue if an independent verification of the valve lineup and release calculation is immediately performed

D✓ (1) Unit Supervisor ONLY

(2) must be terminated and a new release permit initiated

Excerpt from 2-OI-27:



BFN Unit 2	**Condenser Circulating Water System**	**2-OI-27 Rev. 0099 Page 8 of 127**

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.

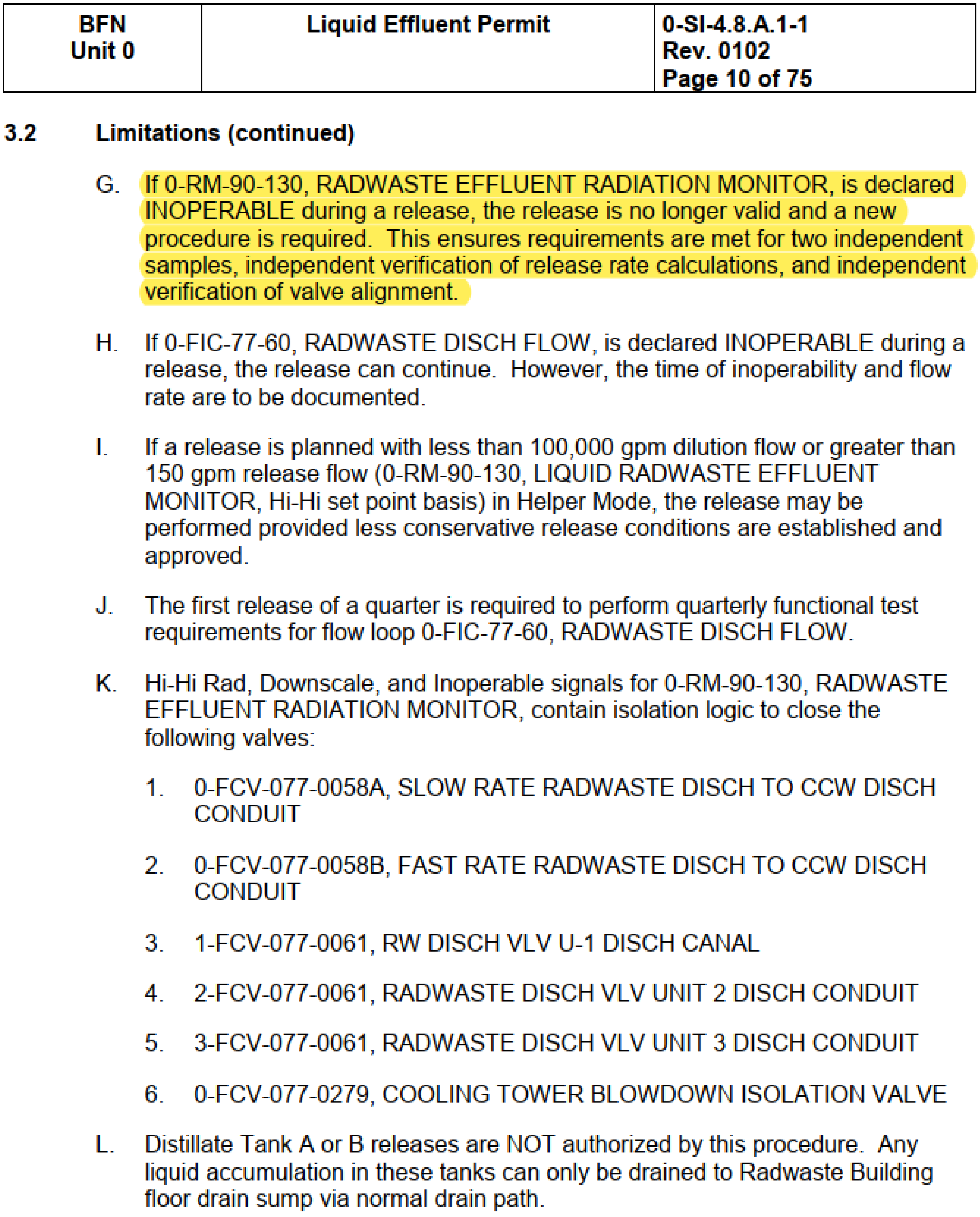
D. Radwaste Discharge Valve, 2-FCV-77-61, automatically closes or is prevented from opening if less than two Unit 2 CCW Pumps are running or Gate 1A is less than full open.

E. During initial startup of the system, the time all three CCW Pump Discharge Valves are fully closed should be minimized. The CCW System is the suction supply for the Raw Cooling Water System.

F. If time permits the Unit 2 condenser Amertap balls should be collected before starting any Unit 2 CCW Pump.

G. Chemistry needs to be notified anytime a CCW pump is removed from service since CCW flow provides dilution water for raw water treatment chemical injection prior to discharge to the river.

Excerpts from 0-SI-4.8.A.1-1:



BFN **Unit 0**	**Liquid Effluent Permit**	**0-SI-4.8.A.1-1** **Rev. 0102** **Page 10 of 75**

3.2 Limitations (continued)

G. If 0-RM-90-130, RADWASTE EFFLUENT RADIATION MONITOR, is declared INOPERABLE during a release, the release is no longer valid and a new procedure is required. This ensures requirements are met for two independent samples, independent verification of release rate calculations, and independent verification of valve alignment.

H. If 0-FIC-77-60, RADWASTE DISCH FLOW, is declared INOPERABLE during a release, the release can continue. However, the time of inoperability and flow rate are to be documented.

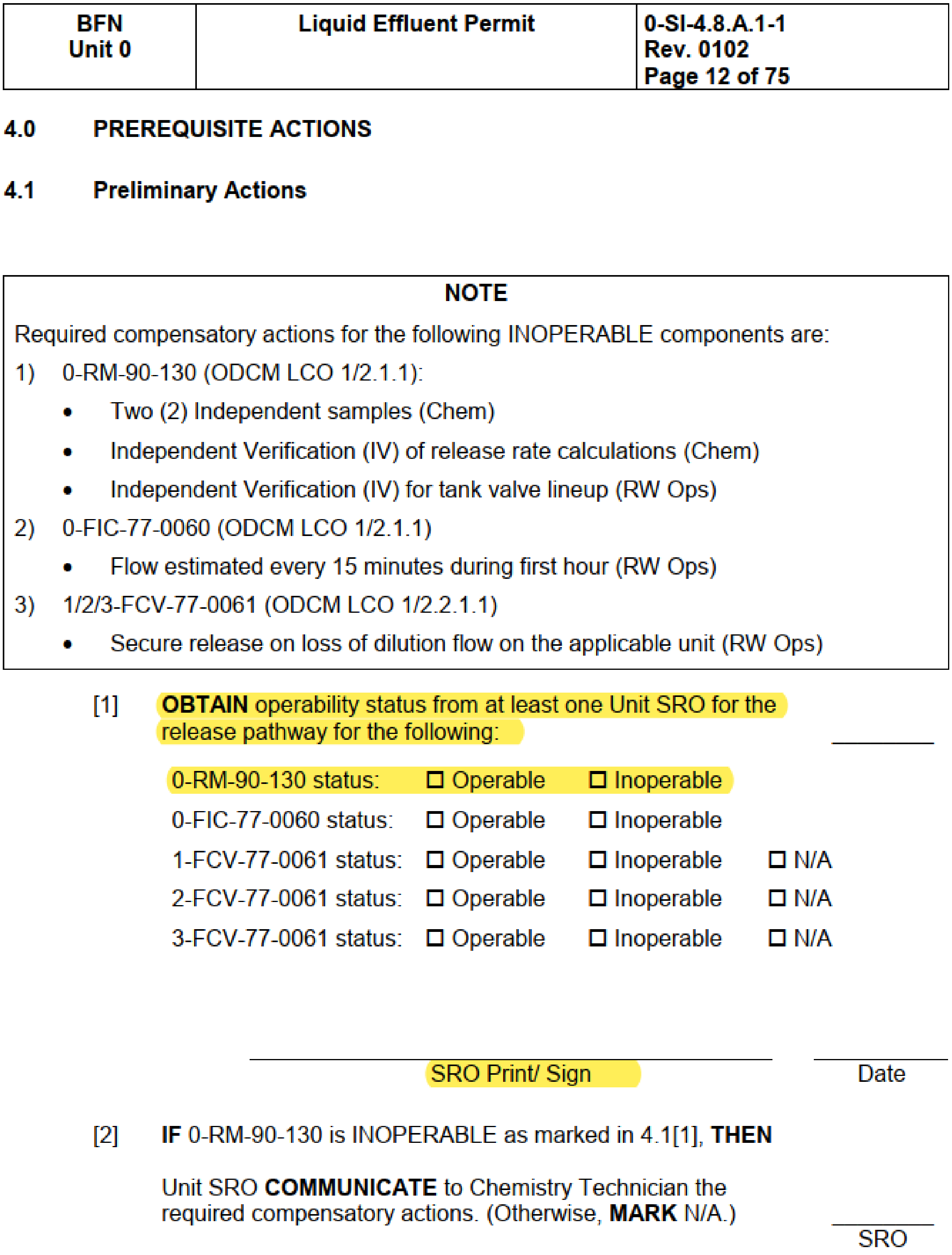
I. If a release is planned with less than 100,000 gpm dilution flow or greater than 150 gpm release flow (0-RM-90-130, LIQUID RADWASTE EFFLUENT MONITOR, Hi-Hi set point basis) in Helper Mode, the release may be performed provided less conservative release conditions are established and approved.

J. The first release of a quarter is required to perform quarterly functional test requirements for flow loop 0-FIC-77-60, RADWASTE DISCH FLOW.

K. Hi-Hi Rad, Downscale, and Inoperable signals for 0-RM-90-130, RADWASTE EFFLUENT RADIATION MONITOR, contain isolation logic to close the following valves:

1. 0-FCV-077-0058A, SLOW RATE RADWASTE DISCH TO CCW DISCH CONDUIT
2. 0-FCV-077-0058B, FAST RATE RADWASTE DISCH TO CCW DISCH CONDUIT
3. 1-FCV-077-0061, RW DISCH VLV U-1 DISCH CANAL
4. 2-FCV-077-0061, RADWASTE DISCH VLV UNIT 2 DISCH CONDUIT
5. 3-FCV-077-0061, RADWASTE DISCH VLV UNIT 3 DISCH CONDUIT
6. 0-FCV-077-0279, COOLING TOWER BLOWDOWN ISOLATION VALVE

L. Distillate Tank A or B releases are NOT authorized by this procedure. Any liquid accumulation in these tanks can only be drained to Radwaste Building floor drain sump via normal drain path.



BFN Unit 0	**Liquid Effluent Permit**	**0-SI-4.8.A.1-1 Rev. 0102 Page 12 of 75**

4.0 PREREQUISITE ACTIONS

4.1 Preliminary Actions

NOTE

Required compensatory actions for the following INOPERABLE components are:

1) 0-RM-90-130 (ODCM LCO 1/2.1.1):
   - Two (2) Independent samples (Chem)
   - Independent Verification (IV) of release rate calculations (Chem)
   - Independent Verification (IV) for tank valve lineup (RW Ops)
2) 0-FIC-77-0060 (ODCM LCO 1/2.1.1)
   - Flow estimated every 15 minutes during first hour (RW Ops)
3) 1/2/3-FCV-77-0061 (ODCM LCO 1/2.2.1.1)
   - Secure release on loss of dilution flow on the applicable unit (RW Ops)

[1] **OBTAIN** operability status from at least one Unit SRO for the release pathway for the following:

0-RM-90-130 status: ☐ Operable ☐ Inoperable
0-FIC-77-0060 status: ☐ Operable ☐ Inoperable
1-FCV-77-0061 status: ☐ Operable ☐ Inoperable ☐ N/A
2-FCV-77-0061 status: ☐ Operable ☐ Inoperable ☐ N/A
3-FCV-77-0061 status: ☐ Operable ☐ Inoperable ☐ N/A

SRO Print/ Sign Date

[2] **IF** 0-RM-90-130 is INOPERABLE as marked in 4.1[1], **THEN**

Unit SRO **COMMUNICATE** to Chemistry Technician the required compensatory actions. (Otherwise, **MARK** N/A.)

SRO

Supports Distractors B(2), D(2):

BFN **Unit 0**	**Liquid Effluent Permit**	**0-SI-4.8.A.1-1** **Rev. 0102** **Page 9 of 75**

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

None

3.2 Limitations

A. An administrative limit more conservative than that specified by 10 CFR 20 for liquid releases provides an additional margin of safety in the event of unforeseen problems. It is defined as the Sum of the Effluent Concentration Limit (ECL) Fraction being less than or equal to 1.0. The administrative limit can be waived only with approval of the Chemistry Manager or designee.

B. Unit SRO is required to authorize all releases for affected unit.

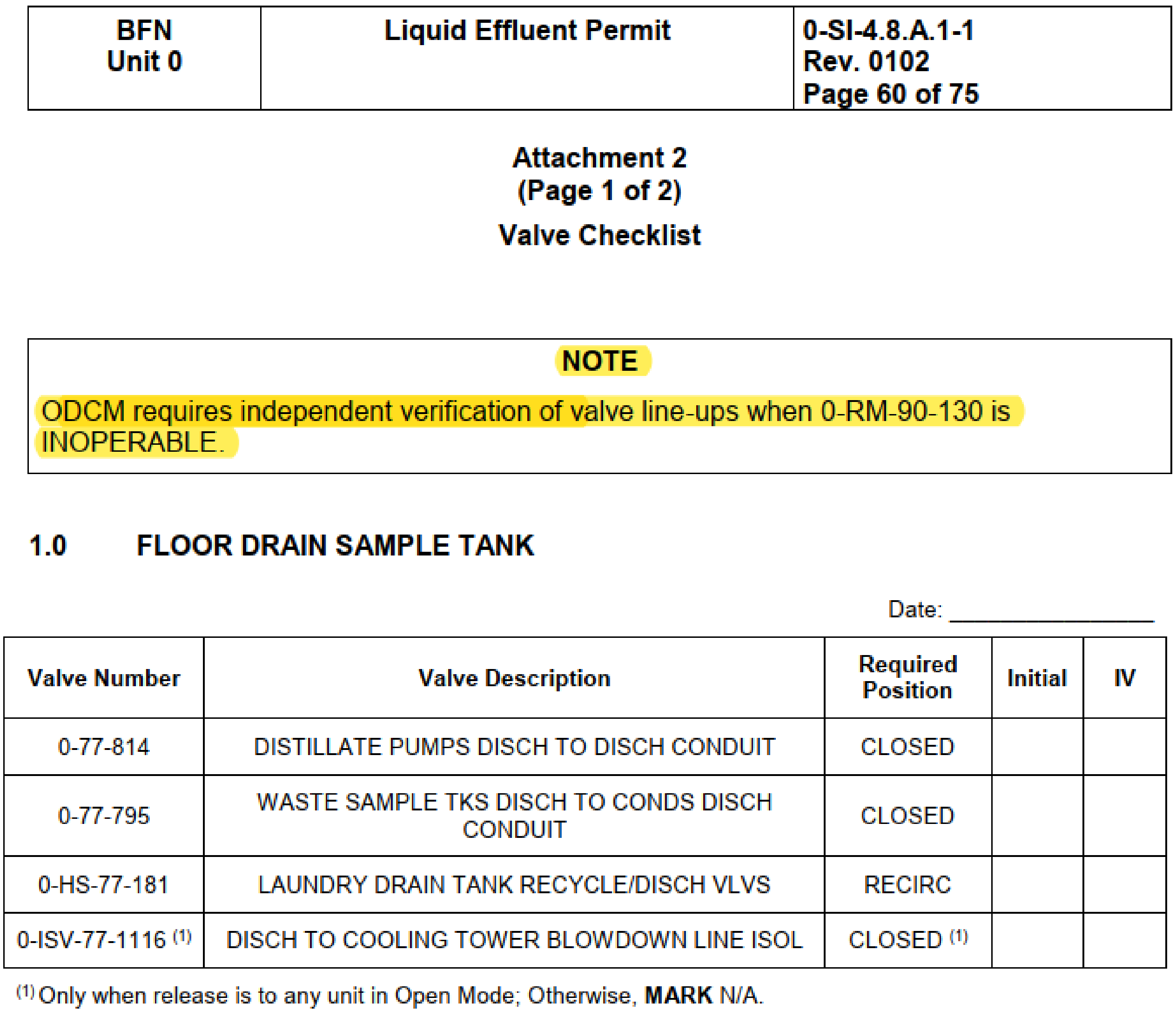
C. Sample analysis count is required to be started within one hour of sample collection to ensure Lower Limit of Detection (LLD) requirements are met.

D. NPDES pH analysis is required to be performed within 15 minutes of sample collection.

E. If 0-RM-90-130, RADWASTE EFFLUENT RADIATION MONITOR, is Inoperable per ODCM criteria, releases can continue with isolation logic bypassed provided:

1. Two independent samples are analyzed,
2. Two qualified Chemistry personnel independently check release rate calculations,
3. Two qualified Operations personnel check the valve alignment per Attachment 2, Valve Checklist,
4. A Unit SRO authorizes bypassing of isolation logic for 0-RM-90-130, RADWASTE EFFLUENT RADIATION MONITOR, and
5. Isolation jumpers are to be removed after each release.

F. During performance of this procedure, if Plant Release Mode (Open or Helper) changes or Condenser Circulating Water Pumps (CCWPs) are removed from service causing dilution flow to decrease, the release is to be terminated until a new procedure is initiated or until original dilution flow is restored.



BFN **Unit 0**	**Liquid Effluent Permit**	**0-SI-4.8.A.1-1** **Rev. 0102** **Page 60 of 75**

Attachment 2
(Page 1 of 2)

Valve Checklist

NOTE

ODCM requires independent verification of valve line-ups when 0-RM-90-130 is INOPERABLE.

1.0 FLOOR DRAIN SAMPLE TANK

Date: ________________

Valve Number	Valve Description	Required Position	Initial	IV
0-77-814	DISTILLATE PUMPS DISCH TO DISCH CONDUIT	CLOSED		
0-77-795	WASTE SAMPLE TKS DISCH TO CONDS DISCH CONDUIT	CLOSED		
0-HS-77-181	LAUNDRY DRAIN TANK RECYCLE/DISCH VLVS	RECIRC		
0-ISV-77-1116 (1)	DISCH TO COOLING TOWER BLOWDOWN LINE ISOL	CLOSED (1)		

(1) Only when release is to any unit in Open Mode; Otherwise, **MARK** N/A.

Examination Outline Cross-reference:	Level	RO	SRO
Radiation Control	Tier #	------	3
G2.3.14 (10CFR 55.43.4 – SRO Only)	Group #	------	------
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits.	K/A #	G2.3.14	
	Importance Rating	------	3.8

Proposed Question: **# 98**

The **MINIMUM** required Emergency Classification for entering 0-EOI-4, Radioactivity Release Control, is a/an ___**(1)**___.

During implementation of 0-EOI-4 and in accordance with the EOI Program Manual Bases, operation of Turbine Building Ventilation preserves ___**(2)**____.

A. (1) NOUE
(2) equipment operability

B. (1) NOUE
(2) building accessibility

C. (1) ALERT
(2) equipment operability

D. (1) ALERT
(2) building accessibility

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that there are multiple radiological levels from indications that are also a NOUE. The same instruments have thresholds for declaring an ALERT as well. Both refer to liquid release levels, and the WRGERMS radiation monitor. Due to the overlap of monitored parameters, it is plausible think an NOUE especially without a provided reference. Second part is incorrect but plausible in that in accordance with EOI-3, Secondary Containment Control, equipment operability is a concern in the Reactor Building.

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-EOI-4, Radioactive Release Control, the entry condition is gaseous offsite radioactivity release rate above that requiring an ALERT IC RA1. For second part, in accordance with EOIPM Section 0-V-K, Radioactivity Release Control Bases, operation of the Turbine Building Ventilation System preserves Turbine Building accessibility.

SRO Level Justification: Tests the candidate's knowledge concerning control of radiation releases related to the restoration of Turbine Building ventilation during the execution of EOI-4, Radioactivity Releases using Bases knowledge 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(K), Rev. 0	(Attach if not previously provided)
	EOIPM 0-V(J), Rev. 1	
	0-EOI-4, Rev 9	

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.204, Obj. 12 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 21-04 #98	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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: **# 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) building accessibility
(2) monitored

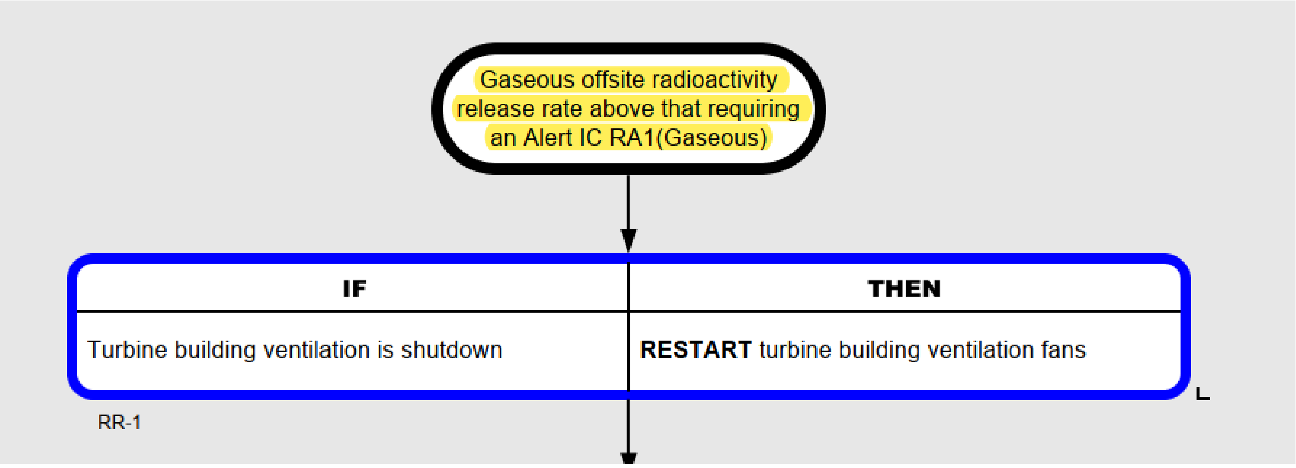
B. (1) building accessibility
(2) unmonitored

C. (1) equipment operability
(2) monitored

D. (1) equipment operability
(2) unmonitored

Proposed Answer: **A**

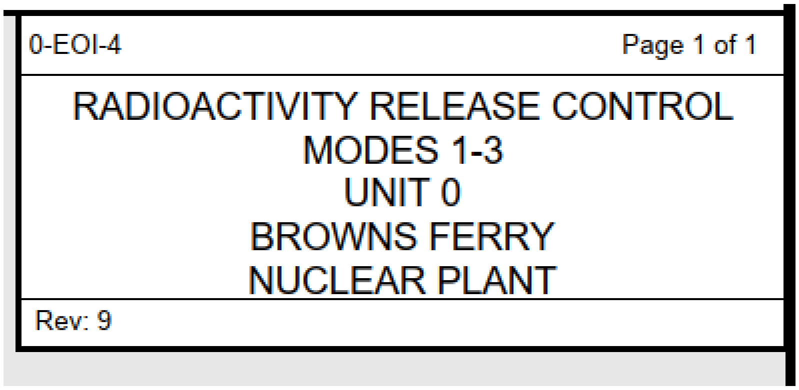
Excerpt from 0-EOI-4: Listing the entry condition



Gaseous offsite radioactivity release rate above that requiring an Alert IC RA1(Gaseous)

IF	THEN
Turbine building ventilation is shutdown	**RESTART** turbine building ventilation fans

RR-1



0-EOI-4 Page 1 of 1

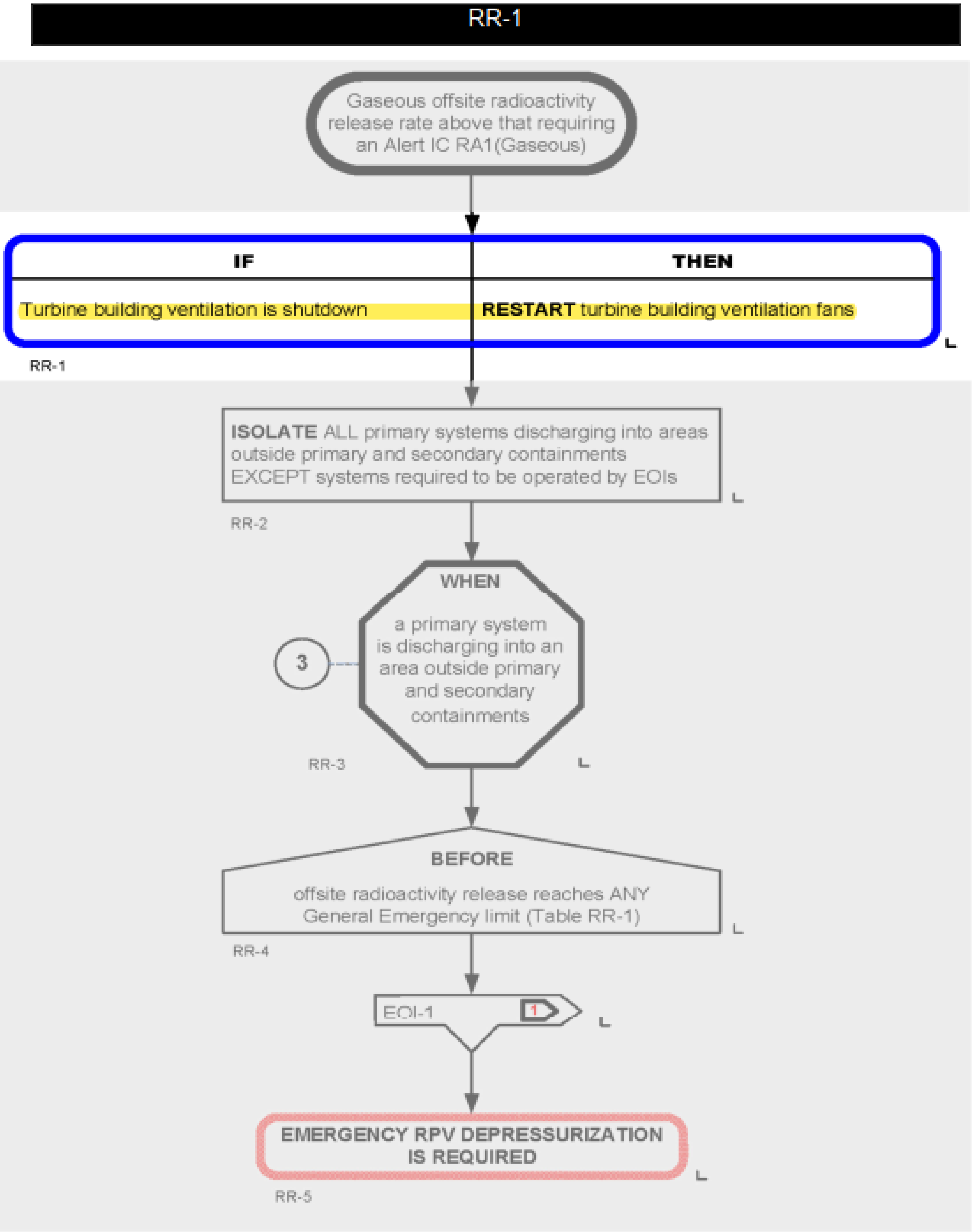
RADIOACTIVITY RELEASE CONTROL
MODES 1-3
UNIT 0
BROWNS FERRY
NUCLEAR PLANT

Rev: 9

Excerpts from EOIPM 0-V(K):

BFN **Unit 0**	**EOI-4, Radioactivity Release Control Bases**	**EOIPM Section 0-V(K)** **Rev. 0000** **Page 8 of 17**

1.0 EOI-4, RADIOACTIVITY RELEASE CONTROL BASES (continued)


RR-1

Gaseous offsite radioactivity release rate above that requiring an Alert IC RA1(Gaseous)

IF	THEN
Turbine building ventilation is shutdown	**RESTART** turbine building ventilation fans

RR-1

ISOLATE ALL primary systems discharging into areas outside primary and secondary containments EXCEPT systems required to be operated by EOIs

RR-2

WHEN a primary system is discharging into an area outside primary and secondary containments

RR-3

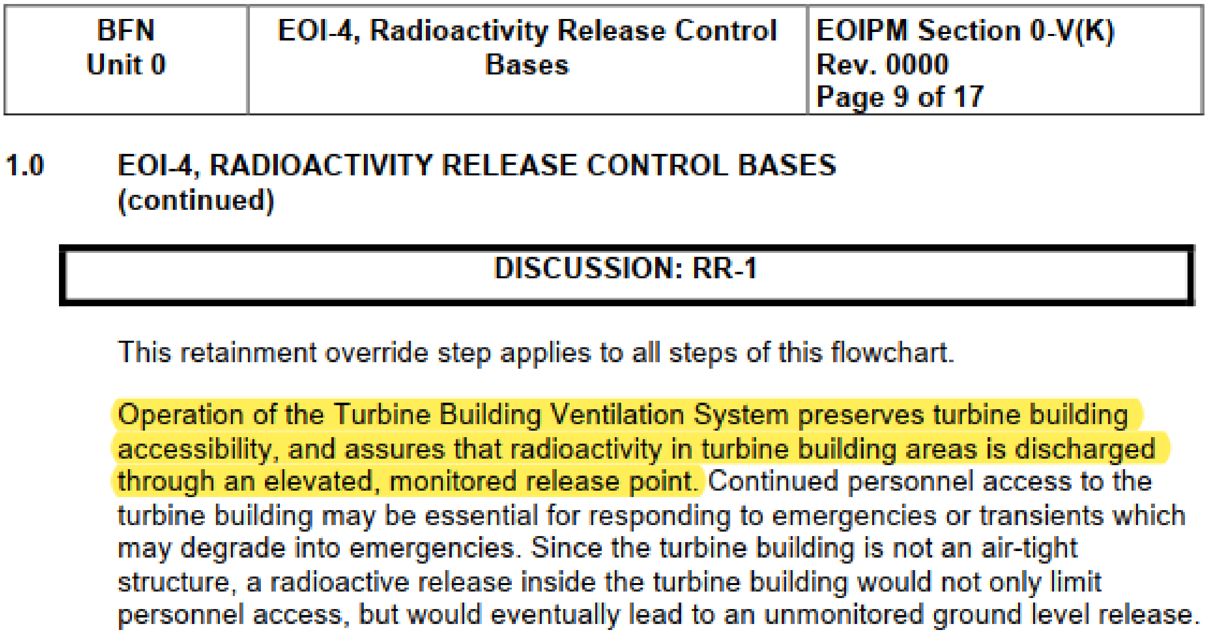
BEFORE offsite radioactivity release reaches ANY General Emergency limit (Table RR-1)

RR-4

EOI-1

EMERGENCY RPV DEPRESSURIZATION IS REQUIRED

RR-5



BFN Unit 0	EOI-4, Radioactivity Release Control Bases	EOIPM Section 0-V(K) Rev. 0000 Page 9 of 17

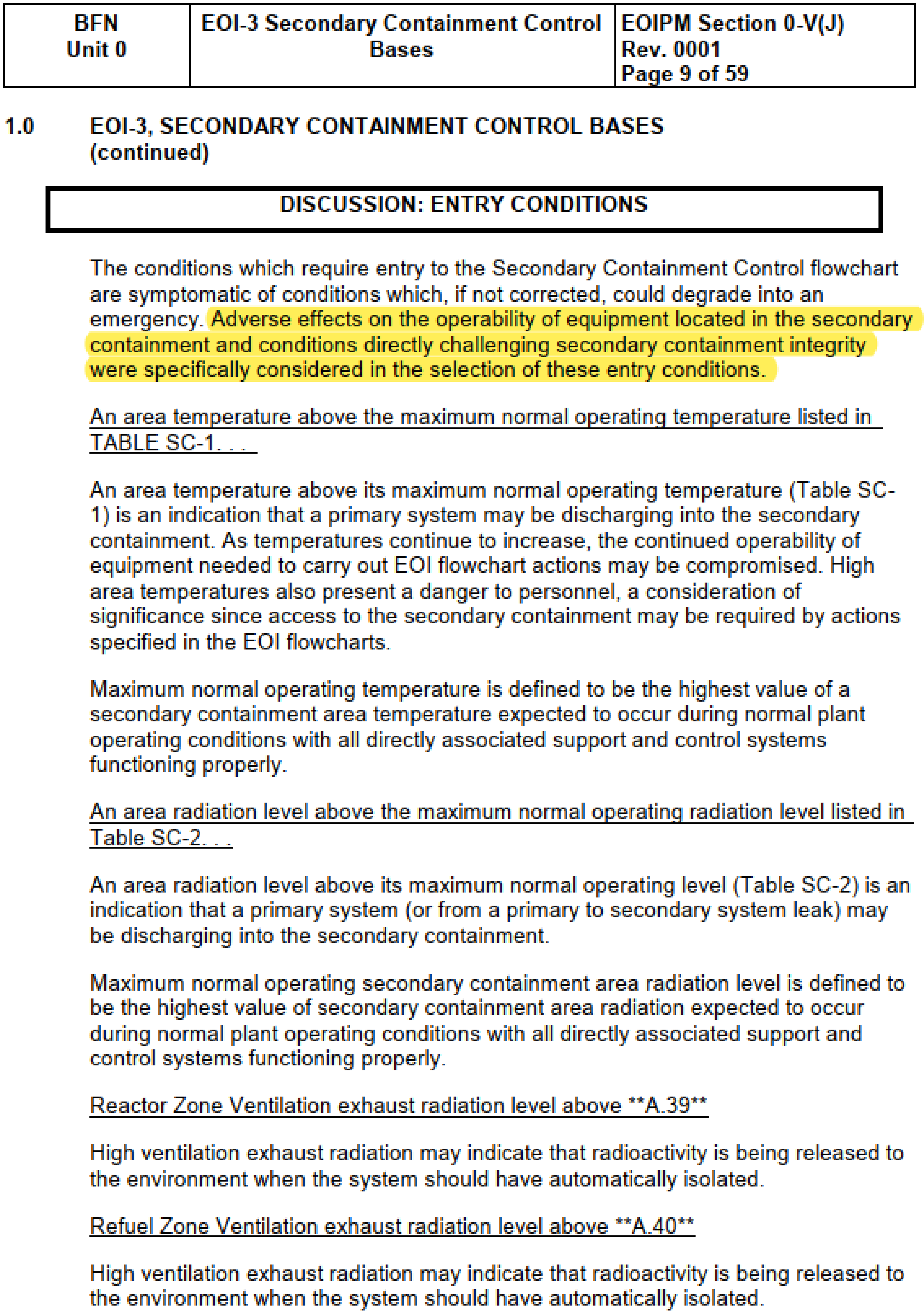
1.0 EOI-4, RADIOACTIVITY RELEASE CONTROL BASES (continued)

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 EOIPM Section 0-V(J): Supports Distractors A(2), C(2)



BFN Unit 0	EOI-3 Secondary Containment Control Bases	EOIPM Section 0-V(J) Rev. 0001 Page 9 of 59

1.0 EOI-3, SECONDARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: ENTRY CONDITIONS

The conditions which require entry to the Secondary Containment Control flowchart are symptomatic of conditions which, if not corrected, could degrade into an emergency. Adverse effects on the operability of equipment located in the secondary containment and conditions directly challenging secondary containment integrity were specifically considered in the selection of these entry conditions.

An area temperature above the maximum normal operating temperature listed in TABLE SC-1. . .

An area temperature above its maximum normal operating temperature (Table SC-1) is an indication that a primary system may be discharging into the secondary containment. As temperatures continue to increase, the continued operability of equipment needed to carry out EOI flowchart actions may be compromised. High area temperatures also present a danger to personnel, a consideration of significance since access to the secondary containment may be required by actions specified in the EOI flowcharts.

Maximum normal operating temperature is defined to be the highest value of a secondary containment area temperature expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

An area radiation level above the maximum normal operating radiation level listed in Table SC-2. . .

An area radiation level above its maximum normal operating level (Table SC-2) is an indication that a primary system (or from a primary to secondary system leak) may be discharging into the secondary containment.

Maximum normal operating secondary containment area radiation level is defined to be the highest value of secondary containment area radiation expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

Reactor Zone Ventilation exhaust radiation level above **A.39**

High ventilation exhaust radiation may indicate that radioactivity is being released to the environment when the system should have automatically isolated.

Refuel Zone Ventilation exhaust radiation level above **A.40**

High ventilation exhaust radiation may indicate that radioactivity is being released to the environment when the system should have automatically isolated.

Examination Outline Cross-reference:	Level	RO	SRO
Emergency Procedures / Plan **G2.4.28 (10CFR 55.43.5 – SRO Only)**	Tier #	------	3
Knowledge of the procedures relating to a security event (ensure that the test item contains no safeguards information).	Group #	------	------
	K/A #	G2.4.28	
	Importance Rating	------	4.1

Proposed Question: **# 99**

Site Security personnel are currently engaged with a hostile force in the Turbine Building breezeway and a TVA spokesman has made an official announcement to the public and news agencies regarding the ongoing event.

Given the condition above and in accordance with security procedures, site personnel will be notified to **(1)** .

In accordance with NPG-SPP-03.5, Regulatory Reporting Requirements, the **FIRST** report is required to be made to the NRC within **(2)** .

[**REFERENCE PROVIDED]**

A. (1) evacuate
(2) 1 hour

B. (1) evacuate
(2) 4 hours

C. (1) shelter in place
(2) 1 hour

D. (1) shelter in place
(2) 4 hours

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that in accordance with EPIP-4, SITE AREA EMERGENCY, evacuation can be conducted in other situations just not when hostile action is in progress in the protected area. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that NPG-SPP-03.5 Attachment 1, Section 3.1.C.4 states that 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. Although correct, a 4 hour report is not the first report required to be made to the NRC.

C **CORRECT:** *(See attached)* (In accordance with EPIP-1, Emergency Classification Procedure, see BFN Exam team for safeguards information justification to specific BFN security procedures). In accordance with EPIP-4, SITE AREA EMERGENCY, Attachment 2 - with a Security Event Imminent Threat, personnel within the Protected Area (implied by Turbine Building breezeway location) will shelter in place or remain sheltered.
For second part, in accordance with NPG-SPP-03.5, Regulatory Reporting Requirements, Attachment 1, Section 3.1.B.2, states the criteria requiring an NRC 1-hour notification is if the Declaration of any of the Emergency classes specified in the licensee's approved Emergency Plan has been met. Given that site security personnel are currently engaged with a hostile force in the Turbine Building breezeway, then in accordance with EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3, a Site Area Emergency – HS1 exist in the Protected Area.

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

SRO Level Justification: Test candidate's knowledge regarding security event response and the related NRC Reporting Requirements. SRO only because of link to 10 CFR 55.43 (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.

Technical Reference(s):	NPG-SPP 3.5, Rev. 17	(Attach if not previously provided)
	EPIP-1, Rev. 61	
	EPIP-4, Rev. 43	

Proposed references to be provided to applicants during examination: **NPG-SPP-3.5 Attachment 1 (Pages 1 through 18)**
EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3

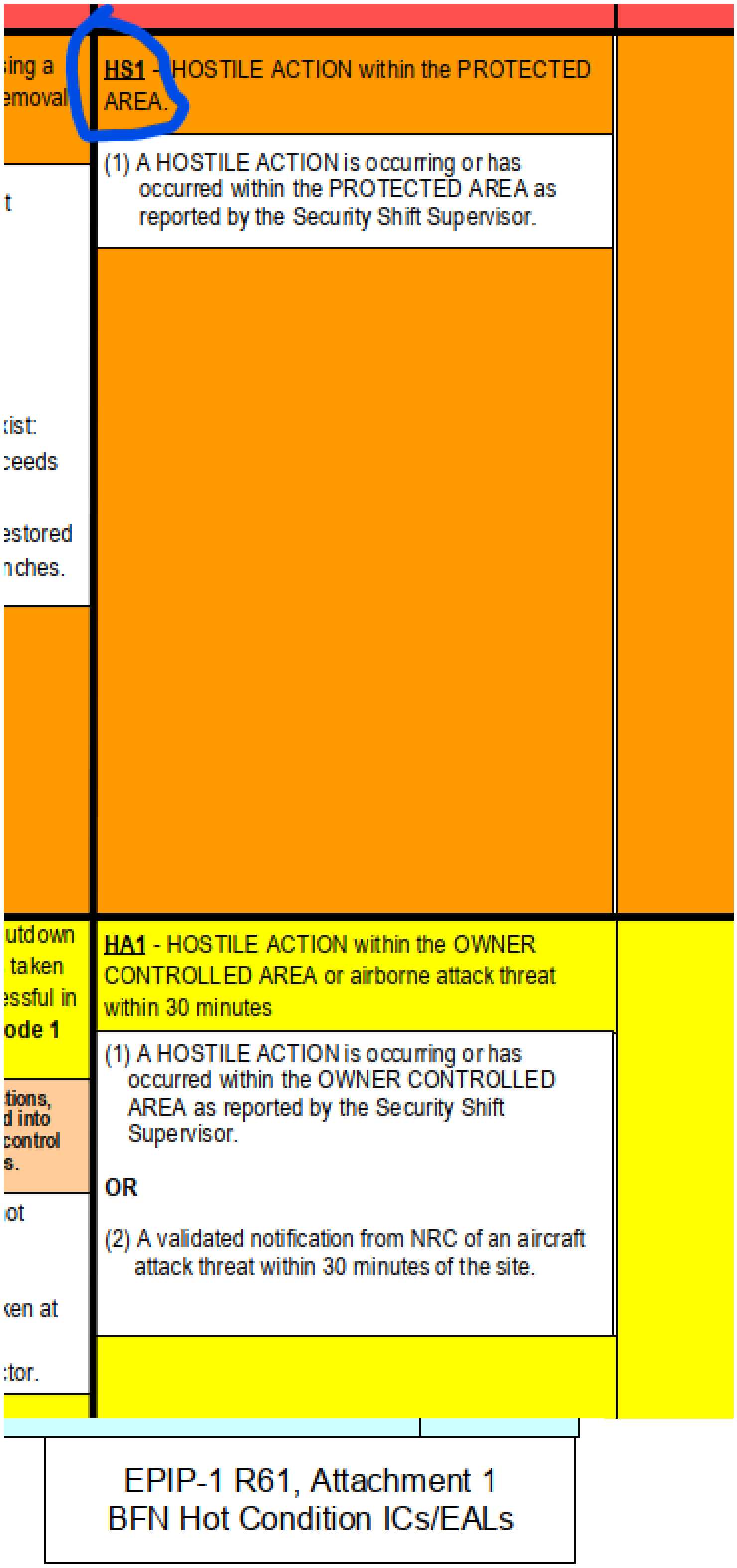
Learning Objective: OPL171.092, Obj. 2 (As available)

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**

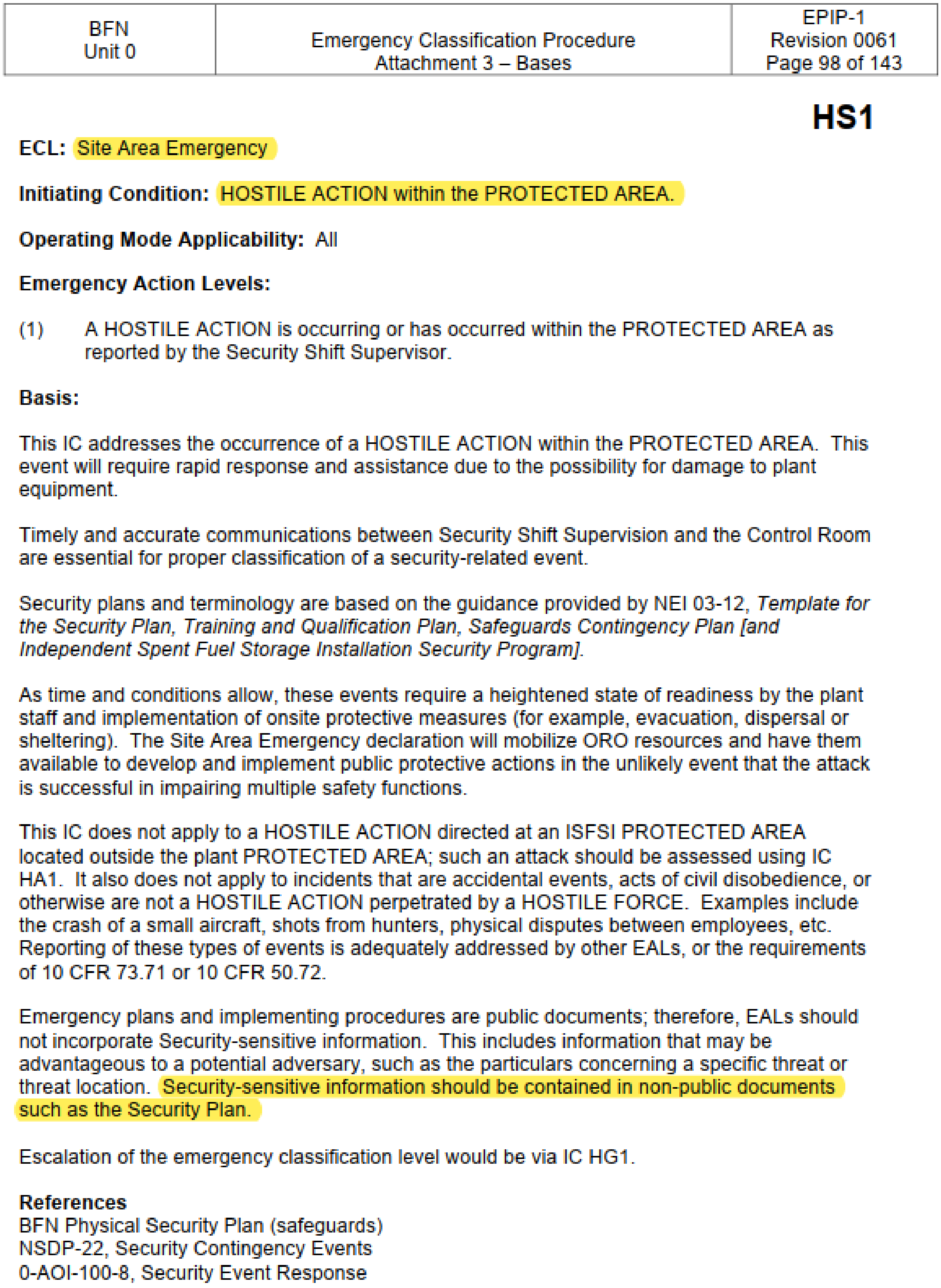
Excerpt from EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3:



ing a emoval	**HS1** - HOSTILE ACTION within the PROTECTED AREA.	
t ist: ceeds estored nches.	(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.	
utdown taken essful in **ode 1**	**HA1** - HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes	
tions, d into control s. ot ken at tor.	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor. **OR** (2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	

EPIP-1 R61, Attachment 1
BFN Hot Condition ICs/EALs

Excerpt from EPIP-1:



BFN Unit 0	Emergency Classification Procedure Attachment 3 – Bases	EPIP-1 Revision 0061 Page 98 of 143

HS1

ECL: Site Area Emergency

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Levels:

(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room are essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (for example, evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize ORO resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

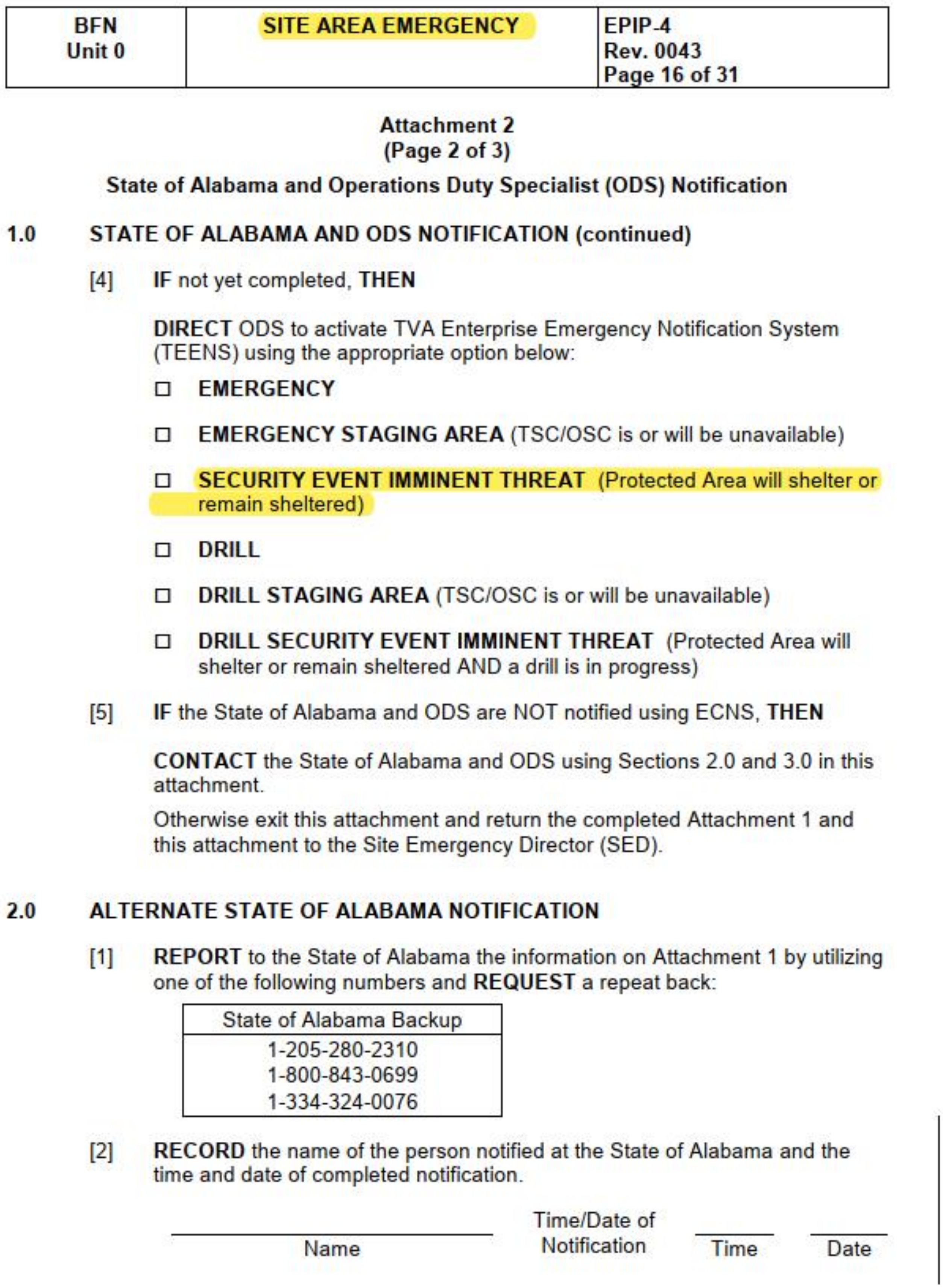
This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR 73.71 or 10 CFR 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.

Escalation of the emergency classification level would be via IC HG1.

References
BFN Physical Security Plan (safeguards)
NSDP-22, Security Contingency Events
0-AOI-100-8, Security Event Response

Excerpt from EPIP-4:



BFN **Unit 0**	**SITE AREA EMERGENCY**	**EPIP-4** **Rev. 0043** **Page 16 of 31**

Attachment 2
(Page 2 of 3)

State of Alabama and Operations Duty Specialist (ODS) Notification

1.0 STATE OF ALABAMA AND ODS NOTIFICATION (continued)

[4] **IF** not yet completed, **THEN**

DIRECT ODS to activate TVA Enterprise Emergency Notification System (TEENS) using the appropriate option below:

- ☐ **EMERGENCY**
- ☐ **EMERGENCY STAGING AREA** (TSC/OSC is or will be unavailable)
- ☐ **SECURITY EVENT IMMINENT THREAT** (Protected Area will shelter or remain sheltered)
- ☐ **DRILL**
- ☐ **DRILL STAGING AREA** (TSC/OSC is or will be unavailable)
- ☐ **DRILL SECURITY EVENT IMMINENT THREAT** (Protected Area will shelter or remain sheltered AND a drill is in progress)

[5] **IF** the State of Alabama and ODS are NOT notified using ECNS, **THEN**

CONTACT the State of Alabama and ODS using Sections 2.0 and 3.0 in this attachment.

Otherwise exit this attachment and return the completed Attachment 1 and this attachment to the Site Emergency Director (SED).

2.0 ALTERNATE STATE OF ALABAMA NOTIFICATION

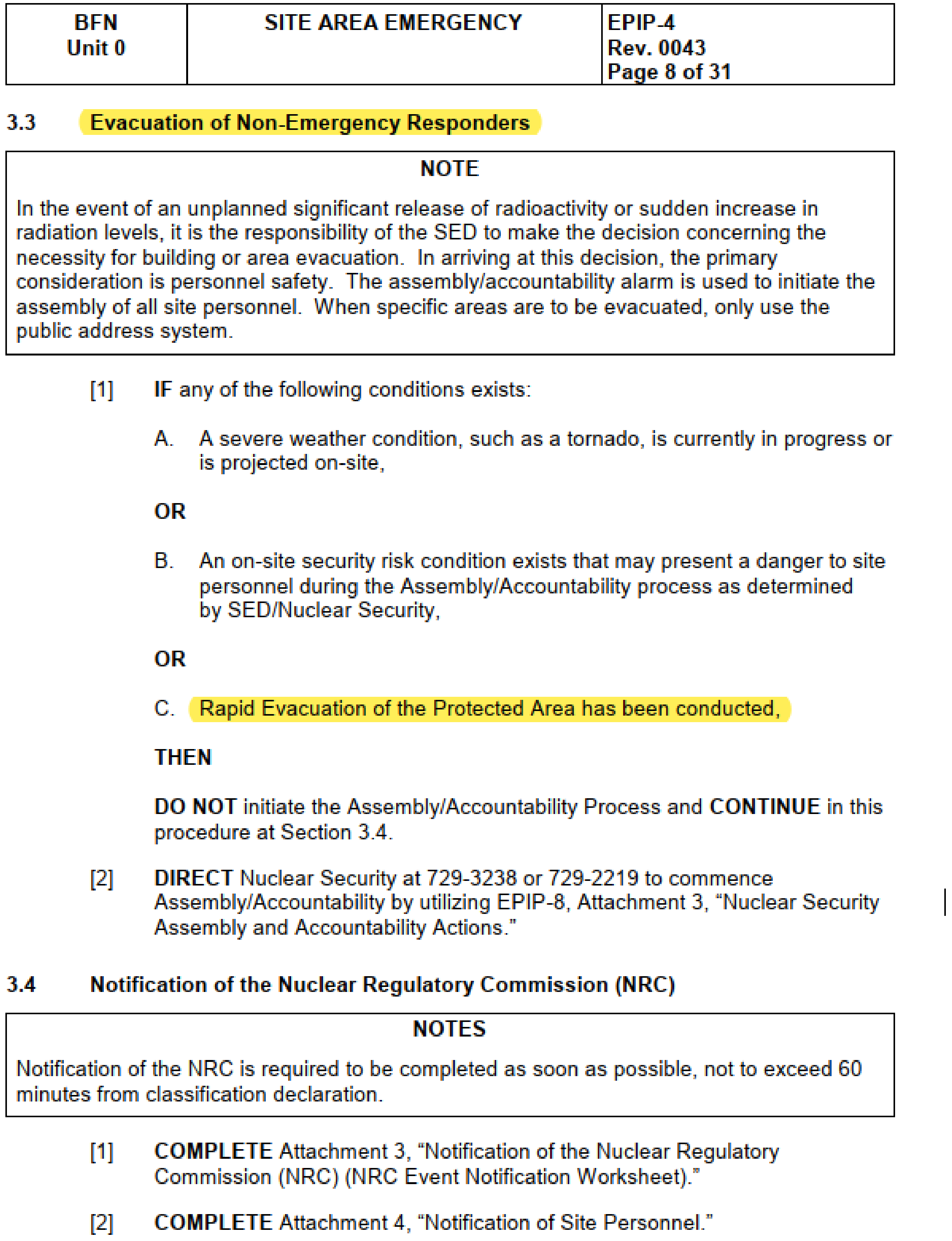
[1] **REPORT** to the State of Alabama the information on Attachment 1 by utilizing one of the following numbers and **REQUEST** a repeat back:

State of Alabama Backup
1-205-280-2310 1-800-843-0699 1-334-324-0076

[2] **RECORD** the name of the person notified at the State of Alabama and the time and date of completed notification.

______________	Time/Date of Notification	______	______
Name		Time	Date

Supports Distractors A(1)/B1):



BFN **Unit 0**	**SITE AREA EMERGENCY**	**EPIP-4** **Rev. 0043** **Page 8 of 31**

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, Attachment 3, “Nuclear Security Assembly and Accountability Actions.”

3.4 Notification of the Nuclear Regulatory Commission (NRC)

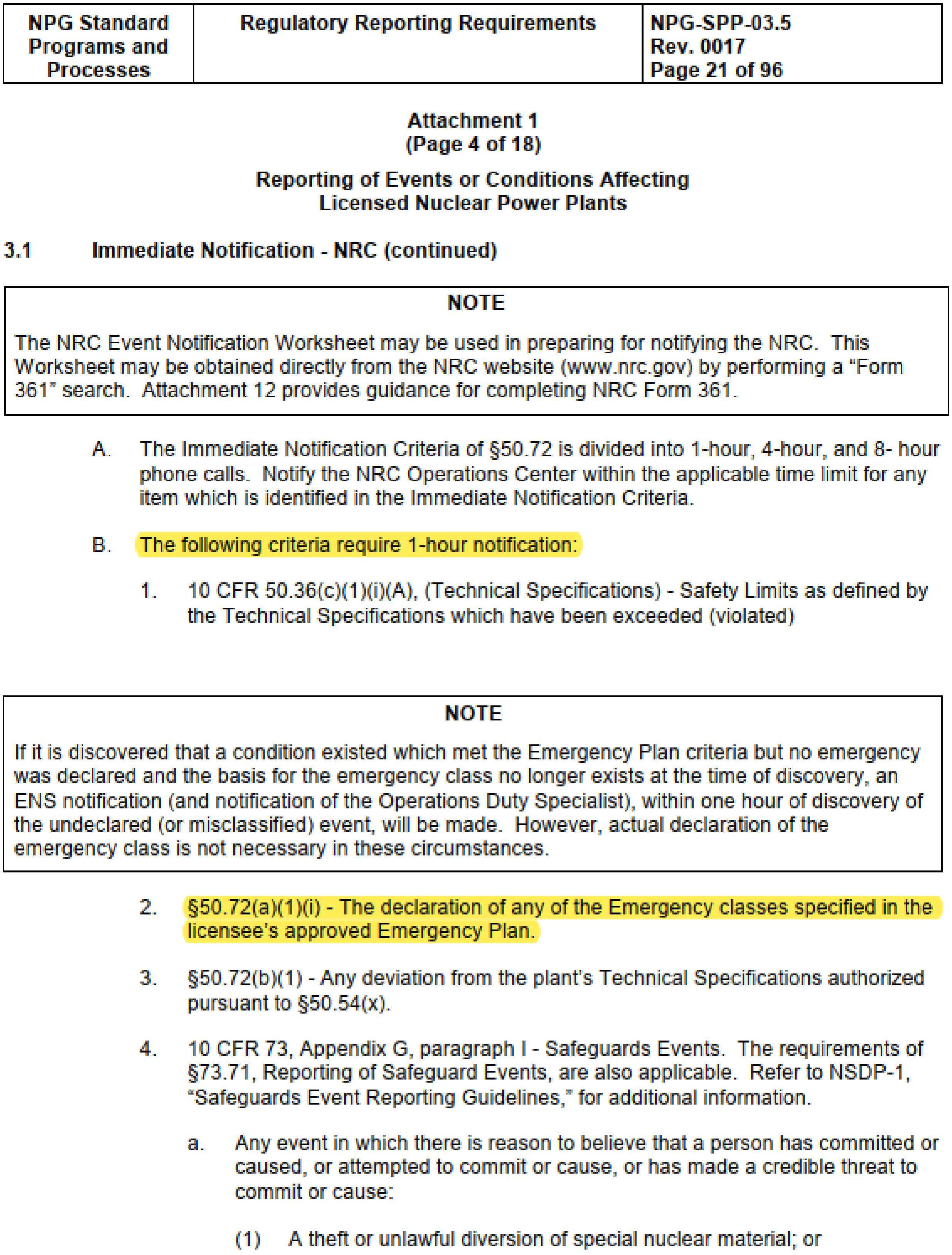
NOTES

Notification of the NRC is required to be completed as soon as possible, not to exceed 60 minutes from classification declaration.

[1] **COMPLETE** Attachment 3, “Notification of the Nuclear Regulatory Commission (NRC) (NRC Event Notification Worksheet).”

[2] **COMPLETE** Attachment 4, “Notification of Site Personnel.”

Excerpts from NPG-SPP-03.5:



NPG Standard Programs and Processes	**Regulatory Reporting Requirements**	**NPG-SPP-03.5** **Rev. 0017** **Page 21 of 96**

Attachment 1
(Page 4 of 18)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

NOTE

The NRC Event Notification Worksheet may be used in preparing for notifying the NRC. This Worksheet may be obtained directly from the NRC website (www.nrc.gov) by performing a "Form 361" search. Attachment 12 provides guidance for completing NRC Form 361.

A. The Immediate Notification Criteria of §50.72 is divided into 1-hour, 4-hour, and 8- hour phone calls. Notify the NRC Operations Center within the applicable time limit for any item which is identified in the Immediate Notification Criteria.

B. The following criteria require 1-hour notification:

1. 10 CFR 50.36(c)(1)(i)(A), (Technical Specifications) - Safety Limits as defined by the Technical Specifications which have been exceeded (violated)

NOTE

If it is discovered that a condition existed which met the Emergency Plan criteria but no emergency was declared and the basis for the emergency class no longer exists at the time of discovery, an ENS notification (and notification of the Operations Duty Specialist), within one hour of discovery of the undeclared (or misclassified) event, will be made. However, actual declaration of the emergency class is not necessary in these circumstances.

2. §50.72(a)(1)(i) - The declaration of any of the Emergency classes specified in the licensee's approved Emergency Plan.

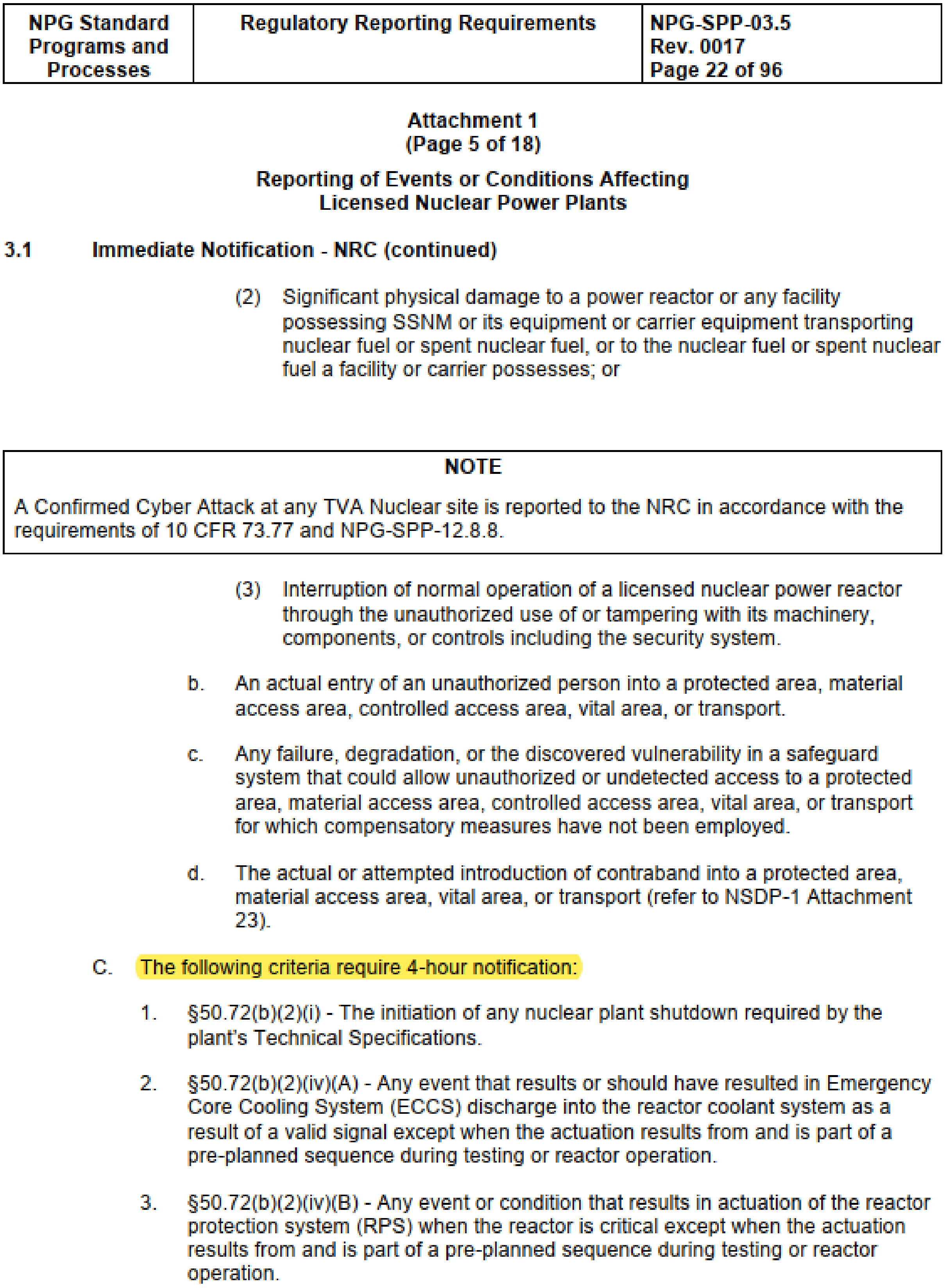
3. §50.72(b)(1) - Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x).

4. 10 CFR 73, Appendix G, paragraph I - Safeguards Events. The requirements of §73.71, Reporting of Safeguard Events, are also applicable. Refer to NSDP-1, "Safeguards Event Reporting Guidelines," for additional information.

   a. Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:

      (1) A theft or unlawful diversion of special nuclear material; or

Supports Distractors B(2)/D(2):



NPG Standard Programs and Processes	**Regulatory Reporting Requirements**	**NPG-SPP-03.5** **Rev. 0017** **Page 22 of 96**

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.

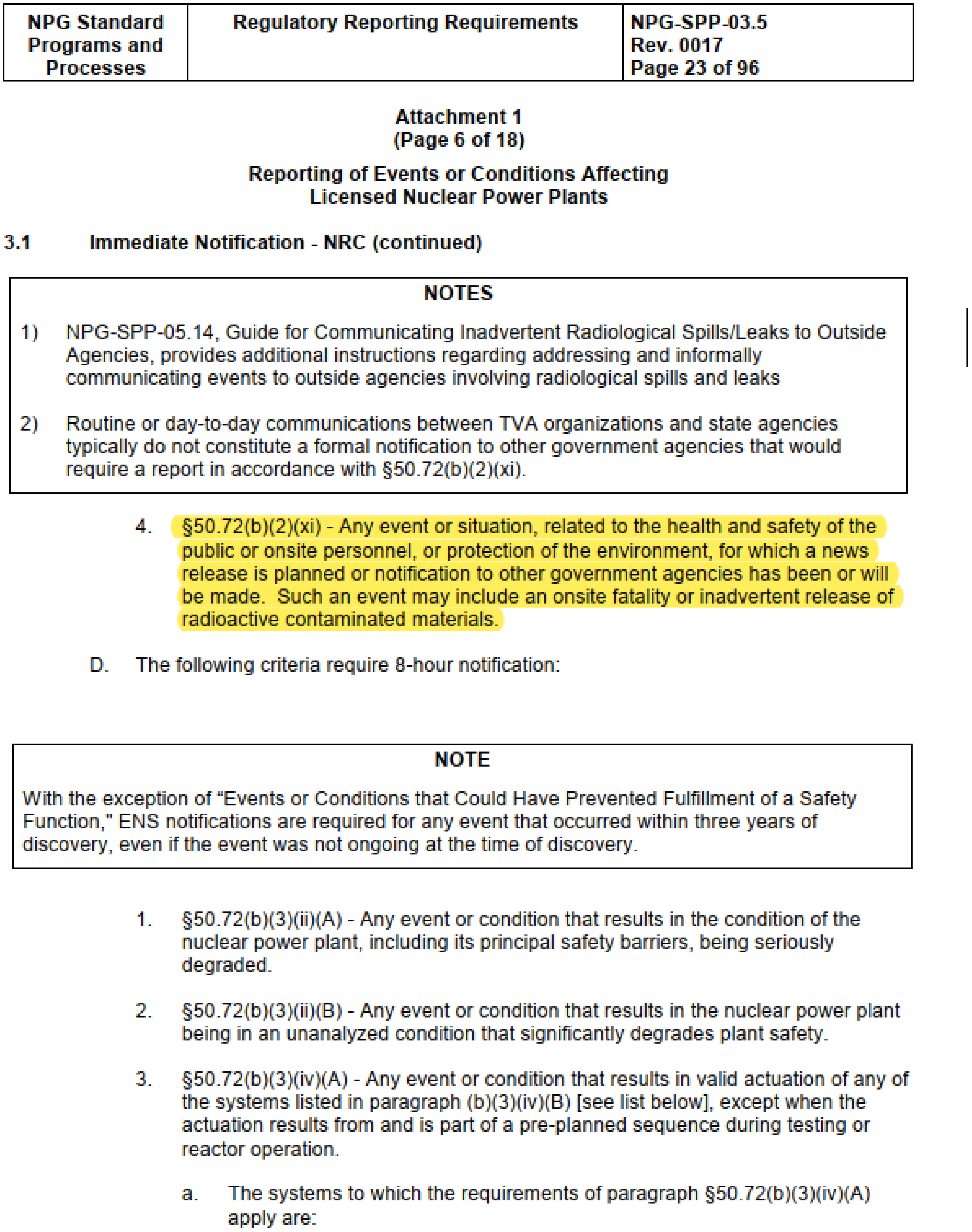
b. An actual entry of an unauthorized person into a protected area, material access area, controlled access area, vital area, or transport.

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. 0017** **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, Guide for Communicating Inadvertent Radiological Spills/Leaks to Outside Agencies, 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:

Examination Outline Cross-reference:	Level	RO	SRO
Emergency Procedures / Plan	Tier #	------	3
G2.4.38 **(10CFR 55.43.5 – SRO Only)**	Group #	------	------
Ability to take actions required by the facility emergency plan implementing procedures, including supporting or acting as emergency coordinator	K/A #	G2.4.38	
	Importance Rating	------	4.4

Proposed Question: **# 100**

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 in accordance with Emergency Plan procedures?

A. Notifications to the State.

B. Notifications to Site Personnel.

C. Conducting Site Accountability Actions.

D. Determining Protective Action Recommendations.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that while the SM/SED cannot delegate the actual decision of the Protective Action Recommendations (PARs) on Appendix A of EPIP-5, in accordance with OPDP-1, Conduct of Operations, SM/SED can delegate a designee some other duties and/or actions, including having an SRO fax the Notification to the State. *(See attached)*

B INCORRECT: Incorrect but plausible in that the SM will be the SED during the event until the Technical Support Center (TSC) is staffed. The SM designee will be performing EPIP-5, Appendix D, Notification of Site Personnel and returns the completed form to the SED. *(See attached)*

C INCORRECT: Incorrect but plausible in that in accordance with EPIP-8, Personnel Accountability and Evacuation, 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. *(See attached)*

D **CORRECT**: (*See attached)* In accordance with OPDP-1, Conduct of Operations, the Shift Manager (SM) functions as the Site Emergency Director (SED) until relieved, and cannot delegate classification of an emergency or Protective Action Recommendations. All other duties may be delegated to another qualified SRO as allowed by site specific procedures. In accordance with EPIP-5, General Emergency, the SM/SED cannot delegate Protective Action Recommendations.

SRO Level Justification: Test the candidate's ability to take actions as the Shift Manager (whom is a licensed SRO) in the implementation of the Emergency Plan along with the delegated duties that can be performed by a licensed SRO. SRO only because of link to 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 recall of specific facts as it relates to an emergency condition.

Technical Reference(s): OPDP-1, Rev. 52 (Attach if not previously provided)
EPIP-1, Rev. 61
EPIP-5, Rev. 58
EPIP-8, Rev. 34

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.075, Obj. 4 (As available)

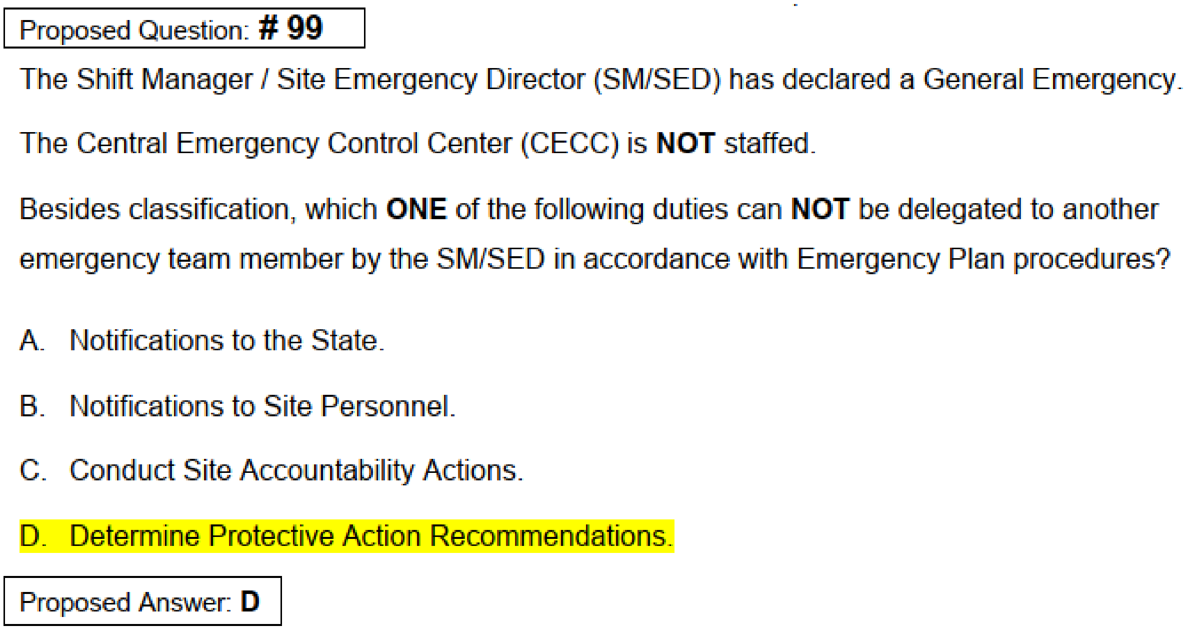
Question Source:	Bank #	BFN 21-04 #99	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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: **# 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 in accordance with Emergency Plan procedures?

A. Notifications to the State.

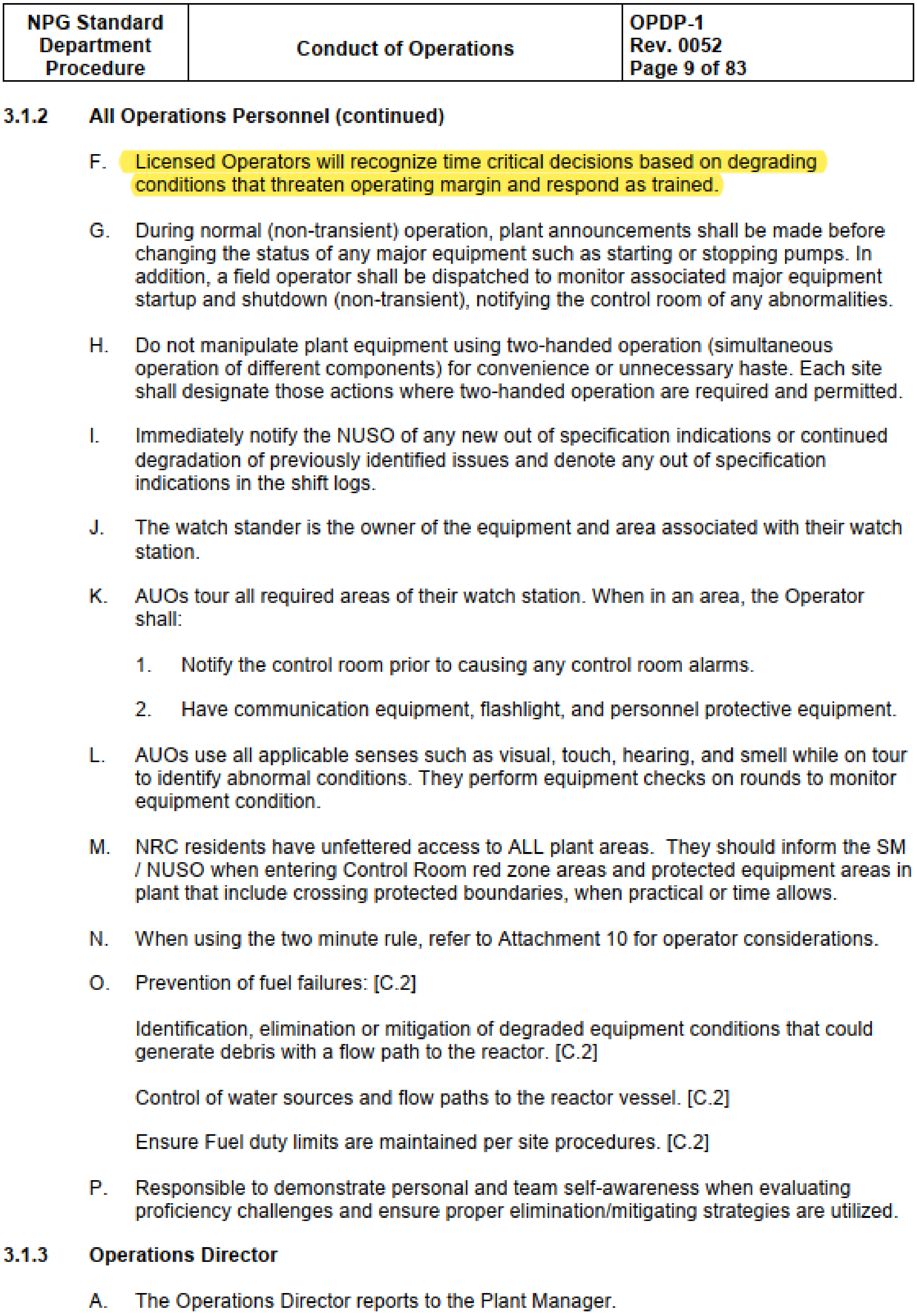
B. Notifications to Site Personnel.

C. Conduct Site Accountability Actions.

D. Determine Protective Action Recommendations.

Proposed Answer: **D**

Excerpts from OPDP-1:



NPG Standard Department Procedure	**Conduct of Operations**	**OPDP-1** **Rev. 0052** **Page 9 of 83**

3.1.2 All Operations Personnel (continued)

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.

H. Do not manipulate plant equipment using two-handed operation (simultaneous operation of different components) for convenience or unnecessary haste. Each site shall designate those actions where two-handed operation are required and permitted.

I. Immediately notify the NUSO of any new out of specification indications or continued degradation of previously identified issues and denote any out of specification indications in the shift logs.

J. The watch stander is the owner of the equipment and area associated with their watch station.

K. AUOs tour all required areas of their watch station. When in an area, the Operator shall:

1. Notify the control room prior to causing any control room alarms.
2. Have communication equipment, flashlight, and personnel protective equipment.

L. AUOs use all applicable senses such as visual, touch, hearing, and smell while on tour to identify abnormal conditions. They perform equipment checks on rounds to monitor equipment condition.

M. NRC residents have unfettered access to ALL plant areas. They should inform the SM / NUSO when entering Control Room red zone areas and protected equipment areas in plant that include crossing protected boundaries, when practical or time allows.

N. When using the two minute rule, refer to Attachment 10 for operator considerations.

O. Prevention of fuel failures: [C.2]

Identification, elimination or mitigation of degraded equipment conditions that could generate debris with a flow path to the reactor. [C.2]

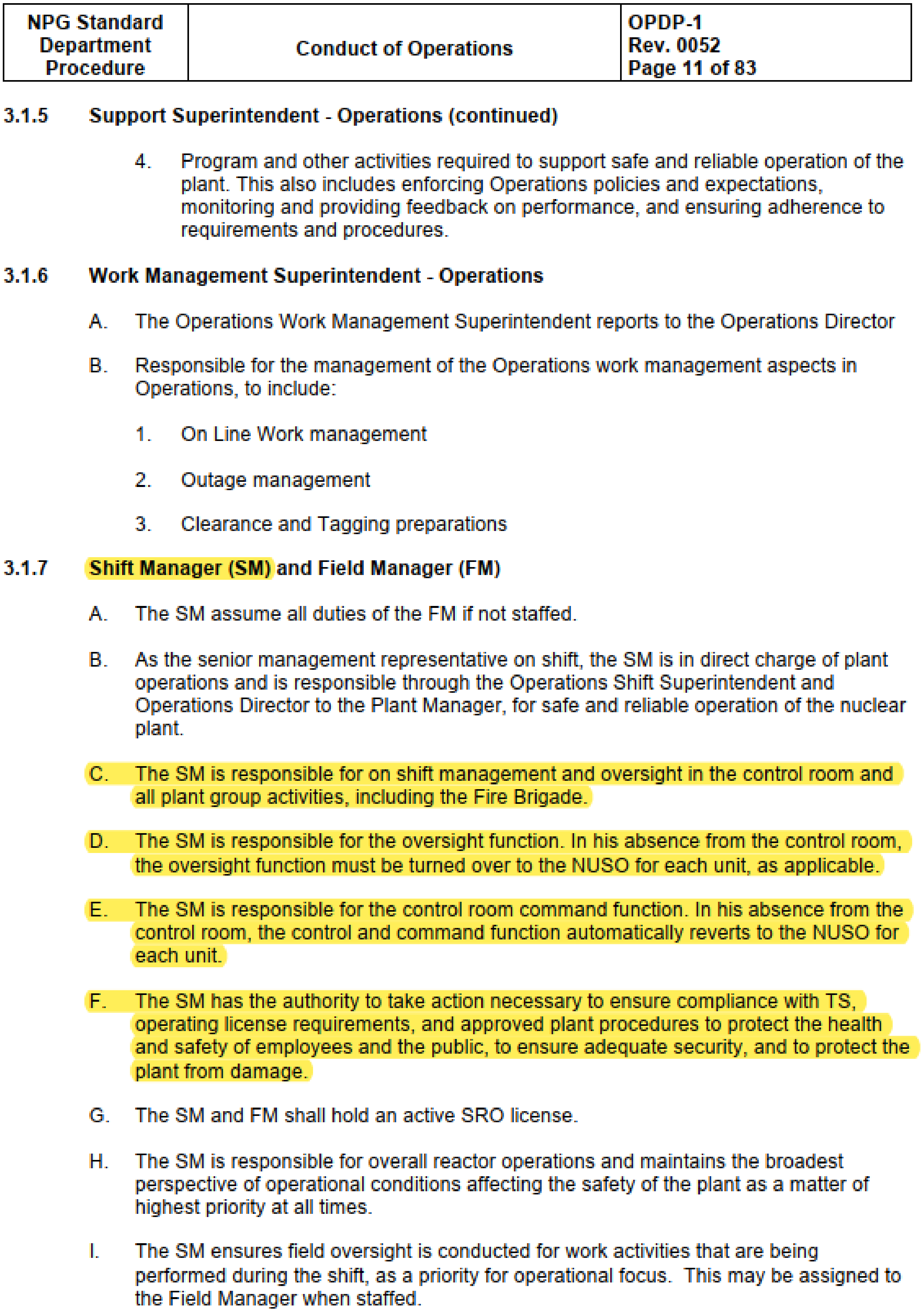
Control of water sources and flow paths to the reactor vessel. [C.2]

Ensure Fuel duty limits are maintained per site procedures. [C.2]

P. Responsible to demonstrate personal and team self-awareness when evaluating proficiency challenges and ensure proper elimination/mitigating strategies are utilized.

3.1.3 Operations Director

A. The Operations Director reports to the Plant Manager.



NPG Standard Department Procedure	**Conduct of Operations**	**OPDP-1** **Rev. 0052** **Page 11 of 83**

3.1.5 Support Superintendent - Operations (continued)

4. Program and other activities required to support safe and reliable operation of the plant. This also includes enforcing Operations policies and expectations, monitoring and providing feedback on performance, and ensuring adherence to requirements and procedures.

3.1.6 Work Management Superintendent - Operations

A. The Operations Work Management Superintendent reports to the Operations Director

B. Responsible for the management of the Operations work management aspects in Operations, to include:

1. On Line Work management
2. Outage management
3. Clearance and Tagging preparations

3.1.7 Shift Manager (SM) and Field Manager (FM)

A. The SM assume all duties of the FM if not staffed.

B. 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.

C. The SM is responsible for on shift management and oversight in the control room and all plant group activities, including the Fire Brigade.

D. 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.

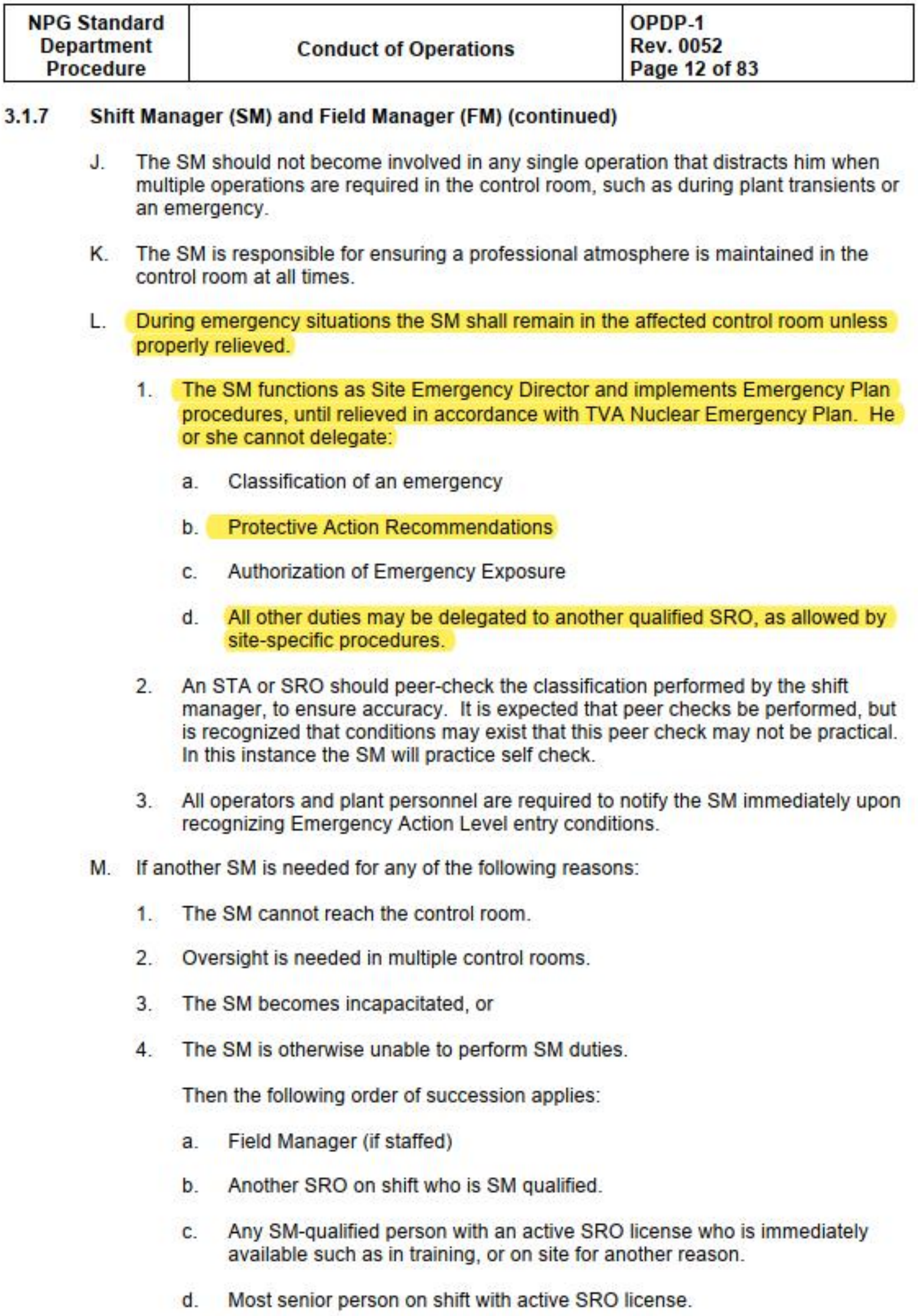
E. The SM is responsible for the control room command function. In his absence from the control room, the control and command function automatically reverts to the NUSO for each unit.

F. 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.

G. The SM and FM shall hold an active SRO license.

H. 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.

I. The SM ensures field oversight is conducted for work activities that are being performed during the shift, as a priority for operational focus. This may be assigned to the Field Manager when staffed.



NPG Standard Department Procedure	**Conduct of Operations**	**OPDP-1** **Rev. 0052** **Page 12 of 83**

3.1.7 Shift Manager (SM) and Field Manager (FM) (continued)

J. 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.

K. The SM is responsible for ensuring a professional atmosphere is maintained in the control room at all times.

L. 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.
2. An STA or SRO should peer-check the classification performed by the shift manager, to ensure accuracy. It is expected that peer checks be performed, but is recognized that conditions may exist that this peer check may not be practical. In this instance the SM will practice self check.
3. All operators and plant personnel are required to notify the SM immediately upon recognizing Emergency Action Level entry conditions.

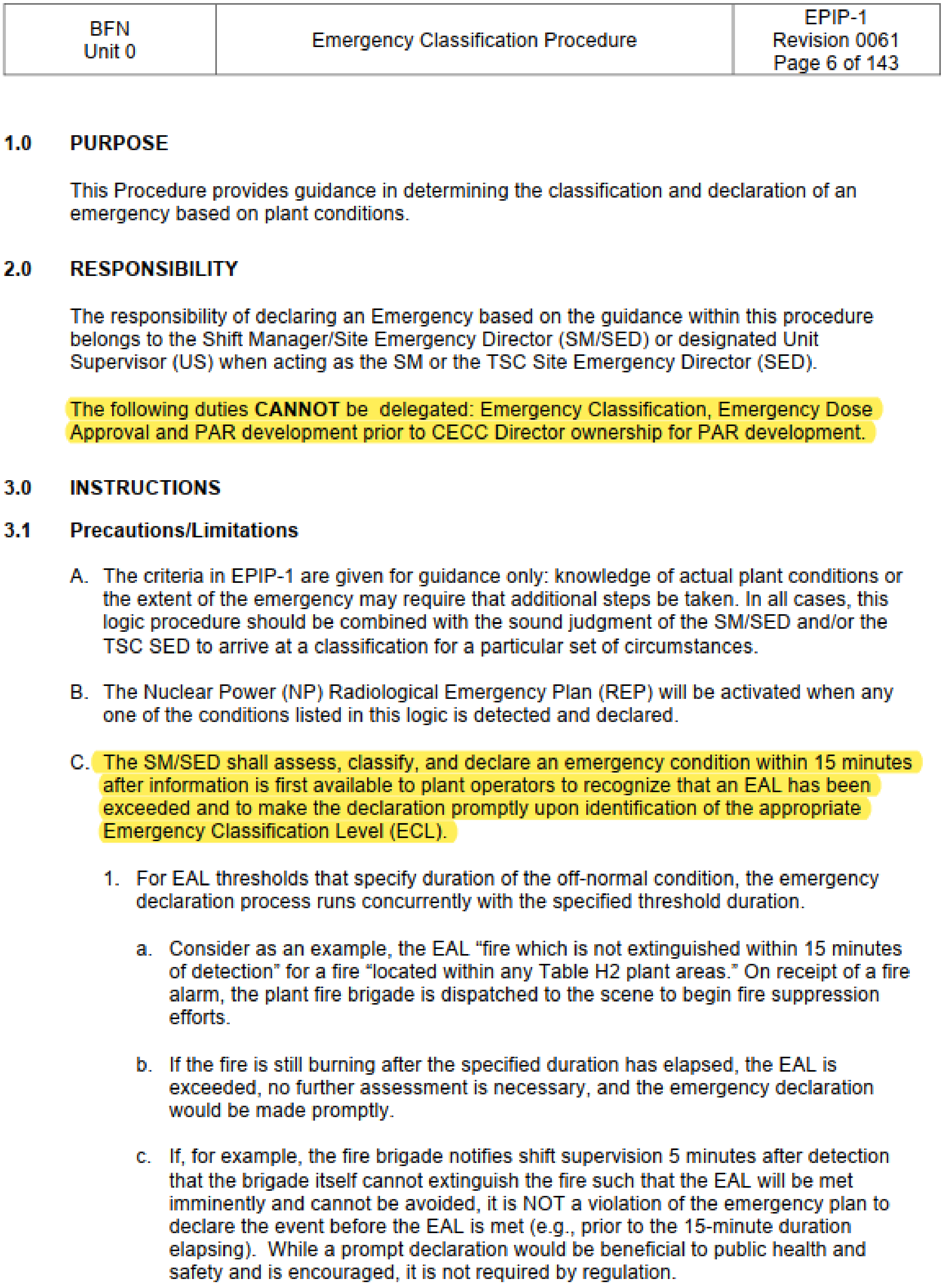
M. If another SM is needed for any of the following reasons:

1. The SM cannot reach the control room.
2. Oversight is needed in multiple control rooms.
3. The SM becomes incapacitated, or
4. The SM is otherwise unable to perform SM duties.

    Then the following order of succession applies:

    a. Field Manager (if staffed)
    b. Another SRO on shift who is SM qualified.
    c. Any SM-qualified person with an active SRO license who is immediately available such as in training, or on site for another reason.
    d. Most senior person on shift with active SRO license.

Excerpt from EPIP-1:



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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).

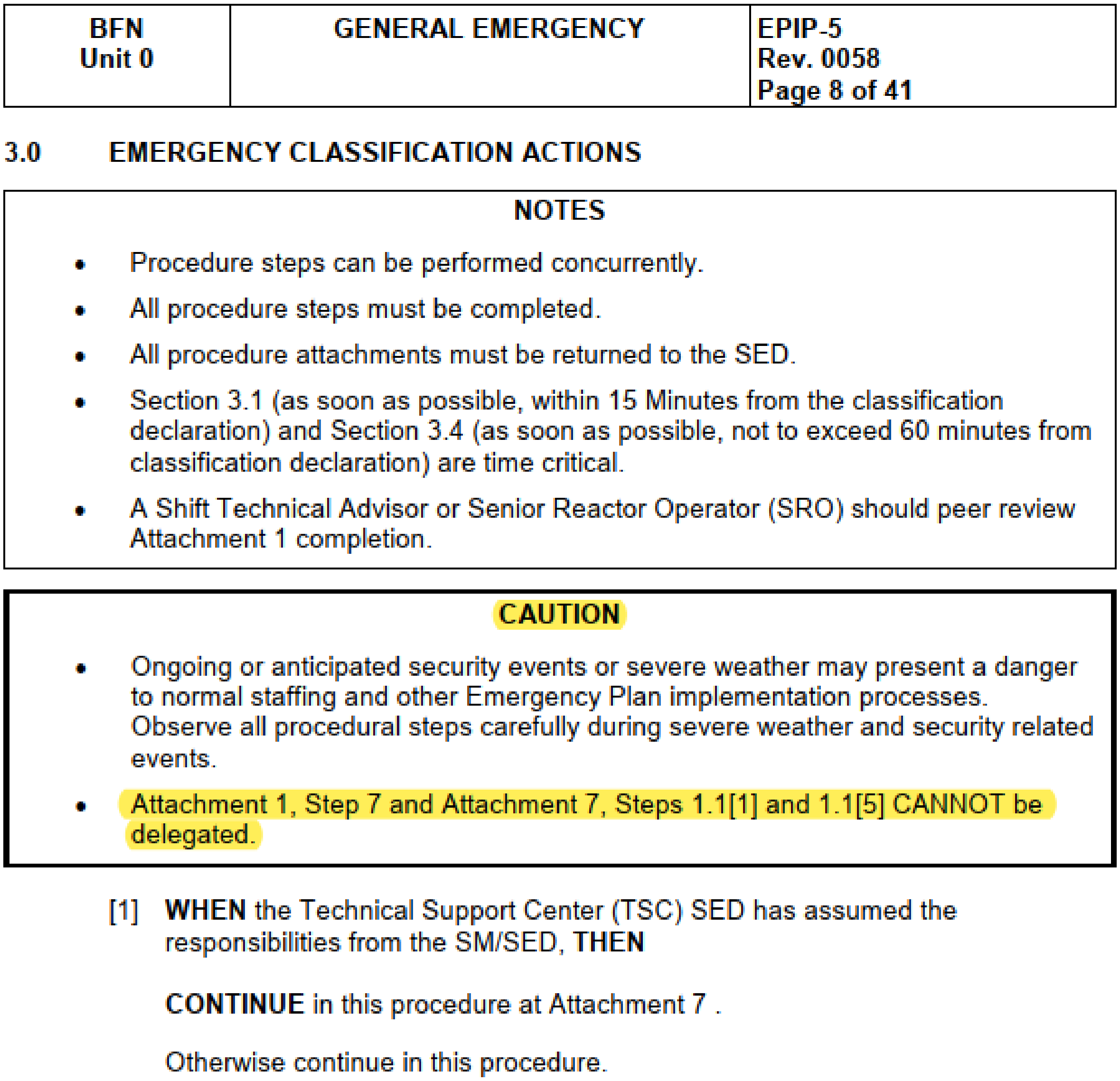
1. For EAL thresholds that specify duration of the off-normal condition, the emergency declaration process runs concurrently with the specified threshold duration.

   a. Consider as an example, the EAL "fire which is not extinguished within 15 minutes of detection" for a fire "located within any Table H2 plant areas." On receipt of a fire alarm, the plant fire brigade is dispatched to the scene to begin fire suppression efforts.

   b. If the fire is still burning after the specified duration has elapsed, the EAL is exceeded, no further assessment is necessary, and the emergency declaration would be made promptly.

   c. If, for example, the fire brigade notifies shift supervision 5 minutes after detection that the brigade itself cannot extinguish the fire such that the EAL will be met imminently and cannot be avoided, it is NOT a violation of the emergency plan to declare the event before the EAL is met (e.g., prior to the 15-minute duration elapsing). While a prompt declaration would be beneficial to public health and safety and is encouraged, it is not required by regulation.

Excerpts from EPIP-5:



BFN **Unit 0**	**GENERAL EMERGENCY**	**EPIP-5** **Rev. 0058** **Page 8 of 41**

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 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 Attachment 1 completion.

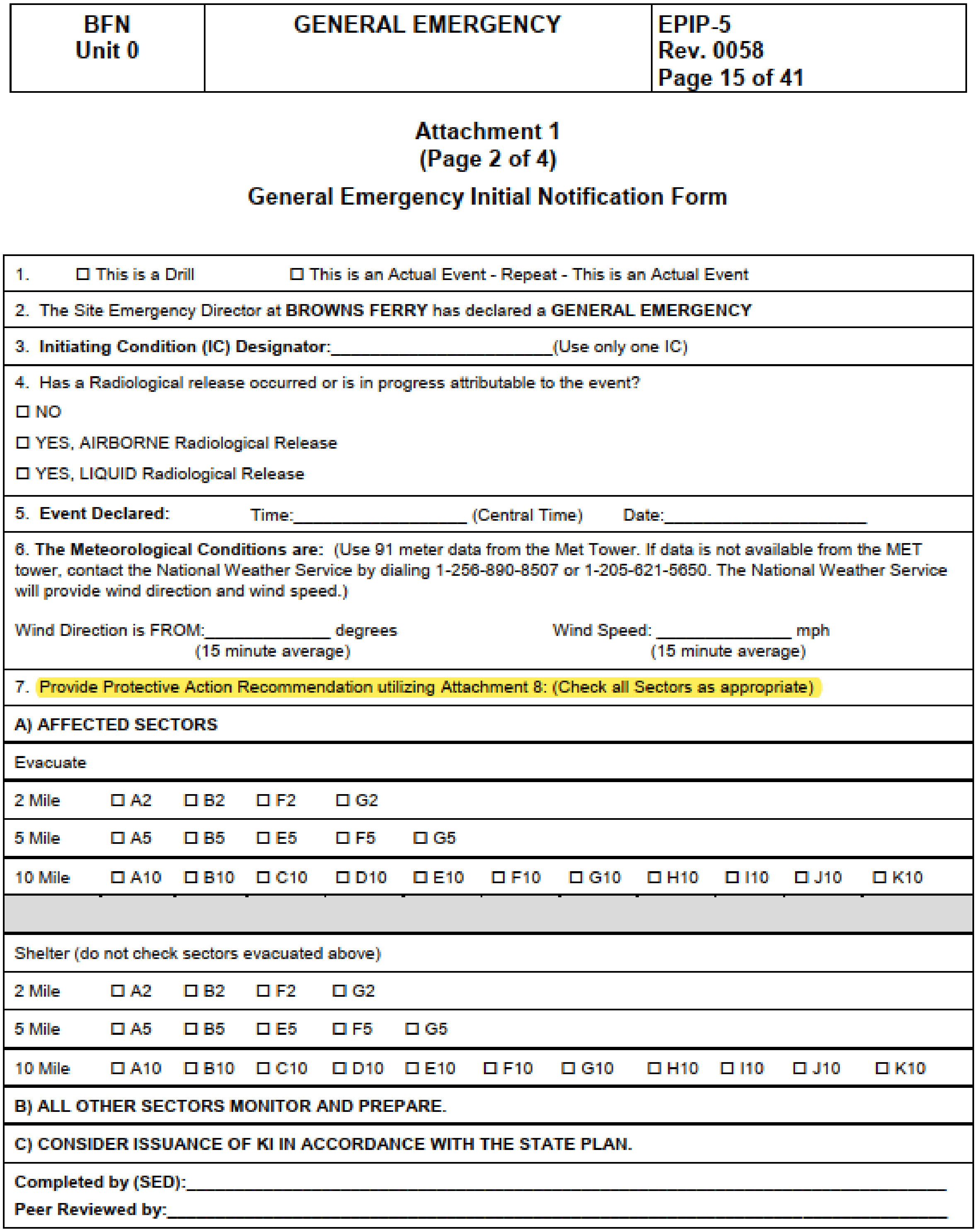
CAUTION

- 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.
- Attachment 1, Step 7 and Attachment 7, 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 Attachment 7 .

Otherwise continue in this procedure.



BFN **Unit 0**	**GENERAL EMERGENCY**	**EPIP-5** **Rev. 0058** **Page 15 of 41**

Attachment 1
(Page 2 of 4)

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. Has a Radiological release occurred or is in progress attributable to the event?

☐ NO

☐ YES, AIRBORNE Radiological Release

☐ YES, LIQUID Radiological Release

5. **Event Declared:** Time:___________________ (Central 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 Attachment 8: (Check all Sectors as appropriate)

A) AFFECTED SECTORS

Evacuate

2 Mile	☐ A2	☐ B2	☐ F2	☐ G2							
5 Mile	☐ A5	☐ B5	☐ E5	☐ F5	☐ G5						
10 Mile	☐ A10	☐ B10	☐ C10	☐ D10	☐ E10	☐ F10	☐ G10	☐ H10	☐ I10	☐ J10	☐ K10

Shelter (do not check sectors evacuated above)

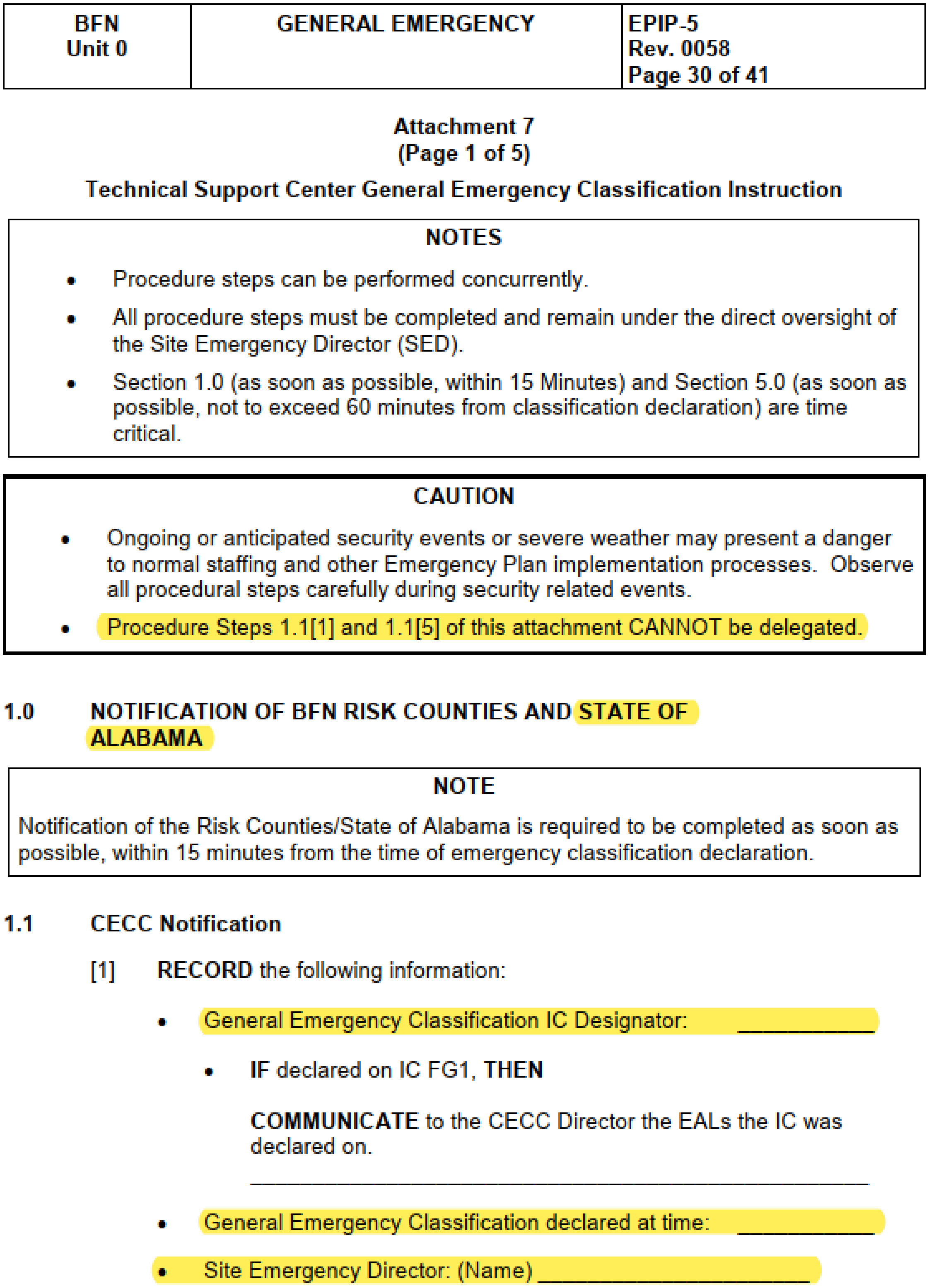
2 Mile	☐ A2	☐ B2	☐ F2	☐ G2							
5 Mile	☐ A5	☐ B5	☐ E5	☐ F5	☐ G5						
10 Mile	☐ A10	☐ B10	☐ C10	☐ D10	☐ E10	☐ F10	☐ G10	☐ H10	☐ I10	☐ J10	☐ K10

B) ALL OTHER SECTORS MONITOR AND PREPARE.

C) CONSIDER ISSUANCE OF KI IN ACCORDANCE WITH THE STATE PLAN.

Completed by (SED):____

Peer Reviewed by:____



BFN **Unit 0**	**GENERAL EMERGENCY**	**EPIP-5** **Rev. 0058** **Page 30 of 41**

Attachment 7
(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 attachment 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, within 15 minutes from the time of emergency classification declaration.

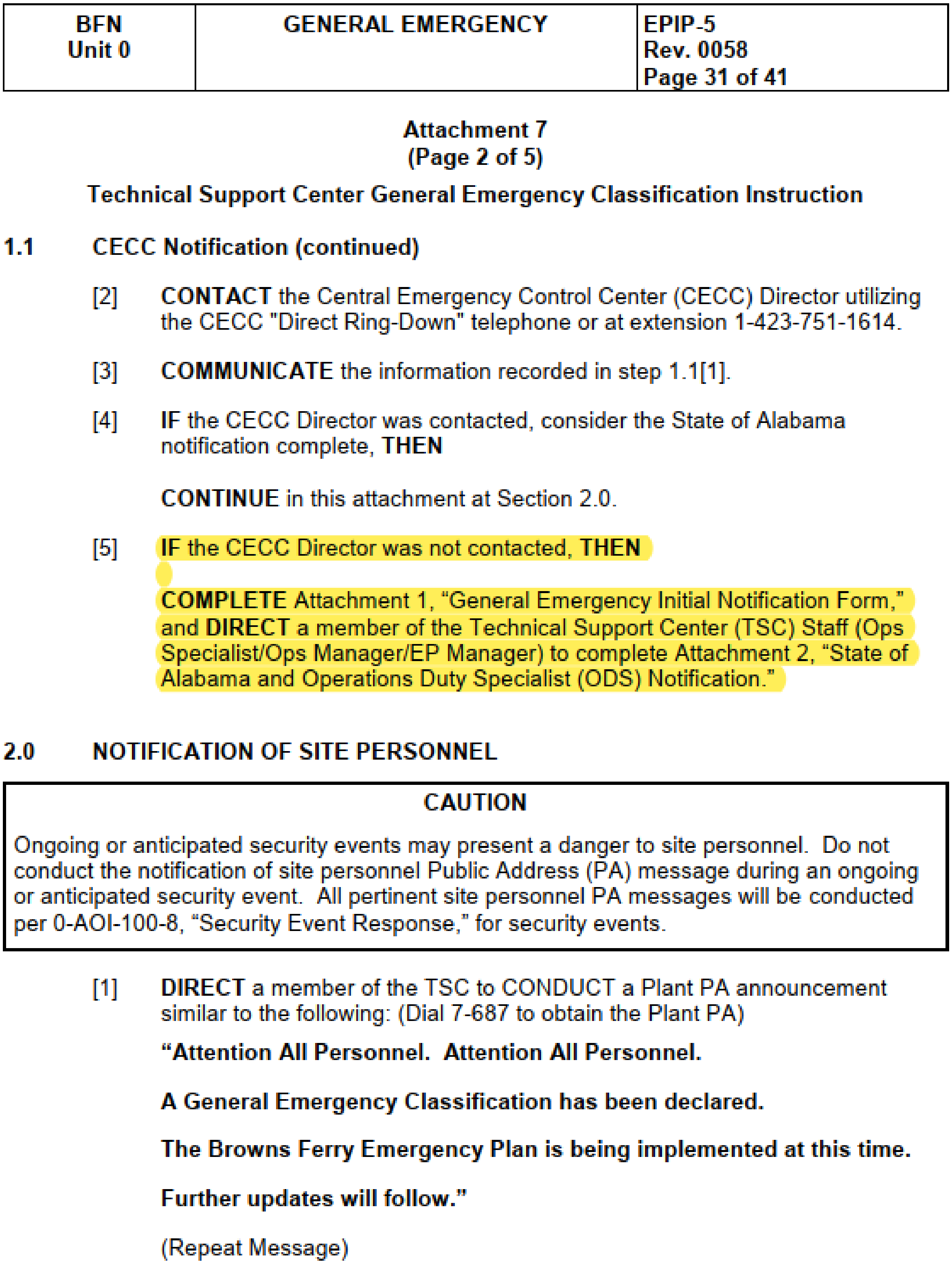
1.1 CECC Notification

[1] **RECORD** the following information:

- General Emergency Classification IC Designator: ____________
    - **IF** declared on IC FG1, **THEN**

      COMMUNICATE to the CECC Director the EALs the IC was declared on.

      ____
- General Emergency Classification declared at time: ____________
- Site Emergency Director: (Name) ______________________



BFN Unit 0	GENERAL EMERGENCY	EPIP-5 Rev. 0058 Page 31 of 41

Attachment 7
(Page 2 of 5)

Technical Support Center General Emergency Classification Instruction

1.1 CECC Notification (continued)

[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].

[4] **IF** the CECC Director was contacted, consider the State of Alabama notification complete, **THEN**

CONTINUE in this attachment at Section 2.0.

[5] **IF** the CECC Director was not contacted, **THEN**

COMPLETE Attachment 1, “General Emergency Initial Notification Form,” and **DIRECT** a member of the Technical Support Center (TSC) Staff (Ops Specialist/Ops Manager/EP Manager) to complete Attachment 2, “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.

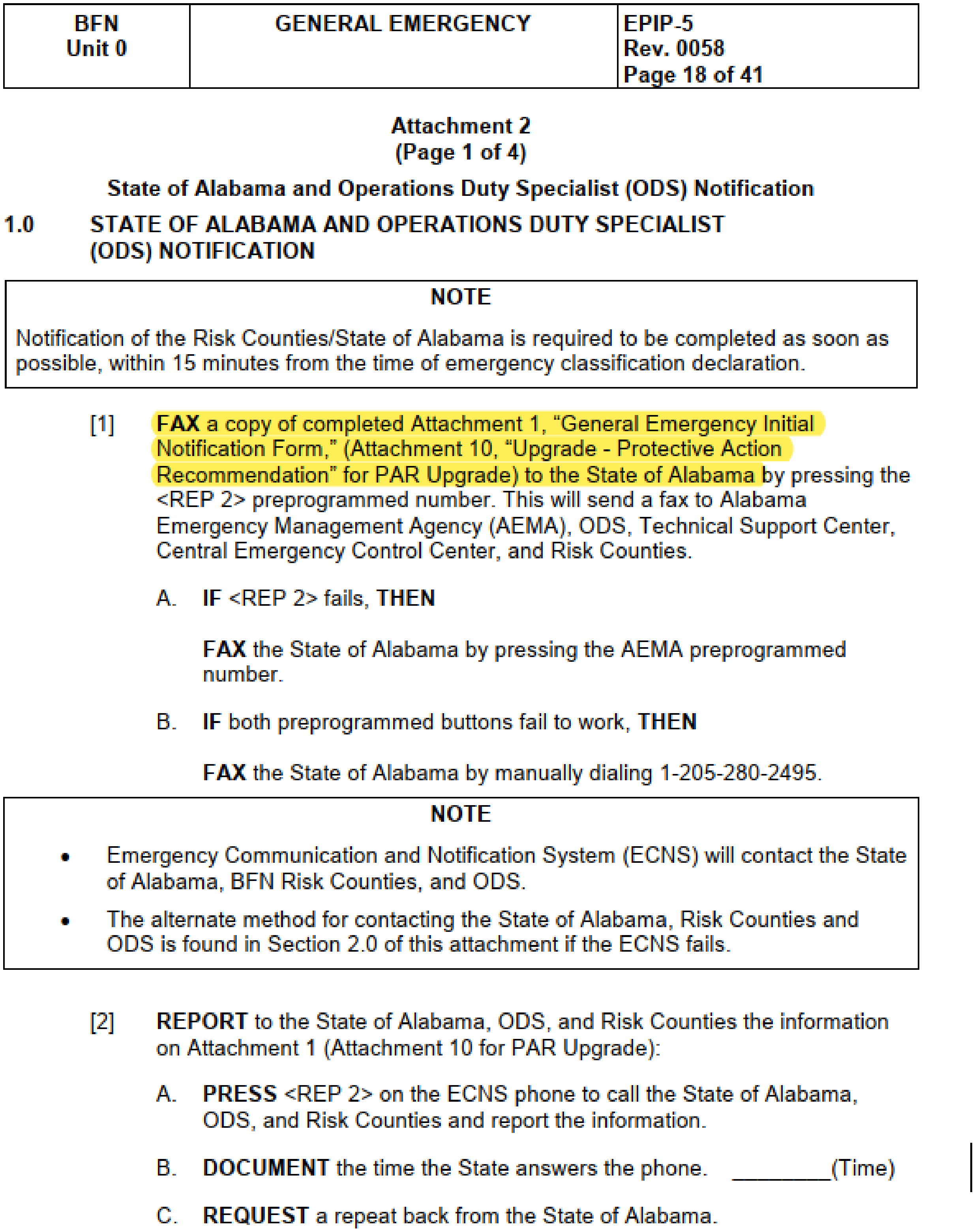
A General Emergency Classification has been declared.

The Browns Ferry Emergency Plan is being implemented at this time.

Further updates will follow.”

(Repeat Message)

Supports Distractor (A):



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Attachment 2
(Page 1 of 4)

State of Alabama and Operations Duty Specialist (ODS) Notification

1.0 STATE OF ALABAMA AND OPERATIONS DUTY SPECIALIST (ODS) NOTIFICATION

NOTE

Notification of the Risk Counties/State of Alabama is required to be completed as soon as possible, within 15 minutes from the time of emergency classification declaration.

[1] **FAX** a copy of completed Attachment 1, “General Emergency Initial Notification Form,” (Attachment 10, “Upgrade - Protective Action Recommendation” for PAR Upgrade) to the State of Alabama by pressing the <REP 2> preprogrammed number. This will send a fax to Alabama Emergency Management Agency (AEMA), ODS, Technical Support Center, Central Emergency Control Center, and Risk Counties.

A. **IF** <REP 2> fails, **THEN**

FAX the State of Alabama by pressing the AEMA preprogrammed number.

B. **IF** both preprogrammed buttons fail to work, **THEN**

FAX the State of Alabama by manually dialing 1-205-280-2495.

NOTE

- Emergency Communication and Notification System (ECNS) will contact the State of Alabama, BFN Risk Counties, and ODS.
- The alternate method for contacting the State of Alabama, Risk Counties and ODS is found in Section 2.0 of this attachment if the ECNS fails.

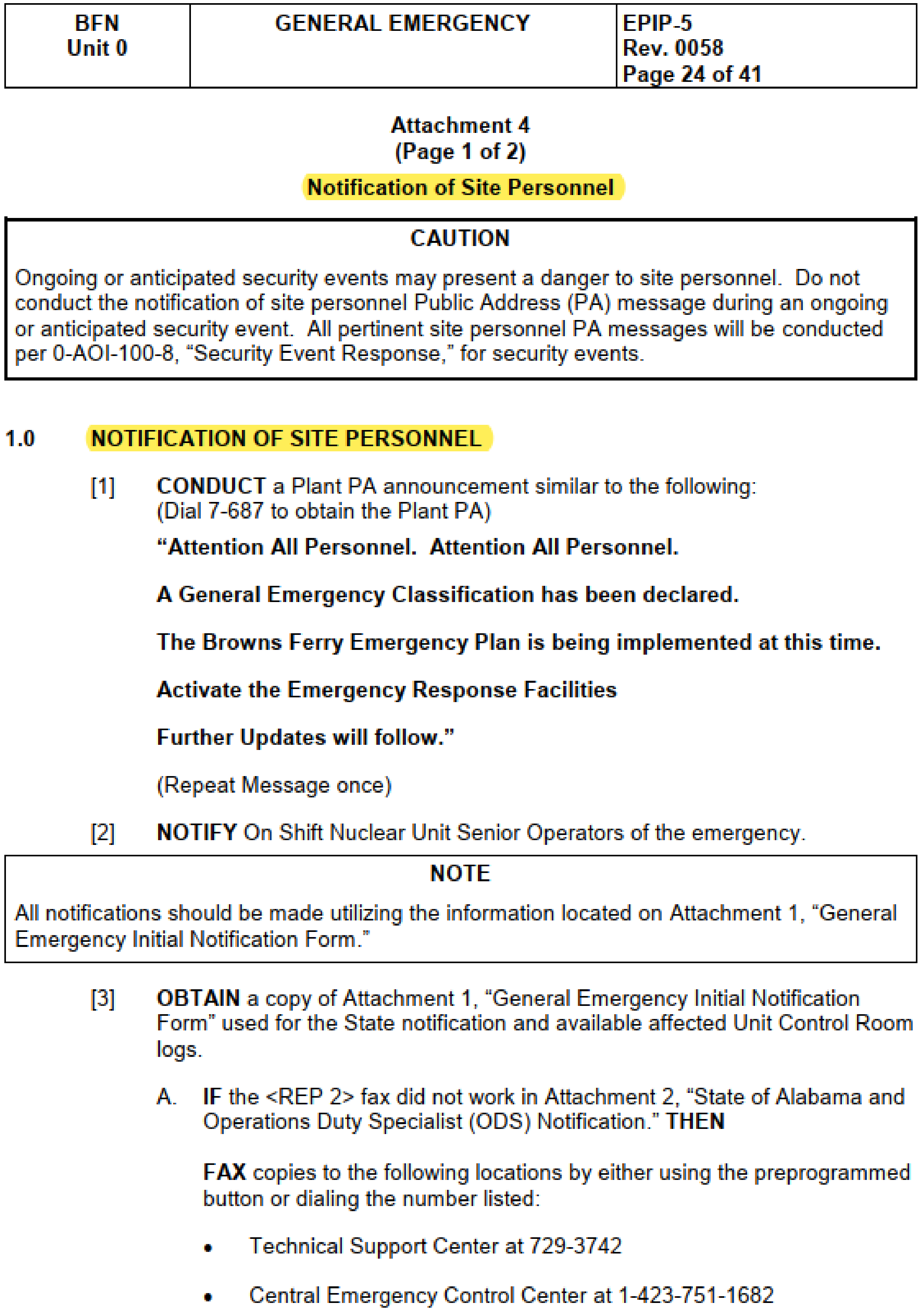
[2] **REPORT** to the State of Alabama, ODS, and Risk Counties the information on Attachment 1 (Attachment 10 for PAR Upgrade):

A. **PRESS** <REP 2> on the ECNS phone to call the State of Alabama, ODS, and Risk Counties and report the information.

B. **DOCUMENT** the time the State answers the phone. _________(Time)

C. **REQUEST** a repeat back from the State of Alabama.

Supports Distractor (B):



BFN Unit 0	GENERAL EMERGENCY	EPIP-5 Rev. 0058 Page 24 of 41

Attachment 4
(Page 1 of 2)
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.

1.0 NOTIFICATION OF SITE PERSONNEL

[1] **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.

Activate the Emergency Response Facilities

Further Updates will follow.”

(Repeat Message once)

[2] **NOTIFY** On Shift Nuclear Unit Senior Operators of the emergency.

NOTE

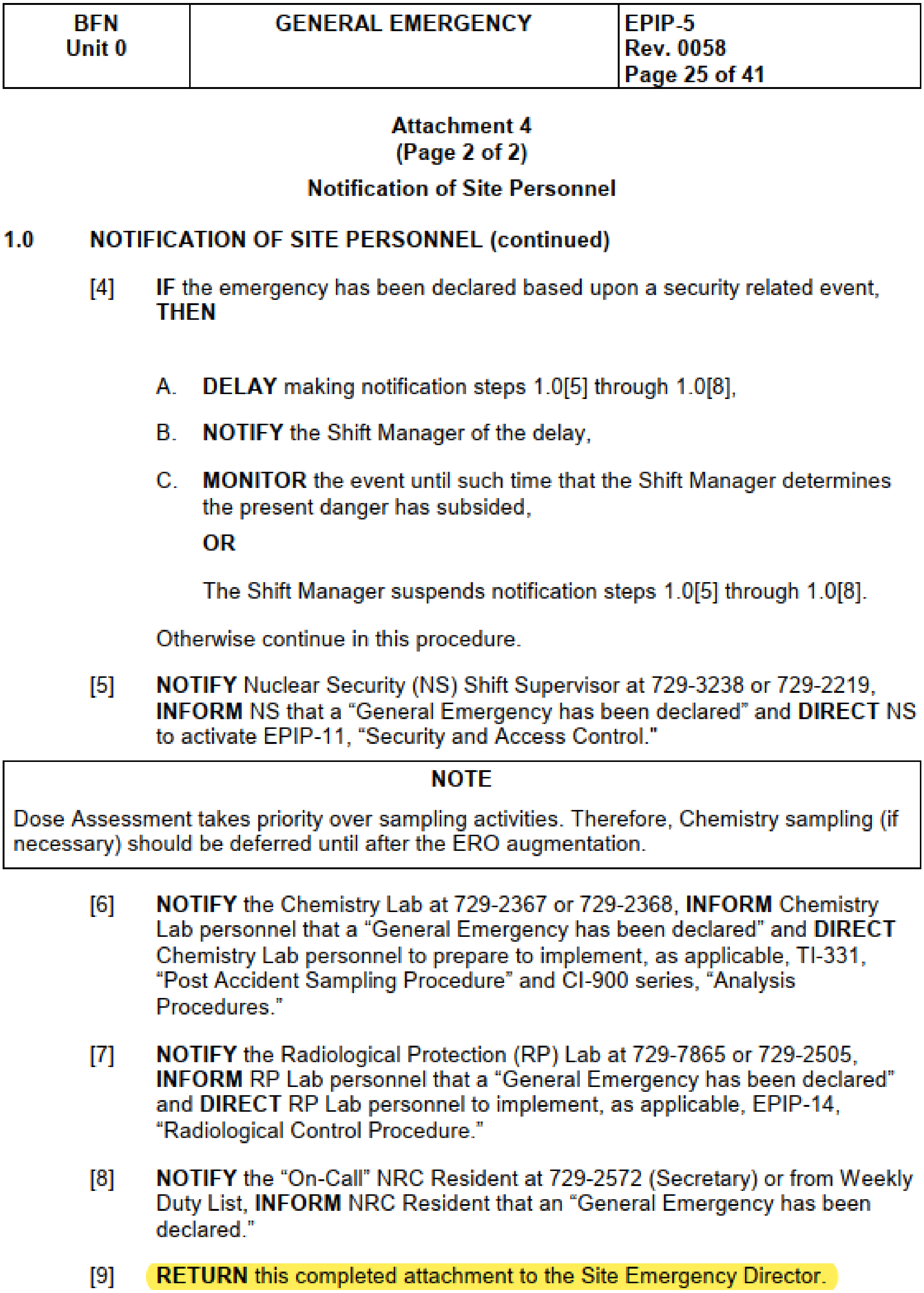
All notifications should be made utilizing the information located on Attachment 1, “General Emergency Initial Notification Form.”

[3] **OBTAIN** a copy of Attachment 1, “General Emergency Initial Notification Form” used for the State notification and available affected Unit Control Room logs.

A. **IF** the <REP 2> fax did not work in Attachment 2, “State of Alabama and Operations Duty Specialist (ODS) Notification.” **THEN**

FAX copies to the following locations by either using the preprogrammed button or dialing the number listed:

- Technical Support Center at 729-3742
- Central Emergency Control Center at 1-423-751-1682



BFN Unit 0	GENERAL EMERGENCY	EPIP-5 Rev. 0058 Page 25 of 41

Attachment 4
(Page 2 of 2)
Notification of Site Personnel

1.0 NOTIFICATION OF SITE PERSONNEL (continued)

[4] **IF** the emergency has been declared based upon a security related event, **THEN**

A. **DELAY** making notification steps 1.0[5] through 1.0[8],

B. **NOTIFY** the Shift Manager of the delay,

C. **MONITOR** the event until such time that the Shift Manager determines the present danger has subsided,

OR

The Shift Manager suspends notification steps 1.0[5] through 1.0[8].

Otherwise continue in this procedure.

[5] **NOTIFY** Nuclear Security (NS) Shift Supervisor at 729-3238 or 729-2219, **INFORM** NS that a "General Emergency has been declared" and **DIRECT** NS to activate EPIP-11, "Security and Access Control."

NOTE

Dose Assessment takes priority over sampling activities. Therefore, Chemistry sampling (if necessary) should be deferred until after the ERO augmentation.

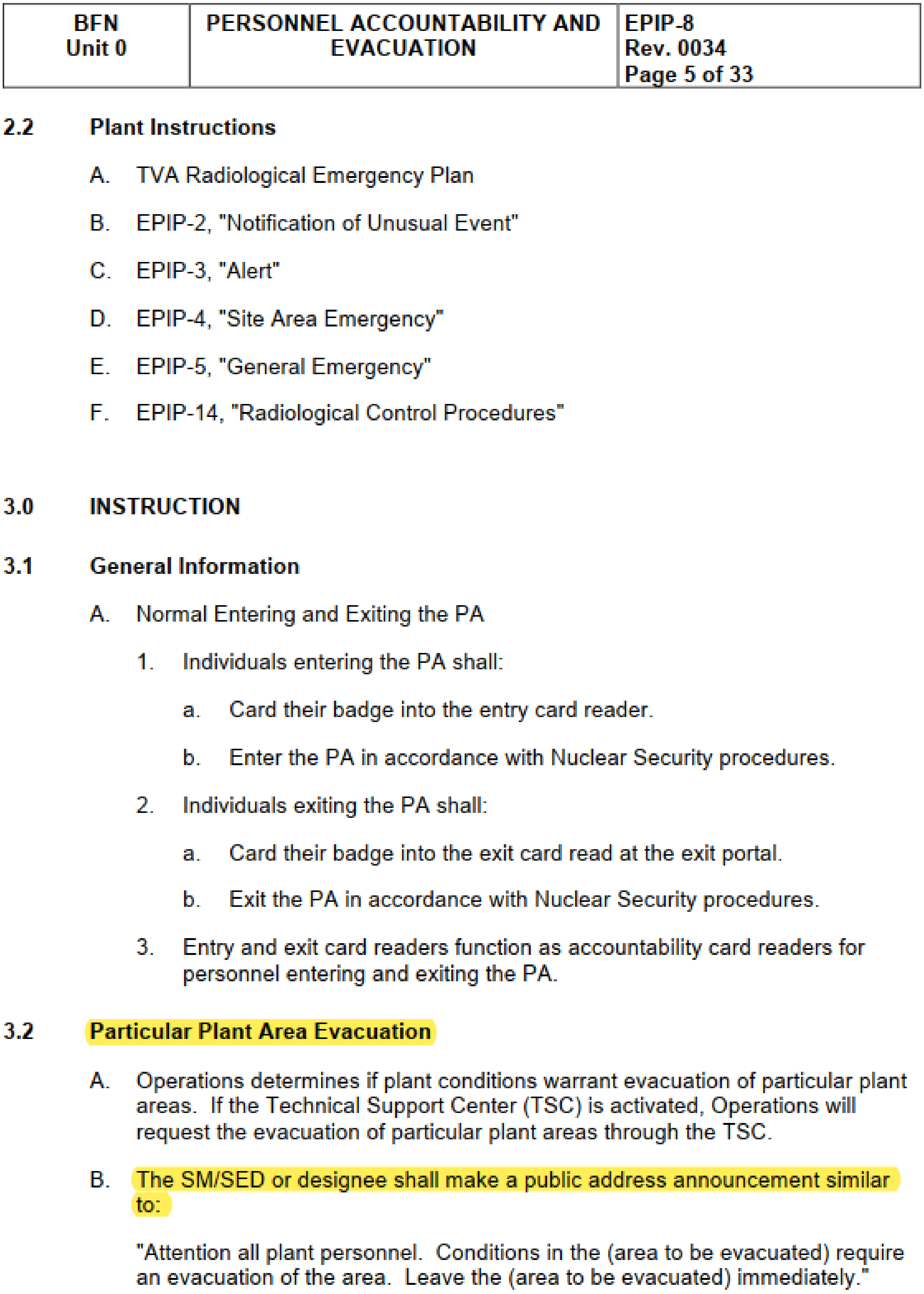
[6] **NOTIFY** the Chemistry Lab at 729-2367 or 729-2368, **INFORM** Chemistry Lab personnel that a "General Emergency has been declared" and **DIRECT** Chemistry Lab personnel to prepare to implement, as applicable, TI-331, "Post Accident Sampling Procedure" and CI-900 series, "Analysis Procedures."

[7] **NOTIFY** the Radiological Protection (RP) Lab at 729-7865 or 729-2505, **INFORM** RP Lab personnel that a "General Emergency has been declared" and **DIRECT** RP Lab personnel to implement, as applicable, EPIP-14, "Radiological Control Procedure."

[8] **NOTIFY** the "On-Call" NRC Resident at 729-2572 (Secretary) or from Weekly Duty List, **INFORM** NRC Resident that an "General Emergency has been declared."

[9] **RETURN** this completed attachment to the Site Emergency Director.

Excerpts from OPDP-8: Supports Distractor (C)



BFN **Unit 0**	**PERSONNEL ACCOUNTABILITY AND EVACUATION**	**EPIP-8** **Rev. 0034** **Page 5 of 33**

2.2 Plant Instructions

A. TVA Radiological Emergency Plan

B. EPIP-2, "Notification of Unusual Event"

C. EPIP-3, "Alert"

D. EPIP-4, "Site Area Emergency"

E. EPIP-5, "General Emergency"

F. EPIP-14, "Radiological Control Procedures"

3.0 INSTRUCTION

3.1 General Information

A. Normal Entering and Exiting the PA

1. Individuals entering the PA shall:

   a. Card their badge into the entry card reader.

   b. Enter the PA in accordance with Nuclear Security procedures.

2. Individuals exiting the PA shall:

   a. Card their badge into the exit card read at the exit portal.

   b. Exit the PA in accordance with Nuclear Security procedures.

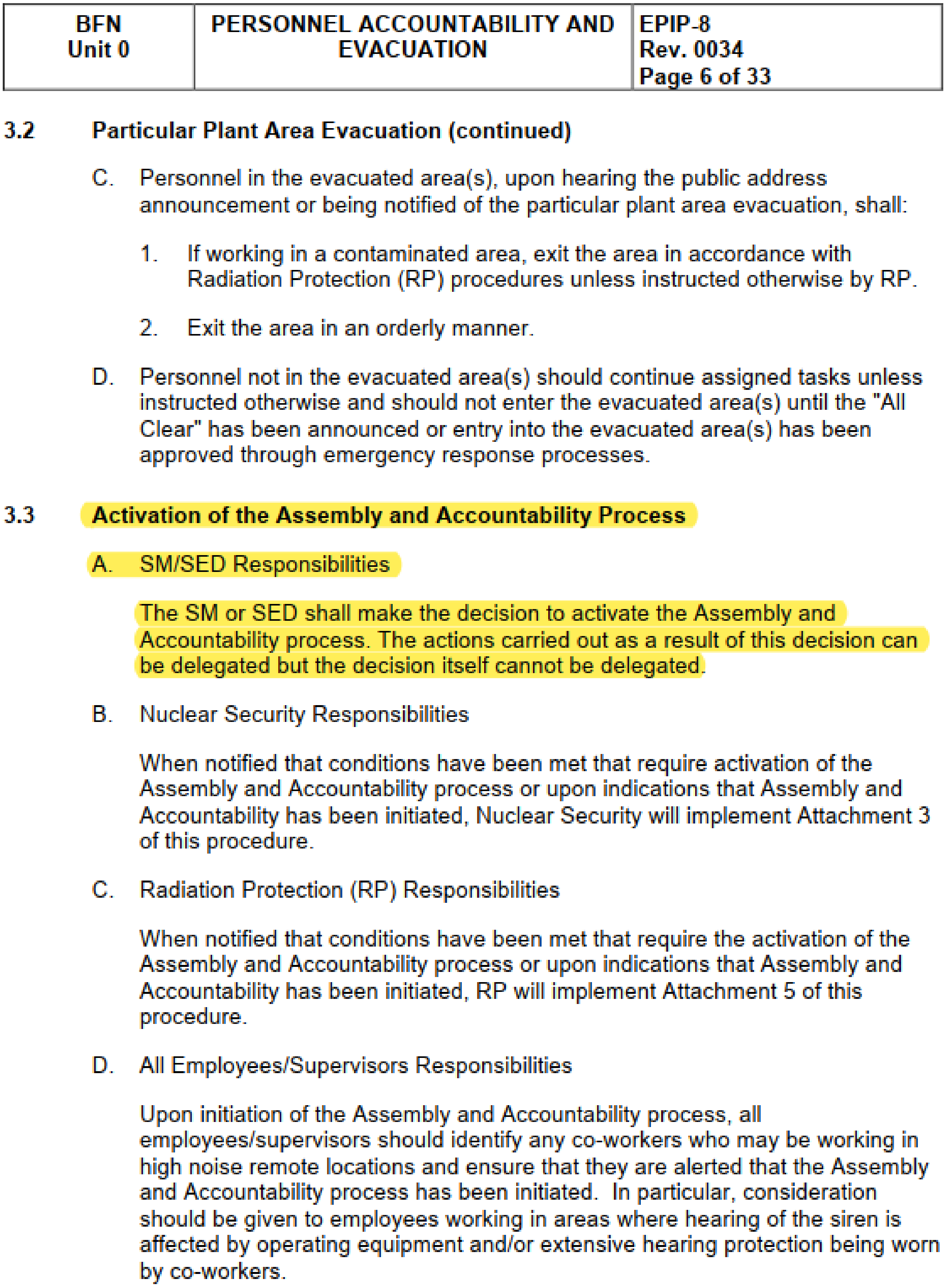
3. Entry and exit card readers function as accountability card readers for personnel entering and exiting the PA.

3.2 Particular Plant Area Evacuation

A. Operations determines if plant conditions warrant evacuation of particular plant areas. If the Technical Support Center (TSC) is activated, Operations will request the evacuation of particular plant areas through the TSC.

B. The SM/SED or designee shall make a public address announcement similar to:

"Attention all plant personnel. Conditions in the (area to be evacuated) require an evacuation of the area. Leave the (area to be evacuated) immediately."



BFN Unit 0	**PERSONNEL ACCOUNTABILITY AND EVACUATION**	**EPIP-8 Rev. 0034 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 Attachment 3 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 Attachment 5 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.

Facility: BFN Scenario Number: NRC-1
Scenario Source: NEW Op-Test Number: 22-04
Examiners: ____________________ Operators: NUSO: ____________________
____________________ OATC: ____________________
____________________ BOP: ____________________

Initial Conditions: 100 % Reactor Power. 2A RHR Pump is tagged for a scheduled outage. 2B EHC Pump is tagged for breaker maintenance.

Turnover: Reduce Reactor Power in accordance with the Reactivity Control Plan (RCP) and start 2C Condensate Booster Pump.

Critical Tasks:

1. To prevent an uncontrolled RPV depressurization when Reactor Water Level cannot be restored and maintained above (-) 122 inches, inhibit ADS.
2. With an injection system(s) operating and the Reactor shutdown and at pressure, after Reactor Water Level lowers to (-)162 inches, direct Emergency Depressurization before Reactor Water Level lowers to (-)180 inches.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	R-OATC R-NUSO	Reduce Reactor Power for Condensate Booster Pump Start
2.	N/A	N-BOP N-NUSO	Start 2C Condensate Booster Pump
3.	HP01	C-BOP MC-BOP TS-NUSO	Inadvertent High Pressure Coolant Injection (HPCI) Initiation
4.	FCV-64-61	TS-NUSO	Loss of Power to 2-FCV-74-61, RHR SYSTEM I DRYWELL SPRAY INBOARD VALVE
5.	ED05	C-OATC MC-OATC C-NUSO	Loss of Unit Preferred
6.	FW18 TH21	M-ALL	Earthquake / Feedwater Leak / LOCA / Emergency Depressurization
7.	RP07	C-OATC MC-OATC C-NUSO	Electrical Anticipated Transient Without Scram (ATWS)
8.	RC04	C-BOP MC-BOP C-NUSO	Reactor Core Isolation Cooling (RCIC) Controller Fails to Operate in Automatic

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

Events

1. In preparation for starting 2C Condensate Booster Pump, the crew will reduce Reactor Power to <95% as required by 2-OI-2, Condensate System.

2. The crew will return 2C Condensate Booster Pump to service in accordance with 2-OI-2, Condensate System.

3. HPCI will inadvertently initiate, requiring the crew to trip and lock out the HPCI Turbine. The Auxiliary Oil Pump will fail once the HPCI turbine is locked out. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.

4. Work Control will contact the NUSO to continue the Residual Heat Removal (RHR) outage and report that 2-FCV-74-61, RHR SYSTEM I DRYWELL SPRAY INBOARD VALVE, will be tagged out of service. The NUSO will address Technical Specifications.

5. The Unit Preferred (UPS) Motor Generator (MG) will trip, requiring the crew to respond in accordance with 2-AOI-57-4, Loss of Unit Preferred.

6. The crew will receive notification that an earthquake has occurred in the Northern Alabama area, which will result in a feedwater leak in the Turbine Building. The crew will respond in accordance with 0-AOI-100-5, Earthquake and Alarm Response Procedures; the crew will insert a manual Reactor SCRAM. The earthquake will also cause a Loss of Coolant (LOCA) in the Drywell, which will gradually worsen. The crew will respond in accordance with the Emergency Operating Instructions (EOIs), and eventually Emergency Depressurize the Reactor due to the inability to maintain Reactor Water Level.

7. When the Operator at the Controls (OATC) inserts the Reactor SCRAM an Electrical Anticipated Transient Without SCRAM (ATWS) will occur, requiring the OATC to take action to insert the Control Rods.

8. When Reactor Core Isolation Cooling (RCIC) is started for Reactor Water Level control, the RCIC flow controller will fail to operate in automatic, requiring the crew to take manual control of RCIC to provide Reactor injection.

The Scenario ends 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 any available low-pressure injection system.

Critical Tasks 2

1. **To prevent an uncontrolled RPV depressurization when Reactor Water Level cannot be restored and maintained above (-) 122 inches, inhibit ADS.**
    a. **Safety Significance**
    Maintain Adequate Core Cooling
    Prevent degradation of fission product barrier
    b. **Cues**
    Procedural Compliance
    Reactor Water Level
    c. **Measured by:**
    Observation – ADS logic is inhibited prior to an automatic initiation
    d. **Feedback:**
    RPV Pressure trend
    RPV Level trend
    ADS LOGIC BUS A/B INHIBITED annunciator (Panel 2-9-3C, Windows 18/31)
    e. **Critical Task Failure Criteria:**
    An automatic initiation of ADS occurs
2. **With an injection system(s) operating and the Reactor shutdown and at pressure, after Reactor Water Level lowers to (-) 162 inches, direct Emergency Depressurization before Reactor Water Level lowers to (-) 180 inches.**
    a. **Safety Significance**
    Maintain Adequate Core Cooling
    Prevent degradation of fission product barrier
    b. **Cues**
    Procedural Compliance
    Water Level trend
    c. **Measured by:**
    Observation – when it is determined that Reactor Water Level cannot be restored and maintained above (-) 180 inches, the crew proceeds without delay in a controlled manner to open 6 MSRVs.
    d. **Feedback**
    RPV Pressure trend
    MSRV status indications
    e. **Critical Task Failure Criteria:**
    Emergency Depressurization is not directed when Reactor Water Level cannot be restored and maintained above (-) 180 inches.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 1 Page 1 of 4

Event Description: Reduce Reactor Power for Condensate Booster Pump Start

Symptoms/Cues: Crew is cued by the turnover sheet or by the Simulator Operator as requested by the Chief Examiner

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, Reduce Reactor Power for Condensate Booster Pump Start after assuming the shift, request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to reduce Reactor Power.** **NOTE: The crew may elect to hold a re-focus reactivity brief for Reactor Power reduction.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to reduce Reactor Power in preparation for Condensate Booster Pump start.** **If contacted by the crew as the Reactor Building Assistant Unit Operator (AUO) acknowledge any direction given.**
	NUSO	In accordance with 2-OI-2, Condensate System, Precautions and Limitations 3.1.B, directs the Operator at the Controls (OATC) to reduce Reactor Power to less than 95% in accordance with 2-OI-68, Reactor Recirculation System, Reactivity Control Plan U2-NRC1-RCP-2204, and 2-GOI-100-12, Power Maneuvering.
	CREW	2-OI-2, Condensate System 3.0 Precautions and Limitations 3.1 General Precautions B. Reactor Power should be verified less than or equal to 95% prior to starting or stopping a Condensate Booster Pump or Reactor Feedwater Pump. Pumps must be removed from high pressure to low. 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.

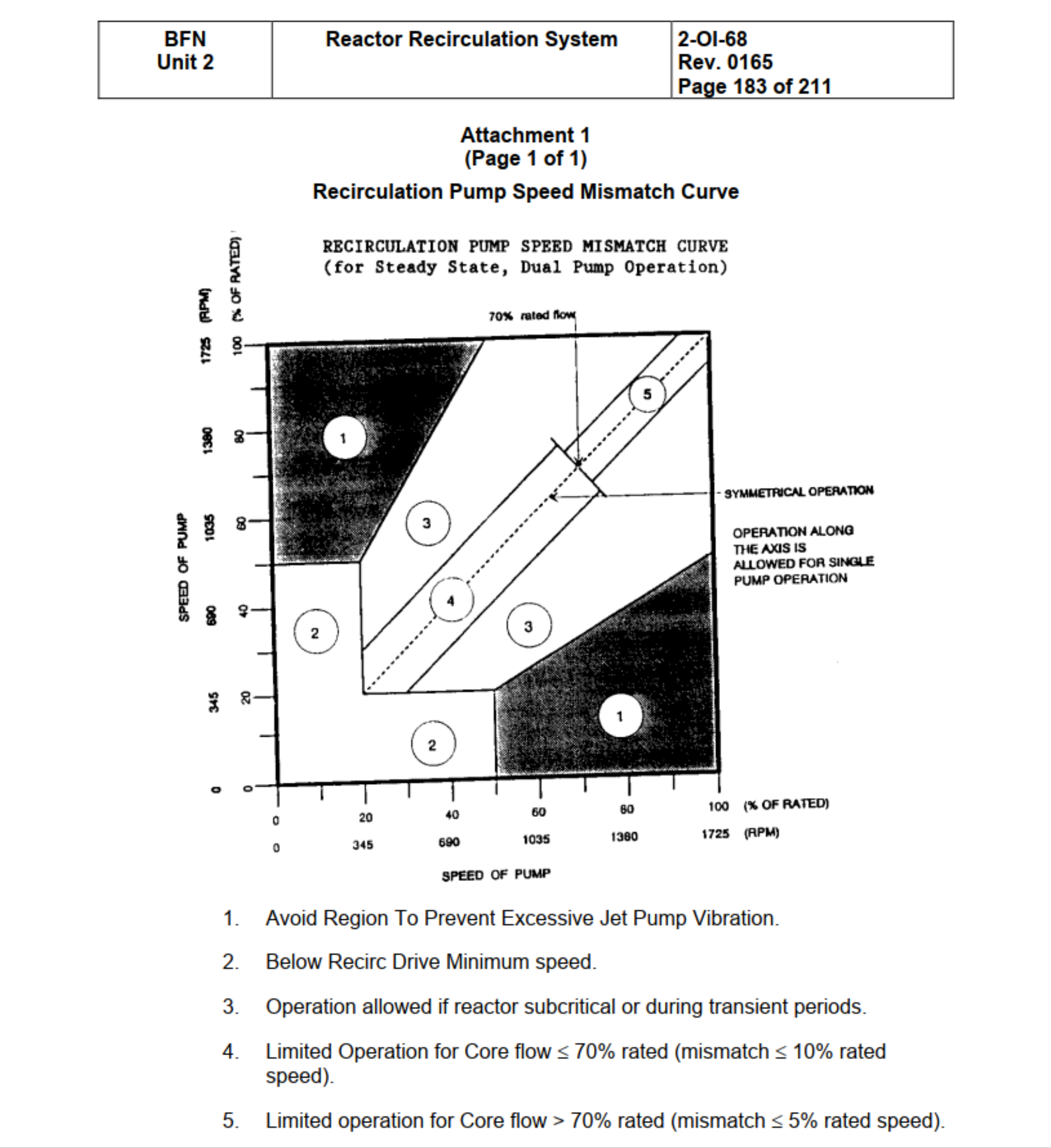
Op Test No.: 22-04	Scenario No. NRC-1	Event No.: 1	Page 2 of 4
Event Description: Reduce Reactor Power for Condensate Booster Pump Start			

Time	Position	Applicant's Actions or Behavior
	OATC	2-GOI-100-12, Power Maneuvering Section 7.0, Performance [7] **REDUCE** Reactor Power by combination of Control Rod insertions and Core Flow changes, as recommended by Reactor Engineer. **REFER TO** 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.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 1 Page 3 of 4

Event Description: Reduce Reactor Power for Condensate Booster Pump Start

Time	Position	Applicant's Actions or Behavior
	NRC	 **BFN Unit 2** | **Reactor Recirculation System** | **2-OI-68** **Rev. 0165** **Page 183 of 211** **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.: 22-04	Scenario No. NRC-1	Event No.: 1	Page 4 of 4

Event Description: Reduce Reactor Power for Condensate Booster Pump Start

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow [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 RAISE SLOW (MEDIUM), 2-HS-96-15A(15B). (Otherwise N/A) AND/OR • LOWER Recirc Pump 2B using SLOW(MEDIUM)(FAST), 2-HS-96-18A(18B)(18C). (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	**End of Event 1. Continue to Event 2, Start 2C Condensate Booster Pump.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 2 Page 1 of 4

Event Description: Start 2C Condensate Booster Pump

Symptoms/Cues: Crew is cued by the turnover sheet or by the Simulator Operator as requested by the Chief Examiner.

Time	Position	Applicant's Actions or Behavior
	NRC	**If the crew does not proceed to Event 2, Start 2C Condensate Booster Pump, 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 start 2C Condensate Booster Pump.** **If contacted by the crew as the Turbine Building AUO acknowledge any direction given.**
	NRC	**The Crew may elect to conduct a re-focus brief for starting 2C Condensate Booster Pump.**
	NUSO	Directs the Balance of Plant Operator (BOP) to start 2C Condensate Booster Pump in accordance with 2-OI-2, Condensate System.
	BOP	2-OI-2, Condensate System 3.0 Precautions and Limitations 3.1 General Precautions B. Reactor Power should be verified less than or equal to 95% prior to starting or stopping a Condensate Booster Pump or Reactor Feedwater Pump. Pumps must be removed from high pressure to low. 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.

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 2 Page 2 of 4

Event Description: Start 2C Condensate Booster Pump

Time	Position	Applicant's Actions or Behavior
	BOP	2-OI-2, Condensate System Section 5.3, Startup of Condensate/Condensate Booster Pumps [2] **START** a Condensate Booster Pump as follows: [2.1] **CHECK** the following initial conditions are satisfied: • Condensate System in operation in accordance with this instruction. • Three Demineralizers in service with their E Valves full open, to accommodate system flow and to prevent the Demineralizer Bypass Valve from opening. • **ENSURE** limiting conditions for Condensate Booster Pump operation in 0-OI-57A, Switchyard and 4160V AC Electrical System are met.
	Driver	**If contacted as the Turbine Building AUO concerning the status of the Condensate Demineralizer System, report that all Demineralizers are in service.**
	BOP	**NOTE** Unless otherwise directed, all control switch manipulations will be performed from Control Room Panel 2-9-6. [2.2] **REVIEW** the Precautions and Limitations in Section 3.0. [2.3] **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 3.1B) [2.4] **DISPATCH** personnel to the Condensate Booster Pump to be started. • CONDENSATE BOOSTER PUMP 2C
	Driver	**If contacted as the Turbine Building AUO, acknowledge any direction given.**

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 2 Page 3 of 4

Event Description: Start 2C Condensate Booster Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[2.4.1] **ENSURE** oil tank level is greater than 3/4.
	Driver	**If oil tank level is requested, report that the oil tank is more than 3/4 full.**
	BOP	[2.4.2] At Junction Box 2069 TB El 557', **START** CONDENSATE BOOSTER PUMP AUX OIL PUMP using the following, at TB El 557' South CBP area. • 2-HS-002-0143, CONDENSATE BOOSTER PUMP 2C AUX OIL PUMP
	Driver	**If contacted as the Turbine Building AUO to start 2C Condensate Booster Pump, insert event 2 and report that 2C Condensate Booster Pump Aux Oil Pump is running.**
	BOP	**NOTE** The pump vendor recommends venting Condensate Booster Pump casings anytime that the pumps are started Directs the Turbine Building AUO to vent 2C Booster Pump casing in accordance with step [2.5].
	Driver	**When directed to vent 2C Condensate Booster Pump casing in accordance with Step [2.5], report that venting is complete.**
	BOP	[2.6] **START** CONDENSATE BOOSTER PUMP, using one of the following. • 2-HS-2-68A, CONDENSATE BOOSTER PUMP 2C [2.7] **ADJUST**, as required, 2-FIC-2-29, CNDS FLOW CONTROL SHORT CYCLE, to maintain operating Condensate Booster Pump Amperage greater than 200 amps.

Op Test No.: 22-04	Scenario No. NRC-1	Event No.: 2	Page 4 of 4

Event Description: Start 2C Condensate Booster Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[2.8] **WHEN** Condensate Flow and Pressure stabilizes, **THEN RETURN** the E valves for in-service Demineralizer Vessels to normal configuration for plant conditions.
	Driver	**When directed to place the Demineralizer E Valves in a normal configuration, report that all E Valves are in a normal configuration for plant conditions.**
	BOP	[2.9] N/A [2.10] **ENSURE** Condensate Booster Pump Aux Oil Pumps shut down on the 2C Condensate Booster Pump. [2.11] **IF** a Condensate Pump or Condensate Booster Pump was started without a CONDENSATE PUMP COOLER FAN or CONDENSATE BOOSTER PUMP COOLER FAN, **THEN ENSURE** a cooler fan is started prior to exceeding 220ºF motor winding temperature.
	Driver	**If directed to verify that the 2C Condensate Booster Pump Aux Oil Pump has shut down, inform the operator that it has been shut down. If asked the status of the Cooler Fan, report that the Cooler Fan is running.**
	BOP	**NOTE** The Hydrogen Water Chemistry System shall be only lined up to inject hydrogen to operating Condensate Booster Pump(s). [3] **IF** desired, **ALIGN** Hydrogen Water Chemistry System to the Condensate System. **REFER** TO 2-OI-4, Hydrogen Water Chemistry System. (Otherwise N/A).
	Driver	**Acknowledge the direction to place Hydrogen Water Chemistry in service.**
	NRC	**End of Event 2. Request that the driver insert Event 3, Inadvertent High Pressure Coolant Injection (HPCI) Initiation**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 3 Page 1 of 4

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; HPCI will start and inject to the Reactor

Time	Position	Applicant's Actions or Behavior
	Driver	**When Requested by the Chief Examiner, insert Event 3 to cause an inadvertent initiation of HPCI.**
	BOP	Recognizes and reports that HPCI has initiated; acknowledges and reports the following alarm on Panel 2-9-3F: • HPCI TURB EXH DRAIN POT LEVEL HIGH, Window 33 Verifies that no Emergency Core Cooling System (ECCS) initiation signal exists and informs the NUSO.
	NUSO	Verifies that there no ECCS initiation signals and directs the BOP to trip and lock out HPCI in accordance with ODM 4.20, Strategies for Successful Transient Mitigation, Section 4.4.2, Operator Expectations for Reactor Water Level Control During Transients.
	NRC	**NOTE: The HPCI Aux Oil Pump will lose power when placed in "Pull to Lock".**
	BOP	Trips and locks out HPCI.
	OATC	Monitors Reactor Water Level, Reactor Power, and Reactor Pressure. Reports that Reactor Water Level Control has switched to Single Element Control (if applicable).
	Driver	**When contacted by the crew as the Outside/Work Control NUSO, acknowledge any direction given.** **If contacted as an AUO to go to 2A 250V DC RMOV Board to open the breaker for the HPCI Minimum Flow Valve, acknowledge the direction.** **If contacted as the Reactor Engineer, acknowledge any direction given.**
	NRC	**If Reactor Water Level Control transfers to Single Element Control, the crew may elect to transfer Reactor Water Level Control back to Three Element Control.** **If the crew does not decide to place Reactor Water Level Control in Three Element Control, end of Event 3; request that the driver insert Event 4 after the Tech Spec call is complete.**

Op Test No.: 22-04	Scenario No. NRC-1	Event No.: 3	Page 2 of 4

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

Time	Position	Applicant's Actions or Behavior
	NUSO	If necessary, directs the OATC to transfer Reactor Water Level Control to Three Element Control in accordance with 2-OI-3, Reactor Feedwater System, Section 8.31, Transferring RFWCS Between Single Element and Three Element Control.
	OATC	2-OI-3, Reactor Feedwater System Section 8.31, Transferring RFWCS Between Single Element and Three Element Control **NOTES** 1) This section is performed at Panel 2-9-5. 2) The RFW Control System will not allow THREE ELEMENT operation until ALL of the following permissives are met: • Total Steam Flow greater than 16% • Reactor Water Level Control (PDS), 2-LIC-46-5, in AUTO <u>and</u> at least one RFPT Speed Control (PDS) in AUTO • A Valid Total Steam Flow signal exists (Four good steam line flows, or at least two good steam line flow signals <u>and</u> a valid Turbine First Stage pressure signal) • A Valid Total Feedwater Line Flow signal exists (Two good Feedwater line flow signals, or one good Feedwater line flow <u>and</u> three valid individual RFP flow signals) • At least one valid Narrow Range Level signal [1] **IF** desired to transfer level control from Single Element to Three Element, **THEN** (Otherwise N/A): [1.1] **ENSURE** conditions in Note 2 are met for placing level control in Three Element. [1.2] **CHECK** stable Steam Flow and Feedwater Flow. [1.3] **DEPRESS** 2-HS-46-6/3, THREE ELEMENT pushbutton, and **CHECK** green backlight for pushbutton illuminates. [1.4] **CHECK** green backlight for 2-HS-46-6/1, SINGLE ELEMENT pushbutton, is extinguished. [1.5] **CHECK** Reactor Water Level stable. [2] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04	Scenario No. NRC-1	Event No.: 3	Page 3 of 4

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

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. NOTE: LCO 3.0.4.b is not applicable to HPCI. Condition: C. HPCI System inoperable. D. HPCI System inoperable <u>AND</u> Condition A entered.	
	NUSO	**REQUIRED ACTION:** C.1 Verify by administrative means RCIC System is OPERABLE. <u>AND</u> C.2 Restore HPCI System to OPERABLE status.	**COMPLETION TIME:** C.1 – Immediately C.2 – 14 days

Op Test No.: 22-04	Scenario No. NRC-1	Event No.: 3	Page 4 of 4

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	**COMPLETION TIME:** C.1 – Immediately C.2 – 14 days
	NUSO	**REQUIRED ACTION:** D.1 Restore HPCI System to OPERABLE status. AND D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status	**COMPLETION TIME:** D.1 – 72 hours D.2 – 72 hours
	Driver	**If contacted as the Work Control/Outside NUSO to prepare Protected Equipment tags for RCIC, acknowledge the direction.** **If contacted as the Work Control/Outside NUSO to investigate the failure of HPCI to isolate acknowledge the direction.**	
	NRC	**End of Event 3. Request that the driver insert Event 4, Loss of Power to 2-FCV-74-61, RHR SYSTEM I DRYWELL SPRAY INBOARD VALVE**	

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 4 Page 1 of 1

Event Description: Loss of Power to 2-FCV-74-61, RHR SYSTEM I DRYWELL SPRAY INBOARD VALVE

Symptoms/Cues: Event is cued by the simulator booth when requested by the Chief Examiner

Time	Position	Applicant's Actions or Behavior	
	Driver	**When requested by the Chief Examiner, contact the NUSO the Work Control and state that 2-FCV-74-61, RHR SYSTEM I DRYWELL SPRAY INBOARD VALVE will be tagged out of service for RHR Loop I outage.** **After the NUSO acknowledges the report, after 30 seconds insert Event 3 to remove power from 2-FCV-74-61.**	
	NUSO	Informs the crew that 2-FCV-74-61 will be inoperable. Technical Specification 3.6.2.5 – RHR Drywell Spray LCO 3.6.2.5 Four RHR drywell spray subsystems shall be OPERABLE. APPLICABILITY: MODES 1, 2, and 3. Condition: B. Two RHR Drywell Spray subsystems inoperable.	
	NUSO	**REQUIRED ACTION** B.1 Restore one RHR Drywell Spray subsystem to OPERABLE status.	**COMPLETION TIME** 7 days
	NRC	**End of Event 4. Request that the driver insert Event 5, Loss of Unit Preferred.**	

Op Test No.: 22-04	Scenario No. NRC-1	Event No.: 5	Page 1 of 3

Event Description: Loss of Unit Preferred

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; Unit Preferred Power will be lost

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause a loss of Unit Preferred Power.**
	NRC	**Numerous alarms will be received as a result of this malfunction, and not all will be covered in this scenario guide.**
	BOP	Acknowledges alarms and reports the following alarm: • UNIT PREFERRED SUPPLY ABNORMAL, 2-9-8B, Window 35
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure and 2-AOI-57-4, Loss of Unit Preferred.
	BOP	2-9-8B, Alarm Response Procedure UNIT PREFERRED SUPPLY ABNORMAL, Window 35 Operator Action: A. **IF** 120V AC Unit Preferred is lost, **THEN REFER** TO 2-AOI-57-4, Loss of Unit Preferred. B. **NOTIFY** Unit NUSO, Unit 1 and Unit 3.
	Driver	**If contacted as the Unit 1 or 3 NUSO, acknowledge any information provided.**
	BOP	C. After determining situation, **REFER TO** appropriate portion of 0-OI-57C, 208/120V AC Electrical System. D. N/A E. N/A
	OATC	2-AOI-57-4, Loss of Unit Preferred 4.0 Operator Actions: 4.1 Immediate Actions: None

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 5 Page 2 of 3

Event Description: Loss of Unit Preferred

Time	**Position**	**Applicant's Actions or Behavior**
	OATC	4.2 Subsequent Actions [1] **ENSURE** the following: • No Control Rod movement as indicated by stable Reactor Power • RFW Control System is maintaining Reactor Water Level • Recirc Flow Control System maintaining Recirc Pump speeds • EHC Control System maintaining Reactor Pressure and Turbine Control parameters • **ENSURE** TIP ISOLATION [2] **IF** ANY EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). (Otherwise N/A) **CAUTION** While RPIS and the Process Computer are inoperable, Control Rod movement may only be performed by manual Reactor SCRAM. **REFER TO** 2-AOI-85-4, Loss of RPIS. [3] **IF** Control Rod movement is required while RPIS and the Process Computer are inoperable, **THEN INSERT** a MANUAL SCRAM. REFER TO 2-AOI-100-1, Reactor SCRAM. (Otherwise N/A) **NOTE** 2-FCV-85-11A/B, CRD SYS FLOW CONTROL VALVE 1A/B closes on a loss of power to Panel 2-9-9 Cabinet 6 Unit Preferred resulting in a loss of normal cooling water flow to CRD seals. CRD Temperatures should be monitored and operation with 2-FCV-85-11A/B closed, limited to less than 1 hour. 2-FCV-85-11A/B can be manually opened, if required. REFER TO 2-OI-85, Control Rod Drive System.

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Event Description: Loss of Unit Preferred

Time	**Position**	**Applicant's Actions or Behavior**
	OATC	[4] **PERFORM** the following for the CRD System: [4.1] **MONITOR** CRD temperatures while 2-FCV-85-11(A/B) are closed. [4.2] **IF** CRD Seal Temperatures rise to the alarm setpoint OR the Unit Preferred system cannot be restored within one hour, **THEN DISPATCH** personnel to **MANUALLY OPEN** 2-FCV-85-11(A/B). REFER TO 2-OI-85, Control Rod Drive System. (Otherwise N/A) [4.3] **IF** Cabinet 5 fails to transfer or is otherwise de-energized, and Cabinet 6 is energized **THEN PLACE** controller 2-FIC-85-11 in manual and adjust until normal Drive Water Pressure is obtained.
	Driver	**If contacted as Electrical Maintenance or the Work Control NUSO to investigate, acknowledge the direction.**
	NRC	**End of Event 5. Request that the driver insert Event 6, Earthquake / Feedwater Leak / LOCA / Emergency Depressurization.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 1 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; Feed Flow indications will indicate a leak in the system

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, contact the Control Room as the Shift Manager and inform the NUSO that Unit 1 has received the following alarms:** • **START OF STRONG MOTION ACCELEROGRAPH, 1-9-22C, Window 5** • **1/2 SSE RESPONSE SPECTRUM EXCEEDED, 1-9-22C, Window 6** **Relay the following information to the NUSO:** **"The National Earthquake Information Center has confirmed that an earthquake has occurred in Northern Alabama, centered approximately 5 miles east of Athens, AL. Plant data is still being analyzed.** **Unit 1 has the lead on performing the actions of 0-AOI-100-5, Earthquake.**
	Driver	**Once the NUSO has received the earthquake report, insert Event 6 to start the Feedwater Leak.**
	NRC	**When the Crew inserts a manual Reactor SCRAM based on the Feedwater Leak, a Recirc Pump Suction Line break will occur to cause a leak inside the Drywell.**
	NRC	**Numerous alarms will be received during this event, and not all are covered in this guide. Depending on when the Crew discovers the Feedwater Leak, the alarms received before the Reactor SCRAM may vary.**
	BOP	Announces and reports the following alarms as they are received: • CONDENSATE DEMIN ABNORMAL, 2-9-6B, Window 6 • RFPT A / B / C ABNORMAL, 2-9-6C, Window 1 / 8 / 15 • CONDENSATE BOOSTER PUMP A / B / C SUCTION PRESSURE LOW, 2-9-6A, Window 19 / 20 / 21 • REACTOR WATER LEVEL ABNORMAL, 2-9-5A, Window 8

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 2 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	CREW	As the Feedwater Leak is ramped in, given the alarms and indications in the Control Room concludes there is a Feedwater Leak and inserts a manual Reactor SCRAM.
	NRC	**The Crew may enter 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level high/Low based on alarms received.** **Depending on when the Crew diagnoses the Feedwater Leak, they may elect to perform a runback before the Reactor SCRAM.**
	OATC	2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level high/Low 4.0 Operator Actions 4.1 Immediate Actions [1] **IF** Condensate Booster Pump(s) OR Reactor Feedwater Pump(s) Low Suction Pressure annunciators are in alarm due to loss of Condensate Pump(s), **THEN PERFORM** the following: [1.1] **DEPRESS** RECIRC PUMPS MID POWER RUNBACK push-button, 2-HS-68-43. [1.2] **CHECK** Reactor Power lowers to 66% or as directed by US. [2] **IF** Reactor Water Level approaches +2 inches, **THEN INITIATE** a manual Reactor SCRAM.
	NUSO	Directs the OATC to insert a Mid-Power or Core Flow Runback (If the crew elects to insert a runback before the Reactor SCRAM).

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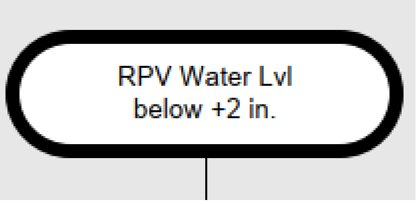
Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

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). [2] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 4 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	OATC	[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%
	OATC	2-AOI-100-1, Reactor SCRAM Operator Actions 4.1 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.
	NRC	**Event 7, Electrical Anticipated Transient Without SCRAM, is automatically on simulator setup. See page 37 of 40 for Event 7 actions.**
	NUSO	After all Control Rods are in (see Event 7), enters 2-EOI-1, RPV Control based on Reactor Water Level < 2". 

Form 3.3-2 Required Operator Actions

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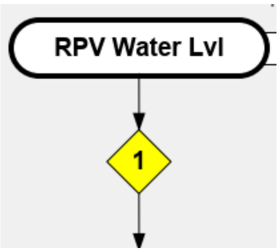
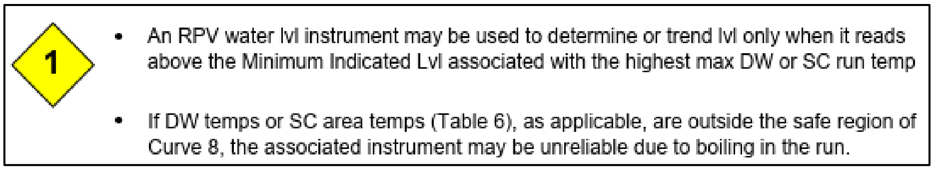
Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>




- 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.

RC/L-1

ENSURE each as required:

- PCIS isolations (Groups 1, 2, and 3)
- ECCS
- RCIC

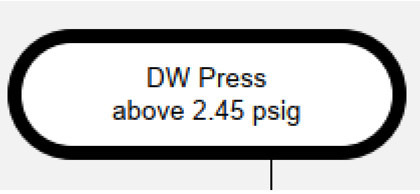
RC/L-2

<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Raising RPV Water Level above (+) 51 in. will facilitate use of shutdown cooling, steam-driven injection systems, or Alternate Injection Subsystems (Table L-2)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>PC Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 6 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant’s Actions or Behavior
	NUSO	Directs the OATC to isolate the Feedwater Leak by closing the High Pressure Heater Outlet Valves, and may direct Feed and Condensate Pumps tripped.
	NRC	**The Feedwater Leak malfunction is on the common header downstream of the High Pressure Heaters. HPCI and RCIC will not feed the leak, but the Emergency High Pressure Makeup Pump (EHPM) WILL feed the leak.**
	OATC	Isolates the Feedwater Leak by closing the High Pressure Heater Outlet Valves.
	NUSO	May direct the OATC or BOP to inject to the Reactor using 2-EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System. Because HPCI is not available, directs the BOP to maintain Reactor Water Level using 2-EOI-Appendix-5C, Injection System Lineup RCIC.
	NRC	**Event 8, Reactor Core Isolation Cooling (RCIC) Controller Fails to Operate in Automatic is automatically inserted when RCIC Speed exceeds 500 rpm on startup. No action is required by the driver to insert Event 8. See page 38 of 40 for Event 8 actions.**
	BOP	Based on Drywell Pressure and Temperature indications, determines that there is a leak in the Drywell. Acknowledges and reports the following alarms as they are received: • PRIMARY CONTAINMENT N2 PRESSURE HIGH, 2-9-3B, Window 10 • DRYWELL NORMAL OPERATING PRESSURE HIGH, 2-9-3B, Window 19 • DRYWELL PRESSURE APPROACHING SCRAM, 2-9-3B, Window 30 • DRYWELL PRESSURE ABNORMAL, 2-9-5B, Window 31
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure(s) and Abnormal Operating Instructions. When Drywell Pressure exceeds 2.45 psig, enters 2-EOI-2, Primary Containment Control. 

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 7 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

<table>
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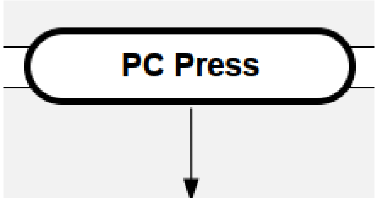
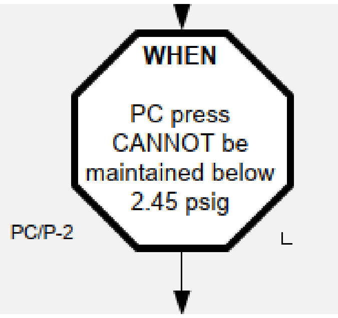
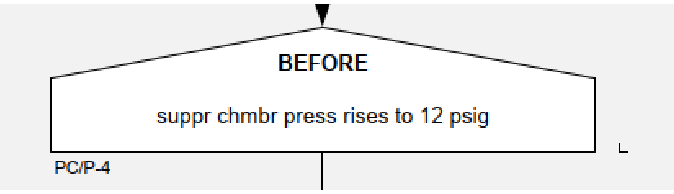
<tr><td></td><td>NUSO</td><td>


<p>PC/P-1</p>
<p>MONITOR and CONTROL PC Pressure below 2.45 psig using the Vent System (AOI-64-1).</p>


<p>PC/P-3</p>
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>PC 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>PC Pressure CANNOT be maintained above (-) 2 psig</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Chamber Pressure approaches 0 psig</td><td>BEFORE 0 psig, STOP Suppression Chamber Sprays</td></tr>
<tr><td>Drywell Pressure approaches 0 psig</td><td>BEFORE 0 psig, STOP Drywell Sprays</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 8 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / 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>


PC/P-5

<table>
<tr><td colspan="2">INITIATE Suppression Chamber Sprays
➢ Use only sources NOT required for continuous RPV injection (Appendix 17C)</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Needed to augment Suppression Chamber Sprays</td><td>Use sources external to PC only if PC Water Level and Drywell Pressure can be restored and maintained in the safe area of Curve 7</td></tr>
</table>

Directs the BOP to spray the Suppression Chamber in accordance with 2-EOI-17C, RHR System Operation Suppression Chamber Sprays.
</td></tr>
<tr><td></td><td>BOP</td><td>
2-EOI-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 OUTBOARD INJECTION VALVE BYPASS SELECT in BYPASS
- PLACE 2-HS-74-155B, LPCI SYS II OUTBOARD INJECTION VALVE BYPASS SELECT in BYPASS

[3] N/A

[4] N/A
</td></tr>
</table>

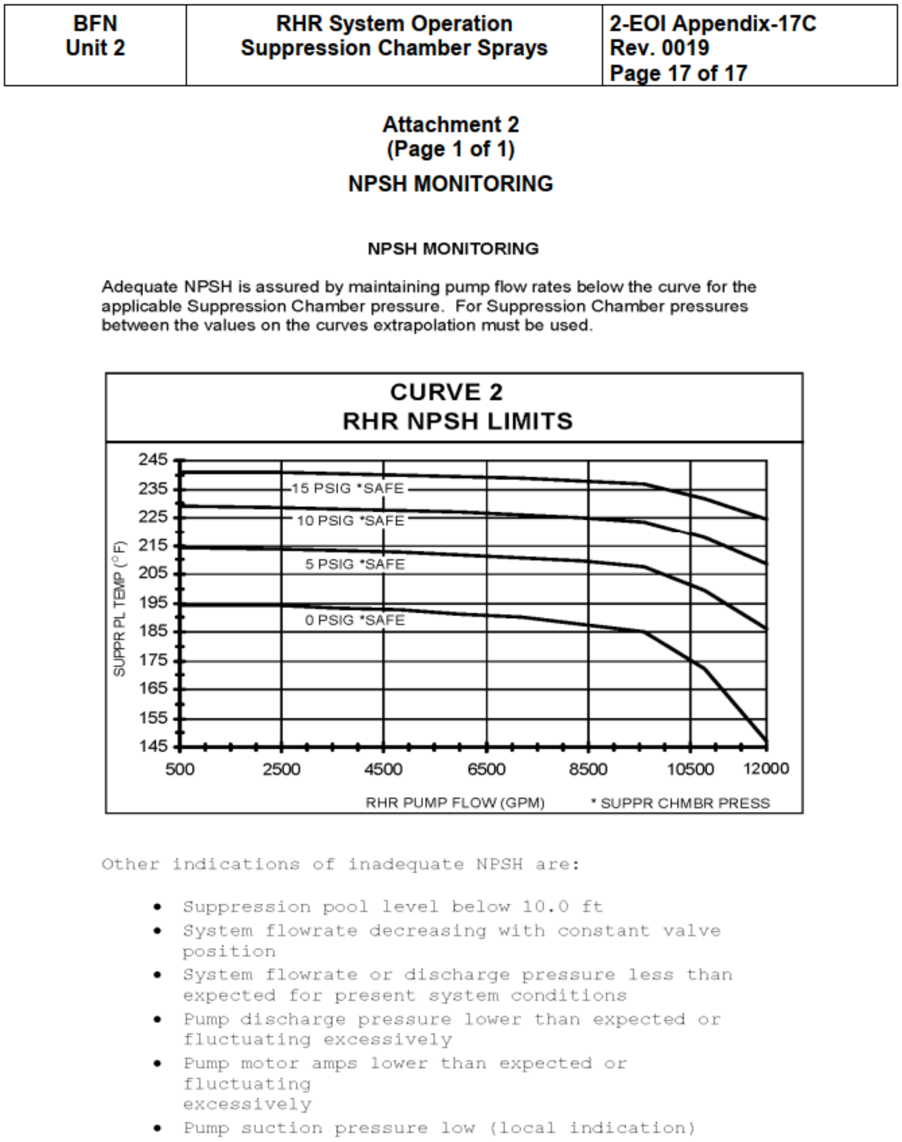
Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 9 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	BOP	[5] **INITIATE** Suppression Chamber Sprays as follows: [5.1] **ENSURE** at least one RHRSW Pump supplying each EECW header. [5.2] **IF** <u>EITHER</u> of the following exists: • LPCI initiation signal is <u>NOT</u> present, **OR** • Directed by NUSO, **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 SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, switch in SELECT. [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 ISOLATION VALVE. [5.7] **OPEN** 2-FCV-74-58 (72), RHR SYS 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.

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 10 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	BOP	

BFN Unit 2	RHR System Operation Suppression Chamber Sprays	2-EOI Appendix-17C Rev. 0019 Page 17 of 17

Attachment 2
(Page 1 of 1)

NPSH MONITORING

NPSH MONITORING

Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used.

CURVE 2
RHR NPSH LIMITS

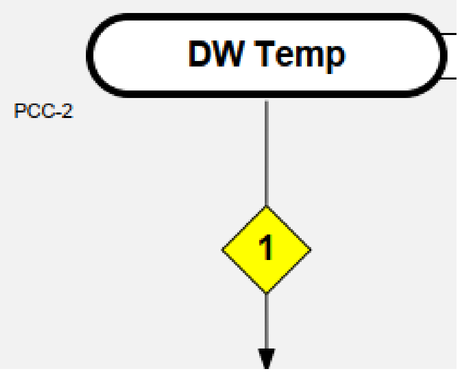
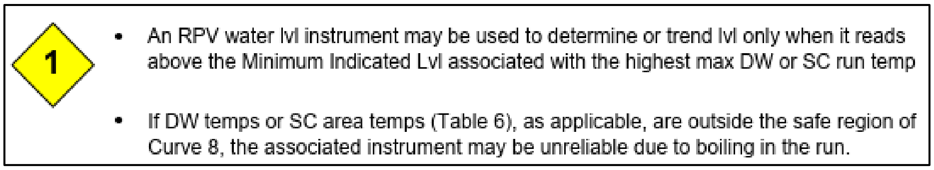
Other indications of inadequate NPSH are:

- Suppression pool level below 10.0 ft
- System flowrate decreasing with constant valve position
- System flowrate or discharge pressure less than expected for present system conditions
- Pump discharge pressure lower than expected or fluctuating excessively
- Pump motor amps lower than expected or fluctuating excessively
- Pump suction pressure low (local indication)

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 11 of 20

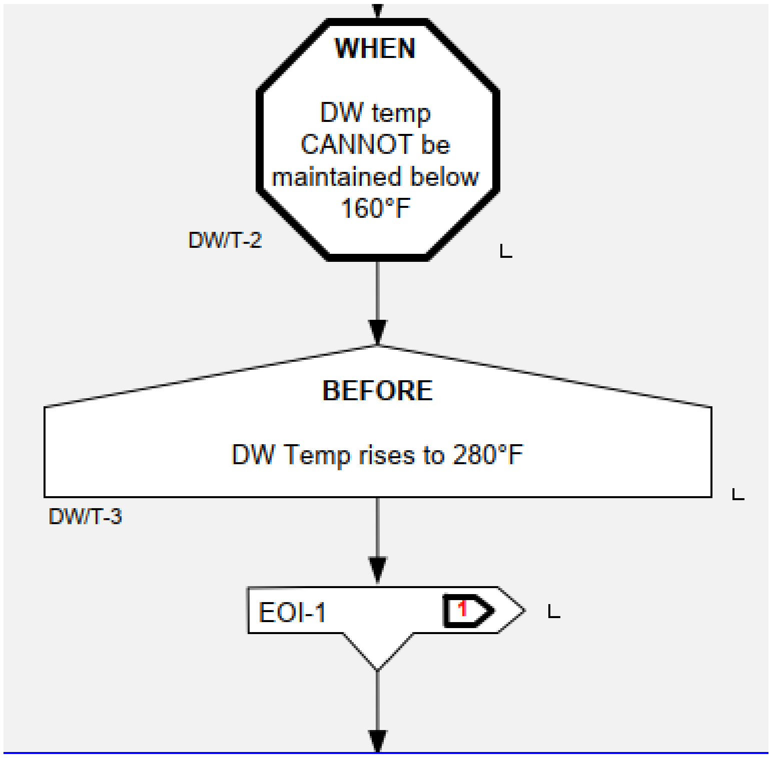
Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	BOP	[5.12] **ENSURE** RHRSW Pump supplying desired RHR Heat Exchanger(s). [5.13] **THROTTLE** the following in-service RHRSW Outlet Valves to obtain 1700 to 4500 gpm RHRSW Flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE [5.14] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	Driver	**If contacted as Chemistry, acknowledge any report(s) given.**
	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. DW/T-1 **MONITOR** and **CONTROL** Drywell Temperature below 160°F using all available Drywell Cooling

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 12 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	
	NUSO	When Suppression Chamber Pressure exceeds 12 psig OR Drywell Temperature cannot be maintained below 160 ºF AND Drywell Temperature is in the safe area of Curve 5, directs the BOP to spray the Drywell in accordance with 2-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].

Op Test No.: 22-04	Scenario No. NRC-1	Event No.: 6	Page 13 of 20

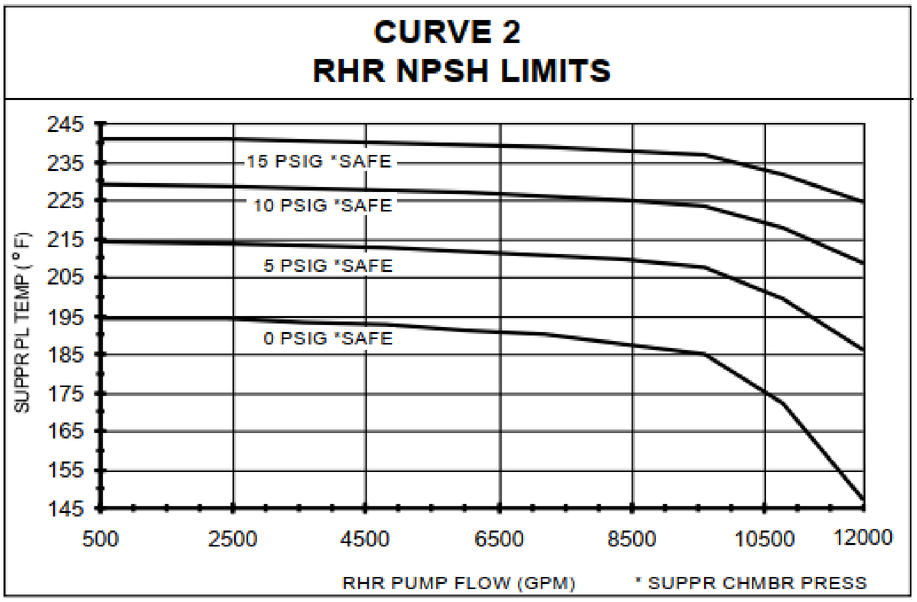
Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	BOP	[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 SYS I OUTBD INJECTION VALVE BYPASS SELECT IN BYPASS • **PLACE** 2-HS-74-155B, LPCI SYS II OUTBD INJECTION VALVE BYPASS SELECT IN BYPASS [3] **ENSURE** Recirc Pumps and Drywell Blowers shutdown. [4] N/A [5] N/A [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 NUSO, **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-74-67, RHR SYSTEM II LPCI INBOARD INJECT VALVE, is OPEN, **THEN ENSURE CLOSED** 2-FCV-74-66, RHR SYSTEM II LPCI OUTBOARD INJECT VALVE. [6.5] **ENSURE** OPERATING the desired System II RHR Pump(s) for Drywell Spray.
	NRC	**NOTE: Loop I of Drywell Sprays is not available due to Event 4.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 14 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	BOP	[6.6] **OPEN** the following valves: • 2-FCV-74-74, RHR SYSTEM II DRYWELL SPRAY OUTBOARD VALVE • 2-FCV-74-75, RHR SYSTEM II DRYWELL SPRAY INBOARD VALVE [6.7] **ENSURE CLOSED** 2-FCV-074-0030, RHR SYSTEM II MINIMUM FLOW VALVE. [6.8] **IF** Additional Drywell Spray Flow is necessary, **THEN PLACE** the second System II RHR Pump in service. [6.9] **MONITOR** RHR Pump NPSH using Attachment 2. **Attachment 2** **(Page 1 of 1)** **NPSH MONITORING** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used.  **CURVE 2** **RHR NPSH LIMITS**

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 15 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

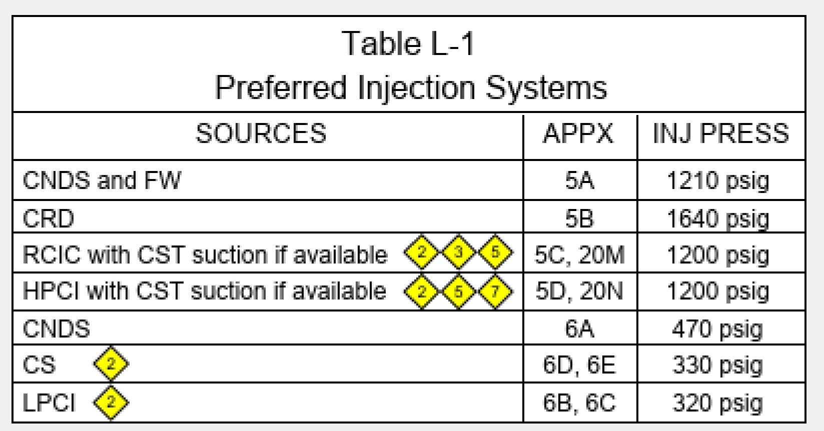
Time	Position	Applicant's Actions or Behavior
	BOP	[6.10] **ENSURE** RHRSW Pump supplying desired RHR Heat Exchanger(s). [6.11] **THROTTLE** the following in-service RHRSW Outlet Valves to obtain 1700 to 4500 gpm RHRSW flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE [6.12] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	Driver	**If contacted as Chemistry, acknowledge any information or direction given.**
	BOP	Monitors Drywell Pressure and if necessary, stops Drywell Spray Flow if Drywell Pressure approaches 0 psig.
	NUSO	(Continuing actions in accordance with 2-EOI-1, RPV Control Modes 1-3) RC/L-3 **RESTORE** and **MAINTAIN** RPV Water Level between +2 in. and +51 in. with ANY Preferred Injection Systems (Table L-1) ➢ OK to use ANY Alternate Injection Subsystems (Table L-2) 

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 ◇5	5C, 20M	1200 psig
HPCI with CST suction if available ◇2 ◇5 ◇7	5D, 20N	1200 psig
CNDS	6A	470 psig
CS ◇2	6D, 6E	330 psig
LPCI ◇2	6B, 6C	320 psig

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 16 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

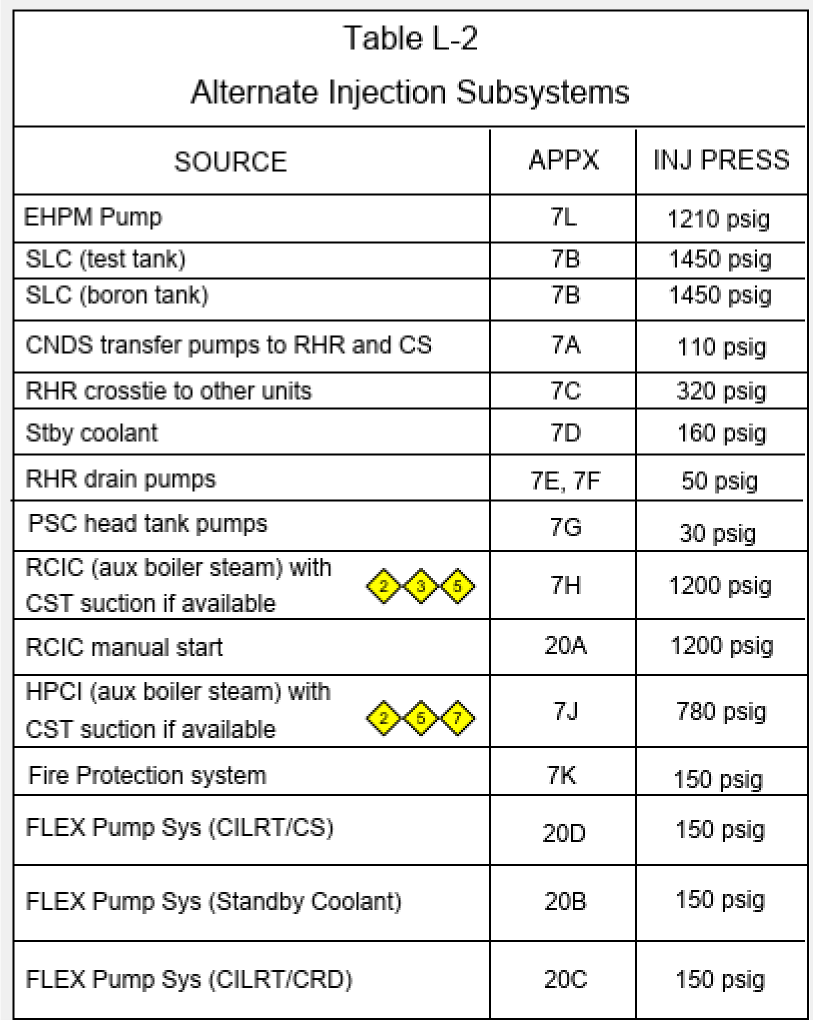
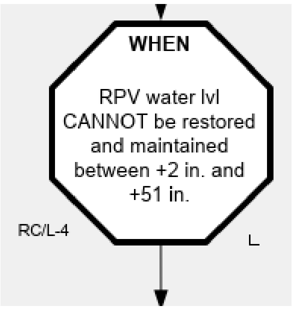
Time	**Position**	**Applicant's Actions or Behavior**
	NUSO	

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 ◇5	7H	1200 psig
RCIC manual start	20A	1200 psig
HPCI (aux boiler steam) with CST suction if available ◇2 ◇5 ◇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

	NUSO	

WHEN

RPV water lvl CANNOT be restored and maintained between +2 in. and +51 in.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 17 of 20

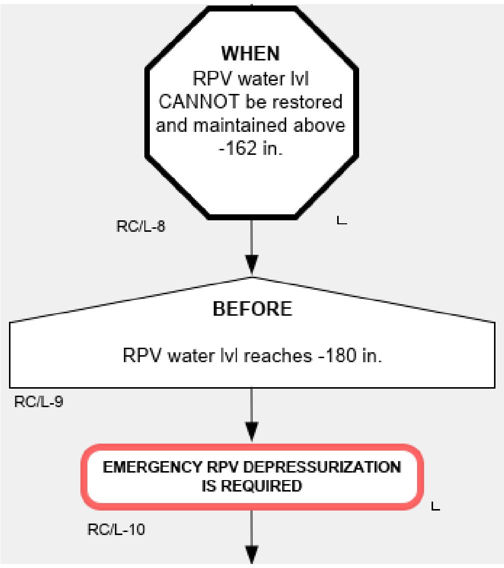
Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>RC/L-5
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level can be restored and maintained above (+) 2 inches</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/L-6
<table>
<tr><td>IF RPV Water Level CANNOT be maintained above (-) 122 inches
THEN inhibit ADS</td></tr>
</table>
</td></tr>
<tr><td></td><td>CREW</td><td>Critical Task:
To prevent an uncontrolled RPV depressurization when Reactor Water Level cannot be restored and maintained above (-) 122 inches, inhibit ADS.

Critical Task Failure Criteria:
An automatic initiation of ADS occurs.</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to inhibit ADS when it is determined that Reactor Water Level cannot be maintained above (-) 122 inches.</td></tr>
<tr><td></td><td>NUSO</td><td>RC/L-7
<table>
<tr><td colspan="2">RESTORE and MAINTAIN RPV Water Level above (-) 162 in. with ANY Preferred Injection Systems (Table L-1)
➢ OK to use ANY Alternate Injection Subsystems (Table L-2)</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>2 or more injection subsystems (Table L-3) CANNOT be lined up</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 18 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	
	CREW	**Critical Task:** **With an injection system(s) operating and the Reactor shutdown and at pressure, after Reactor Water Level lowers to (-) 162 inches, direct Emergency Depressurization before Reactor Water Level lowers to (-) 180 inches.** **Critical Task Failure Criteria:** **Emergency Depressurization is not directed when Reactor Water Level cannot be restored and maintained above (-) 180 inches.**
	NUSO	Determines that Emergency Depressurization is required. 2-EOI-1, RPV Control Modes 1-3 Emergency RPV Depressurization

Op Test No.: 22-04	Scenario No. NRC-1	Event No.: 6	Page 19 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	RC/P-6

IF	THEN
RPV Water Level CANNOT be determined	**NO ACTION REQUIRED**
It is anticipated that available injection subsystems (Table L-3) alone CANNOT assure Adequate Core Cooling	**NO ACTION REQUIRED**
It is anticipated that Containment Water Level will rise above 44 feet	**NO ACTION REQUIRED**

RC/P-7

IF Drywell Pressure is above 2.45 psig

THEN PREVENT injection from ONLY those CS and LPCI Pumps NOT required to assure adequate core cooling (APPX 4)

RC/P-8

EMERGENCY DEPRESSURIZE the RPV

IF Suppression Pool Water Level is above 5.5 ft **THEN OPEN** 6 MSRVs (ADS Valves preferred)

- ➢ OK to exceed 100°F/hr cooldown rate

IF	THEN
Drywell Control Air is or becomes unavailable	**NO ACTION REQUIRED**
Less than 4 MSRVs can be opened AND RPV press is 60 psi or more above Suppression Chamber Pressure	**NO ACTION REQUIRED**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 6 Page 20 of 20

Event Description: Earthquake / Feedwater Leak / LOCA / Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NRC	**End of Event 6. 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 any low pressure systems, end of Scenario.**

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 7 Page 1 of 1

Event Description: Electrical Anticipated Transient Without Scram (ATWS)

Symptoms/Cues: Event is automatically initiated when the crew inserts a manual Reactor SCRAM; Rods will remain out when a Reactor SCRAM is inserted

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 7, Electrical Anticipated Transient Without Scram (ATWS), is automatically entered on simulator setup. No action is required to insert this event.**
	OATC	2-AOI-100-1, Reactor SCRAM [3] **IF** all Control Rods can **NOT** be verified fully inserted, **THEN INITIATE** Alternate Rod Insertion (ARI).
	Driver	**Verify that the ATWS malfunction clears and Control Rods insert when the OATC initiates ARI.**
	OATC	After initiating ARI, verifies that Control Rods insert and informs the NUSO. Continues with Reactor SCRAM actions. [4] **IF** Reactor Power is 5% or BELOW, **THEN**: (otherwise MARK N/A) **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 Performs Subsequent Actions in accordance with 2-AOI-100-1, Reactor SCRAM, Section 4.2.
	NRC	**End of Event 7.**

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 8 Page 1 of 2

Event Description: Reactor Core Isolation Cooling (RCIC) Controller Fails to Operate in Automatic

Symptoms/Cues: Event is automatically initiated when RCIC speed exceeds 500 rpm; the RCIC Flow Controller will indicate an automatic control failure

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 8, Reactor Core Isolation Cooling (RCIC) Controller Fails to Operate in Automatic, is automatically entered on RCIC startup. No action is required to insert Event 8. Verify that malfunction RC04 is active when RCIC Speed exceeds 500 rpm.**
	BOP	2-EOI-Appendix-5C, Injection System Lineup RCIC [1] N/A [2] **ENSURE RESET** auto isolation logic using 2-XS-71-51A(B), RCIC AUTO-ISOL LOGIC A (B) RESET pushbuttons. [3] **ENSURE RESET** and **OPEN** 2-FCV-71-9, RCIC TURBINE TRIP/THROTTLE VALVE. [4] **ENSURE** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with setpoint at 620 gpm. [5] **OPEN** 2-FCV-71-34 RCIC PUMP MINIMUM FLOW VALVE. [6] **OPEN** 2-FCV-71-39 RCIC PUMP INJECTION VALVE. [7] **OPEN** 2-FCV-71-25 RCIC LUBE OIL COOLING WATER VALVE. [8] **PLACE** 2-HS-71-31A, RCIC VACUUM PUMP, in START. **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 ºF may result in equipment damage. [9] **OPEN** 2-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC turbine.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-1 Event No.: 8 Page 2 of 2

Event Description: Reactor Core Isolation Cooling (RCIC) Controller Fails to Operate in Automatic

Time	Position	Applicant's Actions or Behavior
	BOP	[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 MINIMUM FLOW VALVE, closes as flow rises above 120 gpm.
	BOP	Determines that the RCIC Flow Controller has failed in automatic. Takes manual control of the RCIC Flow Controller to inject to the Reactor.
	BOP	[11] **ADJUST** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. [12] **IF** <u>BOTH</u> of the following exist: • RCIC Initiation signal is NOT present, **AND** • RCIC flow is below 60 gpm, **THEN** ENSURE OPEN 2-FCV-71-34, RCIC PUMP MINIMUM FLOW VALVE
	NRC	**End of Event 8.**

UNIT 2 SHIFT TURNOVER MEETING			**Today**
MODE 1	**DAYS ON LINE** **198**	**Total Drywell Leakage (gpm)** **1.55**	**Protected Equipment** 2A EHC Pump RHR Loop II
Rx Power **100%**	PRA (Phoenix) - GREEN 500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (gpm)** **0.11**	Core Spray Loop I, Core Spray Loop II HPCI
MWe **1311**	**Last breaker closure** **10/30/21 04:31**	**Equipment Drain (gpm)** **1.44**	EDG C/D 4KV SD Board C/D, 2A 250V RMOV Board

☐ 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

Day 1 of RHR Loop I (2A RHR Pump) outage, LCO: 3.5.1.A (7 days), 3.6.2.3.A (30 days), 3.6.2.4.A (30 days), 3.6.2.5.A (30 days)

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

2A RHR Pump tagged for RHR Loop I scheduled outage.

2B EHC Pump tagged for breaker maintenance.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Support maintenance on Loop I of RHR.

Support maintenance on 2B EHC Pump Breaker.

Start 2C Condensate Booster Pump (CBP) in accordance with 2-OI-2, Section 5.3. Limiting conditions for Condensate Booster Pump operation in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System are met.

After starting 2C CBP, await instruction from the Reactor Engineer for power ascension.

OPERATOR WORK AROUNDS OWAs – U0-1/U2-0 Burdens - U0 – 0/U2-0 Operator Challenges – U0 – 12/U2 - 6

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Facility: BFN
Scenario Source: NEW
Scenario Number: NRC-2
Op-Test Number: 22-04

Examiners: ____________________________
____
____

Operators: NUSO: ________________________
OATC: ________________________
BOP: ________________________

Initial Conditions: 100 % Reactor Power.

Turnover: 2A RBCCW Pump is tagged for bearing replacement. The EHPM is tagged for breaker inspection.

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 or control Reactor Water Level such that no automatic ADS actuation occurs.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Turbine Bypass Valve Test
2.	TC12B	TS-NUSO	Bypass Valve Inoperable
3.	NMAPRMGAIN(1)	C-OATC C-NUSO	APRM Fails High
4.	RD01A	C-OATC MC-OATC C-NUSO	Control Rod Drive (CRD) Pump Trip
5.	ZLOHS713A_1 ZLOHS713A_2 FCV-71-3	C-BOP MC-BOP TS-NUSO	Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close
6.	TV02A-M	C-NUSO R-OATC	Main Turbine Vibration
7.	RD09A RD09B	M-ALL	Hydraulic Anticipated Transient Without SCRAM (ATWS)
8.	TC02	C-BOP C-NUSO	Bypass Valves Fail Closed

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

Events

1. The crew will test the Main Turbine Bypass Valves in accordance with 2-OI-47, Turbine-Generator System, Section 6.7.

2. The second Bypass Valve that is tested will fail to operate. The Nuclear Unit Senior Operator (NUSO) will declare the Bypass Valve inoperable and will address Technical Specifications.

3. APRM 1 will fail high, requiring the crew to respond in accordance with Alarm Response Procedures and eventually bypass APRM in accordance with 2-OI-92B, Average Power Range Monitoring.

4. 2A CRD pump will trip, requiring the OATC to respond in accordance with 2-AOI-85-3, CRD System Failure.

5. Reactor Core Isolation Cooling (RCIC) will receive an inadvertent isolation signal but not all valves will automatically close, requiring manual action to ensure RCIC is isolated. The Nuclear Senior Unit Operator (NUSO) will address Technical Specifications.

6. The Main Turbine will experience a high vibration condition. The crew will respond in accordance with Alarm Response Procedures, and reduce Reactor Power. Following the Reactor Power Reduction, the vibrations will worsen, requiring the crew to insert a manual Reactor SCRAM.

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, the Turbine Bypass Valves will slowly fail closed, requiring the crew to take action to control Reactor Pressure.

The Scenario ends when the crew has inserted all Control Rods, has control of Reactor Pressure using Safety Relief Valves within a band prescribed by the NUSO, and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using available 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 or control Reactor Water Level such that no automatic ADS actuation occurs.**

   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.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 1 Page 1 of 4

Event Description: Turbine Bypass Valve Test

Symptoms/Cues: Crew is cued by the turnover sheet or by the Simulator Operator as requested by the Chief Examiner.

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, Turbine Bypass Valve Test after assuming the shift, request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to complete the surveillance.** **NOTE: The crew may elect to hold a re-focus brief.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to commence the Turbine Bypass Valve Test in accordance with 2-OI-47, Turbine-Generator System.** **If contacted by the crew as the Reactor Building Assistant Unit Operator (AUO) acknowledge any direction given.**
	NUSO	Directs the Balance of Plant Operator (BOP) to conduct the Turbine Bypass Valve Test in accordance with 2-OI-47, Turbine-Generator System, Section 6.7.

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 1 Page 2 of 4

Event Description: Bypass Valve Inoperable

Time	Position	Applicant's Actions or Behavior
	BOP	2-OI-47, Turbine-Generator System Section 6.7, Turbine Bypass Valve Test **NOTES** 1) This test will be performed as required by 2-SR-3.7.5.1, Main Turbine Bypass Valve Stroke Time Test. 2) This test is performed at BYPASS VALVE TEST switches, located on Panel 2-9-7, unless otherwise specified. 3) The test involves depressing and holding BV TEST pushbuttons. When the pushbutton is depressed, the test logic will slowly ramp the Bypass Valve open. The Bypass Valve will remain open until the pushbutton is released. The open (red) light will illuminate when the valve is approximately greater than 5% open and the closed (green) light will illuminate when the valve is approximately less than 95% open. 4) Allow at least two minutes to elapse between each Bypass Valve being tested to allow system parameters to stabilize. 5) The BPV TEST screen on the EHC WORK STATION can be referred to when performing this test. [1] **CHECK** the following initial conditions are satisfied: • EHC System is in service. Refer to 2-OI-47A. • **CHECK** CONDENSER A, B OR C VACUUM LOW, 2-PA-47-125 (2-XA-55-7B, Window 17) alarm is reset [2] **OBTAIN** Unit NUSO approval to perform this test. [3] **REVIEW** all Precautions and Limitations Section 3.0.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04	Scenario No. NRC-2	Event No.: 1	Page 3 of 4

Event Description: Bypass Valve Inoperable

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **PERFORM** the following to test Bypass Valve 1: [4.1] **DEPRESS** and **HOLD** 2-HS-1-61, BPV-1 TEST pushbutton, **AND CHECK** the following: • Pushbutton backlight illuminates • 2-IL-1-61A red light illuminates when the valve is approximately greater than 10% open • Between approximately 5% to 95% open both 2-IL-1-61A red light and 2-IL-1-61B green light are illuminated • At approximately greater than 95% open, 2-IL-1-61B green light is extinguished [4.2] **AFTER** valve indicates approximately 100% open, **THEN RELEASE** 2-HS-1-61, BPV-1 TEST pushbutton, and **CHECK** the following: • Pushbutton backlight extinguishes. • The 2-IL-1-61B green light illuminates when the valve is approximately less than 95% open • Between approximately 95% to 5% open, both 2-IL-1-61A red light and 2-IL-1-61B green light are illuminated • At approximately less than 5% open, 2-IL-1-61A red light is extinguished • Bypass Valve 1 indicates closed and 2-IL-1-61B green light is illuminated
	BOP	Allows two (2) minutes to pass before proceeding with Bypass Valve #2.
	NRC	**Event 2, Bypass Valve Inoperable, is automatically entered on Simulator Setup during the next step. No action is required by the driver to insert Event 2. See page 5 of 40 for Event 2 actions.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04	Scenario No. NRC-2	Event No.: 1	Page 4 of 4

Event Description: Bypass Valve Inoperable

Time	Position	Applicant's Actions or Behavior
	BOP	[5] **PERFORM** the following to test Bypass Valve 2: [5.1] **WHEN** at least two minutes have elapsed from last bypass valve test, **THEN DEPRESS** and HOLD 2-HS-1-62, BPV-2 TEST pushbutton, and **CHECK** the following: • Pushbutton backlight illuminates • The 2-IL-1-62A red light illuminates when the valve is approximately greater than 5% open
	BOP	Determines that Bypass Valve # 2 failed to operate as expected and informs the Nuclear Unit Senior Operator (NUSO).
	NRC	**End of Event 1. Proceed to Event 2, Bypass Valve Inoperable.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 2 Page 1 of 1

Event Description: Bypass Valve Inoperable

Symptoms/Cues: Event is automatically initiated during Event 1, Turbine Bypass Valve Test

Time	Position	Applicant's Actions or Behavior	
	NRC	**Event 2 is automatically entered on Simulator setup; no driver action is required.**	
	NUSO	Technical Specification 3.7.5 – Main Turbine Bypass System LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE. OR The following limits are made applicable: a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR. Applicability: Thermal Power ≥ 25% RTP Condition: A. Requirements of the LCO not met.	
	NUSO	**REQUIRED ACTION:** A.1 Satisfy the requirements of the LCO.	**COMPLETION TIME:** A.1 – 2 hours
	NRC	**End of Event 2. Request that the driver insert Event 3, APRM Fails High.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 3 Page 1 of 3

Event Description: APRM Fails High

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; APRM 1 will fail high

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to cause APRM 1 to fail high.**
	OATC	Acknowledges and reports the following alarms to the NUSO: • CONTROL ROD WITHDRAWAL BLOCK, 2-9-5A, Window 7 • APRM UPSCALE, 2-9-5A, Window 11 • APRM HIGH/INOP OR OPRM TRIP, 2-9-5A, Window 25
	NUSO	Acknowledges the Operator at the Controls' (OATC) report and directs the OATC to respond in accordance with the appropriate Alarm Response Procedures (ARP).
	BOP	2-9-5A, Alarm Response Procedure CONTROL ROD WITHDRAWL BLOCK, Window 7 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. **REFER TO** Tech Spec Table 3.3.2.1-1, TRM Table 3.3.4-1.
	BOP	2-9-5A, Alarm Response Procedure APRM UPSCALE, Window 11 A. **CHECK** alarm by multiple indications. B. **CHECK** for Rod Block. C. **STOP** any power rise or rod withdrawal in progress. D. **NOTIFY** Reactor Engineer. E. **REFER TO** Tech Spec TRM Table 3.3.4-1.
	Driver	**If contacted as the Reactor Engineer, acknowledge any report or direction given.**

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 3 Page 2 of 3

Event Description: APRM Fails High

Time	Position	Applicant's Actions or Behavior
	BOP	2-9-5A, Alarm Response Procedure APRM HIGH/INOP OR OPRM TRIP, Window 25 A. **CHECK** alarm by multiple indications. B. With NUSO permission, **BYPASS** initiating channel to reset the alarm. **REFER TO** 2-OI-92B, Average Power Range Monitoring. C. N/A D. N/A E. N/A F. N/A G. **NOTIFY** Reactor Engineer H. **CHECK** Thermal Limits on RUNMON Recommends to the NUSO that APRM 1 be bypassed due to failure.
	NUSO	I. **REFER** TO Tech Spec Table 3.3.1.1-1, TRM Table 3.3.4-1. J. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5. Directs the OATC to bypass APRM 1 in accordance with 2-OI-92B, Average Power Range Monitoring.
	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.**
	BOP	2-OI-92B, Average Power Range Monitoring Section 6.1, Bypassing APRM/OPRM Channel **CAUTION** NPG-SPP-10.4 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.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 3 Page 3 of 3

Event Description: APRM Fails High

Time	Position	Applicant's Actions or Behavior
	BOP	[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.
	NRC	**End of Event 3. Request that the driver insert Event 4, CRD Pump Trip.**

Op Test No.: 22-04	Scenario No. NRC-2	Event No.: 4	Page 1 of 2

Event Description: Control Rod Drive (CRD) Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; 2A CRD Pump will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to trip 2A CRD Pump.**
	NRC	**CRD Temperature alarms will be received in approximately 4 and a half minutes after the CRD Pump trip.**
	OATC	Informs the NUSO that 2A CRD Pump has tripped.
	NUSO	Directs 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 AND the 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 MANUAL at the minimum setting.
	Driver	**If contacted as Unit 1, acknowledge any reports given. If the Crew requests to start 1B CRD Pump for use in Unit 2, report that Unit 2 may use 1B CRD Pump.**
	OATC	[1.2] **START** associated standby CRD Pump using 2-HS-85-2A, CRD PUMP 1B. [1.3] **OPEN** 2-HS-85-8A,CRD PUMP 1B DISCH TO U2. [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. [2] N/A

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 4 Page 2 of 2

Event Description: Control Rod Drive (CRD) Pump Trip

Time	Position	Applicant's Actions or Behavior
	OATC	Subsequent Actions: **NOTES** 1) Operation with loss of CRD System should be minimized since CRD Mechanism Cooling Water, Reactor Recirc Pump Seal Purge Flow, 2B Reactor Water Cleanup Recirc Pump Seal Water Flow, and CRD HCU Accumulator Charging Water will be lost. Technical Specification Section 3.1.3, and 3.1.5 should be consulted. 2) Do NOT attempt to start a CRD Pump which has tripped for unknown reasons before verifying locally that the pump, motor and breaker are in operating condition. 3) Steps 4.2[1] and 4.2[2] should be performed concurrently. [1] N/A [2] N/A [3] N/A [4] **IF** CRD System Hydraulic Pressure is restored, **THEN RESTORE** Control Rod Drive System alignment. **REFER TO** 2-OI-85, Control Rod Drive System. (Otherwise N/A) [5] **NOTIFY** Reactor Engineer of CRD System failure.
	Driver	**If contacted as the Reactor Engineer, Electrical Maintenance, System Engineer, AUO, or Work Control NUSO acknowledge any report or direction given.**
	NRC	**End of Event 4. Request that the driver insert Event 5, Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 5 Page 1 of 5

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. RCIC will receive an isolation signal, but 2-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE will not automatically close

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause a RCIC System Isolation.**
	NRC	**2-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE, will not automatically close when the isolation signal is received. The valve will close when an operator gives it a close signal manually.**
	BOP	Acknowledges and reports the following alarms to the NUSO: • RCIC TURBINE TRIPPED, 2-9-3B, Window 14 • RCIC TURBINE EXHAUST DISC RUPTURED, 2-9-3B, Window 35
	NUSO	Directs the BOP to respond in accordance with the appropriate ARPs.
	BOP	Alarm Response Procedure, 2-ARP-9-3B RCIC TURBINE TRIPPED, Window 14 Operator Action: A. **DETERMINE** cause. B. **IF** valid alarm, **THEN REFER TO** 2-OI-71, Reactor Core Isolation Cooling System, Section 8.0. C. **REFER TO** Tech Spec 3.5.3.
	BOP	Alarm Response Procedure, 2-ARP-9-3B RCIC TURBINE EXHAUST DISC RUPTURED, Window 35 Operator Action: A. **REFER TO** 2-AOI-64-2C, Group 5 Reactor Core Isolation Cooling Isolation.

Op Test No.: 22-04	Scenario No. NRC-2	Event No.: 5	Page 2 of 5

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior
	BOP	2-AOI-64-2C, Group 5 Reactor Core Isolation Cooling Isolation 3.0 AUTOMATIC ACTIONS A. RCIC Turbine Trips B. The following valves CLOSE: (Indications for the valves are on Panel 2-9-3). 1. 2-FCV-71-2, RCIC STEAM LINE INBOARD ISOLATION VALVE. 2. 2-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE. 3. 2-FCV-71-34, RCIC PUMP MINIMUM FLOW VALVE. 4. 2-FCV-71-9, RCIC TURBINE TRIP/THROTTLE VALVE. 4.1 Immediate Actions [1] **ENSURE** automatic actions occur.
	BOP	Determines that 2-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE did not automatically close and manually closes 2-FCV-71-3. Informs the NUSO concerning the status of 2-FCV-71-3.
	BOP	4.2 Subsequent Actions [1] **IF** ANY EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). [2] **DISPATCH** an operator to the RCIC Turbine room to investigate.
	Driver	**If contacted as an AUO, Mechanical Maintenance, or the Work Control NUSO acknowledge any report or direction given.**
	BOP	[3] **CHECK** the following monitors for a rise in activity or area temperature: [3.1] 2-RR-90-1, AREA RADIATION, Bank 2 Point 6 & 9 (2-RI-90-26 & 29), Panel 2-9-2. [3.2] 2-MON-90-50, AIR PARTICULATE MONITOR RADIATION (2-RM-90-57), Panel 2-9-2. [3.3] 2-RM-90-250, Reactor and Turbine Building Exhaust Radiation on CHEMISTRY CAM MONITOR CONTROLLER, 0-MON-90-361A, Panel 1-9-2.
	Driver	**If contacted as Unit 1, there is no apparent rise in 2-MON-90-250 activity.**

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 5 Page 3 of 5

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior
	BOP	[3.4] 2-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE (Panel 2-9-22) [4] **ENSURE OPEN** 2-FCV-1-55 and -56, MAIN STEAM LINE INBOARD and OUTBOARD DRAIN ISOLATION VALVE, to drain RCIC steam lines. [5] **IF** the isolation occurred due to Exhaust Diaphragm High Pressure, **THEN DISPATCH** an AUO to check 2-SHV-071-0014, RCIC TURBINE EXHAUST SHUTOFF VALVE, DRAIN LINE SHUTOFF VALVE LOCKED OPEN. [6] N/A [7] N/A [8] N/A [9] **REFER TO** 2-OI-71, Reactor Core Isolation Cooling System and **PLACE** RCIC in the condition directed by the NUSO.
	Driver	**If contacted as the Work Control/Outside NUSO or AUO to investigate the cause, prepare tags, or address Protected Equipment Requirements acknowledge the direction.** **If contacted as an AUO to check ATUs in the Aux Instrument Room, acknowledge the direction.**
	NUSO	Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs) LCO – Each PCIV, except Reactor Building to Suppression Chamber Vacuum Breakers, shall be OPERABLE 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.

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 5 Page 4 of 5

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior	
	NUSO	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.	
	NUSO	**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. AND A.2 ------------------NOTE------------------- Isolation devices in high radiation areas may be verified by use of administrative means. --- Verify the affected penetration flow path is isolated.	**COMPLETION TIME** A.1 – 4 hours except for Main Steam Line AND A.2 – Once per 31 days for isolation devices outside Primary Containment AND 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

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 5 Page 5 of 5

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.5.3, RCIC System LCO 3.5.3 The RCIC System shall be OPERABLE ------------------------------- NOTE -------------------------------------- LCO 3.0.4.b is not applicable to RCIC -- Condition: A. RCIC System inoperable.	
	NUSO	**REQUIRED ACTION** A.1 Verify by administrative means High Pressure Coolant Injection is OPERABLE. <u>AND</u> A.2 Restore RCIC System to OPERABLE status.	**COMPLETION TIME** A.1 – Immediately A.2 – 14 days
	NRC	**End of Event 5. Request that the driver insert Event 6, Main Turbine Vibration.**	

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 6 Page 1 of 5

Event Description: Main Turbine Vibration

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. Main Turbine vibration level will rise above the alarm setpoint, but not above the "immediate trip" action level until after the power reduction

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to cause high vibration on the Main Turbine.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • MAIN TURBINE VIBRATION HIGH, 2-9-7B, Window 32
	NUSO	Directs the BOP to respond in accordance with the appropriate ARP.
	BOP	2-9-7B, Alarm Response Procedure MAIN TURBINE VIBRATION HIGH, Window 32 Operator Action: A. **VALIDATE** alarm by checking the following: • On EHC WORKSTATION, Turbine Vibration screen • On ICS, MAIN TURBINE BEARINGS (TURBBRG) screen • 2-MON-47-94, TURBINE GENERATOR VIBRATION (Panel 2-9-7) • Computer points 47-1501A&B thru 47-1512A&B B. **IF** alarm is valid, **THEN PERFORM** the following: 1. **DETERMINE** cause by checking PROBABLE CAUSE section above. 2. **REDUCE** load and **OBSERVE** vibration.
	NUSO	Directs the OATC to reduce Main Turbine Load by reducing Reactor Power.
	NRC	**The crew may elect to reduce Reactor Power via either a manual Runback, Master Recirc Speed Control, or individual Recirc Pump Speed Controls and may stop the method selected at any point deemed necessary.**

Op Test No.: 22-04	Scenario No. NRC-2	Event No.: 6	Page 2 of 5

Event Description: Main Turbine Vibration

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%.
	OATC	**(If the Crew elects to use an Upper Power Runback)** [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%

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 6 Page 3 of 5

Event Description: Main Turbine Vibration

Time	Position	Applicant's Actions or Behavior
	OATC	**(If the Crew elects to use a Mid-Power Runback)** [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%
	OATC	**(If the Crew elects to use a Core Flow Runback)** [4] **IF** desired to reduce Core Flow to approximately 60%, **THEN** (Otherwise N/A): [4.1] **DEPRESS** RECIRC PUMPS CORE FLOW RUNBACK Pushbutton, 2-HS-68-44. [4.2] **CHECK** the following: • Pushbutton backlight blinks until setpoint is reached • Core Flow lowers to approximately 60%
	OATC	**(If the Crew elects to use individual or Master Speed Control)** 2-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow NOTES 1) Thermal Limits are shown in 0-TI-248, Unit 2 Core Operating Limits Report (COLR) and 2-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the later part of the operating cycle (Coast down) 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.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 6 Page 4 of 5

Event Description: Main Turbine Vibration

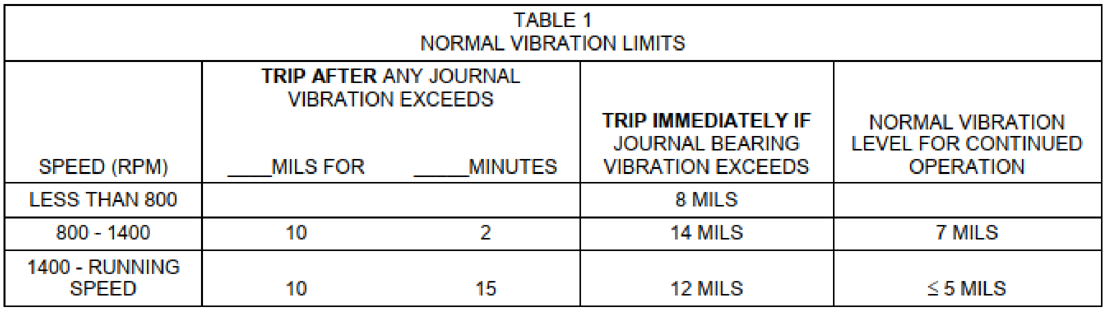
Time	Position	Applicant's Actions or Behavior
	OATC	[1] **IF** desired to control Recirc Pumps 2A and/or 2B speed with Recirc Individual Control, **THEN PERFORM** the following: • **LOWER** Recirc Pump 2A using 2-HS-96-17A(17B)(17C), SLOW(MEDIUM)(FAST), (Otherwise N/A) AND/OR • **LOWER** Recirc Pump 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	**Based on the vibrations not lowering following the power reduction, the Crew may elect to insert a manual Reactor SCRAM.** **If the Crew does not SCRAM after the power reduction, request that the driver insert Event 16 to cause Main Turbine Vibrations to rise above the TURBINE TRIP action level in 2-ARP-9-7B, Window 32 (MAIN TURBINE VIBRATION HIGH).**
	BOP	(Continuing actions of Alarm Response Procedure 2-9-7B, Window 32, MAIN TURBINE VIBRATION HIGH) 3. **IF** any of the vibration limits requiring a trip are met in Table 1, **THEN DEPRESS** Turbine TRIP pushbutton, 2-HS-47-67D: 

TABLE 1 NORMAL VIBRATION LIMITS				
	TRIP AFTER ANY JOURNAL VIBRATION EXCEEDS			
SPEED (RPM)	____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

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 6 Page 5 of 5

Event Description: Main Turbine Vibration

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs the OATC to insert a manual Reactor SCRAM and enter 2-AOI-100-1, Reactor SCRAM.
	NRC	**End of Event 6. Event 7, Hydraulic Anticipated Transient Without SCRAM (ATWS) is automatically entered on Simulator setup. No action is required by the driver to insert Event 7.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 1 of 12

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Symptoms/Cues: Event is automatically initiated during simulator setup. Control Rods will not fully insert into the Core when the OATC inserts a manual SCRAM for Event 6

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.** **Two (2) minutes after the Reactor MODE SWITCH is placed in shutdown, all Turbine Bypass Valves will be closed due to a Control Unit Failure. See Event 8, Bypass Valves Fail Closed on page 33 of 40. No action is required by the driver to insert Event 8.**
	Driver	**During Event 7, when contacted as the Outside (or Work Control) NUSO acknowledge direction to perform EOI Appendices and enter events as necessary:** • **Event 20 – 2-EOI-Appendix-1F, Manual SCRAM** • **Event 21 – 2-EOI-Appendix-2, Defeating ARI Logic Trips** • **Event 22 – 2-EOI-Appendix-8A, Bypassing Group RPV Low Low Low Level Isolation Interlocks** • **Event 23 – 2-EOI-Appendix-8E, Bypassing Group 6 RPV Low Level and High Drywell Pressure Isolation Interlocks** • **Event 24 – 2-EOI-Appendix-1D, Insert Control Rods Using Reactor Manual Control System (Close 2-FCV-85-586, CHARGING WATER ISOLATION)** • **Event 25 – Open 2-FCV-85-586, CHARGING WATER ISOLATION** **Once the event(s) requested have finished their time delay, report completion of the various EOI Appendices to the Control Room.**
	OATC	2-AOI-100-1, Reactor SCRAM [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. [4] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 2 of 12

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	[5] **IF** Reactor Power is ABOVE 5% or unknown, **THEN PERFORM** the following: [5.1] **REPORT** the following to the Unit NUSO: • REACTOR SCRAM, Mode Switch in SHUTDOWN • Control Rods out • Reactor Power • Continuing with ATWS Actions [5.2] **ENSURE** shutdown both Recirc Pumps.
	NRC	**NOTE: The start time for the Critical Task below (STOP and PREVENT) begins when Recirc Pump Speeds are equal to zero.**
	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.**
	OATC	[5.3] **STOP** and **PREVENT** injection from CONDENSATE and FEEDWATER per EOI APPENDIX 4. • Maintain Reactor Water Level (-) 180 inches to (-) 50 inches [5.4] **STOP** and **PREVENT** injection from HPCI, RHR, and Core Spray per EOI APPENDIX 4. [5.5] **INITIATE** SLC and **ENSURE** injection.

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 3 of 12

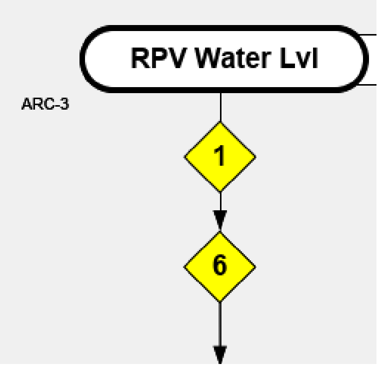
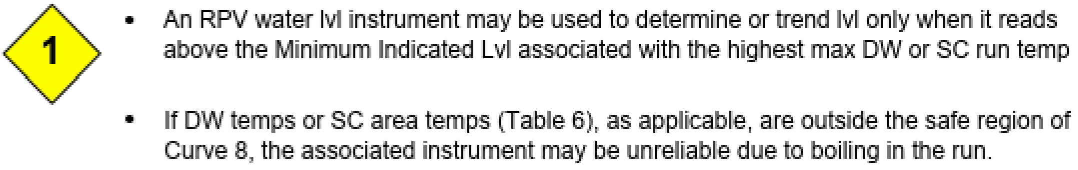
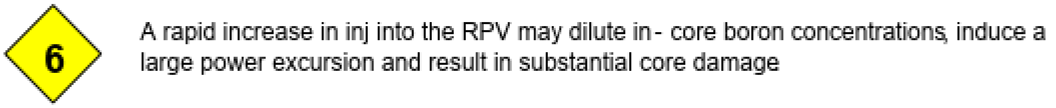
Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	CREW	**Critical Task:** **With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion inhibit ADS or control Reactor Water Level such that no automatic ADS actuation occurs.** **Critical Task Failure Criteria:** **ADS automatic initiation with Control Rods out.**
	BOP	[5.6] **INHIBIT** ADS LOGIC. [5.7] **BYPASS** Group 1 RPV Low-Low-Low Level isolation interlocks per EOI APPENDIX 8A. (SEE PAGE 29 of 40 FOR ACTIONS)
	OATC	[5.8] **WHEN** Reactor Water Level reaches (-) 50 inches, **THEN REPORT** "ATWS Actions Complete" • Reactor Water Level and trend • Reactor Power Level • Reactor Pressure and trend • MSIV position • SLC IS/IS NOT injecting
	OATC	[5.9] N/A [5.10] **IF** Any Main steam line is OPEN, **THEN CALL** for 2-EOI-APPENDIX-8E, Bypassing Group 6 Low RPV Level and High Drywell Pressure Isolation interlocks.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 4 of 12

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	Enters 2-EOI-1A, ATWS RPV Control.   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.  6 A rapid increase in inj into the RPV may dilute in- core boron concentrations, induce a large power excursion and result in substantial core damage ARC/L-1 **INHIBIT** ADS ARC/L-2 **ENSURE** each as required: • PCIS isolations (Groups 1, 2, and 3) • ECCS • RCIC

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 5 of 12

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-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>PC Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table>
ARC/L-4
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Reactor Power is above 5% or unknown
AND
RPV Water Level is above (-) 50 in.</td><td>A</td></tr>
<tr><td>ALL Level/Power conditions exist (Table M-6)</td><td>B</td></tr>
<tr><td>Reactor Power is above 10% or unknown
AND
RPV Water Level is below -162 in.
AND
MSRV open/cycling OR Drywell Pressure above 2.4 psig</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>

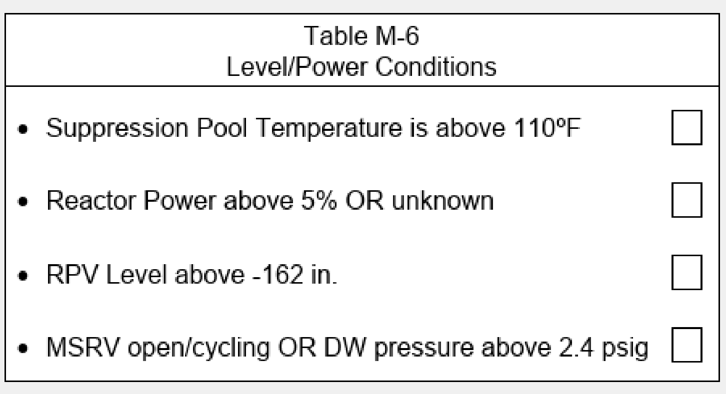
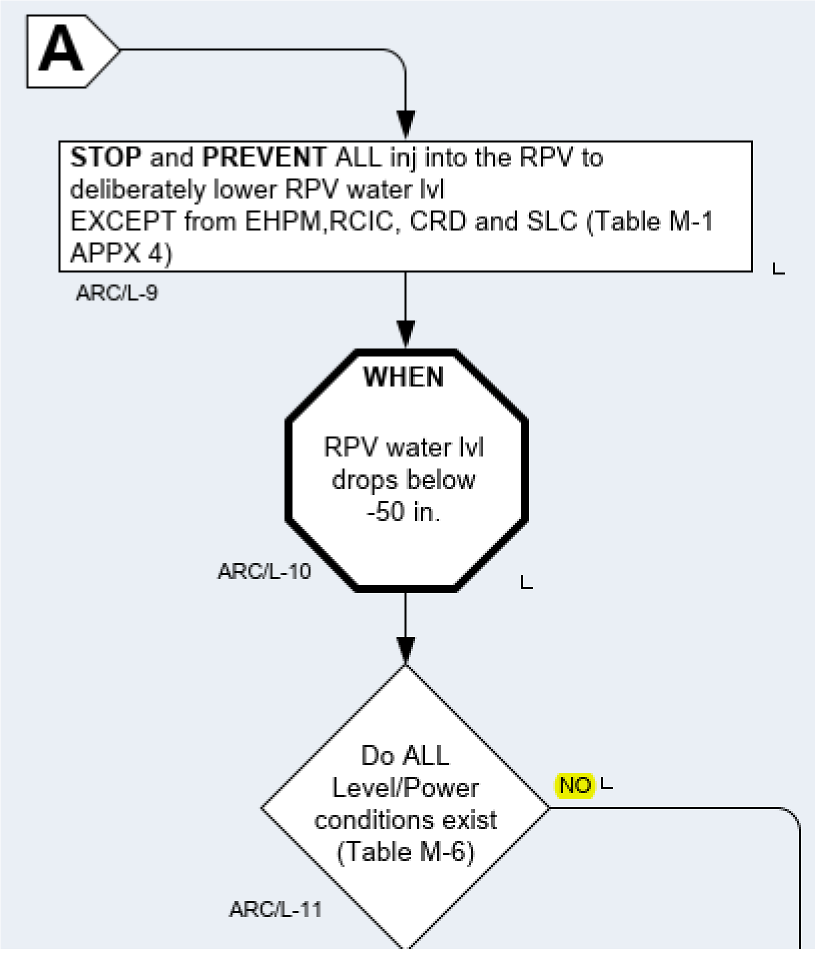



Table M-6
Level/Power Conditions
<table>
<tr><td>• Suppression Pool Temperature is above 110°F</td><td>☐</td></tr>
<tr><td>• Reactor Power above 5% OR unknown</td><td>☐</td></tr>
<tr><td>• RPV Level above -162 in.</td><td>☐</td></tr>
<tr><td>• MSRV open/cycling OR DW pressure above 2.4 psig</td><td>☐</td></tr>
</table>
</td></tr>
</table>

Form 3.3-2 Required Operator Actions

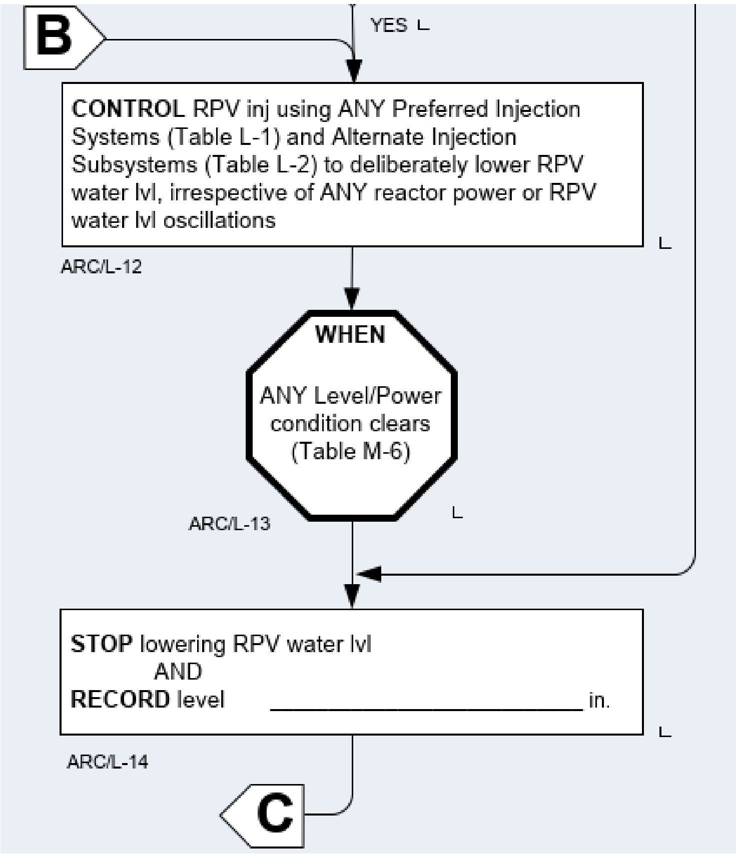
Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 6 of 12

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 7 of 12

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	**Position**	**Applicant's Actions or Behavior**
	NUSO	
	NUSO	Directs the OATC to maintain Reactor Water Level in any band between (-) 50 and (-) 180 inches to keep Reactor Power low.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 8 of 12

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

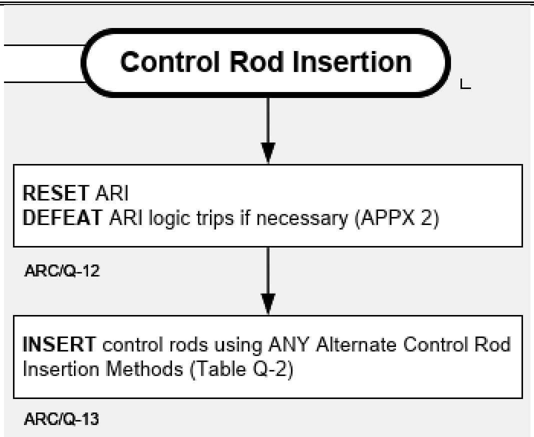
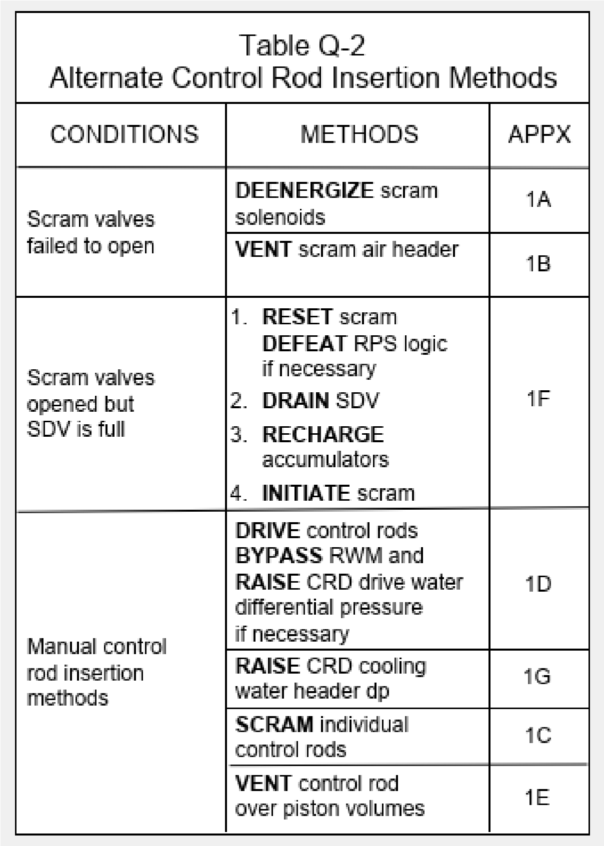
Time	Position	Applicant's Actions or Behavior
	NUSO	 **Control Rod Insertion** **RESET** ARI **DEFEAT** ARI logic trips if necessary (APPX 2) ARC/Q-12 **INSERT** control rods using ANY Alternate Control Rod Insertion Methods (Table Q-2) ARC/Q-13 

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

Op Test No.: 22-04	Scenario No. NRC-2	Event No.: 7	Page 9 of 12

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	BOP	2-EOI-APPENDIX-8A, Bypassing Group 1 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] N/A END OF EOI APPENDIX 8A
	OATC	Drives Control Rods using 2-EOI-Appendix-1D, Insert Control Rods Using Reactor Manual Control System [1] **VERIFY** at least one CRD Pump in service. **NOTES** 1) Closing 2-85-586, CHARGING WATER ISOLATION valve may reduce the effectiveness of EOI Appendix 1A or 1B. 2) A ladder may be required to perform the following step. REFER TO Tools and Equipment, Attachment 1. 3) IF necessary, an alternate ladder is available at the HCU Modules, EAST and West banks. It is stored by the CRD Charging Cart.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 10 of 12

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>OATC</td><td>

[2] **IF** Reactor SCRAM or ARI CANNOT be reset, **THEN DISPATCH** personnel to close 2-SHV-85-586, CHARGING WATER SHUTOFF (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:

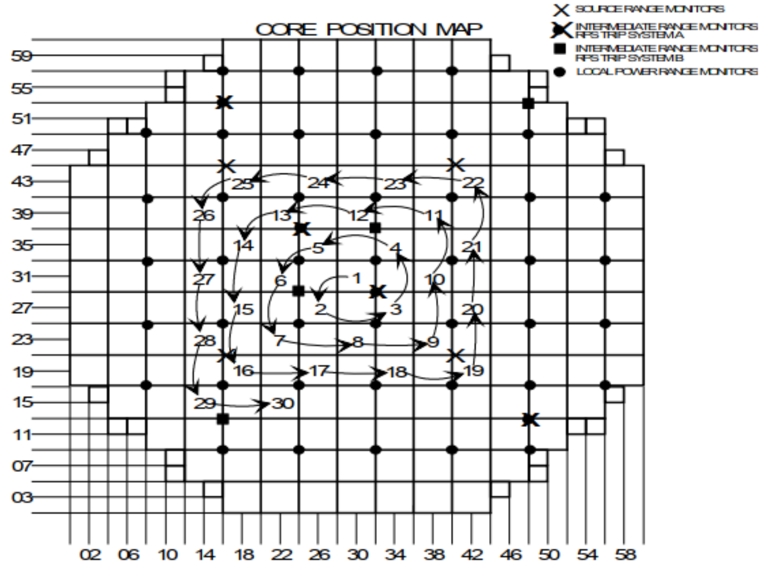
- [5.1] **SELECT** Control Rod.
- [5.2] **PLACE** CRD NOTCH OVERRIDE switch in EMERGENCY ROD IN position UNTIL Control Rod is NOT moving inward.
- [5.3] **REPEAT** Steps 1.0[5.1] and 1.0[5.2] for each Control Rod to be inserted.

[6] **WHEN** NO further Control Rod movement is possible or desired, **THEN DISPATCH** personnel to verify open 2-SHV-85-586, CHARGING WATER SHUTOFF (RB NE, El 565 ft).

</td></tr>
<tr><td></td><td>OATC</td><td>

Attachment 2
(Page 1 of 1)

Core Position Map



(EXAMPLE ONLY)

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Op Test No.: 22-04	Scenario No. NRC-2	Event No.: 7	Page 11 of 12

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	2-EOI-APPENDIX-1F, Manual SCRAM [1] **VERIFY** Reactor Scram and ARI reset. [1.1] **IF** ARI CANNOT be reset, **THEN EXECUTE** EOI Appendix 2, Defeating ARI Logic Trips concurrently with Step 1.0[1.2]of this procedure. [1.2] N/A for the Control Room, this step is performed locally in the Auxiliary Instrument Room [2] **WHEN** RPS Logic has been defeated, **THEN RESET** Reactor SCRAM. [3] **VERIFY** OPEN Scram Discharge Volume Vent and Drain Valves. [4] **DRAIN** SDV UNTIL the following annunciators clear: • WEST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 2-9-4, 2-XA-55-4A, Window 1) • EAST CRD DISCH VOL WTR LVL 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.**

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 7 Page 12 of 12

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	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
	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.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 8 Page 1 of 7

Event Description: Bypass Valves Fail Closed

Symptoms/Cues: Event is automatically initiated when the crew places the Reactor MODE SWITCH in SHUTDOWN. Turbine Bypass Valves will all be closed after 60 seconds

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8 is automatically entered upon Simulator setup. No action is required by the driver to insert Event 8.**
	CREW	Diagnoses a Reactor Pressure rise when Bypass Valves are all closed. Determines that the Bypass valves have closed.
	NUSO	Directs the BOP to maintain Reactor Pressure using 2-EOI-APPENDIX-11A, Alternate RPV Pressure Control Systems MSRVs.
	BOP	2-EOI-APPENDIX-11A, Alternate RPV Pressure Control Systems MSRVs [1] N/A [2] N/A

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 8 Page 2 of 7

Event Description: Bypass Valves Fail Closed

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **OPEN** MSRVs using the following sequence to control RPV Pressure as directed by the NUSO:

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

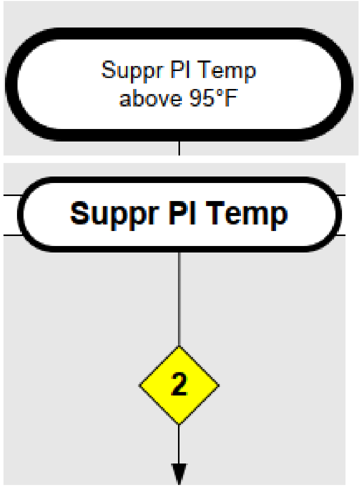
END OF EOI APPENDIX 11A

Time	Position	Applicant's Actions or Behavior
	BOP	Acknowledges and reports the following alarm when received: • SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, 2-9-3E, Window 12

Form 3.3-2 Required Operator Actions

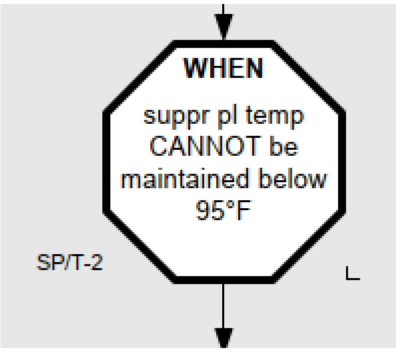
Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 8 Page 3 of 7

Event Description: Bypass Valves Fail Closed

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	(If received) 2-ARP-9-3E,Alarm Response Procedure SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, Window 12 Operator Action: A. **IF** alarm is valid, **THEN ENTER** 2-EOI-2, Primary Containment Control.
	NUSO	Enters 2-EOI-2, Primary Containment Control.
	NUSO	 2 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 SP/T-1 **MONITOR** and **CONTROL** Suppression Pool Temperature below 95°F using available Suppression Pool Cooling (APPX 17A).

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 8 Page 4 of 7

Event Description: Bypass Valves Fail Closed

Time	Position	Applicant's Actions or Behavior
	NUSO	 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. [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

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 8 Page 5 of 7

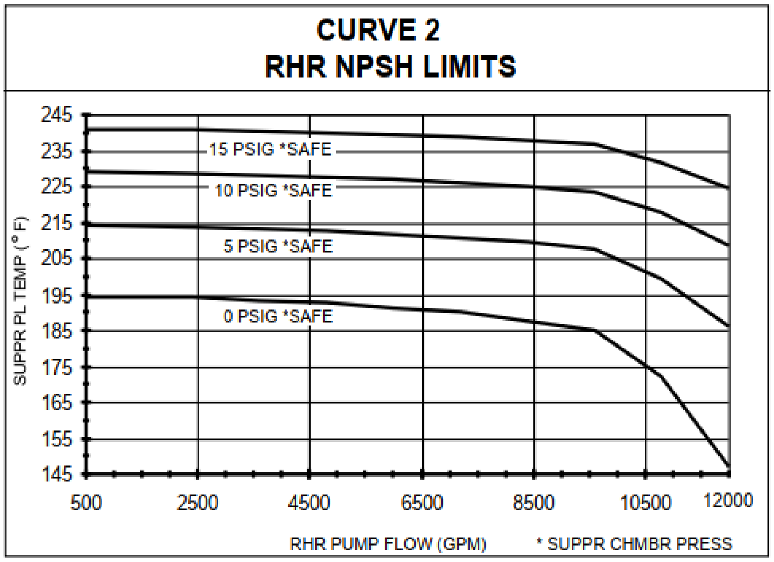
Event Description: Bypass Valves Fail Closed

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: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE [2.4] **IF** Directed by NUSO, **THEN PLACE** 2-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** 2-XS-74-121(129), RHR SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, in SELECT. [2.6] **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 INJECT VALVE. [2.7] **OPEN** 2-FCV-74-57(71), RHR SYSTEM 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.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-2 Event No.: 8 Page 6 of 7

Event Description: Bypass Valves Fail Closed

Time	Position	Applicant's Actions or Behavior
	BOP	[2.9] **THROTTLE** OPEN 2-FCV-74-59(73), RHR SYSTEM I(II) SUPPRESSION POOL COOLING/TEST VALVE, to maintain EITHER of the following as indicated on 2-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 2-FCV-74-7(30), RHR SYSTEM I(II) MINIMUM FLOW VALVE. [2.11] **MONITOR** RHR Pump NPSH using Attachment 1.
	BOP	**Attachment 1** **(Page 1 of 1)** **NPSH Monitoring** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used.  **CURVE 2** **RHR NPSH LIMITS**

Op Test No.: 22-04	Scenario No. NRC-2	Event No.: 8	Page 7 of 7

Event Description: Bypass Valves Fail Closed

Time	Position	Applicant's Actions or Behavior
	BOP	[2.12] **NOTIFY** Chemistry that RHRSW is aligned to in service RHR Heat Exchangers.
	Driver	**When contacted as Chemistry, acknowledge any reports or direction given.**
	NRC	**End of Event 8. Once the crew has inserted all Control Rods, has control of Reactor Pressure using Safety Relief Valves with a band prescribed by the NUSO, and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

UNIT 2 SHIFT TURNOVER MEETING			**Today**
MODE 1	**DAYS ON LINE** **389**	**Total Drywell Leakage (gpm)** **1.55**	**Protected Equipment** None
Rx Power **100%**	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (gpm)** **0.11**	
MWe **1280**	**Last breaker closure** **4/23/21 05:41**	**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

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

EHPM tagged for breaker inspection.

2A RBCCW Pump tagged for bearing replacement. The spare RBCCW Pump is aligned to Unit 3.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Support maintenance on the EHPM and 2A RBCCW Pump.

OPERATOR WORK AROUNDS OWAs – U0-1/U2-0 Burdens - U0 – 0/U2-0 Operator Challenges – U0 – 12/U2 - 6

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Facility: BFN
Scenario Source: NEW
Scenario Number: NRC-3
Op-Test Number: 22-04

Examiners: ____________________________
____
____

Operators: NUSO: ____________________________
OATC: ____________________________
BOP: ____________________________

Initial Conditions: 100 % Reactor Power.

Turnover: 2B Bus Duct Fan is tagged for maintenance. 1B CRD Pump is tagged to add oil. Instrument Mechanics are preparing for maintenance in the Aux Instrument Room. Perform Turning Gear Auto Start Test in accordance with 2-OI-47B, Main Turbine Lube Oil System, Sect. 6.3.

Critical Tasks:

1. Start the Standby Stator Cooling Water Pump before the Turbine Trip Timer times out and trips the Main Turbine, resulting in an automatic Reactor SCRAM.
2. When Drywell Sprays are required, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit Curve to prevent challenging Primary Containment negative pressure capability.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Turning Gear Oil Pump Auto Start Test
2.	RD22	C-OATC MC-OATC C-NUSO	Control Rod Drive (CRD) Flow Transmitter Fails High
3.	N/A	TS-NUSO	2A SLC Pump Low Oil Level
4.	AD01M	C-BOP MC-BOP C-NUSO	Steam Relief Valve (SRV) Fails Open
5.	N/A	R-OATC R-NUSO	Reactor Power Reduction
6.	RD04R3011	C-OATC TS-NUSO	Control Rod Drift Out
7.	HS-35-35A PMP-35-36	C-BOP MC-BOP C-NUSO	Stator Cooling Water Pump Trip
8.	AD01M PC05M	M-ALL	SRV Fails Open / Tail Pipe Leak
9.	FCV-74-74 FCV-74-60	C-BOP C-NUSO	Drywell Spray Failure
10.	FW15C	C-OATC MC-OATC C-NUSO	2C Reactor Feedwater Pump (RFPT) Trip

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

Events

1. The crew will perform the Turning Gear Auto Start Test in accordance with 2-OI-47B, Main Turbine Lube Oil System, Section 6.3.

2. The CRD Flow Transmitter 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. The Reactor Building Assistant Unit Operator (AUO) will report that there is no visible oil level for 2A Standby Liquid Control (SLC) Pump. The Nuclear Senior Unit Operator (NUSO) will address Technical Specifications.

4. Steam Relief Valve (SRV) 2-FCV-1-179, MAIN STEAM LINE A RELIEF VALVE will open inadvertently, requiring the crew to respond in accordance with 2-AOI-1-1, Relief Valve Stuck Open.

5. As a result of the failed open SRV, the crew will reduce Reactor Power in accordance with 2-AOI-1-1, Relief Valve Stuck Open.

6. 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 Specifications.

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-XA-35-100, 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. SRV 2-FCV-1-179, MAIN STEAM LINE A RELIEF VALVE will fail again open and will not be able to be closed. When the SRV fails open, a tail pipe leak will occur in the Drywell, causing Drywell Pressure and Temperature to rise. The crew will respond in accordance with the Emergency Operating Instructions (EOIs).

9. When the Balance of Plant Operator (BOP) attempts to spray the Drywell in accordance with 2-EOI-APPENDIX-17B, RHR System Operation Drywell Sprays, the first loop used will fail, requiring the crew to switch loops of Drywell Spray.

10. Following the Reactor SCRAM and during Reactor Feedwater Pump (RFPT) Automatic SCRAM Response, 2C RFPT will trip, requiring action by the crew to maintain Reactor Water Level using another RFPT or other high-pressure injection system.

The Scenario ends when the crew is spraying the Drywell and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high-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 an automatic 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. **When Drywell Sprays are required, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit Curve to prevent challenging Primary Containment negative pressure capability.**

   a. **Safety Significance:**
   Precludes failure of Primary Containment

   b. **Cues:**
   Procedural compliance.
   High Drywell and Suppression Chamber Pressure.
   High Drywell Temperature.

   c. **Measured by:**
   Observation – the crew verifies Primary Containment parameters prior to initiating Drywell Sprays.

   d. **Feedback:**
   Drywell Pressure and Temperature lowering.
   Suppression Chamber Pressure lowering.
   RHR Flow to Containment Cooling and Sprays.

   e. **Critical Task Failure Criteria:**
   Drywell Sprays are initiated with Drywell Pressure and Temperature outside the safe are of the Drywell Spray Initiation Curve (Curve 5).

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 1 Page 1 of 3

Event Description: Turning Gear Oil Pump Auto Start Test

Symptoms/Cues: Crew is cued by the turnover sheet or by the Simulator Operator as requested by the Chief Examiner

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, Turning Gear Oil Pump Auto Start Test after assuming the shift, request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to perform the test.** **NOTE: The crew may elect to hold a re-focus brief prior to commencing the test.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to perform the Turning Gear Oil Pump Auto Start Test.** **If contacted by the crew as the Turbine Building Assistant Unit Operator (AUO) acknowledge any direction given and report that you are standing by in the Unit 2 Turbine Building.**
	NUSO	Directs BOP to perform 2-OI-47B, Main Turbine Lube Oil System Section 6.3, Turning Gear Oil Pump Start Test
	BOP	2-OI-47B, Main Turbine Lube Oil System Section 6.3, Turning Gear Oil Pump Start Test **NOTES** 1) This test should be performed monthly during normal operation of the Turbine. 2) Steps are performed at Panel 2-9-7 in the Control Room and locally at the Main Turbine Oil Tank (MTOT), elevation 586', Column T9-K Line. **CAUTION** If the Turning Gear Oil Pump (TGOP) fails the auto start test, the turbine may continue to operate at load. However, the problem with the Turning Gear Oil Pump is required to be corrected immediately. [1] **CHECK** that the Turning Gear Oil Pump (TGOP) is NOT running.

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 1 Page 2 of 3

Event Description: Turning Gear Oil Pump Auto Start Test

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Turbine Building AUO to verify that the TGOP is not running, acknowledge the direction and report that it is not running.**
	BOP	[2] **CHECK** 2-HS-47-11A, TURNING GEAR OIL PUMP switch is in AUTO. [3] **PLACE** and **HOLD** 2-HS-47-21A, TGOP AUTO START TEST A and B handswitch, in the TEST A position. [4] **INSTRUCT** Operator stationed at the Main Turbine Oil Tank (MTOT) to complete the following: [4.1] **CHECK** by shaft rotation that the TGOP starts. [4.2] **CHECK** Discharge Pressure, as indicated by 2-PI-47-48, TURBINE TURNING GEAR OIL PUMP DISCHARGE PRESSURE, is above 40 psig.
	Driver	**When contacted as the Turbine Building AUO to perform steps [4.1] and [4.2], report that the TGOP has started and Discharge Pressure is above 40 psig.**
	BOP	[5] **RELEASE** 2-HS-47-21A, TGOP AUTO START TEST A and B handswitch. [6] **STOP** 2-HS-47-11A, TURNING GEAR OIL PUMP. [7] **PLACE** and **HOLD** 2-HS-47-21A, TGOP AUTO START TEST A and B handswitch in the Test B position. [8] **INSTRUCT** Operator stationed at the MTOT to CHECK Turning Gear Oil Pump starts.
	Driver	**When contacted as the Turbine Building AUO to perform Step [8], report that the TGOP has started.**
	BOP	[9] **CHECK** MAIN TURBINE BEARING OIL HEADER PRESSURE LOW 2-PA-47-106 (2-XA-55-7A, Window 27) is ILLUMINATED. [10] **RELEASE** 2-HS-47-21A, TGOP AUTO START TEST A AND B handswitch. [11] N/A [12] **STOP** 2-HS-47-11A, TURNING GEAR OIL PUMP. [13] **ENSURE** 2-PA-47-106, MAIN TURB BEARING OIL HEADER PRESSURE LOW (2-XA-55-7A, Window 27) is RESET. [14] **ENSURE** 2-HS-47-11A, TURNING GEAR OIL PUMP handswitch is in AUTO.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 1 Page 3 of 3

Event Description: Turning Gear Oil Pump Auto Start Test

Time	Position	Applicant's Actions or Behavior
	NRC	**End of Event 1. Request that the driver insert Event 2, Control Rod Drive (CRD) Flow Transmitter Fails High.**

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 2 Page 1 of 2

Event Description: Control Rod Drive (CRD) Flow Transmitter Fails High

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. The CRD Flow Controller will fail high and the CRD Flow Control Valves will be closed.

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, insert Event 2, Control Rod Drive (CRD) Flow Transmitter Fails High.**
	OATC	Acknowledges and reports the following alarm to the 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	2-ARP-9-5A, Alarm Response Procedure ACCUMULATOR CHARGING WATER HEADER PRESSURE 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 Transmitter 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 Transmitter 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.: 22-04	Scenario No. NRC-3	Event No.: 2	Page 2 of 2

Event Description: Control Rod Drive (CRD) Flow Transmitter Fails High

Time	**Position**	**Applicant's Actions or Behavior**
	OATC	2-OI-85, Control Rod Drive System Section 8.34, 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. [3] N/A
	NRC	**End of Event 2. Request that the Driver insert Event 3, 2A SLC Pump Low Oil Level.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 3 Page 1 of 1

Event Description: 2A SLC Pump Low Oil Level

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner

Time	Position	Applicant's Actions or Behavior	
	Driver	**When requested by the Chief Examiner, call the NUSO and report as the Reactor Building AUO that 2A SLC Pump oil sight glass is cracked, there is oil on the floor, and there is no oil in the sight glass. If directed to open the circuit breaker for 2A SLC Pump, wait two (2) minutes and insert Event 3. If contacted as the Work Control/Outside NUSO, acknowledge direction to protect 2B SLC Pump and tag 2A SLC pump.**	
	NUSO	Technical Specification 3.1.7 – Standby Liquid Control System LCO 3.1.7 Two SLC subsystems shall be OPERABLE. Applicability: Modes 1, 2, and 3. Condition: A. One SLC subsystem inoperable.	
	NUSO	**REQUIRED ACTION:** A.1 Restore SLC subsystem to OPERABLE status.	**COMPLETION TIME:** A.1 – 7 Days
	NRC	**End of Event 3. Request that the driver insert Event 4, Steam Relief Valve (SRV) Fails Open.**	

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 4 Page 1 of 3

Event Description: Steam Relief Valve (SRV) Fails Open

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. SRV 2-FCV-1-179, MAIN STEAM LINE A RELIEF VALVE, will inadvertently open

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to cause SRV 1-179 to fail open.**
	NRC	**SRV 1-179 is NOT an Automatic Depressurization System (ADS) Valve.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • MAIN STEAM RELIEF VALVE OPEN, 2-9-3C, Window 25
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the Alarm Response Procedure and 2-AOI-1-1, Relief Valve Stuck Open.
	BOP	2-ARP-9-3C, Alarm Response Procedure MAIN STEAM RELIEF VALVE OPEN, Window 25 Operator Actions: A. **CHECK** 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE, on Panel 2-9-47 and SRV Tailpipe Flow Monitor on Panel 2-9-3 for raised temperature and flow indications. B. **REFER TO** 2-AOI-1-1, Relief Valve Stuck Open. C. N/A
	BOP	2-AOI-1-1, Relief Valve Stuck Open **NOTE** Once a MSRV is operated, a time delay of 15 to 30 seconds can be expected before a response can be detected on 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE. ICS can be used to monitor the discharge tailpipe temperature, but the appropriate indications on 2-TR-1-1 must be confirmed.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 4 Page 2 of 3

Event Description: Steam Relief Valve (SRV) Fails Open

Time	**Position**	**Applicant's Actions or Behavior**
	BOP	4.1 Immediate Action [1] **IDENTIFY** stuck open Relief Valve by **OBSERVING** the following: • 2-FMT-1-4, SRV TAILPIPE FLOW MONITOR, on Panel 2-9-3, <u>OR</u> • 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE recorder, on Panel 2-9-47
	OATC	[2] **IF** Relief Valve transient occurred while operating above 90% power, **THEN REDUCE** Reactor Power to ≤ 90% RTP with Recirc Flow.
	NRC	**See Event 5, Reactor Power Reaction on page 10 of 40 for Reactor Power reduction actions.**
	NRC	**SRV 1-179 will close when cycled in Step [3].**
	BOP	[3] **WHILE OBSERVING** the indications for the affected Relief Valve on the Acoustic Monitor; **CYCLE** the affected Relief Valve Control Switch as required up to three times: • CLOSE to OPEN to CLOSE positions [4] N/A
	BOP	Section 4.2.4, Other Actions and Documentation [1] **NOTIFY** Reactor Engineering of current conditions
	Driver	**When contacted as the Reactor Engineer, acknowledge any direction or information given.**
	BOP	[2] **IF** ANY EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). [3] **REFER TO** Technical Specifications Sections 3.5.1 and 3.4.3 for Automatic Depressurization System and Relief Valve operability requirements. [4] **INITIATE** Suppression Pool Cooling as necessary to maintain Suppression Pool Temperature less than 95°F. [5] N/A [6] **INITIATE** a Condition Report (CR) for the valve.

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 4 Page 3 of 3

Event Description: Steam Relief Valve (SRV) Fails Open

Time	Position	Applicant's Actions or Behavior
	NUSO	Technical Specification 3.4.3 – Safety/Relief Valves (S/RVs) LCO 3.4.3 The safety function of 12 S/RVs shall be OPERABLE. Determines that an information only LCO would be required, as 12 SRVs are operable.
	NRC	**End of Event 4 and 5. Request that the driver insert Event 6, Control Rod Drift Out.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 5 Page 1 of 3

Event Description: Reactor Power Reduction

Symptoms/Cues: Event is initiated by the crew in response to Event 4 (SRV 1-179 Fails Open)

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 5 is entered by the Crew in response to Event 4. No action is required by the driver to enter Event 5.**
	NRC	**The crew may elect to reduce Reactor Power to less than 90% by any of the following means:** • **Upper Power Runback, followed by lowering flow as necessary using the Recirc Master Control Pushbuttons** • **Mid-Power Runback; stopping the Reactor Power change as necessary below 90%** • **Core Flow Runback; stopping the Reactor Power change as necessary below 90%** • **Recirc Master Control Pushbuttons** – **2-HS-96-33, LOWER SLOW** – **2-HS-96-34, LOWER MEDIUM** – **2-HS-96-35, LOWER FAST**
	OATC	Reduces Reactor Power using Core Flow in accordance with 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.

Op Test No.: 22-04	Scenario No. NRC-3	Event No.: 5	Page 2 of 3

Event Description: Reactor Power Reduction

Symptoms/Cues: Event is initiated by the crew in response to Event 4 (SRV 1-179 Fails Open)

Time	Position	Applicant's Actions or Behavior
	OATC	[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-96-33, **LOWER** SLOW • 2-HS-96-34, **LOWER** MEDIUM • 2-HS-96-35, **LOWER** FAST
	OATC	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%.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04	Scenario No. NRC-3	Event No.: 5	Page 3 of 3

Event Description: Reactor Power Reduction

Time	Position	Applicant's Actions or Behavior
	OATC	[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. [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 5. The crew will continue with actions of Event 4.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04	Scenario No. NRC-3	Event No.: 6	Page 1 of 5

Event Description: Control Rod Drift Out

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Control Rod 30-11 will drift out

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6, Control Rod Drift Out.**
	OATC	Acknowledges and reports the following alarm to the NUSO: • CONTROL ROD DRIFT, 2-9-5A, Window 28 • CONTROL ROD WITHDRAWL BLOCK, 2-9-5A, Window 7 • RBM DOWNSCALE, 2-9-5A, Window 31
	NUSO	Directs the OATC to respond in accordance with the appropriate Alarm Response and Abnormal Operating Procedures.
	OATC	2-ARP-9-5A, Alarm Response Procedure CONTROL ROD DRIFT, Window 28 Operator Action: A. **DETERMINE** which rod is drifting from Full Core Display. B. N/A 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

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 6 Page 2 of 5

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	OATC	Subsequent Actions: [1] N/A [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 or an AUO acknowledge any information or 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] **DECLARE** Control Rod INOPERABLE per Tech Spec 3.1.3. [10] **REFER** TO 2-AOI-85-7 Mispositioned Control Rod. [11] **INITIATE** Condition Report/Work Order. [12] **NOTIFY** Reactor Engineer to perform the following: • **EVALUATE** condition of the 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.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 6 Page 3 of 5

Event Description: Control Rod Drift Out

Time	Position	Applicant’s Actions or Behavior
	OATC	[14 – 16] N/A [17] **NOTIFY** Reactor Engineer to **EVALUATE** impact on preconditioning envelope, prior to returning to normal power operation.
	OATC	2-AOI-85-7, Mispositioned Control Rod (if entered) Immediate Actions: None Subsequent Actions: [1] **STOP** all intentional Control Rod movement. [2] **IF** Control Rod is determined to be mispositioned, **THEN NOTIFY** the following: • Reactor Engineer (RE) • Shift Technical Advisor (STA) • Unit Supervisor • Shift Manager (SM) • Operations Superintendent [3] N/A [4] N/A [5] **CHECK** the following radiation recorders for rise in radiation activity to determine if fuel damage occurred: • MAIN STEAM LINE RADIATION, 2-RR-90-135 (Panel 2-9-2) • OFFGAS RADIATION, 2-RR-90-266 (Panel 2-9-2) • OG PRETREATMENT RAD MON RTMR, 2-RM-90-157, on Panel 2-9-10 • OFFGAS RAD MON RTMR, 2-RM-90-160, on Panel 2-9-10 [6] N/A

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 6 Page 4 of 5

Event Description: Control Rod Drift Out

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>OATC</td><td colspan="2">[7] INTIATE a Service Request/PER for Control Rod error or mispositioned Control Rod.
[8] N/A
[9] NOTIFY Reactor Engineer to perform the following when time permits:
• EVALUATE possible consequences
• DOCUMENT in Reactor Engineer log
[10] N/A</td></tr>
<tr><td></td><td>Driver</td><td colspan="2">If contacted as the Reactor Engineer, acknowledge any information or direction given.</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">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.</td></tr>
<tr><td></td><td>NUSO</td><td>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.</td><td>COMPLETION TIME:
C.1 – 3 hours
C.2 – 4 hours</td></tr>
</table>

Op Test No.: 22-04	Scenario No. NRC-3	Event No.: 6	Page 5 of 5

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	NRC	**Tech Spec 3.10.8, Shutdown Margin (SDM) Test – Refueling is not applicable to current plant conditions.**
	NRC	**End of Event 6. Request that the Driver insert Event 7, Stator Colling Water Pump Trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 7 Page 1 of 3

Event Description: Stator Cooling Water Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 2A Stator Cooling Water Pump will trip, and 2B Stator Cooling Water Pump will not automatically 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, 2A 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 an automatic Reactor SCRAM.** **Critical Task Failure Criteria:** **The operating crew fails to start the Standby Stator Cooling Water Pump and a 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	2-ARP-9-7A, Alarm Response Procedure GEN STATOR COOLANT SYS ABNORMAL, Window 22 **NOTE** The control room alarm typer can be used to confirm this alarm. 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.

Op Test No.: 22-04	Scenario No. NRC-3	Event No.: 7	Page 2 of 3
Event Description: Stator Cooling Water Pump Trip			

Time	**Position**	**Applicant's Actions or Behavior**
	BOP	2. N/A 3. N/A 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	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
	Driver	**If contacted as the Turbine Building AUO, Work Control NUSO, or Electrical Maintenance, acknowledge any report or direction given.** **After 2 minutes, report as the Turbine Building AUO that 2A Stator Cooling Water Pump is hot to the touch.**
	BOP	2-ARP-9-8A, Alarm Response Procedure 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%.)

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 7 Page 3 of 3

Event Description: Stator Cooling Water Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	C. N/A D. **DISPATCH** personnel to Stator Coolant Unit to investigate.
	NRC	**End of Event 7. Request that that driver insert Event 8, SRV Fails Open / Tail Pipe Leak.**

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 1 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. SRV 1-179 will fail open and cannot be closed.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to fail SRV 1-179 OPEN and cause an SRV Tail Pipe leak in the Drywell.**
	NRC	**SRV 1-179 previously failed open and was closed earlier in this scenario. The crew will be unable to close SRV 1-179.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • MAIN STEAM RELIEF VALVE OPEN, 2-9-3C, Window 25.
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the Alarm Response Procedure and 2-AOI-1-1, Relief Valve Stuck Open.
	BOP	2-ARP-9-3C, Alarm Response Procedure MAIN STEAM RELIEF VALVE OPEN, Window 25 Operator Actions: A. **CHECK** 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE, on Panel 2-9-47 and SRV Tailpipe Flow Monitor on Panel 2-9-3 for raised temperature and flow indications. B. **REFER TO** 2-AOI-1-1, Relief Valve Stuck Open. C. N/A
	BOP	2-AOI-1-1, Relief Valve Stuck Open **NOTE** Once a MSRV is operated, a time delay of 15 to 30 seconds can be expected before a response can be detected on 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE. ICS can be used to monitor the discharge tailpipe temperature, but the appropriate indications on 2-TR-1-1 must be confirmed.

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 2 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Time	Position	Applicant's Actions or Behavior
	BOP	4.1 Immediate Action [1] **IDENTIFY** stuck open Relief Valve by **OBSERVING** the following: • 2-FMT-1-4, SRV TAILPIPE FLOW MONITOR, on Panel 2-9-3, OR • 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE recorder, on Panel 2-9-47 [2] N/A [3] **WHILE OBSERVING** the indications for the affected Relief Valve on the Acoustic Monitor; **CYCLE** the affected Relief Valve Control Switch as required up to three times: • CLOSE to OPEN to CLOSE positions [4] N/A
	BOP	4.2 Subsequent Action 4.2.1 N/A 4.2.2 Attempt to close valve from Panel 2-9-3: [1] **PLACE** the SRV TAILPIPE FLOW MONITOR POWER SWITCH in the OFF position. [2] **PLACE** the SRV TAILPIPE FLOW MONITOR POWER SWITCH in the ON position. [3] N/A [4] **PLACE** 2-XS-1-202, MSRV AUTO ACTUATION LOGIC INHIBIT in INHIBIT. [5] N/A [6] **PLACE** 2-XS-1-202, MSRV AUTO ACTUATION LOGIC INHIBIT in AUTO.

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 3 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Time	Position	Applicant's Actions or Behavior

BOP

NOTES

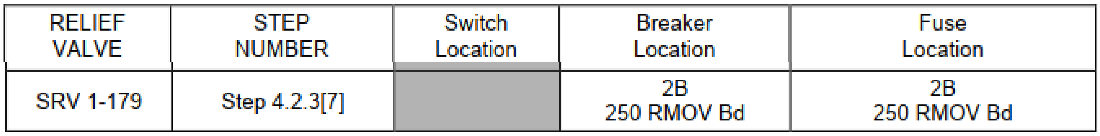
1) Only the appropriate sections for the stuck open relief valve are required to be performed.

2) The ADS Valves that have more than one power supply will AUTO TRANSFER on a loss of power, and are NORMAL SEEKING.

3) ADS Relief Valves with hand-switches on Panel 25-32 are listed below and should be operated from that location first.

4) When opening breakers and pulling fuses, opening the breakers is the preferred method when time permits. However, the breakers with multiple locations will require opening each breaker to de-energize the control circuit. In this case, pulling the fuses from Panel 25-32 may be quicker than opening the breakers.

[7] If the SRV Valve did not close, **THEN** PERFORM the appropriate section from the table below.



RELIEF VALVE	STEP NUMBER	Switch Location	Breaker Location	Fuse Location
SRV 1-179	Step 4.2.3[7]		2B 250 RMOV Bd	2B 250 RMOV Bd

4.2.3 Attempt to close valve from outside the Control Room:

NOTES

1) 2-PCV-1-179 controls have been removed from Panel 25-32.

2) Attachment 1 may be address for fuse and breaker information.

[7] **IF** 2-PCV-1-179 is NOT closed, **THEN PERFORM** the following:

[7.1] **REMOVE** the power from 2-PCV-1-179 by performing one of the following:

A. **OPEN** breaker (Preferred method) 2B 250V RMOV, compartment 8C2

OR

B. In 2B 250V RMOV, compartment R8A (backside) **PULL** Fuse FU2-1-179 (1FU and 2FU) as necessary.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04	Scenario No. NRC-3	Event No.: 8	Page 4 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed to pull fuses or open the breaker for SRV 1-179, wait 2 minutes and insert Event 18 remove power from SRV 1-179. Report to the crew that their chosen method of de-energizing SRV 1-179 is complete (either breaker open or fuses pulled).**
	BOP	[7.2] **IF** the valve does NOT close, **THEN CLOSE** breaker or **REINSTALL** fuses removed in Step 4.2.3 [7.1]. [7.3] **CONTINUE** at Step 4.2.4
	Driver	**When directed to close the breaker or re-install fuses, acknowledge the direction. Wait 30 seconds and insert Event 28 to restore power to SRV 1-179. Report to the Control Room that either the breaker is closed or fuses have been replaced (which ever method was selected by the crew).**
	BOP	4.2.4 Other actions and documentation [1] **NOTIFY** Reactor Engineering of current conditions.
	Driver	**If contacted as the Reactor Engineer, acknowledged any information or direction given.**
	BOP	[2] **IF** ANY EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). [3] **REFER TO** Technical Specifications Sections 3.5.1 and 3.4.3 for Automatic Depressurization System and relief valve operability requirements. [4] **INITIATE** Suppression Pool Cooling as necessary to maintain Suppression Pool Temperature less than 95°F. [5] **IF** the Relief Valve can **NOT** be closed **AND** Suppression Pool Temperature can **NOT** be maintained less than or equal to 95°F, **THEN PLACE** the Reactor in Mode 4 in accordance with 2-GOI-100-12A, Unit Shutdown From Power Operation To Cold Shutdown. [6] Initiate a Condition Report (CR) for the valve.
	CREW	Monitors Drywell Temperature and Drywell/Suppression Chamber Pressure.
	NUSO	Directs the OATC to insert a manual Reactor SCRAM before the Reactor SCRAMS on high Drywell Pressure (2.45 psig).

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 5 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Time	Position	Applicant's Actions or Behavior
	NRC	**Only the Immediate Actions for the Reactor SCRAM are listed below due to the large number of subsequent actions in the Abnormal Operating Instruction.**
	OATC	2-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 2-HS-99-5A/S3A, REACTOR SCRAM A, and 2-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 3-9-5. [2] **PLACE** 2-HS-99-5A/S1, REACTOR MODE SWITCH in SHUTDOWN. [3] N/A [4] **IF** Reactor Power is 5% or BELOW, **THEN 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
	NRC	**EXAMINER NOTE: When RFPT 2A and 2B speed lowers below 750 RPM following the SCRAM, Event 10, 2C Reactor Feedwater Pump (RFPT) Trip, is automatically entered. No action is required by the driver to insert Event 10. See page 38 of 40 for Event 9 actions.**
	BOP	Monitors and reports Suppression Chamber and Drywell Pressure, and reports when Drywell Pressure exceeds 2.45 PSIG to the NUSO.
	NUSO	When Drywell Pressure exceeds 2.45 PSIG, enters 2-EOI-1, RPV Control, and 2-EOI-2, Primary Containment Control. Updates the crew on entering the EOIs.

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 6 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	2-EOI-1, RPV Control MODES 1-3

RC-1

ENSURE Reactor SCRAM

RC-2

Can it be determined that the Reactor will remain subcritical without Boron under all conditions	**YES**

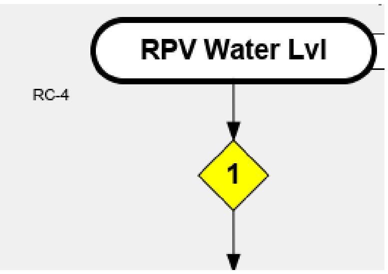
RC-3

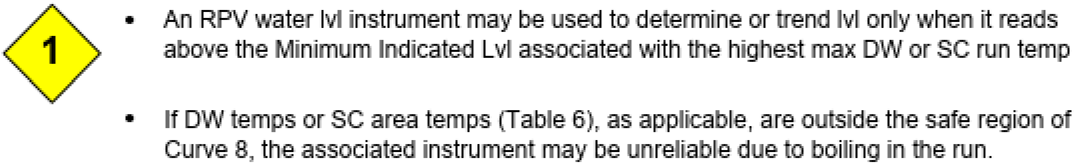
IF	THEN
It is determined that Core Damage is occurring due to a loss of cooling	**NO ACTION REQUIRED**
RPV Water Level CANNOT be determined	**NO ACTION REQUIRED**
It is anticipated that available injection subsystems (Table L-3) alone CANNOT assure Adequate Core Cooling	**NO ACTION REQUIRED**

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 7 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Time	Position	Applicant's Actions or Behavior




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.

NUSO

RC/L-1

ENSURE each as required:
PCIS isolations (Groups 1, 2 and 3)
ECCS
RCIC

RC/L-2

IF	THEN
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**
PC Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7	**NO ACTION REQUIRED**

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 8 of 17

Event Description: SRV Fails Open / Tail Pipe 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/L-3

RESTORE and **MAINTAIN** RPV Water Level between (+) 2 in. and (+) 51 in. with ANY Preferred Injection Systems (Table L-1)

➢ OK to use ANY Alternate Injection Subsystems (Table L-2)

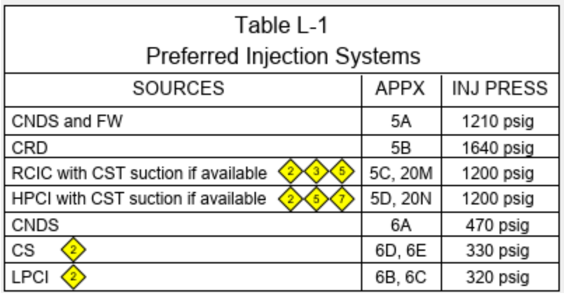


Table L-1
Preferred Injection Systems

<table>
<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 ◇5</td><td>5C, 20M</td><td>1200 psig</td></tr>
<tr><td>HPCI with CST suction if available ◇2 ◇5 ◇7</td><td>5D, 20N</td><td>1200 psig</td></tr>
<tr><td>CNDS</td><td>6A</td><td>470 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>

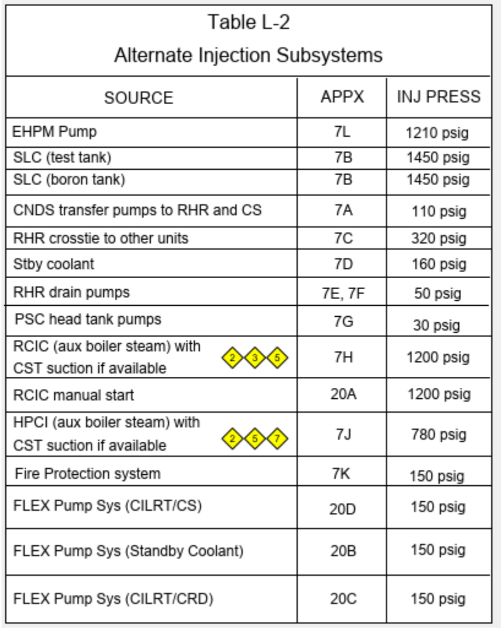


Table L-2
Alternate Injection Subsystems

<table>
<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 ◇5</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 ◇5 ◇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>


</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 9 of 17

Event Description: SRV Fails Open / Tail Pipe 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-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>A high Drywell Pressure ECCS signal exists (2.45 psig)</td><td>PREVENT injection from ONLY those Core Spray and LPCI Pumps NOT required for RPV injection before depressurizing below 330 psig (APPX 4)</td></tr>
<tr><td>EMERGENCY DEPRESSURIZATION is required or has been required</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Emergency Depressurization is anticipated</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Pool Temperature and Level CANNOT be maintained in a safe area of Curve 3 at the existing RPV Pressure
OR
Suppression Pool Level CANNOT be maintained in the safe area of Curve 4</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-2
<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 Drywell 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>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 10 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

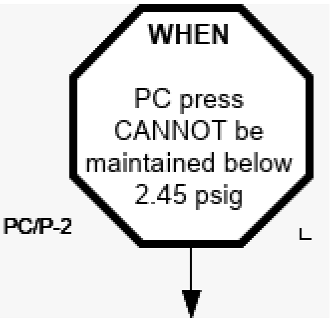
<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>


PC/C-1

<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>It is determined that Core Damage is occurring due to loss of cooling</td><td>NO ACTION REQUIRED</td></tr>
</table>

PC/P-1

MONITOR and CONTROL Primary Containment Pressure below 2.45 psig using the Vent system (AOI-64-1)



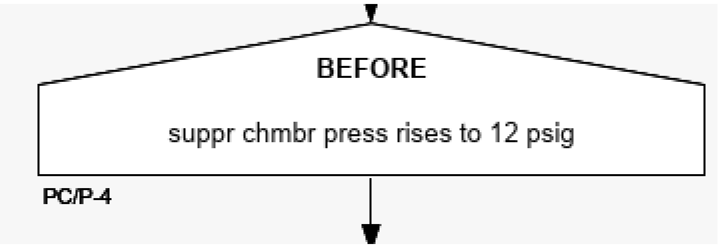
</td></tr>
</table>

Form 3.3-2 Required Operator Actions

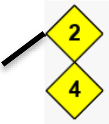
Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 11 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

<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-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>PC 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>PC Pressure CANNOT be maintained above (-) 2 psig</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Chamber Pressure approaches 0 psig</td><td>BEFORE 0 psig, STOP Suppression Chamber Sprays</td></tr>
<tr><td>Drywell Pressure approaches 0 psig</td><td>BEFORE 0 psig, STOP Drywell Sprays</td></tr>
</table>



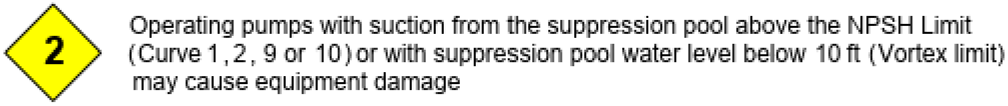
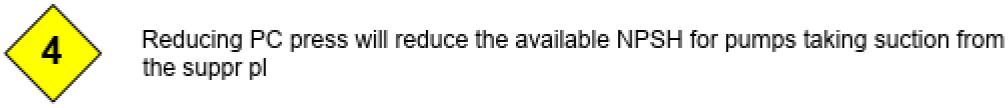
PC/P-5



<table>
<tr><td colspan="2">INITIATE Suppression Chamber Sprays
➢ Use only sources NOT required for continuous RPV injection
(APPX 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>
</table>

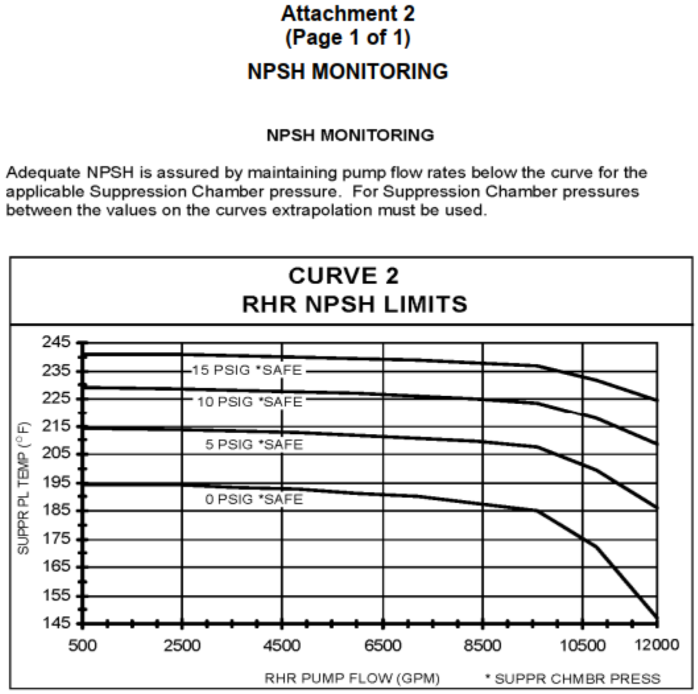
Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 12 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Time	**Position**	**Applicant's Actions or Behavior**
	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  Reducing PC press will reduce the available NPSH for pumps taking suction from the suppr pl
	NUSO	Directs the BOP to spray the Suppression Chamber in accordance with 2-EOI-Appendix-17C.
	BOP	2-EOI-Appendix-17C, RHR System Operation Suppression Chamber Sprays [1] <u>BEFORE</u> 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 SYSTEM I OUTBOARD INJECTION VALVE BYPASS SELECT in BYPASS • PLACE 2-HS-74-155B, LPCI SYSTEM 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 NUSO, **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 SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, switch in SELECT.

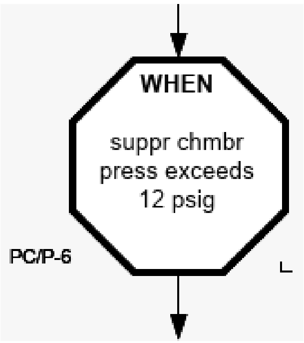
Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 13 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Time	Position	Applicant's Actions or Behavior
	BOP	[5.4] **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. [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 CHAMBERHBR/POOL ISOLATION VALVE. [5.7] **OPEN** 2-FCV-74-58 (72), RHR SYS 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 SYS 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.
	BOP	 **Attachment 2** **(Page 1 of 1)** **NPSH MONITORING** **NPSH MONITORING** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. **CURVE 2** **RHR NPSH LIMITS**

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 14 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

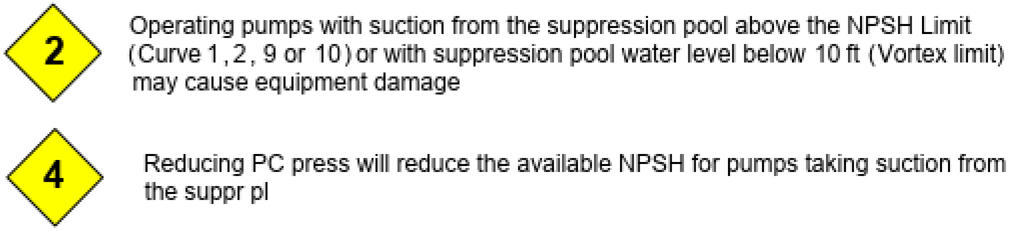
<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>[5.12] ENSURE RHRSW Pump supplying desired RHR Heat Exchanger(s).
[5.13] THROTTLE the following in-service RHRSW Outlet Valves to obtain 1700 to 4500 gpm RHRSW Flow:
• 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE
• 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE
• 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE
• 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE
[5.14] NOTIFY Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.</td></tr>
<tr><td></td><td>Driver</td><td>When notified as Chemistry, acknowledge any information or direction given.</td></tr>
<tr><td></td><td>NUSO</td><td>

PC/P-7 2 4
<table>
<tr><td colspan="2">IF Suppression Pool Level is below 19 ft 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 for continuous RPV Injection (APPX 17B, 20L)</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Needed to Augment Drywell Sprays</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 15 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

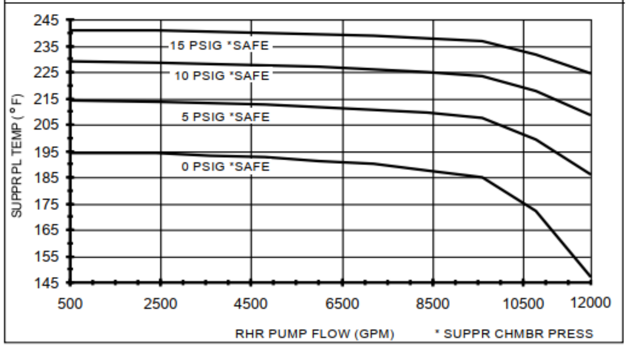
Time	Position	Applicant's Actions or Behavior
	NUSO	 2 Operating pumps with suction from the suppression pool above the NPSH Limit (Curve 1, 2, 9 or 10) or with suppression pool water level below 10 ft (Vortex limit) may cause equipment damage 4 Reducing PC press will reduce the available NPSH for pumps taking suction from the suppr pl
	Crew	**Critical Task:** **When Drywell Sprays are required, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit Curve to prevent challenging Primary Containment negative pressure capability.** **Critical Task Failure Criteria:** **Drywell Sprays are initiated with Drywell Pressure and Temperature outside the safe are of the Drywell Spray Initiation Curve (Curve 5).**
	NUSO	Directs the BOP to spray the Drywell in accordance with 2-EOI-Appendix-17B, RHR System Operation Drywell Sprays.
	NRC	**Event 9, Drywell Spray Failure, is automatically entered on simulator setup. No action is required by the driver to enter Event 9. See Event 9 on page 38 of 40.**
	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 VALVE BYPASS 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 shutdown. [4] N/A [5] N/A

Form 3.3-2 Required Operator Actions

page

Op Test No.: 22-04 Scenario No. NRC-3 Event No.: 8 Page 16 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

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 NUSO, **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. [6.7] **ENSURE CLOSED** 2-FCV-074-0007 (0030), RHR SYSTEM I (II) MINIMUM FLOW VALVE. [6.8] N/A [6.9] **MONITOR** RHR Pump NPSH using Attachment 2. **Attachment 2** **(Page 1 of 1)** **NPSH MONITORING** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. **CURVE 2** **RHR NPSH LIMITS** 

Op Test No.: 22-04	Scenario No. NRC-3	Event No.: 8	Page 17 of 17

Event Description: SRV Fails Open / Tail Pipe Leak

Time	Position	Applicant's Actions or Behavior
	BOP	[6.10] **ENSURE** RHRSW Pump supplying desired RHR Heat Exchanger(s). [6.11] **THROTTLE** the following in-service RHRSW Outlet Valves to obtain 1700 to 4500 gpm RHRSW Flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE [6.12] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	Driver	**When contacted as Chemistry, acknowledge any direction or information given.**
	NRC	**End of Event 8. When the crew is spraying the Drywell 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.: 22-04 Scenario No. NRC-3 Event No.: 9 Page 1 of 1

Event Description: Drywell Spray Failure

Symptoms/Cues: Event is automatically entered on simulator setup. When the crew attempts to spray the Drywell, the first loop of Drywell Sprays will fail.

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9 is entered on simulator setup; no action is required by the driver to insert Event 9.** **The first loop that the crew uses to spray the Drywell will fail (2-FCV-74-61(75), Drywell Spray Inboard Valve power failure), requiring the crew to use the opposite loop for Drywell Sprays. Suppression Chamber Sprays are not affected.**
	Driver	**Verify that when the crew attempts to spray the Drywell that the Event for the opposite loop is automatically cleared.**
	BOP	Attempts to Spray the Drywell in accordance with 2-EOI-Appendix-17B, RHR System Operation Drywell Sprays. Reports that the power to the first Inboard Spray Valve opened has failed to the NUSO. Continues the Drywell Spray procedure using the other loop. See page 35 of 40 for procedural actions.
	NRC	**End of Event 9. When the crew is spraying the Drywell 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.: 22-04 Scenario No. NRC-3 Event No.: 10 Page 1 of 1

Event Description: 2C Reactor Feedwater Pump (RFPT) Trip

Symptoms/Cues: Event is automatically entered on simulator setup. When the Reactor MODE Switch is placed in SHUTDOWN, 2C RFPT will trip.

Time	Position	Applicant's Actions or Behavior
	Driver	**Verify that 2C RFPT trips when 2A and 2B RFPT speeds are less than 750 rpm.**
	NRC	**During Reactor SCRAM Response, normally 2C RFPT will maintain Reactor Water Level while 2A and 2B RFPTs run back to manual control at 600 rpm. 2C RFPT will trip on Low Oil Pressure when 2A and 2B RFPTs have run back to 750 rpm and will not be able to be restarted.**
	OATC	Determines that 2C RFPT has tripped during SCRAM response and reports to the NUSO.
	OATC	2-EOI-Appendix-5A, Injection Systems Lineup Condensate/Feedwater 1. **IF** it is desired to use a Reactor Feed Pump that is in operation, **THEN CONTINUE** at step 12 to control the operating pump. 12. **SLOWLY ADJUST** RFPT speed UNTIL Feedwater Flow to the RPV is indicated, using ANY of the following methods on Panel 2-9-5: • Individual 2-HS-46-8A(9A), RFPT 2A(2B) SPEED CONTROL RAISE/LOWER switch in MANUAL GOVERNOR, **OR** • Individual 2-SIC-46-8(9), RFPT 2A(2B) SPEED CONTROL in MANUAL, **OR** • 2-LIC-46-5, REACTOR WATER LEVEL CONTROL, in MANUAL with individual 2-SIC-46-8(9), RFPT 2A(2B) SPEED CONTROL in AUTO 13. **ADJUST** RFPT speed as necessary to control injection using the methods of step 12. 14. N/A
	NRC	**End of Event 10. When the crew is spraying the Drywell and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high-pressure systems, end of Scenario.**

UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 1	**DAYS ON LINE** **383**	**Total Drywell Leakage (gpm)**	**Protected Equipment**
			2A Bus Duct Fan
	PRA (Phoenix) -Green	**1.55**	2A CRD Pump
Rx Power	500Kv GRID - Qualified	**Floor Drain (gpm)**	
100%	161Kv Grid -Qualified	**0.11**	
MWe 1280	**Last breaker closure**	**Equipment Drain (gpm)**	
	4/25/21 05:41	**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

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

2B Bust Duct Fan tagged for maintenance.

1B CRD Pump tagged for oil addition.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

IMs are preparing for maintenance in the Aux Instrument Room, and will be up to brief after turnover.

Perform Turning Gear Auto Start Test in accordance with 2-OI-47B, Main Turbine Lube Oil System, Sect. 6.3.

OPERATOR WORK AROUNDS OWAs – U0-1/U2-0 Burdens - U0 – 0/U2-0 Operator Challenges – U0 – 12/U2 - 6

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Facility: BFN Scenario Number: NRC-4
Scenario Source: NEW Op-Test Number: 22-04
Examiners: ____________________________ Operators: NUSO:___________________________
____________________________ OATC: __________________________
____________________________ BOP:____________________________

Initial Conditions: 74 % Reactor Power.

Turnover: 2C Condensate/Condensate Booster/Reactor Feedwater Pump are out of service.

Critical Tasks:

1. With Adequate Core Cooling available, lock out HPCI when Suppression Pool Level cannot be maintained above 12.75 feet to prevent Primary Containment damage.
2. Manually insert a Reactor SCRAM before Suppression Pool Level reaches 11.5 feet, to reduce the amount of energy in the vessel on a lowering Suppression Pool Water Level.
3. Emergency Depressurization is performed when Suppression Pool Water Level cannot be maintained above 11.5 feet to minimize the adverse effects on Suppression Pool heat capacity.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Alternate Reactor / Refuel Zone Fans
2.	HS-66-34A PMP-66-51A	C-BOP MC-BOP C-NUSO	Steam Packing Exhauster (SPE) Discharge Valve Closure – Standby SPE Fails to Auto Start
3.	PMP-23-1A ZLOHS231A_1/2	TS-NUSO	A1 Residual Heat Removal Service Water (RHRSW) Clearance
4.	ED08C	C-OATC MC-OATC C-BOP MC-BOP C-NUSO	2C Unit Board Trip
5.	TH03B	C-OATC TS-NUSO	2B Reactor Recirc Pump Trip
6.	N/A	R-OATC R-NUSO	Urgent Load Reduction
7.	PC14 RH04	M-ALL	Low Suppression Pool Water Level
8.	RP11	C-BOP C-NUSO	Inadvertent Main Steam Isolation Valve (MSIV) Closure

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

Events

1. The Balance of Plant Operator (BOP) will alternate Reactor and Refuel Zone Fans in accordance with 2-OI-30A, Refueling Zone Ventilation System, and 2-OI-30B, Reactor Zone Ventilation System.

2. 2A Steam Packing Exhauster Discharge Valve will close, causing low vacuum in the Steam Packing Exhaust System. The standby Steam Packing Exhauster will fail to automatically start. The crew will respond in accordance with Alarm Response Procedures and 2-OI-47C, Seal Steam System, to start the standby Steam Packing Exhauster.

3. Work Control will contact the crew in order to de-energize and hang a clearance on A1 Residual Heat Removal Service Water (RHRSW) Pump in preparation for scheduled maintenance. The Nuclear Unit Senior Operator will address Technical Specifications.

4. 2C Unit Board will trip and lock out due to an electrical fault. The crew will respond in accordance with Alarm Response procedures and Operating Instructions. In addition to the loss of the running Control Rod Drive (CRD) Pump, the 2C Condenser Circulating Water System (CCW) Pump Discharge Valve will fail to automatically close, requiring manual crew action to prevent a possible loss of Condenser Vacuum.

5. 2B Recirculation Pump will trip, requiring the crew to take action in accordance with 2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease. The NUSO will address Technical Specifications.

6. The Operator at the Controls (OATC) will enter the Urgent Load Reduction Procedure to exit operation in Region I/II of the Power to Flow Map and insert Control Rods until Reactor Power meets the requirements of 2-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease.

7. An un-isolable leak will develop in the Suppression Pool, requiring a Reactor SCRAM and Emergency Depressurization in accordance with Emergency Operating Instruction (EOI) 2, Primary Containment Control.

8. Following the Reactor SCRAM, the Main Steam Isolation Valves (MSIVs) will inadvertently close, requiring the crew to switch Reactor Pressure Control methods.

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: 3

1. **With Adequate Core Cooling available, lock out HPCI when Suppression Pool Level cannot be maintained above 12.75 feet to prevent Primary Containment damage.**
    1. **Safety Significance:**
       Prevent failure of Primary Containment from pressurization of the Suppression Chamber.
    2. **Cues:**
       Procedural Compliance.
       Suppression Pool Level indication.
    3. **Measured by:**
       Observation – HPCI Auxiliary Oil Pump placed in Pull to Lock.
    4. **Feedback:**
       HPCI does not Auto initiate.
       No RPM indication on HPCI.
    5. **Critical Task Failure Criteria:**
       HPCI is operated below a Suppression Pool Water Level of 12.75 feet, thus pressurizing the Suppression Chamber from the HPCI exhaust.
2. **Manually insert a Reactor SCRAM before Suppression Pool Level reaches 11.5 feet, to reduce the amount of energy in the vessel on a lowering Suppression Pool Water Level.**
    1. **Safety Significance:**
       Prevent failure of Primary Containment from over pressurization.
    2. **Cues:**
       Procedural Compliance.
       Suppression Pool Level indication.
    3. **Measured by:**
       Observation – Both RPS SCRAM switches are depressed.
    4. **Feedback:**
       All Control Rods insert to their full in position.
    5. **Critical Task Failure Criteria:**
       The operating crew allows Suppression Pool Water Level to lower past 11.5 feet without inserting a manual SCRAM signal.

3. **Emergency Depressurization is performed when Suppression Pool Water Level cannot be maintained above 11.5 feet to minimize the adverse effects on Suppression Pool heat capacity.**

   1. **Safety Significance:**
      Prevent failure of Primary Containment.

   2. **Cues:**
      Procedural Compliance.
      Suppression Pool Level trend.

   3. **Measured by:**
      Observation – the Nuclear Unit Senior Operator (NUSO) determines (as indicated by announcement or observable transition to 2-C-2, RPV Emergency Depressurization) the Emergency Depressurization is required at or before Suppression Pool Water Level lowers below 11.5 feet.
      AND
      Observation – the Unit Operator opens 6 ADS valves when directed by the Nuclear Unit Senior Operator (NUSO).

   4. **Feedback:**
      Reactor Pressure trend.
      Suppression Pool Water Level.
      MSRV status indication.

   5. **Critical Task Failure Criteria:**
      The crew does not perform Emergency Depressurization without delay and in a controlled manner when Suppression Pool Water Level cannot be maintained above 11.5 feet.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 1 Page 1 of 6

Event Description: Alternate Reactor / Refuel Zone Fans

Symptoms/Cues: Crew is cued by the turnover sheet or by the Simulator Operator as requested by the Chief Examiner

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, Alternate Reactor / Refuel Zone Fans after assuming the shift, request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to alternate Reactor and Refuel Zone Fans.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to alternate Reactor and Refuel Zone Fans.** **If contacted by the crew as the Reactor Building Assistant Unit Operator (AUO) acknowledge any direction given.**
	NUSO	Directs the BOP to alternate Reactor and Refuel Zone Fans in accordance with 2-OI-30A, Refueling Zone Ventilation System, and 2-OI-30B, Reactor Zone Ventilation System.
	BOP	2-OI-30A, Refueling Zone Ventilation System Section 6.1, Alternating Refueling Zone Supply and Exhaust Fans [1] **NOTIFY** Unit 1 and Unit 3 Operators that the Refuel Zone fans are being alternated.
	Driver	**When contacted as Unit 1 and Unit 3 Operators, acknowledge that Unit 2 is alternating Refueling and Reactor Zone Fans.**
	BOP	[2] **VERIFY** the Refueling Zone Supply and Exhaust Fans are operating. **REFER TO** Section 5.1. [3] **REVIEW** precautions and limitations in Section 3.0. (Completed during pre-shift brief)

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 1 Page 2 of 6

Event Description: Alternate Reactor / Refuel Zone Fans

Time	Position	Applicant's Actions or Behavior
	BOP	**NOTES** 1) The preferred method to start the alternate Refueling Zone Supply and Exhaust Fans is to use the common control switch, 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, on Panel 2-9-25. 2) Refueling Zone Supply and Exhaust Dampers, 2-FCO-64-5, 6, 9, and 10 will open or close automatically as necessary when fans are stopped and started. 3) Refueling Zone Supply and Exhaust Fans are alternated every six weeks. [4] **PLACE** 2-HS-64-3A, REFUELING ZONE FANS AND DAMPERS, in OFF. [5] **VERIFY** that the two red lights A(B) extinguish and the two green lights A(B) illuminate above 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS **NOTE** If any damper does not meet the requirements of step 6.1[6], IMMEDIATELY notify the Unit supervisor to evaluate SCIV damper operability (refer to TRM appendix A). If any listed damper indicates not full closed, it should be considered inoperable for its SCIV function, and the required actions of Tech Spec LCO 3.6.4.2 entered for all units. [6] **VERIFY** the red (open) damper position indication lights extinguish and the green (closed) lights illuminate above the following control switches: • 2-HS-64-5, REFUEL ZONE SUPPLY OUTBOARD ISOLATION DAMPER • 2-HS-64-6, REFUEL ZONE SUPPLY INBOARD ISOLATION DAMPER • 2-HS-64-9, REFUEL ZONE EXHAUST OUTBOARD ISOLATION DAMPER • 2-HS-64-10, REFUEL ZONE EXHAUST INBOARD ISOLATION DAMPER

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 1 Page 3 of 6

Event Description: Alternate Reactor / Refuel Zone Fans

Time	Position	Applicant's Actions or Behavior
	BOP	[7] **PLACE** 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, in SLOW A (SLOW B) to start alternate fans. [8] **VERIFY** that the two green lights A (B) extinguish and the two red lights A (B) illuminate above 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS. [9] **VERIFY** the red (open) damper position indication lights illuminate and green (closed) lights extinguish above the following control switches: • 2-HS-64-5, REFUEL ZONE SUPPLY OUTBOARD ISOLATION DAMPER • 2-HS-64-6, REFUEL ZONE SUPPLY INBOARD ISOLATION DAMPER • 2-HS-64-9, REFUEL ZONE EXHAUST OUTBOARD ISOLATION DAMPER • 2-HS-64-10, REFUEL ZONE EXHAUST INBOARD ISOLATION DAMPER [10] **IF** necessary to transfer Refuel Zone Supply and Exhaust Fans to fast, five minutes should be allowed after slow start to allow discharge dampers to fully open, **THEN PERFORM** the following: [10.1] **PLACE** 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, in FAST A (FAST B). [10.2] **VERIFY** that the two green lights A (B) remain extinguished and the two red lights A(B) remain illuminated above 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS. [11] **VERIFY** the following conditions: • 2-PDI-64-22, SUPPLY FANS FILTER DIFFERENTIAL PRESSURE, indicates less than 0.6 inches H_2O at the Reactor Building/Refuel Floor Supply Fan intake room at El 565' • 1-PDS-064-0061A/C, REFUELING ZONE STATIC PRESSURE INTERLOCK, on Refuel Floor Panel 25-220 indicates between (negative) (-) 0.25 inches H_2O and (-) 0.40 inches

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04	Scenario No. NRC-4	Event No.: 1	Page 4 of 6

Event Description: Alternate Reactor / Refuel Zone Fans

Time	Position	Applicant's Actions or Behavior
	BOP	2-OI-30B, Reactor Zone Ventilation System Section 6.1, Alternating Reactor Zone Supply and Exhaust Fans [1] **VERIFY** the Reactor Zone Supply and Exhaust Fans are operating. **REFER TO** Section 5.1. [2] **REVIEW** precautions and limitations in Section 3.0. (Completed during pre-shift brief) **NOTES** 1) The preferred method to start the standby Reactor Zone Supply and Exhaust Fans is to use the common control switch (2-HS-64-11A) on Panel 2-9-25. 2) Reactor Zone supply and exhaust dampers, 2-FCO-64-13, 14, 42, and 43 will open or close automatically as necessary when fans are stopped and started. 3) The Steam Vault Exhaust Booster Fan should normally be in service whenever the Unit is operating with Reactor Building Ventilation in service and fans in fast speed. Operation of the Steam Vault Exhaust Booster Fan with Reactor Zone Exhaust fans out of service is an ALARA concern due to backflow into the Reactor Building lower level ventilation ductwork. However, the Steam Vault Exhaust Booster fan may remain in service with Reactor Zone Exhaust fans out of service to cool the steam tunnel for short durations such as alternating fans, cycling Reactor zone dampers, or RPS power transfers. 4) If any damper does not meet the requirements of Tech Spec, IMMEDIATELY notify the Unit Supervisor to evaluate SCIV damper operability (refer to TRM appendix A). If any listed damper indicates not full closed, it should be considered inoperable for its SCIV function, and the required actions of Tech Spec LCO 3.6.4.2 entered for all units. [3] N/A [4] N/A [5] **PLACE** 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS, in OFF.

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 1 Page 5 of 6

Event Description: Alternate Reactor / Refuel Zone Fans

Time	Position	Applicant's Actions or Behavior
	BOP	[6] **VERIFY** dampers close and fans stop as indicated by illuminated green lights above the following switches: • 2-HS-64-13, REACTOR ZONE SUPPLY OUTBOARD ISOLATION DAMPER • 2-HS-64-14, REACTOR ZONE SUPPLY INBOARD ISOLATION DAMPER • 2-HS-64-42, REACTOR ZONE EXHAUST INBOARD ISOLATION DAMPER • 2-HS-64-43, REACTOR ZONE EXHAUST OUTBOARD ISOLATION DAMPER • 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS [7] **PLACE** 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS, in SLOW A (SLOW B) to start alternate fans. [8] **VERIFY** dampers open and fans start as indicated by illuminated red lights above the following switches: • 2-HS-64-13, REACTOR ZONE SUPPLY OUTBOARD ISOLATION DAMPER • 2-HS-64-14, REACTOR ZONE SUPPLY INBOARD ISOLATION DAMPER • 2-HS-64-42, REACTOR ZONE EXHAUST INBOARD ISOLATION DAMPER • 2-HS-64-43, REACTOR ZONE EXHAUST OUTBOARD ISOLATION DAMPER • 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS [9] **IF** fast speed Reactor Zone Supply and Exhaust Fan operation is required, five minutes should be allowed after slow start for the discharge dampers to fully open, **THEN** [9.1] **PLACE** 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS, in FAST A (FAST B). [9.2] **VERIFY** that the two green lights A(B) remain extinguished and the two red lights A(B) remain illuminated above 2-HS-64-11A, REACTOR ZONE FANS AND DAMPERS.

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 1 Page 6 of 6

Event Description: Alternate Reactor / Refuel Zone Fans

Time	Position	Applicant's Actions or Behavior
	BOP	[10] **VERIFY** the following conditions: A. 2-PDI-64-22, SUPPLY FANS FILTER DIFF PRESS, at the filter in Reactor/Refuel Zone supply fan room, El 565' indicates less than 0.6 inches H_2O. B. 2-PDIC-64-2, REACTOR ZONE PRESS DIFFERENTIAL, on Panel 25-213 located at R10-P El 639', indicates between (-) 0.25 inches and (-) 0.40 inches H_2O. C. **IF** 2-PDIC-64-2, REACTOR ZONE PRESS DIFFERENTIAL, is not between (-) 0.25 inches and (-) 0.40 inches H_2O, **THEN REFER TO** 2-AOI-30B-1, Reactor Building Ventilation Failure. [11] N/A
	BOP	Informs the NUSO that Refuel and Reactor Zone Fans have been alternated and are running in slow speed.
	Driver	**If contacted as an AUO to check Refuel and/or Reactor Zone Differential Pressure locally, report that the respective gauges read between (-) 0.25 and (-) 0.40 inches H_2O.**
	NRC	**End of Event 1; request that the driver insert Event 2, Steam Packing Exhauster (SPE) Discharge Valve Closure – Standby SPE Fails to Auto Start.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 2 Page 1 of 2

Event Description: Steam Packing Exhauster (SPE) Discharge Valve Closure – Standby SPE Fails to Auto Start

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; 2A SPE will trip and the standby SPE will not automatically start

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to cause 2A Steam Packing Exhauster Discharge Valve closure.**
	BOP	Acknowledges and reports the following alarm: • STEAM PACKING EXHAUSTER VACUUM LOW, 2-9-7A, Window 12
	NUSO	Directs the BOP to respond in accordance with 2-ARP-9-7A, Alarm Response Procedure.
	BOP	2-ARP-9-7A, Alarm Response Procedure STEAM PACKING EXHAUSTER VACUUM LOW, 2-9-7A, Window 12 Operator Action: A. **CHECK** following: 1. Standby 2-HS-66-51A, STEAM PACKING EXHAUSTER BLOWER 2B, started. (Both red and green indicating lights at handswitch of running standby blower will be illuminated; no indicating lights may be illuminated at handswitch of tripped blower.) 2. 2-HS-66-35A, STEAM PACKING EXHR 2B DISCHARGE VLV, opens. 3. 2-HS-66-34A, STEAM PACKING EXHR 2A DISCHARGE VLV, closes. B. **IF** standby blower fails to start, **THEN START** standby OR **CHECK** normal in service. **REFER TO** 2-OI-47C, Steam Seal System. C. **IF** blower is running, **PERFORM** the following: 1. **THROTTLE** in-service 2-HS-66-35A, STEAM PACKING EXHAUSTER 2B DISCHARGE VALVE, UNTIL SPE Vacuum, as indicated on 2-PI-66-54, STEAM PACKING EXHAUST VACUUM is between 10" and 12" Vacuum, or as appropriate for plant conditions. **REFER TO** 2-OI-47C, Steam Seal System.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04	Scenario No. NRC-4	Event No.: 2	Page 2 of 2

Event Description: Steam Packing Exhauster (SPE) Discharge Valve Closure – Standby SPE Fails to Auto Start

Time	Position	Applicant's Actions or Behavior
	BOP	2. **ENSURE** 2-FI-2-42, SJAE/OG CNDR CNDS FLOW, between 2×10^6 lbm/hr and 3×10^6 lbm/hr to prevent tripping SPE due to inadequate cooling. **REFER TO** 2-OI-2, Condensate System.
	BOP	Determines that the standby SPE did not automatically start, and starts the Standby Steam Packing Exhauster.
	BOP	D. N/A E. **IF** fans or dampers have failed, **THEN CHECK** electrical feeds on 480V Turb MOV Bd 2C, bkr 7E (SPE A) and 480V Turb MOV Bd 2B, bkr 7E (SPE B). **NOTE** Resetting running pump handswitch target will re-enable light indication and autostart circuit of tripped pump. Do not reset target if cause of blower trip is unknown or it is undesired to allow the blower which tripped to restart in auto. F. N/A
	Driver	**If contacted as the Turbine Building AUO, Work Control/Outside NUSO, or Electrical Maintenance to investigate the reason for the 2A Steam Packing Exhauster Discharge valve closure and/or the failure of the Standby Steam Packing Exhauster to start, acknowledge any information or direction given.**
	NRC	**End of Event 2. Request that the driver insert Event 3, A1 Residual Heat Removal System (RHRSW) Clearance.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 3 Page 1 of 2

Event Description: A1 Residual Heat Removal Service Water (RHRSW) Clearance

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, contact the NUSO as the Work Control NUSO and inform him that A1 RHRSW must be de-energized and tagged out in preparation for maintenance. Inform the NUSO that a team has just been sent to open the circuit breaker and within three (3) minutes the tags will be hanging.** **Wait three (3) minutes and insert Event 3 to de-energize A1 RHRSW Pump.**
	NUSO	Updates the crew and references Technical Specifications. Technical Specification 3.7.1 – Residual Heat Removal Service Water (RHRSW) System LCO 3.7.1 ---------------------------------------NOTE-- The number of required RHRSW pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for ε 24 hours. Four RHRSW subsystems shall be OPERABLE with the number of OPERABLE pumps as listed below: 1. One unit fueled - four OPERABLE RHRSW pumps. 2. Two units fueled - six OPERABLE RHRSW pumps. 3. Three units fueled - eight OPERABLE RHRSW pumps. Applicability: MODES 1, 2 and 3 Condition: A. One required RHRSW Pump inoperable.

Op Test No.: 22-04	Scenario No. NRC-4	Event No.: 3	Page 2 of 2

Event Description: A1 Residual Heat Removal Service Water (RHRSW) Clearance

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** A.1 ---------------NOTES-------------- 1. Only applicable for the 2 units fueled condition. 2. Only four RHRSW pumps powered from separate 4 kV shutdown board are required to be OPERABLE if the other fueled unit has been in MODE 4 or 5 for > 24 hours. -------------------------------------- Verify five RHRSW pumps powered from separate 4KV Shutdown Boards are OPERABLE. OR A.2 Restore required RHRSW Pump to OPERABLE status.	**COMPLETION TIME:** A.1 – Immediately A.2 – 30 days
	NRC	**End of Event 3. Request that the driver insert Event 4, 2C Unit Board Trip.**	

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 4 Page 1 of 4

Event Description: 2C Unit Board Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; 2C Unit Board will be de-energized, causing a loss of electrical loads

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4, 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 AND 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 MANUAL 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 DISCHARGE TO U2.

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Event Description: 2C 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.: 22-04 Scenario No. NRC-4 Event No.: 4 Page 3 of 4

Event Description: 2C 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 2 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.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 4 Page 4 of 4

Event Description: 2C Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Turbine Building AUO to start 2C Condensate Booster Pump Aux Oil Pump, insert Event 14 and report that the Aux Oil Pump is running.**
	NRC	**End of Event 4. Request that the Driver insert Event 5, 2B Reactor Recirculation Pump Trip.**

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 5 Page 1 of 4

Event Description: 2B Reactor Recirc Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; 2B Recirculation Pump will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to trip 2B Recirc Pump.**
	OATC	When 2B Recirc Pump Trips acknowledges and reports the following alarms to the NUSO: • RECIRC DRIVE 3B TRIPPED, 2-9-4B, Window 7 • RECIRC LOOP B OUT OF SERVICE, 2-9-4B, Window 26 • RECIRC DRIVE 3B PROCESS ALARM, 2-9-4B, Window 28 • RECIRC LOOP B DIFFERENTIAL PRESSURE LOW, 2-9-4B, Window 31 • RECIRC DRIVE 3B DRIVE ALARM, 2-9-4B, Window 32 • 4KV RECIRC PUMP BOARD 2 UNDERVOLTAGE, 2-9-8B, Window 22 • MOTOR TRIPOUT, 2-9-8C, Window 33
	NUSO	Directs the OATC to respond in accordance with 2-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease OPRMs Operable, and the Urgent Load Reduction RCP.
	OATC	2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease OPRMs Operable 4.1 Immediate Actions – None 4.2 Subsequent Actions **CAUTION** 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. [1] N/A

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Event Description: 2B Reactor Recirc Pump Trip

Time	Position	Applicant's Actions or Behavior
	OATC	**NOTES** 1) Step 4.2[2] through Step 4.2[10.3] 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 MELLA+ operating domain. REFERENCE Tech Spec 3.4.1. [2] **IF** a single Recirc Pump tripped, **THEN CLOSE** 2-FCV-068-0079, RECIRC PUMP 2B DISCHARGE VALVE.
	OATC	Closes 2-FCV-068-0079, RECIRC PUMP 2B DISCHARGE VALVE and re-opens it after 5 minutes.
	OATC	[2.1] N/A [2.2] **IF** loadline is greater than 67.0%, **THEN IMMEDIATELY** take actions to INSERT Control Rods to less than 67.0% loadline per 0-TI-464, Reactivity Control Plan and 2-OI-85, Control Rod Drive System.
	OATC	Reduces the Rod Line below 67.0% using the Urgent Load Reduction RCP. See Event 6 on page xx of xx for actions.
	OATC	[2.3] N/A [2.4] **WHEN** the Recirc Pump Discharge Valve has been closed for at least five minutes (to prevent reverse rotation of the pump), **THEN OPEN** Recirc Pump 2B Discharge Valve 2-FCV-068-0079 as necessary to maintain Recirc Loop in thermal equilibrium. [2.5] **RAISE** Core Flow to greater than 45%. **REFERENCE** 2-OI-68, Reactor Recirculation System. [2.6] **MAINTAIN** operating Recirc Pump flow less than 46,600 gpm. REFERENCE 2-OI-68, Reactor Recirculation System.

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Event Description: 2B Reactor Recirc Pump Trip

Time	Position	Applicant's Actions or Behavior
	OATC	[2.7] **WHEN** plant conditions allow, **THEN MAINTAIN** operating Jet Pump Loop Flow greater than 41 x 10^6 lbm/hr. [2.8] **NOTIFY** Reactor Engineer to **PERFORM** the following: [2.8.1] 2-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation [2.8.2] 0-TI-248, Core Flow Determination in Single Loop Operation
	Driver	**If contacted as the Outside/Work Control NUSO, an AUO, Electrical Maintenance, or Reactor Engineer, acknowledge any information or direction given.**
	OATC	[3] N/A [4] **REFERENCE** the following ICS screens to help determine the cause of Recirc Pump trip/Core Flow lowering. • VFDPMPA(VFDPMPB) • VFDAAL(VFDBAL) [5] **CHECK** parameters associated with Recirc Drive and Recirc Pump/Motor 2B on ICS and 2-TR-68-71 to determine cause of trip/Core Flow lowering event. [6] **PERFORM** visual inspection of tripped/affected Reactor Recirc Drive.
	Driver	**If contacted as the Reactor Building AUO to visually inspect 2B Recirc Drive, acknowledge any information or direction given.**
	OATC	[7] **PERFORM** visual inspection of Reactor Recirc Pump Drive relay boards for relay targets. [8] **IF** necessary, **THEN** (Otherwise MARK N/A) **REFERENCE** 2-OI-68, Reactor Recirculation System for Reactor Recirc Pump trips. [9] **INITIATE** actions required to make the necessary repairs. [10] **PERFORM** the following for Single Loop Operation: [10.1] **REFERENCE** 2-OI-68 for guidance on single loop operation. [10.2] **REFERENCE** Tech Specs 3.4.1. [10.3] **WHEN** available, **THEN RETURN** tripped Recirc Pump to service. REFERENCE 2-OI-68, Reactor Recirculation System.

Op Test No.: 22-04	Scenario No. NRC-4	Event No.: 5	Page 4 of 4

Event Description: 2B Reactor Recirc Pump Trip

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.4.1, Recirculating Loops Operating LCO 3.4.1 Two Recirculation Loops with matched flows shall be in operation. <u>OR</u> 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 Condition: A. Requirements of the LCO not met.	
	NUSO	**REQUIRED ACTION:** A.1 – Satisfy the requirements of the LCO.	**COMPLETION TIME:** A.1 – 24 hours
	NRC	**When satisfied with Urgent Load Reduction, end of Event 5. Request that the driver insert Event 7, Low Suppression Pool Water Level.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 6 Page 1 of 3

Event Description: Urgent Load Reduction

Symptoms/Cues: Event is initiated by the crew in response to 3B Reactor Recirculation Pump trip

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 6 is initiated by the crew; no action is required by the driver to insert Event 6.**
	NRC	**During Event 6, once the crew has restored plant operation to the green region of the Power to Flow Map and Technical Specifications (for the Recirc Pump Trip/Shutdown) have been addressed, end of Events 5 and 6.**
	OATC	Urgent Load Reduction/Recirculation Pump Trip Contingency Plan RCP Purpose/Overview of Evolution: Maneuver Reactor to maintain acceptable operating conditions during and following unexpected plant conditions. These Urgent Load Reduction steps do not support a shutdown using the Improved BPWS Control Rod Insertion Process. Plan A, Recirculation Pump Trip Contingency Plan Maneuver Steps: Recirculation Pump Trip A1. If both pumps trip (N/A). For a single Recirculation Pump trip, IMMEDIATELY insert rod groups to lower rod line below 67%. A2. Change Core Flow with the operating Recirculation Pump to exit POWER/FLOW Region 2 and until the operating Recirculation Pump Drive Flow is below 46,600 GPM.

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 6 Page 2 of 3

Event Description: Urgent Load Reduction

Time	Position	Applicant's Actions or Behavior
	OATC	Reactivity Maneuver Instructions Step A1 of A2 Description of step: Recirculation Pump Trip Response **IF** both pumps trip, SCRAM the Reactor per 2-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease OPRMs Operable. For a single Recirculation Pump trip, IMMEDIATELY insert Control Rods to lower Load Line below 67% using Shove Sheets. (Stop after any rod) Step A2 of A2 Description of step: Recirculation Pump Trip Response Change speed of the Operating Recirculation Pump until: • Core Flow is 46 - 49% to exit/remain out of POWER/FLOW Map Region II • Operating Recirculation Pump Drive Flow is 46,600 GPM • Operating Recirculation Pump Speed is ≤ 100%
	OATC	Plan B, Urgent Load Reduction Maneuver Steps: Urgent Load Reduction B1. Lower power by reducing Core Flow until either the desired operating power level is reached or Core Flow is 55 - 60% of rated Core Flow. (If a Recirculation Pump trip is imminent, only lower using the pump to be tripped). B2. Insert Control Rods per Shove Sheets until Load Line is between 48% and 55%. B3. Lower Recirculation Pump Speed to -28% (-480 RPM). B4. Insert Control Rods per Shove Sheets until desired power level is reached or SCRAM the Reactor.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 6 Page 3 of 3

Event Description: Urgent Load Reduction

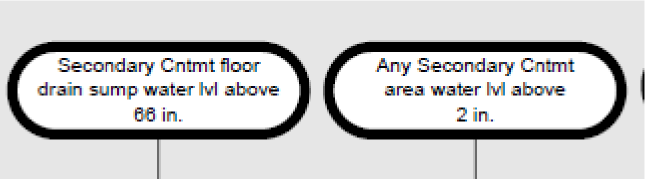
Time	Position	Applicant's Actions or Behavior
	OATC	Reactivity Maneuver Instructions Step B1 of B4 Description of Step: Urgent Load Reduction • Lower power by reducing Core Flow until either the desired operating power level is reached or Core Flow is 55 - 60% of rated Core Flow. (If a Recirculation Pump trip is imminent, only lower using the pump to be tripped) • Insert first four (4) Control Rods from the shove sheet to reduce Load Line Step B2 of B4 Description of Step: Urgent Load Reduction Insert Control Rods per shove sheets until Load Line is between 48% and 55% Step B4 of B4 Description of Step: Urgent Load Reduction Lower Recirculation Pump Speed to ~28% (~480 RPM or ~39% Core Flow). (Control Rods may be adjusted during this step) Step B4 of B4 Description of Step: Urgent Load Reduction Insert Control Rods per shove sheets until desired power level is reached or SCRAM the Reactor (record conditions prior to the SCRAM)
	NRC	**End of Event 6.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 7 Page 1 of 10

Event Description: Low Suppression Pool Water Level

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; Suppression Pool Water Level will lower due to an unisolable leak

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to cause an unisolable Suppression Pool water leak.**
	BOP	Acknowledges and reports Area Flood Level alarms to the NUSO.
	Driver	**When contacted by the Control Room to investigate the Area Flood Level alarms, acknowledge the direction.** **As the Rad Waste Operator report that Sump Level Alarms have been received and both Sump Pumps are running.**
	NUSO	Enters 2-EOI-3, Secondary Containment Control, on Area Water Level alarms. 
	Driver	**Three (3) minutes after being contacted to investigate the flood level alarms, contact the Control Room and inform the crew that there is a leak from the Suppression Chamber. If asked if the leak can be isolated, inform the crew that you are investigating. Wait two (2) minutes and report to the crew that the leak is from the Loop II RHR Pump common suction and is un-isolable.**

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 7 Page 2 of 10

Event Description: Low Suppression Pool Water Level

Time	**Position**	**Applicant's Actions or Behavior**
	NUSO	2-EOI-3, Secondary Containment Control SC-1
	NUSO	SC/L-1 **WHEN** ANY Floor Drain Sump or Area Water Level exceeds its Max Normal level (Table SC-3) Continues TO SC/L-2

SC-1:

IF	**THEN**
Rx 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**
Rx Zone ventilation is isolated AND Rx Zone ventilation exhaust radiation level is below 72 mR/hr	**NO ACTION REQUIRED**
Refuel Zone ventilation is isolated AND Refuel Zone exhaust radiation level is below 72 mR/hr	**NO ACTION REQUIRED**
SBGT CANNOT restore and maintain Secondary Containment DP below 0 inches water AND Radioactivity release from Secondary Containment hinders operation of systems required for damage control or systems required to be operated by EOIs	**NO ACTION REQUIRED**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 7 Page 3 of 10

Event Description: Low Suppression Pool Water Level

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>

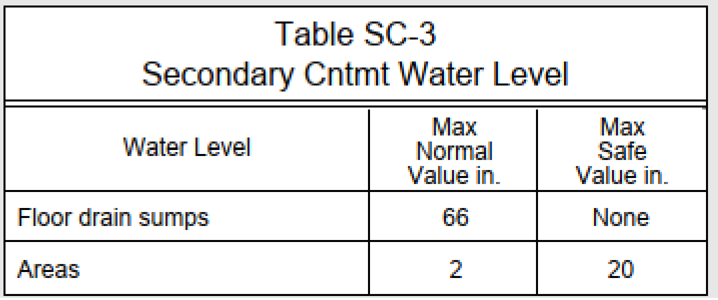
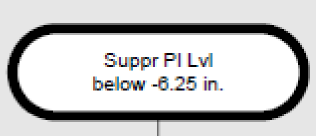
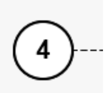



Table SC-3
Secondary Cntmt Water Level
<table>
<tr><th>Water Level</th><th>Max Normal Value in.</th><th>Max Safe Value in.</th></tr>
<tr><td>Floor drain sumps</td><td>66</td><td>None</td></tr>
<tr><td>Areas</td><td>2</td><td>20</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>SC/L-2
OPERATE available Sump Pumps to restore and maintain levels below Max Normal levels (Table SC-3).</td></tr>
<tr><td></td><td>Driver</td><td>When contacted as Rad Waste to operate all available Sump Pumps, acknowledge the direction.</td></tr>
<tr><td></td><td>NUSO</td><td>Enters 2-EOI-2, Primary Containment Control, when Suppression Pool Water Level reaches (-) 6.25".


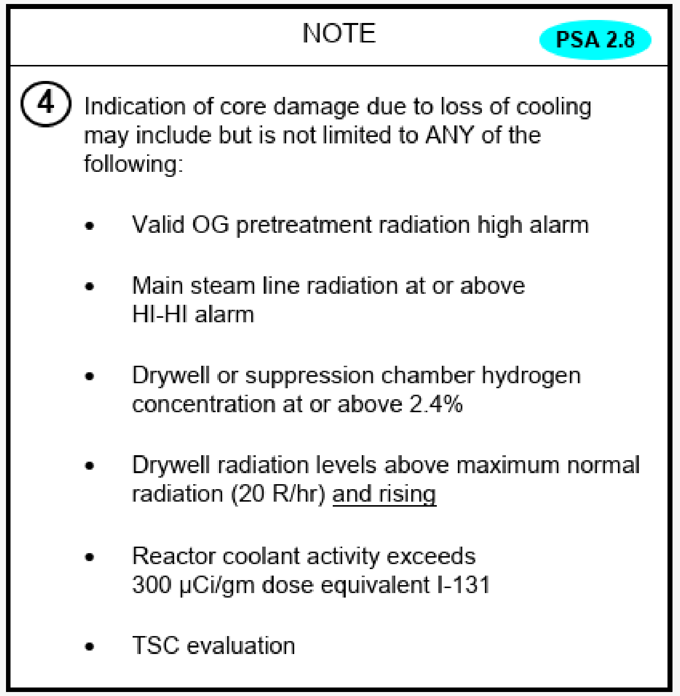
PCC-1


<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>It is determined that core damage is occurring due to loss of cooling</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 7 Page 4 of 10

Event Description: Low Suppression Pool Water Level

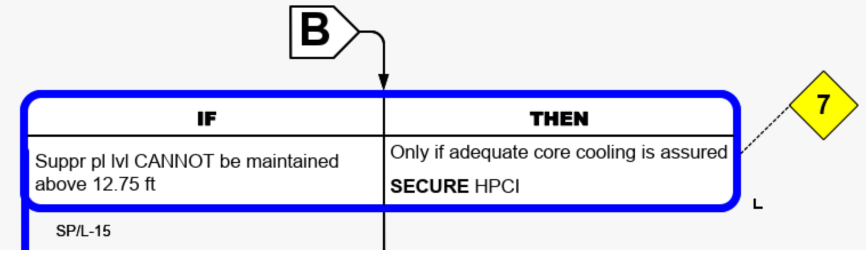
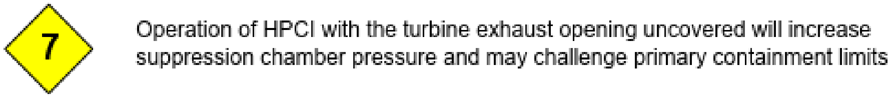
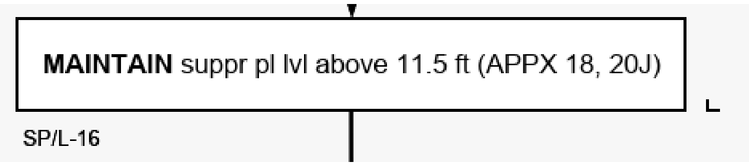
<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>

NOTE PSA 2.8
④ Indication of core damage due to loss of cooling may include but is not limited to ANY of the following:

Valid OG pretreatment radiation high alarm
Main steam line radiation at or above HI-HI alarm
Drywell or suppression chamber hydrogen concentration at or above 2.4%
Drywell radiation levels above maximum normal radiation (20 R/hr) <u>and rising</u>
Reactor coolant activity exceeds 300 µCi/gm dose equivalent I-131
TSC evaluation

SP/L-1
<table>
<tr><td colspan="2">MONITOR and CONTROL Suppression Pool Water Level (-) 6 in. to (-) 1 in. (APPX 18)</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Suppression Pool Water Level CANNOT be maintained below -1 in.</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Pool Water Level CANNOT be maintained above -6 in.</td><td>ACTION REQUIRED
B</td></tr>
</table>
</td></tr>
</table>

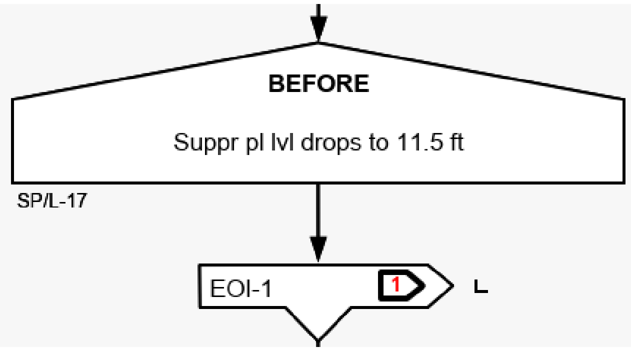
Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 7 Page 5 of 10

Event Description: Low Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	CREW	**Critical Task:** **With Adequate Core Cooling available, lock out HPCI when Suppression Pool Level cannot be maintained above 12.75 feet to prevent Primary Containment damage.** **Critical Task Failure Criteria:** **HPCI is operated below a Suppression Pool Water Level of 12.75 feet, thus pressurizing the Suppression Chamber from the HPCI exhaust.**
	NUSO	  Operation of HPCI with the turbine exhaust opening uncovered will increase suppression chamber pressure and may challenge primary containment limits 
	NUSO	Directs the BOP to maintain Suppression Pool Water Level using 2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup.
	BOP	2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup [1] **IF** Suppression Pool Water makeup is required, **THEN CONTINUE** in this procedure at Step 1.0[5]. [2] N/A [3] N/A [4] N/A [5] N/A

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 7 Page 6 of 10

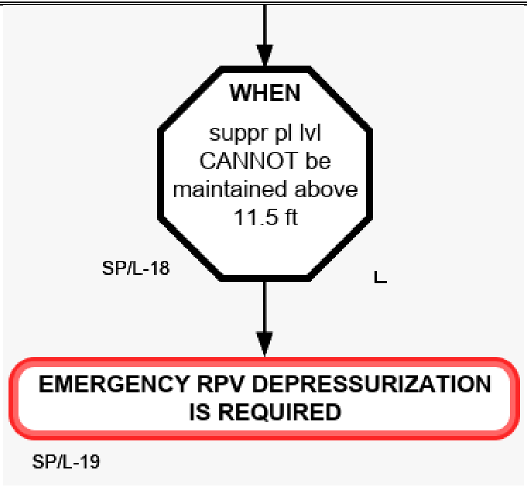
Event Description: Low Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	BOP	[6] **IF** Directed by NUSO to add water to Suppression Pool, **THEN** MAKEUP water to Suppression Pool as follows: [6.1] **VERIFY OPEN** 2-FCV-73-40, HPCI CST SUCTION VALVE. [6.2] **OPEN** 2-FCV-73-30, HPCI PUMP MINIMUM FLOW VALVE.
	NRC	**EXAMINER NOTE: 2-FCV-73-30, HPCI PUMP MINIMUM FLOW VALVE will NOT be open, although it inidicates open to facilitate lowering Suppression Pool Water Level to the Emergency Depressuization trigger value.**
	BOP	[6.3] N/A [6.4] N/A
	CREW	**Critical Task:** **Manually insert a Reactor SCRAM before Suppression Pool Level reaches 11.5 feet, to reduce the amount of energy in the vessel on a lowering Suppression Pool Water Level.** **Critical Task Failure Criteria:** **The operating crew allows Suppression Pool Water Level to lower past 11.5 feet without inserting a manual SCRAM signal.**
	NUSO	
	NRC	**Event 8, Inadvertent Main Steam Isolation Valve (MSIV) Closure, is automatically entered on the MODE Switch being placed in SHUTDOWN following the Reactto SCRAM. No Action is requirred by the driver to insert Event 8. See page xx of xx for Event 8 actions.**
	NUSO	Enters 2-EOI-1, RPV Control MODES 1-3.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 7 Page 7 of 10

Event Description: Low Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	NUSO	
	NUSO	Determines that Emergency Depressurization is required.
	CREW	**Critical Task:** **Emergency Depressurization is performed when Suppression Pool Water Level cannot be maintained above 11.5 feet to minimize the adverse effects on Suppression Pool heat capacity.** **Critical Task Failure Criteria:** **The crew does not perform Emergency Depressurization without delay and in a controlled manner when Suppression Pool Water Level cannot be maintained above 11.5 feet.**
	NUSO	2-EOI-1, RPV Control MODES 1-3 

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 7 Page 8 of 10

Event Description: Low Suppression Pool Water Level

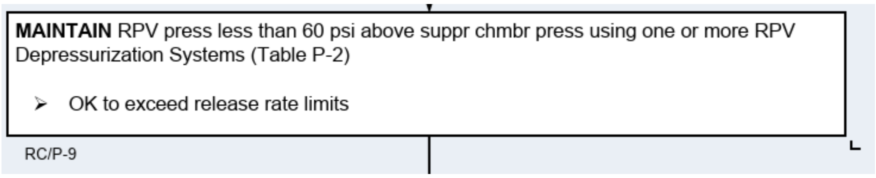
<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
RC/P-6
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that available Injection Subsytems (Table L-3) alone CANNOT assure Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that containment water level will rise above 44 ft</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-7
<table>
<tr><td>IF Drywell Pressure is above 2.45 psig

THEN PREVENT injection from ONLY those CS and LPCI Pumps NOT required to assure Adequate Core Cooling (APPX 4)</td></tr>
</table>
RC/P-8
<table>
<tr><td colspan="2">EMERGENCY DEPRESSURIZE the RPV
IF Suppression Pool Level is above 5.5 ft THEN OPEN 6 MSRVs (ADS Valves preferred)
➢ OK to exceed 100°F/hr cooldown rate</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>
<tr><td>Less than 4 MSRVs can be opened
AND
RPV Pressure is 60 psi or more above Suppression Chamber Pressure</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 7 Page 9 of 10

Event Description: Low Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	NUSO	 **MAINTAIN** RPV press less than 60 psi above suppr chmbr press using one or more RPV Depressurization Systems (Table P-2) ➢ OK to exceed release rate limits RC/P-9
	NUSO	Directs the BOP to open 6 SRVs (ADS preferred). Provides direction to the crew concerning the method to maintain Reactor Water Level following Emergency Depressurization (2-EOI-Appendix-6A, Injection Subsystems Lineup Condensate).
	OATC	2-EOI-Appendix-6A, Injection Subsystems Lineup Condensate **NOTES** 1) All valves are located on Panel 2-9-6 unless otherwise noted within the step. 2) The optimum pump combination for Condensate flood is all available Condensate and Condensate Booster Pumps in service with the RFP Discharge Valves open. [1] **ENSURE CLOSED** the following Feedwater Heater Return Valves: • 2-FCV-3-71, HP HTR 2A1 LONG CYCLE TO CONDENSER • 2-FCV-3-72, HP HTR 2B1 LONG CYCLE TO CONDENSER • 2-FCV-3-73, HP HTR 2C1 LONG CYCLE TO CONDENSER [2] **ENSURE CLOSED** the following RFP Discharge Valves (N/A if Condensate Flood up is anticipated per the FSSs): • 2-FCV-3-19, RFP 2A DISCHARGE VALVE • 2-FCV-3-12, RFP 2B DISCHARGE VALVE • 2-FCV-3-5, RFP 2C DISCHARGE VALVE [3] **ENSURE OPEN** the following Drain Cooler Inlet Valves: • 2-FCV-3-72, DRAIN COOLER 2A5 CNDS INLET ISOLATION VALVE • 2-FCV-3-84, DRAIN COOLER 2B5 CNDS INLET ISOLATION VALVE • 2-FCV-3-96, DRAIN COOLER 2C5 CNDS INLET ISOLATION VALVE

Op Test No.: 22-04	Scenario No. NRC-4	Event No.: 7	Page 10 of 10

Event Description: Low Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	OATC	[4] **ENSURE OPEN** the following Heater Outlet Valves: • 2-FCV-2-124, LP HEATER 2A3 CNDS OUTLET ISOLATION VALVE • 2-FCV-2-125, LP HEATER 2B3 CNDS OUTLET ISOLATION VALVE • 2-FCV-2-126, LP HEATER 2C3 CNDS OUTLET ISOLATION VALVE [5] **ENSURE OPEN** the following Heater Isolation Valves: • 2-FCV-3-38, HP HTR 2A2 FEEDWATER INLET ISOLATION VALVE • 2-FCV-3-31, HP HTR 2B2 FEEDWATER INLET ISOLATION VALVE • 2-FCV-3-24, HP HTR 2C2 FEEDWATER INLET ISOLATION VALVE • 2-FCV-3-75, HP HTR 2A1 FEEDWATER OUTLET ISOLATION VALVE • 2-FCV-3-76, HP HTR 2B1 FEEDWATER OUTLET ISOLATION VALVE • 2-FCV-3-77, HP HTR 2C1 FEEDWATER OUTLET ISOLATION VALVE [6] **ENSURE OPEN** the following RFP Suction Valves: • 2-FCV-2-83, RFP 2A SUCTION VALVE • 2-FCV-2-95, RFP 2B SUCTION VALVE • 2-FCV-2-108, RFP 2C SUCTION VALVE [7] **ENSURE** all available Condensate Pumps running. [8] **ENSURE** all available Condensate Booster Pumps running. [9] **ADJUST** 2-LIC-3-53, RFW START-UP LEVEL CONTROL, on Panel 2-9-5, to control injection. [10] **CHECK** RFW Flow to RPV. **End of 2-EOI-APPENDIX-6A**
	NRC	**End of Event 7, end of Scenario. 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.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 8 Page 1 of 2

Event Description: Inadvertent Main Steam Isolation Valve (MSIV) Closure

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; all MSIVs will inadvertently close when the Reactor MODE Switch is placed in SHUTDOWN

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>NRC</td><td>Event 8 is automatically inserted on simulator setup. No action is required by the driver to insert Event 8.</td></tr>
<tr><td></td><td>CREW</td><td>Determines that MSIVs have closed, causing Reactor Pressure to rise.</td></tr>
<tr><td></td><td>NUSO</td><td>
2-EOI-1, RPV Control MODES 1-3

RC/P-2
<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>Drywell Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
</table>

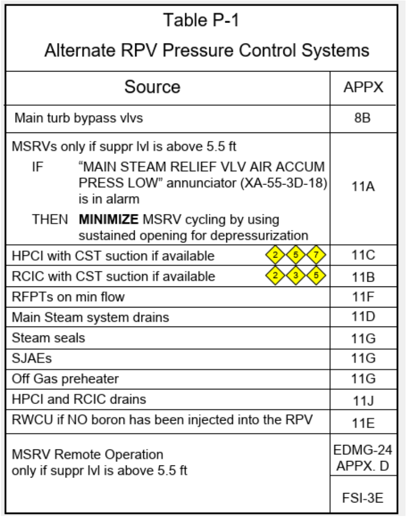



Table P-1
Alternate RPV Pressure Control Systems
<table>
<tr><th>Source</th><th>APPX</th></tr>
<tr><td>Main turb bypass vlvs</td><td>8B</td></tr>
<tr><td>MSRVs only if suppr lvl is above 5.5 ft
IF “MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW” annunciator (XA-55-3D-18) is in alarm
THEN MINIMIZE MSRV cycling by using sustained opening for depressurization</td><td>11A</td></tr>
<tr><td>HPCI with CST suction if available ◇2 ◇5 ◇7</td><td>11C</td></tr>
<tr><td>RCIC with CST suction if available ◇2 ◇3 ◇5</td><td>11B</td></tr>
<tr><td>RFPTs on min flow</td><td>11F</td></tr>
<tr><td>Main Steam system drains</td><td>11D</td></tr>
<tr><td>Steam seals</td><td>11G</td></tr>
<tr><td>SJAEs</td><td>11G</td></tr>
<tr><td>Off Gas preheater</td><td>11G</td></tr>
<tr><td>HPCI and RCIC drains</td><td>11J</td></tr>
<tr><td>RWCU if NO boron has been injected into the RPV</td><td>11E</td></tr>
<tr><td>MSRV Remote Operation
only if suppr lvl is above 5.5 ft</td><td>EDMG-24 APPX. D
FSI-3E</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 22-04 Scenario No. NRC-4 Event No.: 8 Page 2 of 2

Event Description: Inadvertent Main Steam Isolation Valve (MSIV) Closure

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs the BOP to stabilize Reactor Pressure using 2-EOI-Appendix-11A, Alternate RPV Pressure Control Systems MSRVs.
	BOP	[3] **OPEN** MSRVs using the following sequence to control RPV pressure as directed by the NUSO: 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 2-EOI-APPENDIX 11A**
	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 (TAF, (-) 162 inches) using low pressure systems, end of Scenario.**

UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 1	**DAYS ON LINE** **208**	**Total Drywell Leakage (gpm)** **1.55**	**Protected Equipment** Condensate Pump 2A, 2B Condensate Booster Pump 2A, 2B
Rx Power **74%**	PRA (EOOS) -GREEN 500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (gpm)** **0.11**	RFPT 2A, 2B
MWe **912**	**Last breaker closure** **4/10/19 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

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Alternate Reactor and Refuel Zone Fans in accordance with 2-OI-30A, Refueling Zone Ventilation System, and 2-OI-30B, Reactor Zone Ventilation System.

OPERATOR WORK AROUNDS OWAs – U0-1/U2-0 Burdens - U0 – 0/U2-0 Operator Challenges – U0 – 12/U2 - 6

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Facility: BFN Scenario Number: NRC-5
Scenario Source: NEW Op-Test Number: 22-04
Examiners: ____________ Operators: NUSO: ____________
OATC: ____________
BOP: ____________

Initial Conditions: A Reactor Startup is in progress. ~3 % Reactor Power.

Turnover: Continue the Reactor Startup. Contact the Ops Superintendent prior to placing the MODE SWITCH in RUN.

Critical Tasks:

1. With a Loss of Offsite Power and a Station Blackout caused by the failure of two Emergency Diesel Generators (EDGs) to automatically start and tie on to their respective 4KV Shutdown Boards, the crew restores power to one 480V Shutdown Board to exit Station Blackout within 20 minutes of the loss of power.
2. With a loss of EECW Pumps due to a Loss of Offsite Power, the crew restores EECW Flow to the EDGs within 8 minutes.

Event Number	**Malfunction Number**	**Event Type***	**Event Description**
1.	N/A	N-BOP MC-BOP N-NUSO	Warm 2B Reactor Feedwater Pump (RFPT)
2.	N/A	R-OATC R-NUSO	Control Rod Withdrawal
3.	RD12	C-OATC C-NUSO	Control Rod Difficult to Withdraw
4.	HS-75-5A	C-BOP MC-BOP TS-NUSO	Inadvertent Core Spray Pump Start
5.	SCHEDULE STACK	C-BOP MC-BOP C-NUSO	2A Stack Dilution Fan Failure, Standby Fan Fails to Automatically Start
6.	TH31A	I-OATC TS-SRO	Reactor High Pressure A1 Channel Fails High/Half SCRAM (2-PIS-3-22AA)
7.	ED01	M-All	Loss of Offsite Power
8.	DG03B	C-BOP MC-BOP	‘B’ EDG Fails to Tie Automatically
9.	SW07A SW07B SW07C SW07D	C-BOP MC-BOP C-NUSO	Emergency Equipment Cooling Water (EECW) Pumps Fail to Auto Start

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

Events

1. The Balance of Plant (BOP) will warm 2B Reactor Feedwater Pump (RFPT) in accordance with 2-OI-3, Reactor Feedwater System.

2. The crew will continue the Reactor Startup by withdrawing Control Rods in accordance with 2-GOI-100-1A, Unit Startup and Power Operation, and 2-OI-85, Control Rod Drive System.

3. Control Rod 58-31 will be difficult to withdraw during the Reactor Startup, requiring the crew to take action in accordance with 2-OI-85, Control Rod Drive System.

4. 2A Core Spray Pump will inadvertently start, requiring the crew to shut down 2A Core Spray Pump. When contacted, the Assistant Unit Operator (AUO) will report there is no visible oil level on 2A Core Spray Pump. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.

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. Reactor High Pressure A1 Channel Fails High/Half SCRAM (2-PIS-3-22AA), resulting in a Reactor Half SCRAM. The crew will respond in accordance with Alarm Response Procedures. The NUSO will address Technical Specifications.

7. A Loss of Offsite Power will occur, and coupled with various Diesel Generator failures will result in a Station Blackout. The crew will respond in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV)/Station Blackout until the power is restored to one of the 480V Shutdown Boards.

8. Following the Loss of Offsite Power, ‘B’ Emergency Diesel Generator (EDG) will fail to automatically tie onto the ‘B’ 4KV Shutdown Board, requiring the crew to manually close the EDG Output Breaker.

9. Emergency Equipment Cooling Water (EECW) Pumps will fail to auto start when the EDGs restore power to the 4160V Shutdown Boards. The crew will restart the EECW pumps within 8 minutes as required by 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV)/Station Blackout.

The Scenario ends when the crew has restored power to one 480V Shutdown Board, has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using RCIC or HPCI, and has started EECW Pumps.

Critical Tasks 2

1. **With a Loss of Offsite Power and a Station Blackout caused by the failure of two Emergency Diesel Generators to automatically start and tie on to their respective 4KV Shutdown Boards, the crew restores power to one 480V Shutdown Board to exit Station Blackout within 20 minutes of the loss of power.**

   a. **Safety Significance**
   Exit Station Blackout.
   Restores power to components necessary for one loop of Low Pressure ECCS systems.

   b. **Cues**
   Procedural compliance.
   Loss of both 480V Shutdown Boards for Unit 2.

   3. **Measured by:**
   Observation - With a Station Blackout in progress on Unit 2, one 480V Shutdown Board is transferred to its alternate power supply within 20 minutes.

   4. **Feedback**
   Power is restored to one 480V Shutdown Board.
   Power available to Low Pressure ECCS system valves.

   5. **Critical Task Failure Criteria:**
   The operating crew fails to restore power to one 480V Shutdown Board within 20 minutes of the Station Blackout on Unit 2.

2. **With a loss of EECW Pumps due to a Loss of Offsite Power, the crew restores EECW Flow to the EDGs within 8 minutes.**

   1. **Safety Significance**
   Diesel Generator operation without EECW flow will cause EDG failure.

   2. **Cues**
   Procedural compliance.
   EECW Header Flow.
   EECW Pump Run Indications.

   3. **Measured by:**
   Observation – The crew verifies EECW Pumps have restarted as required when power is restored to the 4KV Shutdown Busses.

   4. **Feedback**
   EECW Header Flow.
   EECW Pump Run indications.

   5. **Critical Task Failure Criteria:**
   The operating crew fails to restore EECW Flow to the EDGs within 8 minutes following EDG start upon a Loss of Offsite Power.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 1 Page 1 of 4

Event Description: Warm 2B Reactor Feedwater Pump (RFPT)

Symptoms/Cues: Crew is cued by the turnover sheet or by the Simulator Operator as requested by the Chief Examiner

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, Warm 2B Reactor Feedwater Pump (RFPT) after assuming the shift, request that the Driver contact the Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to warm 2B RFPT.** **NOTE: The crew may elect to hold a re-focus reactivity brief.**
	Driver	**If requested by the Chief Examiner, contact the NUSO as the Shift Manager and direct the crew to warm 2B RFPT.** **If contacted by the crew as the Reactor Building Assistant Unit Operator (AUO) acknowledge any direction given.**
	NUSO	Directs the Operator at the Controls (OATC) to warm 2B RFPT in accordance with 2-OI-3, Reactor Feedwater System, Section 5.6.
	BOP	2-OI-3, Reactor Feedwater System Section 5.6, Warming the Second and Third RFP/RFPT [3.3] **ENSURE OPEN** the following valves to drain Condensate and warm RFPT steam supply lines: [3.3.2] RFPT 2B • 2-FCV-6-127, RFPT 2B HP STOP VALVE ABOVE SEAT DRAIN using handswitch 2-HS-6-127A • 2-FCV-6-128, RFPT 2B HP STOP VALVE BELOW SEAT DRAIN using handswitch 2-HS-6-128A • 2-FCV-6-129, RFPT 2B FIRST STAGE DRAIN VALVE using handswitch 2-HS-6-129A • 2-FCV-006-0155, RFPT 2B HP STEAM SHUTOFF ABOVE SEAT DRAIN using handswitch 2-HS-006-0155B (local control)

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 1 Page 2 of 4

Event Description: Warm 2B Reactor Feedwater Pump (RFPT)

Time	Position	Applicant's Actions or Behavior
	BOP	• RFPT 2B LP STOP VALVE ABOVE SEAT DRAIN, using 2-HS-6-125A using handswitch • 2-FCV-6-126, RFPT 2B LP STOP VALVE BELOW SEAT DRAIN using handswitch 2-HS-6-126A • 2-FCV-006-0156, RFPT 2B LP STEAM SHUTOFF ABOVE SEAT DRAIN using handswitch 2-HS-006-0156B (local control) [3.4] **ENSURE** the following on Panel 2-9-5 (as applicable): • 2-HS-46-9A, RFPT 2B SPEED CONT RAISE/LOWER switch is depressed in MANUAL GOVERNOR position with amber light at switch illuminated [3.5] **ENSURE** one of the following: • RFPT being started has been on Turning Gear or rolling on minimum flow for at least one hour. (Otherwise N/A) • With Unit SRO permission, RFPT is rolling on turning gear or minimum flow and no abnormal rubbing noises or vibration is observed. (Otherwise N/A) [3.6] **REFER TO** 2-OI-47C, Seal Steam System and **ENSURE** Seal Steam being supplied to RFPT being placed in service.
	BOP	[3.7] **ENSURE** applicable switch in NORMAL and amber light extinguished (Panel 2-9-6): • 2-HS-2-122, RFP 2B NPSH TRIP BYPASS **NOTE** When returning Feed Pump to service with Reactor at rated pressure, Step 5.7[6] will provide the instructions to reopen Feed Pump Discharge Valves.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 1 Page 3 of 4

Event Description: Warm 2B Reactor Feedwater Pump (RFPT)

Time	Position	Applicant's Actions or Behavior
	BOP	**CAUTION** [NER/C] Failure to have idle RFP Discharge Valves open during startup after first RFP has been placed in service may result in over-pressurization of piping between RFP Discharge Valve and Discharge Check Valve due to thermal heating. [INPO SER 92-002] [3.8] **ENSURE OPEN** RFP Discharge Valve for pump being placed in service: • 2-FCV-3-12, RFP 2B DISCHARGE VALVE [3.9] **OPEN** applicable valve for pump being started (Panel 2-9-6): • 2-FCV-2-95, RFP 2B SUCTION VALVE **NOTE** Blue light at turbine trip RESET pushbutton should now be illuminated. This indicates all trip signals are reset with control valves closed allowing trip to be reset. [3.10] **CHECK** blue light at turbine trip RESET pushbutton illuminated. [3.11] **PERFORM** RFPT Trip Reset: [3.11.1] N/A [3.11.2] **DEPRESS** RFPT 2B TRIP RESET, 2-HS-3-150A, and **CHECK** the following: • Blue light extinguishes • HP Stop Valves open • LP Stop Valves open
	NRC	**If the candidate asks about the status of steps [3.12] through [3.14], state 'Continue on at step [3.15]'.**

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 1 Page 4 of 4

Event Description: Warm 2B Reactor Feedwater Pump (RFPT)

Time	Position	Applicant's Actions or Behavior
	BOP	[3.15] **PLACE** control switches for the following in AUTO, and **ENSURE OPEN** RFP Minimum Flow Valve for pump being placed in service: • 2-FCV-3-13, RFP 2B MIN FLOW VALVE **NOTE** RFPT speed is raised to approximately 1100 rpm to obtain positive indication of discharge pressure. **CAUTIONS** 1) Do not allow RFPT speed to exceed 1100 rpm until RFPT manual trip has been satisfactorily tested. 2) Do not raise RFP Discharge Pressure to greater than Reactor Pressure to prevent injection to the vessel during RFPT trip testing. [3.16] **START** RFPT from Panel 2-9-6 as follows: • **PLACE** 2-HS-46-138A, RFPT 2B START/LOCAL ENABLE, in START, and **OBSERVE** RFPT accelerates to approximately 600 rpm on 2-SI-46-9A, RFPT 2B SPEED [3.17] **PERFORM** the following locally at RFPT: A. **CHECK** no abnormal rubbing noises or vibration is observed. B. **IF** any abnormal rubbing noises or vibration is observed, **THEN REMOVE** RFPT from operation until condition is corrected. [3.18] **CHECK** RFPT Speed Control in MANUAL GOVERNOR. [3.19] **ADJUST** RFPT Speed Control position. • **USE** 2-HS-46-9A, RFPT 2B SPEED CONT RAISE/LOWER switch as necessary to RAISE RFPT speed to approximately 1100 rpm [3.20] **PLACE** RFPT Turning Gear Motor, in AUTO as follows: • 2-HS-3-127A, RFPT 2B TURNING GEAR MOTOR
	NRC	**The crew has been pre-briefed to stop warming 2B RFPT at procedure step [3.20].**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 2 Page 1 of 7

Event Description: Control Rod Withdrawal

Symptoms/Cues: Crew is cued by the turnover sheet or by the Simulator Operator as requested by the Chief Examiner

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 Nuclear Unit Senior Operator (NUSO) as the Shift Manager and direct the crew to continue the Reactor Startup.**
	NRC	**Event 3, Control Rod Difficult to Withdraw, is automatically entered on simulator setup and will occur during Event 2 on Control Rod 58-31. No action is required by the driver to insert Event 3.**
	SRO	(May 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 Section 5.4, Withdrawal of Control Rods while in Mode 2 **NOTE** 6% to 7% Rated Thermal Power (RTP) is the target power level to prevent rod blocks below 5% RTP or above 8% 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.: 22-04 Scenario No. NRC-5 Event No.: 2 Page 2 of 7

Event Description: Control Rod Withdrawal

Time	**Position**	**Applicant's Actions or Behavior**
	OATC	2-OI-85, Control Rod Drive System Section 6.6, Control Rod Withdrawal Section 6.6.1 Initial Conditions prior to Withdrawing Control Rods **NOTES** 1) If the Control Rod is uncoupled, the four rod display digital read-out and the full Core display digital readout and background light will extinguish for the uncoupled rod and the annunciator CONTROL ROD OVERTRAVEL (2-XA-55-5A, Window 14) will seal in. 2) Coupling integrity is satisfied if CRD Notch Override Switch is used and rod is withdrawn to Position 48. 3) The following steps are performed from Panel 2-9-5 unless noted otherwise. **CAUTIONS** 1) Positioning Control Rods should be done with the utmost diligence and care. Notch Withdrawing Control Rods provides the most deliberate controlled method of withdrawing Control Rods. 2) Never pull Control Rods except in a deliberate, carefully controlled manner, while closely monitoring the Reactor's response. [1] **REVIEW** Precautions and Limitations in Section 3.7 and Section 3.8. [2] **CHECK** the following prior to Control Rod movement: • 2-HS-85-46, CRD POWER in ON • Rod Worth Minimizer is operable and LATCHED into the correct ROD GROUP when Rod Worth Minimizer is enforcing

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 2 Page 3 of 7

Event Description: Control Rod Withdrawal

Time	**Position**	**Applicant's Actions or Behavior**
	OATC	**NOTES** 1) Section 6.6.2 is applicable for all Control Rod withdrawals and addressed as required during or following any Control Rod withdrawal. 2) If rod insertion to position 00 is required and Core Thermal Power is less than or equal to 10%, entry into LCO 3.1.6 may be required. [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. [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 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 Rod Block Monitor (RBM) Rod Block as follows: • **STOP** Control Rod withdrawal (if possible) prior to reaching any RBM Rod Block using the RBM displays on Panel 2-9-5 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 Supervisor's discretion to "REINITIALIZE" the RBM: [6.1] 2-HS-85-46, **PLACE** CRD POWER in the OFF position to deselect the Control Rod. [6.2] 2-HS-85-46, **PLACE** CRD POWER in the ON position.

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 2	Page 4 of 7

Event Description: Control Rod Withdrawal

Time	**Position**	**Applicant's Actions or Behavior**
	OATC	[6.3] **IF** desired, **THEN CONTINUE** to withdrawal Control Rods and **PERFORM** applicable section for Control Rod withdrawal.
	OATC	For each Control Rod to be moved using Control Rod Notch Withdrawal: Section 6.6.3 Control Rod Notch Withdrawal [1] **SELECT** the desired Control Rod by depressing the appropriate 2-XS-85-40, CRD ROD SELECT pushbutton. [2] **ENSURE** CRD DRIVE WATER HEADER DP is between 250 -270 psid on 2-PDI-85-17A by throttling 2-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE as necessary. [3] **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 [4] **CHECK** Rod Worth Minimizer is operable and LATCHED into the correct ROD GROUP when the Rod Worth Minimizer is enforcing. [5] **PLACE** 2-HS-85-48, CRD CONTROL SWITCH, in ROD OUT NOTCH, and **RELEASE**. [6] **OBSERVE** the Control Rod settles into the desired position and the ROD SETTLE light extinguishes. [7] **OBSERVE** control rod settles into desired position **AND** ROD SETTLE light extinguishes. [8] N/A

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 2 Page 5 of 7

Event Description: Control Rod Withdrawal

Time	Position	Applicant's Actions or Behavior
	OATC	For each Control Rod to be moved using Continuous Rod Withdrawal: Section 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 2-SR-3.1.3.5A, Control Rod Coupling Integrity Check. 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** desired Control Rod by depressing appropriate 2-XS-85-40, CRD ROD SELECT. [2] **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

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 2 Page 6 of 7

Event Description: Control Rod Withdrawal

Time	Position	Applicant's Actions or Behavior
	OATC	[3] **ENSURE** Rod Worth Minimizer operable and LATCHED into correct ROD GROUP when the Rod Worth Minimizer is enforcing. [4] **CHECK** Control Rod is being withdrawn to a position greater than three notches. [5] **IF** withdrawing the Control Rod to a position other than "48", **THEN PERFORM** the following: (Otherwise N/A) [5.1] **PLACE AND HOLD** 2-HS-85-47, CRD NOTCH OVERRIDE, in NOTCH OVERRRIDE. [5.2] **PLACE AND HOLD** 2-HS-85-48, CRD CONTROL SWITCH, in ROD OUT NOTCH. [5.3] **WHEN** Control Rod reaches two notches prior to the intended notch, **THEN RELEASE** 2-HS-85-47, CRD NOTCH OVERRIDE, and 2-HS-85-48, CRD CONTROL SWITCH. [5.4] **IF** Control Rod settles at notch before intended notch, **THEN PLACE 2**-HS-85-48, CRD CONTROL SWITCH in ROD OUT NOTCH and **RELEASE**. [5.5] **WHEN** Control Rod settles into the intended notch, **THEN CHECK** the following. • Four Rod display digital readout and the full Core display digital readout and background light remain illuminated. • CONTROL ROD OVERTRAVEL annunciator, 2-XA-55-5A, Window 14, does not alarm. [5.6] **CHECK** the Control Rod settles at intended position and ROD SETTLE light extinguishes. [6] N/A [7] N/A

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 2 Page 7 of 7

Event Description: Control Rod Withdrawal

Time	**Position**	**Applicant's Actions or Behavior**
	OATC	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	**When satisfied with the Reactivity Manipulatiion, end of Event 2. After Event 3 has also been completed, request that the driver insert Event 4, Inadvertent Core Spray Pump Start.**

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 3 Page 1 of 3

Event Description: Control Rod Difficult to Withdraw

Symptoms/Cues: Event is automatically inserted when Control Rod 58-31 is withdrawn.

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 3 will occur automatically during Event 2, Control Rod Withdrawal. No action is required by the driver to insert Event 3.**
	OATC	Determines that Control Rod 58-31 is difficult to withdraw and informs the NUSO.
	NUSO	Directs the OATC to respond in accordance with 2-OI-85, Control Rod Drive System, Section 8.16, Control Rod Difficult to Withdraw.
	OATC	2-OI-85, Control Rod Drive System Section 8.16, Control Rod Difficult to Withdraw [1] **CHECK** the Control Rod will **NOT** notch out. Refer to Section 6.6. [2] **REVIEW** all Precautions and Limitations in Section 3.0. **CAUTION** Never pull Control Rods except in a deliberate, carefully controlled manner, while closely monitoring the Reactor's response. [3] **IF** RWM is enforcing, **THEN ENSURE** RWM is operable and LATCHED in to the correct ROD GROUP. **NOTES** 1) Steps 8.16[4] through 8.16[6] should be used when the Control Rod is at Position 00 while Step 8.16[7] should be used when the Control Rod is at or between Positions 02 and 46. 2) Double clutching of a Control Rod at Position 00 will place the rod at the "overtravel in" stop, independent of the RMCS timer, allowing maximum available time to establish over-piston pressure required to maintain the collet open and prevent the collet fingers from engaging the 00 notch. 3) Step 8.16[4] may be repeated as necessary until it is determined that this method will not free the Control Rod.

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 3	Page 2 of 3

Event Description: Control Rod Difficult to Withdraw

Time	Position	Applicant's Actions or Behavior
	NRC	**EXAMINER NOTE: Step [4] below will NOT be successful. Drive Water Pressure must be raised before the Control Rod can be withdrawn.**
	OATC	[4] **IF** the Control Rod problem is **NOT** believed to be air in the hydraulic system, **THEN PERFORM** the following to double clutch the Control Rod at Position 00: [4.1] **PLACE AND HOLD** 2-HS-85-47, CRD NOTCH OVERRIDE, in EMERG ROD IN, for several seconds. [4.2] **CHECK** the Control Rod full in indication (double green dashes) on the Full Core Display for the associated Control Rod. [4.3] **SIMULTANEOUSLY PLACE** 2-HS-85-47, CRD NOTCH OVERRIDE, in NOTCH OVERRIDE **AND** 2-HS-85-48, CRD CONTROL SWITCH, in ROD OUT NOTCH. [4.4] **WHEN** EITHER of the following occur: • Control Rod begins to move, **OR** • It is determined the rod will **NOT** move, **THEN RELEASE** 2-HS-85-47 AND 2-HS-85-48. [4.5] **IF** the Control Rod successfully notches out, **THEN PROCEED** to Section 6.6 and **WITHDRAW** the Control Rod to the appropriate position. [4.6] **IF** desired, **THEN REPEAT** Steps 8.16[4.1] through 8.16[4.5] several times prior to raising Drive Water Pressure in Step 8.16[5]. [5] **IF** double clutching the Control Rod was unsuccessful, **THEN PERFORM** the following to withdraw the Control Rod using elevated Drive Water Pressure: [5.1] **RAISE** CRD DRIVE WATER HEADER DP to 300 psid on 2-PDI-85-17A using 2-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE. [5.2] **PERFORM** the following to double clutch the Control Rod at Position 00: [5.2.1] **PLACE and HOLD** 2-HS-85-47, CRD NOTCH OVERRIDE in EMERG ROD IN, for several seconds. [5.2.2] **CHECK** the Control Rod full in indication (double green dashes) on the Full Core Display for the associated Control Rod.

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 3	Page 3 of 3

Event Description: Control Rod Difficult to Withdraw

Time	Position	Applicant's Actions or Behavior
	OATC	[5.2.3] **SIMULTANEOUSLY PLACE** 2-HS-85-47, CRD NOTCH OVERRIDE, in NOTCH OVERRRIDE and **PLACE** 2-HS-85-48, CRD CONTROL SWITCH, in ROD OUT NOTCH. [5.2.4] **WHEN** EITHER of the following occur: • Control Rod begins to move, **OR** • It is determined the rod will **NOT** move, **THEN RELEASE** 2-HS-85-47 AND 2-HS-85-48 [5.2.5] **IF** the Control Rod successfully notches out, **THEN LOWER** CRD DRIVE WTR HDR DP, 2-PDI-85-17A, to between 250 psid and 270 psid, using 2-HS-85-23A, CRD DRIVE WATER PRESSURE CONTROL VALVE, and **PROCEED** to Section 6.6.
	OATC	Informs the NUSO that Control Rod 58-31 is no longer difficult to withdraw.
	NUSO	Direct the OATC to continue with Event 2, Control Rod Withdrawal.
	NRC	**End of Event 3. When satisfied with Event 2 and 3, request that the driver insert Event 4, Inadvertent Core Spray Pump Start.**

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 4 Page 1 of 4

Event Description: Inadvertent Core Spray Pump Start

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; 2A Core Spray Pump will inadvertently start

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to cause an inadvertent trip of 2A Core Spray Pump.**
	BOP	Acknowledges and reports the following alarms: • CORE SPRAY SYS II PUMP A START, 2-9-3C, Window 5 • RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE, 2-9-3C, Window 10 • CORE SPRAY PUMP 2A RUNNING, 2-9-23D, Window 33 • RHR SERVICE WATER PUMP B3 RUNNING, 2-9-23C, Window 34 **(only received if the Core Spray Pump runs longer than 28 seconds)** Reports that 2A Core Spray Pump has started and that there are no valid signals that should result in Core Spray Pumps starting.
	SRO	Directs the BOP to stop 2A Core Spray Pump and respond in accordance with Alarm Response Procedures.
	BOP	Alarm Response Procedure, 2-ARP-9-3C CORE SPRAY SYS II PUMP A START, Window 5 Operator Action: A. **CHECK** auto start signals by multiple indications. B. **CHECK** Core Spray Pump 2A operation by motor amps, discharge pressure, and flow on Panel 2-9-3. C. **IF** pump is **NOT** needed, **THEN STOP** Core Spray Pump 2A. D. **WHEN** the auto start signal is reset and Core Spray is **NOT** required for Core Cooling, **THEN DEPRESS** 2-XS-75-61, CORE SPRAY SYSTEM I AUTO INITIATION RESET pushbutton, on Panel 2-9-3. E. **RETURN** system to standby readiness.

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 4 Page 2 of 4

Event Description: Inadvertent Core Spray Pump Start

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 1/2-9-23A CORE SPRAY PUMP 2A RUNNING, Window 37 Operator Action: A. **IF** accident signal is **NOT** present, **THEN** 1. **CHECK** panel 2-9-3 to verify pump is running. 2. **INITIATE** actions to determine why pump is running. 3. **SECURE** pump.
	BOP	Alarm Response Procedure, 2-9-23C RHR SERVICE WATER PUMP B3 RUNNING, Window 34 Operator Action: A. **IF** accident signal is **NOT** present, **THEN PERFORM** the following: 1. **CHECK** panel 9-3 to verify RHRSW Pump B3 Pump is running. 2. **ENSURE** adequate RCW Pressure PI-24-18, panel 9-20. 3. **CHECK** Diesel Generator or Core Spray Pump running.
	Driver	**If informed as Unit 1 and 2, acknowledge that 2A Core Spray Pump inadvertently started, which caused B3 RHRSW Pump to start (if 2A Core Spray Pump is not secured in 28 seconds). If asked, neither Unit 1 nor Unit 3 require that A3 RHRSW Pump remain running.** **If contacted as the Control Bay AUO to check the ATUs in the Aux. Instrument Room, acknowledge the direction.** **When contacted as Maintenance, the Outside/Work Control SRO, or an AUO to investigate, wait 3 minutes and report that the oil level on 2A Core Spray Pump is below the sight glass.**
	BOP	Determines that 2A Core Spray Pump inadvertently started, recommends to the SRO that 2A Core Spray Pump be secured, and secures 2A Core Spray Pump and B3 RHRSW Pump when directed.

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 4 Page 3 of 4

Event Description: Inadvertent Core Spray Pump Start

Time	Position	Applicant's Actions or Behavior
	BOP	0-OI-67, Emergency Equipment Cooling Water System Section 7.2, Return to Standby Readiness of the South Header [1] **ENSURE** Chemistry treatment to associated RHRSW (EECW) Pump discharge line secured. [2] **ENSURE** there are NO emergency Diesel Generators running on any unit. [3] **ENSURE** there are NO Core Spray Pumps running on any unit. [4] **ENSURE** EECW system is **NOT** being used to supply cooling water to RBCCW. [5] **ENSURE** EECW System is **NOT** being used to supply cooling water to the Control Air Compressors. **NOTE** The RHRSW Pump Control Switches and amp meters are located at Control Room Panel 9-3, Unit 1, 2, and 3. **CAUTION** It is normal practice to keep one RHRSW Pump running on each of the North and South EECW headers to ensure the headers remain solid with no trapped air, and to ensure operability requirements. [6] **IF** Pump B3 is to be **SHUT DOWN**, **THEN PLACE** the following switch momentarily to the STOP position: • 0-HS-23-88A/2, RHRSW PUMP B3 EECW SOUTH HDR, on Unit 2 [7] N/A [8] **IF** either pump remains in operation, **THEN CHECK** running current for the remaining pump is less than 61 amps using one of the following: • 0-EI-23-94/2, RHRSW PUMP D3 AMPS, on Unit 2
	NRC	**The candidate may reference Technical Specification 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation; however, no information will be given by the driver concerning the reason for 2A Core Spray Pump auto starting. Therefore, Technical Specification 3.3.5.1 is not required.**

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 4 · Page 4 of 4

Event Description: Inadvertent Core Spray Pump Start

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. NOTE: LCO 3.0.4.b is not applicable to HPCI. Condition: A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump inboth LPCI subsystems inoperable.	
	NUSO	**REQUIRED ACTION:** A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	**COMPLETION TIME:** A.1 – 7 days
	Driver	**If contacted as the Outside/Work Control NUSO or as an AUO to hang a clearance on or open the circuit breaker for 2A Core Spray Pump, wait 2 minutes and insert Event 14. Contact the crew to inform them that the breaker for 2A Core Spray Pump is open.** **If contacted as Engineering to perform an evaluation, acknowledge the direction.** **If contacted as the Outside/Work Control NUSO to address Protected Equipment acknowledge the direction.**	
	NRC	**End of Event 4. Request that the driver insert Event 5, 2B Stack Dilution Fan Failure, Standby Fan Fails to Automatically Start**	

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 5 Page 1 of 1

Event Description: 2A Stack Dilution Fan Failure, Standby Fan Fails to Automatically Start

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; 2A Stack Dilution Fan will trip

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.
	NRC	**It is acceptable if BOP elects to start 2B Stack Dilution Fan in accordance with 2-OI-66, Section 8.6 instead of using the ARP below.**
	BOP	Alarm Response Procedure, 2-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 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, Reactor High Pressure A1 Channel Fails High/Half SCRAM (2-PIS-3-22AA).**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 6 Page 1 of 5

Event Description: Reactor High Pressure A1 Channel Fails High/Half SCRAM (2-PIS-3-22AA)

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; 2-PIS-3-22AA will fail high and a Reactor half SCRAM will occur

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to fail Reactor High Pressure Channel A1 high.**
	OATC	Acknowledges and reports the following alarms to the NUSO: • RX VESSEL PRESSURE HIGH HALF SCRAM, 2-9-4A, Window 9 • REACTOR CHANNEL 'A' AUTO SCRAM, 2-9-5B, Window 1
	NUSO	Directs the OATC to respond in accordance with the appropriate Alarm Response Procedures.
	OATC	2-ARP-9-4A, Alarm Response Procedure RX VESSEL PRESSURE HIGH HALF SCRAM, 2-9-4A, Window 9 Operator Action: A. **CONFIRM** alarm by multiple indications. B. N/A C. **DISPATCH** personnel to the sensors to check for abnormal conditions. D. **IF** alarm is NOT valid or initiating condition is corrected, **THEN PERFORM** the following: 1. **RESET** Half SCRAM with SRO permission. 2. **REFER TO** 2-OI-99, Reactor Protection System. 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.
	OATC	2-ARP-9-5B, Alarm Response Procedure REACTOR CHANNEL 'A' AUTO SCRAM, 3-9-5B, Window 1 Operator Action: A. **CHECK** channel A relays dropped out by checking SCRAM solenoid and backup SCRAM valve lights extinguished. B. **IF** any EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). C. N/A

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 6 Page 2 of 5

Event Description: Reactor High Pressure A1 Channel Fails High/Half SCRAM (2-PIS-3-22AA)

Time	Position	Applicant's Actions or Behavior
	OATC	D. N/A E. With NUSO permission, **RESET** half-SCRAM. **REFER TO** 3-OI-99, Reactor Protection System. F. **CHECK** first-out printer to determine initiating cause.
	OATC	2-ARP-9-5B, Alarm Response Procedure RPS ANALOG TRIP UNIT TROUBLE, 2-9-5B, Window 23 Operator Action: A. **DISPATCH** personnel to Aux Instrument Room to investigate cause. B. **IF** a power supply light on ATU Panel 3-9-83, -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. **FREQUENTLY MONITOR** other input parameters. E. **IF** necessary, **INITIATE** a Work Order (WO) for Instrument Group to troubleshoot and repair. F. **IF** alarm is **NOT** valid, **THEN REFER TO** 0-OI-55, Annunciator System.
	Driver	**When directed to investigate as the Control Bay AUO or Outside/Work Control NUSO, wait 3 minutes and inform the crew that 2-PIS-3-22AA, Reactor Pressure A1 Channel is failed high with a Gross Failure. Investigation is in progress.**
	NRC	**After the NUSO addresses Technical Specifications, request that the driver contact the crew with the report from the Instrument Mechanics concerning the cause.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 6 Page 3 of 5

Event Description: Reactor High Pressure A1 Channel Fails High/Half SCRAM (2-PIS-3-22AA)

<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">

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

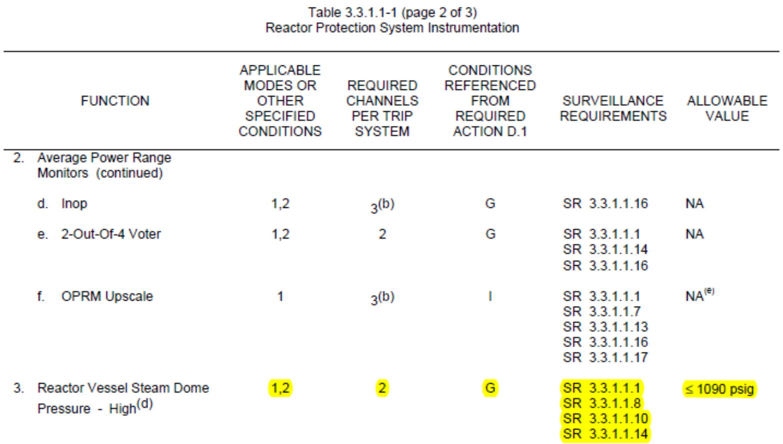


Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

<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>2. Average Power Range Monitors (continued)</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>d. Inop</td><td>1,2</td><td>3(b)</td><td>G</td><td>SR 3.3.1.1.16</td><td>NA</td></tr>
<tr><td>e. 2-Out-Of-4 Voter</td><td>1,2</td><td>2</td><td>G</td><td>SR 3.3.1.1.1
SR 3.3.1.1.14
SR 3.3.1.1.16</td><td>NA</td></tr>
<tr><td>f. OPRM Upscale</td><td>1</td><td>3(b)</td><td>I</td><td>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</td><td>NA(e)</td></tr>
<tr><td>3. Reactor Vessel Steam Dome Pressure - High(d)</td><td>1,2</td><td>2</td><td>G</td><td>SR 3.3.1.1.1
SR 3.3.1.1.8
SR 3.3.1.1.10
SR 3.3.1.1.14</td><td>≤ 1090 psig</td></tr>
</table>

Condition:

A. One or more required channels inoperable.

</td></tr>
<tr><td></td><td>NUSO</td><td>

REQUIRED ACTION:

A.1 Place channel in trip

<u>OR</u>

A.2 NOTE: Not applicable to Functions 2.a, 2.b, 2.c, 2.d, or 2.f. Place associated trip system in trip

</td><td>

COMPLETION TIME:

A.1 – 12 hours

A.2 – 12 hours

</td></tr>
<tr><td></td><td>NRC</td><td colspan="2">When the NUSO has addressed Technical Specifications, request that the driver contact the Control Room with the cause for the Reactor High Pressure trip.</td></tr>
</table>

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 6 Page 4 of 5

Event Description: Reactor High Pressure A1 Channel Fails High/Half SCRAM (2-PIS-3-22AA)

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, inform the crew that Instrument Mechanics in the Auxiliary Instrument Room reported that while performing the initial test setup for 2-SR-3.3.6.1.5(1B/A1), Primary Containment Isolation System Main Line Low Pressure Instrument Channel A1 Calibration, they inadvertently connected to 2-PIS-3-22AA, Reactor Pressure A1 channel, and tripped the ATU.** **If contacted as an Instrument Mechanic, report that no adjustments were made to 2-PIS-3-22AA, Reactor Pressure A1 Channel, that all test equipment has been removed, the instrument is within testing periodicity, and no retests are required.** **If contacted as the Shift Manager, acknowledge any report given. State to the applicant that demonstration of operability is not required – if asked for guidance, ask the candidate for a recommendation and agree with any recommendation.**
	Driver	**If directed to reset the Gross Failure, insert Event 16 and Report that the Gross Failure is reset.**
	NUSO	Directs the OATC to reset the Half SCRAM in accordance with 2-OI-99, Reactor Protection System, Section 6.1, Reset of One RPS Trip Logic Channel.
	OATC	Resets the Half Reactor SCRAM on Panel 2-9-5. 2-OI-99, Reactor Protection System Section 6.1, Reset of One RPS Trip Logic Channel. [1] **ENSURE** Reactor Protection System in prestartup/standby readiness alignment in accordance with Section 4.0. [2] **REVIEW** all Precautions and Limitations in Section 3.0. Refer to sections 5.1, 5.2, 8.1, or 8.2 [3] **ENSURE** RPS Bus for tripped channel ENERGIZED. [4] **ENSURE** trip signals **NOT** present.

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 6 Page 5 of 5

Event Description: Reactor High Pressure A1 Channel Fails High/Half SCRAM (2-PIS-3-22AA)

Time	Position	Applicant's Actions or Behavior
	OATC	[5] **MOMENTARILY PLACE** SCRAM RESET, 2-HS-99-5A/S5, as follows: [5.1] RESET FIRST. (Group 2/3) [5.2] RESET SECOND. (Group 1/4) [5.3] NORMAL [6] **CHECK** the following conditions: A. All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED. B. The following four lights ILLUMINATED: • 2-IL-99-5A/AB, SYSTEM A BACKUP SCRAM VALVE • 2-IL-99-5A/CD, SYSTEM B BACKUP SCRAM VALVE 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 "NOT TRIP" 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 "NOT TRIP" for RPS "B".
	NRC	**End of Event 6. Request that the driver insert Event 7, Loss of Offsite Power.**

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 7	Page 1 of 12

Event Description: Loss of Offsite Power

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner; a loss of 500 KV and 161 KV Offsite Power will occur, resulting in a Reactor SCRAM

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to cause a Loss of all Offsite Power.**
	NRC	**Event 8, 'B' Diesel Generator fails to Tie Automatically, and Event 9, Emergency Equipment Cooling Water (EECW) Pumps Fail to Auto Start will automatically occur during Event 7. See the respective Event pages (Event 8 – page 37 of 39 and Event 9 – page 38 of 39) for actions. No action is required by the driver to insert Event 8 or 9.**
	NRC	**Due to time constraints, it is not expected that the NUSO reference Technical Specifications for this Event. Technical Specification evaluation should not be used to evaluate the applicant's Technical Specification competency.**
	Crew	Determines that a loss of Offsite Power has occurred.
	SRO	Directs the crew to respond in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout and 2-AOI-100-1, Reactor SCRAM.
	NRC	**The subsequent actions contained in 2-AOI-100-1, Reactor SCRAM, are too numerous to list here, therefore only immediate actions are listed below.**
	OATC	2-AOI-100-1, Reactor SCRAM 4.1 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** REACTOR MODE SWITCH, 2-HS-99-5A/S1, in SHUTDOWN. [3] N/A

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 7	Page 2 of 12

Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	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
	OATC	**NOTES** 1) Perform steps 4.1[5.3] and 4.1[5.4] in parallel. 2) Step 4.1[5.7] should be reported IMMEDIATELY when that Condition is reached. 3) Step 4.1[5.8] may be performed before step 4.1[5.7] if Reactor Water Level is slowly lowering. [5] N/A
	OATC	Provides a Reactor SCRAM report to the NUSO. Responds in accordance with 2-AOI-100-1, Reactor SCRAM.
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout **NOTE** Performing this instruction, in conjunction with an earthquake, may require resetting the individual Diesel Generator's 86G Lockout Relay and the Field Breaker (both locally at the Diesel Generator electrical cabinet). Immediate Actions: 1] **ENSURE** Diesel Generators have started and tied to respective 4kV Shutdown Boards, **THEN DISPATCH** personnel to Diesel Generators.

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 7	Page 3 of 12

Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	BOP	Determines 'B' did not tie onto 'B' 4KV Shutdown Board. **SEE EVENT 8 ACTIONS (Page 37 of 42)**
	BOP	[2] **ENSURE** two Emergency Equipment Cooling Water (EECW) Pumps (not using the same EECW strainer) are in service supplying Diesel Generators. **SEE EVENT 9 ACTIONS (Page 38 of 42)**
	BOP	[4] **PERFORM** the following to ensure at least one train of Diesel Generator Room Fans are energized: • **CHECK** 480V DSL Aux Board A or B energized
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout 4.2 Subsequent actions **NOTES** 1) SBO Unit does attachment 12 only; the other two units perform subsequent actions of AOI. 2) The following is the preferred pump combinations of Unit 1 and Unit 2 RHR Pumps used in SDC: 1B/2D, 1A/2C, 1A/2D, 1B/2C. By using these pumps it ensures that a loss of a Diesel during a Station Blackout would not result in a loss of Shutdown cooling for both Unit 1 and Unit 2. [1] **IF** ANY EOI entry condition is met, **THEN REFER TO** the appropriate EOI(s). (Otherwise N/A)

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 7	Page 4 of 12

Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	NUSO	Enters 2-EOI-1, RPV Control. Directs Reactor Pressure Control in accordance with 2-EOI-APPENDIX-11A, Alternate RPV Pressure Control Systems MSRVs (see below). Maybe direct either of the following EOI Apendices for Reactor Water Level Control: • 2-EOI-APPENDIX-5D, Injection System Lineup HPCI (see page 29 of 39) • 2-EOI-APPENDIX-5C, Injections System Lineup RCIC (see page 31 of 39)
	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 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

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 7	Page 5 of 12

Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	BOP	[4] N/A [5] N/A [6] N/A **END OF 2-EOI-APPENDIX-11A**
	BOP	2-EOI-APPENDIX-5D, Injection System Lineup HPCI [1] N/A [2] N/A [3] N/A [4] **VERIFY** at least one Standby Gas Train (SGT) in operation.
	Driver	**Standby Gas Trains (SGT) should already be running, but if not and if contacted to start a SGT, insert Event 17 to start 'A' SGT.**
	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 140º F may result in equipment damage. [5] **VERIFY** 2-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

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 7 Page 6 of 12

Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	BOP	**NOTE** HPCI Auxiliary Oil Pump will NOT start UNTIL 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE starts to open. [6] N/A [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 MINIMUM 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 HPCI Turbine. [12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine speed accelerates. B. 2-CKV-73-45, HPCI SYSTEM CHECK VALVE, 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 MINIMUM FLOW VALVE, closes as flow exceeds approximately 1200 GPM. **CAUTION** 2-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 Minimum Flow Valve may be required for pump min flow protection.

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 7 Page 7 of 12

Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	BOP	[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] **IF** It is desired to align HPCI suction to the Suppression Pool, **THEN PERFORM** the following: A. **OPEN** 2-FCV-73-26, HPCI SUPPRESSION POOL INBOARD SUCTION VALVE. B. **OPEN** 2-FCV-73-27, HPCI SUPPRESSION POOL OUTBOARD SUCTION VALVE. C. **WHEN** 2-FCV-73-26, HPCI SUPPRESSION POOL INBOARD SUCTION VALVE and 2-FCV-73-27, HPCI SUPPRESSION POOL OUTBOARD SUCTION VALVE, are fully open, **THEN VERIFY CLOSED** 2-FCV-73-40, HPCI CST SUCTION VALVE. **NOTE** Step 1.0[17]B must be performed promptly following Step 1.0[17]A to avoid loss of suction path. [17] N/A **END OF 2-EOI-APPENDIX-5D**
	BOP	2-EOI-Appendix-5C, Injection System Lineup RCIC [1] N/A [2] **ENSURE RESET** auto isolation logic using 2-XS-71-51A(B) RCIC AUTO-ISOLATION LOGIC A (B) RESET pushbuttons. [3] **ENSURE RESET** and **OPEN** 2-FCV-71-9, RCIC TURBINE TRIP/THROTTLE VALVE. [4] **ENSURE** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with setpoint at 620 GPM.

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 7 Page 8 of 12

Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	BOP	[5] **OPEN** 2-FCV-71-34, RCIC PUMP MINIMUM FLOW VALVE. [6] **OPEN** 2-FCV-71-39, RCIC PUMP INJECTION VALVE. [7] **OPEN** 2-FCV-71-25, RCIC LUBE OIL COOLING WATER VALVE. [8] **PLACE** 2-HS-71-31A, RCIC VACUUM PUMP in START. **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 240º F may result in equipment damage. [9] **OPEN** 2-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC turbine. [10] **CHECK** proper RCIC operation by observing the following: A. RCIC Turbine Speed accelerates above 2100 RPM. B. Flow to RPV controlled automatically at 620 GPM. C. 2-FCV-71-34, RCIC PUMP MINIMUM FLOW VALVE closes as flow rises above 120 GPM. [11] **ADJUST** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. [12] **IF** BOTH of the following exist: • RCIC Initiation signal is NOT present, **AND** • RCIC flow is below 60 GPM, **THEN ENSURE OPEN** 2-FCV-71-34, RCIC PUMP MINIMUM FLOW VALVE

Op Test No.: 22-04	Scenario No. NRC-5	Event No.: 7	Page 9 of 12

Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	BOP	[13] N/A [14] N/A **END OF 2-EOI-APPENDIX-5C**
	BOP	Continuing 0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout 4.2 Subsequent actions [2] **IF** any Unit is under a Station Blackout, **THEN ONLY PERFORM** Attachment 12 for that Unit: (Otherwise N/A).
	NRC	**The Station Blackout time starts when power is lost to all 480V Shutdown Boards.**
	CREW	**Critical Task: START: _________ STOP: _________** **With a Loss of Offsite Power and a Station Blackout caused by the failure of two Emergency Diesel Generators to automatically start and tie on to their respective 4KV Shutdown Boards, the crew restores power to one 480V Shutdown Board to exit Station Blackout within 20 minutes of the loss of power.** **Critical Task Failure Criteria:** **The operating crew fails to restore power to one 480V Shutdown Board within 20 minutes of the Station Blackout on Unit 2.**

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 7 Page 10 of 12

Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	BOP	Determines that Unit 2 is in a Station Blackout condition due to the following plant conditions: • 'B' Diesel Generator has failed to tie onto 'B' 4KV Shutdown Board • 'D' EDG did not automatically and will not manually start
	SRO	Acknowledges that Unit 2 is in a Station Blackout condition. Directs the BOP to transfer 2B 480V Shutdown Board to the alternate power supply to exit Station Blackout.
	BOP	[3] N/A [4] **CHECK** automatic actions and **PERFORM** any that failed to occur. **NOTE** The following subsequent actions may be performed out of order, depending on plant conditions. [5] **REFER TO** 2-AOI-78-1, Fuel Pool Cooling System Failure for a complete Loss of AC POWER, as necessary.

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 7 Page 11 of 12

Event Description: Loss of Offsite Power

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>NOTES
1) EECW supply valves to the Control Air Compressors and Reactor Building Closed Cooling Water (RBCCW) are air operated. If initial air pressure is low, air compressors may trip on high temperature, until cooling water flow is established.
2) At US discretion, 0-FCV-67-53, EECW NORTH HDR SUPPLY VLV TO AIR COMP, can be placed in the open position with hand switch. The valve will automatically come open once EECW pressure is above setpoint. REFER TO 0-OI-67, Emergency Equipment Cooling Water System, for valve operation.
3) 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, EECW NORTH HEADER SUPPLY TO RBCCW, 2-FCV-67-50 EECW NORTH HEADER SUPPLY TO RBCCW, or 3-FCV-67-51 EECW SOUTH HEADER SUPPLY TO RBCCW, opens.</td></tr>
<tr><td></td><td>BOP</td><td>[6] WHEN EECW header pressure is restored above the reset pressure setpoint (psig) for the valves listed below, THEN
<table>
<tr><th></th><th>Common</th><th>Unit 2</th></tr>
<tr><td>0-FCV-67-53</td><td>106</td><td></td></tr>
<tr><td>FCV-67-50</td><td></td><td>91</td></tr>
<tr><td>FCV-67-51</td><td></td><td>109</td></tr>
</table>
RESET EECW supplies to Control Air Compressors and RBCCW, at Unit 1 Panel 1-LPNL-925-0032 and Unit 2, 3 Panels 2(3)-25-32. REFER TO the EECW to the RCW Crossties for Control Air & RBCCW section of 0-OI-67, Emergency Equipment Cooling Water System.</td></tr>
</table>

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 7 Page 12 of 12
Event Description: Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	BOP	[7] **START** Control Air Compressors A, D and G as required and **MONITOR** system pressure. **REFER TO** 0-AOI-32-1, Loss of Control and Service Air Compressors. [7.1] **IF** an Air Compressor trips on high temperature, **THEN** (Otherwise N/A) **NOTIFY** Unit Supervisor for instructions. [8] **REFER TO** 2-AOI-32-2, Loss of Control Air, as necessary. [9] **PLACE** RPS MG Sets A and B in service. **REFER TO** 2-OI-99, Reactor Protection System. [10] **START** the Diesel Driven Fire Pump. **REFER TO** 0-OI-26, Fire Protection System.
	Driver	**If directed as the Outside/Work Control SRO to reset EECW, restore Control Air, restore RPS, or start the Diesel Driven Fire Pump, insert the following events as required:** • **Event 27 – Reset EECW** • **Event 28 – Restore Control Air** • **Event 29 – Restore RPS** • **Event 30 – Start Diesel Driven Fire Pump**
	NRC	**When the crew has restored power to a 480V Shutdown Board, has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using RCIC or HPCI, and has started EECW Pumps, end of scenario.**

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 8 Page 1 of 1

Event Description: 'B' EDG Fails to Tie Automatically

Symptoms/Cues: Event is automatically inserted on simulator setup; 'B' EDG will not automatically tie onto 'B' 4KV Shutdown Board upon the Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8 is automatically entered on simulator setup and will occur during Event 7, Loss of Offsite Power. No action is required by the driver to insert Event 8.**
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout **NOTE** Performing this instruction, in conjunction with an earthquake, may require resetting the individual Diesel Generator's 86G Lockout Relay and the Field Breaker (both locally at the Diesel Generator electrical cabinet). Immediate Actions: 1] **ENSURE** Diesel Generators have started and tied to respective 4kV Shutdown Boards, **THEN DISPATCH** personnel to Diesel Generators.
	BOP	Determines that 'B' Diesel Generator did not tie onto 'B' 4KV Shutdown Board and closes the Diesel Generator Output Breaker.
	NRC	**End of Event 8. Either continue in Event 7, Loss of Offsite Power, or when has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using RCIC or HPCI, and has started EECW Pumps, end of scenario.**

Form 3.3-2 Required Operator Actions

Op Test No.: 22-04 Scenario No. NRC-5 Event No.: 9 Page 1 of 1

Event Description: Emergency Equipment Cooling Water (EECW) Pumps Fail to Auto Start

Symptoms/Cues: Event is automatically inserted on simulator setup; EECW Pumps will not start as expected following the automatic start of Diesel Generators

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9 is automatically entered on simulator setup and will occur during Event 7, Loss of Offsite Power. No action is required by the driver to insert Event 9.**
	NRC	**The Critical Task start time commences when EDGs start following the loss of Offsite Power.**
	CREW	**Critical Task: START: _________ STOP: _________** **With a loss of EECW Pumps due to a Loss of Offsite Power, the crew restores EECW Flow to the EDGs within 8 minutes.** **Critical Task Failure Criteria:** **The operating crew fails to restore EECW Flow to the EDGs within 8 minutes following EDG start upon a Loss of Offsite Power.**
	BOP	[2] **ENSURE** two Emergency Equipment Cooling Water (EECW) Pumps (not using the same EECW strainer) are in service supplying Diesel Generators.
	BOP	Starts two EECW Pumps to supply Cooling Water Flow to EDGs.
	NRC	**End of Event 9. Either continue in Event 7, Loss of Offsite Power, or when has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using RCIC or HPCI, and has started EECW Pumps, end of scenario.**

UNIT 2 SHIFT TURNOVER MEETING			**Today**
MODE **2**	**DAYS ON LINE** **0**	**Total Drywell Leakage (gpm)** **1.55**	**Protected Equipment** **None**
Rx Power **~3%**	PRA (EOOS) -GREEN 500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (gpm)** **0.11**	
MWe	**Last breaker closure**	**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

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

Reactor Startup.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Warm 2B RFPT in accordance with 2-OI-3, Reactor Feedwater System, starting at Section 5.6. [3.3]. Seal Steam is already in service.

Continue the Reactor Startup Contact the Ops Superintendent prior to placing the MODE Switch in RUN.

OPERATOR WORK AROUNDS OWAs – U0-1/U2-0 Burdens - U0 – 0/U2-0 Operator Challenges – U0 – 12/U2 - 6

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Inject Boron during an ATWS per EOI-APPENDIX-3A, SLC Injection	
JPM NUMBER:		708A	REVISION:	1

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-063-AL-03 / Inject SLC in accordance with EOI-APPENDIX-3A		
K/A RATINGS:	RO/SRO: 3.9		
K/A No. & STATEMENT:	211000 Standby Liquid Control System A1.08; Ability to predict or monitor changes in parameters associated with operation of the Standby Liquid Control System, including: RWCU system lineup		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	1		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	4 minutes		

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: 708A

RO _______ SRO _______ DATE: ___________

TASK STANDARD: The Examinee is expected to inject SLC in accordance with EOI-APPENDIX-3A and take manual action to isolate Reactor Water Cleanup.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 3-EOI-APPENDIX-3A

VALIDATION TIME: 4 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	08/08/17	All	Initial issue
1	11/01/21	All	JPM Update

Procedure Revisions

Procedure	Revision
2-EOI-APP-3A	6

SIMULATOR SETUP

IC	28
Exam IC	265

Console Operator Instructions	• Reset to IC 265 • Run schedule file 2204 NRC JPM a Unit 2.sch • Verify event file 2204 NRC JPM a Unit 2.evt loads • Place the simulator in RUN to ensure stable conditions

Malfunctions	Description	Event	Severity	Delay	Initial set
FCV-69-1	MOTOR_OPERATED_VALVE RWCU INBOARD ISOLATION VALVE	N/A	N/A	N/A	N/A
FCV-69-2	MOTOR_OPERATED_VALVE RWCU OUTBOARD ISOLATION VALVE	N/A	N/A	N/A	N/A
FCV-69-12	MOTOR_OPERATED_VALVE RWCU DISCH ISOLATION VALVE	N/A	N/A	N/A	N/A

Overrides	Description	Event	Severity	Delay	Initial set
ZLOHS691_1	HS-69-1 RWCU INBOARD SUCTION ISOLATION VALVE	N/A	N/A	N/A	N/A
ZLOHS691_2	HS-69-1 RWCU INBOARD SUCTION ISOLATION VALVE	N/A	N/A	N/A	N/A
ZLOHS692A_1	HS-69-2A RWCU OUTBOARD SUCTION ISOLATION VALVE	N/A	N/A	N/A	N/A
ZLOHS692A_2	HS-69-2A RWCU OUTBOARD SUCTION ISOLATION VALVE	N/A	N/A	N/A	N/A
ZLOHS6912A_1	HS-69-12A RWCU RETURN ISOLATION VALVE	N/A	N/A	N/A	N/A
ZLOHS6912A_2	HS-69-12A RWCU RETURN ISOLATION VALVE	N/A	N/A	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 (BOP) on Unit 2. A Reactor SCRAM has occurred with the following conditions:

- Control Rods failed to insert
- Reactor Power is >5%
- The SRO has entered 2-EOI-1A, ATWS RPV Control
- HPCI is tripped and locked out
- The Operator at the Controls is lowering Reactor Water Level to (-) 50 to (-) 180 inches

INITIATING CUE:

The Unit Supervisor directs you to inject Standby Liquid Control (SLC) in accordance with 2-EOI-APPENDIX-3A, SLC Injection.

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **UNLOCK** and **PLACE** 2-HS-63-6A, SLC PUMP 2A/2B, control switch in START-A or START-B position. Standard: Unlocks and places 2-HS-63-6A, SLC PUMP 2A/2B, control switch in either the START-A or START-B position.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 2: [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) Standard: Verifies that the selected SLC Pump started and is injecting to the RPV by observing the parameters listed above.	______ SAT ______UNSAT ______N/A
Step 3: [3] **IF** Proper system operation CANNOT be verified, **THEN RETURN** to Step 1.0[1] and START the other SLC pump. Standard: Verifies proper operation and does not start the other SLC Pump.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Begin Alternate Path - Reactor Water Cleanup (RWCU) will fail to automatically isolate when SLC is initiated. This malfunction is entered on simulator setup. No action is required by the driver for the Alternate Path.	
Step 4: [4] **VERIFY** RWCU isolation by observing the following: • RWCU Pumps 2A and 2B tripped • 2-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE closed • 2-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE closed **Note to Examiner: Tripping the RWCU Pumps and closing either 2-FCV-69-1 OR 2-FCV-69-2 will meet the critical step.** Standard: Determines that RWCU failed to automatically isolate. In accordance with OPDP-1, Conduct of Operations, Section 3.5.3, Manual Control of Automatic Systems, manually closes 2-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE, and 2-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE and trips RWCU Pumps.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 5: **VERIFY** 2-FCV-69-12, RWCU RETURN ISOLATION VALVE closed Standard: Determines that RWCU failed to automatically isolate. In accordance with OPDP-1, Conduct of Operations, Section 3.5.3, Manual Control of Automatic Systems, manually closes 2-FCV-69-12, RWCU RETURN ISOLATION VALVE.	_____SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 6: [5] **VERIFY** ADS inhibited. Standard: Inhibits ADS. Places ADS LOGIC A INHIBIT, 2-XS-1-159A, and ADS LOGIC B INHIBIT, 2-XS-161A, in the INHIBIT position. Verifies that ADS LOGIC BUS A INHIBITED (2-XA-55-3C, Window 18), and ADS LOGIC BUS B INHIBITED (2-XA-55-3C, Window 31) alarms are received.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 7: [6] **MONITOR** Reactor Power for downward trend. Standard: Monitors all available APRMs/IRMs for downward trend in Reactor Power.	_____SAT ______UNSAT ______N/A
Step 8: [7] **MONITOR** 2-LI-63-1A, SLC STORAGE TANK LEVEL, and CHECK that level is dropping approximately 1% per minute. Standard: Monitors SLC STORAGE TANK LEVEL, 2-LI-63-1A, and observes SLC Tank Level is lowering.	_____SAT ______UNSAT ______N/A
Cue: “Another Operator will secure SLC when necessary”.	

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 the Balance of Plant Operator (BOP) on Unit 2. A Reactor SCRAM has occurred with the following conditions:

- Control Rods failed to insert
- Reactor Power is >5%
- The SRO has entered 2-EOI-1A, ATWS RPV Control
- HPCI is tripped and locked out
- The Operator at the Controls is lowering Reactor Water Level to (-) 50 to (-) 180 inches

INITIATING CUE:

The Unit Supervisor directs you to inject Standby Liquid Control (SLC) in accordance with 2-EOI-APPENDIX-3A, SLC Injection.

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Align RCIC Suction to the Suppression Pool per EOI-Appendix-5C, Injection System Lineup RCIC	
JPM NUMBER:		751A	REVISION:	0

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-31 / Lineup Injection Subsystems – RCIC in accordance with EOI-Appendix-5C		
K/A RATINGS:	RO/SRO: 3.8		
K/A No. &STATEMENT:	217000 RCIC Reactor Core Isolation Cooling A4.03; Ability to manually operate and/or monitor in the Control Room: System Valves		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	2		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	4 minutes		

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*

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: ________________________________ JPM Number: 751A

RO _______ SRO _______ DATE: ___________

TASK STANDARD: The Examinee is expected to align RCIC Suction to the Suppression Pool per EOI-Appendix-5C and respond to a valve malfunction.

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: 4 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	11/01/21	All	Initial issue

Procedure Revisions

Procedure	Revision
2-EOI-APP-5C	8

SIMULATOR SETUP

IC	28
Exam IC	258

Console Operator Instructions	• Reset to IC 258 • Run schedule file 2204 NRC JPM b Unit 2.sch • Verify Event file 2204 NRC JPM b Unit 2.evt loads • Place the simulator in RUN to ensure stable conditions

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 the Balance of Plant Operator (BOP) on Unit 2. A Reactor SCRAM has occurred with the following conditions:

- The Nuclear Unit Senior Operator (NUSO) has entered 2-EOI-1, RPV Control Modes 1-3
- Reactor Core Isolation Cooling (RCIC) is in service in accordance with 2-EOI-Appendix-5C, Injection System Lineup RCIC
- Reactor Pressure is being maintained by the Bypass Valves

INITIATING CUES:

The Unit Supervisor directs you to align the RCIC Suction to the Suppression Pool in accordance with 2-EOI-Appendix-5C.

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: [13] **IF** desired to align suction to the Suppression Pool, **THEN PERFORM** the following: A. **OPEN** 2-FCV-71-17, RCIC SUPPR POOL INBD SUCTION VALVE. Standard: Opens 2-FCV-71-17.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 2: B. **OPEN** 2-FCV-71-18, RCIC SUPPRESSION POOL OUTBOARD SUCTION VALVE. Standard: Opens 2-FCV-71-18.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER NOTE: Begin Alternate Path – 2-FCV-71-19, CST SUCTION VALVE, will fail to automatically close when both 2-FCV-71-17 and 2-FCV-71-18 are fully open.	
Step 3: C. **WHEN** 2-FCV-71-17 and 2-FCV-71-18 are fully open **THEN ENSURE CLOSED** 2-FCV-71-19, RCIC CST SUCTION VALVE. Standard: Manually closes 2-FCV-71-19.	**Critical Step** ______ SAT ______UNSAT ______N/A
Examiner Cue: “Another operator will continue this procedure”.	

STOP TIME: ______

Provide to Applicant

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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 (BOP) on Unit 2. A Reactor SCRAM has occurred with the following conditions:

- The Nuclear Unit Senior Operator (NUSO) has entered 2-EOI-1, RPV Control Modes 1-3
- Reactor Core Isolation Cooling (RCIC) is in service in accordance with 2-EOI-Appendix-5C, Injection System Lineup RCIC
- Reactor Pressure is being maintained by the Bypass Valves

INITIATING CUES:

The Unit Supervisor directs you to align the RCIC Suction to the Suppression Pool in accordance with 2-EOI-Appendix-5C.

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Place HPCI in Pressure Control per EOI-Appendix-11C, HPCI in Pressure Control and Respond to HPCI Steam Leak
JPM NUMBER:	627A	REVISION:	1

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-55 / Lineup Alternate RPV Pressure Control Systems – HPCI Test Mode in Accordance with EOI Appendix 11C		
K/A RATINGS:	RO 4.7 SRO 4.0		
K/A STATEMENT:	206000 HPCI High-Pressure Coolant Injection System A2.10; Ability to predict the impacts of the following on the HPCI system and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System Isolation		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	4		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	4 minutes		

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*

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 627A

RO _______ SRO _______ DATE: ___________

TASK STANDARD: The Examinee is expected to place HPCI in Reactor Pressure control and respond to incomplete isolation on a HPCI Steam Leak.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.
OF-5 A solid understanding of plant design, engineering, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 2-EOI Appendix-11C, 2-ARP-9-3F, 2-AOI-64-2B

VALIDATION TIME: 8 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
2	11/02/21	ALL	JPM Update

Procedure Revisions

Procedure	Revision
2-EOI Appendix-11C	8
2-ARP-9-3F	41
2-AOI-64-2B	17

SIMULATOR SETUP

IC	N/A
Exam IC	259

Console Operator Instructions	• Reset to IC 259 • Place the simulator in RUN to ensure stable conditions • Run schedule file 2204 NRC JPM c UNIT 2.SCH • Verify event file 2204 NRC JPM c UNIT 2.EVT

Malfunctions	Description	Event	Severity	Delay	Initial set
HP09	HPCI ISOLATION VALVE AUTO CLOSE FAILURE (FCV-73-2,3)	N/A	N/A	N/A	N/A
HP08	HPCI STEAM LEAK INTO HPCI ROOM	1	100	90	N/A

Schedule File(s): **2204 NRC JPM c UNIT 2.SCH**
Event File(s): **2204 NRC JPM c 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. A Reactor SCRAM has occurred with the following conditions:

- The Nuclear Unit Senior Operator (NUSO) has entered 2-EOI-1, RPV Control Modes 1-3
- Main Steam Isolation Valves are closed
- Another operator is controlling Reactor Water Level in accordance with 2-EOI-Appendix-5C, Injection System Lineup RCIC

INITIATING CUE:

The NUSO has directed you to place HPCI in Reactor Pressure Control and maintain 800 – 1000 psig in accordance with 2-EOI-Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode.

START TIME _______

STEP / STANDARD	SAT / UNSAT
Step 1: 2-EOI-Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode **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 140º F may result in equipment damage. [1] **IF** Suppression Pool Level drops below 12.75 ft, **THEN TRIP** HPCI and **CONTROL** RPV Pressure using other options. Expected Action(s): Verifies Suppression Pool Water Level >12.75 ft and makes note of the Suppression Pool Water Level requirement.	_____SAT _____UNSAT _____N/A
Step 2: [2] **IF** Emergency RPV Depressurization is required, **OR** Steam Cooling is required, **THEN EXECUTE** EOI Appendix 16C and 16D as necessary to bypass HPCI Low RPV Pressure and Test Mode Isolation Interlocks. Expected Action(s): Determines step is N/A.	_____SAT _____UNSAT _____N/A
Step 3: [3] **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 Level Suction Transfer Interlock. Expected Action(s): Verifies Suppression Pool Water Level is >5.25 inches and makes note of the Suppression Pool Water Level requirement.	_____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **IF** HPCI Turbine is operating, **THEN PERFORM** the following to trip HPCI: (Otherwise, N/A) A. **ENSURE** HPCI is no longer needed for RPV injection. B. **CHECK** HPCI Auto initiation signals are clear. C. **DEPRESS** 2-XS-73-59, HPCI AUTO-INIT RESET pushbutton. D. **DEPRESS** and HOLD 2-HS-73-18A, HPCI TURBINE TRIP until Step 1.0[4]. F. E. **CLOSE** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE. F. **WHEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE indicates CLOSED, **THEN RELEASE** 2-HS-73-18A, HPCI TURBINE TRIP. G. **CLOSE** 2-FCV-73-44, HPCI PUMP INJECTION VALVE. Expected Action(s): Determines that HPCI is not operating and marks this step as N/A.	_____SAT _____UNSAT _____N/A
Driver Note: If contacted as Unit 1 or 3 to start a Standby Gas Train (SGT), insert events as follows: • **SGT A – Event 1** • **SGT B – Event 2** • **SGT C – Event 3** **Once a SGT has been started, inform the candidate "A/B/C (as appropriate) SGT has been started".**	
Step 5: [5] START HPCI as follows: A. **ENSURE** at least one Standby Gas Train (SGT) is in operation. B. **ENSURE** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller in AUTO and set for 5300 gpm. Expected Action(s): Verifies at least one SGTS train is in operation and that HPCI flow controller is in AUTO and set for 5300 gpm.	_____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 6: **NOTE** HPCI Auxiliary Oil Pump will NOT start UNTIL 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, starts to open. C. **PLACE** 2-HS-73-47A, HPCI AUXILIARY OIL PUMP handswitch in START. Expected Action(s): Places HPCI AUXILIARY OIL PUMP handswitch in START.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 7: D. **PLACE** 2-HS-73-10A, HPCI STEAM PACKING EXHAUSTER, in START. Expected Action(s): Places HPCI STEAM PACKING EXHAUSTER handswitch in START.	_____SAT ______UNSAT ______N/A
Step 8: E. **OPEN** the following valves: • 2-FCV-73-36, HPCI/RCIC CST TEST VALVE • 2-FCV-73-35, HPCI PUMP CST TEST VALVE • 2-FCV-73-30, HPCI PUMP MINIMUM FLOW VALVE **EXAMINER NOTE: Opening 2-FCV-73-30, HPCI PUMP MINIMUM FLOW VALVE is NOT a Critical Step** Expected Action(s): Opens 2-FCV-73-36, 2-FCV-73-35, and 2-FCV-73.30.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 9: F. **OPEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start the HPCI Turbine. Expected Action(s): Opens 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 10: G. **ENSURE** HPCI Auxiliary Oil Pump starts and turbine accelerates above 2400 rpm. Expected Action(s): Verifies HPCI turbine speed accelerated above 2400 rpm.	_____SAT ______UNSAT ______N/A
Step 11: [6] **CHECK** proper HPCI minimum flow valve operation as follows: A. **IF** HPCI flow is above 1200 gpm, **THEN CHECK CLOSED** 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE. B. **IF** HPCI flow is below 600 gpm, **THEN CHECK OPEN** 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE. Expected Action(s): If initially opened, verifies that the HPCI Minimum Flow Valve closes as applicable.	_____SAT ______UNSAT ______N/A
EXAMINER NOTE: JPM Step 12 (next page) - Failure Criteria is met if HPCI Pump Discharge Instantaneous Pressure reaches 1500 psig.	

STEP / STANDARD	SAT / UNSAT
Step 12: 7. **THROTTLE** 2-FCV-73-35, HPCI PUMP CST TEST VLV, to control HPCI pump discharge pressure at or below 1100 psig. 8. **ADJUST** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller to Control RPV pressure Expected Action(s): Throttles and adjusts CST test valve and HPCI Flow Controller as necessary to maintain HPCI Pump Discharge Pressure at or below 1100 psig and control Reactor Pressure.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER NOTE: Begin Alternate Path – when satisfied with the candidate's actions of placing HPCI in Reactor Pressure Control, request that the Simulator Booth Operator insert Event 1 to cause a steam leak in the HPCI Room. **The candidate may choose to respond in accordance with the ARP or the AOI. The actions for both are outlined following this step. See Step [14] for ARP actions, and Step [15] for AOI actions.**	
Driver Note: When requested by the Examiner, insert Event 1 to cause a steam leak in the HPCI Room.	
Step 14: **HPCI STEAM LINE FLOW EXCESSIVE, 2-ARP-9-3F (window 18)** **EXAMINER NOTE: In accordance with OPDP-1, Conduct of Operations, the candidate will take prompt manual action to accomplish the desired function of malfunctioned automatic actions.** Automatic Action: A. 2-FCV-73-18, HPCI TURBINE STOP VALVE closes. B. 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE closes C. The following HPCI steam supply valves close: • 2-FCV-73-2, HPCI STEAM LINE INBOARD ISOLATION VALVE • 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE • 2-FCV-73-81, HPCI STEAM LINE WARM-UP VALVE D. The following HPCI Suppression Pool suction valves close: • 2-FCV-73-26, HPCI SUPPR POOL INBOARD SUCTION	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
VALVE • 2-FCV-73-27, HPCI SUPPR POOL OUTBOARD SUCTION VALVE E. The following amber lights indicating auto isolation seal-in will illuminate: • 2-IL-73-58A, HPCI AUTO ISOL LOGIC A • 2-IL-73-58B, HPCI AUTO ISOL LOGIC B Expected Action(s): Determines 2-FCV-73-2 did not close. Closes 2-FCV-73-2. Determines 2-FCV-73-3 did not close. Closes 2-FCV-73-3.	
EXAMINER CUE: Another Operator will continue this procedure.	
Step 15: OPERATOR ACTION: A. **IF** annunciation is valid, **THEN REFER TO** 2-AOI-64-2b. B. **REFER TO** Tech Spec 3.5.1, and 3.3.6.1 Expected Action(s): Refers to 2-AOI-64-2b. Notifies the NUSO of Tech Spec 3.5.1 and 3.3.6.1. **EXAMNIER NOTE: Acknowledge any Tech Spec information provided by the candidate.**	_____SAT ______UNSAT ______N/A
Step 16: **2-AOI-64-2B, Group 4 High Pressure Coolant Injection Isolation** **4.1 Immediate Actions** [1] **ENSURE** automatic actions occur. Expected Action(s): Determines 2-FCV-73-2 did not close. Closes 2-FCV-73-2. Determines 2-FCV-73-3 did not close. Closes 2-FCV-73-3.	**Critical Step** __ SAT ___UNSAT ___N/A
EXAMINER CUE: Another Operator will continue this procedure.	

STOP TIME ______

Provide to Applicant

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INITIAL CONDITIONS:

You are an Operator on Unit 2. A Reactor SCRAM has occurred with the following conditions:

- The Nuclear Unit Senior Operator (NUSO) has entered 2-EOI-1, RPV Control Modes 1-3
- Main Steam Isolation Valves are closed
- Another operator is controlling Reactor Water Level in accordance with 2-EOI-Appendix-5C, Injection System Lineup RCIC

INITIATING CUE:

The NUSO has directed you to place HPCI in Reactor Pressure Control and maintain 800 – 1000 psig in accordance with 2-EOI-Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode.

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Vent the Drywell per OI-64, Primary Containment System
JPM NUMBER:	750	REVISION:	0

TASK APPLICABILITY:	☐ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-064-NO-03 / Vent the Drywell with Standby Gas Treatment Fan		
K/A RATINGS:	RO 4.4 SRO 4.3		
K/A STATEMENT:	223001 PCS Primary Containment Systems and Auxiliaries A2.07; Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System Isolation: High Drywell Pressure		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	5		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	6 minutes		

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*

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 750

RO _______ SRO _______ DATE: ___________

TASK STANDARD: The Examinee is expected to perform actions necessary to reduce Drywell Pressure by venting in accordance with plant Operating Procedures.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 2-OI-64-1

VALIDATION TIME: 6 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	11/03/21	ALL	Initial issue

Procedure Revisions

Procedure	Revision
2-OI-64	129

SIMULATOR SETUP

IC	N/A
Exam IC	260

Console Operator Instructions	- Reset to IC 260 - Place the simulator in RUN to ensure stable conditions - Verify that the candidate has been pre-briefed on the procedure

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A					

Schedule File(s): **N/A**
Event File(s): **N/A**

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INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following conditions:

- Drywell Pressure has risen to nearly 1.50 psig and must be lowered
- Prestartup/Standby Readiness Requirements of 2-OI-64, Primary Containment System, Section 4.0 is complete
- Stack Dilution Fans are in operation in accordance with 2-OI-66, Off-Gas System

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to vent the Drywell in accordance with 2-OI-64, Primary Containment System.

Note: Another Operator is standing by for data logging (as necessary)

START TIME _______

STEP / STANDARD	SAT / UNSAT
Step 1: 2-OI-64, Primary Containment System Section 6.1 Venting the Drywell with Standby Gas Treatment Fan Section 6.1.1 Venting Lineup [1] **REVIEW** all Precautions and Limitations. **REFER TO** Section 3.0 Standard: This step should have been completed during the pre-brief, but the candidate may review the Precautions and Limitations.	_____SAT _____UNSAT _____N/A
Step 2: [2] **ENSURE** all Prestartup/Standby Readiness requirements in Section 4.0 are satisfied. Standard: As this information is given in the Initial Conditions, the candidate marks this step as complete.	_____SAT _____UNSAT _____N/A
Step 3: [3] **ENSURE** Stack Dilution Fans in Operation Standard: As this information is given in the Initial Conditions, the candidate marks this step as complete.	_____SAT _____UNSAT _____N/A
Step 4: [4] **ENSURE** Group 6 Isolation Signal (Ventilation Systems) **NOT** present. Standard: Verifies that a Group 6 Isolation Signal is not present.	_____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 5: [5] **ENSURE** CLOSED 2-FCV-64-29, DRYWELL VENT INBOARD ISOLATION VALVE using 2-HS-64-29. Standard: Verifies 2-FCV-64-29 closed.	_____SAT _____UNSAT _____N/A
Step 6: [6] **ENSURE** CLOSED 2-FIC-84-19, PATH B VENT FLOW CONTROL (Panel 2-9-55). Standard: Verifies 2-FIC-84-19 closed.	_____ SAT _____UNSAT _____N/A
Step 7: [7] **ENSURE** 2-FIC-84-20, PATH A VENT FLOW CONTROLLER, in AUTO and set at 100 SCFM (Panel 2-9-55). Standard: Ensures 2-FIC-84-20 is in AUTO and set to 100 SCFM.	_____ SAT _____UNSAT _____N/A
Step 8: [8] **IF** the Drywell DP Compressor is in operation, **THEN STOP** the compressor using 2-HS-64-142A, DRYWELL DP COMPRESSOR AND VALVES CONTROL. (Otherwise N/A) Standard: Stops the Drywell Compressor if necessary.	_____ SAT _____UNSAT _____N/A
Step 9: [9] **ENSURE CLOSED** 2-FCV-64-139, DRYWELL DP COMPRESSOR SUCTION ISOLATION VALVE, using 2-ZI-64-139. Standard: Ensures 2-FCV-64-139 is closed.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 10: [10] **NOTIFY** Unit 1 and 3 Control Room that Unit 3 Drywell venting with Standby Gas is about to start. Standard: **EXAMINER CUE: Acknowledge any reports or information provided by the candidate.** Informs the Unit 1 and 3 Control Rooms that Unit 2 will be venting the Drywell.	_____SAT _____UNSAT _____N/A
Step 11: **NOTE** 1) A Standby Gas Treatment Fan must be running for 2-FCV-84-20 to operate. 2) Depending upon plant conditions, it is possible that flow may indicate less than 100 SCFM even with the thumbwheel of 2-FIC-84-20 on Panel 2-9-55 adjusted to the fully open position. 3) Drywell Pressure is usually controlled to maintain the Drywell and Suppression Chamber Differential Pressure between 1.15 and 1.30 psid to provide a margin to the Tech Spec limit. **CAUTION** Path A and B Vent valves, 2-FCV-84-19 and 2-FCV-84-20, isolate on Standby Gas duct high pressure of 1 psig. [11] **PLACE** a Standby Gas Treatment Train in service. Standard: The candidate may either contact another Unit to have a Standby Gas Train (SGT) started or start ‘C’ SGT at Panel 2-9-25.	_____SAT _____UNSAT _____N/A
Driver Note: If the candidate requests that a Standby Gas Train be started, insert Event 1 to start ‘A’ SGT.	

STEP / STANDARD	SAT / UNSAT
Step 12: Section 6.12 Venting Drywell [1] **ENSURE** Section 6.1.1 has been completed. (N/A if venting has not been completely secured.) [2] **RECORD** initial data for Drywell Venting in 3-SI-4.7.A.2.a, Primary Containment Consumption and Leakage, if required. Standard: Ensure Section 6.1.1 has completed.	______ SAT ______UNSAT ______N/A
Step 14: [3] **CLOSE**, 2-FCV-64-34, using 2-HS-64-34, SUPPRESSION CHAMBER INBOARD ISOLATION VALVE. Standard: Closes 2-FCV-64-34.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 15: [4] **ENSURE** OPEN 2-FCV-64-31 using 2-HS-64-31, DRYWELL INBOARD ISOLATION VALVE. Standard: Ensures 2-FCV-64-31 is open.	_____SAT ______UNSAT ______N/A
Step 16: [5] **OPEN** 2-FCV-84-20 using 2-HS-64-35, CONTROL DRYWELL/SUPPRESSION CHAMBER VENT. Standard: Opens 2-FCV-84-20.	**Critical Step** ___ SAT ___UNSAT ___N/A

STEP / STANDARD	SAT / UNSAT
Step 17: [6] **ENSURE** flow at approximately 100 SCFM on PATH A VENT FLOW CONTROLLER 2-FIC-84-20 (Panel 2-9-55). [7] **MONITOR** Drywell Pressure. Standard: Monitors flow on 2-FIC-84-20 and Drywell Pressure.	_____SAT _____UNSAT _____N/A
EXAMINER CUE: Another Operator will continue this procedure.	

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.

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INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following conditions:

- Drywell Pressure has risen to nearly 1.50 psig and must be lowered
- Drywell – Suppression Pool D/P is abnormal
- Prestartup/Standby Readiness Requirements of 2-OI-64, Primary Containment System, Section 4.0
- Stack Dilution Fans are in operation in accordance with 2-OI-66, Off-Gas System

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to vent the Drywell in accordance with 2-OI-64, Primary Containment System.

Note: Another Operator is standing by for data logging (as necessary)

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Transfer 2C 4KV Unit Board from the Start Bus to the USST Per 0-OI-57A, Switchyard And 4160V AC Electrical System
JPM NUMBER:	725	REVISION:	1

TASK APPLICABILITY:	☐ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-57A-NO-01 / Perform Control Room Transfer of 4KV Unit Board Power Supplies		
K/A RATINGS:	RO / SRO: 3.7		
K/A No. &STATEMENT:	262001 A.C. Electrical Distribution A4.01: Ability to manually operate and/or monitor in the Control Room: Breakers and disconnects		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	6		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	6 minutes		

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*

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: ______________________________ JPM Number: 725

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to transfer 4KV Unit Board 2C from Start Bus to USST in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 0-OI-57A

VALIDATION TIME: 6 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	09/06/18	All	Initial issue
1	11/04/21	All	JPM Update

Procedure Revisions

Procedure	Revision
0-OI-57A	172

SIMULATOR SETUP

IC	28
Exam IC	261

Console Operator Instructions	• **Reset to IC 261** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 0-OI-57A, Switchyard and 4160V AC Electrical System, Section 8.14.2.**

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 the Balance of Plant Operator on Unit 2. 2C 4KV Unit Board was transferred to 1A Start Bus for maintenance on 4KV UNIT BD 2C NORM FDR BKR 1216. Maintenance has been completed.

INITIATING CUES:

The Unit Supervisor directs you to transfer 2C 4KV Unit Board from the Start Bus to the USST in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System, Section 8.14.2.

START TIME:__________

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 0-OI-57A, Switchyard and 4160V AC Electrical System. Step 1: [1] Review all Precautions and Limitation in Section 3.0. Standard: Completed during pre-brief, but the candidate may review Precautions and Limitations.	_____ SAT _____UNSAT _____N/A
NOTE TO DRIVER/EXAMINER: Acknowledge as NSS/Security of the possible loss of power to Security Systems while transferring 4KV Unit Board 2C Power Supplies.	
Step 2: [2] Notify NSS of possible loss of power to Security Systems prior to transferring 4KV Unit Board 2C. Standard: Informs NSS/Security of the possible loss of power to Security Systems while transferring 4KV Unit Board 2C to the USST.	_____ SAT _____UNSAT _____N/A
NOTE All procedural steps are performed from Control Room Panel 2-9-8, unless specified. Step 3: [2.1] **PLACE** 2-XS-202-1, 4KV BD/BUS/XFMR VOLTAGE SELECT switch to USST 2A. Standard: Places 4KV BD/BUS/XFMR VOLTAGE SELECT switch to USST 2A.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
<u>Step 4:</u> [2.2] **CHECK** USST 2A Voltage on 2-EI-57-28, 4KV BOARD VOLTS, is between 3950 and 4400 Volts. <u>Standard:</u> Verifies that USST 2A Voltage is between 3950 and 4400 Volts.	______ SAT ______UNSAT ______N/A
<u>Step 5:</u> [2.3] **PLACE** and **HOLD** 2-HS-57-9, 4KV UNIT BD 2C NORMAL FEEDER BREAKER 1216 to CLOSE. <u>Standard:</u> Places and holds 2-HS-57-9, 4KV UNIT BD 2C NORMAL FEEDER BREAKER 1216 in the CLOSE position.	**Critical Step** ______ SAT ______UNSAT ______N/A
<u>Step 6:</u> [2.4] **PLACE** 2-HS-57-11, 4KV UNIT BD 2C ALTERNATE FEEDER BREAKER 1426 to TRIP. <u>Standard:</u> Places 2-HS-57-11, 4KV UNIT BD 2C ALTERNATE FEEDER BREAKER 1426, in TRIP.	**Critical Step** ______ SAT ______UNSAT ______N/A
<u>Step 7:</u> [2.5] **CHECK CLOSED** the 4KV UNIT BD 2C NORMAL FEEDER BREAKER 1216. [2.6] **CHECK OPEN** 4KV UNIT BD 2C ALTERNATE FEEDER BREAKER 1426. <u>Standard:</u> Checks closed 4KV UNIT BOARD 2C NORMAL FEEDER BREAKER 1216 and checks open the ALTERNATE FEEDER BREAKER 1426.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 8: [2.7] **RELEASE** BKRs 1216 and 1426 switches. Standard: Releases hand switches for breakers 1216 and 1426.	_____ SAT _____UNSAT _____N/A
Step 9: [2.8] **PLACE** 2-XS-202-1, 4KV BD/BUS/XFMR VOLTAGE SELECT switch to UNIT BD 2C. Standard: Places 2-XS-202-1, 4KV BD/BUS/XFMR VOLTAGE SELECT switch to UNIT BD 2C.	_____ SAT _____UNSAT _____N/A
Step 10: [2.9] **CHECK** 4KV UNIT BOARD 2C Voltage greater than 3950 and less than 4400 volts. Standard: Verifies 4KV Unit Board 2C Voltage is between 3950 and 4400 Volts.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: Once JPM Step 10 is completed, inform the candidate "Another Operator will continue 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 the Balance of Plant Operator on Unit 2. 2C 4KV Unit Board was transferred to 1A Start Bus for maintenance on 4KV UNIT BD 2C NORM FDR BKR 1216. Maintenance has been completed.

INITIATING CUES:

The Unit Supervisor directs you to transfer 2C 4KV Unit Board from the Start Bus to the USST in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System, Section 8.14.2.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Perform 2-SR-3.3.1.1.8(11) Reactor Protection System Manual SCRAM Functional Test
JPM NUMBER:		290A	REVISION: 2

TASK APPLICABILITY:	☐ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-085-AL-12 / Respond to Control Rod Drift		
K/A RATINGS:	RO: 4.2 SRO: 3.9		
K/A No. &STATEMENT:	212000 Reactor Protection System A2.19: 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: Partial RPS actuation		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	7		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	12 minutes		

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: 290A

RO _____ SRO _____ DATE: _____________

TASK STANDARD: The Examinee is expected to perform 2-SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test, on RPS A and B and recognize that when RPS B is tested, two Control Rods SCRAM, requiring the candidate to insert a manual Reactor SCRAM.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.
OF-2 Operating the Plant with a Conservative Bias.

REFERENCES/PROCEDURES NEEDED: 2-SR-3.3.1.1.8(11), 2-AOI-85-5, 2-AOI-100-1

VALIDATION TIME: 12 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	09/18/18	All	JPM update; updated format
2	11/04/21	All	JPM Update

Procedure Revisions

Procedure	Revision
2-SR-3.3.1.1.8(11)	5
2-AOI-85-5	24
2-AOI-100-1	118

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• **Ensure the candidate has been pre-briefed on 2-SR-3.3.1.1.8(11)** • **Reset to IC 28** • **Load and run schedule file ILT 2204 NRC JPM – f – 290A.sch** • **Ensure Event File ILT 2204 NRC JPM – f – 290A.evt starts when the schedule file is started** • **Ensure Desk Operator ICS Screen is open to 3.3.1.4.12 (Group → OPS-SI → Group 3311412)** • **Place the simulator in RUN to ensure stable conditions**

Malfunctions	Description	Event	Severity	Delay	Initial set
RD10R3443	INDIVIDUALLY SCRAM ANY CONTROL ROD	1	N/A	N/A	N/A
RD10R3435	INDIVIDUALLY SCRAM ANY CONTROL ROD	1	N/A	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:

Unit 2 is operating at 100% Power, with no equipment out of service. You are the Operator at the Controls on Unit 2.

INITIATING CUES:

The Unit Supervisor directs you to perform 2-SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test, starting at Step [1] of Section 6.2, RPS Manual SCRAM Channel A Functional Test.

START TIME:___________

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 2-SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test, before commencing the JPM.	
Step 1: 6.2 RPS Manual SCRAM Channel A Functional Test [1] **MOMENTARILY DEPRESS** 2-HS-99-5A/S3A**,** REACTOR SCRAM A, Panel 2-9-5. Standard: Momentarily depresses 2-HS-99-5A/S3A**,** REACTOR SCRAM A, Panel 2-9-5.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 2: [2] **VERIFY** the following on Panel 2-9-5: • Control Rod Logic Reset Solenoid Group A indicating lights (4) extinguished • One 2-IL-99-5A/AB, SYSTEM A BACK-UP SCRAM VALVE LIGHT, and 2-IL-99-5A/CD, SYSTEM B BACK-UP SCRAM VALVE LIGHT, extinguished • 2-HS-99-5A/S3A, REACTOR SCRAM A push-button illuminated. • Annunciator 2-9-5B, Window 8, REACTOR CHANNEL A MAN SCRAM (2-XA-55-5B) in alarm Standard: Verifies the conditions listed above are met.	______ SAT ______UNSAT ______N/A
Step 3: [3] **VERIFY** ICS point SOE033 indicates TRIP. Standard: Verifies ICS Point SOE033 indicates TRIP.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **MOMENTARILY PLACE** 2-HS-99-5A-S5, SCRAM RESET, Panel 2-9-5, to GP 1/4 position. Standard: Places 2-HS-99-5A-S5, SCRAM RESET, to GP 1/4 position.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 5: [5] **VERIFY** the following on Panel 2-9-5: • Control Rod Logic Reset Solenoid Group A lights 1 and 4 illuminated, and lights 2 and 3 extinguished • Both System A Backup SCRAM Valve lights 2-IL-99-5A/AB illuminated • One System B Backup SCRAM Valve light 2-IL-99-5A/CD illuminated, and one System B Backup SCRAM Valve light 2-IL-99-5A/CD extinguished • Annunciator 2-9-5B, Window 8, REACTOR CHANNEL A MAN SCRAM (2-XA-55-5B) will **NOT** reset • 2-HS-99-5A/S3A, REACTOR SCRAM A push button illuminated Standard: Verifies the conditions listed above are met.	______ SAT ______UNSAT ______N/A
Step 6: [6] **MOMENTARILY DEPRESS** 2-HS-99-5A/S3A**,** REACTOR SCRAM A, Panel 2-9-5. Standard: Momentarily depresses 2-HS-99-5A/S3A**,** REACTOR SCRAM A, Panel 2-9-5.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [7] **VERIFY** the following on Panel 2-9-5: • Control Rod Logic Reset Solenoid Group A lights 1 and 4 extinguished • One 2-IL-99-5A/AB, System A Backup SCRAM Valve light, extinguished Standard: Verifies the conditions listed above are met.	______ SAT ______UNSAT ______N/A
Step 8: [8] **MOMENTARILY PLACE** 2-HS-99-5A-S5, SCRAM RESET, Panel 2-9-5, to GP 2/3 position. Standard: Places 2-HS-99-5A-S5, SCRAM RESET, to GP 2/3 position.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 9: [9] **VERIFY** the following on Panel 2-9-5: • Control Rod Logic Reset Solenoid Group A lights 2 and 3 illuminated, and lights 1 and 4 extinguished • Both 2-IL-99-5A/CD, SYSTEM B BACKUP SCRAM VALVE, lights illuminated • One 2-IL-99-5A/AB, SYSTEM A BACKUP SCRAM VALVE light illuminated, and one 2-IL-99-5A/AB, SYSTEM A BACKUP SCRAM VALVE light extinguished • Annunciator 2-9-5B, Window 8, REACTOR CHANNEL A MAN SCRAM (2-XA-55-5B), will not reset • 2-HS-99-5A/S3A, REACTOR SCRAM A push button illuminated Standard: Verifies the conditions listed above are met.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 10: [10] **VERIFY** ICS point SOE033 indicates NOTTRIP. Standard: Verifies ICS Point SOE033 indicates NOTTRIP.	______ SAT ______UNSAT ______N/A
Step 11: [11] **MOMENTARILY PLACE** 2-HS-99-5A-S5, SCRAM RESET, Panel 2-9-5 to GP 1/4 position. Standard: Momentarily places 2-HS-99-5A-S5, SCRAM RESET, Panel 2-9-5, to GP 1/4 position.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 12: [12] **VERIFY** the following on Panel 2-9-5: • Control Rod Logic Reset Solenoid Group A red indicating lights (4) illuminated • Control Rod Logic Reset Solenoid Group B red indicating lights (4) illuminated • 2-IL-99-5A/AB, SYSTEM A BACK-UP SCRAM VALVE, lights illuminated • 2-IL-99-5A/CD, SYSTEM B BACK-UP SCRAM VALVE, lights illuminated • 2-HS-99-5A/S3A, REACTOR SCRAM A, push-button extinguished • Annunciator 2-9-5B, Window 8, REACTOR CHANNEL A MAN SCRAM (2-XA-55-5B) reset Standard: Verifies the conditions listed above are met.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 13: [13] **VERIFY** the following computer points indicate NOTTRIP. - ICS point SOE033 - ICS point SOE034 Standard: Verifies ICS Point SOE033 and SOE034 indicate NOTTRIP.	______ SAT ______UNSAT ______N/A
EXAMINER NOTE: Beginning of Alternate Path. When 2-HS-99-5A/S3B, Reactor SCRAM B, is depressed in the step [6.3], Control Rods 34-43 and 34-35 will SCRAM.	
Step 14: 6.3 RPS Manual SCRAM Channel B Functional Test [1] **MOMENTARILY DEPRESS** 2-HS-99-5A/S3B, Reactor SCRAM B, push-button, Panel 2-9-5. Standard: Depresses 2-HS-99-5A/S3B, Reactor SCRAM B. Determines that two Control Rods have SCRAMMED.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 15: 2-AOI-85-5, Rod Drift In Immediate Action: [1] If multiple Rods are drifting into the core, **THEN MANUALLY SCRAM** the Reactor. Refer to 2-AOI-100-1, Reactor SCRAM. Standard: Inserts a Manual Reactor SCRAM by depressing 2-HS-99-5A/S3A and 2-HS-99-5A/S3B, Reactor SCRAM A and B on Panel 2-9-5.	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER CUE: Once JPM Step 15 is completed, inform the candidate "Another Operator will continue the Reactor SCRAM 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:

Unit 2 is operating at 100% Power, with no equipment out of service. You are the Operator at the Controls on Unit 2.

INITIATING CUES:

The Unit Supervisor directs you to perform 2-SR-3.3.1.1.8(11), Reactor Protection System Manual SCRAM Functional Test, starting at Step [1] of Section 6.2, RPS Manual SCRAM Channel A Functional Test.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Align CAD to Drywell Control Air per EOI Appendix-8G, Crosstie CAD to Drywell Control Air
JPM NUMBER:	39	REVISION:	2

TASK APPLICABILITY:	☐ SRO-U	☐ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-74 / Coordinate Crosstieing CAD Trains A and B to Drywell Control Air in Accordance with EOI Appendix 8G		
K/A RATINGS:	RO / SRO 3.4		
K/A No. &STATEMENT:	295019 Partial or Complete Loss of Instrument Air AA1.01: Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Backup air supply		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	8		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	10 minutes		

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: 39

RO _______ SRO _______ DATE: __________

TASK STANDARD: The Examinee is expected to perform operations to align Containment Air Dilution systems A and B to the Drywell Air System.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 2-EOI-APPENDIX-8G

VALIDATION TIME: 10 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	08/14/18	All	Converted JPM to new format/procedure update
2	11/04/21	All	JPM Update

Procedure Revisions

Procedure	Revision
2-EOI-Appendix 8G	7

SIMULATOR SETUP

IC	N/A
Exam IC	263

Console Operator Instructions	• **Reset to IC 263** • **Place the simulator in RUN to ensure stable conditions**

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. The Reactor has SCRAMMED. 2-EOI-1, RPV Control Modes 1-3, has been followed to RC/P-3.

INITIATING CUE:

The Unit Supervisor has directed you to perform 2-EOI-Appendix-8G, Crosstie CAD to Drywell Control Air**.**

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **OPEN** the following valves: • 0-FCV-84-5, CAD SYSTEM A N2 SHUTOFF VALVE (Unit 1, Panel 1-9-54) • 0-FCV-84-16, CAD SYSTEM B N2 SHUTOFF VALVE (Unit 1, Panel 1-9-55). Standard: Opens 0-FCV-84-5, CAD SYSTEM A N2 SHUTOFF VALVE, and 0-FCV-84-16, CAD SYSTEM B N2 SHUTOFF VALVE.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [2] **ENSURE** 0-PI-84-6, N2 VAPORIZER A OUTLET PRESSURE, and 0-PI-84-17, N2 VAPORIZER B OUTLET PRESSURE, indicate approximately 100 PSIG (Unit 1, Panel 1-9-54 and1-9-55). Standard: Verifies 0-PI-84-6, N2 VAPORIZER A OUTLET PRESSURE, and 0-PI-84-17, N2 VAPORIZER B OUTLET PRESSURE, (located on back of Unit 2 Panel 9-54) indicate approximately 100 PSIG.	_____ SAT _____UNSAT _____N/A
Step 3: [3] **PLACE** keylock switch 2-HS-84-48, CAD A CROSS TIE TO DRYWELL CONTROL AIR, in OPEN (Unit 2, Panel 2-9-54). Standard: Places 2-HS-84-48, CAD A CROSS TIE TO DRYWELL CONTROL AIR, in open.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **CHECK OPEN** 2-FSV-84-48, CAD A CROSS TIE TO DRYWELL CONTROL AIR, (Unit 2, Panel 2-9-54). Standard: Checks open 2-FSV-84-48, CAD A CROSS TIE TO DRYWELL CONTROL AIR.	______ SAT ______UNSAT ______N/A
Step 5: [5] **IF** unable to open 2-FSV-84-48, then **PERFORM** the following at the west side header (RB el 580, top of clean room, west side) (otherwise N/A): Standard: Marks this step as N/A.	______ SAT ______UNSAT ______N/A
Step 6: [6] **PLACE** keylock switch 2-HS-84-49, CAD B CROSS TIE TO DRYWELL CONTROL AIR, in OPEN (Unit 2, Panel 2-9-55). Standard: Places 2-HS-84-49, CAD B CROSS TIE TO DRYWELL CONTROL AIR, in open.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 7: [7] **CHECK OPEN** 2-FSV-84-49, CAD B CROSS TIE TO DRYWELL CONTROL AIR (Unit 2, Panel 2-9-55). Standard: Checks open 2-FSV-84-49, CAD B CROSS TIE TO DRYWELL CONTROL AIR.	_____SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 8: [8] **IF** unable to open 2-FSV-84-49, then **PERFORM** the following at the east side header (RB el 580, top of clean room, NE corner) (otherwise N/A): Standard: Marks this step as N/A.	_____SAT _____UNSAT _____N/A
Step 9: [9] **CHECK** MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW, 2-PA-32-31, alarm cleared (2-XA-55-3D, Window 18). Standard: Checks MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW (2-XA-55-3D, Window 18) alarm clear.	_____SAT _____UNSAT _____N/A
Cue: “Another Operator will monitor CAD System Operation”.	

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 has SCRAMMED. 2-EOI-1, RPV Control Modes 1-3, has been followed to RC/P-3.

INITIATING CUE:

The Unit Supervisor has directed you to perform 2-EOI-Appendix-8G, Crosstie CAD to Drywell Control Air.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Emergency Venting Primary Containment per 2-EOI-Appendix 13
JPM NUMBER:	55A	REVISION:	2

TASK APPLICABILITY:	☐ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-63 / Emergency Vent Primary Containment in Accordance with EOI Appendix 13		
K/A RATINGS:	RO 3.5 SRO 3.6		
K/A STATEMENT:	288000A2.01 K/A 288000 PVS 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 operations: High Drywell Pressure		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	5		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	10 minutes		

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: 55A

RO _______ SRO _______ DATE: __________

TASK STANDARD: The Examinee is expected to vent Primary Containment to lower Drywell Pressure.

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: 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
1	3/29/17	All	JPM Update
2	11/08/21	All	JPM Update

Procedure Revisions

Procedure	Revision
2-EOI-Appendix 13	10

SIMULATOR SETUP

IC	N/A
Exam IC	284

Console Operator Instructions	• **Reset to Exam IC 284** • **Place the simulator in RUN to ensure stable conditions** • **When the candidate requests the jumper for 2-FCV-64-30 be installed, insert event 1.**

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A					

Remotes	Description	Event	Severity	Delay	Initial set
PC04	Bypass Isln on FCV-64-30 to allow Drywell Vent	1	NA	NA	BYPASS

Overrides	Description	Event	Severity	Delay	Initial set
HS-64-221A	Hardened Supr Chbr Vent Inbd Isol Vlv	NA	NA	NA	CLOSE

****

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 large leak inside Primary Containment has developed
- The Reactor has been SCRAMMED
- Another Operator has been assigned to control Reactor Water Level and Reactor Pressure

INITIATING CUE:

The Unit Supervisor directs you to vent Primary Containment in accordance with 2-EOI-Appendix-13, Emergency Venting Primary Containment.

START TIME _____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **NOTIFY** Shift Manager / SED of the following: • Emergency Venting of Primary Containment is in progress • Off-Gas Release Rate Limits will be exceeded Standard: Notifies Shift Manager/SED.	_____ SAT _____UNSAT _____N/A
EMAINER CUE: As Shift Manager/SED acknowledge report that Emergency Primary Containment Venting is in progress and Off-Gas release limits will be exceeded	
Step 2: **NOTES** 1) HARDENED CONTAINMENT VENT VALVES 2-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. [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] Standard: Verifies Suppression Pool level below 26 ft using 2-LI-64-159A and makes note of action required if the Suppression Pool cannot be vented.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [2.2] **PLACE** keylock switch 2-HS-64-222B, HARDENED SUPPRESSION CHAMBER VENT OUTBOARD PERMISSIVE, in PERM Standard: Places 2-HS-64-222B in the PERM position.	______ SAT ______UNSAT ______N/A
Step 4: [2.3] **CHECK** blue indicating light above 2-HS-64-222B, HARDENED SUPPRESSION CHAMBER VENT OUTBOARD PERMISSIVE, illuminated. Standard: Verifies BLUE indicating lamp above 2-HS-64-222B illuminated.	______ SAT ______UNSAT ______N/A
Step 5: [2.4] **OPEN** 2-FCV-64-222, HARDENED SUPPRESSION CHAMBER VENT OUTBOARD ISOLATION VALVE. Standard: Places 2-HS-64-222A in the OPEN position and verifies 2-FCV-64-222 OPEN.	______ SAT ______UNSAT ______N/A
Step 6: [2.5] **PLACE** keylock switch 2-HS-64-221B, HARDENED SUPPRESSION CHAMBER VENT INBOARD PERMISSIVE, in PERM. Standard: Places 2-HS-64-221B in the PERM position.	**Critical Step** ______ SAT ______UNSAT ______N/A

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td><u>Step 7:</u>

[2.6] CHECK blue indicating light above 2-HS-64-221B, HARDENED SUPPRESSION CHAMBER VENT INBOARD PERMISSIVE, illuminated.

<u>Standard:</u>

Verifies BLUE indicating lamp above 2-HS-64-221B illuminated.</td><td>______ SAT
______UNSAT
______N/A</td></tr>
<tr><td colspan="2">EXAMINER NOTE: Alternate path starts with Step 8.</td></tr>
<tr><td><u>Step 8:</u>

[2.7] OPEN 2-FCV-64-221, HARDENED SUPPRESSION CHAMBER VENT INBOARD ISOL VALVE.

<u>Standard:</u>

Places 2-HS-64-221 in the OPEN position and recognizes that the valve did NOT open. The Operator may dispatch an Assistant Unit Operator (AUO) to locally operate 2-FCV-64-221. Reports that the Suppression Chamber cannot be vented and proceeds to Step 9.</td><td>Critical Step
______ SAT
______UNSAT
______N/A</td></tr>
<tr><td colspan="2">EXAMINER CUE: If the Operator dispatches an AUO to locally operate 2-FCV-64-221, report that Elevation 565 of the Reactor Building is not accessible.
(See Note under Step 2)</td></tr>
<tr><td colspan="2">EXAMINER CUE: Acknowledge that the Suppression Chamber cannot be vented and that operator is continuing to Step 9.</td></tr>
<tr><td><u>Step 9:</u>

[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 (RP) that Reactor Building is being evacuated due to imminent failure of Primary Containment vent ducts.
[3.3] EVACUATE <u>ALL</u> Reactor Buildings using P.A. System.

<u>Standard:</u>

Notifies the SM/SED that containment integrity failure is possible, notifies RP that the Reactor Building will be evacuated due to imminent failure of Primary Containment vent ducts, and makes P.A. announcement to evacuate Reactor Building.</td><td>______ SAT
______UNSAT
______N/A</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: As SM/SED acknowledge report of possible containment integrity failure, As RP acknowledge report of Reactor Building evacuation due to imminent failure of Primary Containment vent ducts.	
Step 10: [3.4] **START** ALL available SGTS trains. Standard: Determines that all trains of SGTS are already in service by observing SGT OPERATING lights on 2-9-20.	______ SAT ______UNSAT ______N/A
Step 11: [3.5] **ENSURE CLOSED** 2-FCV-64-36, DRYWELL/SUPPRESSION CHAMBER VENT TO SGT (Panel 2-9-3). Standard: Verifies 2-FCV-64-36 is closed on Panel 9-3.	______ SAT ______UNSAT ______N/A
Step 12: [3.6] **ENSURE OPEN** the following dampers (Panel 2-9-25): • 2-FCO-64-40, REACTOR ZONE EXHAUST TO SGTS • 2-FCO-64-41, REACTOR ZONE EXHAUST TO SGTS Standard: Verifies dampers 2-FCO-64-40 and 2-FCO-64-41 are open on Panel 2-9-25.	______ SAT ______UNSAT ______N/A
Step 13: [3.7] **ENSURE CLOSED 2-**FCV-64-29, DRYWELL VENT INBOARD ISOLATION VALVE (Panel 2-9-3 or Panel 2-9-54). Standard: Verifies 2-FCV-64-29 is closed on Panel 9-3 or Panel 9-54.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 14: [3.8] **DISPATCH** personnel to Unit 3 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 OUTBOARD ISOLATION VALVE, is in place. Standard: Contacts an AUO or an extra operator to perform step [3.8].	**Critical Step** ______ SAT ______UNSAT ______N/A
DRIVER: When called to install jumper for 2-FCV-64-30 • **Insert Event 2** • **Inform the operator that you are using time compression, and report that the jumper for 2-FCV-64-30 is in place.**	
Step 15: [3.9] **ENSURE OPEN** 2-FCV-64-30, DRYWELL VENT OUTBOARD ISOLATION VALVE (Panel 2-9-3). Standard: Verifies open 2-FCV-64-30 on panel 9-3.	______ SAT ______UNSAT ______N/A
Step 16: **CAUTION** 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, SUPPRESSION CHAMBER/DRYWELL VENT ISOLATION BYPASS SELECT, to DRYWELL (Panel 2-9-54). Standard: Places keylock switch 2-HS-84-36 in the DRYWELL position.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 17: [3.11] **ENSURE OPEN 2-**FCV-64-29, DRYWELL VENT INBOARD ISOLATION VALVE (Panel 2-9-54). Standard: Verifies that 2-FCV-64-29 opens on Panel 9-54.	______ SAT ______UNSAT ______N/A
Step 18: [3.12] **CHECK** Drywell and Suppression Chamber pressure lowering. [3.13] **MAINTAIN** Primary Containment pressure below 55 psig using 2-FCV-64-29, DRYWELL VENT INBOARD ISOLATION VALVE, as directed by SRO. Standard: Checks that containment pressures are lowering.	______ SAT ______UNSAT ______N/A
CUE: Another operator will continue this procedure.	

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 large leak inside Primary Containment has developed
- The Reactor has been SCRAMMED
- Another Operator has been assigned to control Reactor Water Level and Reactor Pressure

INITIATING CUE:

The Unit Supervisor directs you to vent Primary Containment in accordance with 2-EOI-Appendix-13, Emergency Venting Primary Containment.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Align Components per 1-AOI-100-2, Control Room Abandonment, Attachment 3, Part B	
JPM NUMBER:		754-U1	REVISION:	0

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-AB-05 / Respond to Control Room Abandonment		
K/A RATINGS:	RO / SRO: 3.6		
K/A No. &STATEMENT:	295016 Control Room Abandonment AA1.04; Ability to operate and/or monitor the following as they apply to Control Room Abandonment: AC Electrical Distribution.		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	6		

EVALUATION LOCATION:	☒ In-Plant	☐ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	15 minutes		

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: _754-U1_

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to align electrical switchboards as required due to Control Room abandonment.

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-100-2

VALIDATIONTIME: __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

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	11/17/21	All	Initial issue

Procedure Revisions

Procedure	Revision
1-AOI-100-2	24

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, check protected equipment
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.

- The Unit 1 Control Room has been abandoned
- The Nuclear Unit Senior Operator (NUSO) has entered 1-AOI-100-2, Control Room Abandonment

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to complete 1-AOI-100-2, Attachment 3, Part B.

Another Operator has already **UNLOCKED** and **OPENED** 1-SHV-032-0332, CONTROL AIR SHUTOFF VALVE.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: 1-AOI-100-2, Control Room Abandonment **CAUTION** Failure to place control switch for each component in desired position prior to transferring to emergency may result in inadvertent actuation of the component. Aligns Drywell Cooling Unit 1A5 Blower Breaker as follows: (see table below) Expected action(s): **EXAMINER CUE: As switches are manipulated (simulated) as required by the procedure, inform that candidate that the switch has been positioned as performed.** Proceeds to the 480V Reactor MOV Board 1C (Unit 1 Reactor Building Elevation 565'), locates Breaker 1A, and simulates positioning switches as required by procedure.	**Critical Step** _____ SAT _____UNSAT _____N/A

Breaker Number	**Component**	**Required Position**
1A	1-HS-070-0041C, DRYWELL COOLING UNIT 1A5 BLOWER	STOP
	1-XS-070-0041, DRYWELL COOLING UNIT 1A5 BLOWER MOTOR TRANSFER SWITCH	EMERG

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>

<u>Step 2</u>:

Aligns Emergency Feeder Breaker from 480V Shutdown Board 1A as follows:

Breaker Number	**Component**	**Required Position**
2D	1-43-268-001C/02D, MANUAL NORM/EMER TRANSFER SWITCH	EMER

<u>Expected action(s):</u>

EXAMINER CUE: As switches are manipulated (simulated) as required by the procedure, inform that candidate that the switch has been positioned as performed.

Locates Breaker 2D and simulates positioning switches as required by procedure.

</td><td>

Critical Step

_____ SAT

_____UNSAT

_____N/A

</td></tr>
<tr><td>

<u>Step 3</u>:

Aligns RWCU Blowdown Valve Breaker as follows:

Breaker Number	**Component**	**Required Position**
7C	1-HS-069-0016C, RWCU BLOWDOWN TO MAIN CONDENSER	CLOSE
	1-XS-069-0016, RWCU BLOWDOWN TO MAIN CONDENSER	EMERGENCY

<u>Expected action(s):</u>

EXAMINER CUE: As switches are manipulated (simulated) as required by the procedure, inform that candidate that the switch has been positioned as performed.

Locates Breaker 7C and simulates positioning switches as required by procedure.

</td><td>

Critical Step

_____ SAT

_____UNSAT

_____N/A

</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: After completion of JPM Step 3, inform the candidate that 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 1.

- The Unit 1 Control Room has been abandoned
- The Nuclear Unit Senior Operator (NUSO) has entered 1-AOI-100-2, Control Room Abandonment

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to complete 1-AOI-100-2, Attachment 3, Part B.

Another Operator has already **UNLOCKED** and **OPENED** 1-SHV-032-0332, CONTROL AIR SHUTOFF VALVE.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Perform 2-EOI-APPENDIX-2, Defeating ARI Logic Trips
JPM NUMBER:	314-U2	REVISION:	2

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-26 / Defeat ARI Logic Trips in accordance with EOI Appendix-2		
K/A RATINGS:	RO / SRO: 4.2		
K/A No. &STATEMENT:	295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA1.01; Ability to operate and/or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Reactor protection system.		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	7		

EVALUATION LOCATION:	☒ In-Plant	☐ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	6 minutes		

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: 314-U2

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to perform operations necessary to bypass High-Pressure Coolant Injection isolation signals.

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-EOI-APPENDIX-2

VALIDATIONTIME: 6 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	12/3/19	All	Initial issue
1	1/14/2020	All	JPM updated
2	11/16/21	All	JPM updated

Procedure Revisions

Procedure	Revision
2-EOI-APPENDIX-2	4

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, check protected equipment
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 Reactor SCRAM has occurred
- The Nuclear Unit Senior Operator (NUSO) has entered 2-EOI-1A, ATWS RPV Control

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to defeat Alternate Rod Insertion (ARI) Logic trips in accordance with 2-EOI-APPENDIX-2, Defeating ARI Logic Trips.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: 2-EOI-APPENDIX-2, Defeating ARI Logic Trips [1] **REFER** to Attachment 1 and **OBTAIN** two keys. Expected action(s): **EXAMINER CUE: When the candidate begins to go to the Unit 2 Control Room to obtain the keys, state "You have the two required keys".** Refers to Attachment 1 and states that two keys will be retrieved from the Unit 2 Control Room.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 2: [2] **PLACE** 2-HS-068-0118B, ATWS MODE SWITCH, in TEST position on 2-LPNL-925-0613, ATWS CHANNEL B 250 VDC LOGIC CABINET. Expected action(s): **EXAMINER CUE: When the candidate simulates placing 2-HS-068-0118B in TEST, state "The ATWS Mode Switch is in the TEST position".** Proceeds to 4KV Shutdown Board Room '3A', and simulates placing 2-HS-068-0118B, ATWS MODE SWITCH, in TEST.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [3] **PLACE** 2-HS-068-0118A, ATWS MODE SWITCH, in TEST position on 2-LPNL-925-0716, ATWS CHANNEL A 250 VDC LOGIC CABINET. Expected action(s): **EXAMINER CUE: When the candidate simulates placing 2-HS-068-0118A in TEST, state “The ATWS Mode Switch is in the TEST position”.** Simulates placing 2-HS-068-0118B, ATWS MODE SWITCH, in TEST.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 4: [4] **NOTIFY** Unit Operator that ARI logic trips are defeated and ARI is reset. Expected action(s): **EXAMINER CUE: Acknowledge any report given to the Control Room/Unit Operator.** Informs the Unit 2 Control Room/Unit Operator that ARI logic trips are defeated and that ARI is reset.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: Acknowledge the candidate’s report that the task is complete.	

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.

- A Reactor SCRAM has occurred
- The Nuclear Unit Senior Operator (NUSO) has entered 2-EOI-1A, ATWS RPV Control

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to defeat Alternate Rod Insertion (ARI) Logic trips in accordance with 2-EOI-APPENDIX-2, Defeating ARI Logic Trips.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Perform 2-EOI-APPENDIX-16L, Bypassing HPCI High Temperature Isolation
JPM NUMBER:	755-U2	REVISION:	0

TASK APPLICABILITY:	☐ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-90 / Bypass HPCI High Temperature Isolations in Accordance with EOI Appendix 16L		
K/A RATINGS:	RO: 4.7 SRO: 4.0		
K/A No. &STATEMENT:	206000 HPCI High-Pressure Coolant Injection System A2.10; Ability to (a) predict the impacts of the following on the High-Pressure Coolant Injection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System isolation.		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	2		

EVALUATION LOCATION:	☒ In-Plant	☐ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	15 minutes		

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: 755-U2

RO _____ SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to perform operations necessary to bypass High-Pressure Coolant Injection isolation signals.

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-EOI-APPENDIX-16L

VALIDATIONTIME: 15 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

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	11/16/21	All	Initial Issue

Procedure Revisions

Procedure	Revision
2-EOI-APPENDIX-16L	2

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, check protected equipment
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 Reactor SCRAM has occurred
- High Pressure Coolant Injection (HPCI) is the only available high-pressure makeup source and is required to maintain Reactor Water Level
- A steam leak has developed in the HPCI Room

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to bypass the HPCI High Temperature Isolation signal in accordance with 1-EOI-APPENDIX-16L, Bypassing HPCI High Temperature Isolation.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

START TIME:___________

STEP / STANDARD	SAT / UNSAT
Step 1: 2-EOI-APPENDIX-16L, Bypassing HPCI High Temperature Isolation [1] **PROCEED** to the Unit 2 Auxiliary Instrument Room. Expected action(s): Goes to the Unit 2 Auxiliary Instrument Room.	______ SAT ______UNSAT ______N/A
Step 2: [2] **REFER** to Attachment 1 and **OBTAIN** necessary tools and equipment. Expected action(s): **EXAMINER CUE: When the candidate states that the fuse pullers and insulating sleeves are located in the EOI Equipment Storage Box in the Auxiliary Instrument Room, state "You have all required tools/equipment".** Simulates obtaining fuse pullers and insulating sleeves from the EOI Equipment Storage Box.	______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [3] **REFER** to Attachment 2 and **INSTALL** an insulating sleeve over each moving contact finger for the following relay on Panel 2-9-32, front: • Relay 23A-K34, contact 7 • Relay 23A-K34, contact 8 Expected action(s): **EXAMINER CUES:** • **When the candidate locates Panel 2-9-32 and simulates opening the cover for Relay 23A-K34, state "The relay cover is open".** • **When the candidate simulates placing the insulating sleeves on Contacts 7 and 8, state "The insulating sleeve has been placed on Contact 7(8)".** Locates Panel 2-9-32, simulates opening the cover for Relay 23A-K34, and simulates placing an insulating sleeve over Contacts 7 and 8.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 4: [4] **REFER** to Attachment 2 and **INSTALL** an insulating sleeve over each moving contact finger for the following relay on Panel 2-9-39, front: • Relay 23A-K6, contact 7 • Relay 23A-K6, contact 8 Expected action(s): **EXAMINER CUES:** • **When the candidate locates Panel 2-9-39 and simulates opening the cover for Relay 23A-K6, state "The relay cover is open".** • **When the candidate simulates placing the insulating sleeves on Contacts 7 and 8, state "The insulating sleeve has been placed on Contact 7(8)".** Locates Panel 2-9-39, simulates opening the cover for Relay 23A-K6, and simulates placing an insulating sleeve over Contacts 7 and 8.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 5: [5] NOTIFY Unit Operator that the HPCI high temperature isolation is bypassed. Expected action(s): **EXAMINER CUE: Acknowledge any report given to the Control Room/Unit Operator.** Informs the Unit 2 Control Room/Unit Operator that the HPCI High Temperature Isolation has been bypassed.	______ SAT ______UNSAT ______N/A
EXAMINER CUE: Acknowledge the candidate's report that the task is complete.	

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.

- A Reactor SCRAM has occurred
- High Pressure Coolant Injection (HPCI) is the only available high-pressure makeup source and is required to maintain Reactor Water Level
- A steam leak has developed in the HPCI Room

INITIATING CUES:

The Nuclear Unit Senior Operator directs you to bypass the HPCI High Temperature Isolation signal in accordance with 1-EOI-APPENDIX-16L, Bypassing HPCI High Temperature Isolation.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Drywell Leakage Calculation
JPM NUMBER:	556	REVISION:	1

TASK APPLICABILITY:	☐SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	U-000-SU-06 / Perform Instrument Checks and Observations SI			
K/A RATINGS:	RO 4.4			
K/A STATEMENT:	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.			
RELATED PRA INFORMATION:	None			
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: 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*

OPERATOR: ________________________________ JPM Number: __556__

RO ____ SRO ____ DATE: _______

TASK STANDARD: The Examinee is expected to calculate Drywell Floor and Equipment Sump leak rates to support a surveillance and determine if leak rates are met Technical Specification Acceptance Criteria (AC).

Operator Fundamental evaluated:
OF-3 Operating the Plant with a Conservative Bias

PRA: NA

REFERENCES/PROCEDURES NEEDED: 2-SR-2, Instrument Checks and Observations,
Unit 2 Technical Specification 3.4.4

VERIFICATION TIME: __20 min.______

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
1	02/01/2022	ALL	Updated JPM

Procedure Revisions

Procedure	Revision

****

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 2 is operating at 100% Rated Thermal Power (RTP), online for 100 days.

It is Saturday at 0800, and the Drywell Floor and Equipment Drain sumps have just completed pumping down.

The 0800 readings for 2-SR-2, Instrument Checks and Observations, are as follows:

• Drywell Floor Drain	22643 gals
• Drywell Equipment Drain	28005 gals

INITIATING CUE:

You are a Unit Operator, the Nuclear Unit Senior Operator (NUSO) has directed you to complete:

- 2-SR-2, Instrument Checks and Observations, for the Drywell Floor and Equipment Drain Sumps

- Determine if **ALL** Acceptance Criteria (AC) is met

Answer:

____

____

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____

KEY

BFN Unit 2	Instrument Checks and Observations	2-SR-2 Rev. 0087 Page 25 of 157

Attachment 2
(Page 4 of 90)

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

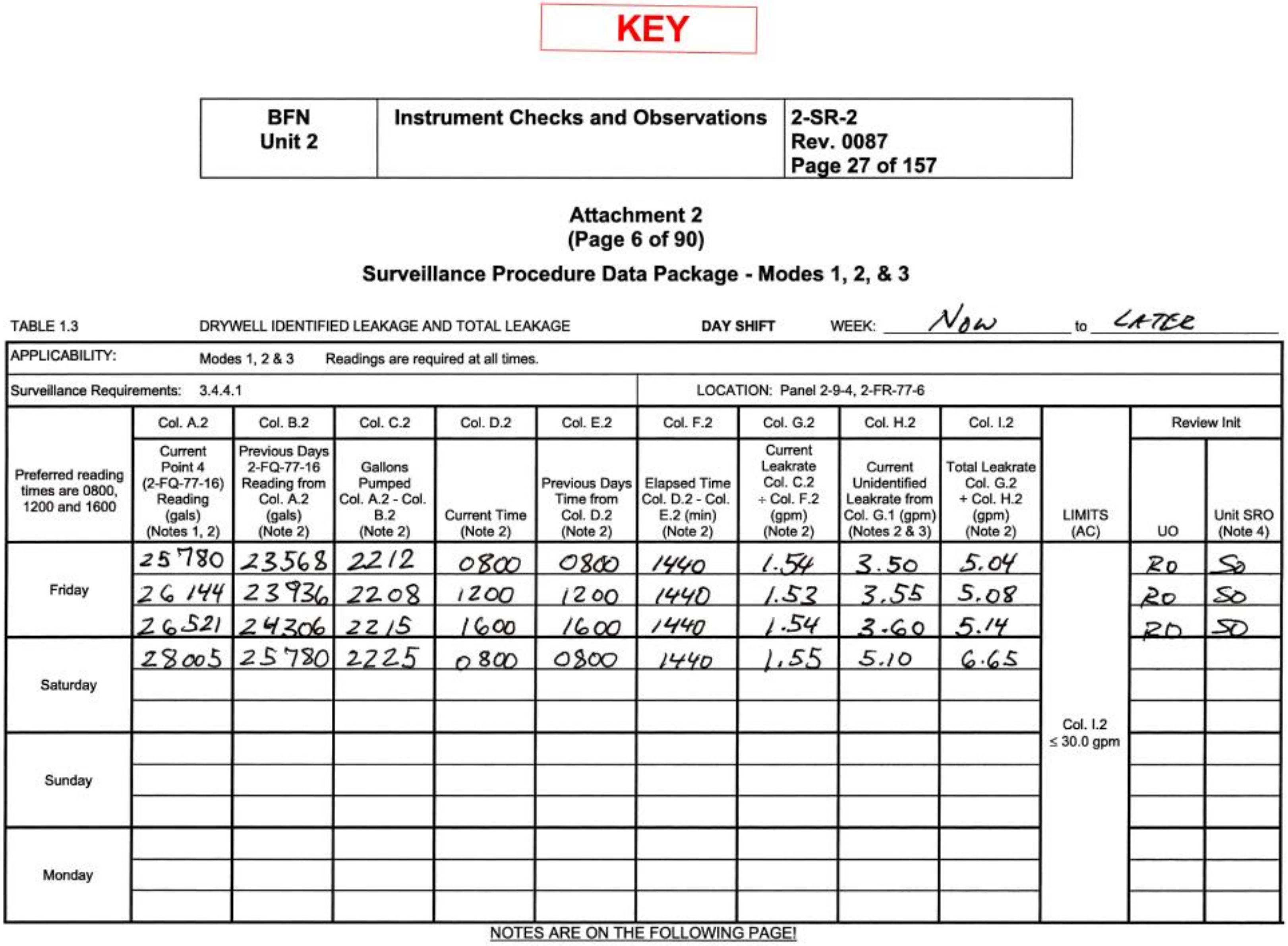
TABLE 1.2 DRYWELL UNIDENTIFIED LEAKAGE DAY SHIFT WEEK: Now to LATER

APPLICABILITY: Modes 1, 2 & 3 Readings are required at all times.

Surveillance Requirements: 3.4.4.1 LOCATION: Panel 2-9-4, 2-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 (2-FQ-77-6) Reading (gals) (Notes 1, 2)	Previous Days 2-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 SRO (Note 4)
Friday	15299	10259	5040	0800	0800	1440	3.50	3.50	0	Col. G.1 ≤ 5.0 gpm and Col. I.1 ≤ 2 gpm (Note 3)	RO	SO
	16223	11,111	5112	1200	1200	1440	3.55	3.50	+.05		RO	SO
	17155	11,971	5184	1600	1600	1440	3.60	3.50	+.10		RO	SO
Saturday	22643	15299	7344	0800	0800	1440	5.10	3.50	+1.60			
Sunday												
Monday												

NOTES ARE ON THE FOLLOWING PAGE!



KEY

BFN Unit 2	Instrument Checks and Observations	2-SR-2 Rev. 0087 Page 27 of 157

Attachment 2
(Page 6 of 90)

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

TABLE 1.3 DRYWELL IDENTIFIED LEAKAGE AND TOTAL LEAKAGE **DAY SHIFT** WEEK: Now to LATER

APPLICABILITY: Modes 1, 2 & 3 Readings are required at all times.

Surveillance Requirements: 3.4.4.1 LOCATION: Panel 2-9-4, 2-FR-77-6

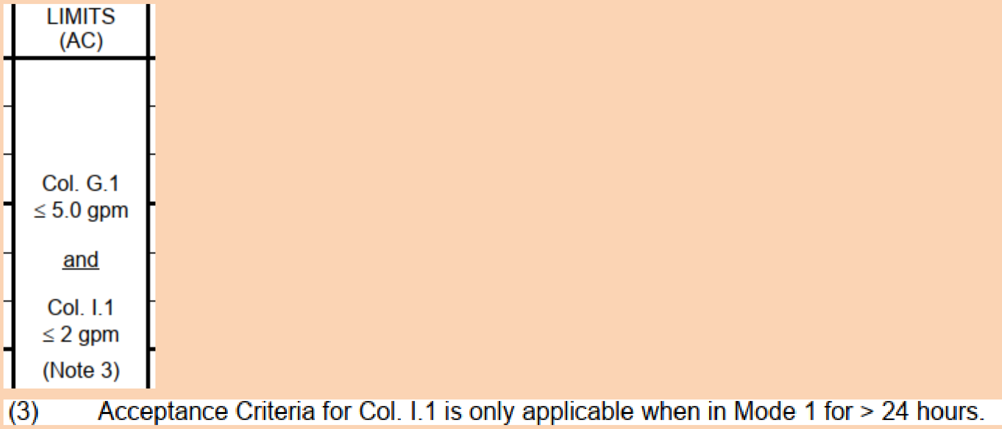
	Col. A.2	Col. B.2	Col. C.2	Col. D.2	Col. E.2	Col. F.2	Col. G.2	Col. H.2	Col. I.2		Review Init	
Preferred reading times are 0800, 1200 and 1600	Current Point 4 (2-FQ-77-16) Reading (gals) (Notes 1, 2)	Previous Days 2-FQ-77-16 Reading from Col. A.2 (gals) (Note 2)	Gallons Pumped Col. A.2 - Col. B.2 (Note 2)	Current Time (Note 2)	Previous Days Time from Col. D.2 (Note 2)	Elapsed Time Col. D.2 - Col. E.2 (min) (Note 2)	Current Leakrate Col. C.2 ÷ Col. F.2 (gpm) (Note 2)	Current Unidentified Leakrate from Col. G.1 (gpm) (Notes 2 & 3)	Total Leakrate Col. G.2 + Col. H.2 (gpm) (Note 2)	LIMITS (AC)	UO	Unit SRO (Note 4)
Friday	25780	23568	2212	0800	0800	1440	1.54	3.50	5.04	Col. I.2 ≤ 30.0 gpm	RO	SO
	26144	23936	2208	1200	1200	1440	1.53	3.55	5.08		RO	SO
	26521	24306	2215	1600	1600	1440	1.54	3.60	5.14		RO	SO
Saturday	28005	25780	2225	0800	0800	1440	1.55	5.10	6.65			
Sunday												
Monday												

NOTES ARE ON THE FOLLOWING PAGE!

START TIME:________________

STEP / STANDARD	SAT / UNSAT
Step 1: (2-SR-2, Instrument Checks and Observations, Attachment 2 (page 4 of 90), Table **1.2** – DRYWELL **UNIDENTIFIED** LEAKAGE, Column **G.1**) Completes 2-SR-2 for Drywell Unidentified Leakage for 0800 Saturday morning, and calculates current Drywell Unidentified Leakrate (Drywell Floor Drain). Expected Action(s): Examinee calculates current Drywell Unidentified Leakrate in Column **G.1** using the given Drywell Floor Drain readings of 22643 gals: 7344 gals / 1440 minutes = 5.10 gpm (± .05 gpm is acceptable margin)	**Critical Step** ____ SAT _____UNSAT _____N/A
Step 2: (2-SR-2, Instrument Checks and Observations, Attachment 2 (page 4 of 90), Table 1.2, Column **I.1**) Calculates Change in Leakrate. Expected Action(s): Examinee calculates a Change in Leakrate in Column I.1: 5.10 – 3.50 gpm (Column G.1– Column H.1) = 1.60 gpm (± .05 gpm is acceptable margin) Determines the current Change in Leakrate of 1.60 gpm is ≤ 2 gpm as required by 2-SR-2.	**Critical Step** ____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: (2-SR-2, Instrument Checks and Observations, Attachment 2 (page 6 of 90), Table **1.3** - DRYWELL **IDENTIFIED** LEAKAGE **AND TOTAL LEAKAGE**, Column **G.2**) Completes 2-SR-2 for Drywell Identified Leakage and Total Leakage for 0800 Saturday morning, and calculates Drywell Identified Leakrate (Drywell Equipment Drain). Expected Action(s): Examinee calculates a current Drywell Identified Leakrate in Column **G.2** using the given Drywell Equipment Drain readings of 28005 gals: 2225 gals / 1440 minutes = 1.55 gpm (± .05 gpm is acceptable margin)	**Critical Step** ____ SAT _____UNSAT _____N/A
Step 4: (2-SR-2, Instrument Checks and Observations, Attachment 2 (page 6 of 90), Table 1.3, Column **I.2**). Calculates Total Leakrate. Expected Action(s): Examinee calculates a Total Leakrate of 6.65 gpm in Column **I.2**: 1.55 + 5.10 gpm (Column G.2– Column H.2) = 6.65 gpm (± .05 gpm is acceptable margin) Determines the current Total Leakrate of 6.65 gpm is ≤ 30.0 gpm as required by 2-SR-2 to meet the Acceptance Criteria (AC).	**Critical Step** ____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 5: Compares the calculated current Drywell Unidentified Leakrate (5.10 gpm) from Column G.1 in JPM Step 1 above and Change in Leakrate ((+) 1.60 gpm) from Column I.1 in JPM Step 2 above to determine if the Acceptance Criteria (AC) LIMITS of ≤ 5.0 gpm and ≤ 2.0 gpm respectively have been met.  LIMITS (AC) Col. G.1 ≤ 5.0 gpm and Col. I.1 ≤ 2 gpm (Note 3) (3) Acceptance Criteria for Col. I.1 is only applicable when in Mode 1 for > 24 hours. Technical Specification 3.4.4, RCS Operational LEAKAGE LCO 3.4.4 RCS operational LEAKAGE shall be limited to: a. No pressure boundary LEAKAGE; b. ≤ 5 gpm unidentified LEAKAGE; and c. ≤ 30 gpm total LEAKAGE averaged over the previous 24 hour period; and d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1. APPLICABILITY: MODES 1, 2, and 3. Expected Action(s): Determines the current Drywell Unidentified Leakrate (5.10 gpm) change does NOT meet the Acceptance Criteria (AC) LIMITS of ≤ 5 gpm. 2-SR-2, Att. 2 (page 4 of 90), Table 1.2 requires both limits to be acceptable for the AC to be met.	**Critical Step** ____ SAT _____UNSAT _____N/A
EXAMINER NOTE: The Candidate may elect to reference Tech Spec 3.4.4, but it is not required since the AC LIMITS are listed on 2-SR-2.	

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: Once the Operator completes JPM Step 5 above, 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:

Unit 2 is operating at 100% Rated Thermal Power (RTP), online for 100 days.

It is Saturday at 0800, and the Drywell Floor and Equipment Drain sumps have just completed pumping down.

The 0800 readings for 2-SR-2, Instrument Checks and Observations, are as follows:

• Drywell Floor Drain	22643 gals
• Drywell Equipment Drain	28005 gals

INITIATING CUE:

You are a Unit Operator, the Nuclear Unit Senior Operator (NUSO) has directed you to complete:

- 2-SR-2, Instrument Checks and Observations, for the Drywell Floor and Equipment Drain Sumps
- Determine if **ALL** Acceptance Criteria (AC) is met

Answer:

____

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PROVIDE to APPLICANT

BFN Unit 2	Instrument Checks and Observations	2-SR-2 Rev. 0087 Page 25 of 157

Attachment 2
(Page 4 of 90)

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

TABLE 1.2 DRYWELL UNIDENTIFIED LEAKAGE DAY SHIFT WEEK: Now to LATER

APPLICABILITY: Modes 1, 2 & 3 Readings are required at all times.

Surveillance Requirements: 3.4.4.1 LOCATION: Panel 2-9-4, 2-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 (2-FQ-77-6) Reading (gals) (Notes 1, 2)	Previous Days 2-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 SRO (Note 4)
Friday	15299	10259	5040	0800	0800	1440	3.50	3.50	0	Col. G.1 ≤ 5.0 gpm and Col. I.1 ≤ 2 gpm (Note 3)	RO	SO
	16223	11,111	5112	1200	1200	1440	3.55	3.50	+.05		RO	SO
	17155	11,971	5184	1600	1600	1440	3.60	3.50	+.10		RO	SO
Saturday												
Sunday												
Monday												

NOTES ARE ON THE FOLLOWING PAGE!

PROVIDE to APPLICANT

BFN Unit 2	Instrument Checks and Observations	2-SR-2 Rev. 0087 Page 27 of 157

Attachment 2
(Page 6 of 90)

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

TABLE 1.3 DRYWELL IDENTIFIED LEAKAGE AND TOTAL LEAKAGE **DAY SHIFT** WEEK: Now to LATER

APPLICABILITY: Modes 1, 2 & 3 Readings are required at all times.

Surveillance Requirements: 3.4.4.1 LOCATION: Panel 2-9-4, 2-FR-77-6

Preferred reading times are 0800, 1200 and 1600	Col. A.2	Col. B.2	Col. C.2	Col. D.2	Col. E.2	Col. F.2	Col. G.2	Col. H.2	Col. I.2	LIMITS (AC)	Review Init	
	Current Point 4 (2-FQ-77-16) Reading (gals) (Notes 1, 2)	Previous Days 2-FQ-77-16 Reading from Col. A.2 (gals) (Note 2)	Gallons Pumped Col. A.2 - Col. B.2 (Note 2)	Current Time (Note 2)	Previous Days Time from Col. D.2 (Note 2)	Elapsed Time Col. D.2 - Col. E.2 (min) (Note 2)	Current Leakrate Col. C.2 ÷ Col. F.2 (gpm) (Note 2)	Current Unidentified Leakrate from Col. G.1 (gpm) (Notes 2 & 3)	Total Leakrate Col. G.2 + Col. H.2 (gpm) (Note 2)		UO	Unit SRO (Note 4)
Friday	25780	23568	2212	0800	0800	1440	1.54	3.50	5.04	Col. I.2 ≤ 30.0 gpm	RO	SO
	26144	23936	2208	1200	1200	1440	1.53	3.55	5.08		RO	SO
	26521	24306	2215	1600	1600	1440	1.54	3.60	5.14		RO	SO
Saturday												
Sunday												
Monday												

NOTES ARE ON THE FOLLOWING PAGE!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Determine Adequate Performance Of License Reactivation
JPM NUMBER:	661	REVISION:	3

TASK APPLICABILITY:	☐SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	N/A			
K/A RATINGS:	RO 3.3			
K/A STATEMENT:	2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.			
RELATED PRA INFORMATION:	None			
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

OPERATOR: ______________________________ JPM Number: __661__

RO ____ SRO ____ DATE: _______

TASK STANDARD: The Examinee is expected to determine which of the reactivating personnel have correctly completed the license reactivation process.

Operator Fundamental evaluated:
OF-4 Working Effectively as a Team

PRA: NA

REFERENCES/PROCEDURES NEEDED: OPDP-10

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	02/10/2022	ALL	Updated JPM

Procedure Revisions

Procedure	Revision
OPDP-10	11

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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:

3 off-shift licensed personnel are returning to shift from rotating assignments and are reactivating their licenses. The following table gives information about hours worked under direction of an activated licensee, tours performed, etc.

License	Pre-activation Meeting	Shift 1	Shift 2	Shift 3	Shift 4	Shift 5	Shift 6	Plant Tour
RO1	Ops Training Manager And Shift Manager	12 hrs U-3 RO	12 hrs U-3 RO	12 hrs U-1 RO	12 hrs tagging SST RO	12 hrs U-2 RO	12 hrs U-2 RO	completed plant tour with STA (SRO)
RO2	Ops Training Manager And Ops Superintendent	12 hrs U-2 RO	12 hrs U-2 RO Called for Random Drug test during shift – missed end of shift turnover	12 hrs U-1 RO	12 hrs U-3 RO	12hrs U-3 RO	N/A	completed plant tour with SST (RO)
RO3	Ops Training Manager And Ops Superintendent	12 hrs U-2 RO	12 hrs U-2 RO	12 hrs U-3 RO	12 hrs U-3 RO	12 hrs U-1 RO	N/A	completed plant tour with U1 RB AUO

INITIATING CUES:

The Shift Manager has tasked you to determine which of these personnel, if any, have completed the requirements for license reactivation.

If any personnel do not meet the requirements for license reactivation state the reason(s) why.

KEY

****

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:

3 off-shift licensed personnel are returning to shift from rotating assignments and are reactivating their licenses. The following table gives information about hours worked under direction of an activated licensee, tours performed, etc.

License	Pre-activation Meeting	Shift 1	Shift 2	Shift 3	Shift 4	Shift 5	Shift 6	Plant Tour	Performance Step
RO1	Ops Training Manager And Shift Manager	12 hrs U-3 RO	12 hrs U-3 RO	12 hrs U-1 RO	12 hrs tagging SST RO	12 hrs U-2 RO	12 hrs U-2 RO	completed plant tour with STA (SRO)	**Does Not Meet**
RO2	Ops Training Manager And Ops Superintendent	12 hrs U-2 RO	12 hrs U-2 RO Called for Random Drug test during shift – missed end of shift turnover	12 hrs U-1 RO	12 hrs U-3 RO	12hrs U-3 RO	N/A	completed plant tour with SST (RO)	Meets requirements
RO3	Ops Training Manager And Ops Superintendent	12 hrs U-2 RO	12 hrs U-2 RO	12 hrs U-3 RO	12 hrs U-3 RO	12 hrs U-1 RO	N/A	completed plant tour with U1 RB AUO	**Does Not Meet**

INITIATING CUES:

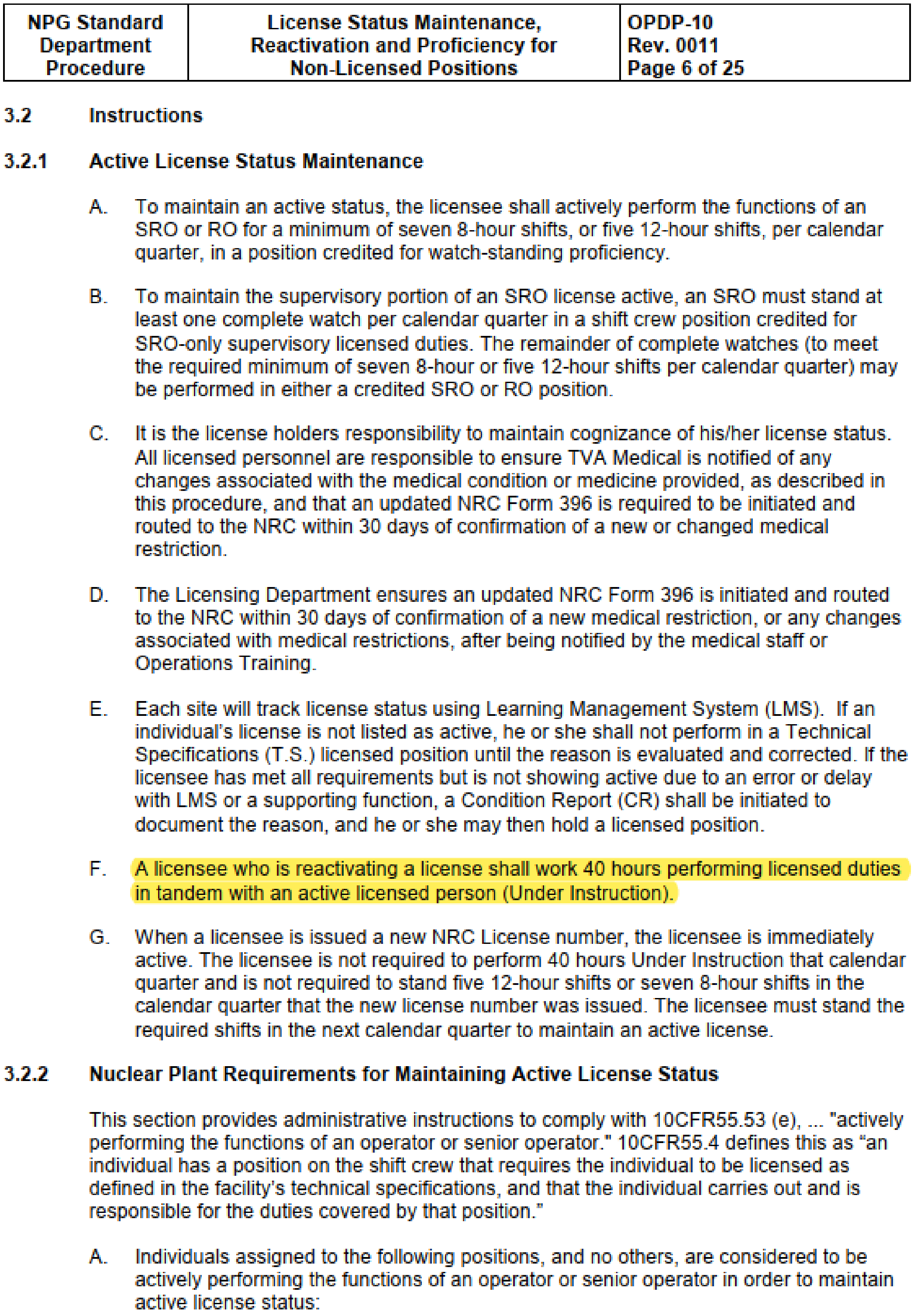
The Shift Manager has tasked you to determine which of these personnel, if any, have completed the requirements for license reactivation. If any personnel do not meet the requirements for license reactivation state the reason(s) why.

Answer:

RO-1 did not interview with the Operations Superintendent, contrary to OPDP-10. The Shift Manager is **NOT** an authorized alternative for the Operations Superintendent.

RO-3 performed the plant tour with a non-licensed Operator, contrary to OPDP-10

KEY



NPG Standard Department Procedure	License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions	OPDP-10 Rev. 0011 Page 6 of 25

3.2 Instructions

3.2.1 Active License Status Maintenance

A. To maintain an active status, the licensee shall actively perform the functions of an SRO or RO for a minimum of seven 8-hour shifts, or five 12-hour shifts, per calendar quarter, in a position credited for watch-standing proficiency.

B. To maintain the supervisory portion of an SRO license active, an SRO must stand at least one complete watch per calendar quarter in a shift crew position credited for SRO-only supervisory licensed duties. The remainder of complete watches (to meet the required minimum of seven 8-hour or five 12-hour shifts per calendar quarter) may be performed in either a credited SRO or RO position.

C. It is the license holders responsibility to maintain cognizance of his/her license status. All licensed personnel are responsible to ensure TVA Medical is notified of any changes associated with the medical condition or medicine provided, as described in this procedure, and that an updated NRC Form 396 is required to be initiated and routed to the NRC within 30 days of confirmation of a new or changed medical restriction.

D. The Licensing Department ensures an updated NRC Form 396 is initiated and routed to the NRC within 30 days of confirmation of a new medical restriction, or any changes associated with medical restrictions, after being notified by the medical staff or Operations Training.

E. Each site will track license status using Learning Management System (LMS). If an individual's license is not listed as active, he or she shall not perform in a Technical Specifications (T.S.) licensed position until the reason is evaluated and corrected. If the licensee has met all requirements but is not showing active due to an error or delay with LMS or a supporting function, a Condition Report (CR) shall be initiated to document the reason, and he or she may then hold a licensed position.

F. A licensee who is reactivating a license shall work 40 hours performing licensed duties in tandem with an active licensed person (Under Instruction).

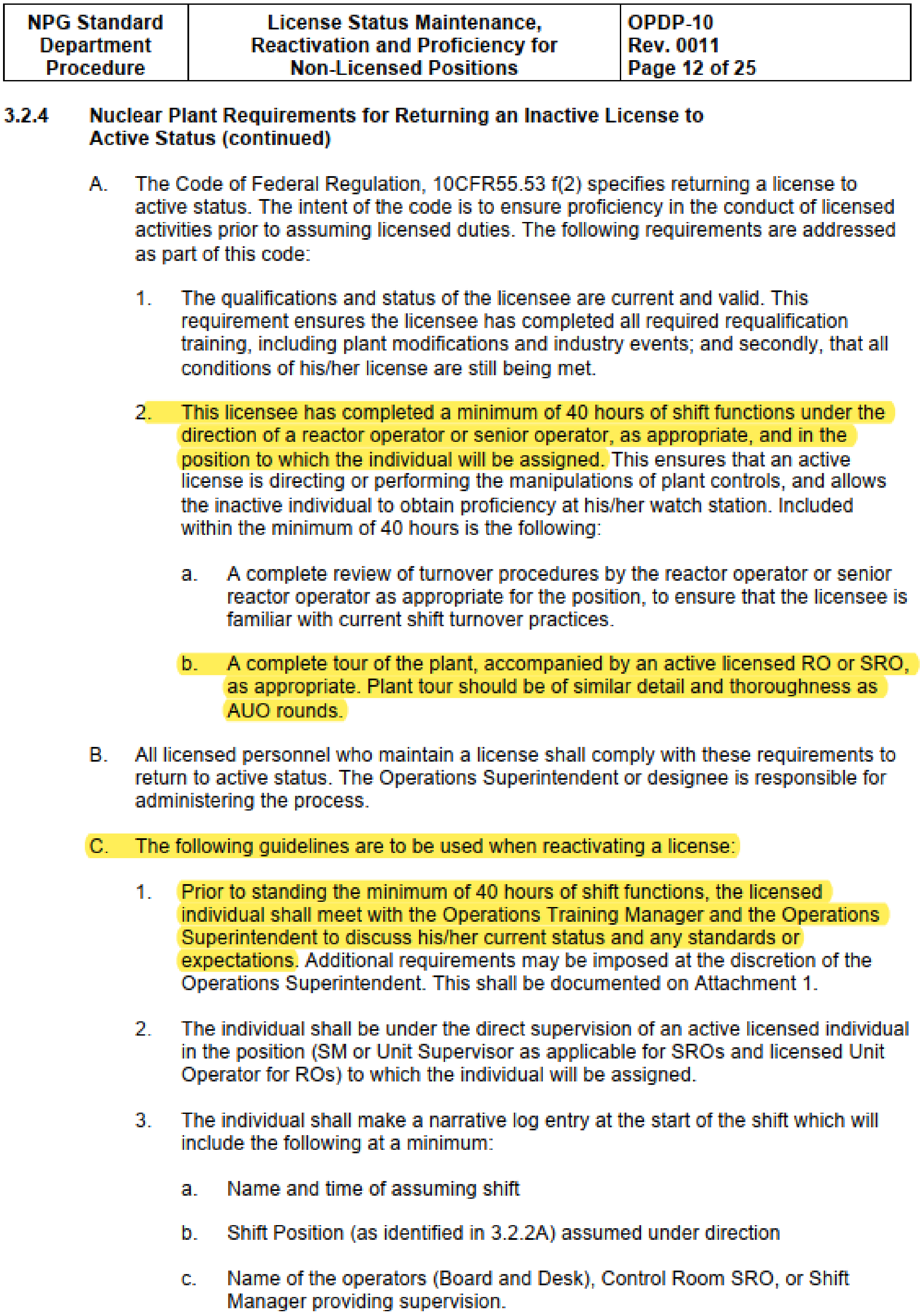
G. When a licensee is issued a new NRC License number, the licensee is immediately active. The licensee is not required to perform 40 hours Under Instruction that calendar quarter and is not required to stand five 12-hour shifts or seven 8-hour shifts in the calendar quarter that the new license number was issued. The licensee must stand the required shifts in the next calendar quarter to maintain an active license.

3.2.2 Nuclear Plant Requirements for Maintaining Active License Status

This section provides administrative instructions to comply with 10CFR55.53 (e), ... "actively performing the functions of an operator or senior operator." 10CFR55.4 defines this as "an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications, and that the individual carries out and is responsible for the duties covered by that position."

A. Individuals assigned to the following positions, and no others, are considered to be actively performing the functions of an operator or senior operator in order to maintain active license status:

KEY



NPG Standard Department Procedure	License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions	OPDP-10 Rev. 0011 Page 12 of 25

3.2.4 Nuclear Plant Requirements for Returning an Inactive License to Active Status (continued)

A. The Code of Federal Regulation, 10CFR55.53 f(2) specifies returning a license to active status. The intent of the code is to ensure proficiency in the conduct of licensed activities prior to assuming licensed duties. The following requirements are addressed as part of this code:

1. The qualifications and status of the licensee are current and valid. This requirement ensures the licensee has completed all required requalification training, including plant modifications and industry events; and secondly, that all conditions of his/her license are still being met.

2. This licensee has completed a minimum of 40 hours of shift functions under the direction of a reactor operator or senior operator, as appropriate, and in the position to which the individual will be assigned. This ensures that an active license is directing or performing the manipulations of plant controls, and allows the inactive individual to obtain proficiency at his/her watch station. Included within the minimum of 40 hours is the following:

   a. A complete review of turnover procedures by the reactor operator or senior reactor operator as appropriate for the position, to ensure that the licensee is familiar with current shift turnover practices.

   b. A complete tour of the plant, accompanied by an active licensed RO or SRO, as appropriate. Plant tour should be of similar detail and thoroughness as AUO rounds.

B. All licensed personnel who maintain a license shall comply with these requirements to return to active status. The Operations Superintendent or designee is responsible for administering the process.

C. The following guidelines are to be used when reactivating a license:

1. Prior to standing the minimum of 40 hours of shift functions, the licensed individual shall meet with the Operations Training Manager and the Operations Superintendent to discuss his/her current status and any standards or expectations. Additional requirements may be imposed at the discretion of the Operations Superintendent. This shall be documented on Attachment 1.

2. The individual shall be under the direct supervision of an active licensed individual in the position (SM or Unit Supervisor as applicable for SROs and licensed Unit Operator for ROs) to which the individual will be assigned.

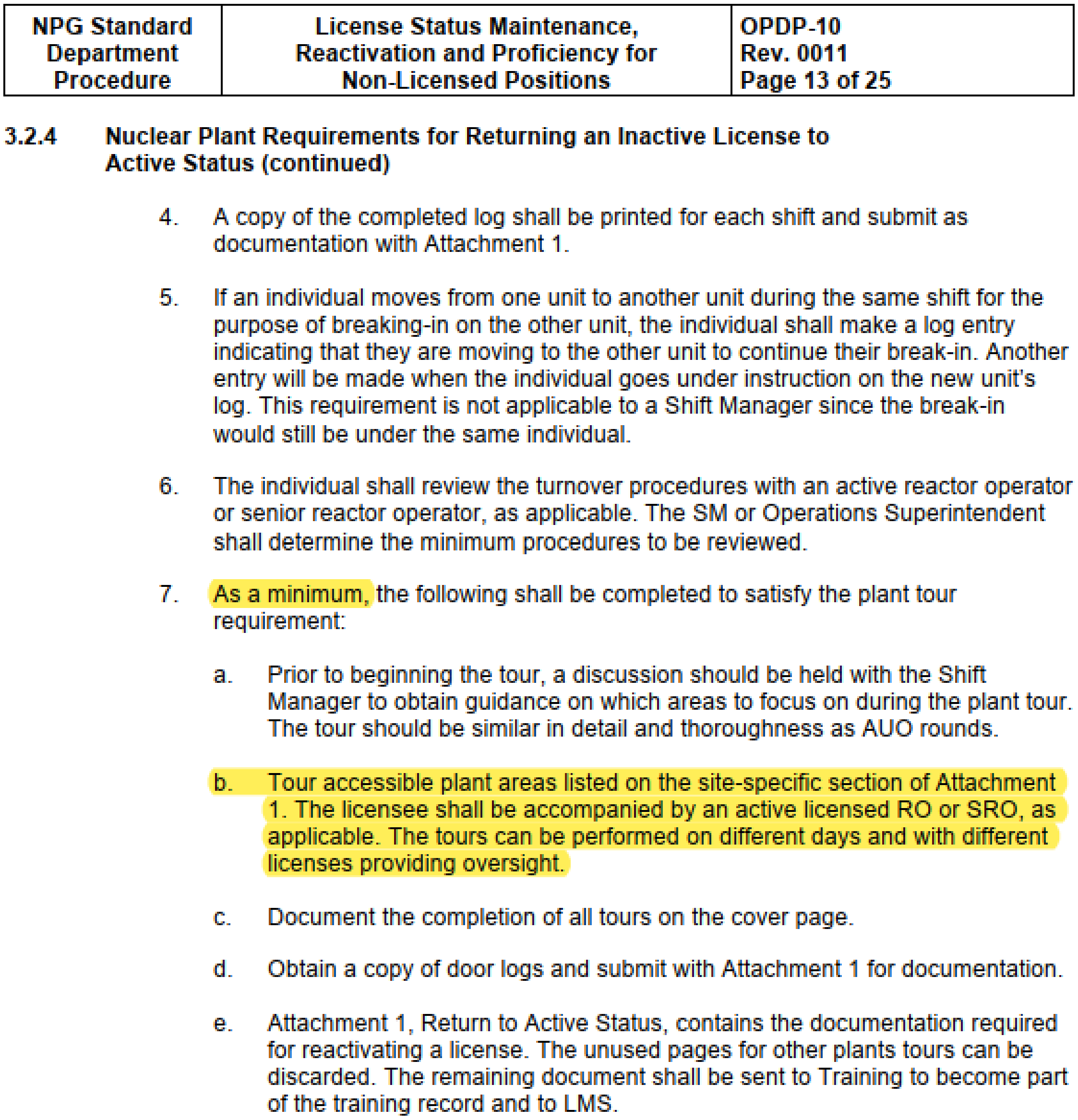
3. The individual shall make a narrative log entry at the start of the shift which will include the following at a minimum:

   a. Name and time of assuming shift

   b. Shift Position (as identified in 3.2.2A) assumed under direction

   c. Name of the operators (Board and Desk), Control Room SRO, or Shift Manager providing supervision.

KEY



NPG Standard Department Procedure	**License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions**	**OPDP-10** **Rev. 0011** **Page 13 of 25**

3.2.4 Nuclear Plant Requirements for Returning an Inactive License to Active Status (continued)

4. A copy of the completed log shall be printed for each shift and submit as documentation with Attachment 1.

5. If an individual moves from one unit to another unit during the same shift for the purpose of breaking-in on the other unit, the individual shall make a log entry indicating that they are moving to the other unit to continue their break-in. Another entry will be made when the individual goes under instruction on the new unit's log. This requirement is not applicable to a Shift Manager since the break-in would still be under the same individual.

6. The individual shall review the turnover procedures with an active reactor operator or senior reactor operator, as applicable. The SM or Operations Superintendent shall determine the minimum procedures to be reviewed.

7. As a minimum, the following shall be completed to satisfy the plant tour requirement:

   a. Prior to beginning the tour, a discussion should be held with the Shift Manager to obtain guidance on which areas to focus on during the plant tour. The tour should be similar in detail and thoroughness as AUO rounds.

   b. Tour accessible plant areas listed on the site-specific section of Attachment 1. The licensee shall be accompanied by an active licensed RO or SRO, as applicable. The tours can be performed on different days and with different licenses providing oversight.

   c. Document the completion of all tours on the cover page.

   d. Obtain a copy of door logs and submit with Attachment 1 for documentation.

   e. Attachment 1, Return to Active Status, contains the documentation required for reactivating a license. The unused pages for other plants tours can be discarded. The remaining document shall be sent to Training to become part of the training record and to LMS.

START TIME:__________

STEP / STANDARD	SAT / UNSAT
Step 1: Uses OPDP-10, License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions, to analyze the information provided to determine which personnel meet the requirements for license reactivation. Expected Action(s): Examinee determines that RO-2 meets the requirements for license reactivation in accordance with OPDP-10.	**Critical Step** ___ SAT ___UNSAT ___N/A
Step 2: States the reason why RO-1 does not meet the requirements for license reactivation. Expected Action(s): RO-1 did not interview with the Operations Superintendent, contrary to OPDP-10. The Shift Manager is **NOT** an authorized alternative for the Operations Superintendent.	**Critical Step** ___ SAT ___UNSAT ___N/A
Step 3: States the reason why RO-3 does not meet the requirements for license reactivation. Expected Action(s): RO-3 performed the plant tour with a non-licensed Operator, contrary to OPDP-10.	**Critical Step** ___ SAT ___UNSAT ___N/A
EXAMINER CUE: Once the Operator completes JPM Step 3 above, 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:

3 off-shift licensed personnel are returning to shift from rotating assignments and are reactivating their licenses. The following table gives information about hours worked under direction of an activated licensee, tours performed, etc.

License	Pre-activation Meeting	Shift 1	Shift 2	Shift 3	Shift 4	Shift 5	Shift 6	Plant Tour
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RO2	Ops Training Manager And Ops Superintendent	12 hrs U-2 RO	12 hrs U-2 RO Called for Random Drug test during shift – missed end of shift turnover	12 hrs U-1 RO	12 hrs U-3 RO	12hrs U-3 RO	N/A	completed plant tour with SST (RO)
RO3	Ops Training Manager And Ops Superintendent	12 hrs U-2 RO	12 hrs U-2 RO	12 hrs U-3 RO	12 hrs U-3 RO	12 hrs U-1 RO	N/A	completed plant tour with U1 RB AUO

INITIATING CUES:

The Shift Manager has tasked you to determine which of these personnel, if any, have completed the requirements for license reactivation.

If any personnel do not meet the requirements for license reactivation state the reason(s) why.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Perform Jet Pump Mismatch and Operability SR
JPM NUMBER:	680	REVISION:	2

TASK APPLICABILITY:	☐ SRO-U	☐ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-068-SU-05: Perform Jet Pump Mismatch and Operability		
K/A RATINGS:	RO 3.7		
K/A STATEMENT:	2.2.12 Knowledge of surveillance procedures		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	EQUIPMENT CONTROL - ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	20 minutes		

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: 680_

RO _______ DATE: __________

TASK STANDARD: Using plant parameters, the Examinee is expected to perform a Jet Pump Mismatch and Operability surveillance to determine if Acceptance Criteria (AC) is met.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 2-SR-3.4.2.1, Jet Pump Mismatch and Operability

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

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
2	11/15/21	All	JPM Update

Procedure Revisions

Procedure	Revision
2-SR-3.4.2.1	51

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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:

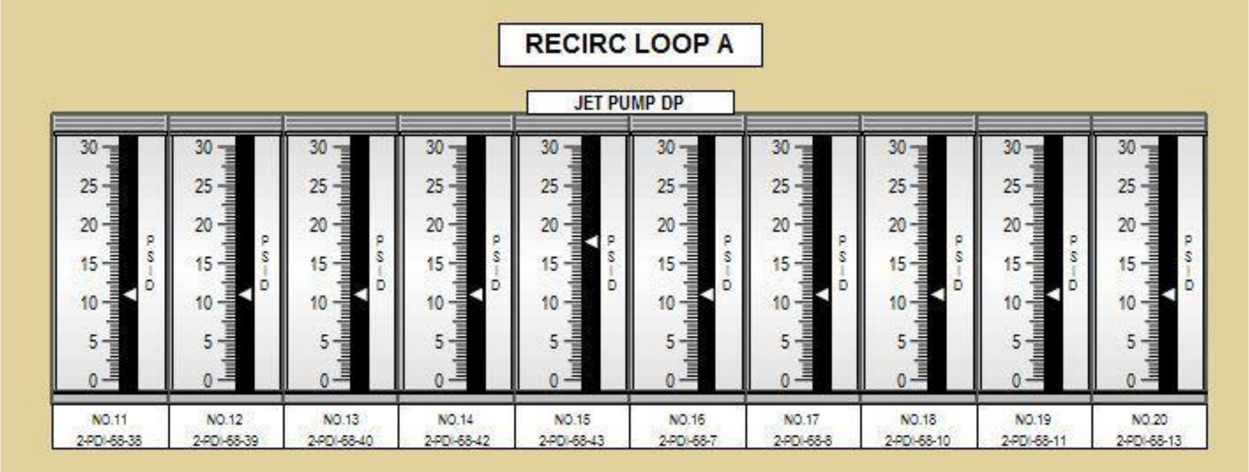
Unit 2 is operating at 100% Rated Thermal Power (RTP) and you are the Unit 2 Operator.

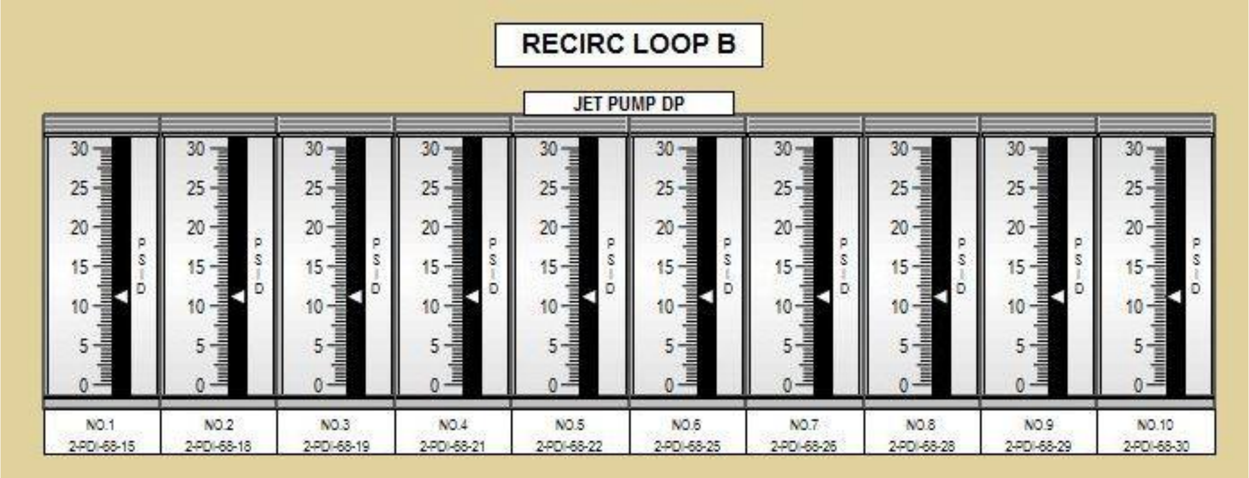
INITIATING CUE:

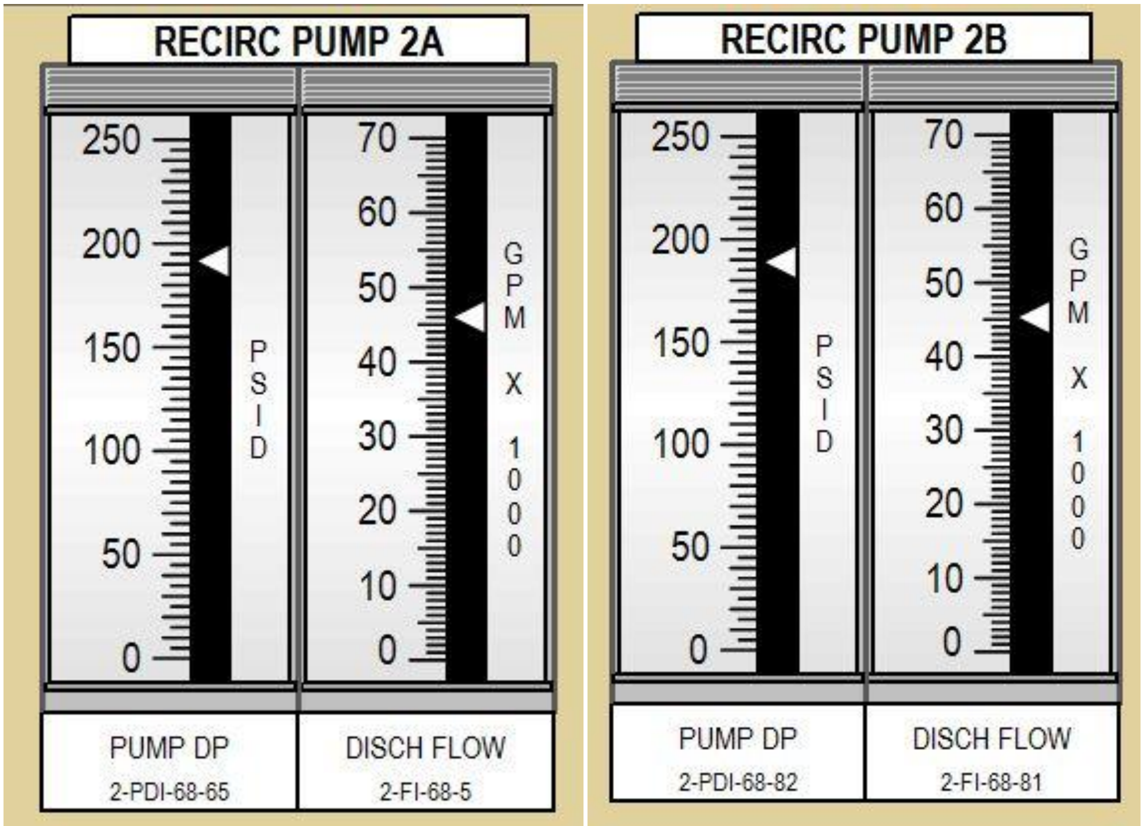
Perform 2-SR-3.4.2.1; Jet Pump Mismatch and Operability starting at step 7.2.1, Core Power and Flow Readings using the provided readings below:

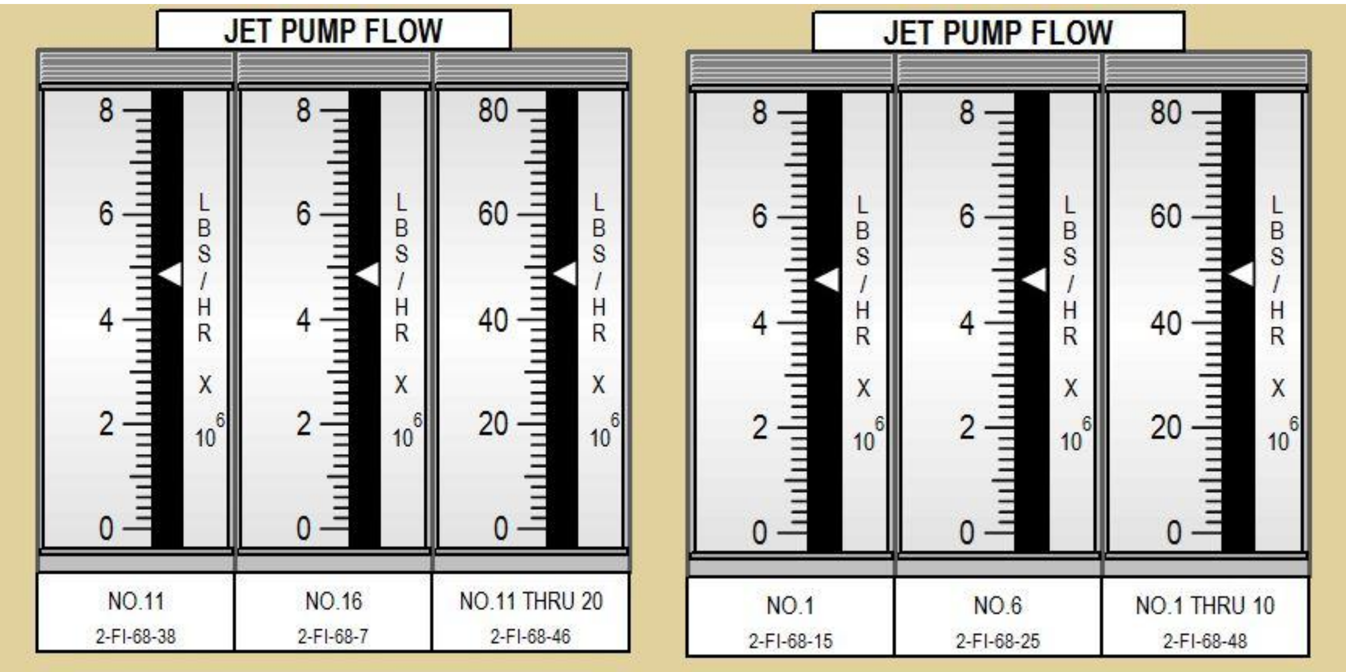
CALC002 = 3952 CMWT
2-XR-68-50 Green Pen = 18.0 psid
2-XR-68-50 Red Pen = 97.1 Mlb/hr

Recirc Motor Speeds
2A - 2-SI-68-59 = 1456 rpm
2B - 2-SI-68-71 = 1456 rpm


RECIRC LOOP A
JET PUMP DP
30
25
20
15
10
5
0
PSID
NO.11 2-PDI-68-38
NO.12 2-PDI-68-39
NO.13 2-PDI-68-40
NO.14 2-PDI-68-42
NO.15 2-PDI-68-43
NO.16 2-PDI-68-7
NO.17 2-PDI-68-8
NO.18 2-PDI-68-10
NO.19 2-PDI-68-11
NO.20 2-PDI-68-13


RECIRC LOOP B
JET PUMP DP
30
25
20
15
10
5
0
PSID
NO.1 2-PDI-68-15
NO.2 2-PDI-68-18
NO.3 2-PDI-68-19
NO.4 2-PDI-68-21
NO.5 2-PDI-68-22
NO.6 2-PDI-68-25
NO.7 2-PDI-68-26
NO.8 2-PDI-68-28
NO.9 2-PDI-68-29
NO.10 2-PDI-68-30


RECIRC PUMP 2A
250
200
150
100
50
0
PSID
70
60
50
40
30
20
10
0
GPM X 1000
PUMP DP
2-PDI-68-65
DISCH FLOW
2-FI-68-5
RECIRC PUMP 2B
250
200
150
100
50
0
PSID
70
60
50
40
30
20
10
0
GPM X 1000
PUMP DP
2-PDI-68-82
DISCH FLOW
2-FI-68-81


JET PUMP FLOW
8
6
4
2
0
LBS/HR X 10^6
NO.11
2-FI-68-38
NO.16
2-FI-68-7
80
60
40
20
0
NO.11 THRU 20
2-FI-68-46
JET PUMP FLOW
NO.1
2-FI-68-15
NO.6
2-FI-68-25
NO.1 THRU 10
2-FI-68-48

START TIME _____

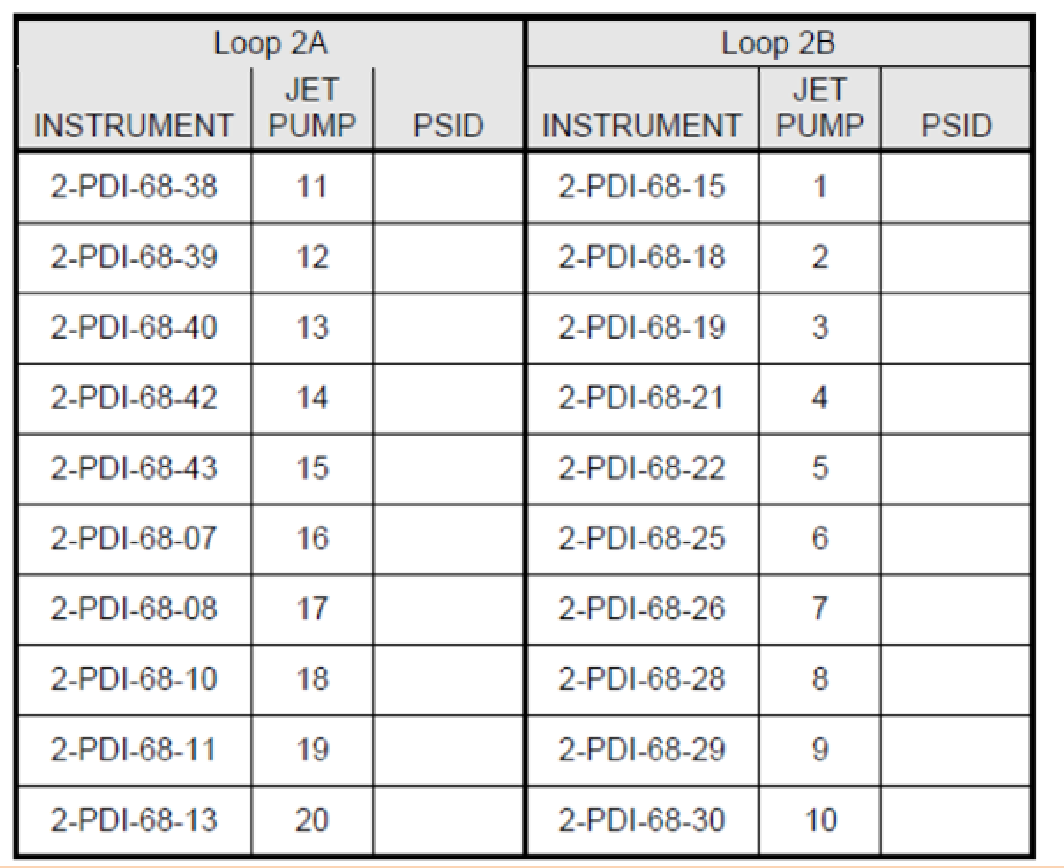
STEP / STANDARD	SAT / UNSAT
Step 1: **7.2.1 Core Power and Flow Readings** [1] **RECORD** the Core Thermal Power from Core Power and Flow Log. (N/A if ICS is NOT available) Point CALC002 __________CMWT Expected Action(s): The candidate is expected to record 3952 CMWT as given.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 2: [2] **RECORD** the Core Plate Differential Pressure from ICS point 68-52 or 2-XR-68-50 (Green Pen). (N/A if not available) Core Press Drop 68-52 ___________PSID Expected Action(s): The candidate is expected to record 18.0 PSID as given.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 3: [3] **RECORD** the Total Core Flow. Total Core Flow (Red Pen) 2-XR-68-50 _________ Mlb/hr Expected Action(s): The candidate is expected to record 97.1 Mlb/hr as given.	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 4: **7.2.2 Recirculation Pump Loops** [1] **RECORD** the Recirc Pump 2A and 2B Motor Speeds for operating Recirc Pumps and **MARK** instrumentation used. Pump Motor 2A √ Pump Motor 2B √ 2-SI-68-59 ☐ 2-SI-68-71 ☐ 2-SIT-068-0059 ☐ 2-SIT-068-0071 ☐ 2-SI-96-61 ☐ 2-SI-96-73 ☐ ________ RPM ________ RPM Expected Action(s): The candidate is expected to record the following as given. Pump Motor 2A √ Pump Motor 2B √ 2-SI-68-59 ☑ 2-SI-68-71 ☑ 2-SIT-068-0059 ☐ 2-SIT-068-0071 ☐ 2-SI-96-61 ☐ 2-SI-96-73 ☐ _1456__ RPM __1456__ RPM	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 5: [2] **RECORD** the Recirc Pump Discharge flows. Loop 2A Loop 2B 2-FI-68-5 2-FI-68-81 ________ gpm X 1000 ________ gpm X 1000 Expected Action(s): The candidate is expected to record the Recirc Pump Discharge flows as follows: Loop 2A Loop 2B 2-FI-68-5 2-FI-68-81 __45 to 47__ gpm X 1000 __44 to 46__ gpm X 1000	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT
Step 6: [3] **RECORD** the Recirc Loop 2A and 2B Jet Pump flows. Loop 2A Loop 2B 2-FI-68-46 2-FI-68-48 ________ Mlb/hr ________ Mlb/hr Expected Action(s): The candidate is expected to record the Recirc Loop 2A and 2B Jet Pump flows as follows: Loop 2A Loop 2B 2-FI-68-46 2-FI-68-48 __48 to 50___Mlb/hr _48 to 50___Mlb/hr	**Critical Step** ______ SAT ______UNSAT ______N/A

Step 7: **7.2.3 Jet Pump Loops**

[1] **RECORD** the following Differential Pressure readings below:



Loop 2A			Loop 2B		
INSTRUMENT	JET PUMP	PSID	INSTRUMENT	JET PUMP	PSID
2-PDI-68-38	11		2-PDI-68-15	1	
2-PDI-68-39	12		2-PDI-68-18	2	
2-PDI-68-40	13		2-PDI-68-19	3	
2-PDI-68-42	14		2-PDI-68-21	4	
2-PDI-68-43	15		2-PDI-68-22	5	
2-PDI-68-07	16		2-PDI-68-25	6	
2-PDI-68-08	17		2-PDI-68-26	7	
2-PDI-68-10	18		2-PDI-68-28	8	
2-PDI-68-11	19		2-PDI-68-29	9	
2-PDI-68-13	20		2-PDI-68-30	10	

Expected Action(s):

The candidate is expected to record the following Differential Pressure readings:

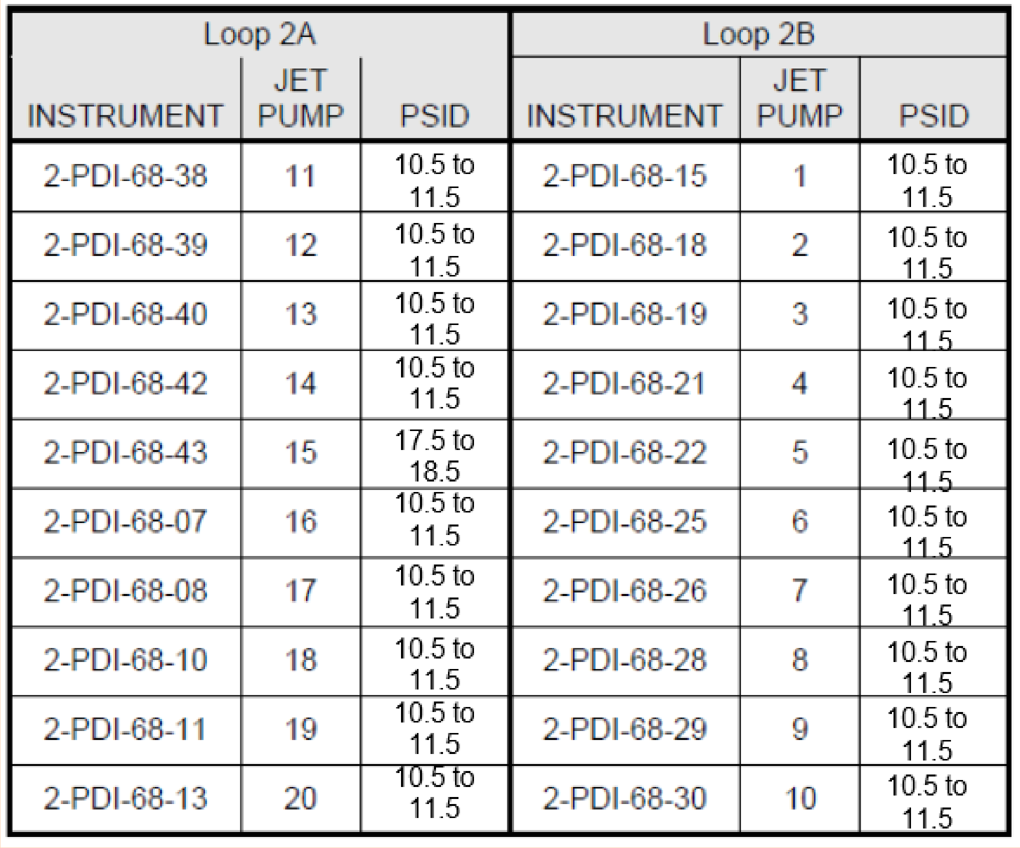
Critical Step

______ SAT

______UNSAT

______N/A

STEP / STANDARD	SAT / UNSAT



Loop 2A			Loop 2B		
INSTRUMENT	JET PUMP	PSID	INSTRUMENT	JET PUMP	PSID
2-PDI-68-38	11	10.5 to 11.5	2-PDI-68-15	1	10.5 to 11.5
2-PDI-68-39	12	10.5 to 11.5	2-PDI-68-18	2	10.5 to 11.5
2-PDI-68-40	13	10.5 to 11.5	2-PDI-68-19	3	10.5 to 11.5
2-PDI-68-42	14	10.5 to 11.5	2-PDI-68-21	4	10.5 to 11.5
2-PDI-68-43	15	17.5 to 18.5	2-PDI-68-22	5	10.5 to 11.5
2-PDI-68-07	16	10.5 to 11.5	2-PDI-68-25	6	10.5 to 11.5
2-PDI-68-08	17	10.5 to 11.5	2-PDI-68-26	7	10.5 to 11.5
2-PDI-68-10	18	10.5 to 11.5	2-PDI-68-28	8	10.5 to 11.5
2-PDI-68-11	19	10.5 to 11.5	2-PDI-68-29	9	10.5 to 11.5
2-PDI-68-13	20	10.5 to 11.5	2-PDI-68-30	10	10.5 to 11.5

EXAMINER NOTE: PSID band given due to increments. 2-PDI-68-43, JET PUMP 15 is the abnormal reading.

STEP / STANDARD	SAT / UNSAT
Step 8: **7.3 Tech Spec 3.4.1.1 - Recirculation Loop Mismatch Verification with Both Recirculation Loops in Operation Checks** [1] **CALCULATE** percent of Rated Core Flow (%WT) using data obtained in Section 7.2.1.[3] as follows. (Step 7.2.1[3] ÷102.5) X 100 = % Core Flow (__________ ÷ 102.5) X 100 = ________ Expected Action(s): The candidate is expected to calculate %WT using data obtained in Section 7.2.1.[3]: (Step 7.2.1[3] ÷102.5) X 100 = % Core Flow (97.1 Mlb/hr ÷ 102.5) X 100 = _94.7-94.8 %__	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 9: [2] **CALCULATE** the absolute value for Recirculation Loop Jet Pump Flow Mismatch using data obtained in Section 7.2.2[3] as follows. 2-FI-68-46 - 2-FI-68-48 = Mismatch _______ Mlb/hr - _______ Mlb/hr = _______ Mlb/hr Expected Action(s): The candidate is expected to calculate the absolute value for Recirculation Loop Jet Pump Flow Mismatch using data obtained in Section 7.2.2[3]: 2-FI-68-46 - 2-FI-68-48 = Mismatch _49____Mlb/hr - _49____ Mlb/hr = ___0 - 3____ Mlb/hr	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER NOTE: Procedure Steps [3] and [4] below are Acceptance Criteria (AC) steps.	

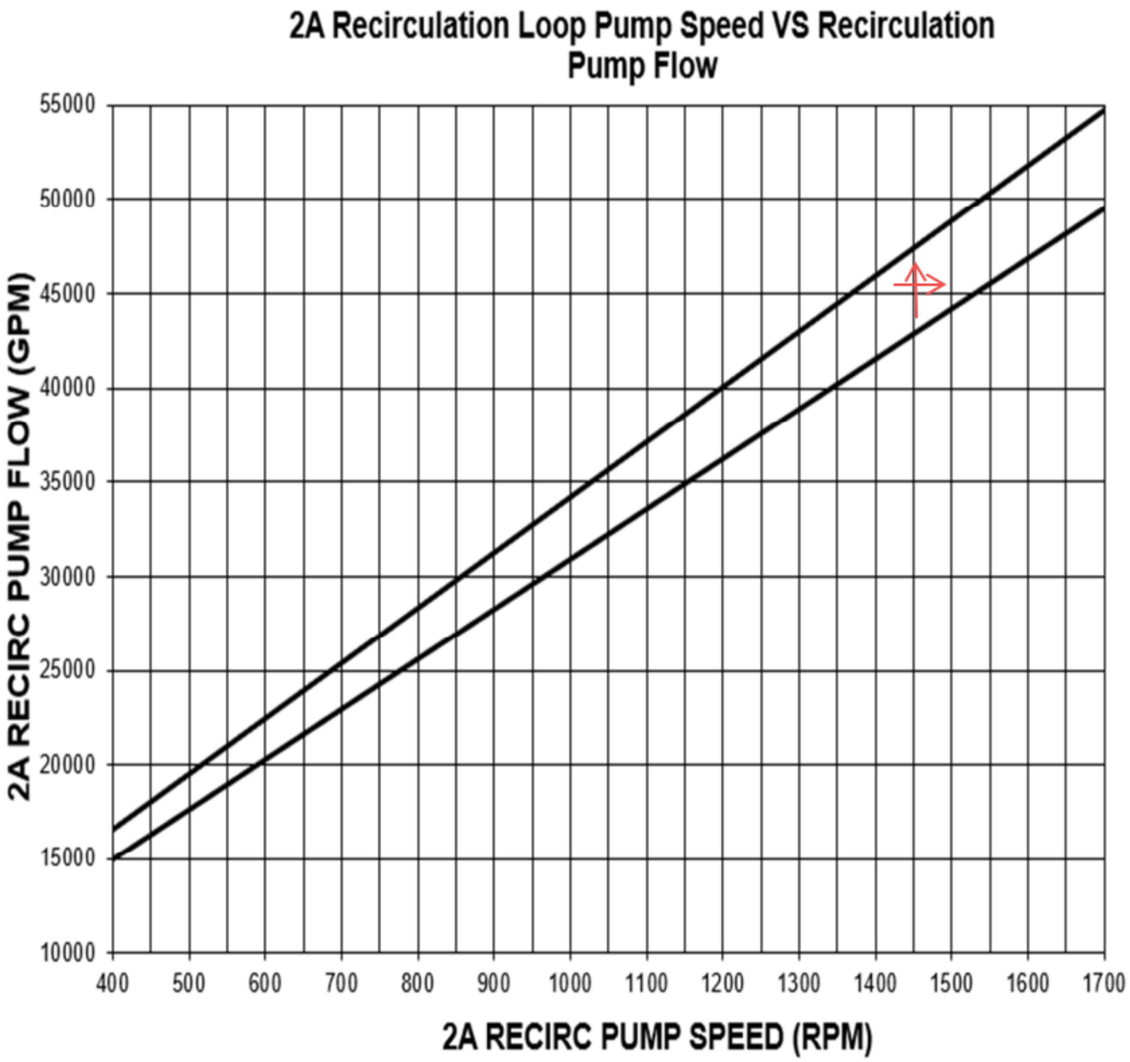
STEP / STANDARD	SAT / UNSAT
Step 10: [3] **IF** %WT is < 70% as recorded in Step 7.3[1], **THEN** **CHECK** Recirculation Loop Jet Pump Flow Mismatch recorded in Step 7.3[2] is ≤ 10.25 Mlb/hr. (Otherwise N/A) Expected Action(s): The candidate is expected to mark step as N/A.	______ SAT ______UNSAT ______N/A
Step 11: [4] **IF** %WT is ≥ 70% as recorded in Step 7.3[1], **THEN** **CHECK** Recirculation Loop Jet Pump Flow Mismatch recorded in Step 7.3[2] is ≤ 5.12 Mlb/hr. (Otherwise N/A) Expected Action(s): The candidate is expected to check Recirculation Loop Jet Pump Flow Mismatch recorded in Step 7.3[2] is ≤ 5.12 Mlb/hr (which is 0 - 3 Mlb/hr) and initials the Acceptance Criteria (AC) step.	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 12: **7.4.1 Loop 2A Recirculation Pump and Jet Pump Flow to Recirculation Pump Speed** [1] Using the 2A Pump Speed recorded in Step 7.2.2[1] and the 2A Pump Flow recorded in Step 7.2.2[2]: **CHECK** the plot falls between the two bold lines on Attachment 1 and **RECORD** below. Plot falls between the bold lines. Yes ☐ No ☐ Expected Action(s): The candidate is expected to check the plot falls between the two bold lines on Attachment 1 as shown below. Plot falls between the bold lines Yes ☒ No ☐	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT

BFN Unit 2	Jet Pump Mismatch and Operability	2-SR-3.4.2.1 Rev. 0051 Page 26 of 32

Attachment 1
(Page 1 of 1)

2A Recirculation Loop Pump Speed VS Recirculation Pump Flow
Good Thru June 1, 2022



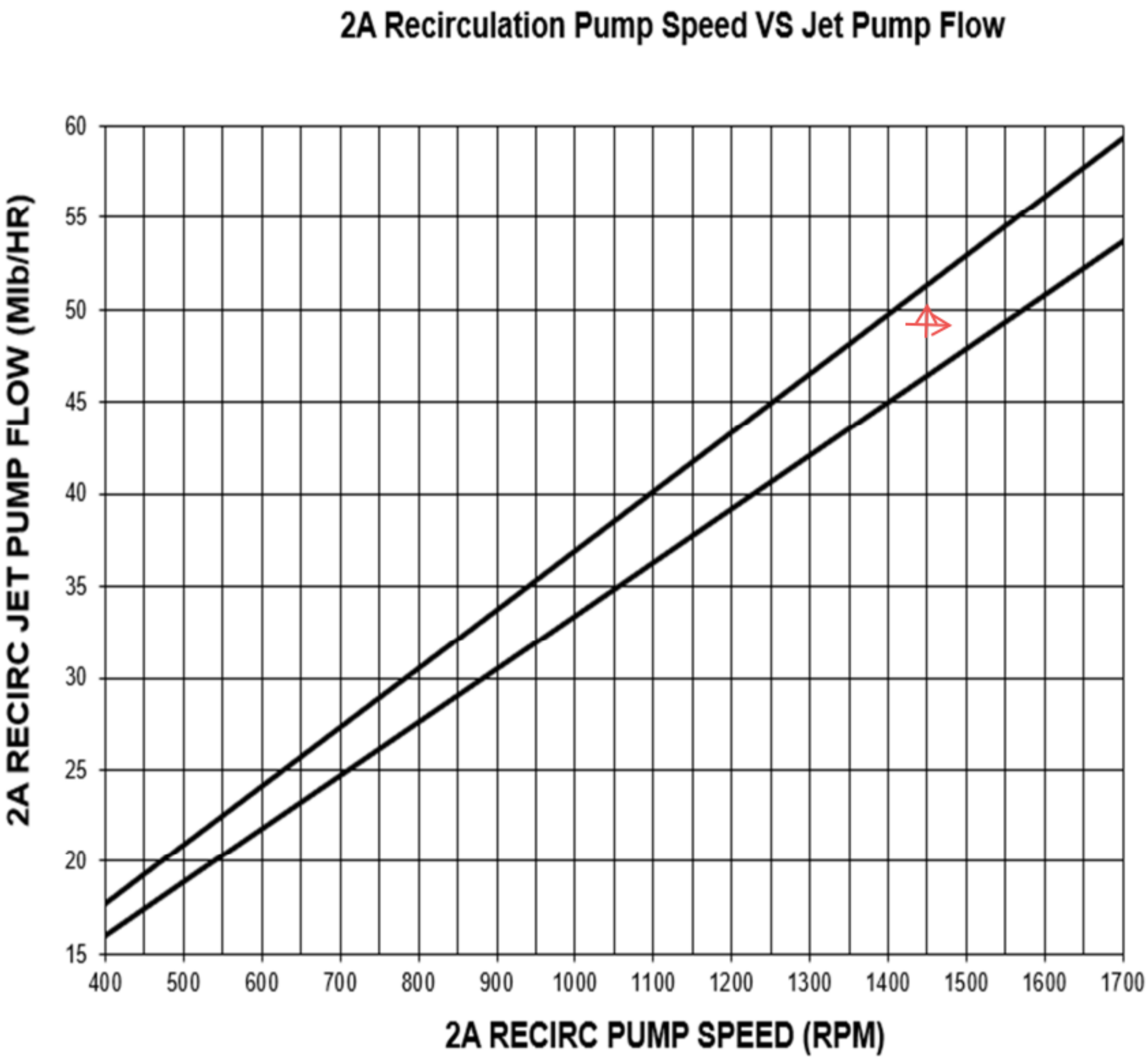
STEP / STANDARD	SAT / UNSAT
<u>Step 13:</u> [2] Using the 2A Pump Speed recorded in Step 7.2.2[1] and the 2A Jet Pump Flow in Step 7.2.2[3]: **CHECK** the plot falls between the two bold lines on Attachment 2 and **RECORD** below. Plot falls between the bold lines. Yes ☐ No ☐ <u>Expected Action(s):</u> The candidate is expected to check the plot falls between the two bold lines on Attachment 2 as shown below. Plot falls between the bold lines. Yes ☒ No ☐	**Critical Step** ______ SAT ______UNSAT ______N/A

STEP / STANDARD	SAT / UNSAT

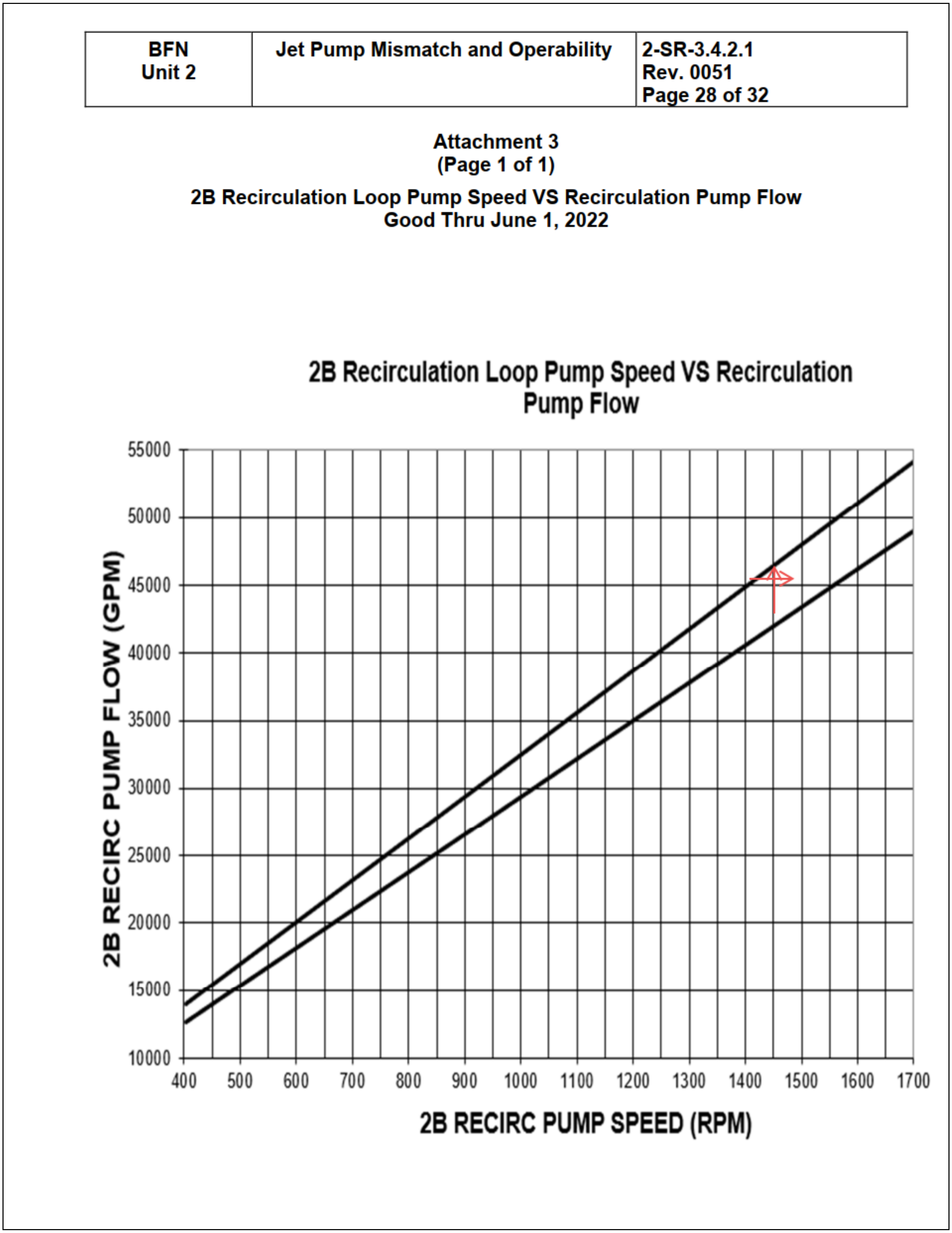
BFN Unit 2	**Jet Pump Mismatch and Operability**	**2-SR-3.4.2.1 Rev. 0051 Page 27 of 32**

Attachment 2
(Page 1 of 1)

2A Recirculation Pump Speed VS Jet Pump Flow
Good Thru June 1, 2022



STEP / STANDARD	SAT / UNSAT
<u>Step 14:</u> [3] Using Steps 7.4.1[1] and 7.4.1[2] from above: **DETERMINE** if the Jet Pump Loop 2A criteria is satisfied by marking below if both steps are marked as **Yes**. Jet Pump Loop 2A criteria is satisfied. Yes ☐ No ☐ <u>Expected Action(s):</u> The candidate is expected to determine if the Jet Pump Loop 2A criteria is satisfied by marking below if both steps are marked as **Yes**. Jet Pump Loop 2A criteria is satisfied. Yes ☒ No ☐	**Critical Step** ______ SAT ______UNSAT ______N/A
<u>Step 15:</u> **7.4.2 Loop 2B Recirculation Pump and Jet Pump Flow to Recirculation Pump Speed** [1] Using the 2B Pump Speed recorded in Step 7.2.2[1] and 2B Pump Flow recorded in Step 7.2.2[2]: **CHECK** the plot falls between the two bold lines on Attachment 3 and **RECORD** below. Plot falls between the bold lines. Yes ☐ No ☐ <u>Expected Action(s):</u> The candidate is expected to check the plot falls between the two bold lines on Attachment 3 as shown below. Plot falls between the bold lines. Yes ☒ No ☐	**Critical Step** ______ SAT ______UNSAT ______N/A



BFN Unit 2	Jet Pump Mismatch and Operability	2-SR-3.4.2.1 Rev. 0051 Page 28 of 32

Attachment 3
(Page 1 of 1)

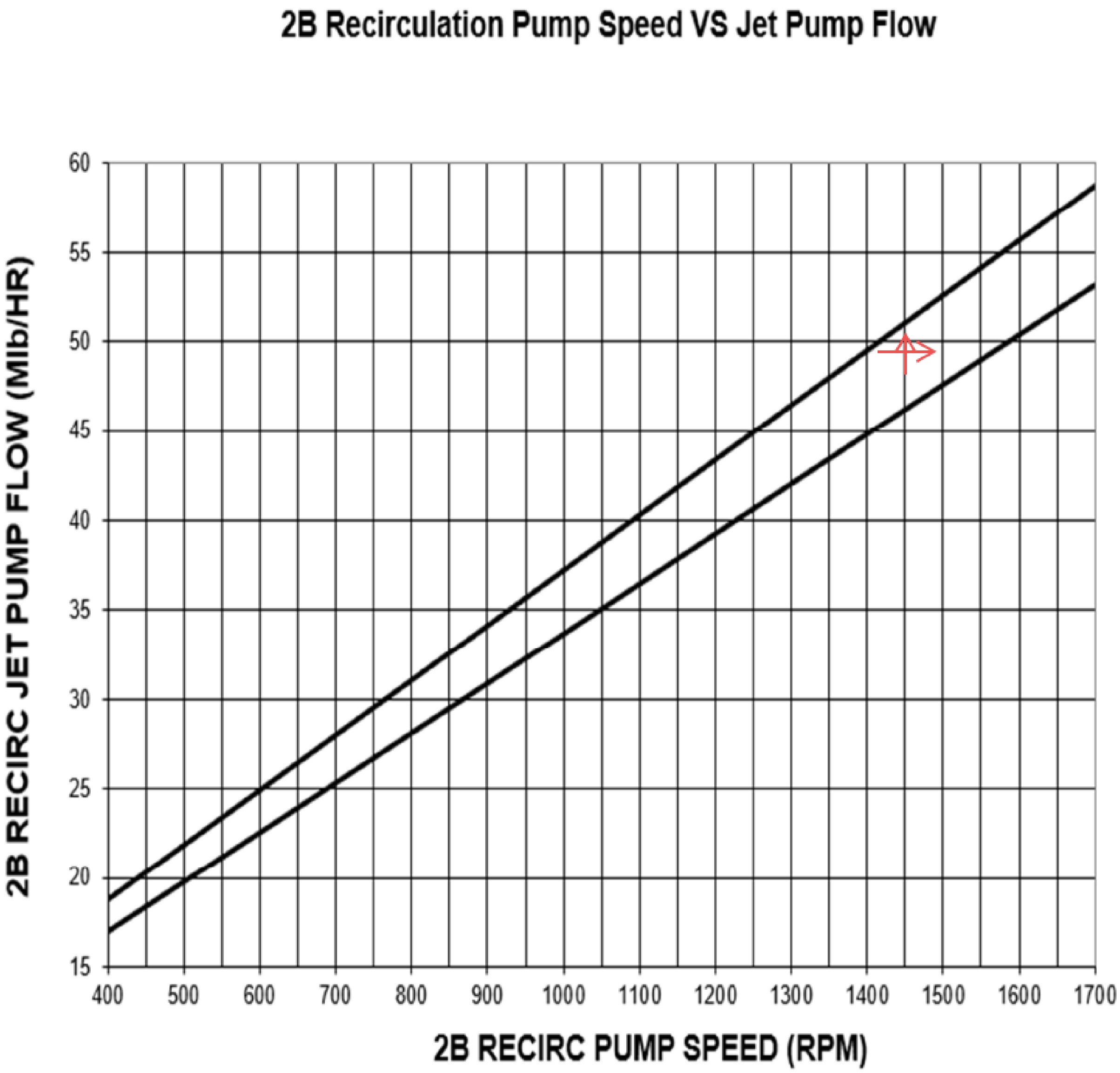
2B Recirculation Loop Pump Speed VS Recirculation Pump Flow
Good Thru June 1, 2022

STEP / STANDARD	SAT / UNSAT
Step 16: [2] Using the 2B Pump Speed recorded in Step 7.2.2[1] and 2B Jet Pump Flow recorded in Step 7.2.2[3]: **CHECK** the plot falls between the two bold lines on Attachment 4 and **RECORD** below. Plot falls between the bold lines. Yes ☐ No ☐ Expected Action(s): The candidate is expected to check the plot falls between the two bold lines on Attachment 4 as shown below. Plot falls between the bold lines. Yes ☒ No ☐	**Critical Step** ______ SAT ______UNSAT ______N/A

BFN Unit 2	Jet Pump Mismatch and Operability	2-SR-3.4.2.1 Rev. 0051 Page 29 of 32

Attachment 4
(Page 1 of 1)

2B Recirculation Pump Speed VS Jet Pump Flow
Good Thru June 1, 2022



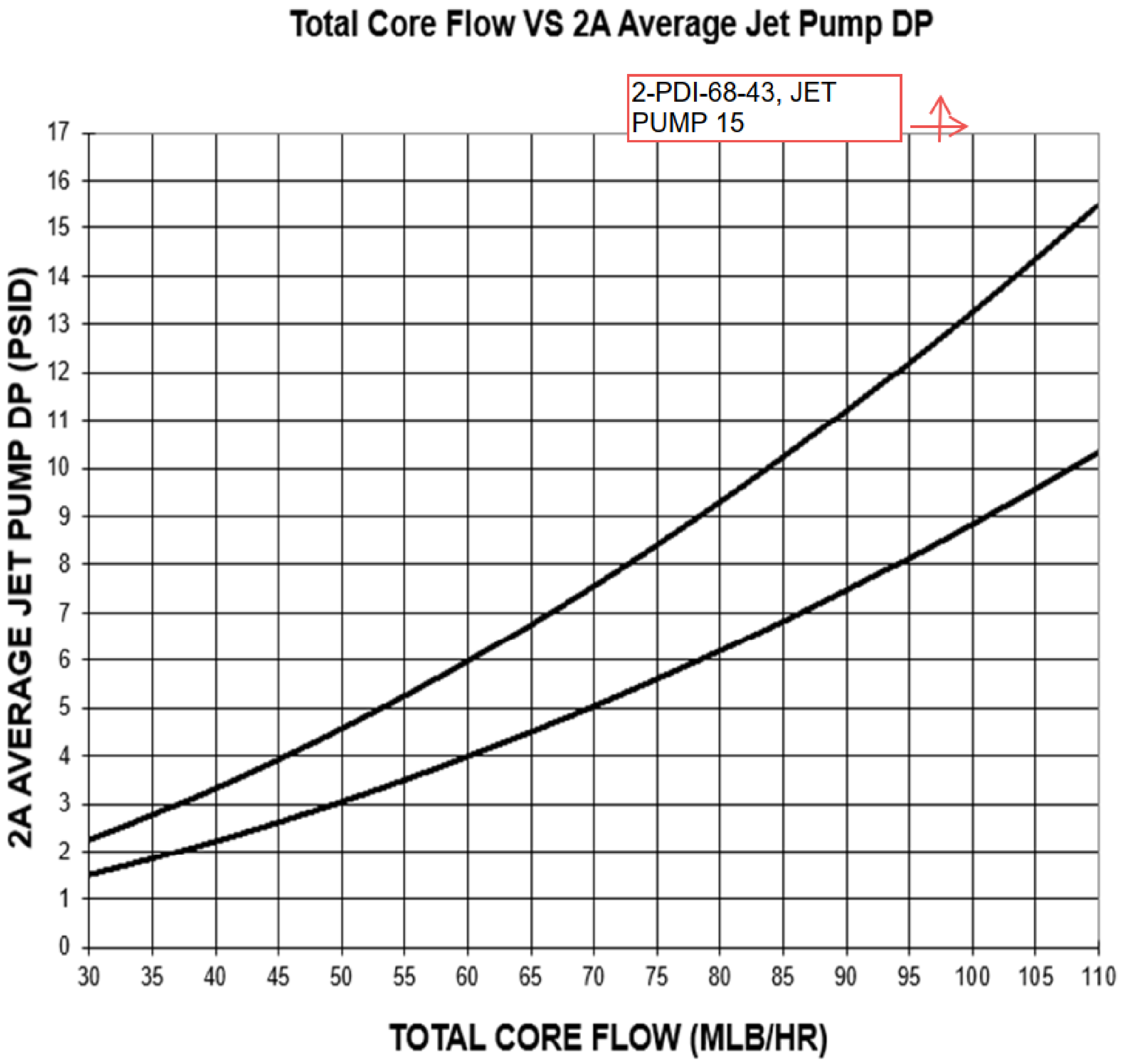
STEP / STANDARD	SAT / UNSAT
Step 17: [3] Using Steps 7.4.2[1] and 7.4.2[2] from above: **DETERMINE** if the Jet Pump Loop 2B criteria is satisfied by marking below if both steps are marked as Yes. Jet Pump Loop 2B criteria is satisfied. Yes ☐ No ☐ Expected Action(s): The candidate is expected to determine if the Jet Pump Loop 2B criteria is satisfied by marking below if both steps are marked as Yes. Jet Pump Loop 2B criteria is satisfied. Yes ☒ No ☐	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 18: **7.4.3 Recirculation Jet Pump Diffuser to Lower Plenum Differential Pressure Verification:** [1] Using the individual 2A Jet Pump DP's recorded in Step 7.2.3[1] **CHECK** each individual Jet Pump DP recorded fall between the two bold lines on Attachment 5 for the recorded Total Flow in Step 7.2.1[3] and **RECORD** results below. 2A Individual DP's are between the bold lines. Yes ☐ No ☐ Expected Action(s): The candidate is expected to check each individual Jet Pump DP recorded fall between the two bold lines on Attachment 5 for the recorded Total Flow in Step 7.2.1[3] and **RECORD** results below. 2A Individual DP's are between the bold lines. Yes ☐ No ☒	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER NOTE: See next page for Attachment 5, 2-PDI-68-43, JET PUMP 15 is 17.5 to 18.5 psid.	

STEP / STANDARD	SAT / UNSAT

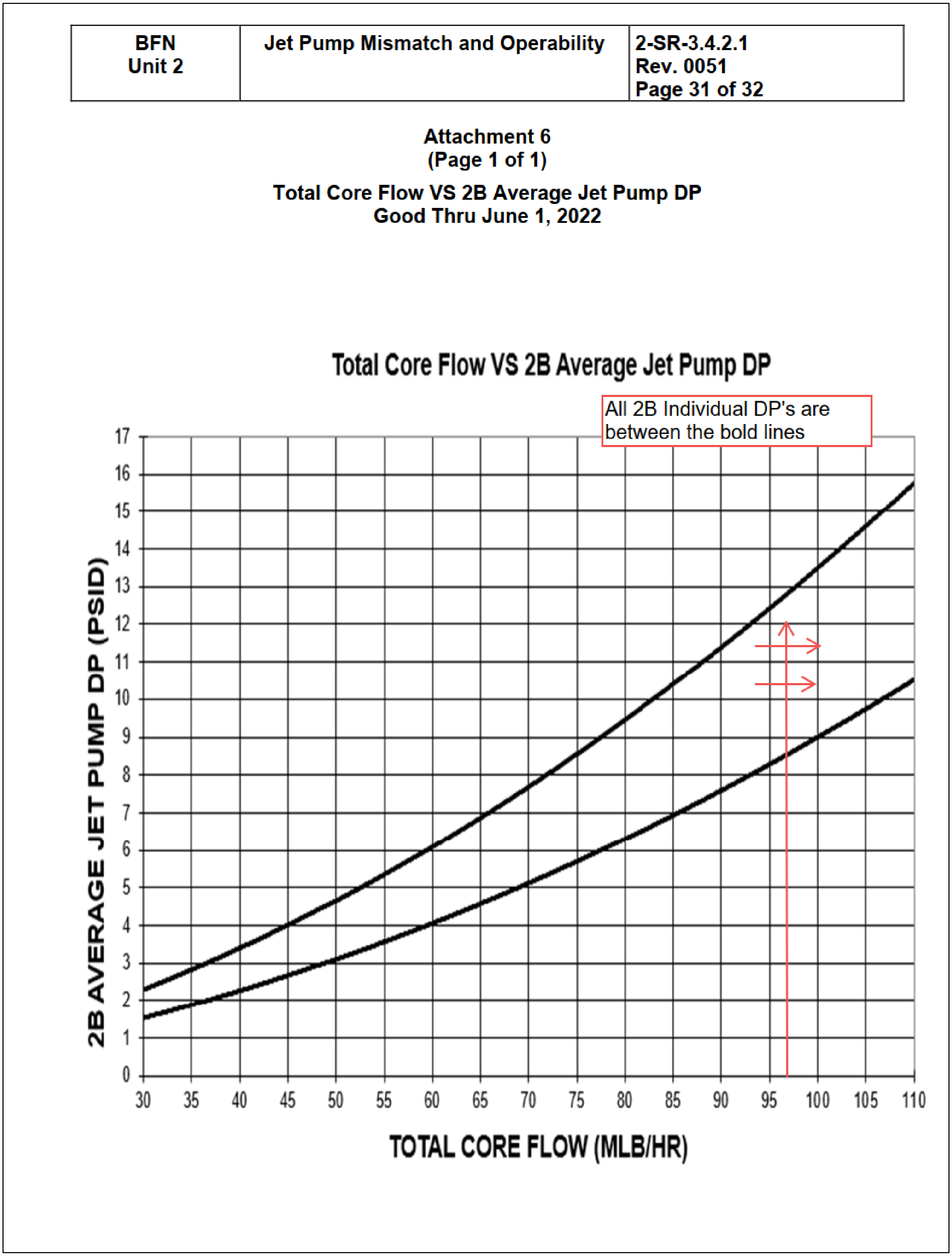
BFN **Unit 2**	**Jet Pump Mismatch and Operability**	**2-SR-3.4.2.1** **Rev. 0051** **Page 30 of 32**

Attachment 5
(Page 1 of 1)

Total Core Flow VS 2A Average Jet Pump DP
Good Thru June 1, 2022



STEP / STANDARD	SAT / UNSAT
Step 19: [2] Using the individual 2B Jet Pump DP's recorded in Step 7.2.3[1] **CHECK** each individual Jet Pump DP recorded fall between the two bold lines on Attachment 6 for the recorded Total Flow in Step 7.2.1[3] and **RECORD** results below. 2B Individual DP's are between the bold lines. Yes ☐ No ☐ Expected Action(s): The candidate is expected to check each individual Jet Pump DP recorded fall between the two bold lines on Attachment 6 for the recorded Total Flow in Step 7.2.1[3] and **RECORD** results below. 2B Individual DP's are between the bold lines. Yes ☒ No ☐	**Critical Step** ______ SAT ______UNSAT ______N/A
EXAMINER NOTE: See next page for Attachment 6.	



BFN Unit 2	Jet Pump Mismatch and Operability	2-SR-3.4.2.1 Rev. 0051 Page 31 of 32

Attachment 6
(Page 1 of 1)

Total Core Flow VS 2B Average Jet Pump DP
Good Thru June 1, 2022

All 2B Individual DP's are between the bold lines

STEP / STANDARD	SAT / UNSAT
Step 20: [3] Using Steps 7.4.3[1] and 7.4.3[2] **DETERMINE** whether the Recirculation Jet Pump Diffuser to Lower Plenum Differential Pressure Verification criteria is satisfied by marking below if both steps are marked as Yes. Jet Pump Diffuser to Lower Plenum Differential Pressure Verification criteria is satisfied Yes ☐ No ☐ Expected Action(s): The candidate is expected to determine whether the Recirculation Jet Pump Diffuser to Lower Plenum Differential Pressure Verification criteria is satisfied by marking below if both steps are marked as Yes. Jet Pump Diffuser to Lower Plenum Differential Pressure Verification criteria is **NOT** satisfied. Yes ☐ No ☒	**Critical Step** ______ SAT ______UNSAT ______N/A
Step 21: **7.4.4 Engineering Judgment/Review Criteria** [1] **IF** any of the following conditions apply: • Following Refueling Outage. (See Caution above) **OR** • The Reactor is in Single Loop Operation (See Caution above) **OR** • If point(s) plotted in Sections 7.4.1 or 7.4.2, AND 7.4.3 fall on or outside the bolded lines, to determine if the graph(s) need updating, **THEN** **PERFORM** Attachment 7, Engineering Judgment/Review: (Otherwise N/A this step and Mark all of Attachment 7 as N/A) Expected Action(s): The candidate is expected to evaluate step and determines that this step is N/A. However, the candidate may ask for an Engineering Judgement (see below EXAMINER CUE).	______ SAT ______UNSAT ______N/A
EXAMINER CUE: JPM Step 21 above – If the candidate ask about Attachment 7, Engineering Judgment/Review, state “The evaluation of Attachment 7, Engineering Judgment/Review is in progress, continue with task”.	

STEP / STANDARD	SAT / UNSAT
Step 22: **7.4.5 Operability Determination**	**Critical Step**

[1] **MARK** the appropriate criteria results for the following.
(N/A any criteria not performed)

Steps	Criteria Results	Yes	No	N/A
7.4.1[3] and 7.4.2[3]	Both Jet Pump Loops steps are marked as YES	☐	☐	☐
7.4.3[3]	Jet Pump DP to criteria is marked as YES.	☐	☐	☐
Attachment 7	Engineering Evaluation is marked as YES.	☐	☐	☐

Expected Action(s):

The candidate is expected to mark the appropriate criteria results for the following. (N/A any criteria not performed)

Steps	Criteria Results	Yes	No	N/A
7.4.1[3] and 7.4.2[3]	Both Jet Pump Loops steps are marked as YES	☒	☐	☐
7.4.3[3]	Jet Pump DP to criteria is marked as YES.	☐	☒	☐
Attachment 7	Engineering Evaluation is marked as YES.	☐	☐	☒

______ SAT

______UNSAT

______N/A

EXAMINER NOTE: JPM Step 22 above – Marking Attachment 7 in the table is NOT Critical.

Step 23: **7.4.5 Operability Determination**

[2] Using the Criteria Results in Step 7.4.5[1]

ENSURE at least one Criteria Results is satisfied and marked as YES.

Expected Action(s):

The candidate is expected to determine at least one Criteria Results is satisfied from 7.4.5[1] and the Acceptance Criteria (AC) step is initialed.

______ SAT

______UNSAT

______N/A

EXAMINER CUE: Once JPM Step 23 is complete, inform the candidate “Another Operator will continue the 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:

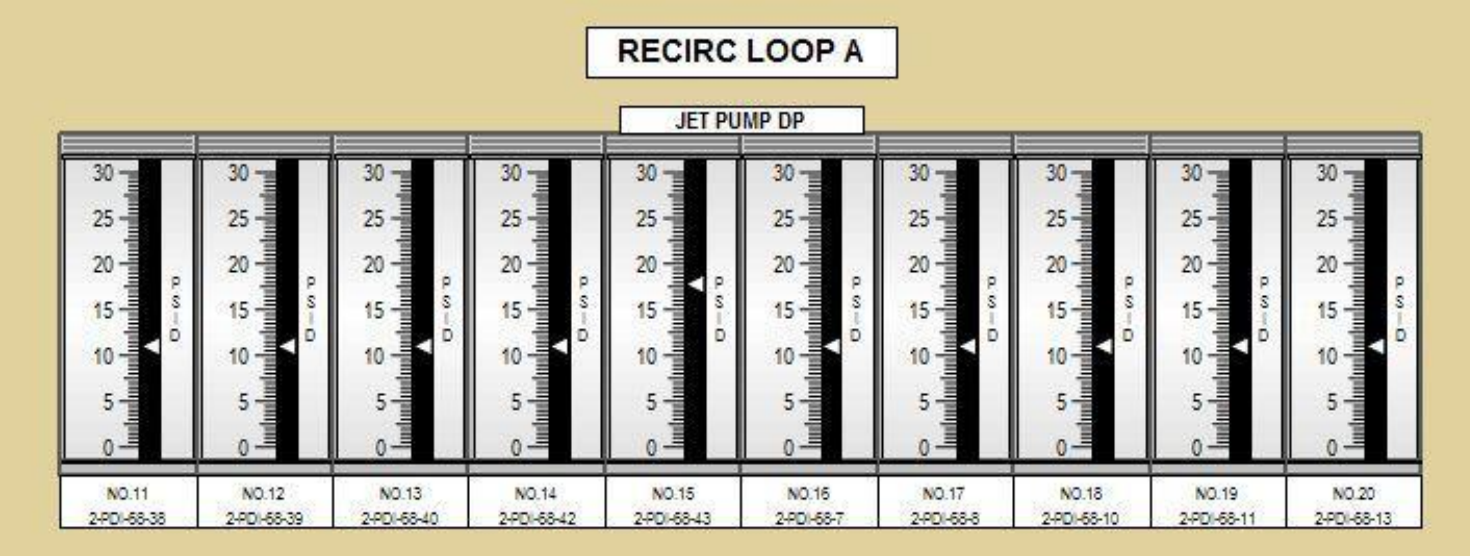
Unit 2 is operating at 100% Rated Thermal Power (RTP) and you are the Unit 2 Operator.

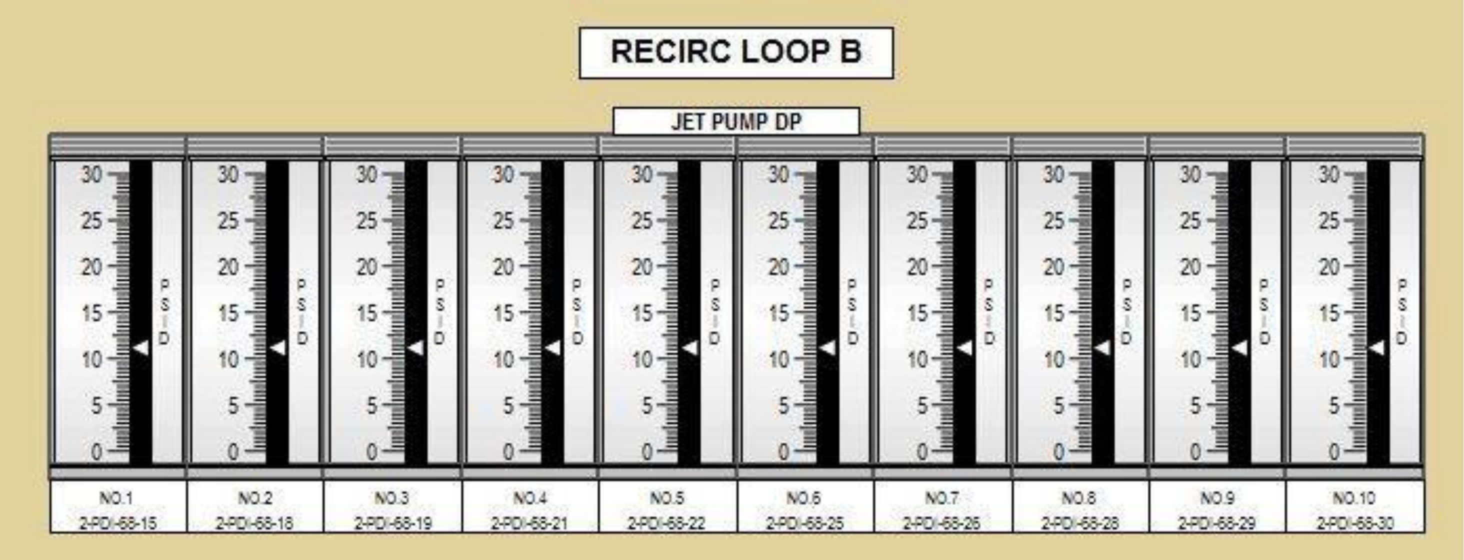
INITIATING CUE:

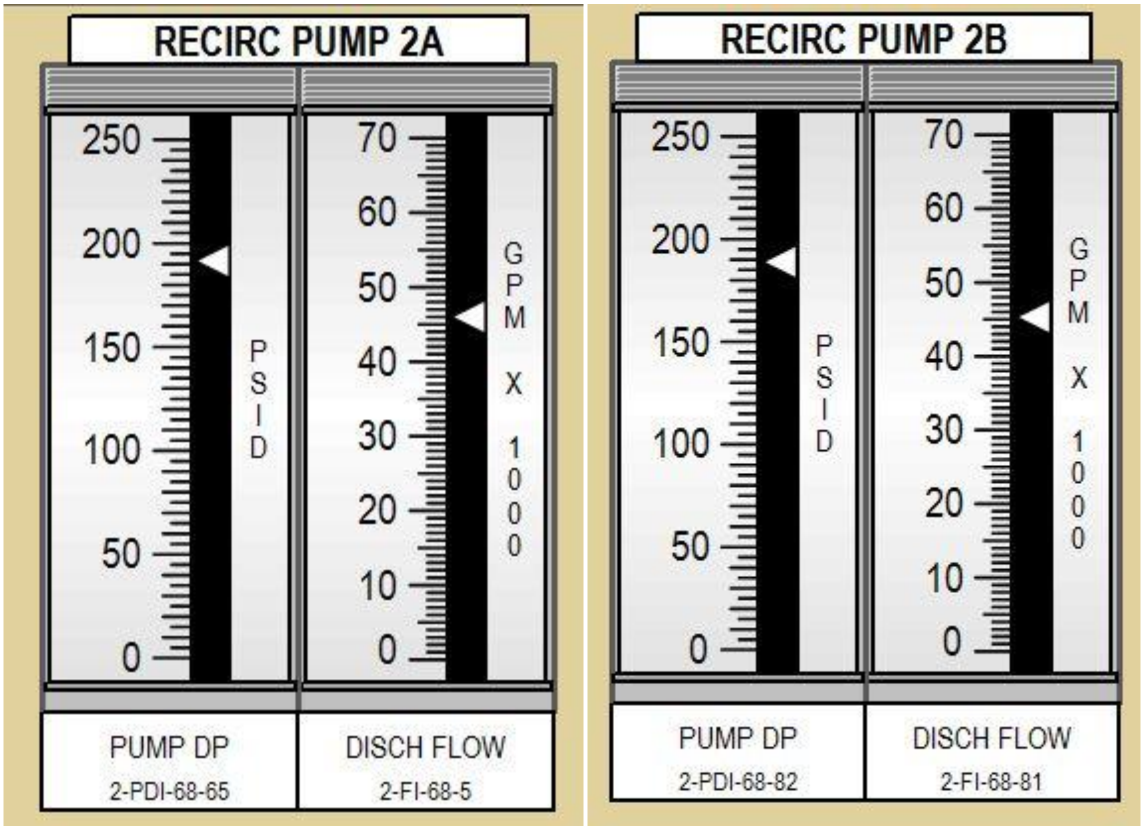
Perform 2-SR-3.4.2.1; Jet Pump Mismatch and Operability starting at step 7.2.1, Core Power and Flow Readings using the provided readings below:

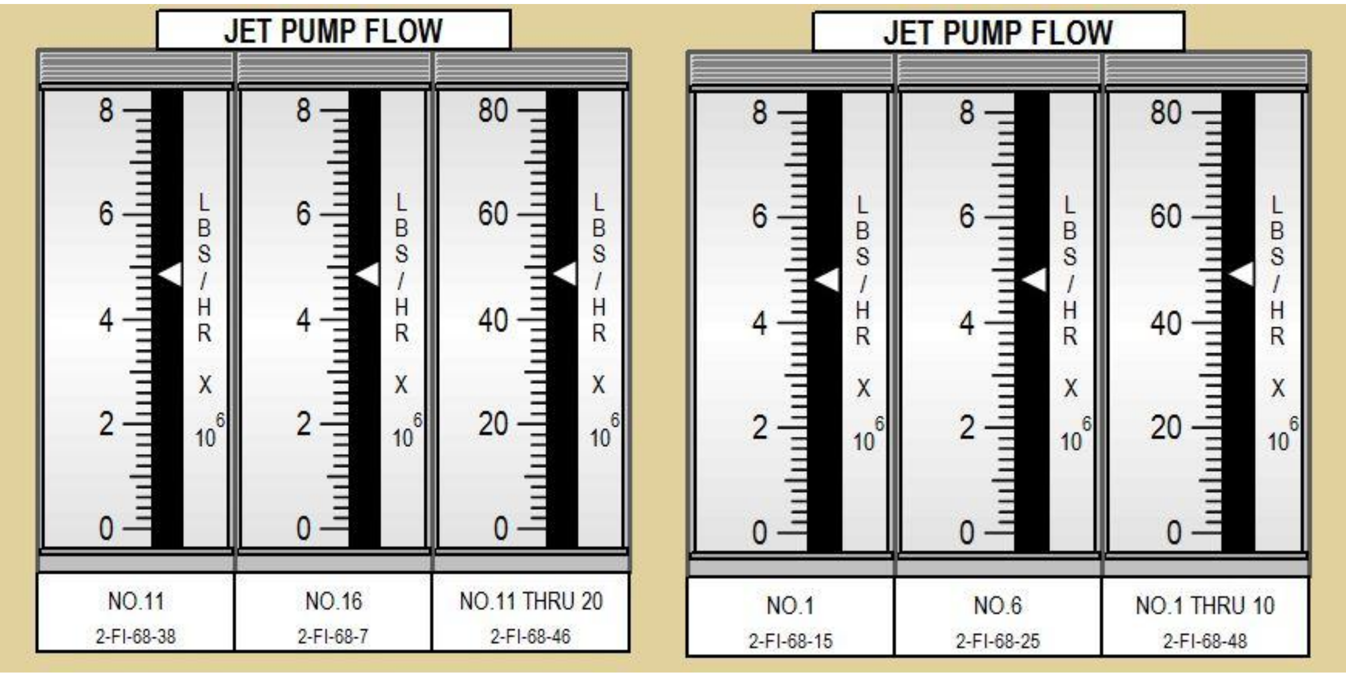
CALC002 = 3952 CMWT
2-XR-68-50 Green Pen = 18.0 psid
2-XR-68-50 Red Pen = 97.1 Mlb/hr

Recirc Motor Speeds
2A - 2-SI-68-59 = 1456 rpm
2B - 2-SI-68-71 = 1456 rpm


RECIRC LOOP A
JET PUMP DP
30
25
20
15
10
5
0
PSID
NO.11 2-PDI-68-38
NO.12 2-PDI-68-39
NO.13 2-PDI-68-40
NO.14 2-PDI-68-42
NO.15 2-PDI-68-43
NO.16 2-PDI-68-7
NO.17 2-PDI-68-8
NO.18 2-PDI-68-10
NO.19 2-PDI-68-11
NO.20 2-PDI-68-13


RECIRC LOOP B
JET PUMP DP
30
25
20
15
10
5
0
PSID
NO.1 2-PDI-68-15
NO.2 2-PDI-68-18
NO.3 2-PDI-68-19
NO.4 2-PDI-68-21
NO.5 2-PDI-68-22
NO.6 2-PDI-68-25
NO.7 2-PDI-68-26
NO.8 2-PDI-68-28
NO.9 2-PDI-68-29
NO.10 2-PDI-68-30


RECIRC PUMP 2A
250
200
150
100
50
0
PSID
70
60
50
40
30
20
10
0
GPM X 1000
PUMP DP
2-PDI-68-65
DISCH FLOW
2-FI-68-5
RECIRC PUMP 2B
250
200
150
100
50
0
PSID
70
60
50
40
30
20
10
0
GPM X 1000
PUMP DP
2-PDI-68-82
DISCH FLOW
2-FI-68-81


JET PUMP FLOW
8
6
4
2
0
LBS/HR X 10^6
NO.11
2-FI-68-38
NO.16
2-FI-68-7
80
60
40
20
0
NO.11 THRU 20
2-FI-68-46
JET PUMP FLOW
NO.1
2-FI-68-15
NO.6
2-FI-68-25
NO.1 THRU 10
2-FI-68-48

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Review a Radiological Work Permit (RWP)
JPM NUMBER:	682	REVISION:	3

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*

OPERATOR: ________________________________ JPM Number: 682

RO ____ SRO ____ DATE: ________

TASK STANDARD: The Examinee is expected to calculate an expected dose between 120 to 127 mrem and determine that the task cannot be completed for the given Radiation Work Permit (RWP).

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
3	03/13/2021	2	Updated task standard

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:

- 10 minutes to close the valve
- 15 minutes 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 **CAN/CANNOT** 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
<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

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 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:

- 10 minutes to close the valve
- 15 minutes 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.

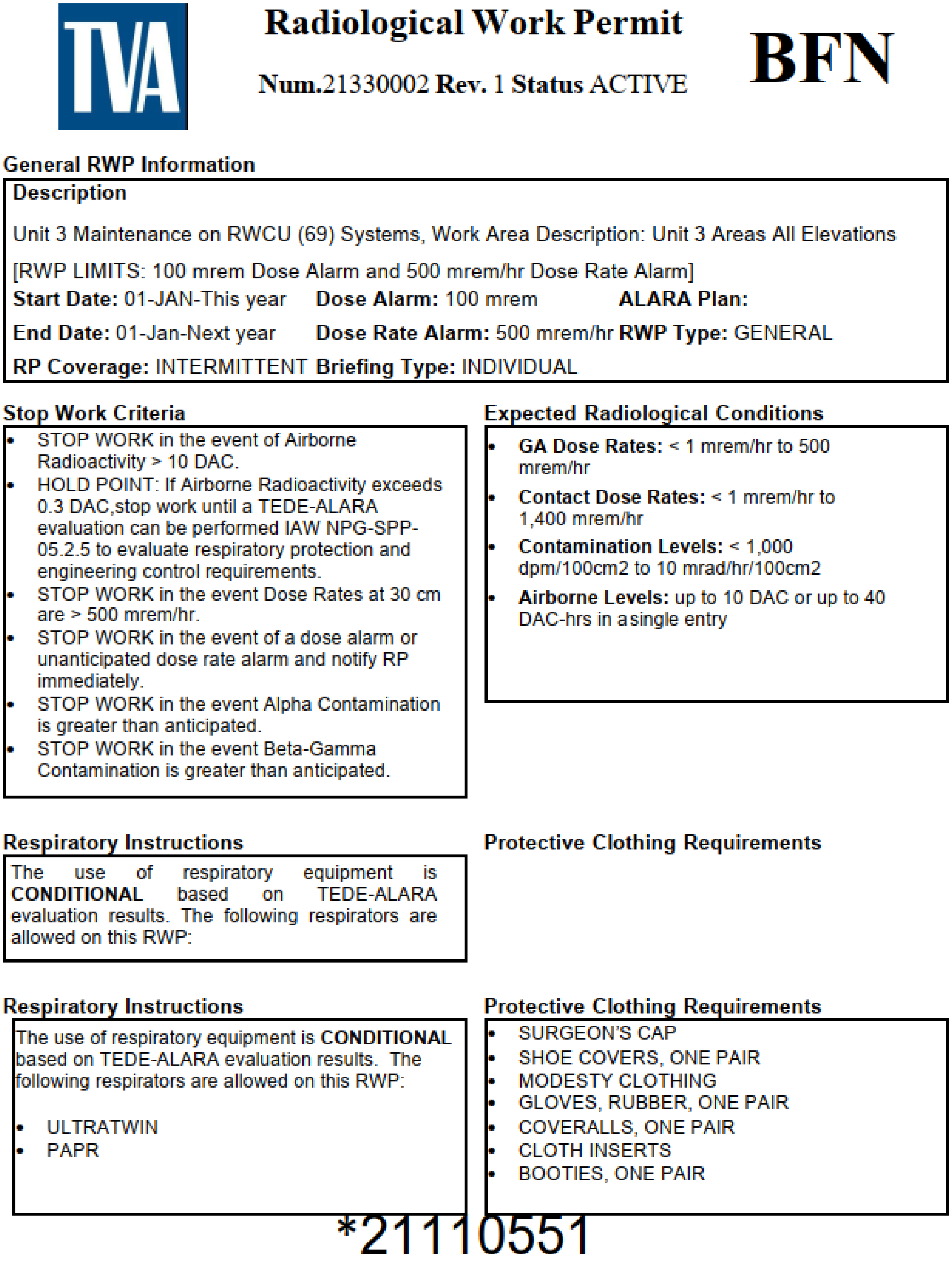
Use the attached Radiological Work Permit **(**RWP) to accomplish your task

INITIATING CUE:

Given the conditions above, determine if this task **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:

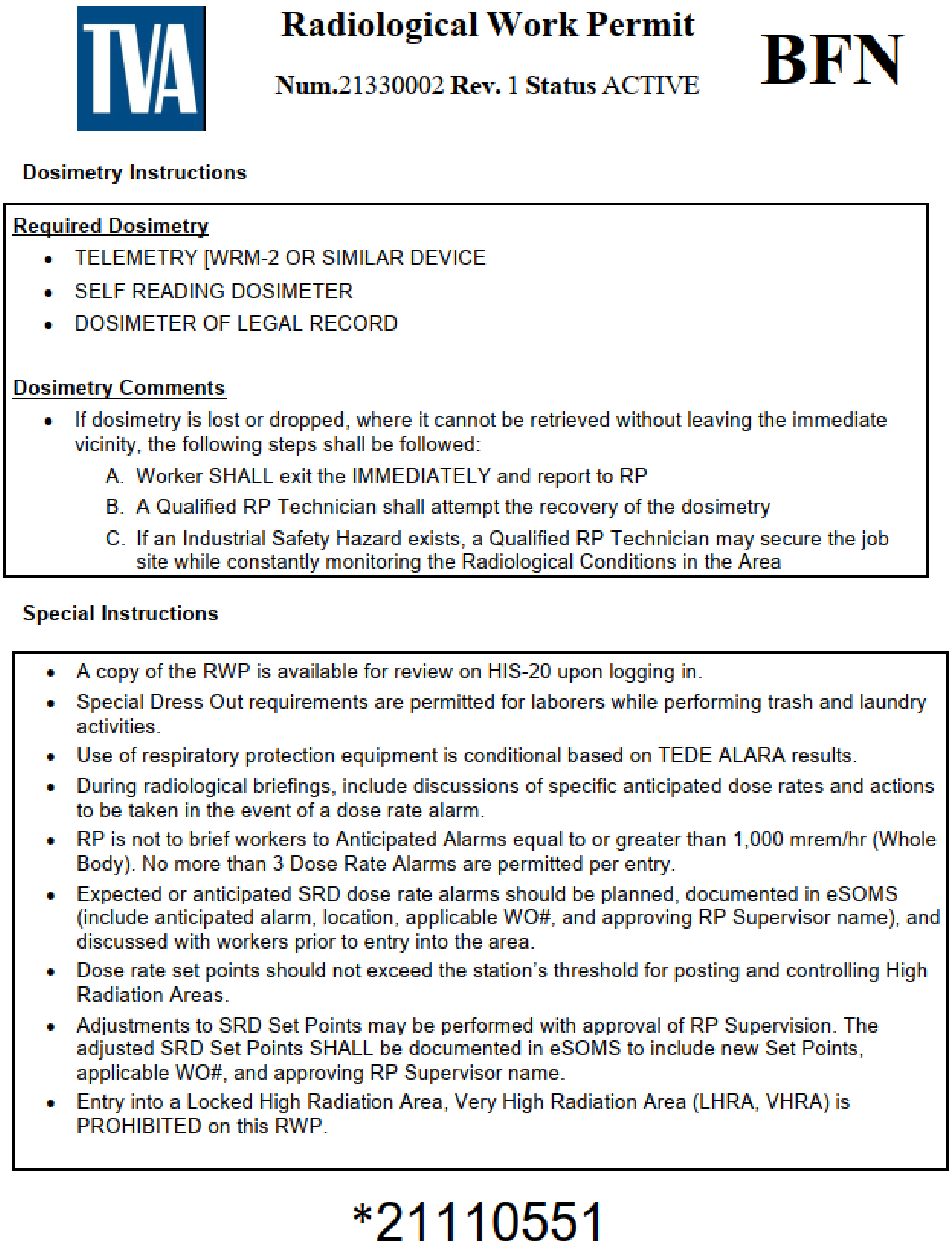
- ULTRATWIN
- PAPR

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

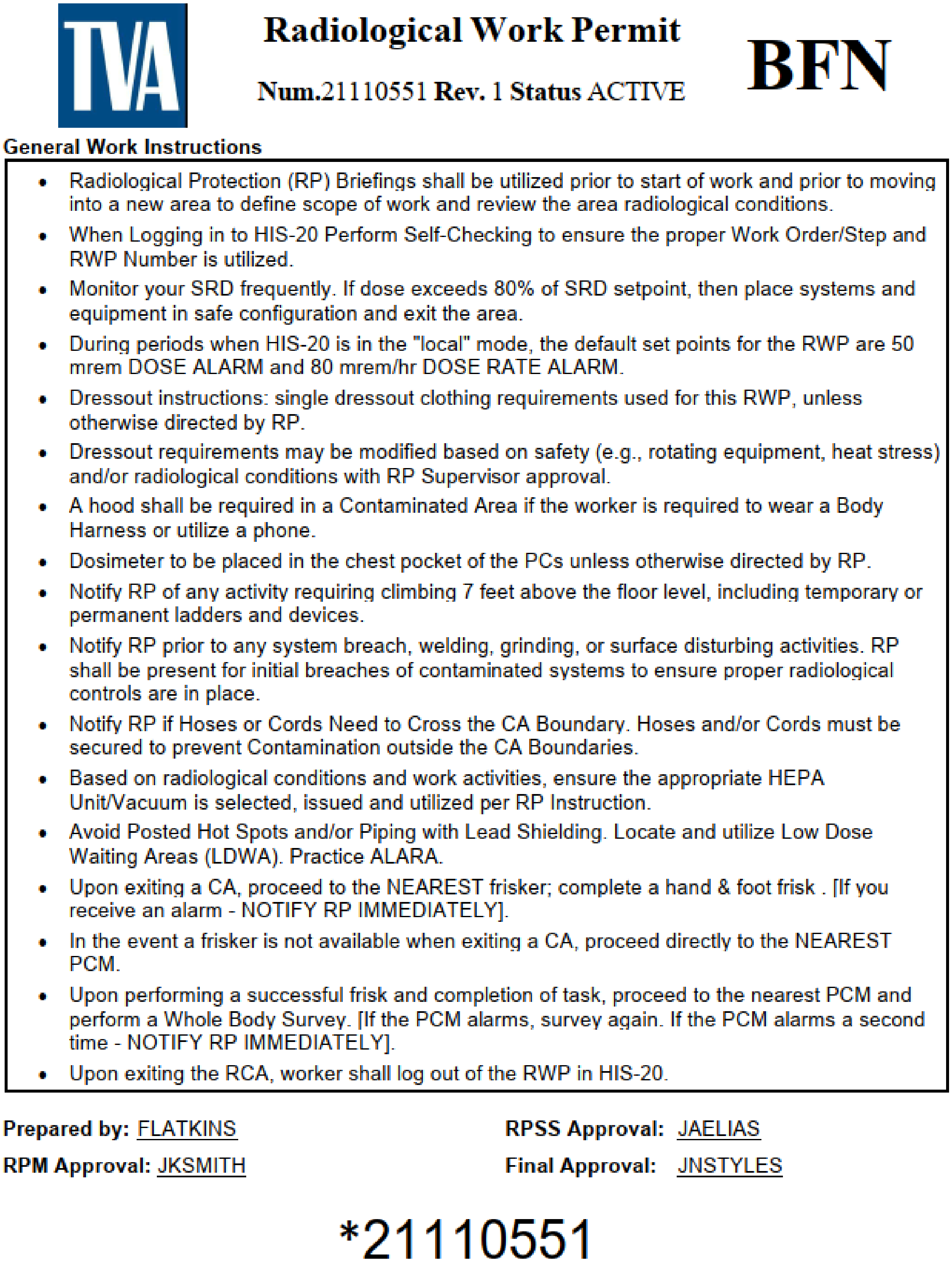
- If dosimetry is lost or dropped, where it cannot be retrieved without leaving the immediate vicinity, the following steps shall be followed:
  - A. Worker SHALL exit the IMMEDIATELY and report to RP
  - B. A Qualified RP Technician shall attempt the recovery of the dosimetry
  - C. If an Industrial 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:	Drywell Leakage Calculation
JPM NUMBER:	556-SRO	REVISION:	1

TASK APPLICABILITY:	☒SRO	☐STA	☐UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	S-000-SU-08 / Review Instrument Checks and Observations SI			
K/A RATINGS:	SRO 4.7			
K/A STATEMENT:	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.			
RELATED PRA INFORMATION:	None			
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: 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: 556-SRO

RO ____ SRO ____ DATE: _______

TASK STANDARD: The Examinee is expected to calculate Drywell Floor and Equipment Sump leak rates to support a surveillance and determine if leak rates meet Acceptance Criteria (AC). Determine what Technical Specification Required Action(s) must be taken (if any).

Operator Fundamental evaluated:
OF-3 Operating the Plant with a Conservative Bias

PRA: NA

REFERENCES/PROCEDURES NEEDED: 2-SR-2, Instrument Checks and Observations,
Unit 2 Technical Specification 3.4.4

VERIFICATION TIME: 20 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
1	02/01/2022	ALL	Updated JPM

Procedure Revisions

Procedure	Revision
2-SR-2	87
Unit 2 Tech Spec 3.4.4	Amend. 253

****

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 2 is operating at 100% Rated Thermal Power (RTP), online for 100 days.

It is Saturday at 0800, and the Drywell Floor and Equipment Drain sumps have just completed pumping down.

The 0800 readings for 2-SR-2, Instrument Checks and Observations, are as follows:

Drywell Floor Drain	22643 gals
Drywell Equipment Drain	28005 gals

INITIATING CUE:

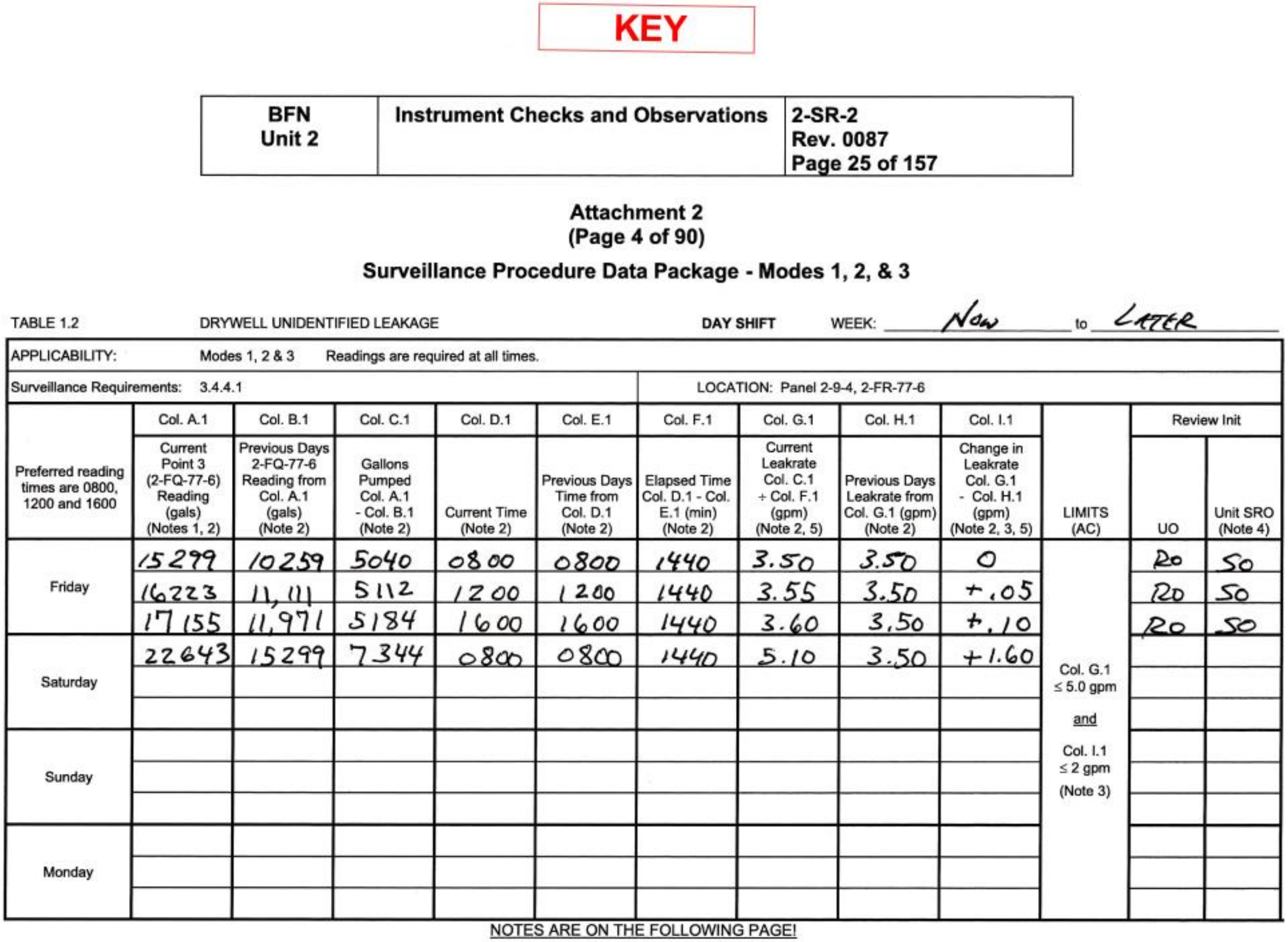
You are the Nuclear Unit Senior Operator (NUSO) required to complete the following:

- 2-SR-2, Instrument Checks and Observations, for the Drywell Floor and Equipment Drain Sumps

- Determine if **ALL** Acceptance Criteria (AC) is met

- Determine what Technical Specification Required Action(s) must be taken (if any)?

Answer:



KEY

BFN Unit 2	Instrument Checks and Observations	2-SR-2 Rev. 0087 Page 25 of 157

Attachment 2
(Page 4 of 90)

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

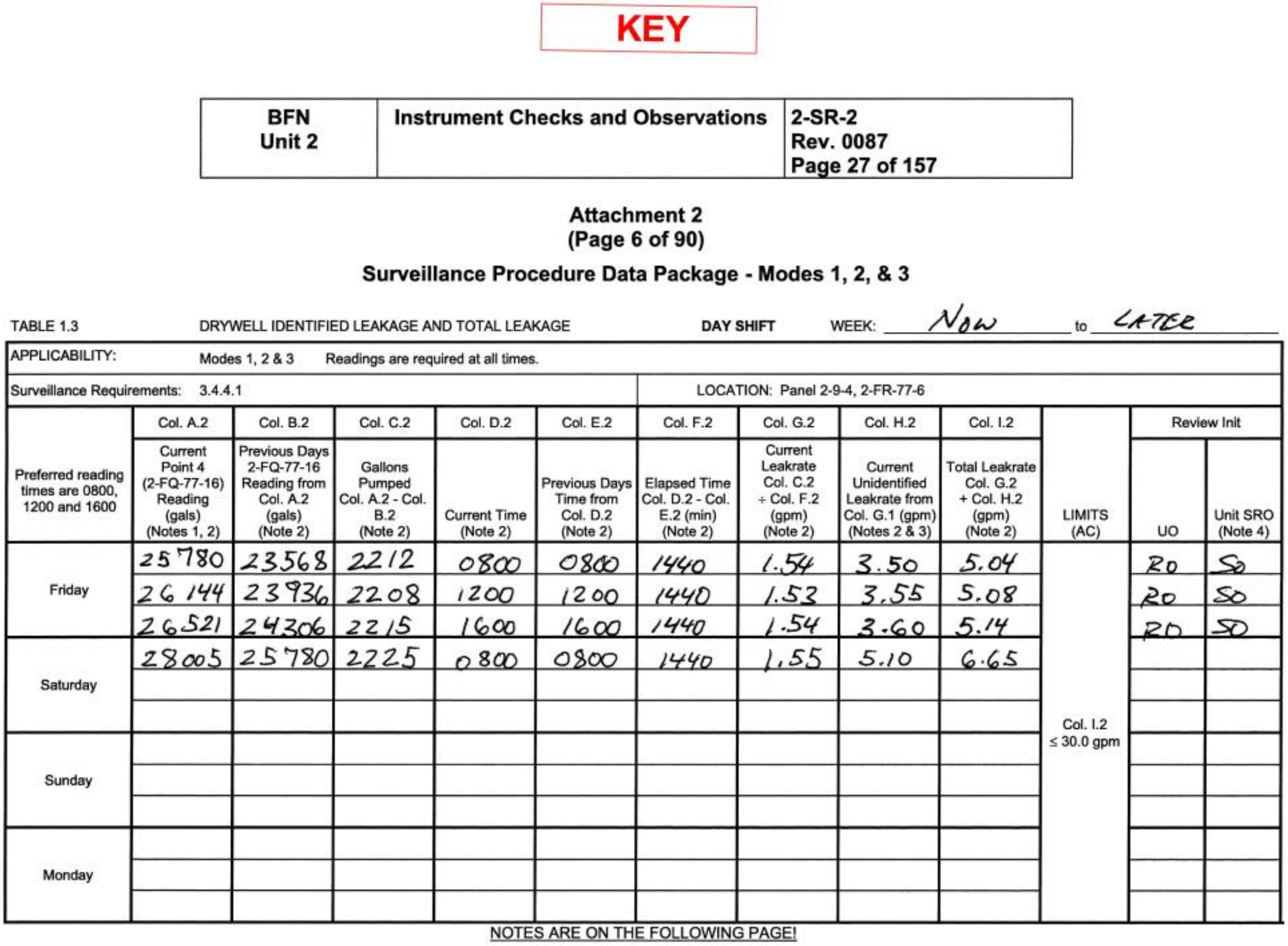
TABLE 1.2 DRYWELL UNIDENTIFIED LEAKAGE **DAY SHIFT** WEEK: Now to LATER

APPLICABILITY: Modes 1, 2 & 3 Readings are required at all times.

Surveillance Requirements: 3.4.4.1 LOCATION: Panel 2-9-4, 2-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 (2-FQ-77-6) Reading (gals) (Notes 1, 2)	Previous Days 2-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 SRO (Note 4)
Friday	15299	10259	5040	0800	0800	1440	3.50	3.50	0		RO	SO
	16223	11,111	5112	1200	1200	1440	3.55	3.50	+.05		RO	SO
	17155	11,971	5184	1600	1600	1440	3.60	3.50	+.10		RO	SO
Saturday	22643	15299	7344	0800	0800	1440	5.10	3.50	+1.60	Col. G.1 ≤ 5.0 gpm and Col. I.1 ≤ 2 gpm (Note 3)		
Sunday												
Monday												

NOTES ARE ON THE FOLLOWING PAGE!



KEY

BFN **Unit 2**	**Instrument Checks and Observations**	**2-SR-2** **Rev. 0087** **Page 27 of 157**

Attachment 2
(Page 6 of 90)

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

TABLE 1.3 DRYWELL IDENTIFIED LEAKAGE AND TOTAL LEAKAGE **DAY SHIFT** WEEK: Now to LATER

APPLICABILITY: Modes 1, 2 & 3 Readings are required at all times.

Surveillance Requirements: 3.4.4.1 LOCATION: Panel 2-9-4, 2-FR-77-6

	Col. A.2	Col. B.2	Col. C.2	Col. D.2	Col. E.2	Col. F.2	Col. G.2	Col. H.2	Col. I.2		Review Init	
Preferred reading times are 0800, 1200 and 1600	Current Point 4 (2-FQ-77-16) Reading (gals) (Notes 1, 2)	Previous Days 2-FQ-77-16 Reading from Col. A.2 (gals) (Note 2)	Gallons Pumped Col. A.2 - Col. B.2 (Note 2)	Current Time (Note 2)	Previous Days Time from Col. D.2 (Note 2)	Elapsed Time Col. D.2 - Col. E.2 (min) (Note 2)	Current Leakrate Col. C.2 ÷ Col. F.2 (gpm) (Note 2)	Current Unidentified Leakrate from Col. G.1 (gpm) (Notes 2 & 3)	Total Leakrate Col. G.2 + Col. H.2 (gpm) (Note 2)	LIMITS (AC)	UO	Unit SRO (Note 4)
Friday	25780	23568	2212	0800	0800	1440	1.54	3.50	5.04	Col. I.2 ≤ 30.0 gpm	RO	SO
	26144	23936	2208	1200	1200	1440	1.53	3.55	5.08		RO	SO
	26521	24306	2215	1600	1600	1440	1.54	3.60	5.14		RD	SO
Saturday	28005	25780	2225	0800	0800	1440	1.55	5.10	6.65			
Sunday												
Monday												

NOTES ARE ON THE FOLLOWING PAGE!

START TIME:________________

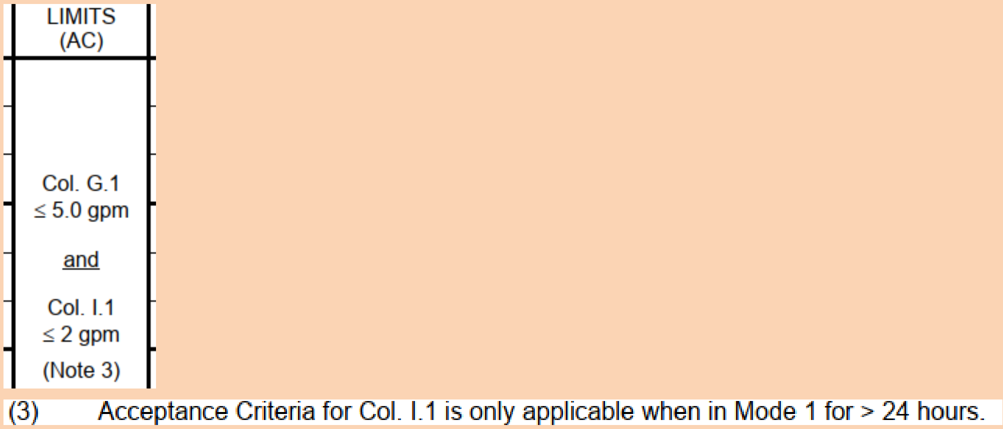
STEP / STANDARD	SAT / UNSAT
Step 1: (2-SR-2, Instrument Checks and Observations, Attachment 2 (page 4 of 90), Table **1.2** – DRYWELL **UNIDENTIFIED** LEAKAGE, Column **G.1**) Completes 2-SR-2 for Drywell Unidentified Leakage for 0800 Saturday morning, and calculates current Drywell Unidentified Leakrate (Drywell Floor Drain). Expected Action(s): Examinee calculates current Drywell Unidentified Leakrate in Column **G.1** using the given Drywell Floor Drain readings of 22643 gals: 7344 gals / 1440 minutes = 5.10 gpm (± .05 gpm is acceptable margin)	**Critical Step** ____ SAT _____UNSAT _____N/A
Step 2: (2-SR-2, Instrument Checks and Observations, Attachment 2 (page 4 of 90), Table 1.2, Column **I.1**) Calculates Change in Leakrate. Expected Action(s): Examinee calculates a Change in Leakrate in Column I.1: 5.10 – 3.50 gpm (Column G.1– Column H.1) = 1.60 gpm (± .05 gpm is acceptable margin) Determines the current Change in Leakrate of 1.60 gpm is ≤ 2 gpm as required by 2-SR-2.	**Critical Step** ____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: (2-SR-2, Instrument Checks and Observations, Attachment 2 (page 6 of 90), Table **1.3** - DRYWELL **IDENTIFIED** LEAKAGE **AND TOTAL LEAKAGE**, Column **G.2**) Completes 2-SR-2 for Drywell Identified Leakage and Total Leakage for 0800 Saturday morning, and calculates Drywell Identified Leakrate (Drywell Equipment Drain). Expected Action(s): Examinee calculates a current Drywell Identified Leakrate in Column **G.2** using the given Drywell Equipment Drain readings of 28005 gals: 2225 gals / 1440 minutes = 1.55 gpm (± .05 gpm is acceptable margin)	**Critical Step** ____ SAT _____UNSAT _____N/A
Step 4: (2-SR-2, Instrument Checks and Observations, Attachment 2 (page 6 of 90), Table 1.3, Column **I.2**). Calculates Total Leakrate. Expected Action(s): Examinee calculates a Total Leakrate of 6.65 gpm in Column **I.2**: 1.55 + 5.10 gpm (Column G.2– Column H.2) = 6.65 gpm (± .05 gpm is acceptable margin) Determines the current Total Leakrate of 6.65 gpm is ≤ 30.0 gpm as required by 2-SR-2 to meet the Acceptance Criteria (AC).	**Critical Step** ____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT

Step 5:

Compares the calculated current Drywell Unidentified Leakrate (5.10 gpm) from Column G.1 in JPM Step 1 above and Change in Leakrate ((+) 1.60 gpm) from Column I.1 in JPM Step 2 above to determine if the Acceptance Criteria (AC) LIMITS of ≤ 5.0 gpm and ≤ 2.0 gpm respectively have been met.



LIMITS (AC)
Col. G.1 ≤ 5.0 gpm and Col. I.1 ≤ 2 gpm (Note 3)

(3) Acceptance Criteria for Col. I.1 is only applicable when in Mode 1 for > 24 hours.

Technical Specification 3.4.4, RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;
b. ≤ 5 gpm unidentified LEAKAGE; and
c. ≤ 30 gpm total LEAKAGE averaged over the previous 24 hour period; and
d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

Expected Action(s):

Determines the current Drywell Unidentified Leakrate (5.10 gpm) does NOT meet the Acceptance Criteria (AC) LIMITS of ≤ 5 gpm. 2-SR-2, Att. 2 (page 4 of 90), Table 1.2 requires both limits to be acceptable for the AC to be met.

Critical Step

____ SAT

_____UNSAT

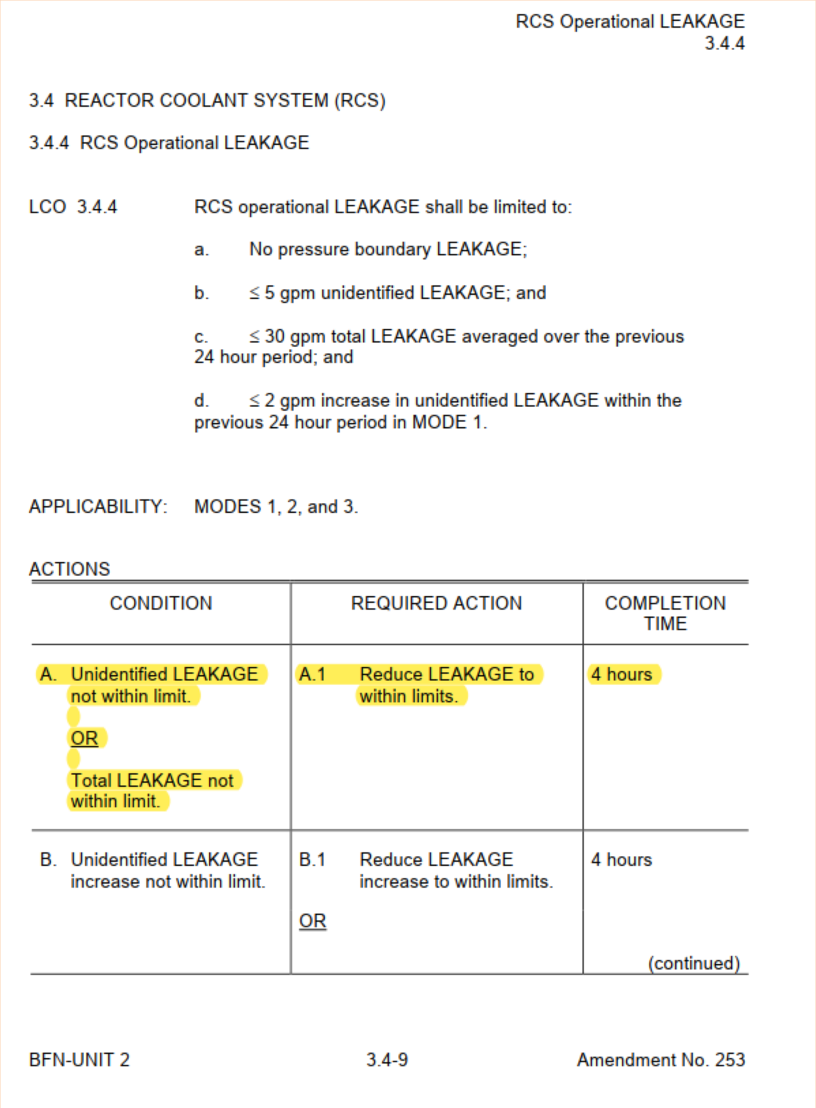
_____N/A

EXAMINER NOTE: The Candidate may elect to reference Tech Spec 3.4.4 for the Limits, but it is not required since the AC LIMITS are listed on 2-SR-2.

STEP / STANDARD	SAT / UNSAT

Step 6:

Determine what Technical Specification Required Action(s) must be taken (if any).



RCS Operational LEAKAGE
3.4.4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE; and

c. ≤ 30 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce LEAKAGE increase to within limits. OR	4 hours

(continued)

BFN-UNIT 2 3.4-9 Amendment No. 253

Expected Action(s):

Determines the current Drywell Unidentified Leakrate (5.10 gpm) is **NOT** within the limit of Technical Specification 3.4.4, RCS Operational LEAKAGE LCO Requirement of ≤ 5 GPM. Condition A is applicable and the REQUIRED ACTION A.1 is to Reduce LEAKAGE to within limits within (4) four hours.

Critical Step

____ SAT

____ UNSAT

_____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: Once the Operator completes JPM Step 6 above, 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:

Unit 2 is operating at 100% Rated Thermal Power (RTP), online for 100 days.

It is Saturday at 0800, and the Drywell Floor and Equipment Drain sumps have just completed pumping down.

The 0800 readings for 2-SR-2, Instrument Checks and Observations, are as follows:

• Drywell Floor Drain	22643 gals
• Drywell Equipment Drain	28005 gals

INITIATING CUE:

You are the Nuclear Unit Senior Operator (NUSO) required to complete the following:

- 2-SR-2, Instrument Checks and Observations, for the Drywell Floor and Equipment Drain Sumps
- Determine if **ALL** Acceptance Criteria (AC) is met
- Determine what Technical Specification Required Action(s) must be taken (if any)?

Answer:

____

____

____

____

____

PROVIDE to APPLICANT

BFN **Unit 2**	**Instrument Checks and Observations**	**2-SR-2** **Rev. 0087** **Page 25 of 157**

Attachment 2
(Page 4 of 90)

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

TABLE 1.2 DRYWELL UNIDENTIFIED LEAKAGE DAY SHIFT WEEK: Now to LATER

APPLICABILITY: Modes 1, 2 & 3 Readings are required at all times.

Surveillance Requirements: 3.4.4.1 LOCATION: Panel 2-9-4, 2-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 (2-FQ-77-6) Reading (gals) (Notes 1, 2)	Previous Days 2-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 SRO (Note 4)
Friday	15299	10259	5040	0800	0800	1440	3.50	3.50	0	Col. G.1 ≤ 5.0 gpm and Col. I.1 ≤ 2 gpm (Note 3)	RO	SO
	16223	11,111	5112	1200	1200	1440	3.55	3.50	+.05		RO	SO
	17155	11,971	5184	1600	1600	1440	3.60	3.50	+.10		RO	SO
Saturday												
Sunday												
Monday												

NOTES ARE ON THE FOLLOWING PAGE!

PROVIDE to APPLICANT

BFN **Unit 2**	**Instrument Checks and Observations**	**2-SR-2** **Rev. 0087** **Page 27 of 157**

Attachment 2
(Page 6 of 90)

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

TABLE 1.3 DRYWELL IDENTIFIED LEAKAGE AND TOTAL LEAKAGE **DAY SHIFT** WEEK: Now to LATER

APPLICABILITY: Modes 1, 2 & 3 Readings are required at all times.

Surveillance Requirements: 3.4.4.1 LOCATION: Panel 2-9-4, 2-FR-77-6

	Col. A.2	Col. B.2	Col. C.2	Col. D.2	Col. E.2	Col. F.2	Col. G.2	Col. H.2	Col. I.2		Review Init	
Preferred reading times are 0800, 1200 and 1600	Current Point 4 (2-FQ-77-16) Reading (gals) (Notes 1, 2)	Previous Days 2-FQ-77-16 Reading from Col. A.2 (gals) (Note 2)	Gallons Pumped Col. A.2 - Col. B.2 (Note 2)	Current Time (Note 2)	Previous Days Time from Col. D.2 (Note 2)	Elapsed Time Col. D.2 - Col. E.2 (min) (Note 2)	Current Leakrate Col. C.2 ÷ Col. F.2 (gpm) (Note 2)	Current Unidentified Leakrate from Col. G.1 (gpm) (Notes 2 & 3)	Total Leakrate Col. G.2 + Col. H.2 (gpm) (Note 2)	LIMITS (AC)	UO	Unit SRO (Note 4)
Friday	25780	23568	2212	0800	0800	1440	1.54	3.50	5.04	Col. I.2 ≤ 30.0 gpm	RO	SO
	26144	23936	2208	1200	1200	1440	1.53	3.55	5.08		RO	SO
	26521	24306	2215	1600	1600	1440	1.54	3.60	5.14		RO	SO
Saturday												
Sunday												
Monday												

NOTES ARE ON THE FOLLOWING PAGE!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Determine Protected Equipment Requirements
JPM NUMBER:	753-SRO	REVISION:	2

TASK APPLICABILITY:	☒SRO	☐STA	☐UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	N/A			
K/A RATINGS:	SRO 4.3			
K/A STATEMENT:	2.1.39 Knowledge of conservative decision-making practices			
RELATED PRA INFORMATION:	None			
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*

OPERATOR: ______________________________ JPM Number: 753-SRO

RO ____ SRO ____ DATE: _______

TASK STANDARD: The Examinee is expected to determine Protected Equipment Requirements and Technical Specification Requirements for a given set of plant conditions associated with the Standby Gas Treatment Systems.

Operator Fundamental evaluated:
OF-3 Operating the Plant with a Conservative Bias

PRA: NA

REFERENCES/PROCEDURES NEEDED:
NPG-SPP-07.3.4, Protected Equipment
ODM 4.18, Protected Equipment
Unit 3 Tech Spec 3.6.4.3 and 3.8.1

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	10/03/2018	ALL	New JPM
1	08/15/2019	ALL	Updated JPM
2	02/01/2022	ALL	Updated JPM

Procedure Revisions

Procedure	Revision
ODM 4.18	30
NPG-SPP-07.3.4	11
Unit 3 Tech Spec 3.6.4.3	Amend. 294
Unit 3 Tech Spec 3.8.1	Amend. 266

****

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% Rated Thermal Power (RTP) with 3D Diesel Generator INOPERATIVE with the applicable equipment already protected.

At 1000, you are the Unit 3 Nuclear Unit Senior Operator (NUSO) when Work Control reports to you that the ‘A’ Train of the Standby Gas Treatment System (SGT) motor supply breaker was found tripped and will not reset.

INITIATING CUES:

As the Unit 3 NUSO, determine the following:

- Applicable Unit 3 Technical Specification Requirement(s)
- Time LCO 3.0.3 entry is required
- Additional Protected Equipment Requirement(s) given the Standby Gas Treatment System conditions above (if any)

Answer:

____

____

____

____

____

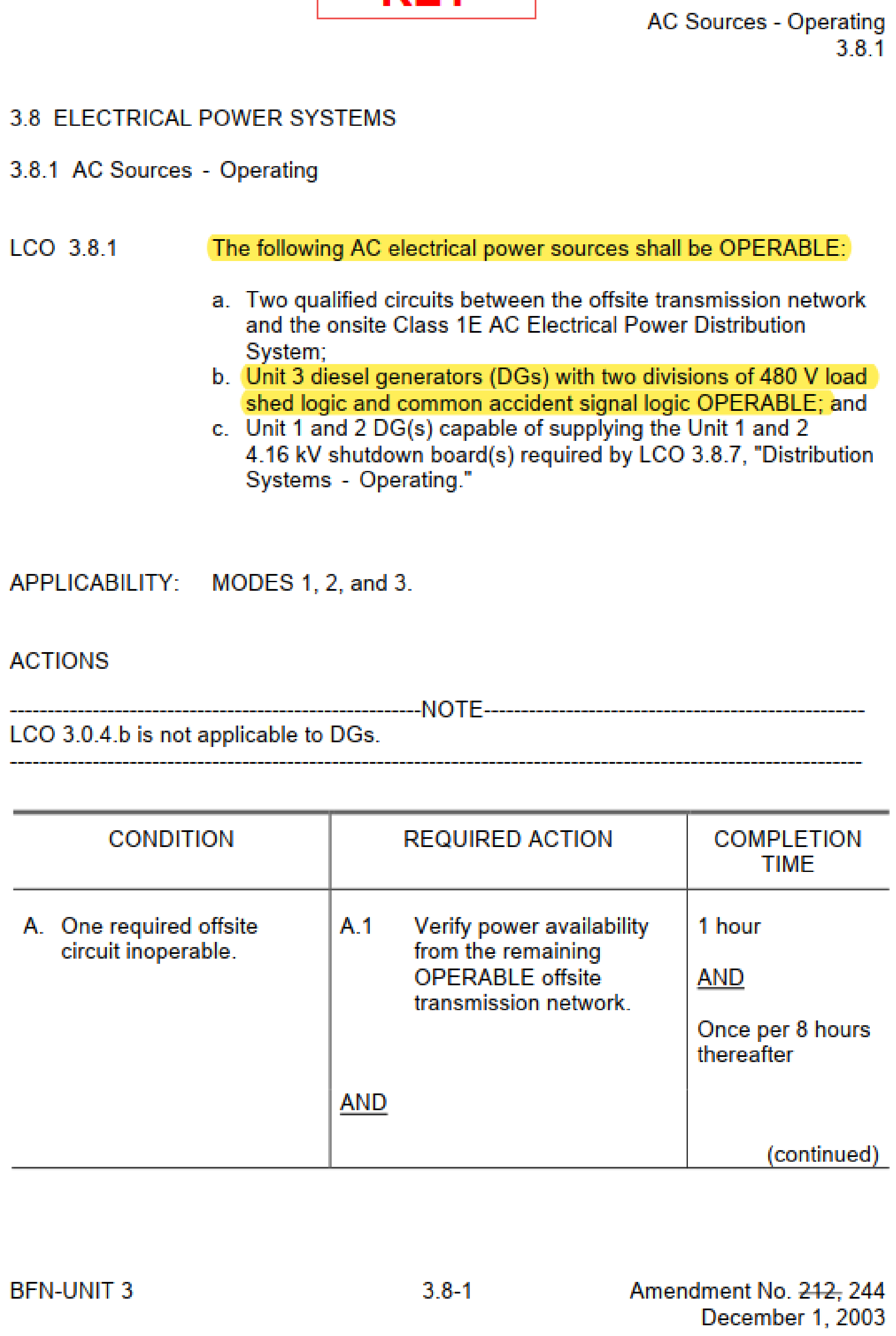
____

____

START TIME:________________

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: The Examinee may elect to reference NPG-SPP-07.3.4, Protected Equipment, however it is not required or considered Critical.	
Step 1: Determine applicable Unit 3 Technical Specification Requirement(s) for the given plant conditions. Expected Action(s): Examinee references Unit 3 Technical Specification 3.8.1 given that 3D Diesel Generator is INOPERABLE. The candidate will indicate that this meets the following: • Tech Spec 3.8.1, CONDITION B – One required Unit 3 DG INOPERABLE • REQUIRED ACTION B.3 – Declare required feature(s), supported by the INOPERABLE Unit 3 DG, INOPERABLE when the redundant required feature(s) are INOPERABLE • COMPLETION TIME – 4 hours from discovery of CONDTION B concurrent with INOPERABILITY of redundant required feature(s)	**Critical Step** ____ SAT ____UNSAT ____N/A
EXAMINER NOTE: 3D Diesel Generator provides alternate power to 4KV Shutdown Board 3ED and 480V Standby Gas Board which provides power to 'C' Train of the Standby Gas Treatment System (SGT). Given that the redundant required 'A' SGT is already INOPERABLE, 'C' SGT is required to be declared INOPERABLE within 4 hours.	

KEY



AC Sources - Operating
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 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.

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)

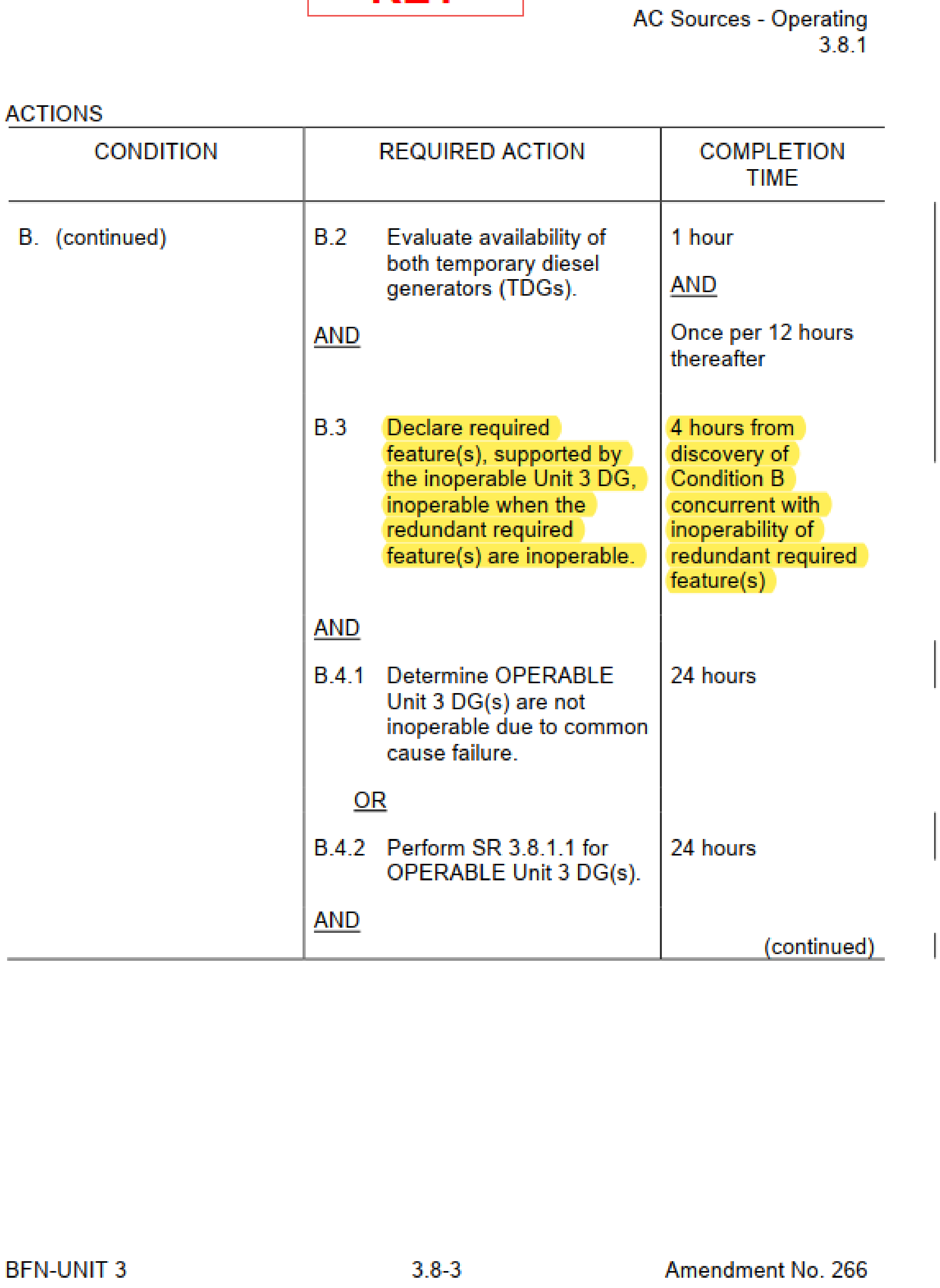
BFN-UNIT 3 3.8-1 Amendment No. ~~212,~~ 244
December 1, 2003

KEY

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 3 DG inoperable.	B.1	Verify power availability from the offsite transmission network.	1 hour AND Once per 8 hours thereafter
	AND		
			(continued)

KEY



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 AND Once per 12 hours thereafter
	AND	
	B.3 Declare required feature(s), supported by the inoperable Unit 3 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)
	AND	
	B.4.1 Determine OPERABLE Unit 3 DG(s) are not inoperable due to common cause failure.	24 hours
	OR	
	B.4.2 Perform SR 3.8.1.1 for OPERABLE Unit 3 DG(s).	24 hours
	AND	
		(continued)

BFN-UNIT 3 3.8-3 Amendment No. 266

STEP / STANDARD	SAT / UNSAT
Step 2: Determine applicable Unit 3 Technical Specification Requirement(s) for the given plant conditions and Time LCO entry is required. Expected Action(s): Examinee references Unit 3 Technical Specification 3.6.4.3 and determines since the 'A' Train of the Standby Gas Treatment System (SGT) motor supply breaker found tripped and will not reset, the SGT System will be declared INOPERABLE. Since 3D Diesel Generator provides alternate power to 4KV Shutdown Board 3ED and 480V Standby Gas Board which provides power to 'C' Train of the Standby Gas Treatment System (SGT), the candidate will indicate that this meets the following: • Tech Spec 3.6.4.3, CONDITION C – Two or three SGT subsystems INOPERABLE • REQUIRED ACTION C.1 – Enter LCO 3.0.3 • COMPLETION TIME – Immediately • LCO entry required 4 hours from discovery of 'C' SGT INOPERABILITY (at 1000) at Time: 1400	**Critical Step** ____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Given that the redundant required 'A' SGT is already INOPERABLE, 'C' SGT is then required to be declared INOPERABLE within 4 hours. If either 'A' or 'C' hasn't been restored within 4 hours, LCO 3.0.3 entry is then required Immediately.	

KEY

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE.

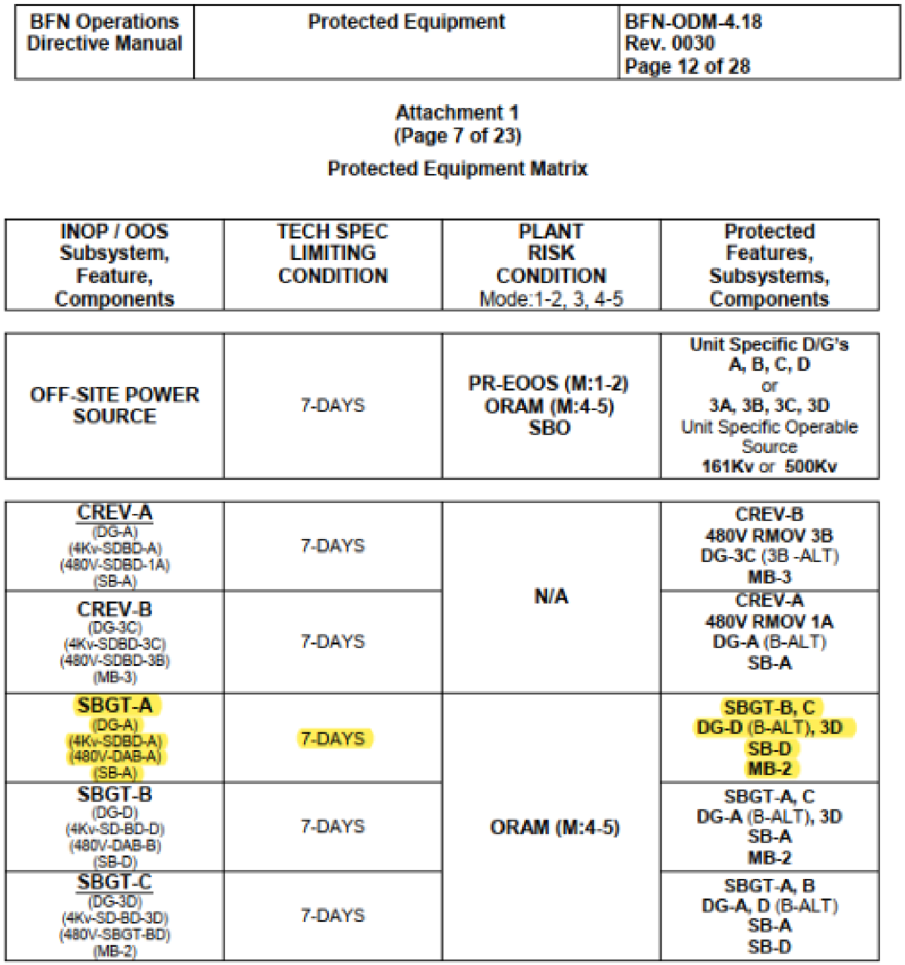
APPLICABILITY: MODES 1, 2, and 3.

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.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Two or three SGT subsystems inoperable.	C.1 Enter LCO 3.0.3.	Immediately

Step 3: Additional Protected Equipment Requirement(s) given the Standby Gas Treatment System conditions (if any) Expected Action(s): Examinee references BFN-ODM-4.18, Protected Equipment, Attachment 1, to determine the equipment required to be protected with 'A' Train of Standby Gas Treatment (SGT) INOPERABLE. The candidate will indicate the following components are required to be protected: • Standby Gas Treatment Trains – B, C • Diesel Generator – D, 3D • Shutdown Battery (SB) - D • Main Bank Battery (MB) - 2	**Critical Step** ____ SAT _____UNSAT _____N/A
EXAMINER NOTE: The Examinee may elect to list other equipment, for example: Diesel Generator – B (B-ALT) as required Protected Equipment from Attachment 1 associated with 'A' Train of Standby Gas Treatment (SGT) being INOPERABLE, however it is not required or considered Critical.	

KEY



BFN Operations Directive Manual	Protected Equipment	BFN-ODM-4.18 Rev. 0030 Page 12 of 28

Attachment 1
(Page 7 of 23)

Protected Equipment Matrix

INOP / OOS Subsystem, Feature, Components	TECH SPEC LIMITING CONDITION	PLANT RISK CONDITION Mode:1-2, 3, 4-5	Protected Features, Subsystems, Components
OFF-SITE POWER SOURCE	7-DAYS	PR-EOOS (M:1-2) ORAM (M:4-5) SBO	Unit Specific D/G's A, B, C, D or 3A, 3B, 3C, 3D Unit Specific Operable Source 161Kv or 500Kv
CREV-A (DG-A) (4Kv-SDBD-A) (480V-SDBD-1A) (SB-A)	7-DAYS	N/A	CREV-B 480V RMOV 3B DG-3C (3B -ALT) MB-3
CREV-B (DG-3C) (4Kv-SDBD-3C) (480V-SDBD-3B) (MB-3)	7-DAYS		CREV-A 480V RMOV 1A DG-A (B-ALT) SB-A
SBGT-A (DG-A) (4Kv-SDBD-A) (480V-DAB-A) (SB-A)	7-DAYS	ORAM (M:4-5)	SBGT-B, C DG-D (B-ALT), 3D SB-D MB-2
SBGT-B (DG-D) (4Kv-SD-BD-D) (480V-DAB-B) (SB-D)	7-DAYS		SBGT-A, C DG-A (B-ALT), 3D SB-A MB-2
SBGT-C (DG-3D) (4Kv-SD-BD-3D) (480V-SBGT-BD) (MB-2)	7-DAYS		SBGT-A, B DG-A, D (B-ALT) SB-A SB-D

EXAMINER CUE: Once the Examinee completes JPM Step 2 above, 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:

Unit 3 is operating at 100% Rated Thermal Power (RTP) with 3D Diesel Generator INOPERATIVE with the applicable equipment already protected.

At 1000, you are the Unit 3 Nuclear Unit Senior Operator (NUSO) when Work Control reports to you that the ‘A’ Train of the Standby Gas Treatment System (SGT) motor supply breaker was found tripped and will not reset.

INITIATING CUES:

As the Unit 3 NUSO, determine the following:

- Applicable Unit 3 Technical Specification Requirement(s)
- Time LCO 3.0.3 entry is required
- Additional Protected Equipment Requirement(s) given the Standby Gas Treatment System conditions above (if any)

Answer:

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Review a completed Surveillance
JPM NUMBER:	746-SRO	REVISION:	6

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*

OPERATOR: ________________________ JPM Number: 746-SRO

SRO _____ DATE: ____________

TASK STANDARD: The Examinee is expected to conduct a review of the Power Availability Surveillance (SR), determine that Board Voltage Acceptance Criteria (AC) has not been met, and determine the required Technical Specification REQUIRED ACTIONs as applicable.

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, Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems, but **NOT** identified by the Unit Operator
(2) 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
5	03/13/2021	ALL	Updated JPM
6	02/10/2022	ALL	Updated JPM

Procedure Revisions

Procedure	**Revision**
3-SR-3.8.7.1	17
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% Rated Thermal Power (RTP). You are the Unit 3 Nuclear Unit Senior Operator (NUSO) and the Balance of Plant Operator (BOP) has just completed 3-SR-3.8.7.1, Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems, and has given it to you for review.

INITIATING CUES:

Conduct a review of 3-SR-3.8.7.1, Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems.

Determine additional REQUIRED ACTION(s) in accordance with Technical Specifications as applicable (if any).

Answer:

____

____

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____

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): Examinee reviews the completed 3-SR-3.8.7.1 and 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 23)	
Step 2: NUSO checks that the BOP has identified any anomalies. Expected Action(s): Examinee reviews the completed 3-SR-3.8.7.1 and notes that the BOP recorded 235 Volts in 7.4 [1] D, 250V DC RMOV BOARD 3A VOLTAGE.	______ SAT ______UNSAT ______N/A

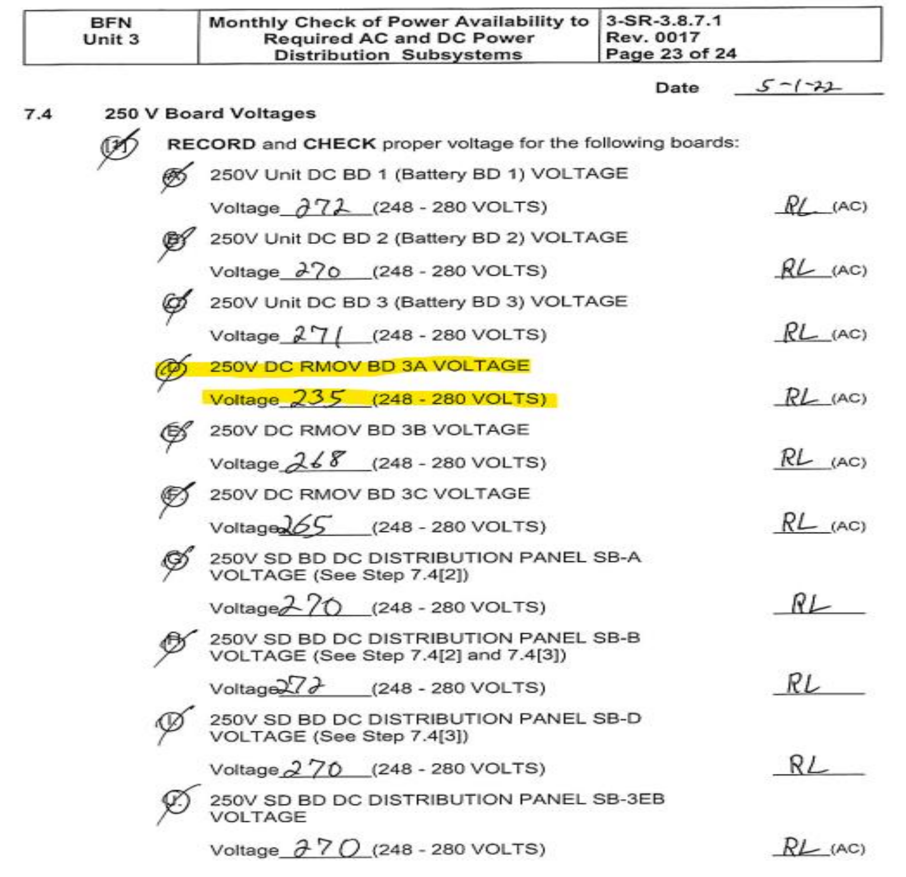
STEP / STANDARD | SAT / UNSAT

Step 3:

NUSO checks that the BOP has identified any anomalies.

Expected Action(s):

Examinee notes that Step 7.4[1] D, 250V DC RMOV BOARD 3A VOLTAGE, CHECK Voltage 248-280 volts, is NOT filled out correctly since Step voltage requirements are not met.
Step 7.4[1] D is an ACCEPTANCE CRITERIA (AC) step that was incorrectly signed off suggesting that the board voltage met the SR requirement.



BFN Unit 3	Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems	3-SR-3.8.7.1 Rev. 0017 Page 23 of 24

Date 5-1-22

7.4 250 V Board Voltages

[1] **RECORD** and **CHECK** proper voltage for the following boards:

(A) 250V Unit DC BD 1 (Battery BD 1) VOLTAGE

Voltage 272 (248 - 280 VOLTS) RL (AC)

(B) 250V Unit DC BD 2 (Battery BD 2) VOLTAGE

Voltage 270 (248 - 280 VOLTS) RL (AC)

(C) 250V Unit DC BD 3 (Battery BD 3) VOLTAGE

Voltage 271 (248 - 280 VOLTS) RL (AC)

(D) 250V DC RMOV BD 3A VOLTAGE

Voltage 235 (248 - 280 VOLTS) RL (AC)

(E) 250V DC RMOV BD 3B VOLTAGE

Voltage 268 (248 - 280 VOLTS) RL (AC)

(F) 250V DC RMOV BD 3C VOLTAGE

Voltage 265 (248 - 280 VOLTS) RL (AC)

(G) 250V SD BD DC DISTRIBUTION PANEL SB-A VOLTAGE (See Step 7.4[2])

Voltage 270 (248 - 280 VOLTS) RL

(H) 250V SD BD DC DISTRIBUTION PANEL SB-B VOLTAGE (See Step 7.4[2] and 7.4[3])

Voltage 272 (248 - 280 VOLTS) RL

(I) 250V SD BD DC DISTRIBUTION PANEL SB-D VOLTAGE (See Step 7.4[3])

Voltage 270 (248 - 280 VOLTS) RL

(J) 250V SD BD DC DISTRIBUTION PANEL SB-3EB VOLTAGE

Voltage 270 (248 - 280 VOLTS) RL (AC)

Critical Step

______ SAT

______UNSAT

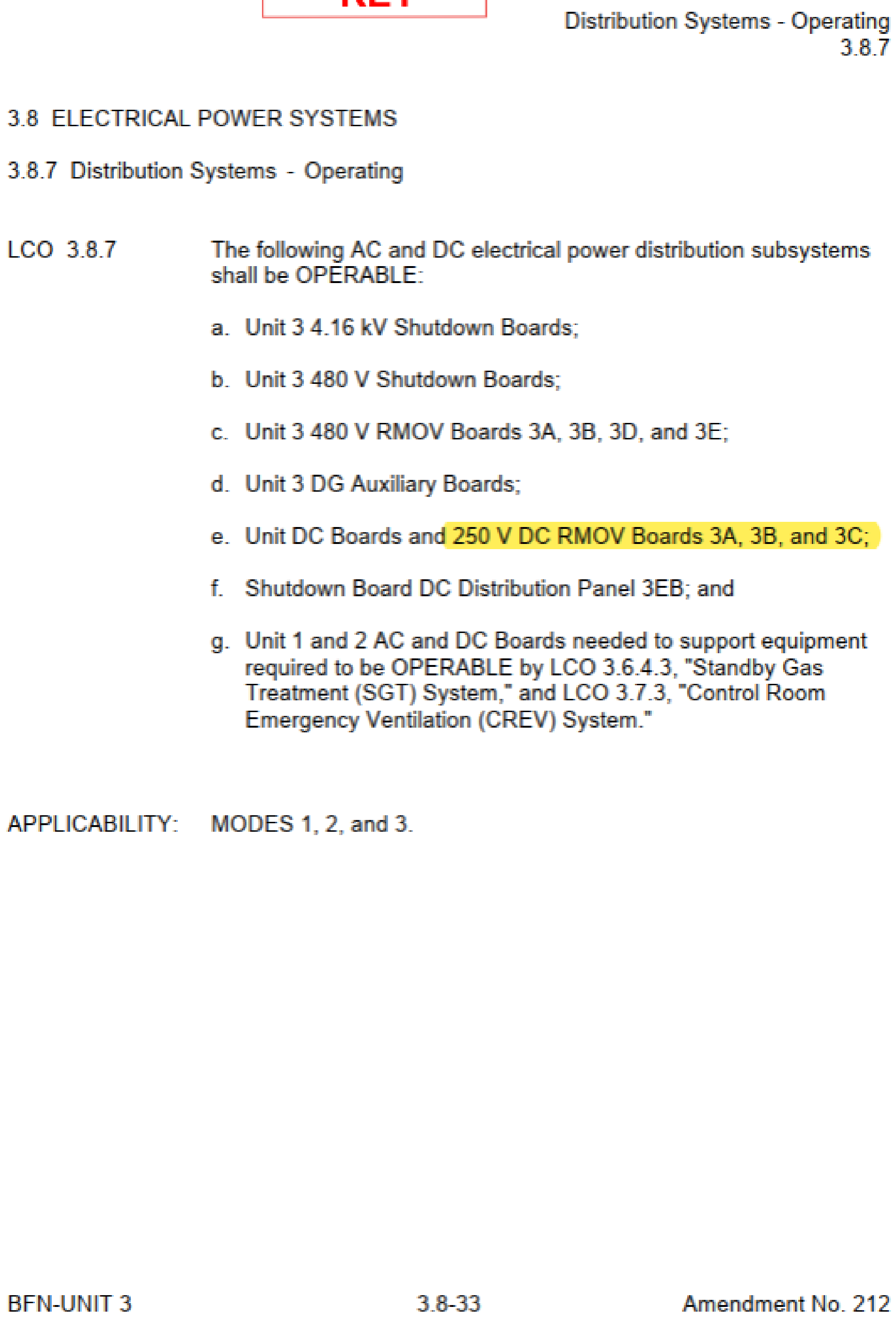
______N/A

EXAMINER CUE: For JPM Step 4 below, the 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.

Step 4: NUSO determines that 250V DC RMOV BOARD 3A voltage makes it INOPERABLE in accordance with Tech Spec 3.8.7. Expected Action(s): Examinee determines that all AC steps are NOT met therefore fails the SR. The NUSO will enter Tech Spec 3.8.7 CONDITION E with the following REQUIRED ACTION: E.1 Restore required Board or Shutdown Board DC Distribution Panel 3EB to OPERABLE status.	**Critical Step** ______ SAT ______UNSAT ______N/A

STOP TIME ___________

KEY



3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Distribution Systems - Operating

LCO 3.8.7 The following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Unit 3 4.16 kV Shutdown Boards;

b. Unit 3 480 V Shutdown Boards;

c. Unit 3 480 V RMOV Boards 3A, 3B, 3D, and 3E;

d. Unit 3 DG Auxiliary Boards;

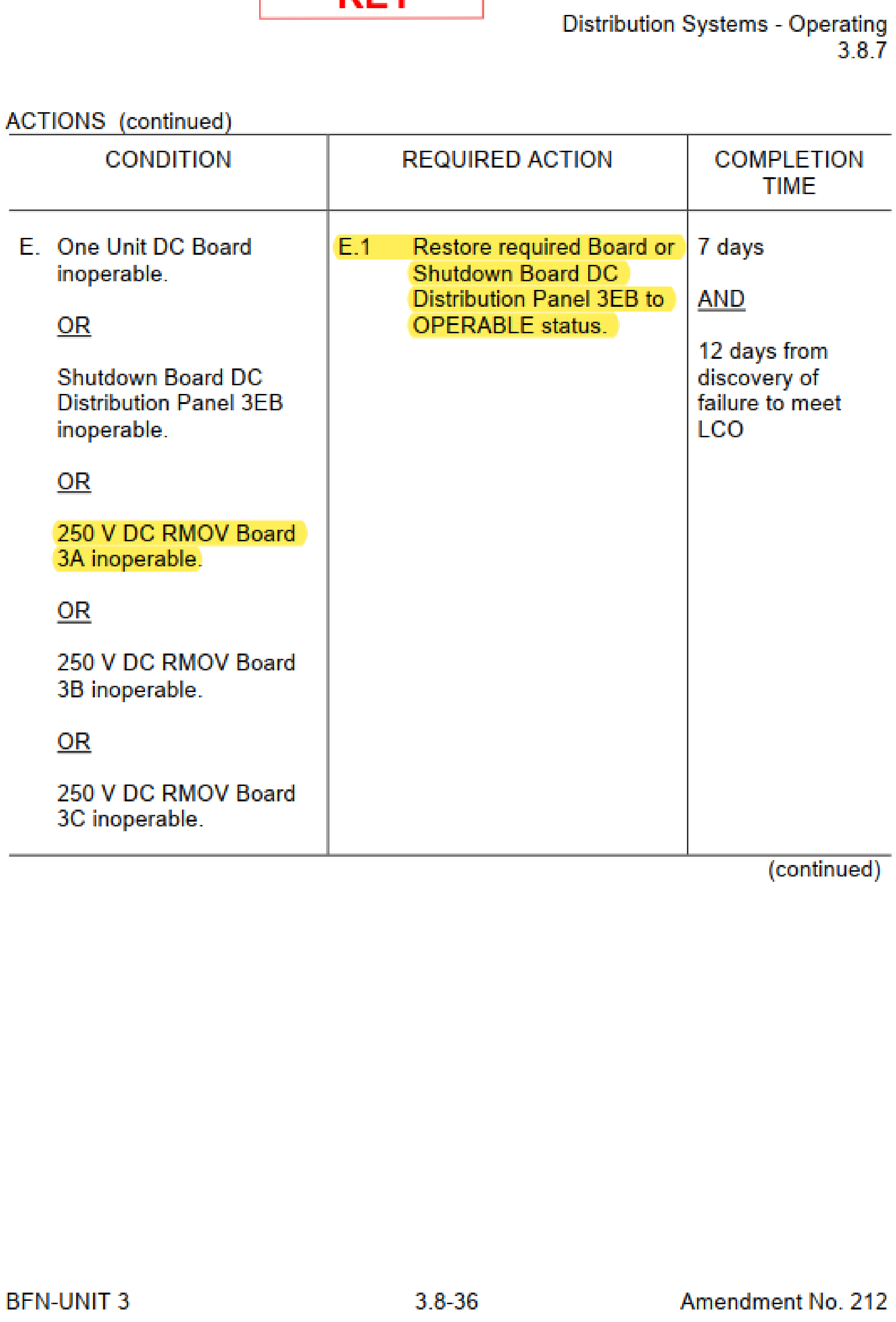
e. Unit DC Boards and 250 V DC RMOV Boards 3A, 3B, and 3C;

f. Shutdown Board DC Distribution Panel 3EB; and

g. Unit 1 and 2 AC and DC Boards 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.

KEY



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One Unit DC Board inoperable. OR Shutdown Board DC Distribution Panel 3EB inoperable. OR 250 V DC RMOV Board 3A inoperable. OR 250 V DC RMOV Board 3B inoperable. OR 250 V DC RMOV Board 3C inoperable.	E.1 Restore required Board or Shutdown Board DC Distribution Panel 3EB to OPERABLE status.	7 days AND 12 days from discovery of failure to meet LCO

(continued)

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% Rated Thermal Power (RTP). You are the Unit 3 Nuclear Unit Senior Operator (NUSO) and the Balance of Plant Operator (BOP) has just completed 3-SR-3.8.7.1, Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems, and has given it to you for review.

INITIATING CUES:

Conduct a review of 3-SR-3.8.7.1, Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems.

Determine additional REQUIRED ACTION(s) in accordance with Technical Specifications as applicable (if any).

Answer:

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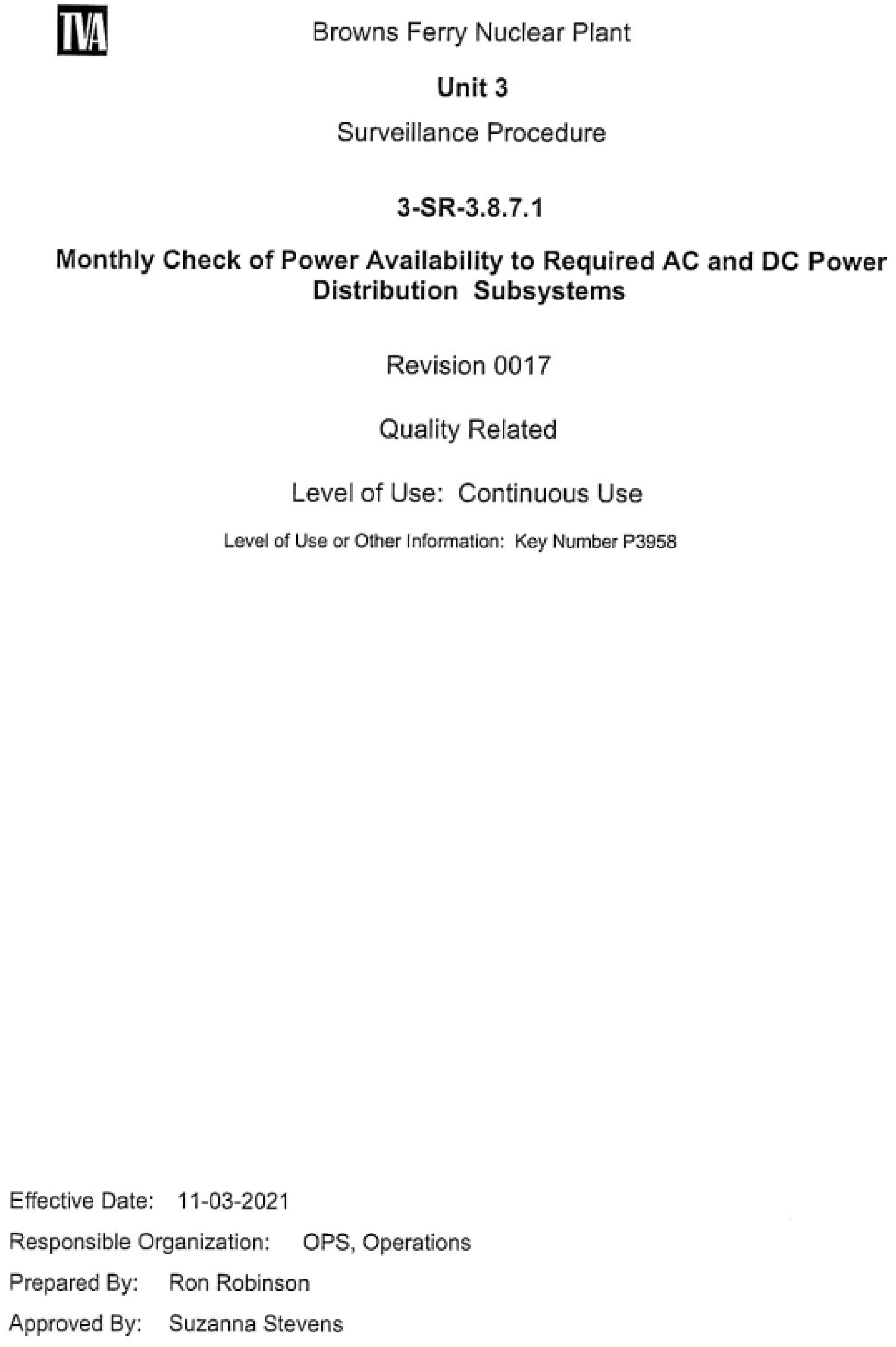


Browns Ferry Nuclear Plant

Unit 3

Surveillance Procedure

3-SR-3.8.7.1

Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems

Revision 0017

Quality Related

Level of Use: Continuous Use

Level of Use or Other Information: Key Number P3958

Effective Date: 11-03-2021

Responsible Organization: OPS, Operations

Prepared By: Ron Robinson

Approved By: Suzanna Stevens

Current Revision Description

Type of Change: Enhancement   Tracking Number: 019

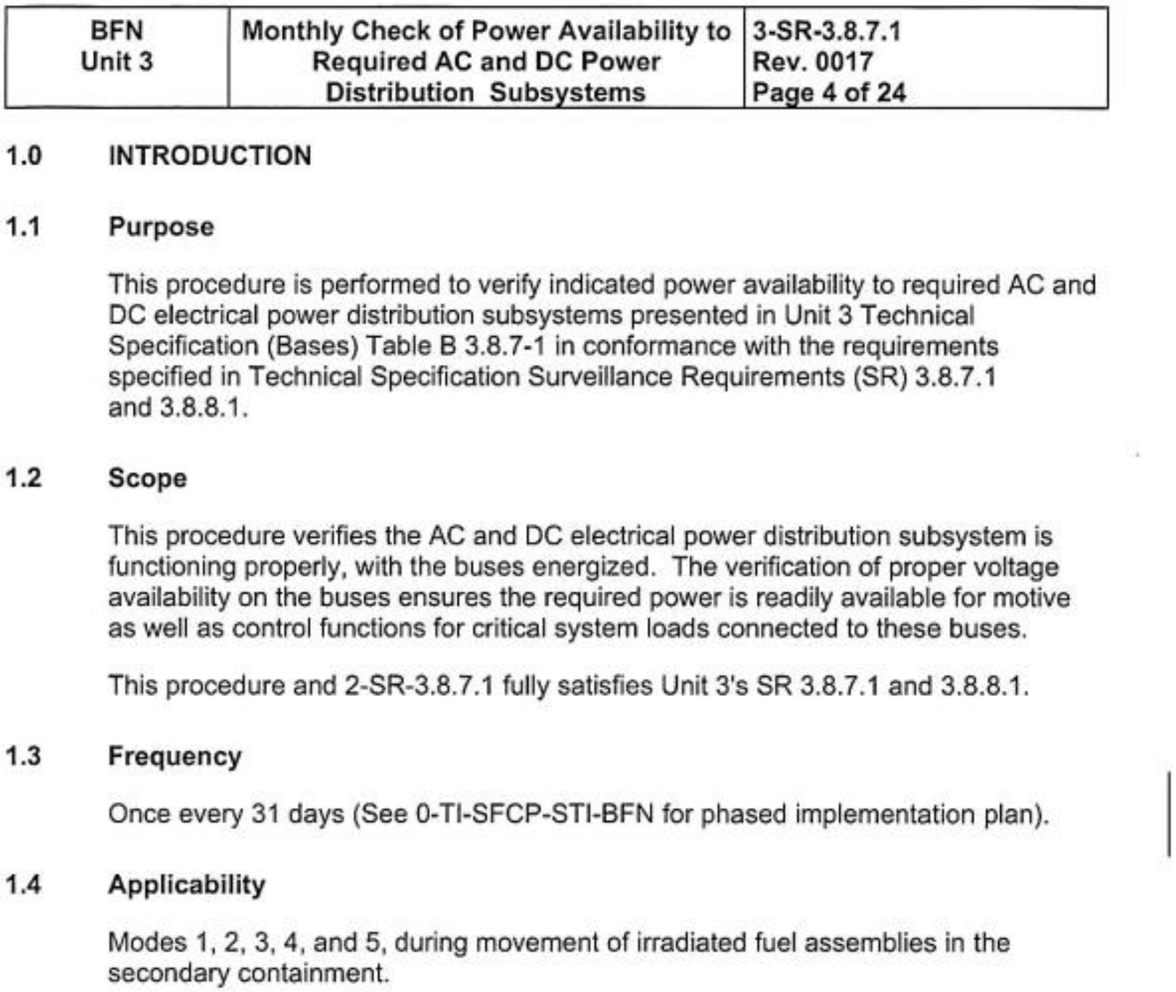
PCRs: N/A

Documentation: CR 1709212

Changed Title from "Weekly" to "Monthly"

Section 1.3 - Changed the frequency from 7 Days to 31 Days based on Surveillance Test Risk Informed Document Evaluation (STRIDE) BFN-SFCP-073 and Risk Assessment BFN-0-21-073.

Table of Contents



BFN Unit 3	Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems	3-SR-3.8.7.1 Rev. 0017 Page 4 of 24

1.0 INTRODUCTION

1.1 Purpose

This procedure is performed to verify indicated power availability to required AC and DC electrical power distribution subsystems presented in Unit 3 Technical Specification (Bases) Table B 3.8.7-1 in conformance with the requirements specified in Technical Specification Surveillance Requirements (SR) 3.8.7.1 and 3.8.8.1.

1.2 Scope

This procedure verifies the AC and DC electrical power distribution subsystem is functioning properly, with the buses energized. The verification of proper voltage availability on the buses ensures the required power is readily available for motive as well as control functions for critical system loads connected to these buses.

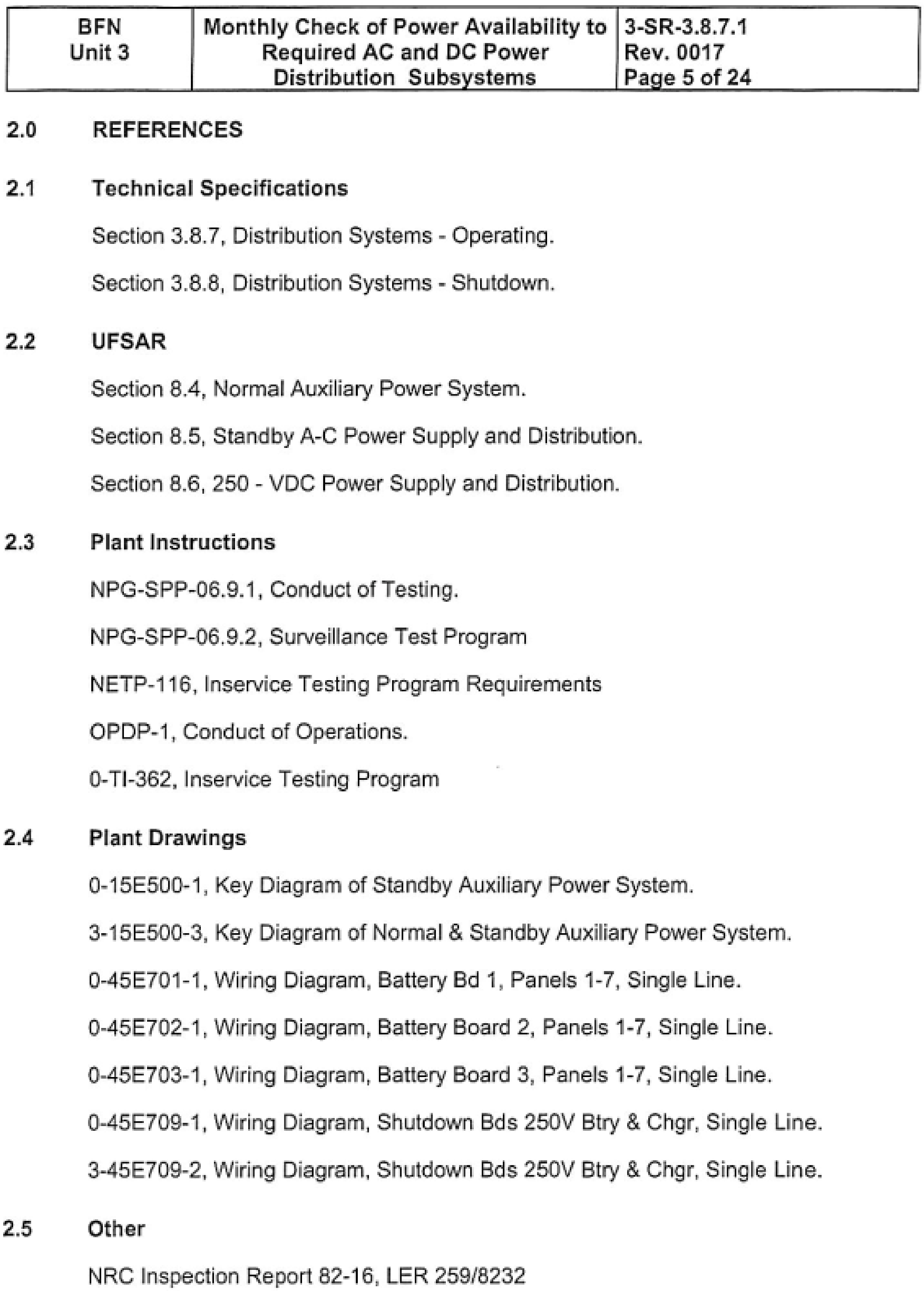
This procedure and 2-SR-3.8.7.1 fully satisfies Unit 3's SR 3.8.7.1 and 3.8.8.1.

1.3 Frequency

Once every 31 days (See 0-TI-SFCP-STI-BFN for phased implementation plan).

1.4 Applicability

Modes 1, 2, 3, 4, and 5, during movement of irradiated fuel assemblies in the secondary containment.



BFN Unit 3	Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems	3-SR-3.8.7.1 Rev. 0017 Page 5 of 24

2.0 REFERENCES

2.1 Technical Specifications

Section 3.8.7, Distribution Systems - Operating.

Section 3.8.8, Distribution Systems - Shutdown.

2.2 UFSAR

Section 8.4, Normal Auxiliary Power System.

Section 8.5, Standby A-C Power Supply and Distribution.

Section 8.6, 250 - VDC Power Supply and Distribution.

2.3 Plant Instructions

NPG-SPP-06.9.1, Conduct of Testing.

NPG-SPP-06.9.2, Surveillance Test Program

NETP-116, Inservice Testing Program Requirements

OPDP-1, Conduct of Operations.

0-TI-362, Inservice Testing Program

2.4 Plant Drawings

0-15E500-1, Key Diagram of Standby Auxiliary Power System.

3-15E500-3, Key Diagram of Normal & Standby Auxiliary Power System.

0-45E701-1, Wiring Diagram, Battery Bd 1, Panels 1-7, Single Line.

0-45E702-1, Wiring Diagram, Battery Board 2, Panels 1-7, Single Line.

0-45E703-1, Wiring Diagram, Battery Board 3, Panels 1-7, Single Line.

0-45E709-1, Wiring Diagram, Shutdown Bds 250V Btry & Chgr, Single Line.

3-45E709-2, Wiring Diagram, Shutdown Bds 250V Btry & Chgr, Single Line.

2.5 Other

NRC Inspection Report 82-16, LER 259/8232

3.0 PRECAUTIONS AND LIMITATIONS

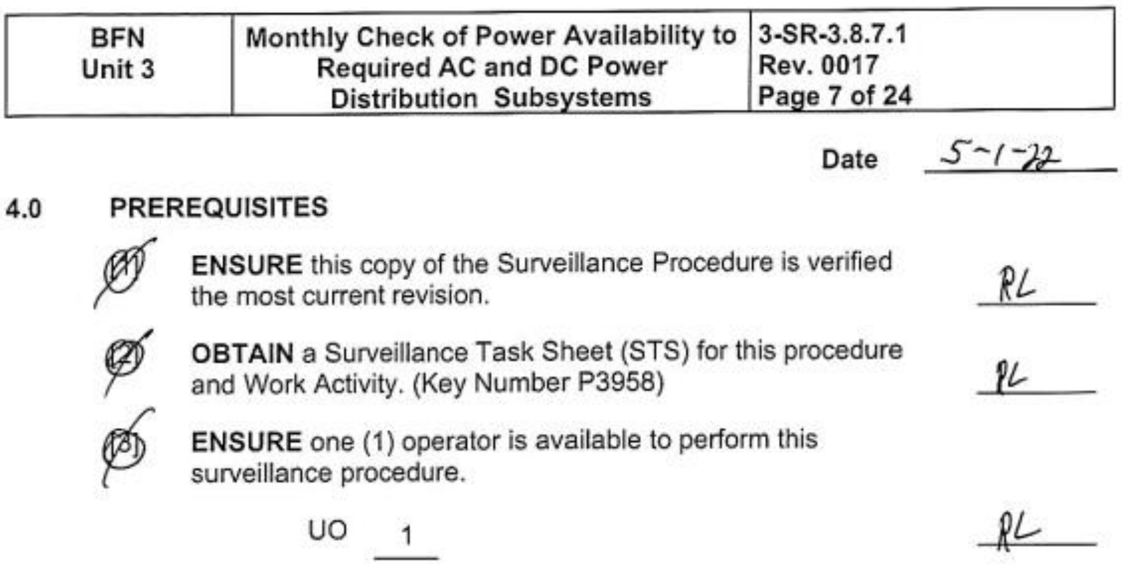
3.1 General Precautions

A. This procedure may be performed in any plant operating condition subject to the approval of the Unit SRO.

B. This procedure will require the performer to obtain various readings from local indications since all required readings are not available from any Control Room.

C. 480V Shutdown Board, 480V Diesel Auxiliary Board, SGT Board, 480V RMOV Board D or E voltages that appear to exceed 500 volts requires EM assistance in verifying the AØ to BØ, BØ to CØ, and CØ to AØ voltages using a multimeter. Measured voltages exceeding 508 volts requires notification of Site Engineering.

D. When the voltage for an electrical board appears to be outside its allowable range, the event should be recorded in the post-test remarks and a Condition Report (CR) should be generated or verified initiated to have Electrical Maintenance verify the board voltages that are the outside the range.



BFN **Unit 3**	**Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems**	**3-SR-3.8.7.1** **Rev. 0017** **Page 7 of 24**

Date 5-1-22

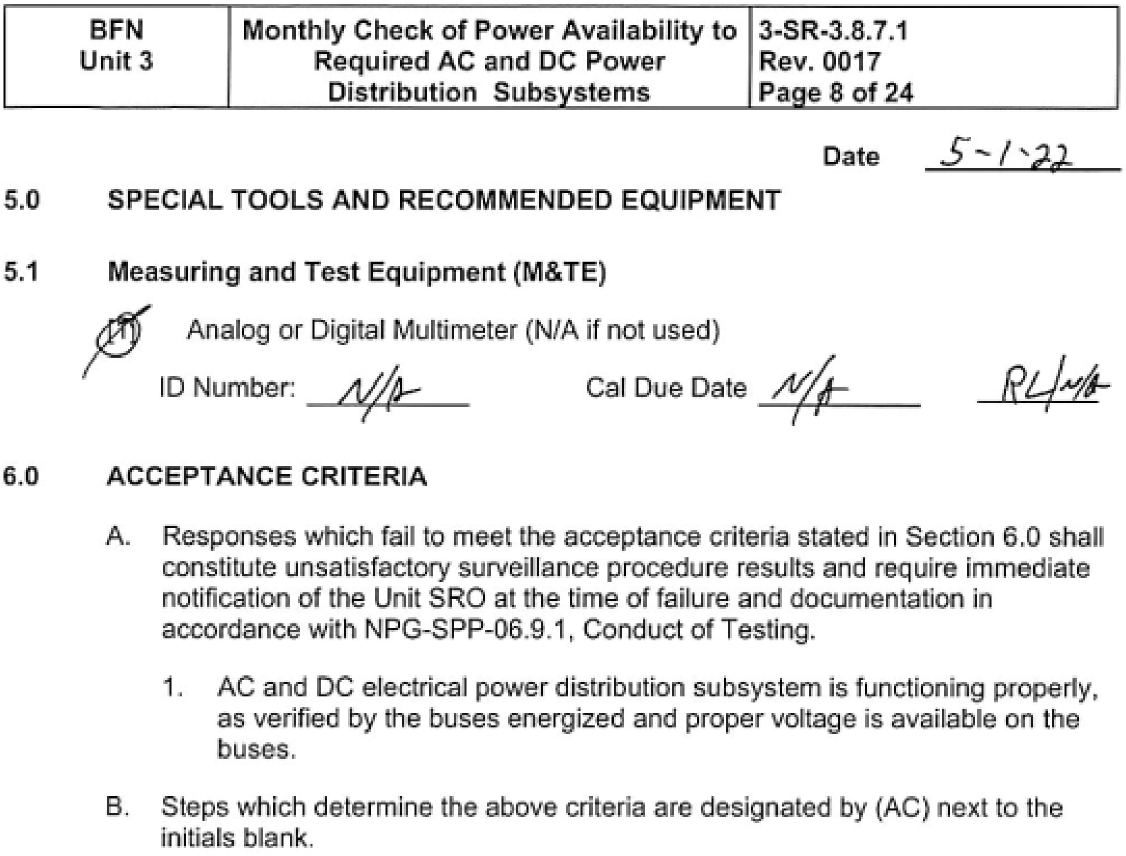
4.0 PREREQUISITES

(1) **ENSURE** this copy of the Surveillance Procedure is verified the most current revision. RL

(2) **OBTAIN** a Surveillance Task Sheet (STS) for this procedure and Work Activity. (Key Number P3958) RL

(3) **ENSURE** one (1) operator is available to perform this surveillance procedure.

UO 1 RL



BFN Unit 3	Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems	3-SR-3.8.7.1 Rev. 0017 Page 8 of 24

Date 5-1-22

5.0 SPECIAL TOOLS AND RECOMMENDED EQUIPMENT

5.1 Measuring and Test Equipment (M&TE)

(1) Analog or Digital Multimeter (N/A if not used)

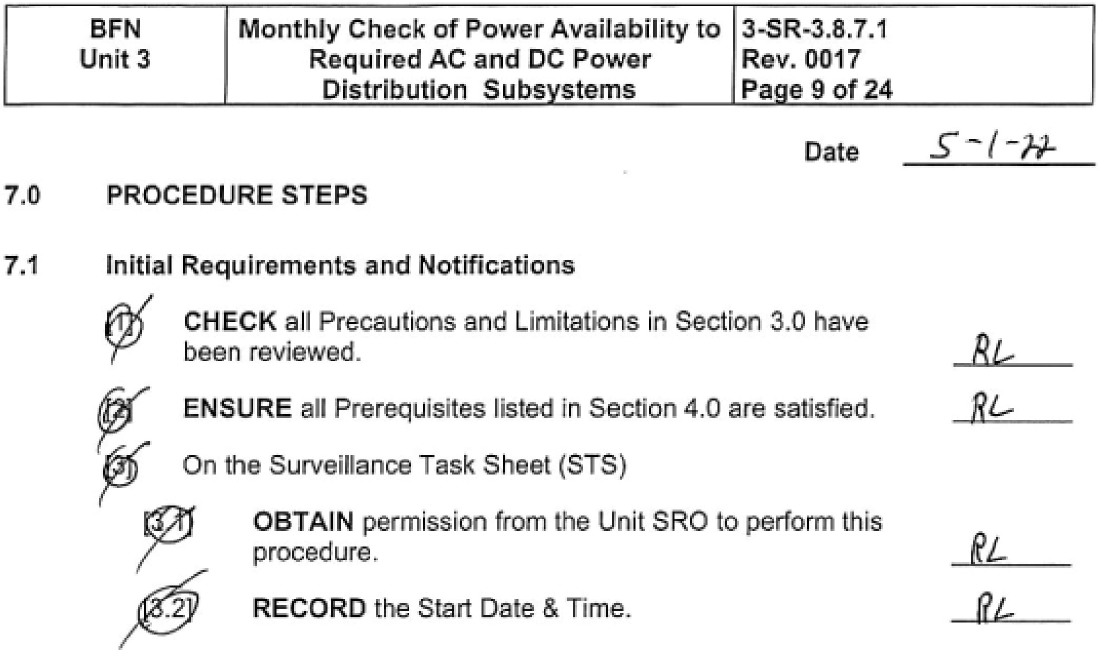
ID Number: N/A Cal Due Date N/A RL N/A

6.0 ACCEPTANCE CRITERIA

A. Responses which fail to meet the acceptance criteria stated in Section 6.0 shall constitute unsatisfactory surveillance procedure results and require immediate notification of the Unit SRO at the time of failure and documentation in accordance with NPG-SPP-06.9.1, Conduct of Testing.

1. AC and DC electrical power distribution subsystem is functioning properly, as verified by the buses energized and proper voltage is available on the buses.

B. Steps which determine the above criteria are designated by (AC) next to the initials blank.



BFN Unit 3	Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems	3-SR-3.8.7.1 Rev. 0017 Page 9 of 24

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7.0 PROCEDURE STEPS

7.1 Initial Requirements and Notifications

[1] **CHECK** all Precautions and Limitations in Section 3.0 have been reviewed. RL

[2] **ENSURE** all Prerequisites listed in Section 4.0 are satisfied. RL

[3] On the Surveillance Task Sheet (STS)

- [3.1] **OBTAIN** permission from the Unit SRO to perform this procedure. RL
- [3.2] **RECORD** the Start Date & Time. RL

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NOTES

1) Voltage checks can be performed in any order.

2) Specific voltmeters are NOT listed to allow maximum flexibility when obtaining readings. However, Control Room indications should be used whenever possible.

7.2 4KV Shutdown Board Voltages

[1] **ENSURE** applicable VOLTAGE SELECT switch positioned to desired 4KV SD BD prior to recording each voltage in Step 7.2[2] as required. (Otherwise N/A) RL

[2] **CHECK** and **RECORD** proper voltage for the following boards:

A. 4KV SD BD A VOLTAGE

Voltage 4200 (3950 - 4400 VOLTS) RL

B. 4KV SD BD B VOLTAGE

Voltage 4250 (3950 - 4400 VOLTS) RL

C. 4KV SD BD D VOLTAGE

Voltage 4200 (3950 - 4400 VOLTS) RL

D. 4KV SD BD 3EA VOLTAGE

Voltage 4300 (3950 - 4400 VOLTS) RL (AC)

E. 4KV SD BD 3EB VOLTAGE

Voltage 4300 (3950 - 4400 VOLTS) RL (AC)

F. 4KV SD BD 3EC VOLTAGE

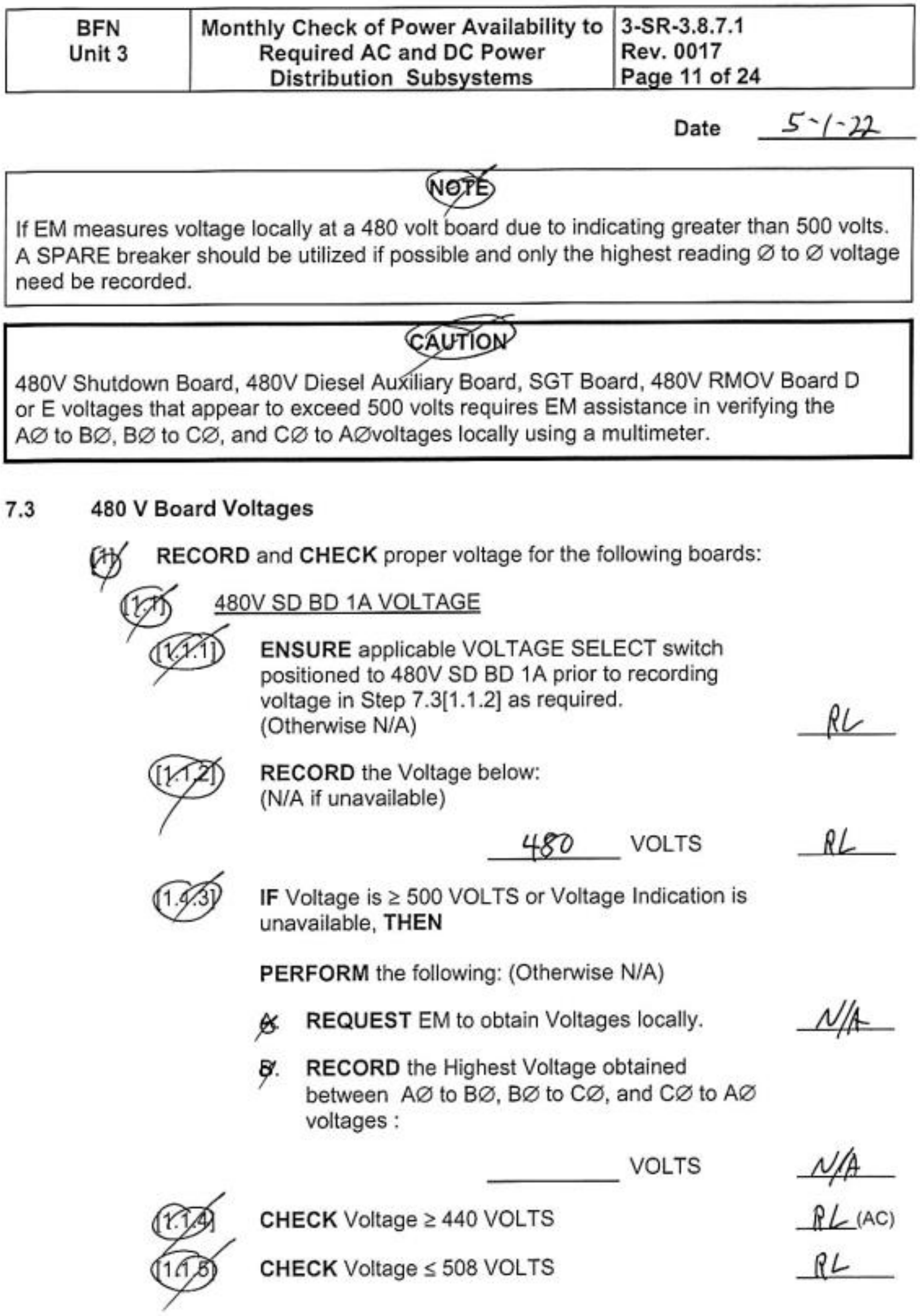
Voltage 4200 (3950 - 4400 VOLTS) RL (AC)

G. 4KV SD BD 3ED VOLTAGE

Voltage 4250 (3950 - 4400 VOLTS) RL (AC)

[3] **CHECK** 4KV SD BD A or 4KV SD BD B VOLTAGE is satisfactory (See Steps 7.2[2]A and 7.2[2]B). RL (AC)

[4] **CHECK** 4KV SD BD D or 4KV SD BD B VOLTAGE is satisfactory (See Steps 7.2[2]B and 7.2[2]C). RL (AC)



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NOTE

If EM measures voltage locally at a 480 volt board due to indicating greater than 500 volts. A SPARE breaker should be utilized if possible and only the highest reading Ø to Ø voltage need be recorded.

CAUTION

480V Shutdown Board, 480V Diesel Auxiliary Board, SGT Board, 480V RMOV Board D or E voltages that appear to exceed 500 volts requires EM assistance in verifying the AØ to BØ, BØ to CØ, and CØ to AØvoltages locally using a multimeter.

7.3 480 V Board Voltages

[1] **RECORD** and **CHECK** proper voltage for the following boards:

[1.1] 480V SD BD 1A VOLTAGE

[1.1.1] **ENSURE** applicable VOLTAGE SELECT switch positioned to 480V SD BD 1A prior to recording voltage in Step 7.3[1.1.2] as required. (Otherwise N/A) RL

[1.1.2] **RECORD** the Voltage below: (N/A if unavailable)

480 VOLTS RL

[1.1.3] **IF** Voltage is ≥ 500 VOLTS or Voltage Indication is unavailable, **THEN**

PERFORM the following: (Otherwise N/A)

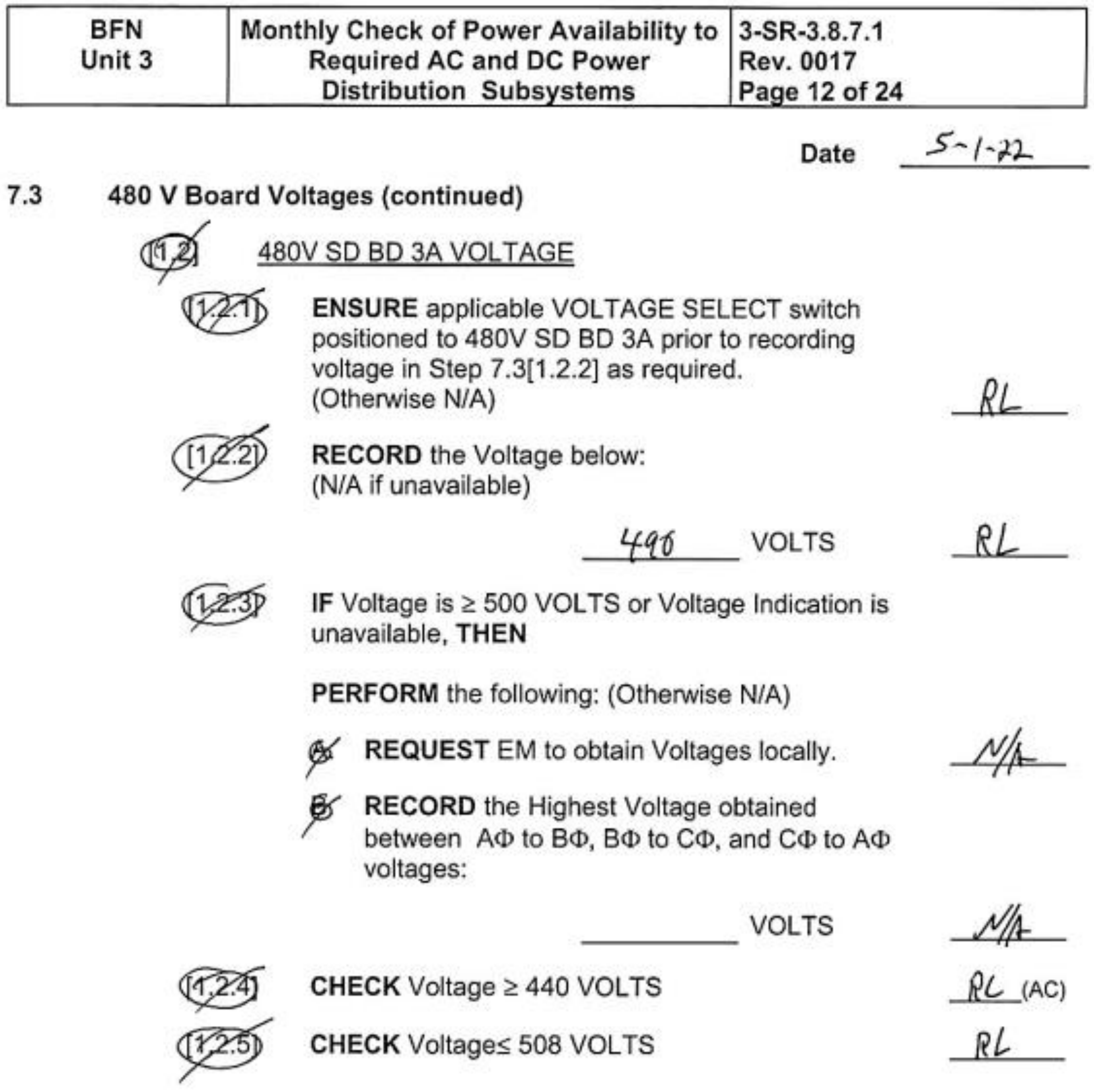
A. **REQUEST** EM to obtain Voltages locally. N/A

B. **RECORD** the Highest Voltage obtained between AØ to BØ, BØ to CØ, and CØ to AØ voltages :

________ VOLTS N/A

[1.1.4] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.1.5] **CHECK** Voltage ≤ 508 VOLTS RL



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7.3 480 V Board Voltages (continued)

[1.2] 480V SD BD 3A VOLTAGE

[1.2.1] **ENSURE** applicable VOLTAGE SELECT switch positioned to 480V SD BD 3A prior to recording voltage in Step 7.3[1.2.2] as required. (Otherwise N/A) RL

[1.2.2] **RECORD** the Voltage below: (N/A if unavailable)

490 VOLTS RL

[1.2.3] **IF** Voltage is ≥ 500 VOLTS or Voltage Indication is unavailable, **THEN**

PERFORM the following: (Otherwise N/A)

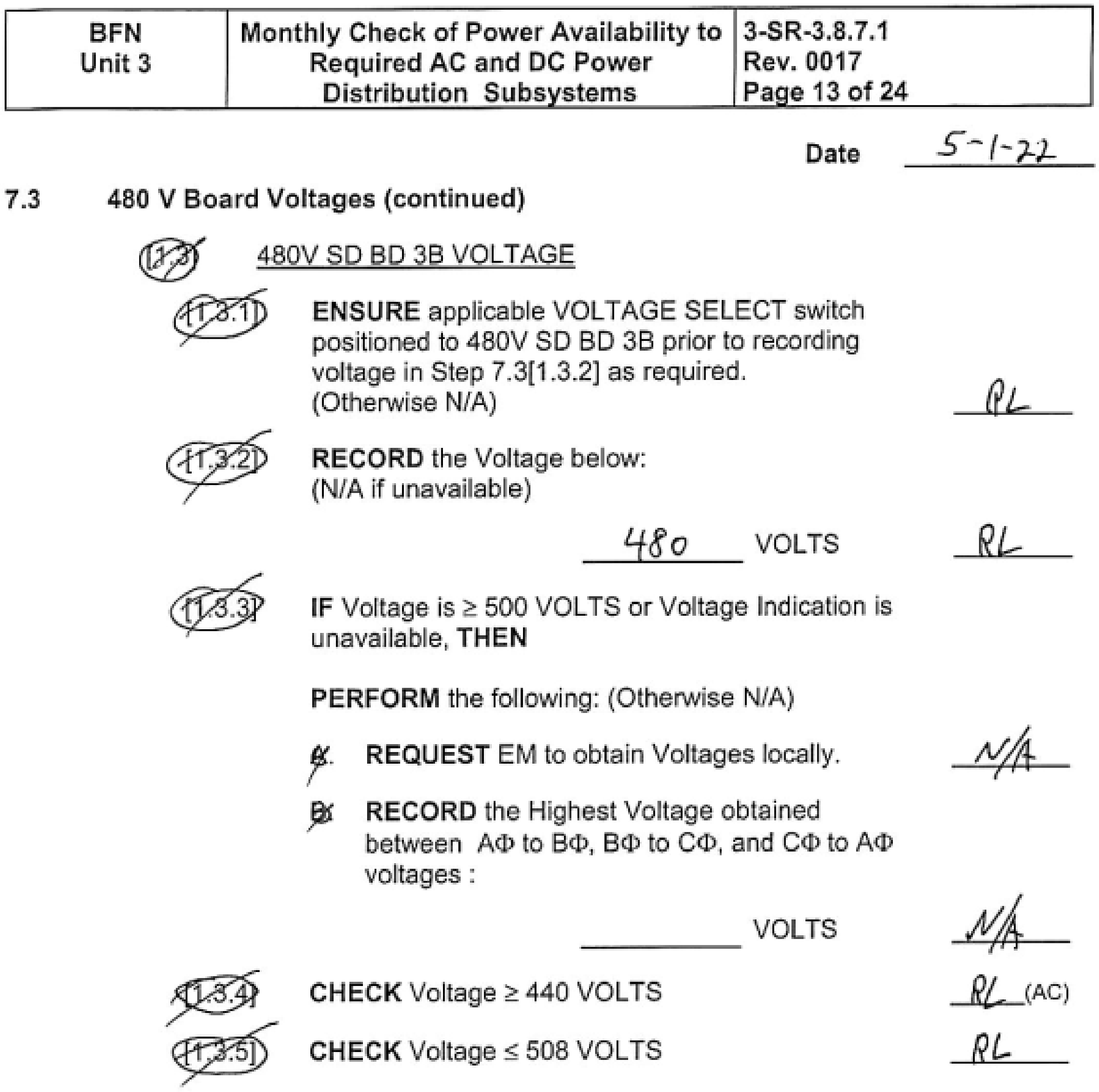
A. **REQUEST** EM to obtain Voltages locally. N/A

B. **RECORD** the Highest Voltage obtained between AΦ to BΦ, BΦ to CΦ, and CΦ to AΦ voltages:

________ VOLTS N/A

[1.2.4] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.2.5] **CHECK** Voltage≤ 508 VOLTS RL



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7.3 480 V Board Voltages (continued)

~~[1.3]~~ 480V SD BD 3B VOLTAGE

~~[1.3.1]~~ **ENSURE** applicable VOLTAGE SELECT switch positioned to 480V SD BD 3B prior to recording voltage in Step 7.3[1.3.2] as required. (Otherwise N/A) RL

~~[1.3.2]~~ **RECORD** the Voltage below: (N/A if unavailable)

480 VOLTS RL

~~[1.3.3]~~ **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

~~[1.3.4]~~ **CHECK** Voltage ≥ 440 VOLTS RL (AC)

~~[1.3.5]~~ **CHECK** Voltage ≤ 508 VOLTS RL

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7.3 480 V Board Voltages (continued)

[1.4] 480V RMOV BD 3A VOLTAGE

[1.4.1] **RECORD** the Voltage below:
(N/A if unavailable)

490 VOLTS RL

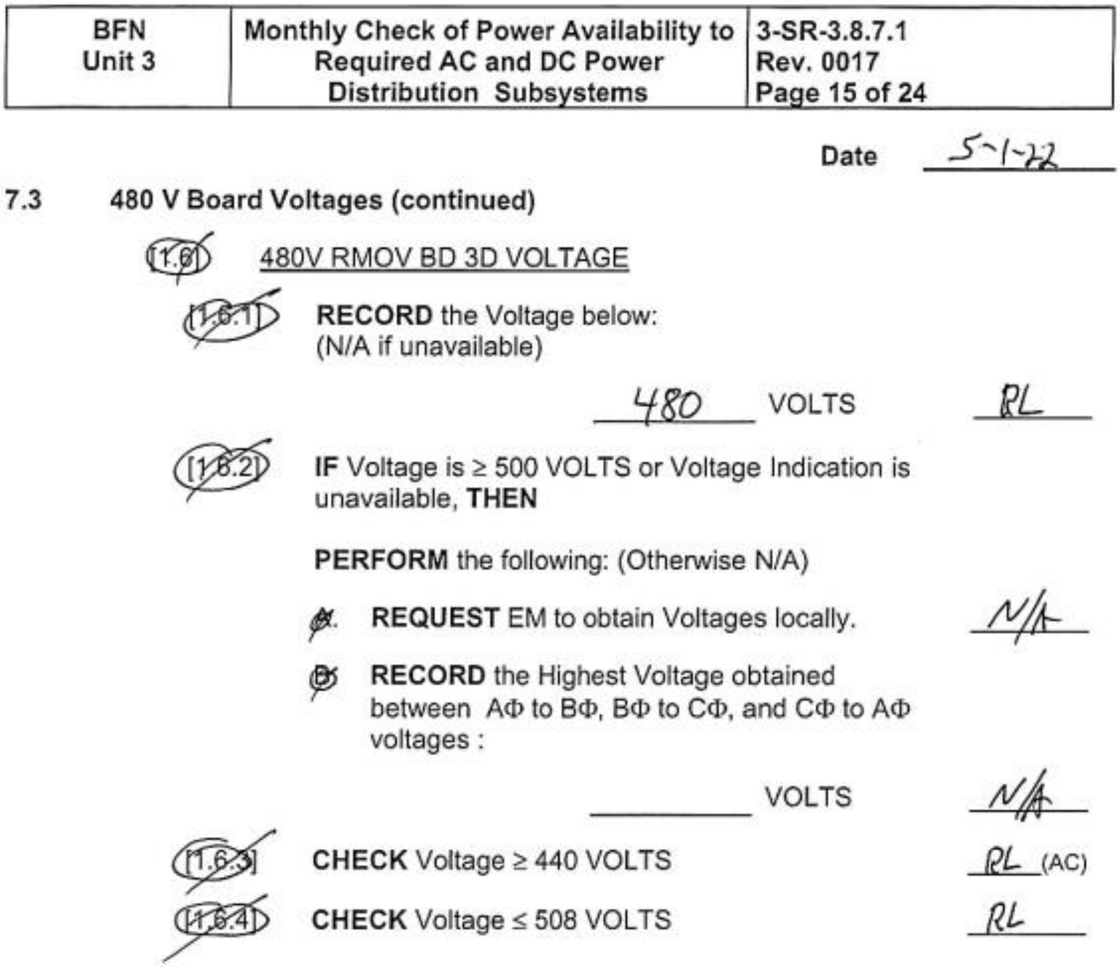
[1.4.2] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.5] 480V RMOV BD 3B VOLTAGE

[1.5.1] **RECORD** the Voltage below:
(N/A if unavailable)

496 VOLTS RL

[1.5.2] **CHECK** Voltage ≥ 440 VOLTS RL (AC)



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7.3 480 V Board Voltages (continued)

[1.6] 480V RMOV BD 3D VOLTAGE

[1.6.1] **RECORD** the Voltage below:
(N/A if unavailable)

480 VOLTS RL

[1.6.2] **IF** Voltage is ≥ 500 VOLTS or Voltage Indication is unavailable, **THEN**

PERFORM the following: (Otherwise N/A)

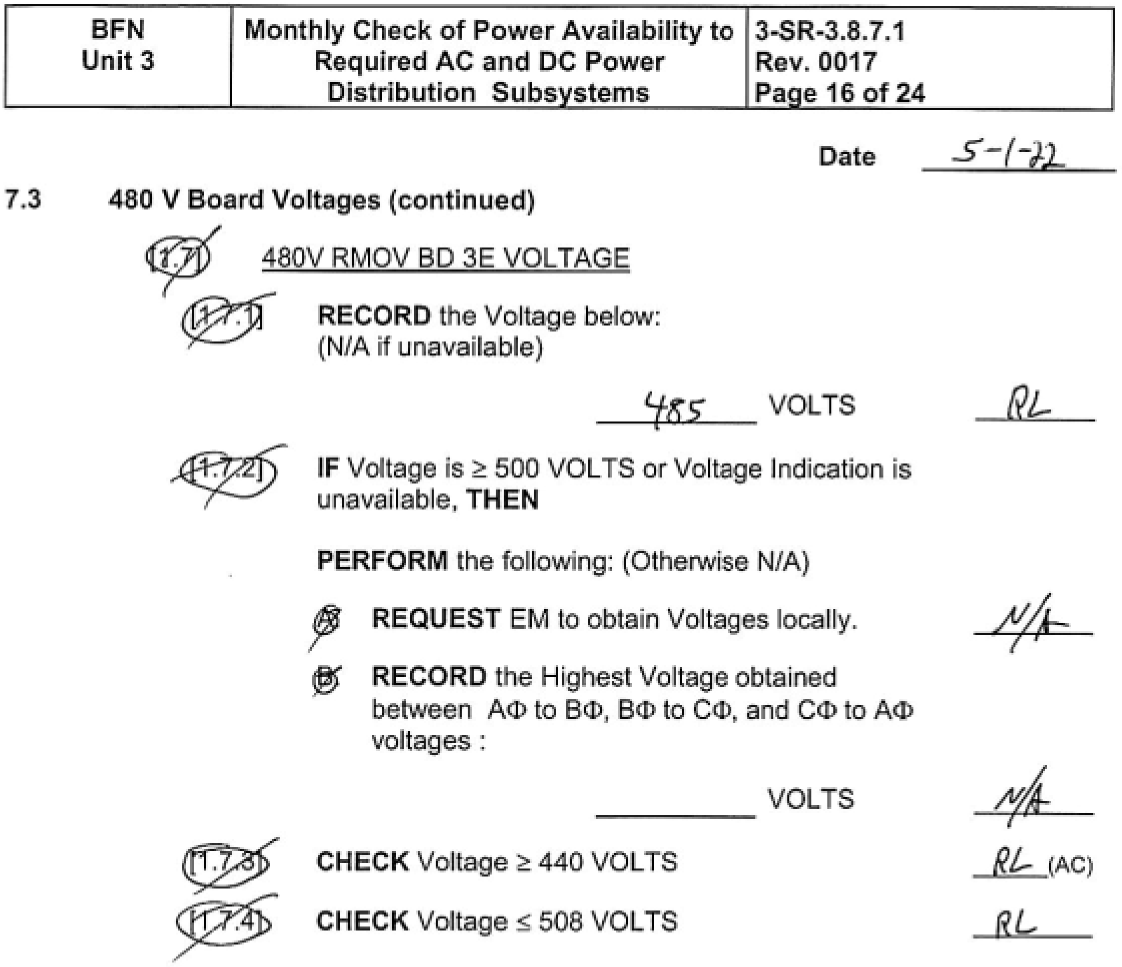
A. **REQUEST** EM to obtain Voltages locally. N/A

B. **RECORD** the Highest Voltage obtained between AΦ to BΦ, BΦ to CΦ, and CΦ to AΦ voltages :

__________ VOLTS N/A

[1.6.3] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.6.4] **CHECK** Voltage ≤ 508 VOLTS RL



BFN Unit 3	Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems	3-SR-3.8.7.1 Rev. 0017 Page 16 of 24

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7.3 **480 V Board Voltages (continued)**

[1.7] 480V RMOV BD 3E VOLTAGE

[1.7.1] **RECORD** the Voltage below:
(N/A if unavailable)

485 VOLTS RL

[1.7.2] **IF** Voltage is ≥ 500 VOLTS or Voltage Indication is unavailable, **THEN**

PERFORM the following: (Otherwise N/A)

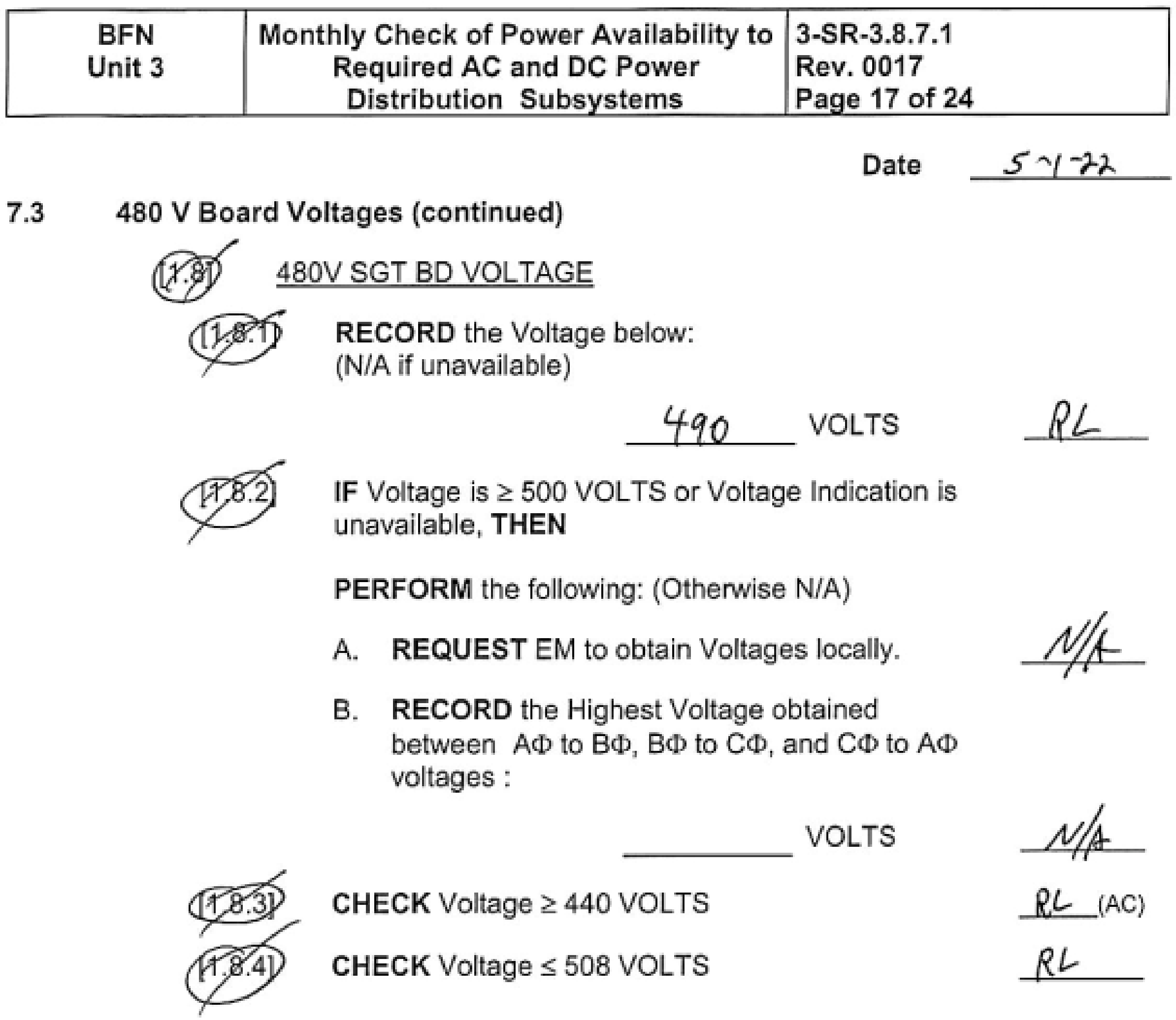
A. **REQUEST** EM to obtain Voltages locally. N/A

B. **RECORD** the Highest Voltage obtained between AΦ to BΦ, BΦ to CΦ, and CΦ to AΦ voltages :

________ VOLTS N/A

[1.7.3] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.7.4] **CHECK** Voltage ≤ 508 VOLTS RL



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7.3 480 V Board Voltages (continued)

[1.8] 480V SGT BD VOLTAGE

[1.8.1] **RECORD** the Voltage below:
(N/A if unavailable)

490 VOLTS RL

[1.8.2] **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

[1.8.3] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.8.4] **CHECK** Voltage ≤ 508 VOLTS RL

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7.3 480 V Board Voltages (continued)

[1.9] 480V DSL AUX BD A VOLTAGE

[1.9.1] **ENSURE** applicable VOLTAGE SELECT switch positioned to 480V DSL AUX BD A prior to recording voltage in Step 7.3[1.9.2] as required. (Otherwise N/A) RL

[1.9.2] **RECORD** the Voltage below: (N/A if unavailable)

485 VOLTS RL

[1.9.3] **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

[1.9.4] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.9.5] **CHECK** Voltage ≤ 508 VOLTS RL

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7.3 **480 V Board Voltages (continued)**

[1.10] 480V DSL AUX BD B VOLTAGE

[1.10.1] **ENSURE** applicable VOLTAGE SELECT switch positioned to 480V DSL AUX BD B prior to recording voltage in Step 7.3[1.10.2] as required. (Otherwise N/A) RL

[1.10.2] **RECORD** the Voltage below: (N/A if unavailable)

480 VOLTS RL

[1.10.3] **IF** Voltage is ≥ 500 VOLTS or Voltage Indication is unavailable, **THEN**

PERFORM the following: (Otherwise N/A)

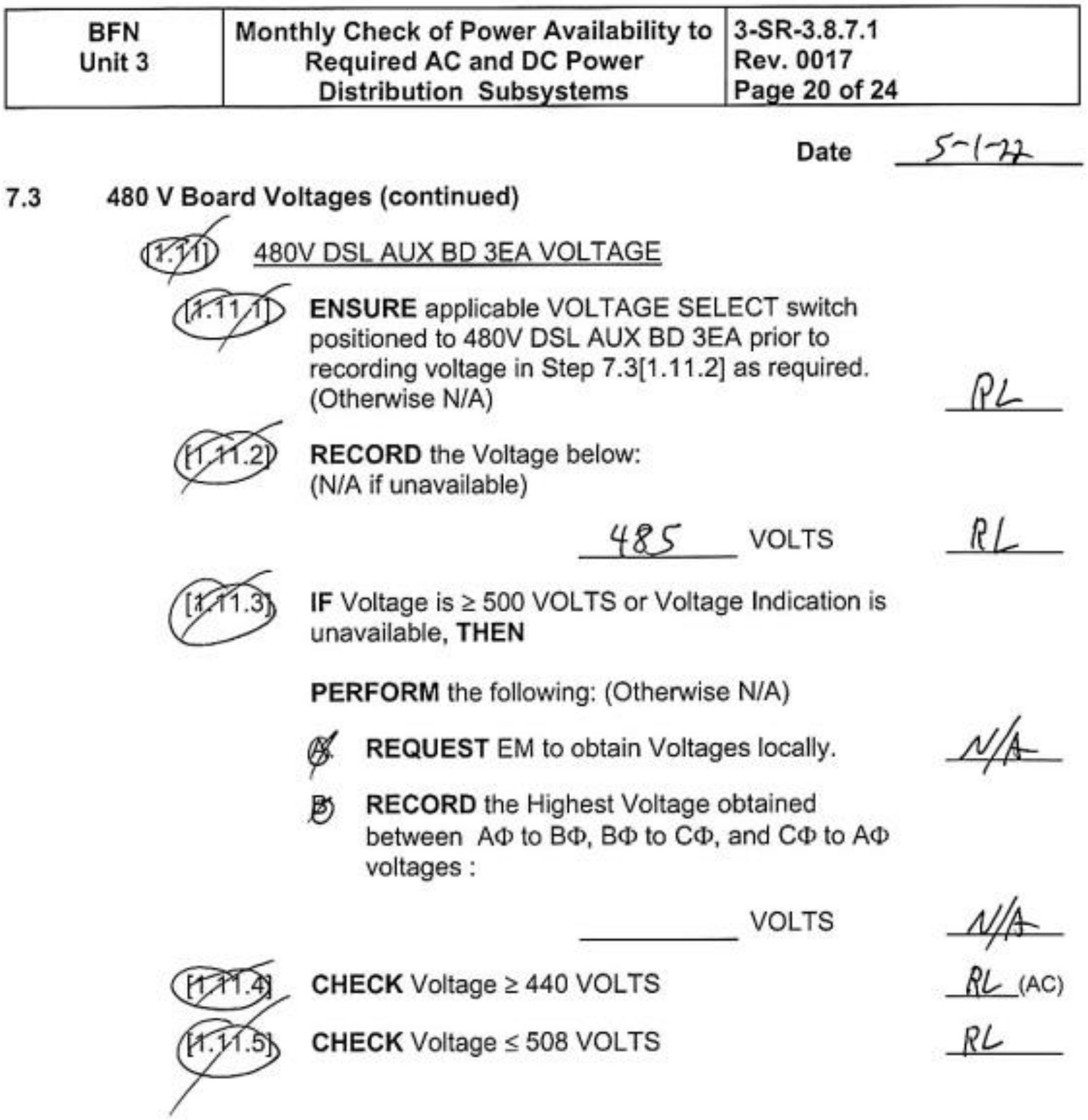
A. **REQUEST** EM to obtain Voltages locally. N/A

B. **RECORD** the Highest Voltage obtained between AΦ to BΦ, BΦ to CΦ, and CΦ to AΦ voltages :

________ VOLTS N/A

[1.10.4] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.10.5] **CHECK** Voltage ≤ 508 VOLTS RL



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7.3 480 V Board Voltages (continued)

[1.11] 480V DSL AUX BD 3EA VOLTAGE

[1.11.1] **ENSURE** applicable VOLTAGE SELECT switch positioned to 480V DSL AUX BD 3EA prior to recording voltage in Step 7.3[1.11.2] as required. (Otherwise N/A) RL

[1.11.2] **RECORD** the Voltage below: (N/A if unavailable)

485 VOLTS RL

[1.11.3] **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

[1.11.4] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.11.5] **CHECK** Voltage ≤ 508 VOLTS RL

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7.3 480 V Board Voltages (continued)

[1.12] 480V DSL AUX BD 3EB VOLTAGE

[1.12.1] **ENSURE** applicable VOLTAGE SELECT switch positioned to 480V DSL AUX BD 3EB prior to recording voltage in Step 7.3[1.12.2] as required. (Otherwise N/A) RL

[1.12.2] **RECORD** the Voltage below: (N/A if unavailable)

490 VOLTS RL

[1.12.3] **IF** Voltage is ≥ 500 VOLTS or Voltage Indication is unavailable, **THEN**

PERFORM the following: (Otherwise N/A)

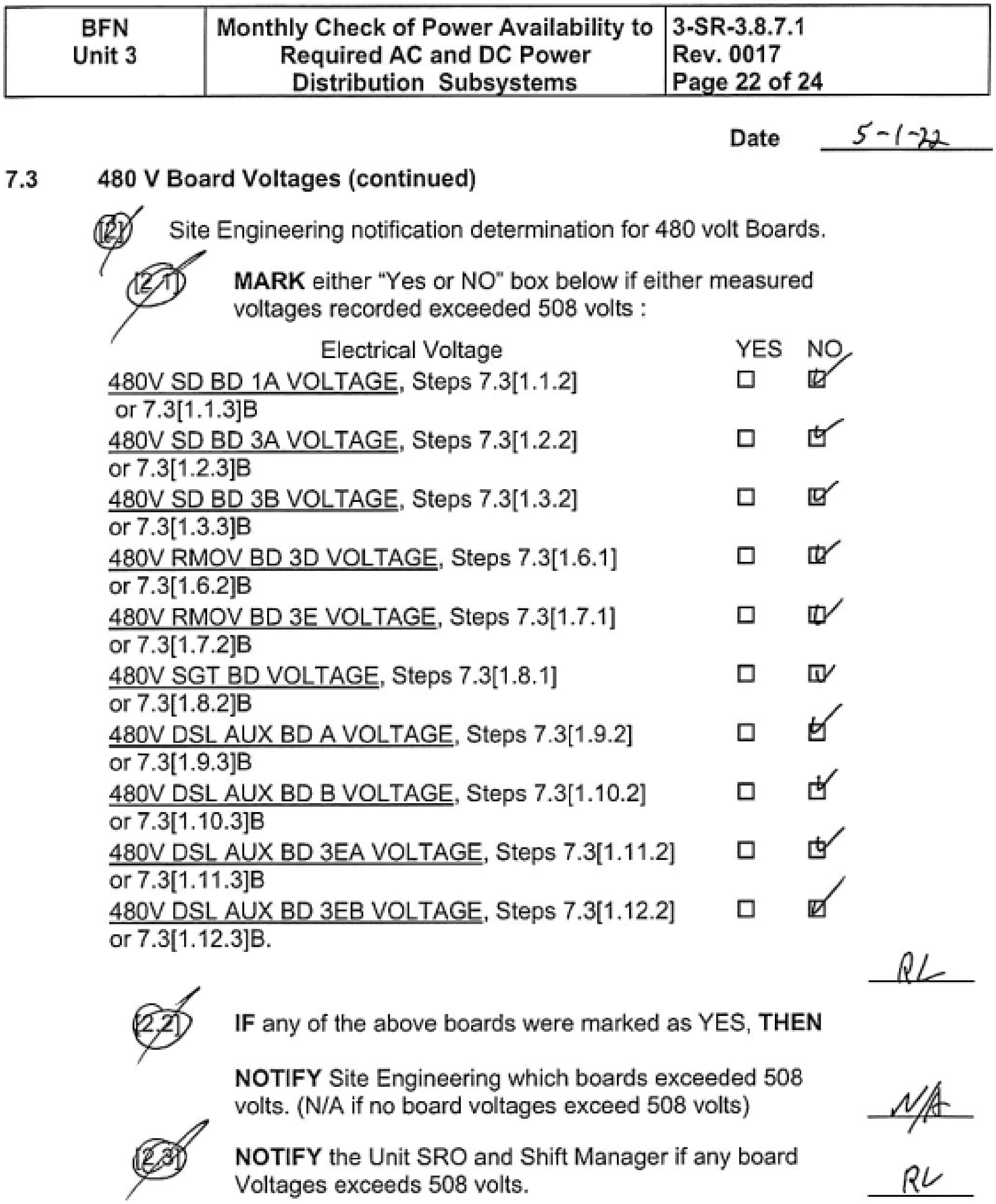
A. **REQUEST** EM to obtain Voltages locally. N/A

B. **RECORD** the Highest Voltage obtained between AΦ to BΦ, BΦ to CΦ, and CΦ to AΦ voltages :

________ VOLTS N/A

[1.12.4] **CHECK** Voltage ≥ 440 VOLTS RL (AC)

[1.12.5] **CHECK** Voltage ≤ 508 VOLTS RL



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7.3 480 V Board Voltages (continued)

[2] Site Engineering notification determination for 480 volt Boards.

[2.1] **MARK** either "Yes or NO" box below if either measured voltages recorded exceeded 508 volts :

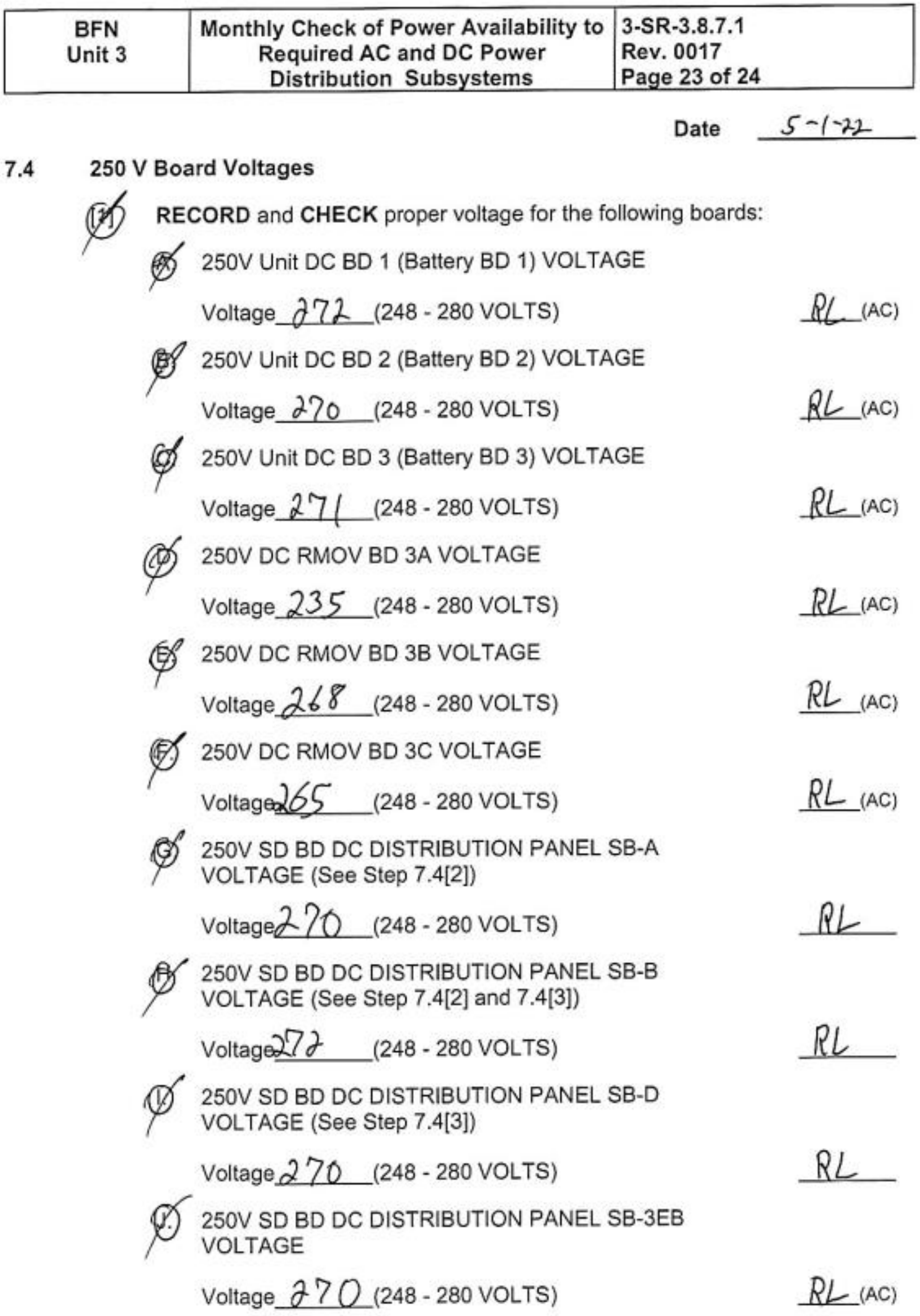
Electrical Voltage	YES	NO
480V SD BD 1A VOLTAGE, Steps 7.3[1.1.2] or 7.3[1.1.3]B	☐	☑
480V SD BD 3A VOLTAGE, Steps 7.3[1.2.2] or 7.3[1.2.3]B	☐	☑
480V SD BD 3B VOLTAGE, Steps 7.3[1.3.2] or 7.3[1.3.3]B	☐	☑
480V RMOV BD 3D VOLTAGE, Steps 7.3[1.6.1] or 7.3[1.6.2]B	☐	☑
480V RMOV BD 3E VOLTAGE, Steps 7.3[1.7.1] or 7.3[1.7.2]B	☐	☑
480V SGT BD VOLTAGE, Steps 7.3[1.8.1] or 7.3[1.8.2]B	☐	☑
480V DSL AUX BD A VOLTAGE, Steps 7.3[1.9.2] or 7.3[1.9.3]B	☐	☑
480V DSL AUX BD B VOLTAGE, Steps 7.3[1.10.2] or 7.3[1.10.3]B	☐	☑
480V DSL AUX BD 3EA VOLTAGE, Steps 7.3[1.11.2] or 7.3[1.11.3]B	☐	☑
480V DSL AUX BD 3EB VOLTAGE, Steps 7.3[1.12.2] or 7.3[1.12.3]B.	☐	☑

RL

[2.2] **IF** any of the above boards were marked as YES, **THEN**

NOTIFY Site Engineering which boards exceeded 508 volts. (N/A if no board voltages exceed 508 volts) N/A

[2.3] **NOTIFY** the Unit SRO and Shift Manager if any board Voltages exceeds 508 volts. RL



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7.4 250 V Board Voltages

[1] **RECORD** and **CHECK** proper voltage for the following boards:

A. 250V Unit DC BD 1 (Battery BD 1) VOLTAGE

Voltage 272 (248 - 280 VOLTS) RL (AC)

B. 250V Unit DC BD 2 (Battery BD 2) VOLTAGE

Voltage 270 (248 - 280 VOLTS) RL (AC)

C. 250V Unit DC BD 3 (Battery BD 3) VOLTAGE

Voltage 271 (248 - 280 VOLTS) RL (AC)

D. 250V DC RMOV BD 3A VOLTAGE

Voltage 235 (248 - 280 VOLTS) RL (AC)

E. 250V DC RMOV BD 3B VOLTAGE

Voltage 268 (248 - 280 VOLTS) RL (AC)

F. 250V DC RMOV BD 3C VOLTAGE

Voltage 265 (248 - 280 VOLTS) RL (AC)

G. 250V SD BD DC DISTRIBUTION PANEL SB-A VOLTAGE (See Step 7.4[2])

Voltage 270 (248 - 280 VOLTS) RL

H. 250V SD BD DC DISTRIBUTION PANEL SB-B VOLTAGE (See Step 7.4[2] and 7.4[3])

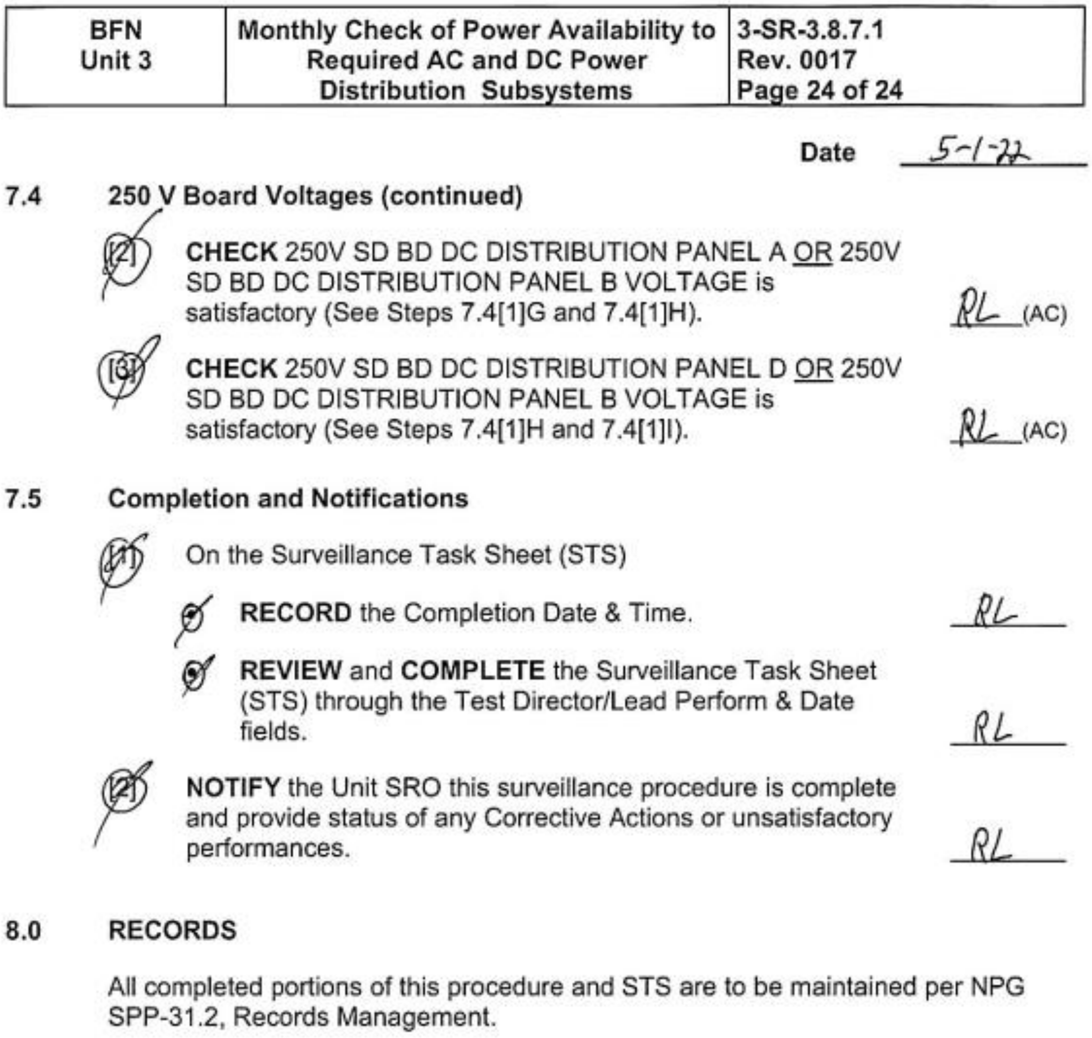
Voltage 272 (248 - 280 VOLTS) RL

I. 250V SD BD DC DISTRIBUTION PANEL SB-D VOLTAGE (See Step 7.4[3])

Voltage 270 (248 - 280 VOLTS) RL

J. 250V SD BD DC DISTRIBUTION PANEL SB-3EB VOLTAGE

Voltage 270 (248 - 280 VOLTS) RL (AC)



BFN **Unit 3**	**Monthly Check of Power Availability to Required AC and DC Power Distribution Subsystems**	**3-SR-3.8.7.1** **Rev. 0017** **Page 24 of 24**

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7.4 250 V Board Voltages (continued)

[2] **CHECK** 250V SD BD DC DISTRIBUTION PANEL A OR 250V SD BD DC DISTRIBUTION PANEL B VOLTAGE is satisfactory (See Steps 7.4[1]G and 7.4[1]H). RL (AC)

[3] **CHECK** 250V SD BD DC DISTRIBUTION PANEL D OR 250V SD BD DC DISTRIBUTION PANEL B VOLTAGE is satisfactory (See Steps 7.4[1]H and 7.4[1]I). RL (AC)

7.5 Completion and Notifications

[1] On the Surveillance Task Sheet (STS)

- **RECORD** the Completion Date & Time. RL
- **REVIEW** and **COMPLETE** the Surveillance Task Sheet (STS) through the Test Director/Lead Perform & Date fields. RL

[2] **NOTIFY** the Unit SRO this surveillance procedure is complete and provide status of any Corrective Actions or unsatisfactory performances. RL

8.0 RECORDS

All completed portions of this procedure and STS are to be maintained per NPG SPP-31.2, Records Management.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Emergency Action Level Classification
JPM NUMBER:	752-SRO	REVISION:	1

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:	EMERGENCY PLAN - ADMIN			

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: Given plant conditions, the Examinee is expected to use Emergency Preparedness procedures to classify an event as a SITE AREA EMERGENCY within 15 minutes and fill out the required notification form also within 15 minutes.

Operator Fundamental evaluated:
OF-1 Monitoring Plant Indications and Conditions Closely

PRA: N/A

REFERENCES/PROCEDURES NEEDED: EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3
EPIPs 1, 2, 3, 4, and 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	2/9/2022	ALL	Updated JPM

Procedure Revisions

Procedure	Revision
EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3	61
EPIP-1	61
EPIP-2	41
EPIP-3	44
EPIP-4	44
EPIP-5	58

****

Classroom: Have copies of Appendix A from EPIP 2, 3, 4, 5

****

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 1, 2, and 3 are operating at 100% Reactor Power with the following plant conditions:

- ‘B’ Diesel Generator (EDG) is tagged
- 1045: Tornado strikes the switchyard causing a total Loss of Offsite Power
- ‘A’ EDG failed to tie to ‘A’ 4KV Shutdown Board
- 1050: Unit 1, 2 and 3 are maintaining Reactor Water Level (+) 2 to (+) 51 inches with RCIC
- 1100: ‘C’ EDG develops a severe oil leak and is emergency shutdown
- 1130: ‘D’ EDG output breaker trips
    - Elect. Maintenance reports that ‘B’ EDG will be returned to service by 1400
- 91 meter Met Tower Data: 15 minute average Wind Direction is from 175 degrees and 15 minute average Wind Speed is 7 mph

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:

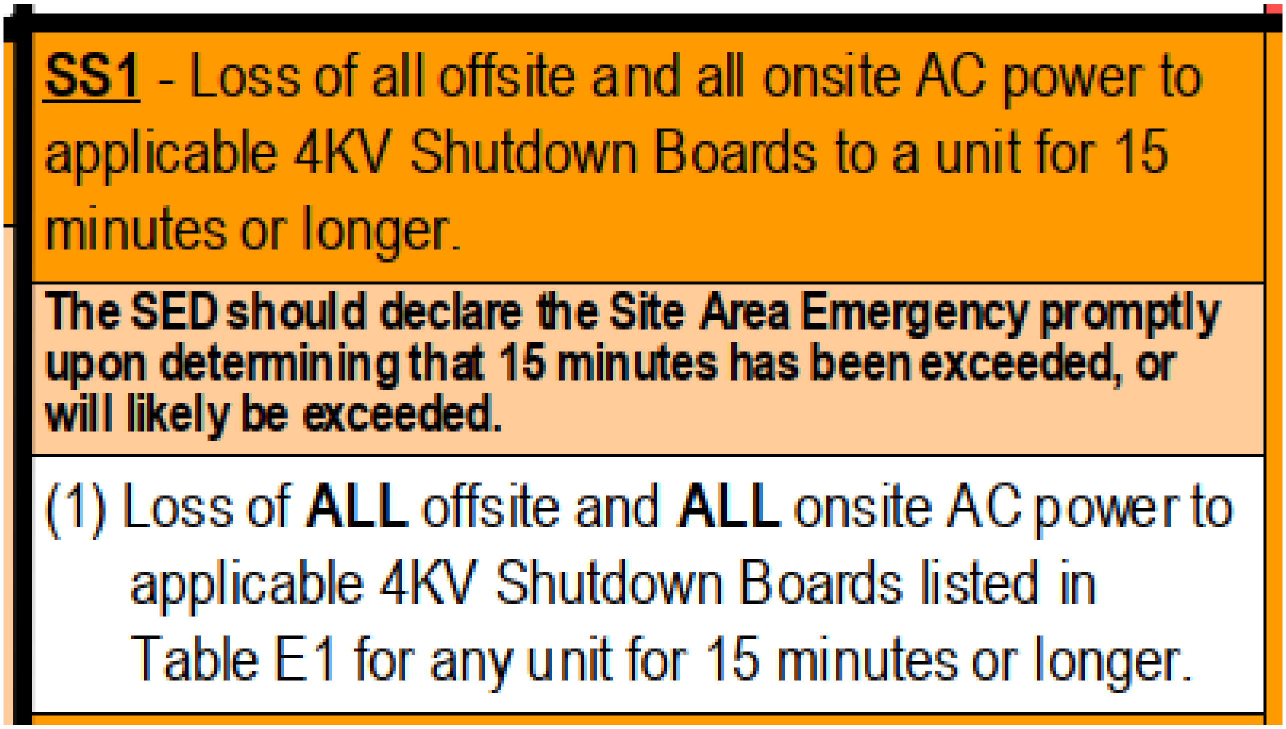
Classify the Event **AND** Record the time of Event Classification,

then **IMMEDIATELY** raise your hand.

This JPM is TIME CRITICAL

KEY

SS1:



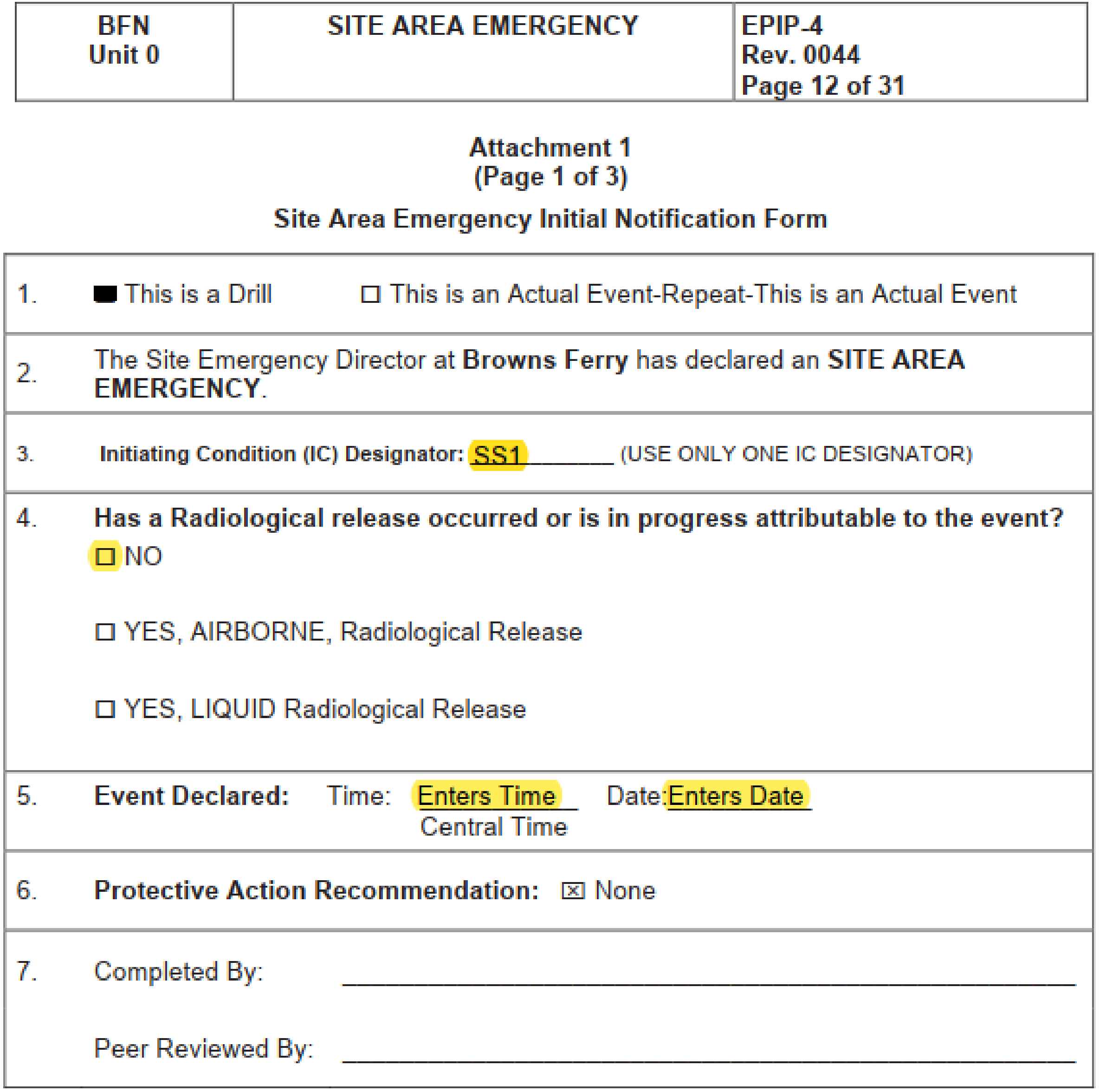
SS1 - Loss of all offsite and all onsite AC power to applicable 4KV Shutdown Boards to a unit for 15 minutes or longer.

The SED should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Loss of **ALL** offsite and **ALL** onsite AC power to applicable 4KV Shutdown Boards listed in Table E1 for any unit for 15 minutes or longer.

KEY

KEY



BFN Unit 0	SITE AREA EMERGENCY	EPIP-4 Rev. 0044 Page 12 of 31

Attachment 1
(Page 1 of 3)

Site Area 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 an **SITE AREA EMERGENCY**.

3. **Initiating Condition (IC) Designator:** SS1_______ (USE ONLY ONE IC DESIGNATOR)

4. **Has a Radiological release occurred or is in progress attributable to the event?**

   ☐ NO

   ☐ YES, AIRBORNE, Radiological Release

   ☐ YES, LIQUID Radiological Release

5. **Event Declared:** Time: Enters Time_ Date:Enters Date
   Central Time

6. **Protective Action Recommendation:** ☒ None

7. Completed By: ____

   Peer Reviewed By: ____

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: Hard copies of EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3 will be available.

EXAMINER NOTE: This JPM has two Time Critical sections. The candidate will have 15 minutes to classify the event once they understand their task, then 15 minutes to complete any required paperwork for Notification after they complete the Classification.

<u>Step 1</u>:

Classifies the Event using EPIP-1, Emergency Classification Procedure

3.1 Precautions/Limitations (page 6 of 143)

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).

<u>Expected Action(s):</u>

Examinee refers to EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3. Given the plant conditions, within 15 minutes, Declares **SITE AREA EMERGENCY - SS1** - Loss of all offsite and onsite AC power to applicable 4KV Shutdown Boards to a unit for 15 minutes or longer based on the following listed in Table E1:

- ‘A’, ‘B’, ‘C’, and ‘D’ EDGs are applicable to their respective 4KV Shutdown Boards ‘A’, ‘B’, ‘C’, and ‘D’ Boards. The expected return to service of at least one EDG in less than the 4 hours prevents the upgrade of the classification to a GENERAL EMERGENCY.

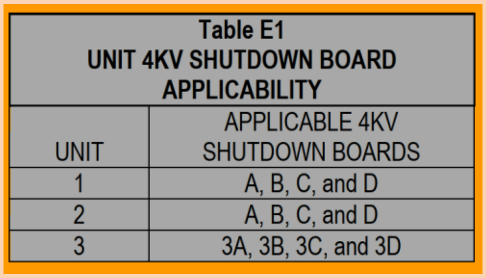


Table E1 UNIT 4KV SHUTDOWN BOARD APPLICABILITY	
UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
1	A, B, C, and D
2	A, B, C, and D
3	3A, 3B, 3C, and 3D

TIME CLASSIFICATION COMPLETE: _____________

Critical Step

_____ SAT

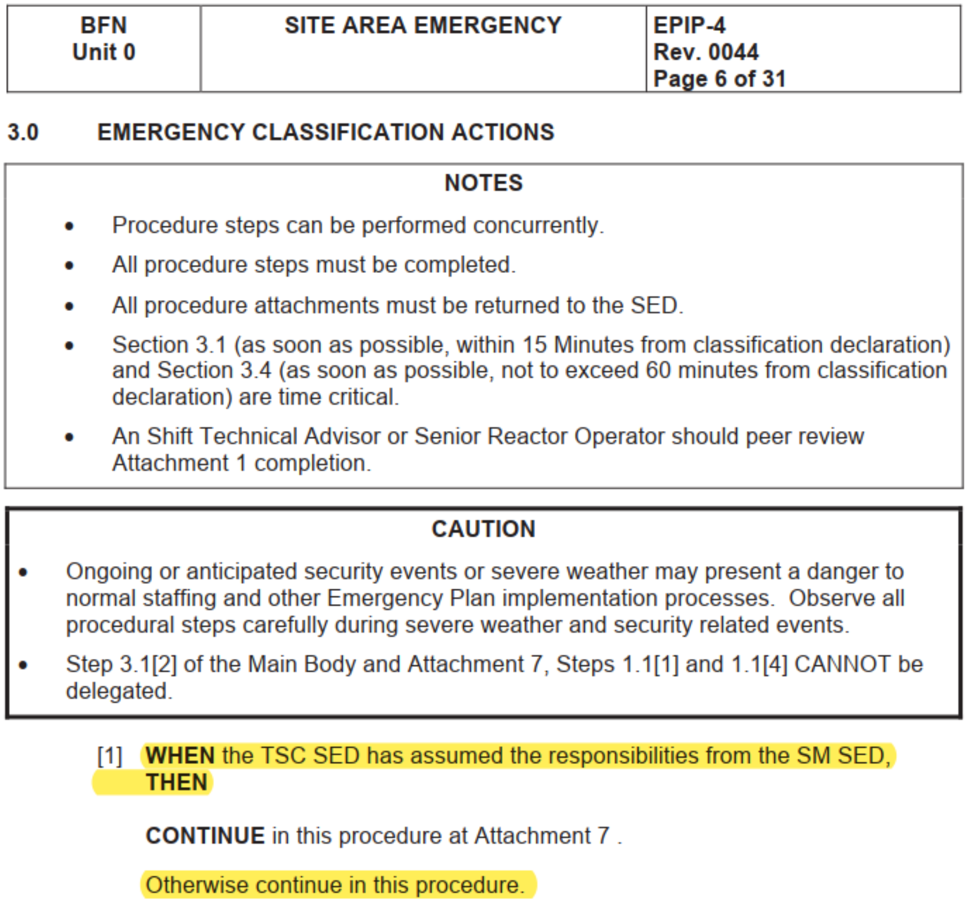
_____UNSAT

_____N/A

STEP / STANDARD	SAT / UNSAT
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. **Record the 2nd TIME START for the candidate's last 15 minute required time in Step 2 below.**	
<u>Step 2</u>: Implements EPIP-4, SITE AREA EMERGENCY. **TIME START** ______________________ <u>Expected Action(s):</u> Examinee implements EPIP-4, SITE AREA EMERGENCY.	**Critical Step** ____ SAT _____UNSAT _____N/A

STEP / STANDARD | SAT / UNSAT

Step 3:



BFN Unit 0	SITE AREA EMERGENCY	EPIP-4 Rev. 0044 Page 6 of 31

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.
- An Shift Technical Advisor or Senior Reactor Operator 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 TSC SED has assumed the responsibilities from the SM SED, **THEN**

CONTINUE in this procedure at Attachment 7 .

Otherwise continue in this procedure.

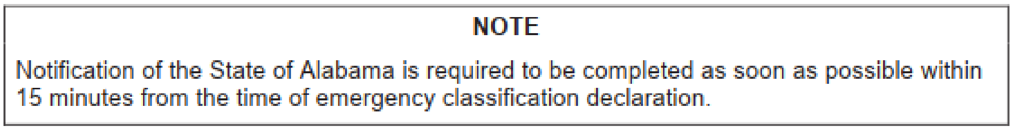
_____ SAT

_____UNSAT

_____N/A

Expected Action(s):

Examinee continues in EPIP-4, Site Area Emergency, as the TSC has not yet been staffed as given in the Initial Conditions.

STEP / STANDARD	SAT / UNSAT
Step 4: **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 SITE AREA EMERGENCY Event Classification: _____ Expected Action(s): Examinee enters the time of Site Area Emergency Event Classification.	_____ SAT _____UNSAT _____N/A
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): Examinee proceeds to Step 3.1[2], as the CECC has not been activated as given in the Initial Conditions.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 6: [2] **COMPLETE** Attachment 1, "Site Area Emergency Initial Notification Form." Expected Action(s): Completes Attachment 1, and simulates notifying the State within 15 minutes by bringing completed EPIP-4, Attachment 1 to the Examiner. The following are **Critical** items on Attachment 1: • (Step 3) Emergency Action Level (EAL) Designator – SS1 • (Step 4) Has a Radiological release occurred or is in progress attributable to the event? ☐ - NO • (Step 5) Time and Date Event Declared	**Critical Step** _____ SAT _____UNSAT _____N/A
Record the 2nd STOP TIME for the candidate's last 15 minute required time below.	
EXAMINER CUE: After candidate hands the examiner their documented INITIATING CUE sheet AND EPIP-4, Attachment 1, inform the candidate "Your task is complete."	

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. Unit 1, 2, and 3 are operating at 100% Reactor Power with the following plant conditions:

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INITIATING CUE:

Classify the Event **AND** Record the time of Event Classification,

then **IMMEDIATELY** raise your hand.

This JPM is TIME CRITICAL

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Determine ACTIONS required to allow releases in accordance with 0-ODCM-001, OFFSITE DOSE CALCULATION MANUAL
JPM NUMBER:	749-SRO	REVISION:	0

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 4.3
K/A STATEMENT:	2.3.11 Ability to control radiation releases
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: 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: 749-SRO

SRO ____ DATE: ________

TASK STANDARD: Given that a Turbine Building Radiation Monitor is taken out of service for maintenance, the Examinee is expected to identify the correct governing plant procedure and the ACTIONS required to allow continued releases.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: ODCM

VALIDATION TIME: 15 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
0	03/15/2021	ALL	Initial revision

Procedure Revisions

Procedure	Revision
0-ODCM-001	25

****

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 Nuclear Unit Senior Operator (NUSO) with the following conditions:

- The Reactor is operating at 100% RTP
- Work Control just notified you that 2-RM-90-249, TURBINE BUILDING EXHAUST RADIATION MONITOR, was just tagged for scheduled maintenance
- Maintenance is expected to last 12 hours

Work Control requests that you ensure compliance with all approved plant procedures for releases.

INITIATING CUE: Given the conditions above, determine the following to ensure compliance for releases via the affected pathway:

- What is/are the required ACTION(s) to allow releases to continue

Answer:

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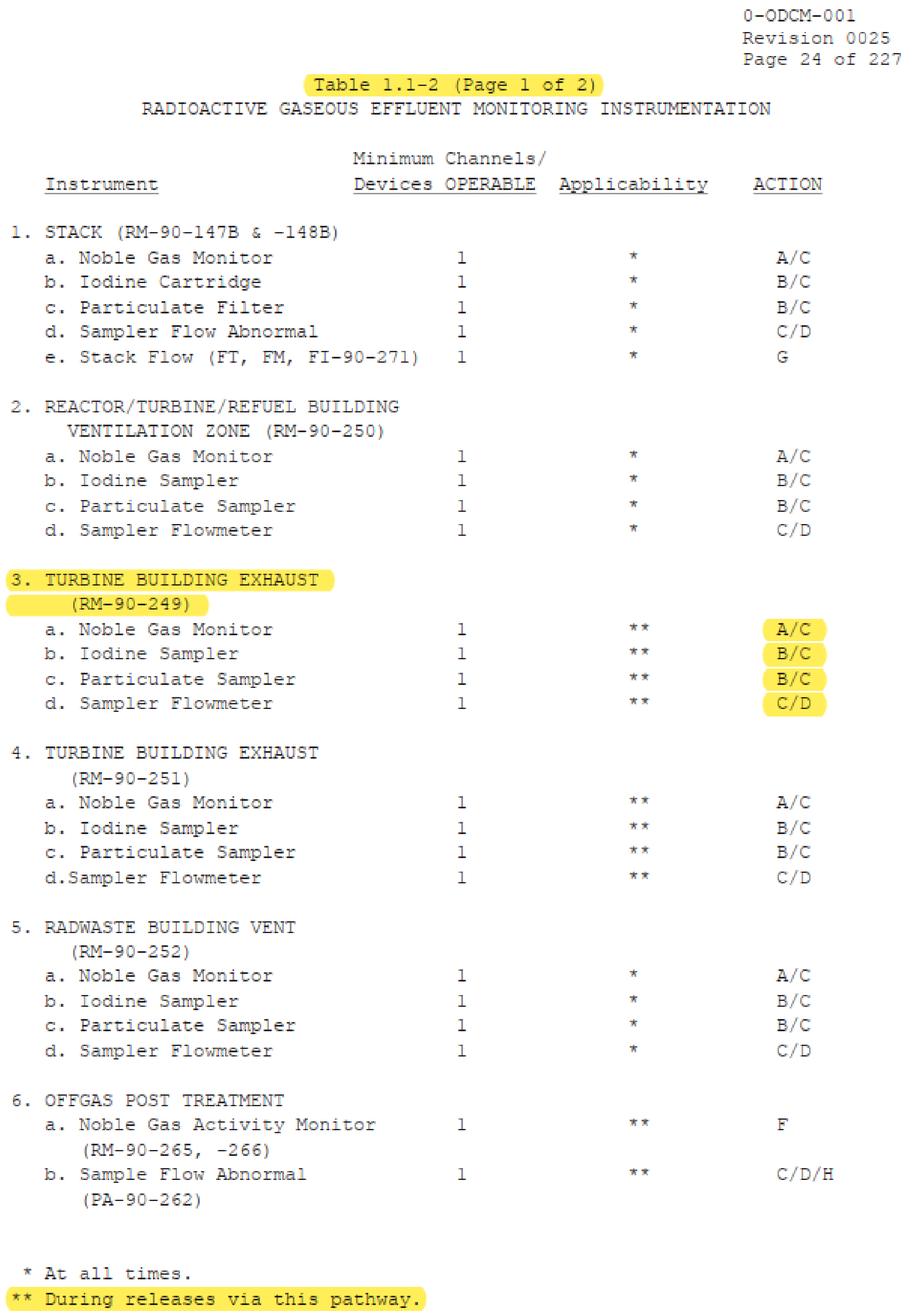
START TIME: __________

STEP / STANDARD	SAT / UNSAT
<u>Step 1</u>: Refers to 0-ODCM-001, OFFSITE DOSE CALCULATION MANUAL (see JPM attached page 6 of 8) <u>Expected Action(s):</u> Examinee refers to 0-ODCM-001, OFFSITE DOSE CALCULATION MANUAL, Table 1.1-2 (Page 1 of 2) for applicability to allow releases to continue via this pathway with 2-RM-90-249, Turbine Building Exhaust Radiation Monitor while tagged out of service.	**Critical Step** _____ SAT _____UNSAT _____N/A
<u>Step 2</u>: Refers to 0-ODCM-001, OFFSITE DOSE CALCULATION MANUAL Table 1.1-2 (Page 2 of 2) (see JPM attached page 7 of 8) <u>Expected Action(s):</u> Given 2-RM-90-249, Turbine Building Exhaust Radiation Monitor being tagged out, examinee determines that the following ACTIONS are required to allow effluent releases via the affected pathway to continue: • ACTION ‘A’ – A temporary monitoring system is installed or grab samples are taken and analyzed at least once every 8 hours • ACTION ‘B’ – Samples are continuously collected with auxiliary sampling equipment for periods of 7 days and analyzed within 48 hours of the end of the sampling period • ACTION ‘D’ – The flow rate is estimated at once per 4 hours	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER NOTE: ACTION ‘C’ is not applicable since the out of service time is expected to exceed the allowable time of 4 hours. **Given in the cue, 2-RM-90-249, Turbine Building Exhaust Radiation Monitor maintenance is expected to last 12 hours.**	



0-ODCM-001
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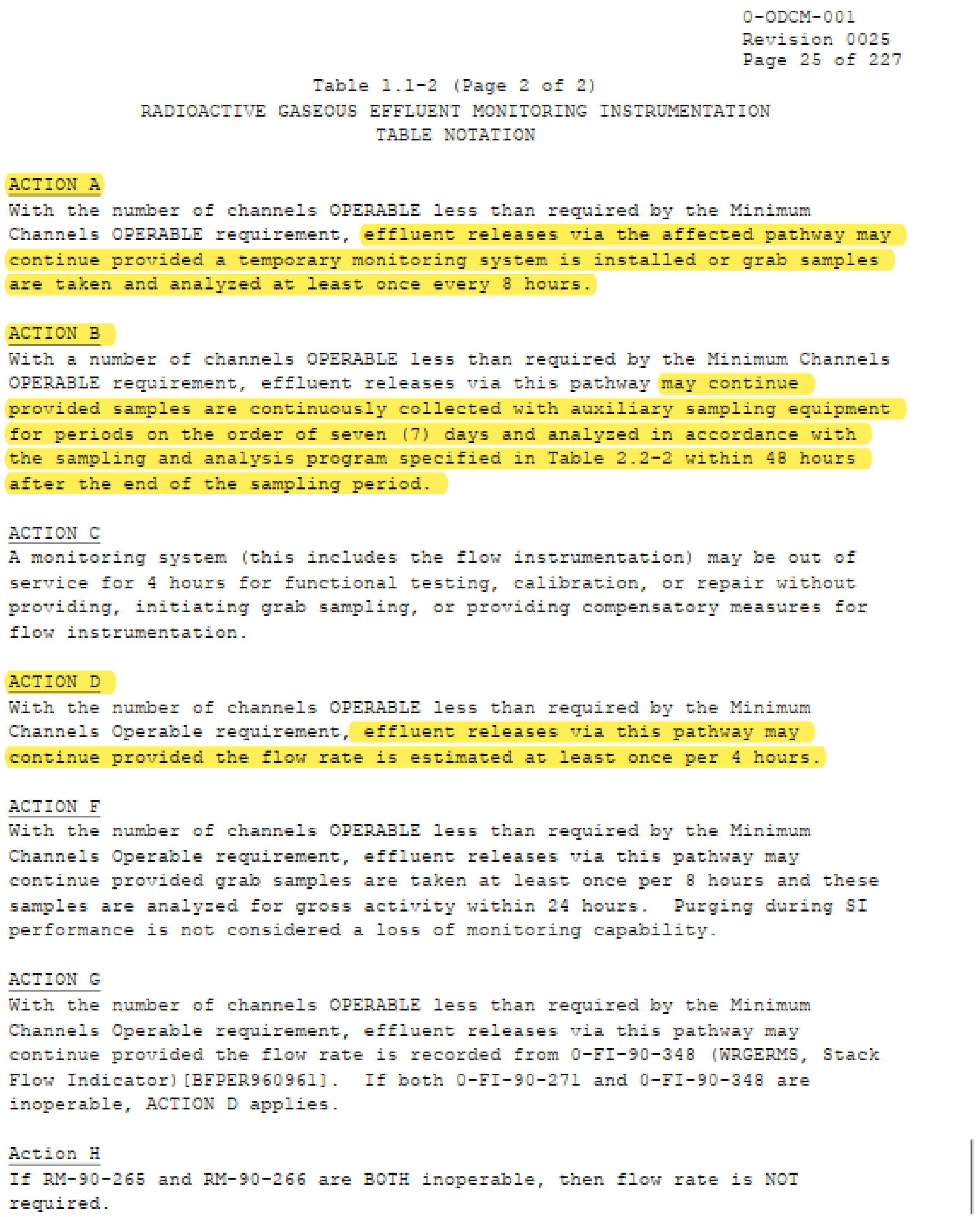
Table 1.1-2 (Page 1 of 2)
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels/ Devices OPERABLE	Applicability	ACTION
1. STACK (RM-90-147B & -148B)			
a. Noble Gas Monitor	1	*	A/C
b. Iodine Cartridge	1	*	B/C
c. Particulate Filter	1	*	B/C
d. Sampler Flow Abnormal	1	*	C/D
e. Stack Flow (FT, FM, FI-90-271)	1	*	G
2. REACTOR/TURBINE/REFUEL BUILDING VENTILATION ZONE (RM-90-250)			
a. Noble Gas Monitor	1	*	A/C
b. Iodine Sampler	1	*	B/C
c. Particulate Sampler	1	*	B/C
d. Sampler Flowmeter	1	*	C/D
3. TURBINE BUILDING EXHAUST (RM-90-249)			
a. Noble Gas Monitor	1	**	A/C
b. Iodine Sampler	1	**	B/C
c. Particulate Sampler	1	**	B/C
d. Sampler Flowmeter	1	**	C/D
4. TURBINE BUILDING EXHAUST (RM-90-251)			
a. Noble Gas Monitor	1	**	A/C
b. Iodine Sampler	1	**	B/C
c. Particulate Sampler	1	**	B/C
d.Sampler Flowmeter	1	**	C/D
5. RADWASTE BUILDING VENT (RM-90-252)			
a. Noble Gas Monitor	1	*	A/C
b. Iodine Sampler	1	*	B/C
c. Particulate Sampler	1	*	B/C
d. Sampler Flowmeter	1	*	C/D
6. OFFGAS POST TREATMENT			
a. Noble Gas Activity Monitor (RM-90-265, -266)	1	**	F
b. Sample Flow Abnormal (PA-90-262)	1	**	C/D/H

* At all times.
** During releases via this pathway.



0-ODCM-001
Revision 0025
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Table 1.1-2 (Page 2 of 2)
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
TABLE NOTATION

ACTION A

With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided a temporary monitoring system is installed or grab samples are taken and analyzed at least once every 8 hours.

ACTION B

With a number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment for periods on the order of seven (7) days and analyzed in accordance with the sampling and analysis program specified in Table 2.2-2 within 48 hours after the end of the sampling period.

ACTION C

A monitoring system (this includes the flow instrumentation) may be out of service for 4 hours for functional testing, calibration, or repair without providing, initiating grab sampling, or providing compensatory measures for flow instrumentation.

ACTION D

With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION F

With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours. Purging during SI performance is not considered a loss of monitoring capability.

ACTION G

With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided the flow rate is recorded from 0-FI-90-348 (WRGERMS, Stack Flow Indicator)[BFPER960961]. If both 0-FI-90-271 and 0-FI-90-348 are inoperable, ACTION D applies.

Action H

If RM-90-265 and RM-90-266 are BOTH inoperable, then flow rate is NOT required.

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 2 Nuclear Unit Senior Operator (NUSO) with the following conditions:

- The Reactor is operating at 100% RTP
- Work Control just notified you that 2-RM-90-249, TURBINE BUILDING EXHAUST RADIATION MONITOR, was just tagged for scheduled maintenance
- Maintenance is expected to last 12 hours

Work Control requests that you ensure compliance with all approved plant procedures for releases.

INITIATING CUE: Given the conditions above, determine the following to ensure compliance for releases via the affected pathway:

- What is/are the required ACTION(s) to allow releases to continue

Answer:

____

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