Facility: **Brunswick** Date of Exam: **December, 2016**

Tier	Group	RO K/A Category Points: K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	SRO-Only Points: A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	4	N/A			4	3	N/A		3	20	4	3	7
	2	1	1	1				2	1			1	7	2	1	3
	Tier Totals	4	4	5				6	4			4	27	6	4	10
2. Plant Systems	1	3	2	3	2	3	2	2	3	2	2	2	26	3	2	5
	2	1	1	1	1	1	2	1	1	1	1	1	12	0	2	1	3
	Tier Totals	4	3	4	3	4	4	3	4	3	3	3	38	5	3	8

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

Note:
1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 Radiation Control K/A is allowed if the K/A 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 associated 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 shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in 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

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)	Form ES-401-1	
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A2	G*	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4					X	X	G2.2.12; Knowledge of surveillance procedures. AA2.05; Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Jet pump operability.	3.7 3.4	
295003 Partial or Complete Loss of AC / 6	X						AK1.03; Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Under voltage/degraded voltage effects on electrical loads.	2.9	
295004 Partial or Total Loss of DC Pwr / 6		X					AK2.01; Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: Battery charger.	3.1	
295005 Main Turbine Generator Trip / 3			X				AK3.04; Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: Main generator trip.	3.2	
295006 SCRAM / 1				X			AA1.06; Ability to operate and/or monitor the following as they apply to SCRAM: CRD hydraulic system.	3.5	
295016 Control Room Abandonment / 7			X				AK3.03; Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: Disabling control room controls.	3.5	
295018 Partial or Total Loss of CCW / 8		X					AK2.02; Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: Plant operations.	3.4	
295019 Partial or Total Loss of Inst. Air / 8					X		AA2.02; Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Status of safety-related instrument air system loads.	3.6	
295021 Loss of Shutdown Cooling / 4		X			X		AK2.05; Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: Fuel pool cooling and cleanup system. AA2.03; Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : Reactor water level.	2.5 3.5	
295023 Refueling Acc / 8				X		X	AA1.04; Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Radiation monitoring equipment. G2.2.25; Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	3.4 4.2	
295024 High Drywell Pressure / 5				X			EA1.05; Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: RPS.	3.9	
295025 High Reactor Pressure / 3			X				EK3.02; Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: Recirculation pump trip: Plant-Specific.	3.9	

295026 Suppression Pool High Water Temp. / 5						X X	G2.4.50; Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. G2.1.23; Ability to perform specific system and integrated plant procedures during all modes of plant operation.	2.7 4.4	
295027 High Containment Temperature / 5									
295028 High Drywell Temperature / 5					X		EA2.01; Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell temperature.	4.0	
295030 Low Suppression Pool Wtr Lvl / 5						X	G2.4.50; Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	4.2	
295031 Reactor Low Water Level / 2	X						EK1.03; Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Water level effects on reactor power.	3.7	
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1	X					 X	EK1.03; Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Boron effects on reactor power (SBLC). G2.4.21; Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.2 4.6	
295038 High Off-site Release Rate / 9				X	 X		EA1.01; Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Stack-gas monitoring system: Plant-Specific. EA2.01; Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: †Off-site.	3.9 4.3	
600000 Plant Fire On Site / 8			X		 X		AK3.04; Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site. AA2.07; Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Whether malfunction is due to common-mode electrical failures.	2.8 3.0	
700000 Generator Voltage and Electric Grid Disturbances / 6					X		AA2.03; Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator current outside the capability curve.	3.5	
K/A Category Totals:	3	3	4	4	3/4	3/3	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	K 1	K 2	K 3	A 1	A2	G*	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3									
295007 High Reactor Pressure / 3									
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2		X					AK2.04; Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: Reactor water cleanup.	2.6	
295010 High Drywell Pressure / 5									
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1						X	G2.4.31; Knowledge of annunciator alarms, indications, or response procedures.	4.1	
295017 High Off-site Release Rate / 9					X		AA2.03; Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: †Radiation levels: Plant-Specific.	3.1	
295020 Inadvertent Cont. Isolation / 5 & 7				X			AA1.02; Ability to operate and/or monitor the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Drywell ventilation/cooling system.	3.2	
295022 Loss of CRD Pumps / 1					X		AA2.02; Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : CRD system status	3.4	
295029 High Suppression Pool Wtr Lvl / 5				X			EA1.01; Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: HPCI: Plant-Specific.	3.4	
295032 High Secondary Containment Area Temperature / 5			X				EK3.01; Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Emergency/normal depressurization.	3.5	
295033 High Secondary Containment Area Radiation Levels / 9									
295034 Secondary Containment Ventilation High Radiation / 9						X	G2.4.8; Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	3.8	
295035 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: Plant-Specific.	3.9	
295036 Secondary Containment High Sump/Area Water Level / 5	X						EK1.02; Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Electrical ground/ circuit malfunction.	2.6	
500000 High CTMT Hydrogen Conc. / 5									
K/A Category Point Totals:	1	1	1	2	1/2	1/1	Group Point Total:		7/3

ES-401	BWR Examination Outline Plant Systems - Tier 2/Group 1 (RO / SRO)												Form ES-401-1	
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode										X		A4.04; Ability to manually operate and/or monitor in the control room: Heat exchanger cooling flow.	3.6	
205000 Shutdown Cooling					X							K5.03; Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Heat removal mechanisms.	2.8	
206000 HPCI	X											K1.10; Knowledge of the physical connections and/or cause/effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following: Condensate storage and transfer system.	3.4	
207000 Isolation (Emergency) Condenser														
209001 LPCS			X								X	K3.03; Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following: Emergency generators. G2.4.35; Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	2.9 4.0	
209002 HPCS														
211000 SLC	X											K1.01; Knowledge of the physical connections and/or cause/effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Core spray line break detection: Plant-Specific.	3.0	
212000 RPS		X									X	K2.01; Knowledge of electrical power supplies to the following: RPS motor-generator sets. 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.	3.2 4.4	

215003 IRM								X				A2.06; Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty range switch.	3.0	
						X						K6.04; Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : Detectors.	4.0	
215004 Source Range Monitor					X							K5.03; Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM: Changing detector position.	2.8	
215005 APRM / LPRM		X										K2.02; Knowledge of electrical power supplies to the following: APRM channels.	2.6	
					X							K5.04; Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: LPRM detector location and core symmetry.	2.9	
217000 RCIC							X					A1.01; Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: RCIC flow.	3.7	
218000 ADS	X											K1.03; Knowledge of the physical connections and/or cause/effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: Nuclear boiler instrument system.	3.7	
			X									K3.01; Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Restoration of reactor water level after a break that does not depressurize the reactor when required.	4.4	
223002 PCIS/Nuclear Steam Supply Shutoff							X					A1.01; Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: System indicating lights and alarms.	3.5	

239002 SRVs				X				X				K4.03; Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Prevents siphoning of water into SRV discharge piping and limits loads on subsequent actuation of SRV's. A2.01; Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open vacuum breakers.	3.1 3.3	
259002 Reactor Water Level Control									X			A3.08; Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: FWCI system initiation: FWCI.	4.0	
261000 SGTS								X		X		A4.02; Ability to manually operate and/or monitor in the control room: Suction valves. A2.09; Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Plant air system failure.	3.1 2.6	
262001 AC Electrical Distribution										X	X	G2.2.40; Ability to apply Technical Specifications for a system. A2.01; Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine/generator trip.	3.4 3.6	
262002 UPS (AC/DC)									X			A3.01; Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: Transfer from preferred to alternate source.	2.8	
263000 DC Electrical Distribution											X	G2.2.37; Ability to determine operability and/or availability of safety related equipment.	3.6	
264000 EDGs						X						K6.03; Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET) : Lube oil pumps.	3.5	

300000 Instrument Air			X					X				A2.01; 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. K3.01; Knowledge of the effect that a loss or malfunction of the Containment air system.	2.9 2.7	
400000 Component Cooling Water				X				X				A2.01; Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Loss of CCW pump. K4.01; Knowledge of CCWS design feature(s) and or interlocks which provide for the following: Automatic start of standby pump.	3.3 3.4	
K/A Category Point Totals:	3	2	3	2	3	2	2	3/3	2	2	2/2	Group Point Total:		26/5

ES-401 BWR Examination Outline Form ES-401-1

Plant Systems - Tier 2/Group 2 (RO / SRO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
201001 CRD Hydraulic						X						K6.02; Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System: Condensate storage tanks.	3.0	
201002 RMCS														
201003 Control Rod and Drive Mechanism											X	G2.4.49; Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.6	
201004 RSCS														
201005 RCIS														
201006 RWM														
202001 Recirculation														
202002 Recirculation Flow Control						X						K6.03; Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM: Recirculation system.	2.8	
204000 RWCU														
214000 RPIS														
215001 Traversing In-Core Probe											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.	4.4	
215002 RBM	X											K1.02; Knowledge of the physical connections and/or cause/effect relationships between ROD BLOCK MONITOR SYSTEM and the following: LPRM.	3.2	
216000 Nuclear Boiler Inst.								X				A2.11; Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Heatup or cooldown of the reactor vessel.	3.2	
219000 RHR/LPCI: Torus/Pool Cooling Mode								X				A2.12; Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve logic failure: Plant-Specific	3.1	
223001 Primary CTMT and Aux.			X									K3.09; Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES will have on following: Nuclear boiler instrumentation.	2.8	

226001 RHR/LPCI: CTMT Spray Mode		X										K2.02; Knowledge of the physical connections and/or cause/effect relationships between RHR/LPCI: CONTAINMENTSPRAY SYSTEM MODE and the following: Pumps.	2.9	
230000 RHR/LPCI: Torus/Pool Spray Mode														
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment									X			A3.01; Ability to monitor automatic operations of the FUEL HANDLING EQUIPMENT including: Crane/refuel bridge movement: Plant-Specific.	2.6	
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator										X		A4.14; Ability to manually operate and/or monitor in the control room: Turbine trip.	3.8	
245000 Main Turbine Gen. / Aux.				X								K4.07; Knowledge of MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS design feature(s) and/or interlocks which provide for the following: Generator voltage regulation.	2.5	
256000 Reactor Condensate														
259001 Reactor Feedwater							X					A1.04; Ability to predict and/or monitor changes in parameters associated with operating the REACTOR FEEDWATER SYSTEM controls including: RFP turbine speed: Turbine-Driven-Only.	2.8	
268000 Radwaste														
271000 Offgas								X				A2.10; Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Offgas system high flow.	3.3	
272000 Radiation Monitoring					X							K5.01; Knowledge of the operational implications of the following concepts as they apply to RADIATION MONITORING SYSTEM: Hydrogen injection operation's effect on process radiation indications: Plant-Specific.	3.2	
286000 Fire Protection														
288000 Plant Ventilation														
290001 Secondary CTMT														
290003 Control Room HVAC														
290002 Reactor Vessel Internals														
K/A Category Point Totals:	1	1	1	1	1	2	1	1/2	1	1	1/1	Group Point Total:		12/3

Facility: **Brunswick**		Date of Exam: **December, 2016**				
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements.	3.8			
	2.1.32	Ability to explain and apply system limits and precautions.	3.8			
	2.1.36	Knowledge of procedures and limitations involved in core alterations.	3.0			
	2.1.5	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.			3.9	
	2.1.43	Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.			4.3	
	Subtotal					
2. Equipment Control	2.2.2	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.	4.6			
	2.2.4	(multi-unit license) Ability to explain the variations in control | board/control room layouts, systems, instrumentation, and procedural actions between units at a facility.	3.6			
	2.2.44	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2			
	2.2.15	Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc. |			4.3	
	2.2.22	Knowledge of limiting conditions for operations and safety limits.			4.7	
	Subtotal					
3. Radiation Control	2.3.12	Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	3.2			
	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9			
	2.3.11	Ability to control radiation releases.			4.3	
	Subtotal					
4. Emergency Procedures / Plan	2.4.20	Knowledge of the operational implications of EOP warnings, cautions, and notes.	3.8			
	2.4.27	Knowledge of “fire in the plant” procedures.	3.4			
	2.4.30	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.			4.1	
	2.4.35	Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.			4.0	
	Subtotal					
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	295024EA1.14	Too similar to 295020AA1.02. Reselected 295024EA1.05 as a replacement.
2/1	259002A3.05	Not representative of how plant operates. Reselected 259002A3.08 as a replacement.
2/1	262002K6.03	Too similar to 26002A3,01. Reselected 215003K6.04
1/1	295028G2.19	Cannot write at SRO level. Replaced by 295023G2.2.25.
1/1	700000AA2.08	Cannot write at SRO level. Replaced by 295021AA2.03.
1/1	295024G2.2.37	Cannot write at SRO level. Replaced by 295037G2.4.21.
2/1	217000A2.09	Cannot write operationally valid question. Replaced with 262001A2.01.
2/2	256000A2.17	Not in plant design. Reselected 219000A2.12.
2/2	268000A2.01	Unable to write a operationally valid and discriminating question. Reselected 216000 A2.11
2/2	2860000K2.03	Unable to write a discerning question. Reselected to 2260001K2.02.
1/2	295013.A2.02	Unable to write to the K/A. The plant doesn't really monitor stratification. Reselected with 295022A.2.02.

Facility: Brunswick	Date of Examination: 11/28/2016
Examination Level: RO SRO	Operating Test Number: Draft

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations #1 (RO, then SRO)	R, N	2.1.25 Perform SJAE Off-Gas Radiation Monitors Channel Check Calculation
Conduct of Operations #2 (All)	R, D	2.1.07 Determine Primary Containment Water Level and Evaluate PCPL-A.
Equipment Control (RO, then SRO)	R, N	2.2.12 Calculate Drywell Leakage Rate.
Radiation Control (All)	R, D	2.3.07 Determine Stay Time Limitations in High Radiation Areas.
Emergency Plan (SRO Only)	R, M	2.4.29 Classify An Emergency per PEP-02.1.

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

* Type Codes & Criteria:

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

Conduct of Operations #1
Perform SJAE Off-Gas Radiation Monitors Channel Check Calculation
RO, then SRO

2.01.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.
3.9/4.2

This is a new JPM developed for the 2016 NRC Initial Exam. The examinee will perform item 108, SJAE Off-Gas Radiation Monitors Channel Check, of 2OI-03.2, Reactor Operator Daily Surveillance Report, and state the status of the channel check. Then the SRO examinees will determine any required actions.

Conduct of Operations #2
Determine Primary Containment Water Level and Evaluate PCPL-A
All

2.01.07 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
4.4/4.7

This is a bank JPM. The examinee will determine Primary Containment water level per EOP-01-UG, Attachment 11. Determine the current region of operation (Safe/Unsafe) on Primary Containment Pressure Limit A (PCPL-A).

Equipment Control
Calculate Drywell Leakage Rate
RO, then SRO

2.2.12 Knowledge of surveillance procedures
3.7/4.1

This is a new JPM developed for the 2016 NRC Initial Exam. The examinee will determine the 24 hour leak rate for the equipment and floor drains, and the 24 hour total leak rate to the drywell IAW Attachment 1, Drywell Leakage Calculation, of 2OI-03.2, Reactor Operator Daily Surveillance Report, for Sunday Nightshift at time 2000.. Then the SRO examinees will determine if TS LCO 3.4.4 is met and if it is not met identify the latest time the Unit is required to be in MODE 3.

Radiation Control
Determine Stay Time Limitations in High Radiation Areas.
All

2.3.07 Ability to comply with radiation work permit requirements during normal and abnormal conditions.
3.5/3.6

This is a bank JPM. The examinee will determine the total dose accumulated for both workers and determine if any Brunswick administrative dose limitations will be exceeded.

Emergency Plan
Classify An Emergency per PEP-02.1
SRO Only

2.4.29 Knowledge of the Emergency Plan
3.1/4.4

This is a modified JPM that was used on the 2012 NRC Initial Exam. Changed EAL from Site Area Emergency to an Alert. The SRO examinees will determine the highest required classification and its EAL Identifier. This JPM is time critical (15 minutes).

Facility: Brunswick	Date of Examination: 11/28/2016
Exam Level: RO SRO-I ***SRO-U***	Operating Test No.: Draft

Control Room Systems:* 8 for RO; 7 for SRO-I; 2 or 3 for SRO-U

System / JPM Title	Type Code*	Safety Function
a. ***Reset Recirc Pump Runback, Both Recirc Pumps trip***	S, N, A	1
b. ***Mechanical Trip Valve Oil Trip Test***	S, N	4
c. ***RCIC Start w/ failure to isolate***	S, P, A, EN	2
d. Suppression Pool Cooing - Service Water Release	S, A, D, L	5
e. Vent Drywell w/ Stack Rad Mon >50% increase	S, A	9
f. Shifting Caswell Beach Lube Water Pumps From The RTGB	S, N	8
g. Substitute Control Rod Position	S, L	7
h. RO ONLY - Test the Main Steam Isolation Valves	S, P	3
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. LEP-01, Heater Drain Pumps	R, D, E, L	2
j. ***LEP-05, SRV operation from RSDP***	N, R, E, L	7
k. ***Rack in E6 Crosstie Breaker***	A, D, E	6

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

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

a. **Reset Recirc Pump Runback with both Recirc Pumps Tripping**

202002 A2.01	Ability to predict the impacts of recirculation pump trip and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations. 3.4/3.4
RO/ISRO/USRO	This is a simulator alternate path JPM that will have the examinees resetting a Recirc Pump runback signal. When the runback is reset both recirc pumps will trip, this will require an immediate operator action to insert a manual reactor scram. This JPM is a new alternate path JPM.

b. **Mechanical Trip Valve Oil Trip Test**

245000 A3.01	Ability to manually operate and/or monitor in the control room: Turbine Trip. 3.6/3.6
RO/ISRO/USRO	This is a new simulator JPM that will require the examinee to perform the Mechanical Trip Valve Oil Trip Test.

c. **RCIC Start Per The Hard Card – Steam line break**

217000 A4.08	Ability to manually operate and/or monitor RCIC system flow. 3.7/3.6
RO/ISRO/USRO	This is a simulator alternate path JPM that will require the examinee to start RCIC for injection per the Hard Card and restore RPV water level. As an alternate path the steam line breaks and RCIC does not auto isolate requiring manual isolation of RCIC. RCIC is an engineered safety feature. This JPM was randomly selected from the previous exam (2015).

d. **Suppression Pool Cooing - Service Water Release**

219000 A4.01	Ability to manually operate and/or monitor in the control room: Pumps. 3.8/3.7
RO/ISRO	This is a banked alternate path simulator JPM that will require the examinee to place the B Loop of RHR in SPC. While placing in service the heat exchanger tubes will rupture causing a release in the service water system. The examinee will have to isolate the release path.

e. **Vent Drywell w/ Stack Rad Mon >50% increase**

261000 A4.04	Ability to manually operate and/or monitor Primary Containment Pressure. 3.3/3.4
RO/ISRO	This is a banked simulator JPM that will require the examinee to vent the Drywell via Standby Gas Treatment. This JPM is alternate path in that Main Stack Rad will rise requiring the examinee to isolate the system.

f. **Shifting Caswell Beach Lube Water Pumps From The RTGB**

400000 A4.01	Ability to manually operate and/or monitor in the control room: CCW indications and control. 3.1/3.0
RO/ISRO	This is a new simulator JPM that will require the examinee to perform shift Caswell Beach Lube Water Pumps From The RTGB.

g. **Substitute Control Rod Position**

201006 A4.06 — Ability to manually operate and/or monitor in the control room: Selected rod position indication. 3.2/3.2

RO/ISRO — This is a banked JPM that will require the examinee to substitute in the Rod Worth Minimizer the indicated rod position.

h. **Test the Main Steam Isolation Valves**

239001 A4.01 — Ability to manually operate and/or monitor the MSIVs in the Control Room. 4.2/4.0

RO — This is a new JPM that will require the examinee to perform post-maintenance testing of a MSIV. This JPM was randomly selected from the previous exam (2015).

i. **LEP-01, Heater Drain Pumps**

295031 EA1.08 — Ability to operate alternate injection system systems as they apply to Reactor Water Level Low. 3.8/3.9

RO/ISRO — This is a banked in-plant JPM that will require the examinee to simulate the Auxiliary Operator actions for Alternate Coolant Injection, Heater Drain Pump Injection per 0EOP-01-LEP-01. This JPM is performed in the RCA.

j. **LEP-05, SRV operation from RSDP**

295016 AA1.08 — Ability to operate and/or monitor Reactor Pressure as it applies to Control Room Abandonment. 4.0/4.0

RO/ISRO/USRO — This is a new in-plant JPM that will require the examinee to simulate the actions associated with performing the field actions for pressure control from the RSDP (Remote Shutdown Panel)

k. **Rack in E6 Crosstie Breaker**

295003 AA1.01 — Ability to operate and/or monitor AC Electrical Distribution System as it applies to a partial or complete loss of A.C. power. 3.7/3.8

RO/ISRO/USRO — This is a banked in-plant alternate path JPM that will require the examinee to simulate manually racking in the crosstie breaker. The charging springs on the breaker will not automatically re-charge and will have to be manually charged.

1. 201001 1

Unit One is in an outage with the condensate system under clearance.
An earthquake results in damage to the CST causing level to slowly lower.

Which one of the following completes the statement below with regards to the effect on the CRD system?

The CRD system will __(1)__ when the CST level reaches approximately __(2)__.

A (1) trip
(2) 3 feet

B (1) trip
(2) 11 feet

C (1) transfer to the backup supply
(2) 3 feet

D (1) transfer to the backup supply
(2) 11 feet

Answer: B

K/A:
201001 Control Rod Drive Hydraulic System
K6 Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System: (CFR: 41.7 / 45.7)
02 Condensate storage tanks

RO/SRO Rating: 3.0/3.1

Tier 2 / Group 2

K/A Match: This meets the K/A because the student has determine the effect of the loss of the CST on the CRD system.

Pedigree: New

Objective: LOI-CLS-LP-008, Obj. 8
Given plant conditions, predict the effect that a loss or malfunction of the following will have on the CRDH System: b. Condensate Storage Tank

Reference: None

Cog Level: High

Explanation: Under normal system operations the CRD system suction is from the condensate system. The alternate supply is from the CST, which will transfer automatically. With the condensate system under clearance these valves would be isolated. The standpipe for the CRD suction is at ~11 feet. The auto transfer for the suction for ECCS is at ~3 feet.

Distractor Analysis:

Choice A: Plausible because the pumps will lose NPSH and trip but the suction is at 11 feet not 3 feet.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because an auto transfer to the CST would occur but in this case an auto transfer to the condensate system is not possible. 3 feet is the suction height for the ECCS system.

Choice D: Plausible because an auto transfer to the CST would occur but in this case an auto transfer to the condensate system is not possible. The second part is correct.

SRO Basis: N/A

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3.0 PRECAUTIONS AND LIMITATIONS

1. This procedure is Reactivity Management related per AD-OP-ALL-0203, Reactivity Management. Those portions of this procedure that move control rods in MODES 1 or 2 are considered a Direct Reactivity manipulation and Reactivity Evolution Category R2 (Reactivity Manipulation, R2). ☐

2. CST level is maintained greater than 11 feet to prevent CRD pumps from losing suction. ☐

2. 201003 1

Unit Two is operating at rated power when a control rod begins to drift out from position 24.

Which one of the following identifies the **first** action to be taken by the operator at the controls (OATC)?

A Initiate a single rod scram.

B Initiate a manual reactor scram.

C Select and fully insert the control rod to position 00.

D Select and attempt to arrest the control rod at position 24.

Answer: D

K/A:
201003 Control Rod and Drive Mechanism
G2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6)

RO/SRO Rating: 4.6/4.4

Tier 2 / Group 2

K/A Match: This meets the K/A because the question is testing the operator action required to control a drifting control rod
(Chief Examiner agreed that operation of the RMCS for a rod drift would meet this K/A)

Pedigree: New

Objective: LOI-CLS-LP-07. Obj. 11b
Describe the possible cause(s) and required operator actions for the following alarms:
A-5 3-2. Control Rod Drift

Reference: None

Cog Level: Fundamental

Explanation: This abnormal positive reactivity addition requires response from the APP before entering the AOP. The APP requires that the operator attempt to arrest the drift at the intended position first, if it cannot be arrested but responds to RMCS to insert to 00, if it does not respond to RMCS to perform a single rod scram. If more than 1 rod drifts then a manual scram is required.

Distractor Analysis:

Choice A: Plausible because if the rod does not move then this is the appropriate action.

Choice B: Plausible because if more than 1 rod is drifting then this would be correct

Choice C: Plausible because this is the correct action if the rod is drifting in or the rod continues to drift after attempting to arrest.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

ROD DRIFT

AUTO ACTIONS

1. RWM withdraw or insert errors possibly causing rod block if reactor power is below the low power setpoint.
2. The reactor power will respond to the drifting rod depending upon the direction of the rod drift and rod worth, and could result in a reactor Scram if the plant is at low power operation.

CAUSE

1. Rod in uneven position due to:
   a. Leaking Scram valve.
   b. High cooling water pressure.
   c. Failure of directional control valves.
   d. Slow to settle due to fuel bundle channel bow.
2. Malfunction in alarm circuit.

OBSERVATIONS

1. "Rod Drift" indication on the full core display.
2. RWM error indications and Rod Block if reactor power is below the low power setpoint.
3. A change in neutron monitoring system meter readings as a result of the drifting rod with possible high flux alarms.
4. If drifting rod is selected, the four-rod group display will indicate an odd control rod position, a blank window, or a changing control rod position in the direction of the drift.
5. High control rod cooling water pressure and/or flow.
6. If control rod drifts to the full in position, a green backlight on the full core display will illuminate with no position readout on RTGB.
7. ROD OUT BLOCK alarm A-05 (2-2) and no withdraw permissive light.
8. Greater than normal settle times causing an odd or no-position to be present when the RMCS timer times out.

ACTIONS

1. Determine if the affected control rod(s) is drifting or if the rod(s) has scrammed using full core display, RPIS, and RWM.
2. Select the drifting rod and determine direction of drift.
   a. Attempt to arrest the drift and latch rod by performing the following:
      1) Apply appropriate insert or withdrawal signals to the rod using RMCS.
      2) If RWM is causing rod blocks, then bypass RWM if directed by Unit CRS.

<u>ACTIONS</u> (Continued)

3. If the rod continues to <u>DRIFT OUT</u>, then perform the following:

CAUTION

A control rod collett piston stuck in the withdraw (unlatched) position will allow the rod to drift full out due to its own weight when insert pressure is removed either by the RMCS or by closing Valve C12-101.

a. Notify Reactor Engineer.
b. Monitor core parameters, main steam line radiation monitors, and off-gas activity.
c. If rod responds to an RMCS insert signal, then fully insert the rod to position 00.
d. If rod fails to latch at position 00, then reapply insert signal to drive the rod full in.
e. If rod fails to respond to RMCS, then initiate a single control rod scram.
f. Refer to 0AOP-03.0.
g. Refer to Technical Specifications 3.1.3.

4. If rod continues to <u>DRIFT IN</u>, then perform the following:
a. Apply an RMCS insert signal and fully insert rod to position 00.

3.0 AUTOMATIC ACTIONS

1. Possible rod block or select block from a failed reed switch or a loss of power

2. CRD pumps trip after a 3 second delay on low suction pressure

4.0 OPERATOR ACTIONS

NOTE

The following should be considered for establishment as critical parameters during performance of this procedure: ☐

- Reactor power
- Control rod position
- Thermal limits

4.1 Immediate Actions

1. **Stop** any power changes in progress. ☐

NOTE

Detected control rod motion without a withdraw or insert command will cause annunciator A-05 3-2, Rod Drift, to alarm. **IF** the annunciator alarms **AND NO** blue scram light(s) are lit on the full core display, the conservative assumption is that rod(s) are drifting. ☐

2. **IF** more than one control rod is drifting,
THEN insert a manual scram **AND enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure. ☐

3. 202002 1
Unit One is at rated power.

Which one of the following identifies the impact of inadvertently closing the 1A Reactor Recirculation Pump 1-B32-F031A, Pump A Disch Vlv?

The 1A Reactor Recirculation pump speed will lower to approximately:

A 20%

B 34%

C 45.4%

D 48%

Answer: B
K/A:
202002 Recirculation Flow Control System
K6 Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM: (CFR: 41.7 / 45.7)
03 Recirculation system

RO/SRO Rating: 2.8/2.8

Tier 2 / Group 2

K/A Match: This meets the K/A because the student has to determine the effect of closing the discharge valve (which causes a loss of recirc) will have on the recirc flow control system.

Pedigree: new

Objective: LOI-CLS-LP-002.1, Obj. 17
Explain the operation of the following VFD limiters and controls: a. Limiter #1 b. Limiter #2

Reference: None

Cog Level: Fundamental

Explanation: Closing of the discharge valve will cause the pump to runback to limiter #1 (34%).

Distractor Analysis:

Choice A: Plausible because this is the minimum speed setting.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is limiter #2 setting for Unit One

Choice D: Plausible because this is limiter #2 setting for Unit Two

SRO Basis: N/A

4. **RFCS VFD Runback #1 Logic (Figure 02-18D and 02-18E)**

The logic for VFD A Runback #1 is shown on Figure 02-18D; the logic for VFD B Runback #1 shown on Figure 02-18E is functionally identical to that for VFD A. The initiating conditions for Runback #1 are:

- Recirculation Pump A Discharge Valve B32-F031A Limit Switch LS-2 opens (equivalent to Discharge Valve Not Full Open);
- Total Feedwater Flow as sensed by DFCS is less than 16.4% for 15 seconds or more.

Unit 1 Specific VFD Parameters

Parameter	Value	Function
1170	98.9% (1661.5 rpm)	VFD Over Speed Trip (Over Speed Alarm at 93.95% or 1578.4 rpm)
2080	92.5% (1554.0 rpm)	Maximum running motor speed (based upon achieving 104.5% Core Flow)
2120	45.4% (762.7 rpm)	Runback #2 Active Maximum Motor Speed
4250	50.8% (853.0 rpm)	Manual Runback Motor Speed Low Limit

Unit 2 Specific VFD Parameters

Parameter	Value	Function
1170	103.7% (1742.2 rpm)	VFD Over Speed Trip (Over Speed Alarm at 98.5% or 1655.1 rpm)
2080	97.9% (1644.7 rpm)	Maximum running motor speed (based upon achieving 104.5% Core Flow)
2120	48% (806.4 rpm)	Runback #2 Active Maximum Motor Speed
4250	53.6% (900.5 rpm)	Manual Runback Motor Speed Low Limit

VFD Parameters Common to Both Units

Parameter	Value	Function
2090	20% (336 rpm)	Minimum Running Motor Speed
2100	34% (571.2 rpm)	Runback #1 Active Maximum Motor Speed

4. 203000 1

A line break has occurred in the Unit Two drywell with the following sequence of events:

1200	Drywell pressure rises above 1.7 psig
1202	RPV pressure drops below 410 psig
1203	RPV level drops to LL3

Which one of the following completes the statement below?

The **earliest** time that the operator can throttle the 2-E11-F048A, Loop 2A RHR Heat Exchanger Bypass Valve is at:

A 1205.

B 1206.

C 1207.

D 1208.

Answer: A

K/A:

203000 RHR / LPCI: Injection Mode

A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

04 Heat exchanger cooling flow

RO/SRO Rating: 3.6/3.6

Tier 2 / Group 1

K/A Match: This meets the K/A because the student has to determine when the HX cooling flow can be operated.

Pedigree: Bank

Objective: LOI-CLS-LP-017, Obj. 09
Given an RHR pump or valve, list the interlocks, permissives and/or automatic actions associated with the RHR pump or valve, including setpoints.

Reference: None

Cog Level: High

Explanation: The heat exchanger bypass valve has a 3 minute timer that starts on a LOCA signal. Drywell pressure greater than 1.7# and reactor pressure is less than 410# is the first LOCA signal. The injection valve has a 5 minute interlock initiated by the same conditions. Another LOCA signal is introduced when reactor water level less than LL3 which provides the plausibility of the distractors.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because this is 3 minutes from the LL3 LOCA signal.

Choice C: Plausible because this is 5 minutes from the low pressure setpoint which is time limit interlock for the injection valve.

Choice D: Plausible because this is 5 minutes from the LL3 LOCA signal.

SRO Basis: N/A

After an initiation signal is received the following actions will occur:

- all four RHR pumps will start 10 seconds after power is available to the E-buses.
- Recirculation pumps are tripped via LL#2
- All valves not needed for LPCI injection automatically isolate and are interlocked shut as previously described.
- Heat exchanger bypass valve F048A/B opens and cannot be throttled for 3 minutes after an initiation signal is received. This ensures a discharge path for the RHR pumps.
- Permissives sent to ADS as RHR pump pressure is sensed > 100 psig. Both pumps in either loop are required to satisfy the ADS permissive, or one core spray loop.
- Minimum flow valve opens if injection flow in loop is < 1000 gpm decreasing after a 10 sec time delay. It automatically shuts as injection valves open and injection flow raises to > 3000 gpm increasing.
- Reactor pressure decreases through the break and/or with actuation of ADS.
- As reactor pressure decreases to 410 psig, the LPCI injection valves F015A(B) auto open. The outboard injection valve F017A(B) can be throttled 5 minutes after the RPV pressure is below 410 psig.
- As pressure reaches 310 psig, recirculation pump discharge and discharge bypass valves shut and are interlocked shut in the attempt to re-flood the core.
- As pressure reaches 200 psig, the RHR system injects into both recirculation system loops by lifting the check valves and overcoming reactor pressure.

5. 205000 1
RHR Loop 2A is operating in the Shutdown Cooling mode of operation with the following parameters:

RHRSW Pump 2A	Operating
RHRSW Flow	4000 gpm
RHR Pump 2A	Operating
RHR Loop A Flow	6000 gpm

Which one of the following completes the statement below?

The required operator action to **lower** the cooldown rate IAW 2OP-17, *Residual Heat Removal System Operating Procedure*, is to throttle **open**:

A 2-E11-F003A, HX 2A Outlet Vlv.

B 2-E11-F017A, Outboard Injection Vlv.

C 2-E11-F048A, HX 2A Bypass Vlv.

D 2-E11-PDV-F068A, HX 2A SW Disch Vlv.

Answer: C
K/A:
205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)
K5 Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.5 / 45.3)
03 Heat removal mechanisms

RO/SRO Rating: 2.8/3.1

Tier 2 / Group 1

K/A Match: This meets the K/A because the student has to know which valve would need to be operated to control the heat removal for SDC.

Pedigree: New

Objective: LOI-CLS-LP-017, Obj. 15
Describe how the reactor cool down rate is controlled when the RHR system is in the Shutdown Cooling mode

Reference: None

Cog Level: High

Explanation: The procedure allows throttling closed the F003 or F068 or throttling open the F048. Throttling open the F048 will bypass some of the RHR flow around the heat exchanger thereby lowering cooldown rate.

Distractor Analysis:

Choice A: Plausible because if the valve was throttled closed this would be correct,

Choice B: Plausible because if the valve was throttled closed this would be correct, although it is not an option that is allowed in the procedure.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because if the valve was throttled closed this would be correct.

SRO Basis: N/A

4. **IF** a lower cooldown rate is desired, **THEN PERFORM** the following, as necessary, for each operating RHR loop while maintaining desired flow rate, **NOT** to exceed 10,000 gpm per loop:

> **CAUTION**
>
> **IF** *E11-F003A(B)* is closed, **THEN** *RHR HEAT EXCHANGER 2A(2B)* inlet temperature, located on *E41-TR-R605*, Point *1(2)*, is **NOT** a valid indication of reactor coolant temperature.

a. **SLOWLY THROTTLE CLOSE** *HX 2A(2B) OUTLET VLV, E11-F003A(B)*, as necessary. ☐

b. **THROTTLE CLOSED** *HX 2A(2B) SW DISCH VLV, E11-PDV-F068A(B)*, as necessary, to reduce RHRSW flow rate. ☐

c. **SLOWLY THROTTLE OPEN** *HX 2A(2B) BYPASS VLV, E11-F048A(B)*, as necessary, maintaining RHR flowrate greater than 4500 gpm. ☐

6. 206000 1

A Group 1 isolation has occurred on Unit One.

HPCI has been placed in the pressure control mode of operation IAW 1OP-19, *High Pressure Coolant Injection System Operating Procedure.*

HPCI flow controller, E41-FIC-R600, is in manual with the output at midscale.

Which one of the following completes the statement below?

If the 1-E41-F008, Bypass To CST Valve, is throttled ___(1)___ too far, this may result in HPCI ___(2)___.

A (1) open
  (2) tripping on overspeed

B (1) open
  (2) operation below 2100 rpm

C (1) closed
  (2) tripping on overspeed

D (1) closed
  (2) operation below 2100 rpm

Answer: B

K/A:

206000 High Pressure Coolant Injection System

K1 Knowledge of the physical connections and/or cause-effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following:

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

10 Condensate storage and transfer system

RO/SRO Rating: 2.8/3.1

Tier 2 / Group 1

K/A Match: This meets the K/A because this is testing the cause-effect relationship between HPCI and the flowpath to the CST.

Pedigree: Bank

Objective: LOI-CLS-LP-019. Obj. 8
Describe the methods available for controlling RPV pressure and/or RPV cooldown when operating the HPCI System in the Pressure Control mode. (LOCT)

Reference: None

Cog Level: high

Explanation: Opening F008 will increase flow, causing turbine speed control to lower turbine speed to maintain desired flow. Opening valve too far can result in RPM below 2100 (OP-19, Section 8.2). Closing F008 will cause turbine speed to increase, but the governor limits turbine speed to a maximum value (4100 RPM) below the overspeed trip.

Distractor Analysis:

Choice A: Plausible because opened is correct and an overspeed condition may be thought correct if the flowpath is considered incorrectly.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because throttling closed will increase the speed of the turbine.

Choice D: Plausible because may be thought correct if the flowpath is considered incorrectly.

SRO Basis: N/A

From OP-19:

CAUTION

Throttling *E41-F008* open may cause turbine speed reduction to less than 2100 rpm, if opened too far.

From the SD:

Operation of the HPCI Turbine below the minimum rated speed of 2100 rpm may result in a failure of the auxiliary oil pump from repeated startup cycles. A loss of the auxiliary oil pump will prevent starting of the HPCI Turbine.

7. 209001 1
Unit Two is operating at rated power.
Due to a circuit malfunction an inadvertent LOCA initiation occurs in the Div II Core Spray logic causing A-03 (2-6), *CORE SPRAY SYSTEM II ACTUATED*, to alarm.

Which one of the following completes both statements below?

Core Spray Pump(s) ___(1)___ will start.

___(2)___ will start.

A (1) 2B ONLY
(2) All DGs

B (1) 2B ONLY
(2) DG2 and DG4 ONLY

C (1) 2A and 2B
(2) All DGs

D (1) 2A and 2B
(2) DG2 and DG4 ONLY

Answer: A
K/A:
209001 Low Pressure Core Spray System
K3 Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following: (CFR: 41.7 / 45.4)
03 Emergency generators

RO/SRO Rating: 2.9/3.0

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the knowledge of a malfunction of the CS logic has on the EDG.

Pedigree: New

Objective: LOI-CLS-LP-018, Obj. 14
List three systems, other than the Core Spray System, which are initiated or isolated by the Core Spray System logic.

Reference: None

Cog Level: High

Explanation: For CS the logic will only start that divisions pump (RHR would start the other divisions pump) for the CS logic to the DGs either divisions signal will start all DGs.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct and since it is divisional for the pump starts the student may think that it would start only the Div II DGs. There are signals that would start divisional DGs.

Choice C: Plausible because the student may think the CS logic is similar to the RHR logic for pump starts and the second part is correct.

Choice D: Plausible because the student may think the CS logic is similar to the RHR logic for pump starts and since it is a Div II logic the student may think that it would start only the Div II DGs. There are signals that would start divisional DGs.

SRO Basis: N/A

CORE SPRAY SYS II ACTUATED

AUTO ACTIONS

1. If E bus was not deenergized, Core Spray Pump 2B starts 15 seconds after receipt of initiation signal
2. If E bus was deenergized, Core Spray pump 2B starts 15 seconds after diesel generator ties onto E bus
3. If open, Full Flow Test Byp Vlv, E21-F015B, closes
4. If closed, Outboard Injection Vlv, E21-F004B, opens
5. When reactor pressure drops to 410 psig, Inboard Injection Vlv, E21-F005B, opens
6. When loop flow is greater than 1500 gpm, Min Flow Bypass Vlv, E21-F031B, closes
7. Div II Non-Intrpt RNA, RNA-SV-5261, and Div I Non-Intrpt RNA, RNA-SV-5262, close
8. Div II Backup N2 Rack Isol Vlv, RNA-SV-5481, and Div I Backup N2 Rack Isol Vlv, RNA-SV-5482, open
9. Fans for Drywell Coolers B and C trip
10. All diesel generators start
11. Nuclear Service Water To Vital Header Valve, SW-V117, opens
12. RBCCW HX Service Water Inlet Valve, SW-V103, closes

CAUSE

1. Reactor low level three (45 inches)
2. High drywell pressure (1.7 psig) in conjunction with low reactor pressure (410 psig)
3. Circuit malfunction

8. 211000 1

Which one of the following completes the statement below concerning Core Spray Line Break Detection differential pressure instrument?

The ___(1)___ leg of this DP instrument senses ___(2)___ core plate pressure via the SLC/Core Differential Pressure penetration.

A (1) variable
(2) below

B (1) variable
(2) above

C (1) reference
(2) below

D (1) reference
(2) above

Answer: D

K/A:

211000 Standby Liquid Control System

K1 Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

01 Core spray line break detection

RO/SRO Rating: 3.0/3.3

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the physical connection of SBLC and CS line break detection.

Pedigree: Last used on 10-1 NRC Exam

Objective: CLS-LP-18, Obj. 10
Explain the principle of operation of the CS Line Break Detection Instrumentation

Reference: None

Cog Level: fundamental

Explanation: This system is comprised of a differential pressure detector which provides Control Room annunciation on detected high DP. The high pressure reference leg of this instrument is exposed to above core plate pressure via the SLC/Core Differential Pressure penetration. The low pressure of this instrument is normally exposed to above core pressure via the Core Spray injection line. This results in the instrument normally measuring core DP (not including core plate DP).

Distractor Analysis:

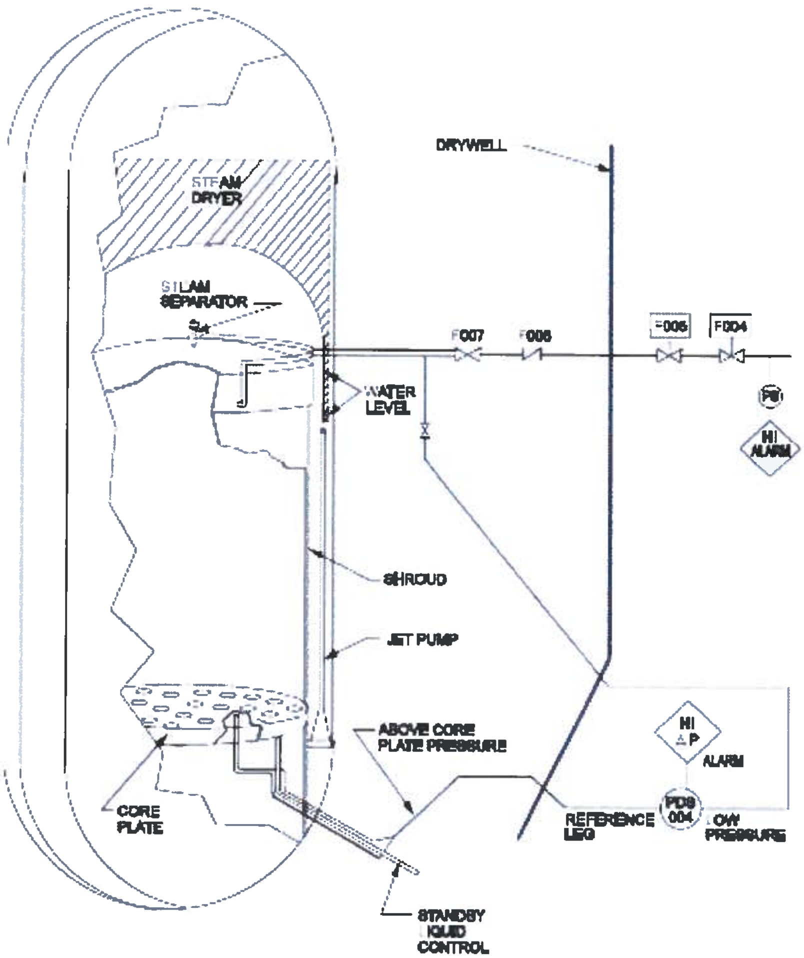
Choice A: Plausible because the examinee may confuse the reference and variable legs and SLC does discharge below the core plate

Choice B: Plausible because the examinee may confuse the reference and variable legs

Choice C: Plausible because it is the reference leg and SLC does discharge below the core plate.

Choice D: Correct Answer, see explanation

SRO Basis: N/A



This system is comprised of a differential pressure detector which provides Control Room annunciation on detected high ΔP. The high pressure reference leg of this instrument is exposed to above core plate pressure via the SLC/Core Differential Pressure penetration. The low pressure of this instrument is normally exposed to above core pressure via the Core Spray injection line. This results in the instrument normally measuring core ΔP (not including core plate ΔP).

A break in the Core Spray injection line between the reactor vessel penetration and the core shroud would expose the low pressure side of the instrument to the lower pressure of the region outside the shroud. This would be sensed as an <u>increased</u> differential pressure and Control Room annunciator would alert the Operator. Although other indications would be available, this alarm would also indicate a break in the line between the E21-F006B(A) check valve and the reactor vessel penetration.

The Core Spray pipe break detection instruments are located on the Reactor Building 20' elevation.

9. 212000 1
Which one of the following identifies the normal power supply to RPS MG Set 2B?

A 480V MCC 2CA

B 480V MCC 2CB

C 480V MCC 2XC

D 480V MCC 2XD

Answer: B
K/A:
212000 Reactor Protection System
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
01 RPS motor-generator sets

RO/SRO Rating: 3.2/3.3

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the power supply to the RPS MG Set

Pedigree: New

Objective: CLS-LP-03, Obj 18b
State the power supplies for the following: RPS MG Set B

Reference: None

Cog Level: Fundamental

Explanation: Power for the Motor Generator Sets is tapped off two phases of the normal 480 VAC MCC 1CA/1CB (2CA/2CB) power supply for the motor through a stepdown transformer (480V to 120V) from E5/E6 (E7/E8). Selectable reserve power to the Bus is provided from 120 VAC 1E5(2E7) or 1E6(2E8), and is normally selected to Division I. In the event that either RPS M-G Set fails to operate, the alternate power source must be manually selected.

Distractor Analysis:

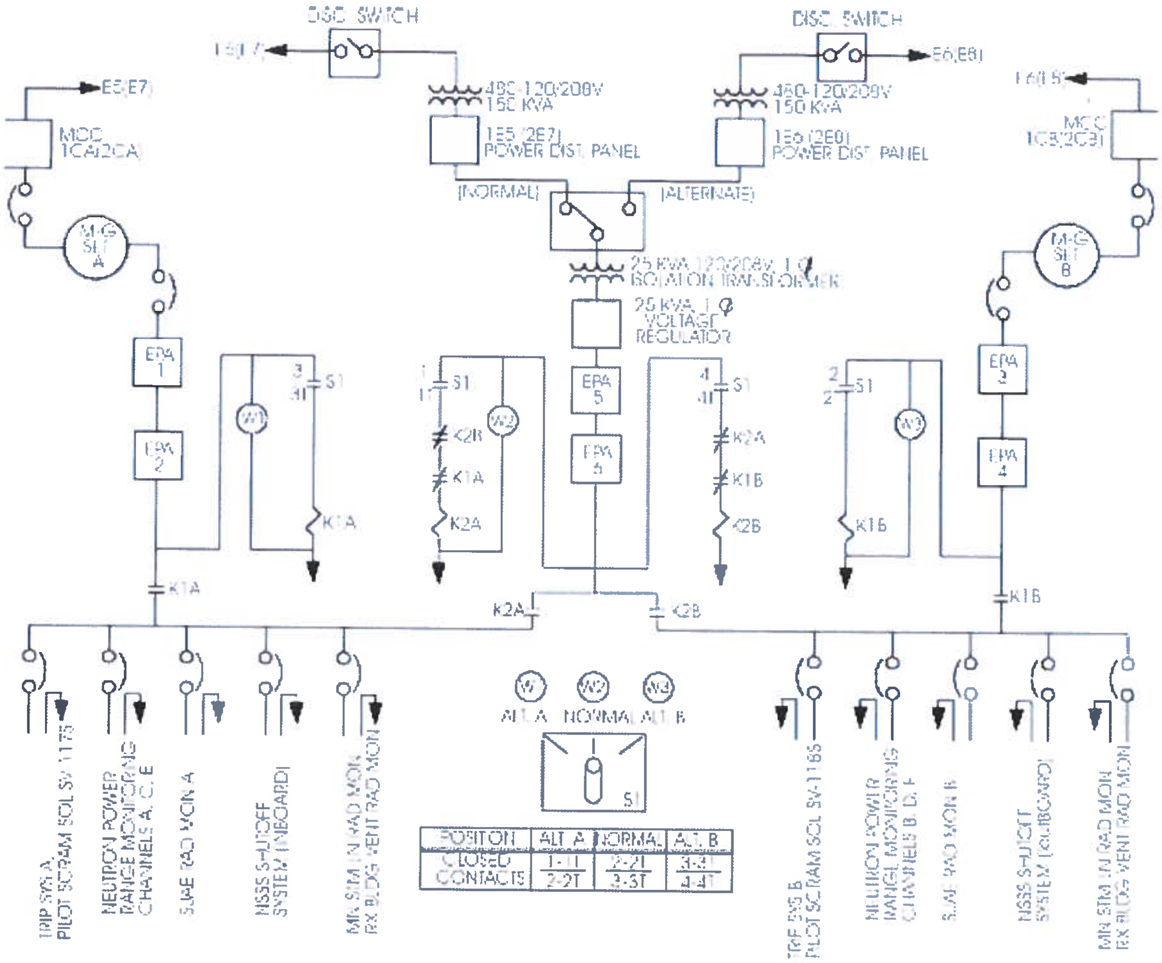
Choice A: Plausible because 2CA supplies RPS MG Set A.

Choice B: Correct Answer, see explanation

Choice C: Plausible because 2E7 is normally the alternate RPS power supply.

Choice D: Plausible because 2E8 is the alternate alternate RPS power supply.

SRO Basis: N/A



POSITION	ALT. A	NORMAL	ALT. B
CLOSED CONTACTS	1-1T 2-2T	2-2T 3-3T	3-3T 4-4T

10. 215002 1
Which one of the following identifies the LPRM detector level that provides input to the Rod Block Monitor system for indication ONLY, and is NOT used for the purpose of generating rod blocks?

A Level A

B Level B

C Level C

D Level D

Answer: A
K/A:
215002 Rod Block Monitor System
K1 Knowledge of the physical connections and/or cause-effect relationships between ROD BLOCK MONITOR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
02 LPRM

RO/SRO Rating: 3.2/3.1

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the connection between RBM and LPRMs

Pedigree: Bank

Objective: LOI-CLS-LP-09.6, Obj 5a
List the PRNMS system signals/conditions that will cause the following actions: APRM / RBM Rod Blocks

Reference: None

Cog Level: Fundamental

Explanation: The level A inputs are sent to RBM-A for processing/output to the LPRM Display Meters on the 4-Rod Display. Level A is for indication only

RBM-A Receives
all four level C inputs
lower left and upper right level B inputs
upper left and lower right level D inputs

RBM-B Receives
all four level C inputs
upper left and lower right level B inputs
lower left and upper right level D inputs

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because LPRMs have a B level that input to the RBMs Rod Blocks.

Choice C: Plausible because LPRMs have a C level that input to the RBMs Rod Blocks.

Choice D: Plausible because LPRMs have a D level that input to the RBMs Rod Blocks.

SRO Basis: N/A

The "A" level LPRM detectors are not used for RBM input processing, while both RBM channels use all "C" level detectors. This gives an accurate representation of actual power around the control rod. The "B" and "D" detectors are distributed evenly between the two RBM channels. An example of LPRM input to a both RBM channels with a four-string rod selected is two "B" level LPRMs, four "C" level LPRMs, and two "D" level LPRMs for each channel.

The RBM circuitry undergoes a nulling and filtering sequence when a rod is selected and therefore a delay of at least 2.5 seconds must be allowed between selection and rod movement. A Rod Inhibit signal is

11. 215003 1
Unit One is performing a startup with the reactor just declared critical.
While ranging IRM G from range 1, the IRM will not change ranges and remains on Range 1.

Which one of the following completes both statements below?

When IRM G indication **first** exceeds ___(1)___ on the 125 scale, annunciator A-05, 2-4, *IRM UPSCALE*, will alarm.

The action required IAW A-05, 2-4, *IRM UPSCALE*, is to ___(2)___.

A (1) 70
(2) place the joystick on P603 for the IRM G to Bypass

B (1) 70
(2) withdraw the IRM G detector to maintain reading on scale

C (1) 117
(2) place the joystick on P603 for the IRM G to Bypass

D (1) 117
(2) withdraw the IRM G detector to maintain reading on scale

Answer: A

K/A:
215003 Intermediate Range Monitor System
A2 Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)
06 Faulty range switch

RO/SRO Rating: 3.0/3.2

Tier 2 / Group 1

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because it is testing what will happen with a faulty range switch and the action required.

Pedigree: New

Objective: LOI-CLS-LP-009.1, Obj. 3a
List the SRM/IRM system signals/conditions that will cause the following actions and the conditions under which each is bypassed: Rod Blocks (LOCT)
LOI-CLS-LP-009.1, Obj. 14a
Given plant conditions and one of the following events, use plant procedures to determine the actions required to control and/or mitigate the consequences of the event: SRM/IRM Upscale alarm (LOCT)

Reference: None

Cog Level: High

Explanation: With the reactor critical the indication will continue to rise. The Upscale alarm will come in at 70 on the 0-125 scale. The Upscale Hi/Inop alarm comes in at 117 on the 0-125 scale. IAW with the APP the action to take is to bypass the IRM. In the case of the SRMs an action to take could be to withdraw the SRM to maintain on scale readings.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct and the second part could be correct if this was an SRM.

Choice C: Plausible because 117 is an alarm setpoint for the IRMs and the second part is correct.

Choice D: Plausible because 117 is an alarm setpoint for the IRMs and the second part could be correct if this was an SRM.

SRO Basis: N/A

IRM UPSCALE

AUTO ACTIONS

1. Rod withdrawal block (bypassed when Reactor Mode Switch in RUN)

CAUSE

1. IRM channel indicates greater than or equal to 70 on 0-125 scale
2. Improper ranging of IRM channels during reactor startup or shutdown
3. IRM detector failure
4. During refuel outages, IRM spiking due to noise generation from work activities in drywell, such as welding
5. Circuit malfunction

OBSERVATIONS

1. IRM channel indicating greater than or equal to 70 on 0-125 scale
2. IRM channel upscale (UPSC ALARM) amber indicating light on
3. ROD OUT BLOCK (A-05 2-2) alarms
4. Rod withdrawal permissive indicating light off

ACTIONS

1. If in progress, stop withdrawal of control rods.
2. Monitor IRM indications to determine affected channel(s).

CAUTION

IRM range switches should be repositioned carefully in order to prevent a reactor scram.

3. Reposition affected IRM range switch to next higher range.
4. If a sudden rise in indicated reactor power occurred on more than one IRM channel, verify correct rod withdrawal sequence is being used and insert in-sequence control rods as necessary to turn power rise.
5. If IRM detector failure or circuit malfunction is suspected, perform the following:
   a. Refer to Technical Specification 3.3.1.1 and TRM 3.3 for IRM channel operability requirements.
   b. Notify Unit CRS.
   c. Bypass affected channel using IRM bypass switch.
   d. Ensure a WR is prepared.

12. 215003 2

A reactor shutdown is in progress.
All IRMs on **range 1** reading between 15 and 20.
IRM B detector is failing downscale.

Which one of the following completes both statements below?

A-05 (1-4) *IRM Downscale,* alarm setpoint is ≤ ___(1)___ on the 125 scale.

When the IRM downscale alarm is received, a rod block ___(2)___ be generated.

A (1) 3
(2) will

B (1) 3
(2) will NOT

C (1) 6.5
(2) will

D (1) 6.5
(2) will NOT

Answer: D

K/A:
215003 Intermediate Range Monitor (IRM) System
K6 Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : (CFR: 41.7 / 45.7)
04 Detectors

RO/SRO Rating: 3.0/3.0

Tier 2 / Group 1

K/A Match: This meets the K/A because this is testing a failure/malfunction of a detector effect on the IRM system (whether it generates a rod block)

Pedigree: New

Objective: LOI-CLS-LP-009-A, Obj. 3a
List the SRM/IRM system signals/conditions that will cause the following actions and the conditions under which each is bypassed: Rod Blocks

Reference: None

Cog Level: High

Explanation: The downscale setpoint for the IRMs is 6.5 on the 125 scale. The rod block is bypassed under these conditions because the IRMs are all on Range 1.

Distractor Analysis:

Choice A: Plausible because 3 is the downscale tech spec setpoint for SRMs and if the IRMs were not all on range 1 a rod block would be generated.

Choice B: Plausible because 3 is the downscale tech spec setpoint for SRMs and the second part is correct.

Choice C: Plausible because the first part is correct and if the IRMs were not all on range 1 a rod block would be generated.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

IRM DOWNSCALE

AUTO ACTIONS

1. Rod withdrawal block (bypassed when IRM range switch for the affected channel is on Range 1 or when the reactor mode switch is in RUN).
2. Computer printout.

CAUSE

1. IRM channel(s) indicating less than or equal to 6.5 on the 0-125 scale when its range switch is not on Range 1.
2. Improper ranging of IRM channels during reactor startup or shutdown.
3. IRM detector not fully inserted.
4. IRM detector failure.
5. Circuit malfunction.

OBSERVATIONS

1. IRM channel indicating less than or equal to 6.5 on the 0-125 scale.
2. IRM downscale (DNSC) white indicating light is on.
3. ROD OUT BLOCK (A-05 2-2) alarm, if affected IRM channel is not on Range 1.
4. If the affected IRM channel(s) is not on Range 1, the rod withdrawal permissive indicating light will be off.

13. 215004 1

Which one of the following identifies the criteria for when SRM detectors can **first** begin to be withdrawn from the core IAW 0GP-02, *Approach To Criticality And Pressurization Of The Reactor*?

A When all IRMs are above range 3.

B When SRM counts reach 2 x 10^5 counts.

C When RTRCT PERMIT light is illuminated.

D When SRM/IRM overlap has been established.

Answer: D

K/A:

215004 Source Range Monitor System

K5 Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM: (CFR: 41.5 / 45.3)

03 Changing detector position

RO/SRO Rating: 2.8/2.8

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing when SRMs can change detector positions.

Pedigree: New

Objective: LOI-CLS-LP-009-A, Obj 3b
List the SRM/IRM system signals/conditions that will cause the following actions and the conditions under which each is bypassed: Retract Permissive (SRM) only (LOCT)

Reference: None

Cog Level: Fundamental

Explanation: When SRM/IRM overlap has been established then SRM can be withdrawn to maintain an indicated SRM count rate between 100 cps and 200,000 cps.

Distractor Analysis:

Choice A: Plausible because this is the logic setpoint at which the SRM can be fully withdrawn

Choice B: Plausible because this is the point at which the SRM must be fully withdrawn.

Choice C: Plausible because this is an indication that is used during the withdrawal of the SRMs

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

NOTE

- SRM/IRM overlap is required to be demonstrated for all operable IRM channels prior to withdrawing SRMs from the fully inserted position. SRM/IRM overlap exists when IRM channels show an increase to at least twice their pre-startup levels and indicate at least 10% of scale (i.e., 12.5 on the digital readout 0-125 scale) before the first SRM channel reaches 5×10^5 cps (Technical Specifications, SR 3.3.1.1.6) .... ☐
- If desired, the level of the highest reading IRM (pre-startup) may be doubled and that value used as overlap criteria for all IRMs. This method will allow the operator to compare IRM channel response to a single value which is at least twice the pre-startup levels of the individual IRMs .... ☐

6.2 Pulling Rods To Achieve Criticality (continued)

NOTE

- With IRM channels below Range 3, the SRM channels will initiate a rod withdrawal block when either of the following conditions exists:
  - ◊ SRM channel indicates greater than 2×10^5 cps .... ☐
  - ◊ SRM channel indicates less than 10^2 cps with its detector **NOT** full in .... ☐
- SRM detectors are withdrawn two at a time so that the reactor flux level conditions are being monitored by channels that are **NOT** being affected by detector movement .... ☐

32. **WHEN** SRM/IRM overlap has been confirmed, **THEN withdraw** SRM detectors as required to maintain an indicated SRM count rate between 10^2 cps and 2×10^5 cps .... ________

CAUTION

Repositioning IRM range switches is performed by one operator, using one hand, on one trip system at a time {8.1.6} .... ☐

33. As reactor power rises, **reposition** the IRM range switches to maintain IRM indication on recorders between 15 and 50 on the 0-125 scale .... ________

34. **WHEN** all OPERABLE IRM channels are above Range 3 **AND** prior to reaching Range 7, **THEN** fully **withdraw** all SRM detectors .... ________

14. 215005 1

Which one of the following identifies the power supply to the APRM channel NUMACs?

A All APRM channels receive 120 VAC power from UPS

B All APRM channels receive 120 VAC power from both RPS Bus A and RPS Bus B

C APRM Channels 1 & 3 receive power from ONLY120 VAC RPS Bus A
APRM Channels 2 & 4 receive power from ONLY120 VAC RPS Bus B

D APRM Channels 1 & 3 receive power from Division I 24/48 VDC
APRM Channels 2 & 4 receive power from Division II 24/48 VDC

Answer: B

K/A:

215005 Average Power Range Monitor/Local Power Range Monitor

K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)

02 APRM channels

RO/SRO Rating: 2.6/2.8

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the power supply to the NUMACs.

Pedigree: Modified from 2015 NRC Exam

Objective: LOI-CLS-LP-09.6, Objective 7a
Describe the operational relationships between the PRNMS and the following:
Reactor Protection System

Reference: None

Cog Level: Fundamental

Explanation: Each APRM channel NUMAC is equipped with a dual power supply arrangement with one supply from RPS Bus A and the other supply from RPS Bus B. All four APRM channels maintain power on loss of either supply as long as the other supply is available

Distractor Analysis:

Choice A: Plausible because UPS supplies power to the APRM ODA and recorder

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the power supply arrangement for the voters.

Choice D: Plausible because other ranges of nuclear instrumentation (SRM/IRM) receive their power from here.

SRO Basis: N/A

2.8.8 PRNMS Power Supplies

The Power Range Neutron monitoring System uses one Quadruple Voltage Power Supply (QLVPS) chassis and four Dual Low Voltage Power Supplies (DLVPS), one for each bay of the PRNMS panel, to provide redundant power to the NUMAC instruments. These LVPS convert 120 VAC to low voltage DC. See Figure 09.6-14.

Each APRM instrument receives power from two power supplies, LVPS 1 and LVPS 4. LVPS 1 is fed from RPS Bus A while LVPS 4 is fed from RPS Bus B. Therefore, a loss of an RPS Bus will not affect operation of the APRM NUMACS. Each RBM instrument also

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4.3.1 Reactor Protection System

APRM channels provide signals to open contacts in the scram trip logic of the RPS System under various conditions discussed previously.

The RPS System provides power to each of the four APRM instruments, which in turn provide power to all subsystems driven from the APRM instruments or NUMAC. Both RPS busses, A and B, provide power to each APRM instrument, as well as, each RBM. Therefore, a loss of one RPS bus will not affect operation of the PRNMS.

The reactor mode switch provides input to each APRM instrument to determine when to enforce the fixed or flow biased scram trip and rod block settings. OPRM circuitry is enabled only when power/flow conditions are met and the mode switch in RUN.

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Which one of the following is the power supply to APRM Channel 4 NUMAC on P608?

A✓ 120 VAC RPS

B. 120 VAC UPS

C. 24/48 VDC Div I

D. 24/48 VDC Div II

15. 215005 2
Which one of the following completes the statement below?

An APRM must have at least ___(1)___ of the assigned LPRMs operable with at least ___(2)___ LPRM inputs per axial level operable.

A (1) 18
(2) 2

B (1) 18
(2) 3

C (1) 17
(2) 2

D (1) 17
(2) 3

Answer: D
K/A:
215005 Average Power Range Monitor/Local Power Range Monitor
K5 Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: (CFR: 41.5 / 45.3)
04 LPRM detector location and core symmetry

RO/SRO Rating: 2.9/3.2

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the knowledge of the LPRM inputs per axial level that are required and the minimum number of inputs for core symmetry that are required.

Pedigree: New

Objective: LOI-CLS-LP-09.6, Obj. 13b
Given plant conditions, predict the effect of a single or multiple LPRM failure on the following: APRM

Reference: None

Cog Level: Fundamental

Explanation: An APRM channel must have a minimum of 3 LPRM inputs per level and a total of 17 LPRM inputs to be operable

Distractor Analysis:

Choice A: Plausible because an OPRM requires 18 LPRMs with at least 2 LPRM inputs to each cell.

Choice B: Plausible because an OPRM requires 18 LPRMs and 3 per level is correct for APRMS.

Choice C: Plausible because 17 is correct for APRMs and OPRMs require at least 2 LPRM inputs to each cell.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

4.2.1 LPRM

LPRM System failure, depending on the extent or failure type, can cause the loss of LPRM functions including the loss of indication, incorrect operation of rod block or scram protection. Generally, the following symptoms are exhibited for LPRM failure for the affected LPRM:

- Indicates upscale, accompanied by an upscale alarm.
- Indicates downscale, accompanied by a downscale alarm.
- Indicator reads erratically.

The results of an LPRM failure may lead to an APRM or OPRM becoming inoperable. An APRM channel must have a minimum of 3 LPRM inputs per level and a total of 17 LPRM inputs to be operable.

An OPRM cell must have a minimum of 2 LPRM inputs to each cell and a total of 18 cells to be operable.

16. 216000 1

A Unit Two plant cooldown is being performed with the following plant conditions:

Reactor water level	175 inches, steady
Reactor pressure band	500 - 700 psig
Drywell ref leg temp	175°F

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

The lowering of reactor pressure causes the N004A/B/C (Narrow Range) reactor water level instruments indicated level error to ___(1)___.

The reactor water level that would correspond to Low level 4 (LL4) is ___(2)___.

A (1) increase
(2) -60 inches

B (1) increase
(2) -65 inches

C (1) decrease
(2) -60 inches

D (1) decrease
(2) -65 inches

Answer: A

K/A:

216000 Nuclear Boiler Instrumentation

A2 Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

11 Heatup or cooldown of the reactor vessel

RO/SRO Rating: 3.2/3.3

Tier 2 / Group 2

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
The first part of the question deals with predicting the effect of a cooldown on indicated level error while the second part has the student has to determine based on the lowering pressure what LL4 value would be which is the value that emergency depressurization would be required. They must utilize the lower end of the pressure band to determine LL4. If LL4 cannot be maintained then ED is required.

Pedigree: New

Objective: LOI-CLS-LP-001.2, Objective 5a
Explain the effect that the following will have on reactor vessel level and/or pressure indications: Plant heatup/cooldown

Reference: 0EOP-01-UG, Attachment 26

Cog Level: High

Explanation: The indicated level error is sensitive to changes in the saturation density of the bulk water as a function of system pressure. The amount of the indicated level error is also a function of the difference in the actual water level and the variable leg instrument tap elevation. As the saturation density increases (pressure decreases) the indicated level error will increase for the narrow and wide range instruments and decrease for the fuel zone and shutdown range instruments due to calibration criteria.
From OI-37.11, TAF, LL4, and LL5 values should be determined based on the reference leg area temperature and RPV pressure compensation curves, using RPV pressure at the low end of the established RPV pressure control band. Based on the low end of the band of 500 psig and < 200°F in the drywell the LL4 value would be -60 inches.

Distractor Analysis:

Choice A: Correct Answer, see explanation

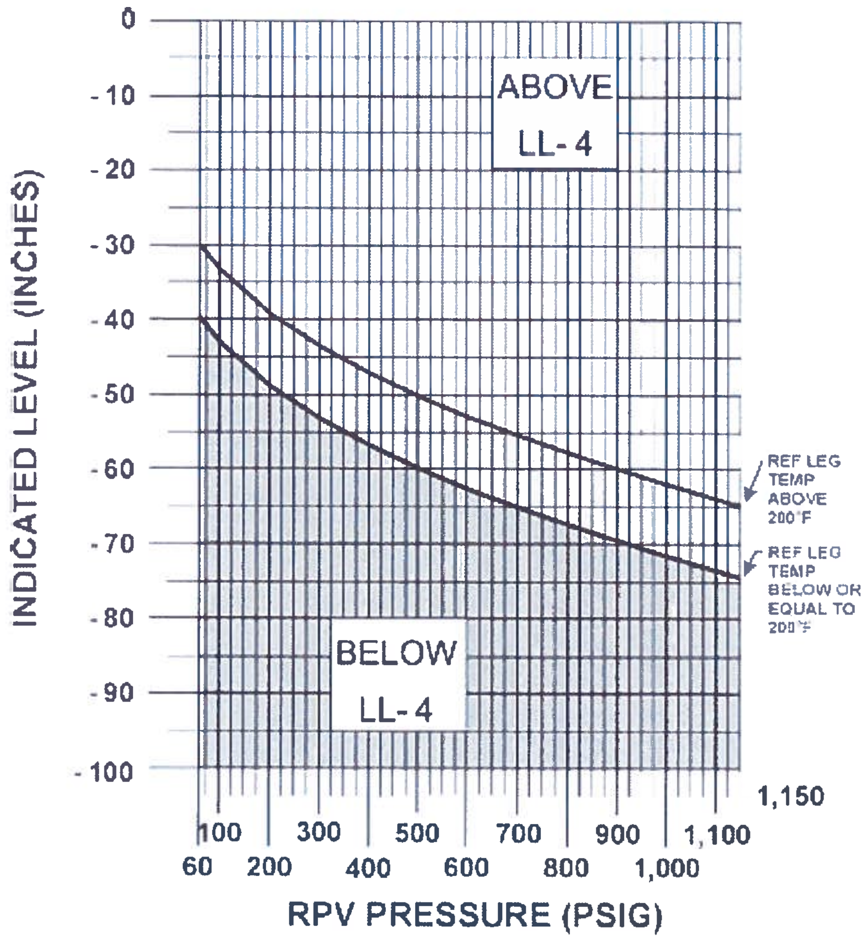
Choice B: Plausible because the first part is correct and the second part would be correct for 700 psig.

Choice C: Plausible because this would be correct for the fuel zone or shutdown range instruments and the second part is correct.

Choice D: Plausible because this would be correct for the fuel zone or shutdown range instruments and the second part would be correct for 700 psig.

SRO Basis: N/A

<< Unit 2 RPV Level at LL 4
(Minimum Steam Cooling RPV Level) >>



When RPV pressure is less than 60 psig, use indicated level. LL-4 is -27.5 inches.

4.1.2 System Pressure (Heat-up and Cool-down)

The indicated level error is sensitive to changes in the saturation density of the bulk water as a function of system pressure. The amount of the indicated level error is also a function of the difference in the actual water level and the variable leg instrument tap elevation. As the saturation density increases (pressure decreases) the indicated level error will increase for the narrow and wide range instruments and decrease for the fuel zone and shutdown range instruments due to calibration criteria. As actual water level decreases, the amount of error will decrease because less vessel water level is acting on the instrument.

5.3.3 1(2)EOP-01-RVCP, Reactor Vessel Control Procedure

1. Level Leg

   a. The CRS directs an initial RPV level band of +166 to +206 inches. The reactor operator actually maintains a RPV level band of +170 to +200 inches to provide additional margin to the reactor scram and turbine trip set points. The CRS may direct a widened band based on plant conditions and other controlling procedures associated with the transient.

   b. If RPV level is above TAF, injection flow should be controlled so as to control the cooldown rate below 100°F/hr.

   c. If RPV level is below TAF, RPV level should be rapidly restored to above TAF, and then injection flow reduced so as to control the cooldown rate below 100°F/hr.

   d. TAF, LL4, and LL5 values should be determined based on the reference leg area temperature and RPV pressure compensation curves, using RPV pressure at the low end of the established RPV pressure control band.

17. 217000 1

Following a loss of feedwater, RCIC automatically initiated and subsequently tripped on low suction pressure.

Current plant status is:
Reactor water level is 150 inches
RCIC flow controller in Manual set at 200 gpm

Subsequently, the following actions are taken:
RCIC suction transferred to Torus
E51-V8, Turbine Trip and Throttle Valve is closed
E51-V8 is re-opened
PF push button on the RCIC flow controller is depressed

Which one of the following identifies the indicated flow on the RCIC flow controller that would be observed for these conditions?

A 0 gpm

B 200 gpm

C 400 gpm

D 500 gpm

Answer: C

K/A:
217000 Reactor Core Isolation Cooling System
A1 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: (CFR: 41.5 / 45.5)
01 RCIC flow

RO/SRO Rating: 3.7/3.7

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the prediction of what RCIC flow will be when operating the RCIC system.

Pedigree: New

Objective: CLS-LP-016-A, Obj.16c
Describe how the following evolutions are performed during operation of the RCIC System: Adjusting RCIC flow in the Reactor Level Control mode.

Reference: None

Cog Level: high

Explanation: The RCIC Turbine is provided with a solenoid operated remote electrical tripping device, which when actuated (in this case by low suction pressure), will close the Turbine Trip and Throttle Valve, E51-V8. Resetting of the remote electrical tripping device may be accomplished from the RTGB. The RCIC system is restarted after auto initiation and turbine trip by fully closing the V-8, and re-opening the V-8. Located on the controller face is a PF (programmable function) pushbutton which when depressed an automatic transfer from manual to automatic at a predetermined setpoint of 400 GPM will result. This button (PF) has no function if the controller is already in automatic.

Distractor Analysis:

Choice A: This is plausible because this answer would be correct for these actions following a high RPV water level trip of RCIC

Choice B: Plausible because this would be correct if the operator did not depress the PF pushbutton.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the PF push button would raise RCIC flow to rated (400 gpm) and not maximum per procedure (500 gpm). Achieving 500 gpm would require the flow control setpoint to be manually raised.

SRO Basis: N/A

From SD-16:

Also located on the controller face is a PF (programmable function) pushbutton. When depressed an automatic transfer from MANUAL to AUTOMATIC at a predetermined setpoint of 400 GPM will result. NOTE: This button (PF) has no function if the controller is already in AUTOMATIC.

For various internal processing failures, the controller is designed to hold the last output and automatically switch to MANUAL giving the operator manual control capability. Barring operator intervention, this failure could result in rising or lowering RCIC flow and would be indicated by the red FAIL lamp on the controller face. Failure display code can then be checked using the side panel keypad. A down scale failure of the controller is possible and would result in turbine operation at well below the normal minimum speed of 2000 rpm. An upscale failure is highly unlikely but would result in turbine speed at or above the maximum running speed of 4600 rpm. Failures associated with the dynamic response are also highly unlikely but would produce either excessively sluggish responses or dynamic instability (full scale oscillations) when in the Automatic mode. Programmable settings internal to the controller are maintained during a loss of 24 Vdc power supply by a lithium battery. If this battery voltage drops to a pre-determined low value, the yellow ALARM light will flash. If the input signals are not within the limits of -6.3% to 106.3% or if the input or output signals are not intact, the Yellow ALARM light will come on solid.

ATTACHMENT 9
Page 1 of 1

<< RCIC Instructional Aid for EOPs >>

RESTARTING RCIC AFTER AUTO INITIATION AND TURBINE TRIP

(2OP-16 Section 8.7)

1. **ENSURE** THE E51-V8 (VALVE POSITION) **AND** E51-V8 (MOTOR OPERATOR) ARE CLOSED. .... ☐
2. **PLACE** RCIC FLOW CONTROL IN MANUAL (M) **AND ADJUST** OUTPUT TO 0%.......... ☐
3. **JOG OPEN** E51-V8 UNTIL THE TURBINE SPEED IS CONTROLLED BY THE GOVERNOR.......... ☐
4. **FULLY OPEN** E51-V8. .... ☐
5. **SLOWLY RAISE** TURBINE SPEED UNTIL FLOW RATE OF AT LEAST 120 GPM. .... ☐
6. **ENSURE** E51-F019 IS CLOSED WITH FLOW GREATER THAN 80 GPM.......... ☐
7. **WHEN** SYSTEM CONDITIONS ARE STABLE, **THEN ADJUST** SETPOINT, **AND TRANSFER** RCIC FLOW CONTROL TO AUTO (A). .... ☐
8. **SLOWLY ADJUST** FLOW RATE USING RCIC FLOW CONTROL IN AUTO (A).......... ☐
9. **ENSURE** THE FOLLOWING:

   BAROMETRIC CNDSR VACUUM PUMP HAS STARTED .... ☐

   SBGT STARTED (2OP-10).......... ☐

   SGT-V8 AND SGT-V9 ARE OPEN.......... ☐

18. 218000 1

Which one of the following completes both statements below concerning the Automatic Depressurization System (ADS) reactor water level inputs from the Nuclear Boiler System?

The ___(1)___ instruments provide LL3 inputs to ADS initiation logic.

The ___(2)___ range instruments provide LL1 inputs to ADS logic.

A (1) Fuel Zone
(2) Narrow

B (1) Fuel Zone
(2) Shutdown

C (1) Wide range
(2) Narrow

D (1) Wide range
(2) Shutdown

Answer: C

K/A:
217000 Automatic Depressurization System
K1 Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
03 Nuclear boiler instrument system

RO/SRO Rating: 3.7/3.8

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the connection between ADS and level indicators.

Pedigree: New

Objective: LOI-CLS-LP-0001.2, Obj 4a
List the systems which receive input from the Vessel Instrumentation system for the following: Level signal

Reference: None

Cog Level: Fundamental

Explanation: B21-LT-N031(Wide Range) provide LL3 initiation from N031A and C for Logic B and from N031B and D for Logic A. B21-LT-N042 (Narrow Range) provide LL1 confirmatory from N042A for Logic B and from N042B for Logic A.

Distractor Analysis:

Choice A: Plausible because the fuel zone instruments covers LL3 (45 inches) and the second part is correct.

Choice B: Plausible because fuel zone instruments covers LL3 (45 inches) and the shutdown range covers the LL1 setpoint (166 inches).

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the wide range is correct and the shutdown range covers the LL1 setpoint (166 inches).

SRO Basis: N/A

4.1.2 Automatic Operation

The ADS logic automatically opens the ADS valves in the event the HPCI System fails to maintain reactor level during a LOCA. The seven ADS valves open automatically when all the following conditions are met on either of two logic channels (A or B) associated with ADS:

- Reactor low water level (LL3 from B21-LTS-N031A and C or B and D).
- Reactor confirmatory low water level (LL1 from B21-LTS-N042A or B).
- Operation of both pumps of an RHR loop or one Core Spray pump as indicated by a pump discharge pressure of 115 psig (either E11-PS-N016A AND C or B AND D or E11-PS-N020A AND C or B AND D for RHR or either E21-PS-N008A AND E11-PS-N009A or E21-PS-N008B AND E21-PS-N009B for CS).
- A time delay of 83 seconds has elapsed (timer B21-TDPU-K5A or B).
- AUTO/INHIBIT switches in AUTO for either or both logic channels A and B.

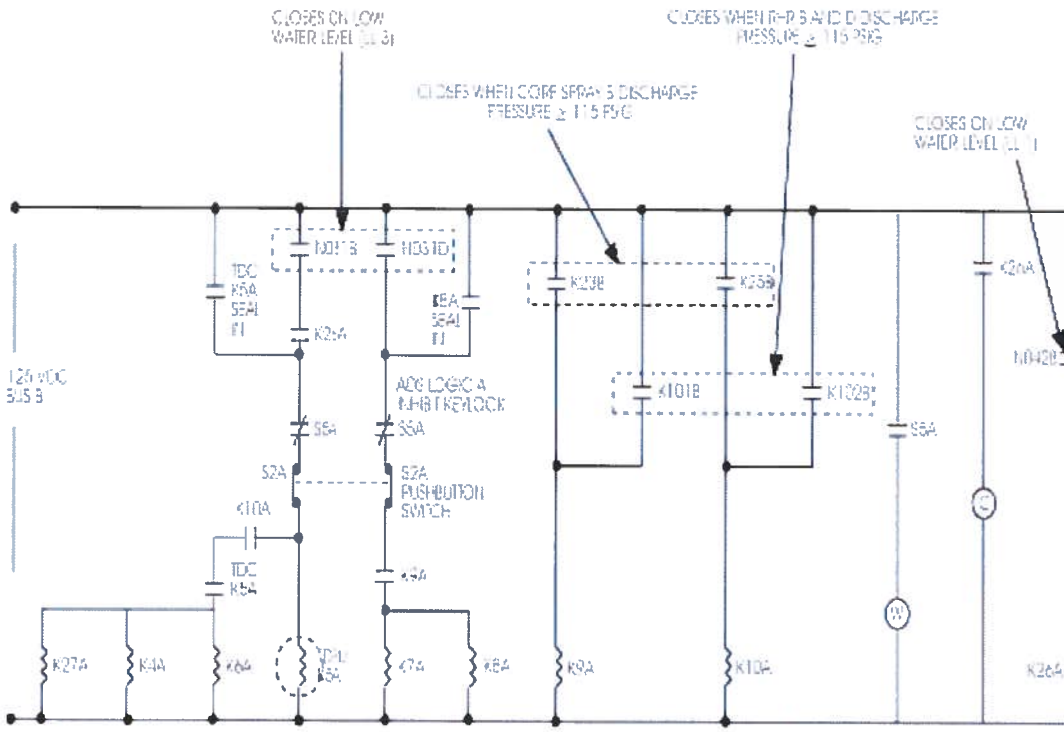
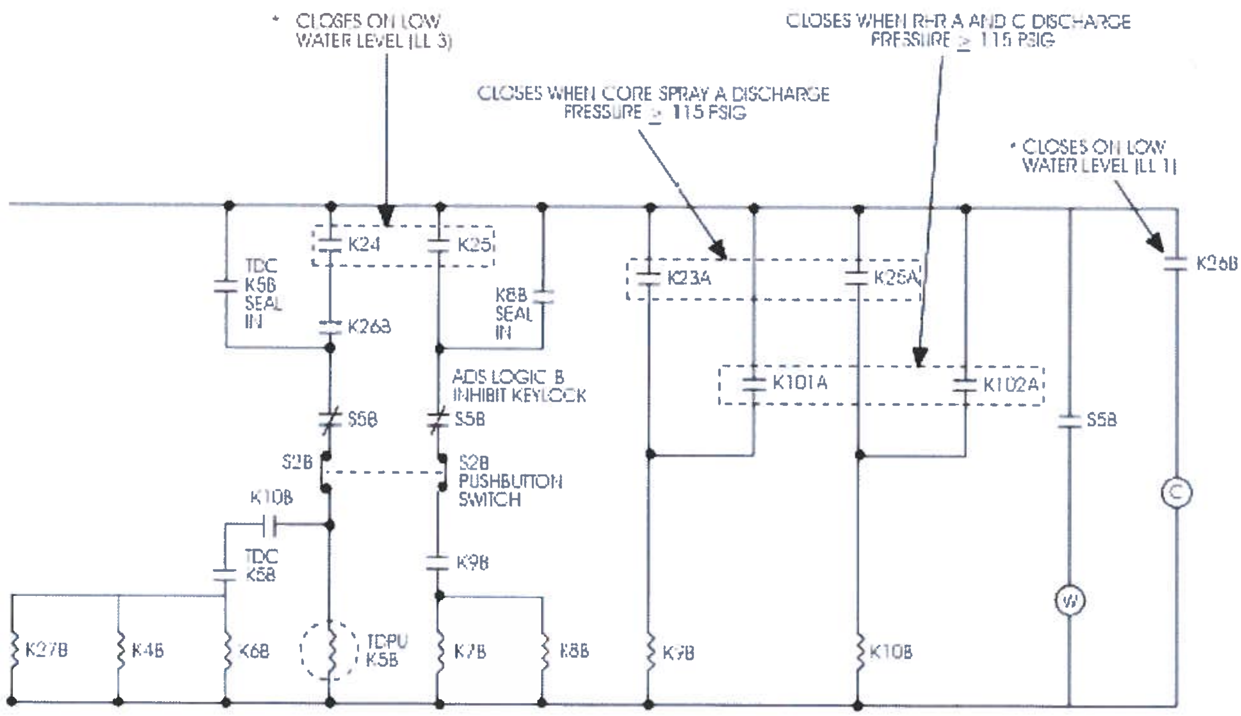


FIGURE 20-10
ADS Logic Channel "A"



19. 218000 2
Unit One is operating at power with Core Spray Pump 1B under clearance.
A small break LOCA occurs simultaneously with a Loss of Off-site Power to both units.

DG1 and DG4 fail to start and tie onto their respective E bus.

The following plant conditions exist on Unit One:

A-03 (5-1) *Auto Depress Timers Initiated*	In alarm
A-03 (6-9) *Reactor Low Wtr Level Initiation*	In alarm
RPV pressure	600 psig
Drywell pressure	13 psig

Which one of the following completes both statements below?

ADS ___(1)___ auto initiate.

After ADS is initiated (either automatically or manually), RPV water level will be restored with RHR Loop ___(2)___.

A (1) will
(2) A

B (1) will
(2) B

C (1) will NOT
(2) A

D (1) will NOT
(2) B

Answer: C
K/A:
217000 Automatic Depressurization System
K3 Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: (CFR: 41.7 / 45.4)
01 Restoration of reactor water level after a break that does not depressurize the reactor when required

RO/SRO Rating: 4.4/4.4

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the knowledge of the effect of the malfunction on auto initiation of ADS and how level will be restored.

Pedigree: Last used on 10-1 NRC exam

Objective: CLS-LP-20 Obj. 16b
Given plant conditions, predict how the following will be affected by a loss or malfunction of ADS/SRVs: Reactor water level

Reference: None

Cog Level: high

Explanation: With the loss of offsite power and 1B CS pump under clearance this would leave only one pump available in each RHR loop. Therefore ADS logic is lost. Level will continue to lower until the ADS valves are manually opened (emergency depressurization) at which time the running low pressure pumps will be able to add water. Injection would be from the A Loop of RHR as the B Loop injection valves do not have power.

Distractor Analysis:

Choice A: Plausible because ADS does have initiation conditions except that the logic will not have the appropriate pumps lined up for injection.

Choice B: Plausible because ADS does have initiation conditions except that the logic will not have the appropriate pumps lined up for injection. B Loop of RHR does not have power to the injection valves

Choice C: Correct Answer, see explanation.

Choice D: Plausible because ADS will not auto initiate but the B Loop of RHR does not have power to the injection valves.

SRO Basis: N/A

SD-20

4.1.2 Automatic Operation

The ADS logic automatically opens the ADS valves in the event the HPCI System fails to maintain reactor level during a LOCA. The seven ADS valves open automatically when all the following conditions are met on either of two logic channels (A or B) associated with ADS:

- Reactor confirmatory low water level (LL1 from B21-LTS-N042A or B).
- Operation of both pumps of an RHR loop or one Core Spray pump as indicated by a pump discharge pressure of 115 psig (either E11-PS-N016A AND C or B AND D or E11-PS-N020A AND C or B AND D for RHR or either E21-PS-N008A AND E11-PS-N009A or E21-PS-N008B AND E21-PS-N009B for CS).
  - A time delay of 83 seconds has elapsed (timer B21-TDPU-K5A or B).
  - AUTO/INHIBIT switches in AUTO for either or both logic channels A and B. Reactor low water level (LL3 from B21-LTS-N031A and C or B and D).

20. 223001 1

Which one of the following completes the statement below concerning the Fuel Zone instruments, N036 and N037, during a loss of drywell cooling?

The reference leg density will ___(1)___ causing the indicated level to read ___(2)___ than actual level.

A (1) rise
(2) higher

B (1) rise
(2) lower

C (1) lower
(2) higher

D (1) lower
(2) lower

Answer: C

K/A:

223001 Primary Containment System and Auxiliaries

K3 Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES will have on following: (CFR: 41.7 / 45.4)

09 Nuclear boiler instrumentation

RO/SRO Rating: 2.8/3.1

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the knowledge of a loss of DW cooling has on instrumentation.

Pedigree: Bank

Objective: LOI-CLS-LP-001.2, Obj. 05c
Explain the effect that the following will have on reactor vessel level and/or pressure indications: High containment (primary and secondary) temperatures.

Reference: None

Cog Level: High

Explanation: The reference leg length is longer than the variable leg length, therefore secondary temp increasing makes the instrument read higher than actual level.

Distractor Analysis:

Choice A: Plausible because density is a function of temperature and the temperature is rising. The second part is correct.

Choice B: Plausible because density is a function of temperature and the temperature is rising. The second part is plausible because if the first part is seen as right then this would be correct.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and the second part is the opposite of the answer.

SRO Basis: N/A

21. 223002 1
Unit One is at 75% power.
The 1A RPS MG set trips.
RPS Bus A has NOT been transferred to an alternate power supply.

Which one of the following identifies the Main Steam Line Isolation Valve (MSIV) logic lamp status on P601 panel?

	Inboard MSIV Logic		Outboard MSIV Logic	
	DC	AC	DC	AC
A	LIT	OFF	LIT	OFF
B	OFF	LIT	OFF	LIT
C	LIT	OFF	OFF	LIT
D	OFF	LIT	LIT	OFF

Answer: C
K/A:
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off
A1 Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: (CFR: 41.5 / 45.5)
01 System indicating lights and alarms

RO/SRO Rating: 3.5/3.5

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the ability to predict the light status on a loss of a power supply

Pedigree: New

Objective: LOI-CLS-LP-012, Objective 12
Given plant conditions, determine how the following will affect PCIS:
c. Loss of RPS

Reference: None

Cog Level: High

Explanation: See Notes Section. RPS A provides power to PCIS Logic A. PCIS Logic A is Inboard AC and Outboard DC indicating lights on P601.

Distractor Analysis:

Choice A: Plausible because first part is correct. Outboard light is DC.

Choice B: Plausible because second part is correct. Inboard light is AC.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the lights are just reversed. This would be true for Loss of RPS B.

SRO Basis: N/A

4.3.10 AC Distribution

RPS MG sets supply power to the following PCIS related components:

RPS Bus A

PCIS Trip System A logic
PCIS Trip Channels A1 and A2 logic
Inboard isolation logic for valves:
- Inboard reactor water sample valve
- Main Steam Line drains
- Shutdown cooling suction
- RWCU
- Inboard RHR Sample valves
- Drywell floor and equipment drains
- CAC/CAMS/PASS for LL1 and High Drywell pressure

Valve operating power:
- Inboard reactor water sample valve
- Inboard RHR Sample valves
- Drywell floor and equipment drains
- Inboard "AC" MSIV solenoids
- Reactor Building Vent Exh Rad Monitor N010A
- Main Steam Line Rad Monitors A and C (alarm function only)

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P601 panel. These lights are arranged above the MSIV control switches as follows:

TABLE 25-3, MSIV ISOLATION SIGNAL STATUS

Light	INBD DC	INBD AC	OUTBD DC	OUTBD AC
Solenoid Power	125 VDC "A"	RPS "A"	125 VDC "B"	RPS "B"
PCIS Logic	B	A	A	B

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22. 234000 1

Which one of the following identifies the affect if both Refuel Bridge hoist grapple hooks are not open five seconds **after** placing the Engage/Release switch to Release?

A Fuel Hoist Interlock is generated.

B Engage amber light extinguishes.

C Fault lockout is generated.

D Grapple hooks will reclose.

Answer: D

K/A:
234000 Fuel Handling
A3 Ability to monitor automatic operations of the FUEL HANDLING EQUIPMENT including: (CFR: 41.7 / 45.7)
01 Crane/refuel bridge movement

RO/SRO Rating: 2.6/3.1

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the ability to monitor the crane grapple hooks auto re-close feature.

Pedigree: New

Objective: LOI-CLS-LP-58.1, Obj 13
Describe the operation of the grapple if the ENGAGE/RELEASE Switch is positioned to RELEASE and both grapple hooks are not open within 5 seconds when the main hoist is loaded.

Reference: None

Cog Level: Fundamental

Explanation: If the grapple does not indicate released (open) within 5 seconds, the solenoid is de-energized and the grapple hooks re-close. The switch must then be taken to the ENGAGE position to reset the logic prior to making another attempt to release the grapple.

Distractor Analysis:

Choice A: Plausible because a Fuel Hoist Interlock is generated for a number of reasons.

Choice B: Plausible because this is an indication of operation of the grapple hooks.

Choice C: Plausible because a fault lockout is generated for a number of reasons.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

23. 239002 1

Which one of the following identifies the SRV component that will prevent siphoning of water into the SRV discharge piping?

A Vacuum breaker

B Check Valve

C T-Quencher

D Sparger

Answer: A

K/A:
239002 Safety Relief Valves
K4 Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)
03 Prevents siphoning of water into SRV discharge piping and limits loads on subsequent actuation of SRV's

RO/SRO Rating: 3.1/3.3

Tier 2 / Group 1

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because it is testing the knowledge of the design feature that prevents siphoning of water.

Pedigree: New

Objective: LOI-CLS-LP-020, Obj. 7d
State the purpose of the following: SRV tailpipe vacuum breakers

Reference: None

Cog Level: Fundamental

Explanation: Following operation of the valve, a vacuum is created in the SRV tailpipe as the steam condenses. Water in the line above the suppression pool water level would cause excessive pressure at the SRVs discharge when and if the valve reopened. For this reason, a vacuum relief valve is provided on each SRV tailpipe to prevent drawing water up into the line due to this steam condensation following SRV operation.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because this is a component that is typically used to provide an anti-siphon break.

Choice C: Plausible because this is a component on the SRV that the steam discharges through and has holes in the pipe which could be thought of an anti-siphon type break.

Choice D: Plausible because this is a component on the SRV that the steam discharges through and has holes throughout the pipe which could be thought of an anti-siphon type break. (The supplemental fuel pool cooling sparger has this design to prevent siphoning of water)

SRO Basis: N/A

24. 241000 1

Which one of the following identifies the criteria for tripping the main turbine IAW the Unit Two Scram Immediate Actions of 0EOP-01-UG, *Users Guide*?

A When APRM's indicate downscale trip.

B When steam flow is less than 3 Mlbs/hr.

C When reactor water level is 160 inches and rising.

D When reactor mode switch is placed in SHUTDOWN.

Answer: A

K/A:

241000 Reactor/Turbine Pressure Regulating System

A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)

14 Turbine trip

RO/SRO Rating: 3.8/3.7

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the ability of tripping the turbine from the control room.

Pedigree: New

Objective: LOI-CLS-LP-300-C, Obj. 2
List the immediate operator actions for a reactor scram.

Reference: None

Cog Level: Fundamental

Explanation: The main turbine is tripped after reactor power is below 2% which is indicated by APRM downscale trip lights illuminated.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because this is a criteria for placing the mode switch to shutdown which is an immediate operator action.

Choice C: Plausible because this is a criteria for a reactor feed pump which is an immediate operator action.

Choice D: Plausible because this is an immediate operator action that is performed on the scram.

SRO Basis: N/A

Unit 2 Scram Immediate Actions (0EOP-01-UG)

SCRAM IMMEDIATE ACTIONS

1. Ensure SCRAM valves OPEN by manual SCRAM or ARI initiation.

2. **WHEN** steam flow less than 3 x 10^6 lb/hr,
   THEN place reactor mode switch in SHUTDOWN.

3. **IF** reactor power below 2% (APRM downscale trip),
   THEN trip main turbine.

4. Ensure master RPV level controller setpoint at +170 inches.

5. **IF:**

   - Two reactor feed pumps running

     AND

   - RPV level above +160 inches

     AND

   - RPV level rising,

   THEN trip one.

25. 245000 1

Which one of the following completes both statements below concerning the Main Generator Voltage Regulator?

The automatic voltage regulator maintains a constant generator ___(1)___ voltage.

While in the automatic voltage regulation mode, the manual voltage regulator setting ___(2)___ automatically follow the automatic setpoint.

A (1) field
(2) does

B (1) field
(2) does NOT

C (1) terminal
(2) does

D (1) terminal
(2) does NOT

Answer: D

K/A:
245000 Main Turbine Generator and Auxiliary Systems
K4 Knowledge of MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)
07 Generator voltage regulation

RO/SRO Rating: 2.5/2.6

Tier 2 / Group 2

K/A Match: This meets the K/A because this is testing the design of the auto regulator as to what it controls and whether the manual regulator automatically follows the auto regulator.

Pedigree: Bank

Objective: LOI-CLS-LP-027.0, Obj 7c
Given a simplified diagram of the Main Generator Voltage Regulator, explain how:
a. the MANUAL regulator controls Generator output voltage
b. the AUTOMATIC regulator controls Generator output voltage
c. to transfer from one Voltage Regulator to the other

Reference: None

Cog Level: Fundamental

Explanation: The AVR controls terminal voltage while the manual regulator controls field voltage. The manual voltage regulator does not track the setpoint of the AVR, this must be manually adjusted in the control room.

Distractor Analysis:

Choice A: Plausible because the MVR controls field voltage and the DG manual voltage regulator does track the auto regulator setpoint.

Choice B: Plausible because the MVR controls field voltage and the second part is correct.

Choice C: Plausible because the first part is correct and the DG manual voltage regulator does track the auto regulator setpoint.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

2.15 Excitation Control (Refer to Figure 27-12)

The Silicon Controlled Rectifier (SCR) bridge circuit is used as a variable DC voltage source to control the exciter field current as required by the AC or DC regulator. The source of the control signal for the SCRs is determined by the Regulator Mode Selector Switch (43CS) located on Panel XU-1. When Manual is selected, the DC regulator maintains a constant generator field voltage that is determined by the Manual Volts Adjust Rheostat. When the Automatic regulator is selected, the AC regulator maintains a constant generator terminal voltage.

2.17.8 Generator Voltage Regulator Differential Voltmeter

This is a standard voltmeter that measures the magnitude and polarity of the difference between the DC regulator output signal and the AC regulator output signal. When shifting control from the DC voltage regulator to the AC regulator or back, it is important to ensure that the signals are the same. As an example, if the meter reads to the clockwise of zero, then the manual regulator output is less than the automatic regulator. If the meter reads counter clockwise of zero, then the manual signal is larger than the automatic signal. The meter indicates 0-10 volts in both directions.

Failure to have the regulator control signals matched when shifting regulator modes may result in transients on the generator output. The severity of the transient would be determined by the direction and magnitude of the mismatch.

26. 259001 1
Unit One Reactor Feed Pump 1B is operating in automatic DFCS control at 4500 RPM. The DFCS control signal to Reactor Feed Pump 1B woodward governor **immediately** fails downscale.

Which one of the following completes the statement below?

Reactor Feed Pump 1B speed will:

A lower to 0 rpm.

B lower to 1000 rpm.

C lower to 2450 rpm.

D remain at 4500 rpm.

Answer: D
K/A:
259001 Reactor Feedwater System
A1 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR FEEDWATER SYSTEM controls including: (CFR: 41.5 / 45.5)
04 RFP turbine speed: Turbine-Driven-Only

RO/SRO Rating: 2.8/2.7

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing the ability to predict the response in parameters.

Pedigree: New

Objective: LOI-CLS-LP-032.2, Obj. 13d
Given plant conditions and one of the following events, use plant procedures to determine the actions required to control and/or mitigate the consequences of the event:
Loss of signal interface between controllers and processor.

Reference: None

Cog Level: High

Explanation: If RFPT A(B) *MAN/DFCS* selector switch is in *DFCS*, and DFCS control signal subsequently drops below 2450 rpm, or increases to greater than 5450 rpm, then Woodward 5009 digital controls will automatically assume RFPT speed control and maintain current pump speed.

Distractor Analysis:

Choice A: Plausible if the student believes that a loss of input signal will cause the controller to use 0 as the input for the speed of the pump. (i.e. HPCI/RCIC controllers will fail to zero)

Choice B: Plausible because an idled RFP is maintained at 1000 rpm.

Choice C: Plausible because 2450 is the low end of the controller function.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

NOTE: If RFPT A(B) *MAN/DFCS* selector switch is in *DFCS*, and DFCS control signal subsequently drops below 2450 rpm, or increases to greater than 5450 rpm, then Woodward 5009 digital controls will automatically assume RFPT speed control and maintain current pump speed. In this condition, the RFPT will only respond to *LOWER/RAISE* speed control switch commands until *MAN/DFCS* selector switch is placed in *MAN*, DFCS *CTRL RESET* pushbutton is depressed, and *MAN/DFCS* selector switch returned to *DFCS*.

3.13 Plant management has recommended one RFPT be idled at 1000 rpm with the discharge valve closed, during conditions with one RFPT in service.

27. 259002 1

Which one of the following completes both statements below concerning the reactor feed pump turbine (RFPT) DFCS controls?

During a RFPT startup, transfer to DFCS control is performed when RFPT speed is approximately ___(1)___.

DFCS will automatically control the speed of the RFPT up to ___(2)___.

A (1) 1000 rpm
(2) 5450 rpm

B (1) 1000 rpm
(2) 6150 rpm

C (1) 2550 rpm
(2) 5450 rpm

D (1) 2550 rpm
(2) 6150 rpm

Answer: C

K/A:
259002 Reactor Water Level Control System
A3 Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: (CFR: 41.7 / 45.7)
01 Runout flow control

RO/SRO Rating: 3.0/3.0

Tier 2 / Group 1

K/A Match: This meets the K/A because this is testing the upper limit of the auto (DFCS) controls which in essence prevent pump runout of the reactor feed pumps.

Pedigree: new

Objective: LOI-CLS-LP-032.2, Obj. 5d
Describe the operation of the DFCS in the following operating modes:
Master Level Control Mode (auto and manual)

Reference: None

Cog Level: Fundamental

Explanation: DFCS will be placed into service with the manual output set at 2550 RPM. The DFCS system will control the RFPT speed from 2450 - 5450 RPMs

Distractor Analysis:

Choice A: Plausible because 1000 RPM is the idle speed of the RFPT and the second part is correct.

Choice B: Plausible because 1000 RPM is the idle speed of the RFPT and 6150 is the overspeed setpoint of the woodward controls.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and 6150 is the overspeed setpoint of the woodward controls.

SRO Basis: N/A

6.1.5 Reactor Feed Pump Startup from Idle Speed to Injection at Low Pressure Conditions (continued)

NOTE

When using RFPT A(B) Lower/Raise speed control switch, reactor feed pump turbine speed will change at a rate of 50 rpm per second. If switch is held in LOWER or RAISE for greater than 3 seconds, the rate of change will rise to 375 rpm per second. ☐

6. **Maintain** RFPT A(B) discharge pressure at least 100 psig greater than reactor pressure by adjusting RFPT A(B) Lower/Raise speed control switch until RFPT speed is approximately 2550 rpm. ______

END R.M. LEVEL R3 REACTIVITY EVOLUTION

7. **Direct** Radwaste Operator to monitor effluent conductivity for each in service CDD. ______

8. **WHEN** RFPT A(B) speed is approximately 2550 rpm,
   THEN raise C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] output to match DFCS Stpt and Speed Stpt on Panel P603 to within 100 rpm. ______

NOTE

- When RFPT A(B) Man/DFCS control switch is placed in DFCS, C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] will control RFPT speed. ☐
- When RFPT A(B) Man/DFCS control switch is placed in DFCS, and DFCS is in control, the RFPT A(B) DFCS Ctrl light will be ON. ☐
- If RFPT A(B) Man/DFCS selector switch is in DFCS and DFCS control signal subsequently drops to less than 2450 rpm or rises to greater than 5450 rpm, Woodward 5009 digital controls will automatically assume RFPT speed control and maintain current pump speed. In this condition, the RFPT will only respond to Lower/Raise speed control switch commands until the Man/DFCS selector switch is placed in MAN, DFCS Ctrl Reset pushbutton is depressed, and the Man/DFCS selector switch returned to DFCS. ☐

9. **Confirm** the following RFPT A(B) speed signals on Panel P603 agree within 100 rpm:
   - DFCS Stpt (speed demand from DFCS) ______
   - Speed Stpt (speed demand from 5009 control) ______
   - Act Spd (actual RFPT speed) ______

10. **Place** Man/DFCS control switch in DFCS. ______

3.0 PRECAUTIONS AND LIMITATIONS (continued)

5. Any of the following conditions will automatically trip a reactor feed pump turbine:

   - RFPT Woodward 5009 overspeed greater than or equal to 6150 rpm .... ☐

28. 261000 1

Unit One primary containment venting is being performed IAW 1OP-10, *Standby Gas Treatment System Operating System,* with the following plant status:

1-VA-1F-BFV-RB, SBGT DW Suct Damper	Open
1-VA-1D-BFV-RB, Reactor Building SBGT Train 1A Inlet Valve	Closed
1-VA-1H-BFV-RB, Reactor Building SBGT Train 1B Inlet Valve	Closed

Which one of the following completes both statements below concerning the predicted SBGT response if drywell pressure rises to 1.9 psig?

1-VA-1F-BFV-RB ___(1)___.

Both 1-VA-1D-BFV-RB and 1-VA-1H-BFV-RB ___(2)___.

A (1) auto closes
(2) auto open

B (1) auto closes
(2) remain closed

C (1) remains open
(2) auto open

D (1) remains open
(2) remain closed

Answer: A

K/A:
261000 Standby Gas Treatment System
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
02 Suction valves

RO/SRO Rating: 3.1/3.1

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the ability to monitor SBGT suction valves.

Pedigree: Last used on 2014 NRC Exam

Objective: LOI-CLS-LP-004.1, Obj 5
List the signals and setpoints that will cause a Secondary Containment isolation

Reference: None

Cog Level: High

Explanation: The filter train fans will automatically start on High Drywell Pressure. The following actions occur: 1) SBGT Reactor Building suction dampers (1D-BFV-RB and 1H-BFV-RB) open, 2) SBGT DW Suct Damper (F-BFV-RB) closes. The SBGT Train A/B Suction & Discharge Valves on U1 do not auto open. These valves on U2 do auto open, so there could be a misconception on these valves (inlet vs. suction dampers).

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because 1F does auto close and SBGT Train 1A/B Suction Valves (1C & 1E) on Unit One only do not auto open

Choice C: Incorrect since SBGT will auto realign from primary containment to the Reactor Building on system initiation

Choice D: Incorrect since SBGT will auto realign from primary containment to the Reactor Building on system initiation and SBGT Train 1A/B Suction Valves (1C & 1E) on Unit One only do not auto open

SRO Basis: N/A

2.1.6 Fan

A 100% capacity, heavy-duty, industrial type Fan and motor assembly is provided in each SBGT filter train. Each Fan will produce the required 2700 - 3300 scfm flow through its associated filter train.

Each Fan is driven by a direct-drive AC motor which is energized from a redundant and separate emergency power supply. The Unit 1 A and B Fans are powered from 480 VAC MCCs 1XE and 1XF respectively and Unit 2 A and B Fans from 2XE and 2XF.

The filter train fans may be operated manually from controls located at RTGB XU-51.

The filter train fans will automatically start if any of the following Secondary Containment isolation conditions exist: (Figure 10-2)

1. Low Reactor Water Level, LL #2
2. High Drywell Pressure
3. Reactor Building Ventilation Radiation (Figure 10-3)

3.2.6 Automatic

1. Upon receipt of an automatic initiation signal both trains of SBGT will start.

Unit 1 ONLY

The dampers associated with Unit 1 SBGT System will receive automatic open signals when an initiation signal is received EXCEPT for the train inlet and outlet dampers, (BFVs-1B,1C,1E,and 1G). Should these normally open dampers be manually closed locally via their CLOSE/OPEN pushbuttons, **they will NOT automatically reopen and the associated SBGT will not automatically start.**

29. 262001 1

Unit One and Unit Two are both operating at rated power.

Which one of the following completes both statements below IAW **Unit One** Tech Spec 3.8.1, AC Sources - Operating, LCO statement?

The Unit Two SAT ___(1)___ required to be OPERABLE.

___(2)___ Diesel Generators are required to be OPERABLE.

A (1) is
(2) Two

B (1) is
(2) Four

C (1) is NOT
(2) Two

D (1) is NOT
(2) Four

Answer: B

K/A:
262001 A.C. Electrical Distribution
G2.2.40 Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)

RO/SRO Rating: 3.4/4.7

Tier 2 / Group 1

K/A Match: This meets the K/A because this is testing the items above the line for TS 3.8.1.

Pedigree: New

Objective: LOI-CLS-LP-050, Obj. 16
Given plant conditions, determine whether given plant conditions meet minimum Technical Specifications requirements associated with the 230 KV Electrical Distribution system.

Reference: None

Cog Level: Fundamental

Explanation: With both Units in Mode 1, both SAT's and both UAT's and all four DGs are required to be operable.

Distractor Analysis:

Choice A: Plausible because the first part is correct and there are only two Unit One DGs but all four are required for the LCO to be met.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the Unit 2 SAT and asking if it is required for Unit 1 TS and whether only the 2 Unit One DGs are required or all of the DGs.

Choice D: Plausible because this is the Unit 2 SAT and asking if it is required for Unit 1 TS and the second part is correct.

SRO Basis: N/A

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 Unit 1 qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Four diesel generators (DGs); and

c. Two Unit 2 qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

NOTE

LCO 3.0.4.b is not applicable to DGs.

30. 262002 1
Unit One is operating at rated power.

Subsequently, breaker AU9, Feed to 480V Substation E5 trips.

Which one of the following completes the statement below?

120V UPS Distribution Panel 1A is:

A de-energized.

B energized from MCC 1CB.

C energized from the Standby UPS.

D energized from 250V DC SWBD A.

Answer: D

K/A:
262002 Uninterruptable Power Supply (A.C. /D.C.)
A3 Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: (CFR: 41.7 / 45.7)
01 Transfer from preferred to alternate source

RO/SRO Rating: 2.8/3.1

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the ability to monitor the transfer to the alternate power source.

Pedigree: New

Objective: LOI-CLS-LP-052, Obj. 5
Given plant conditions, determine the lineup of the primary UPS, the Standby UPS, and their reserve sources.

Reference: None

Cog Level: High

Explanation: The UPS system is normally aligned such the primary inverter is powering UPS loads. The standby inverter is energized but bypassed with the Manual Bypass switch in Bypass Test. The static transfer switch of the Primary inverter (and also the Standby inverter) is receiving an input from the alternate (hard) source. If the primary power source is lost (in this case the loss of E5 which powers MCC CA) the alternate power source from the 250V batteries will keep the loads energized with no need for the inverter to swap to the hard source.

Distractor Analysis:

Choice A: Plausible because the normal power source is lost.

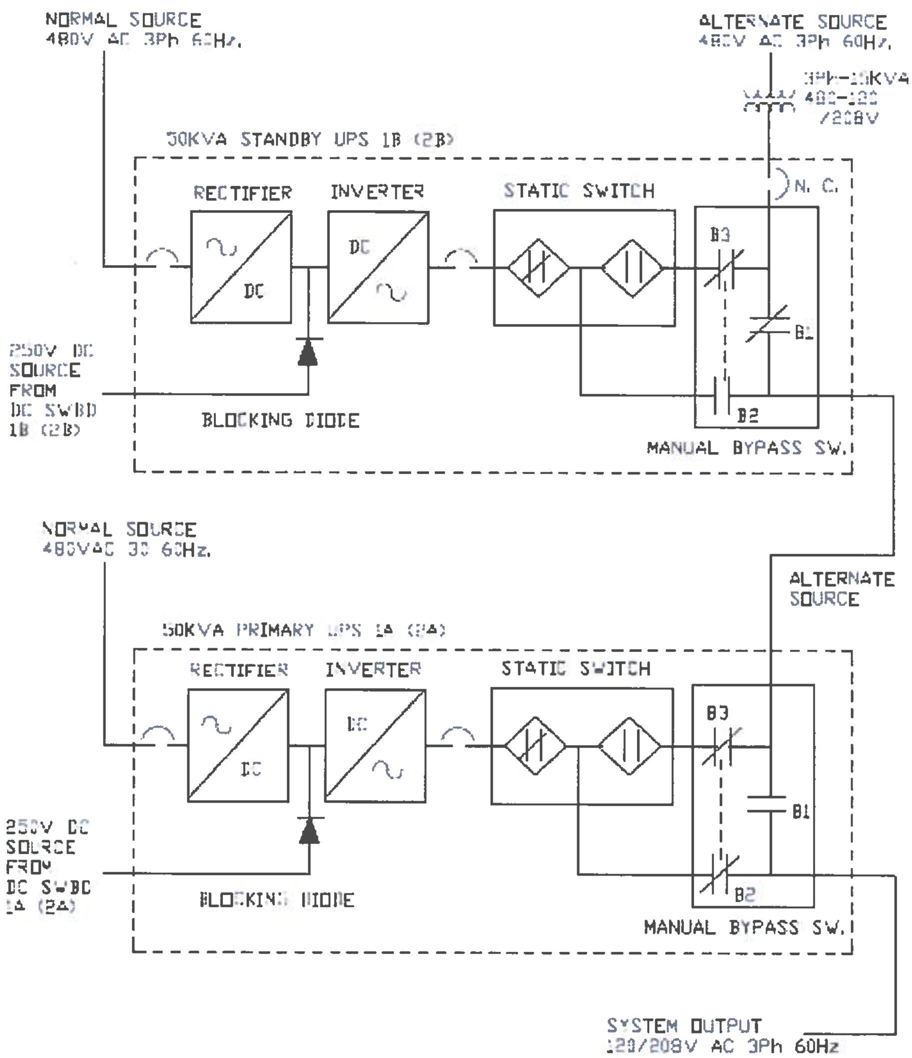
Choice B: Plausible because this is the hard source for the Distribution Panel.

Choice C: Plausible because this is an available power source for the Distribution Panel.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

FIGURE 52-7
Basic Vital UPS System



31. 263000 1

Unit Two is operating at full power when a loss of DC Distribution Panel 4B occurs.

Which one of the following completes both statements below?

RCIC is ____(1)____ for injection from the RTGB.

RCIC ____(2)____ isolation logic has lost power.

A (1) available
  (2) inboard

B (1) available
  (2) outboard

C (1) unavailable
  (2) inboard

D (1) unavailable
  (2) outboard

Answer: D

K/A:

263000 D.C. Electrical Distribution

G2.2.37 Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating: 3.6/4.6

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the operability of RCIC/ADS.

Pedigree: New

Objective: LOI-CLS-LP-051, Obj. 7
Given plant conditions, determine the effect that a loss of DC power will have on the following:
d. Reactor Core Isolation Cooling.
e. Automatic Depressurization System.

Reference: None

Cog Level: High

Explanation: RCIC initiation, trip logic, governor control and outboard isolation logic is powered from Division II 125 VDC panels 4B for unit 2. Without this power source RCIC cannot start either automatically or manually and Div II isol valves will not auto isolate

Distractor Analysis:

Choice A: Plausible because the student may believe that the logic has a backup power supply or confuse the HPCI and RCIC power supplies. Part 2 is plausible because HPCI inboard isolation logic would be inoperable.

Choice B: Plausible because the student may believe that the logic has a backup power supply or confuse the HPCI and RCIC power supplies. Part 2 is correct, see explanation.

Choice C: Part 2 is correct, see explanation. Part 2 is plausible because HPCI inboard isolation logic would be inoperable.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

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<< Plant Effects from Loss of DC Distribution Panel 3B(4B) >>

RCIC: Will **NOT** auto initiate, outboard isolation logic INOPERABLE (E51-F008, -F029, and -F066 will **NOT** auto close), RCIC turbine will **NOT** trip except on overspeed, RCIC flow controller and EGM INOPERABLE (no flow control or indication), E51-F045 will **NOT** auto close on high water level, E51-F004, -F054, and -F026 fail closed. RCIC isolation is required in accordance with APP 1(2)-A-03 1-4.

32. 264000 1
Unit Two has lost off-site power.
DG3 started and tied to its respective E Bus.
Sequence of events:

1200 DG3 ties to E3
1205 DG3 lube oil temperature rises above 190°F
1206 DG3 lube oil pressure drops below 27 psig

Which one of the following identifies when DG3 will trip?

A Immediately at 1205.

B Immediately at 1206.

C 45 seconds after 1205.

D 45 seconds after 1206.

Answer: B
K/A:
264000 Emergency Generators (Diesel/Jet)
K6 Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET): (CFR: 41.7 / 45.7)
03 Lube oil pumps

RO/SRO Rating: 3.5/3.7

Tier 2 / Group 1

K/A Match: This meets the K/A because it is testing the effect of a loss of lube oil on the EDG.

Pedigree: Bank

Objective: LOI-CLS-LP-039, Obj. 4a
Given plant conditions, determine if EDGs will trip: After an auto start (LOCT)

Reference: None

Cog Level: High

Explanation: Hi lube oil temperature bypassed by auto start signal (LOOP and LOCA). Low lube oil pressure trip never bypassed. On a start of the DG the low lube oil trip is bypassed for 45 seconds.

Distractor Analysis:

Choice A: Plausible because hi lube oil temperature is a trip, but it is bypassed on the LOOP.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because there is a 45 second time delay associated with the lube oil trip on an initial start of the EDG.

Choice D: Plausible because there is a 45 second time delay associated with the lube oil trip on an initial start of the EDG.

SRO Basis: N/A

3.3.3 Automatic Stop Control (Figure 39-14)

Under conditions where continued Diesel Generator operation may cause damage to the Diesel itself automatic shutdowns are provided. The shutdown signals will vary dependent upon whether the engine has been started manually or automatically.

When operating due to receipt of an automatic start signal the following trips and lockout are provided:

- Low lube oil pressure — 27 psig
- Overspeed — 575 (561 to 589) rpm

When operating as a result of an initiation from a normal non-emergency start the following trips and lockouts are enforced in addition to those listed above:

- High lube oil temperature — 190°F
- High jacket water temperature — 200°F
- Jacket Water Low pressure — 12 psig

The low lube oil pressure, and low jacket water pressure shutdowns are blocked for the first forty-five second on initiation of an engine start sequence (auto or manual). This permits the conditions to be established which will prevent these shutdowns during engine operation

33. 272000 1
Unit Two is performing a startup IAW 0GP-02, *Approach to Criticality and Pressurization of the Reactor*.

IAW 0GP-02, which one of the following identifies the radiation monitor(s) that will require the alarm setpoints raised when HWC is placed in service?

A D12-RM-K603A,B,C,D, Main Steam Line Rad Monitors

B ARM Channel 2-9, U-2 Turbine Bldg Breezeway

C D12-RR-4599-1,2,3, Main Stack Rad Monitors

D ARM Channel 2-4, Cond Filter-Demin Aisle

Answer: A
K/A:
272000 Radiation Monitoring System
K5 Knowledge of the operational implications of the following concepts as they apply to RADIATION MONITORING SYSTEM: (CFR: 41.7 / 45.4)
01 Hydrogen injection operation's effect on process radiation indications

RO/SRO Rating: 3.2/3.5

Tier 2 / Group 2

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because this is testing the operational implication as to which rad monitor, if asked as the operational effect on the individual rad monitor this would provide no discrimatory value.

Pedigree: New

Objective: LOI-CLS-LP-059, Obj. 8
Explain why Chemistry must be notified when starting and securing the HWC System.

Reference: None

Cog Level: Fundamental

Explanation: The excess Hydrogen injected into the reactor coolant creates the driving force to shift the Nitrogen-16 distribution ratio, resulting in a larger fraction of the Nitrogen-16 forming volatile Ammonia and a smaller fraction forming Nitrites and Nitrates. This additional volatile Ammonia is then carried over in the reactor steam resulting in higher background radiation levels. Any increase in Hydrogen injection rates will result in a proportional increase in background radiation levels and vise-versa.
0GP-02 has a step for ensuring that the rad monitors are adjusted based on this background rad level increase.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because when HWC is placed in service the rad levels will increase minimally and HWC H2 is injected in the reactor feed pumps.

Choice C: Plausible because sufficient decay time is available for N-16 such that radiation levels wouldn't raise that much in this area.

Choice D: Plausible because when HWC is placed in service the rad levels will increase minimally and this is downstream of the HWC O2 injection point.

SRO Basis: N/A

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3.0 PRECAUTIONS AND LIMITATIONS (continued)

15. B21-F032A and B21-F032B (Feedwater Supply Line Isolation Valves), are stop-check valves. These valves are designed to prevent leakage from the reactor into the feedwater system. These valves are not designed to positively close against condensate system pressure. As such, with the reactor depressurized and the condensate system in service, these valves may leak by, causing reactor water level to rise ☐

16. The Main Steam Line Radiation Monitor (MSLRM) High-High Radiation setpoint is adjusted assuming HWC is in service. If HWC is removed from service for an extended period of time (greater than one week), 1(2)OP-59, Hydrogen Water Chemistry System Operating Procedure requires BESS determine if a MSLRM High-High Radiation setpoint adjustment is required ☐

17. The HWC System will normally be placed in service immediately after establishing the following conditions:

 - At least one Condensate Booster Pump feeding the reactor with minimum flow valve closed ☐

 - At least one SJAE operating at greater than or equal to half-load ☐

34. 286000 1
Which one of the following identifies the distribution system that provides the normal power supply to the Unit Two Reactor Building Fire Alarm Control Panel?

A 48 VDC

B 120 VAC

C 125 VDC

D 480 VAC

Answer: B
K/A:
286000 Fire Protection System
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
03 Fire detection system

RO/SRO Rating: 3.6/3.8

Tier 2 / Group 2

K/A Match: This meets the K/A because this is testing the power supply to fire detection.

Pedigree: New

Objective: LOI-CLS-LP-042, Obj. 7
Identify the electrical distribution system which supplies power to the fire/smoke detection circuits.

Reference: None

Cog Level: Fundamental

Explanation: The Detection Systems control panels are powered from 120 VAC (reference Table 42-1) and convert the AC power to DC power for the rest of the system. The Detection Systems control panels are powered from 120 VAC; except Caswell Beach which is 125 VDC.

Distractor Analysis:

Choice A: Plausible because the plant has a 24/48 VDC system and the panel converts the incoming power supply to a DC source.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the Caswell Beach fire detection is supplied from 125 VDC.

Choice D: Plausible because 480 VAC could easily be thought to supply the power based on DG supply in emergencies.

SRO Basis: N/A

TABLE 42-1

Fire Detection Control Panel Power Supplies

COMPONENT	POWER SUPPLY
Control Building Fire Alarm Control Panel	120 VAC Distribution Panel 2G-CB
Radwaste Building Fire Alarm Control Panel	120 VAC Distribution Panel DRWD
Unit 2 Reactor Building Fire Alarm Control Panel	120 VAC Distribution Panel 2C-RX
Unit 1 Reactor Building Fire Alarm Control Panel	Emergency 120 VAC Distribution Panel 1B-RX
Unit 2 Turbine Building Fire Alarm Control Panel	120 VAC Distribution Panel 2C-TB2
Unit 1 Turbine Building Fire Alarm Control Panel	120 VAC Distribution Panel 1C-TB2

35. 295001 1

Unit One is operating at 70% power when the OATC observes indications for a failed jet pump. Subsequently, Recirc Pump 1A trips.

Which one of the following completes both statements below IAW 1AOP-04.0, *Low Core Flow*?

Performance of the jet pump operability surveillance for ___(1)___ Loop Operation is required.

If it is determined that a jet pump has failed, the required action is to ___(2)___.

A (1) Single
(2) reduce reactor power below 25% rated thermal power

B (1) Single
(2) commence unit shutdown IAW 0GP-05, Unit Shutdown

C (1) Two
(2) reduce reactor power below 25% rated thermal power

D (1) Two
(2) commence unit shutdown IAW 0GP-05, Unit Shutdown

Answer: B

K/A:
295001 Partial or Complete Loss of Forced Core Flow Circulation
G2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)

RO/SRO Rating: 3.7/4.1

Tier 1 / Group 1

K/A Match: This meets the K/A because this is testing knowledge of which surv. is required and the action if it has failed the surv.

Pedigree: New

Objective: LOI-CLS-LP-302-C, Obj 4
Given plant conditions and AOP-04.0, determine the required supplementary actions.

Reference: None

Cog Level: High

Explanation: The indications given are for a failed jet pump which IAW the AOP require the surveillance performed for determination of a failed jet pump. Unlike the selection of the power to flow map the PT only looks at the recirc pumps for determination of single loop or two loop operation. The power to flow maps for single loop are not used until the APRM setpoint adjustments are made.

Distractor Analysis:

Choice A: Plausible because single loop is correct and 25% is the requirement for when the PT is required to be performed.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because APRM setpoint adjustments have not been made which is a determination of how to use the power to flow maps and 25% is the requirement for when the PT is required to be performed.

Choice D: Plausible because APRM setpoint adjustments have not been made which is a determination of how to use the power to flow maps and the second part is correct.

SRO Basis: N/A

4.2 Supplementary Actions (continued)

NOTE

Jet pump failure is indicated by the following: ☐

- Reduction in generator megawatt output on GEN-WMR-760 (Net Unit Megawatts)
- Reduction in core thermal power
- Rise in indicated total core flow on B21-R613 (Core Δ Pressure/Core Flow) recorder
- Reduction in core plate differential pressure on B21-R613 (Core Δ Pressure/Core Flow) recorder
- Rise in recirculation loop flow in the loop with a failed jet pump on B32-R614 (Recirculation Flow) recorder

CAUTION

Under conditions of jet pump failure, indicated core flow on Process Computer Point U2CPWTCF and B21-R613 (Core Δ Pressure/Core Flow) recorder, will **NOT** be accurate. Accurate core flow is available from Process Computer Point U2NSSWDP (Core Plate Differential Pressure) or Attachment 1, Estimated Total Core Flow vs. Core Support Plate Delta P for B2C22. Until Step 23.b(1), the operating point on the Power-to-Flow Map will **NOT** be accurate. Indicated total core flow on B21-R613 (Core Δ Pressure/Core Flow) recorder will continue to be inaccurate until the failed jet pump is repaired. ☐

23. **IF** jet pump failure is suspected,
THEN perform the following:

    a. **IF** reactor power is greater than or equal to 25%,
    THEN ensure the following:

    - 0PT-13.1, Reactor Recirculation Jet Pump Operability, is performed for two loop operation ☐

    OR

    - 0PT-13.4, Reactor Recirculation Jet Pump Operability for Single Loop Operation, is performed for single loop operation ☐

b. **<u>IF</u>** any jet pump is determined to be INOPERABLE, **<u>THEN</u> perform** the following:

(1) **Ensure** the input to the Power-to-Flow Map has been changed from WTCF to core plate differential pressure. .... ☐

(2) **Notify** the Duty Reactor Engineer the input to the Power-to-Flow Map has been changed from WTCF to core plate differential pressure. .... ☐

(3) **Commence** unit shutdown in accordance with <u>0GP-05</u>, Unit Shutdown, in compliance with Technical Specification 3.4.2. .... ☐

36. 295003 1

Unit One is operating at rated power.
The load dispatcher reports degraded grid conditions with the following indications:

Generator frequency	59.7 hertz
230 KV Bus 1A voltage	205 KV
230 KV Bus 1B voltage	205 KV
E1 voltage	3690 volts
E2 voltage	3685 volts

Which one of the following completes both statements below?

The ____(1)____ may be damaged with continued operation under these conditions.

IAW 0AOP-22.0, *Grid Instability,* a reactor scram and turbine trip ____(2)____ required.

A (1) main turbine blades
(2) is

B (1) main turbine blades
(2) is NOT

C (1) emergency bus loads
(2) is

D (1) emergency bus loads
(2) is NOT

Answer: D

K/A:
295003 Partial or Complete Loss of A.C. Power
AK1 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.8 to 41.10)
03 Under voltage/degraded voltage effects on electrical loads

RO/SRO Rating: 2.9/3.2

Tier 1 / Group 1

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because this is testing the degraded voltage conditions.

Pedigree: Last used on the 10-2 NRC Exam

Objective: LOI-CLS-LP-302-G, Obj. 4b
Given plant conditions and any of the following AOPs, determine the required supplemental actions: AOP-22.0, Grid Instability

Reference: None

Cog Level: High

Explanation: There are frequency based criteria in AOP-22.0 (Caution directly preceding step 3.2.1) for tripping the turbine to prevent resonance vibration of low pressure blades due to off frequency operation. Time limits include, 5 minute ranges and 1 minute ranges. At this current frequency, the Main Turbine can be operated indefinitely, which will not cause turbine damage. Sustained low voltage provides for higher running currents which will damage running ESF motors. Per the automatic actions section of AOP-22.0, the degraded voltage relays will actuate when emergency bus voltage has dropped below 3700 VAC for 10 seconds. This trips the Master/Slave breakers (BOP bus supply to E Buses) and the DGs start and load.

Distractor Analysis:

Choice A: Plausible because turbine blade damage can occur due to off frequency operation and AOP-22.0 does have a low frequency range limit to scram the reactor and trip the turbine.

Choice B: Plausible because turbine blade damage can occur due to off frequency operation and the second part is correct.

Choice C: Plausible because damage to E Bus loads is correct and AOP-22.0 does have a low frequency range limit to scram the reactor and trip the turbine.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

asymmetrical short circuit current of 200kA RMS.

2.4 Protective Relaying

Protective relaying is designed to isolate any faulted component or portion of the electrical system, while maintaining continuity of power to the unfaulted portion of the system. The most commonly used protective devices include:

1. Undervoltage (27 Device) Relays. Undervoltage relays actuate on a low voltage condition, and usually are time delayed to account for momentary transient conditions, such as fault clearing and bus transfers. The degraded grid voltage relays are provided with a substantially longer time delay to prevent actuation due to motor starting transients. Undervoltage relays provide a variety of protective functions including supply breaker trips and closure permissives, large motor breaker trips and closure permissives, and automatic starting of the Emergency Diesel Generators.

4.2 Abnormal Operation

4.2.1 Abnormal Frequency Conditions

When system frequency reaches 59.8 hertz, Annunciator, UA-06, window 1-2, "GEN BUS UNDER FREQ RELAY" is activated. Operators are directed to respond per AOP-22.0, Generator Abnormal Frequency Conditions. This is done to stabilize loads on the system. One of the most probable causes of an under frequency condition would be the loss of another large generating unit, or units, when the on-line reserve capacity is inadequate for current system loads. Rapid response and close coordination with the load dispatcher are required to ensure system stability.

Abnormal frequency operation can develop resonant frequencies that may induce vibrations in the low pressure turbine blades. The vibration can cause turbine blades to fatigue and possibly fail during operation. The effect increases proportionally in relation to the magnitude of the frequency difference, and the length of time at the abnormal frequency.

3.0 AUTOMATIC ACTIONS

1. IF emergency bus voltage has lowered to less than 3700 volts (approximately equal to BOP bus voltage) for greater than 10 seconds,
   THEN the master/slave breakers to the E bus open and associated diesel generator starts and loads .... ☐

4.2 Supplementary Actions

NOTE

A sudden rise in system frequency may be observed due to additional generation or load shedding. Automatic load shedding (10% of system load) occurs at each of the following frequencies: 59.3, 59.0, and 58.5 Hz. .... ☐

CAUTION

The maximum allowable time at a given frequency is as follows: .... ☐

- Below 58.1 Hz, operation is prohibited
- Between 58.1-58.5 Hz, operation for 1 minute is allowed
- Between 58.6-59.3 Hz, operation for 5 minutes is allowed
- Between 59.4-60.6 Hz, operation is allowed indefinitely
- Between 60.7-61.4 Hz, operation for 5 minutes is allowed
- Between 61.5-61.9 Hz, operation for 1 minute is allowed
- Above 61.9 Hz, operation is prohibited

CAUTION

- Off-frequency operation can stimulate resonance vibration in low pressure blades. .... ☐
- A total loss of off-site power (LOOP) should be anticipated if the turbine is tripped. .... ☐
- With grid voltage or frequency unstable or grid vulnerability identified, diesel generators should **NOT** be paralleled with any E bus connected to the grid since severe load swings may occur and possibly overload the diesel generators. .... ☐

1. **IF** the maximum allowable time at a given frequency is exceeded, **THEN** perform the following:

   a. **IF** reactor power is greater than or equal to 26%, **THEN insert a manual scram**. .... ☐

   b. **Trip the main turbine**. .... ☐

   c. **IF** the unit was scrammed, **THEN enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure. .... ☐

4.2 Supplementary Actions (continued)

10. IF system frequency is high,
THEN:

    a. **Establish** communication with the Load Dispatcher. ☐

    b. **Continue** unit generation as directed by the Unit CRS AND **coordinate** with the Load Dispatcher. ☐

    c. IF tripping the turbine becomes imminent,
    THEN **rapidly reduce** power in an attempt to lower frequency to less than 60.7 Hz prior to tripping the main turbine. ☐

11. IF notified by the Load Dispatcher system voltage is unable
OR will be unable to support a LOCA,
OR abnormal frequency conditions persist,
THEN **follow** the guidelines in 0OI-01.01, BNP Conduct of Operations Supplement. ☐

12. IF any diesel generator is loaded to an E bus connected to the grid,
THEN **restore** the diesel generator to standby in accordance with applicable procedures. ☐

13. IF system voltage is less than 3700 volts for greater than 10 seconds,
THEN **ensure**:

    - The affected E bus master/slave breakers OPEN ☐
    - The affected diesel generator starts and loads ☐

37. 295004 1
Which one of the following completes both statements below?

IAW 0AOP-39.0, *Loss of DC Power*, before 125 VDC battery voltage reaches __(1)__, remove loads as directed by the Unit CRS.

IAW 1EOP-01-SBO, *Station Blackout*, if either division battery chargers can NOT be restored within ___(2)___ then load strip the affected battery.

A (1) 105 volts
(2) 1 hour

B (1) 105 volts
(2) 2 hours

C (1) 129 volts
(2) 1 hour

D (1) 129 volts
(2) 2 hours

Answer: A
K/A:
295004 Partial or Complete Loss of D.C. Power
AK2 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: (CFR: 41.7 / 45.8)
01 Battery charger

RO/SRO Rating: 3.1/3.1

Tier 1 / Group 1

K/A Match: This meets the K/A because this is testing knowledge of the relationship between the loss of DC power and time requirement to re-energize the battery charger.

Pedigree: New

Objective: LOI-CLS-LP-051, Obj. 14
Describe the consequences/problems associated with the following: a. Battery chargers remaining out of service during a loss of off-site power / station blackout.

Reference: None

Cog Level: Fundamental

Explanation: AOP-39.0 directs to load strip before reaching 105 VDC to prevent cell reversal. The alarm for undervoltage comes in at 129 VDC. The station Blackout procedure states that if the battery charger is not energized in 1 hour to load strip the batteries. There is a time critical 2 hour action in the SBO procedure for opening the Reactor Building doors.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct and the second part is a time critical action time limit in the SBO procedure.

Choice C: Plausible because 129 volts is the annunciator setpoint for the batteries and the second part is correct.

Choice D: Plausible because 129 volts is the annunciator setpoint for the batteries and the second part is a time critical action time limit in the SBO procedure.

SRO Basis: N/A

LOSS OF DC POWER	0AOP-39.0
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4.2 **Supplementary Actions**

1. Loss of Battery Chargers:

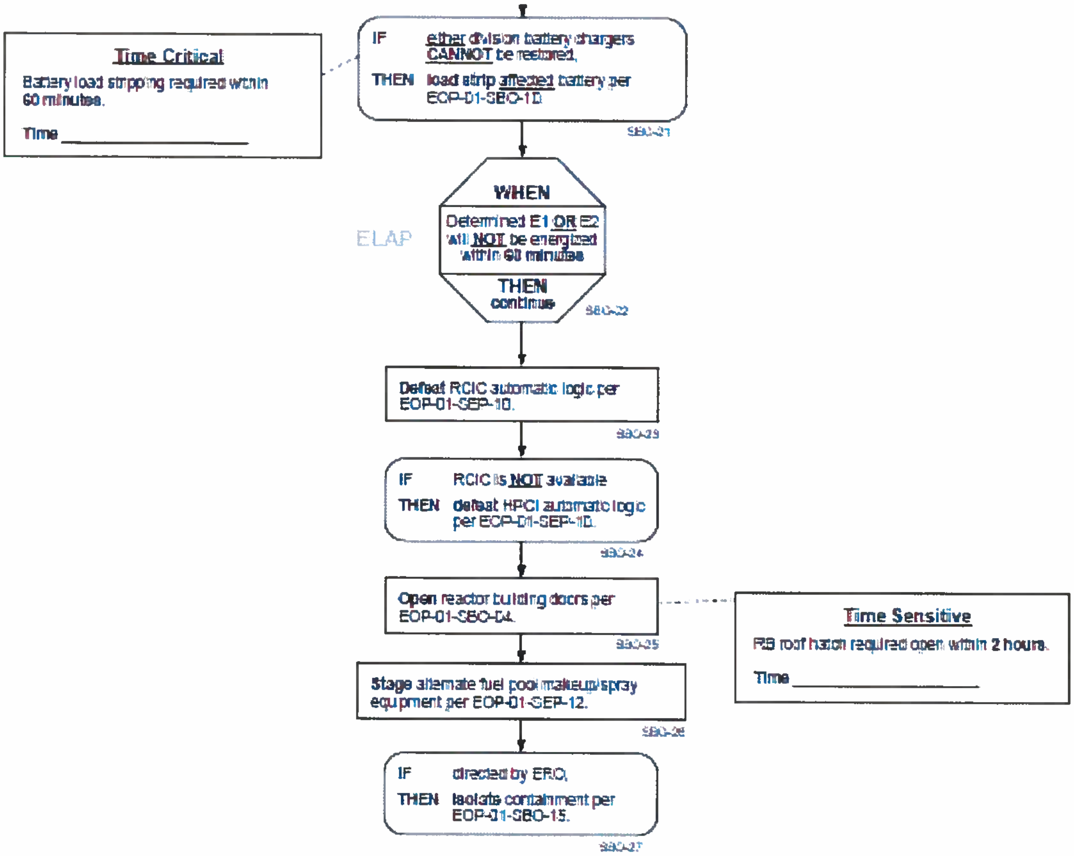
   a. **Monitor** 125V and 24V DC battery voltages. .... ☐

   b. **IF** power has been removed from the battery chargers for greater than 1 hour,
   THEN remove selected loads from the battery based on 0OI-50, 125/250 and 24/48 VDC Electrical Load List and Unit CRS direction. .... ☐

   c. Before 125V DC battery voltage reaches the low voltage limit of 105 volts, **remove** loads as directed by the Unit CRS as necessary to maintain battery voltage greater than 105 volts. .... ☐

   d. Before 24V battery voltage reaches the low voltage limit of 21 volts, **remove** loads as directed by the Unit CRS as necessary to maintain battery voltage greater than 21 volts. .... ☐

   e. **IF** battery charger AC power has been lost due to Station Blackout,
   THEN enter 1EOP-01-SBO(2EOP-01-SBO), Station Blackout. .... ☐


Time Critical
Battery load stripping required within 60 minutes.
Time ________________
IF either division battery chargers CANNOT be restored,
THEN load strip affected battery per EOP-01-SBO-10.
SBO-21
WHEN
Determined E1 OR E2 will NOT be energized within 60 minutes
THEN
continue
SBO-22
ELAP
Defeat RCIC automatic logic per EOP-01-SEP-10.
SBO-23
IF RCIC is NOT available
THEN defeat HPCI automatic logic per EOP-01-SEP-10.
SBO-24
Open reactor building doors per EOP-01-SBO-04.
SBO-25
Time Sensitive
RB roof hatch required open within 2 hours.
Time ________________
Stage alternate fuel pool makeup/spray equipment per EOP-01-SEP-12.
SBO-26
IF directed by ERO,
THEN isolate containment per EOP-01-SBO-15.
SBO-27

38. 295005 1

Which one of the following identifies the reason an operator is directed to trip the main turbine as an immediate action IAW 0AOP-32.0, Plant Shutdown From Outside Control Room?

A To initiate a scram on TSV/TCV closure.

B To prevent cold start challenges to Diesel Generators.

C The turbine cannot be tripped once the Control Room is evacuated.

D To bring bypass valves into operation until Remote Shutdown Panel control is established.

Answer: B

K/A:
295005 Main Turbine Generator Trip
AK3 Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.5 / 45.6)
04 Main generator trip

RO/SRO Rating: 3.2/3.2

Tier 1 / Group 1

K/A Match: This question requires the operator to have knowledge of the reason for turbine/generator trip. AOP-32 was used to include plausibility of distractors.

Pedigree: Bank

Objective: LOI-CLS-LP-302E, Obj. 6
Given plant conditions and entry into 0AOP-32.0, Plant Shutdown From Outside Control Room, explain the basis for a specific caution, note, or series of procedure steps.

Reference: None

Cog Level: Fundamental

Explanation: Following a reactor scram, the turbine control valves throttle shut in an effort to control RPV pressure at the setpoint of 928 psig. Without operator action, the turbine control valves will fully close, causing the generator to motor. Reverse power on the generator will cause a generator primary lockout and auto start of the diesel generators. The main turbine is therefore manually tripped to prevent it from automatically tripping on generator reverse power. This also reduces the number of cold start demands on the diesel generators.

Distractor Analysis:

Choice A: Plausible because a reactor scram is inserted as a step in the AOP, but it is performed earlier.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the procedure states to perform the step prior to exiting the control room but it could still be done at the turbine front standard.

Choice D: Plausible because this would allow use of the bypass valves, but MSIVs are manually closed prior to leaving the control room. This brings SRVs into operation. If MSIVs are not closed prior to leaving the control room, RPS EPA breakers are opened prior to establishing control at Remote Shutdown panel, which would close MSIVs.

SRO Basis: N/A

5.2 Step RSP-2

Perform scram immediate actions.

RSP-2

Step RSP-2 includes the potential for multiple sensor and sensor relay failures in the automatic RPS logic where an automatic reactor scram should have initiated but did not. If needed a manual scram is inserted to accomplish an automatic action which should have taken place. A manual reactor scram is also required when directed from other EOPs and no condition exists which would have automatically initiated a reactor scram (e.g., entry from PCCP because of high torus temperature).

Step RSP-2 also addresses other Reactor Operator scram immediate actions and includes:

- ARI initiation is an additional means of inserting control rods if needed.

- Placing the reactor mode switch to shutdown. When the reactor mode switch is placed in SHUTDOWN position, a diverse and redundant reactor scram signal is generated by the RPS logic. If the mode switch is taken out of RUN prior to RPV pressure decreasing to 835 psig, the MSIV closure due to low main steam line pressure is prevented.

  For Unit 2 only, if the mode switch is taken out of RUN when steam flow is above 33%, the MSIVs will close. Therefore, for Unit 2 the mode switch is placed in SHUTDOWN after steam flow is below $3x10^{6}$ lb/hr.

- Following a reactor scram, the turbine control valves throttle shut in an effort to control RPV pressure at the setpoint of 928 psig. Without operator action, the turbine control valves will fully close, causing the generator to motor. Reverse power on the generator will cause a generator primary lockout and auto start of the diesel generators. The main turbine is therefore manually tripped to prevent it from automatically tripping on generator reverse power. This also reduces the number of cold start demands on the diesel generators.

39. 295006 1

Unit One has entered RSP with the following conditions:

Six control rods are at position 02, all others are fully inserted
B Recirc Pump has tripped

Which one of the following completes both statements below?

The control rods will be inserted by ___(1)___ IAW 0EOP-01-LEP-02, *Alternate Control Rod Insertion*.

After the control rods are inserted, a CRD flow rate of approximately ___(2)___ will be established.

A (1) placing the individual scram test switches to the Scram position
(2) 30 gpm

B (1) placing the individual scram test switches to the Scram position
(2) 45 gpm

C (1) driving rods using RMCS
(2) 30 gpm

D (1) driving rods using RMCS
(2) 45 gpm

Answer: C

K/A:
295006 Scram
AA1 Ability to operate and/or monitor the following as they apply to SCRAM: (CFR: 41.7 / 45.6)
06 CRD hydraulic system

RO/SRO Rating: 3.5/3.6

Tier 1 / Group 1

K/A Match: This meets the K/A because this is testing operation of CRD controls after a scram.

Pedigree: new

Objective: LOI-CLS-LP-300-C, Obj. 10
Given plant conditions and the Reactor Scram Procedure, determine the required operator actions

Reference: None

Cog Level: High

Explanation: Even if the reactor will remain shutdown under all conditions without boron the LEP is used to insert the control rods using RMCS. If more control rods were out then the scram test switches would be an option. If a recirc pump is tripped then CRD flow is set to 30 gpm to minimize the stratification in the bottom head region.

Distractor Analysis:

Choice A: Plausible because this is an option used to insert the control rods in the LEP. The second part is correct.

Choice B: Plausible because this is an option used to insert the control rods in the LEP. The second part is the nominal setting for the CRD flowrate.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and the second part is the nominal setting for the CRD flowrate.

SRO Basis: N/A

7. **WHEN** either:

- All control rods in .... ☐ RO
- Only one control rod **NOT** fully inserted .... ☐ RO
- **NO** more than 10 control rods withdrawn to position 02 **AND** **NO** control rod withdrawn beyond position 02 .... ☐ RO
- Reactor engineering has determined the reactor will remain shutdown under all conditions without boron .... ☐ RO

THEN perform Section 2.2, Control Rod Insertion Verification on Page 7. .... ☐ RO

10. **IF** any control rod **NOT** fully inserted, **THEN insert** control rods.

   a. **Record** in Control Room log the control rod number and position of any rods **NOT** fully inserted. .... ________ RO

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2.2.3 Control Rod Verification Actions (continued)

   b. **Bypass** RWM. .... ________ RO

   c. **Insert** control rods with Emergency Rod In Notch Override switch .... ________ RO

2.6.3 Scram Individual Control Rods Actions (continued)

10. **Unit 1 Only: Insert** control rods with individual scram test switches:

    a. **Identify** any control rod **NOT** inserted to or beyond Position 00. .... ☐ RO

NOTE

- A sound powered phone jack is located on the column beside Panel XU-76 and in Panels XU-12, 58, 49 and 61 .... ☐
- The preferred sound-powered phone switchboard bus for use is Bus 1 .... ☐

    b. **Establish** communication between Panel P610 and Control Room. .... ☐ RO

NOTE

The individual scram test switch SCRAM position is down. .... ☐

    c. **Place individual scram test switch to SCRAM position** for any control rod **NOT** inserted to or beyond Position 00. .... ☐ RO

2.2.3 Control Rod Verification Actions (continued)

(3) IF both CRD pumps running,
THEN stop one CRD pump. .... ________ RO

(4) **Set** the setpoint tape on C11(C12)-FC-R600 (CRD Flow Control) to 30 gpm. .... ________ RO

> **NOTE**
>
> The actions in Section 2.2.3 Step 9.h(5) may be repeated as necessary. .... ☐

(5) **Adjust** cooling water differential pressure, CRD flow rate and drive pressure:

- C11(C12)-FC-R600 (CRD Flow Control) to maintain cooling water differential pressure between 10 and 26 psid. .... ________ RO
- IF a reactor recirculation pump is tripped,
  THEN establish a CRD flow rate of approximately 30 gpm. .... ________ RO
- IF both reactor recirculation pumps running,
  THEN establish a CRD flow rate between 30 and 60 gpm. .... ________ RO

40. 295009 1

A total loss of Unit One feedwater results in reactor water level lowering to 87 inches.
Drywell pressure is 2.1 psig.
Reactor water level is being restored with RCIC and CRD.

Which one of the following completes both statements below?

RVCP ___(1)___ required to be entered.

The expected response of the G31-F001, Inboard RWCU Isolation Valve, and the G31-F004, Outboard RWCU Isolation Valve, is that ___(2)___ should be closed.

A (1) is
(2) ONLY the G31-F004

B (1) is
(2) BOTH

C (1) is NOT
(2) ONLY the G31-F004

D (1) is NOT
(2) BOTH

Answer: B

K/A:
295009 Low Reactor Water Level
AK2 Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: (CFR: 41.7 / 45.8)
04 Reactor water cleanup

RO/SRO Rating: 2.6/2.6

Tier 1 / Group 2

K/A Match: This meets the K/A because this is testing the LL2 relationship to Group 3 (RWCU) isolation.

Pedigree: New

Objective: LOI-CLS-LP-014, Obj 8
Given plant conditions, determine if the RWCU system should have isolated, including expected changes in RWCU System components

Reference: None

Cog Level: High

Explanation: Based on conditions RVCP should be entered. By knowing the entry conditions for RVCP (2# DW pressure) this eliminates the RSP. The low level condition will isolate the F001 and F004. There are some signals that will isolate only the F004 only.

Distractor Analysis:

Choice A: Plausible because the first part is correct and some of the Group 3 signals do only close the F004.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the RSP would be entered, but there is an entry condition for RVCP (2# in the DW). Some of the Group 3 signals do only close the F004.

Choice D: Plausible because the RSP would be entered, but there is an entry condition for RVCP (2# in the DW). The second part is correct.

SRO Basis: N/A

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ATTACHMENT 1
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<< Group Isolation Checklist >>

Group 3 Isolation Signals

Signal	Tech Spec Value	Setpoint Value
Low Level 2	+101 inches	+105 inches
High Differential Flow	73 gpm	43 gpm (after 28.5 minute time delay)
Area High Temperature	150°F	140°F
Area Ventilation ΔT High	50°F	47°F
Non-Regen Hx Outlet Temp Hi	N/A	135°F
SLC Initiation	N/A	N/A
RWCU Outside Pump/Hx Rms	120°F	115°F
RWCU Differential Flow High Time Delay	30 minutes	28.5 minutes

Group 3 Isolation Valves
Control Room - RTGB - Panel H12-P601

Valve Number	Power Supply Unit 1(Unit 2)	Normal Position	Fail Position	Checked
[Note 1] G31-F001	1XC(2XC)/E1(E3)	NO	[Note 2] FAI	
G31-F004	1XDB(2XDB) [DC]	NO	FAI	

Note 1: SLC Initiation and RWCU Non-Regen Hx Outlet Temperature Hi signals do **NOT** isolate the RWCU Inlet Inboard Isolation Valve, G31-F001.

41. 295016 1

> **CAUTION**
>
> There are seven keylock *NORMAL/LOCAL* switches located on Diesel Generator 2 control panel. Six of these are located in a row. The seventh switch is located in the row above the six switches. The six switches in a row must be placed in *LOCAL* before placing the seventh switch in *LOCAL*.

Which one of the following identifies the reason the seventh switch is the **last** one to be placed in *LOCAL* while performing 0ASSD-02, *Control Building*?

To prevent a loss of DG2 caused by the:

A output breaker circuitry not being isolated from the fire area.

B lube oil control circuitry not being isolated from the fire area.

C loss of redundant power supply fuses for the output breaker circuitry.

D loss of redundant power supply fuses for the engine run control circuitry.

Answer: D
K/A:
295016 Control Room Abandonment
AK3 Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.5 / 45.6)
03 Disabling control room controls

RO/SRO Rating: 3.5/3.7

Tier 1 / Group 1

K/A Match: This meets the K/A because the six local switches remove control room controls and the seventh switch supplies an alternate power supply to the equipment.

Pedigree: Bank

Objective: LOI-CLS-LP-304, Obj. 21
Explain why the Diesel Generator NORMAL/LOCAL switches must be placed in LOCAL in a particular sequence.

Reference: None

Cog Level: Fundamental

Explanation: The six switches in a row isolate DG2 engine and generator control circuitry from the control room (the fire area) since a fire induced fault in wiring in the fire area may result in loss of the DG. The seventh switch inserts redundant control power fuses to the circuitry that has been isolated in the event a fault has already resulted in blowing the normal fuses. This seventh switch must be turned last with the potentially faulted circuitry already isolated or the alternate fuses may also blow making the DG unavailable. The DG engine lockout is already tripped if the DG had been running since the operator is directed to trip the DG using emergency stop.
Of the first six switches, they include: - Diesel START/STOP (2 switches) - Diesel Governor (2 switches) - Generator Voltage Regulation (2 switches)

Distractor Analysis:

Choice A: Plausible because this is what one the first six switches are performing.

Choice B: Plausible because a loss of lube oil will prevent the DG from operating.

Choice C: Plausible because the output breaker does have an alternate power supply.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

The Governor Control At Setpoint indicator light provides a status of the DRU speed reference for the 2301A governor. The light is an indicator that the Governor Control System is ready to operate at the Setpoint "speed". During actual operation of the DG, the Governor Control At Setpoint indicator light may or may not be illuminated depending on the speed of the DG.

Voltage Adjust Switches

Two three position (RAISE-NEUT-LOWER) spring return to NEUT switches are provided per engine to permit the adjustment of voltage regulators from the local panel regardless of EDG mode of operation. The auto adjust switch is normally used.

ASSD Keylock Switches

Brass handled two-position NORM - LOCAL ASSD keylock switches on the local engine panels permit the operator to transfer control of the engine and generator to the local control panel. ASSD operations are performed when a fire exists in the plant and components required to be operated may be damaged by the fire.

These switches isolate control room controls and indications to isolate the EDG control circuitry from potential fire induced faults. There are six ASSD switches (2 for EDG run/stop controls, 2 for governor controls, and 2 for voltage regulation controls) located on each local EDG panel. When in the "ASSD" mode, operation of the Diesel engine can only be accomplished by the LOCAL EMERGENCY STOP and LOCAL EMERGENCY START pushbuttons.

In addition to the six ASSD switches, for EDG 2 and 4 only, there is a seventh ASSD switch located above the other six switches. This switch provides an alternate set of control power fuses for EDG control circuitry. This may be necessary since fire induced faults may have blown normal control fuses. When operating the ASSD switches for EDGs 2 or 4, the seventh switch **must be turned last** after the potentially faulted circuitry has been isolated to prevent blowing the alternate fuses, making the EDG unavailable to provide power to Safe Shutdown loads.

42. 295017 1

During accident conditions, the source term from the Unit One Reactor Building must be estimated. Three RB HVAC supply fans and three RB HVAC exhaust fans are running.

IAW 0PEP-03.6.1, *Release Estimates Based on Stack/Vent Readings*, which one of the following is the calculated release rate?

ATTACHMENT 2
Page 1 of 1
Source Term Calculation From #1 RX Gas (1-CAC-AQH-1264-3)

TIME	METER READING (cpm)	FLOW[1] (cfm)	EFFICIENCY[(2)] FACTOR	RELEASE[(3)] RATE (µCi/sec)
1 minute ago	4.0 E+3	43,200 CFM per exhaust fan	1.275 E-5	

(1) If not available use 43,200 cfm per exhaust fan times the number of fans operating.

(2) The efficiency factors can be obtained from 0E&RC-2020 (contact E&RC counting room).

(3) Release Rate = (cpm) x (cfm) x (Efficiency Factor)

A 2.2 E+3 µCi/sec.

B 6.6 E+3 µCi/sec.

C 1.3 E+4 µCi/sec.

D 6.6 E+4 µCi/sec.

Answer: B

K/A:
295017 High Off-Site Release Rate
AA2 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.10 / 43.5 / 45.13)
03 Radiation levels

RO/SRO Rating: 3.1/3.9

Tier 1 / Group 2

K/A Match: This meets the K/A because it is testing the source term for a release off-site.

Pedigree: Bank

Objective: LOI-CLS-LP-301A, Obj. 6
Determine data required for offsite dose projection in accordance with AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment, and PEP-03.6.1, Release Estimates Based Upon Stack/Vent Readings.

Reference: None

Cog Level: High

Explanation: Per Attachment 2 the calculated release rate is:

Meter reading (CPM) X Flow (43,200 per fan X no of discharge fans) X efficiency factor

or

(4 E+3) (43,200 X 3) (1.275 E-5) = 6.6 E+3 mCi/sec

Distractor Analysis:

Choice A: Plausible because it is the calculation without multiplying times the number of running exhaust fans.

Choice B: Correct Answer, see explanation.

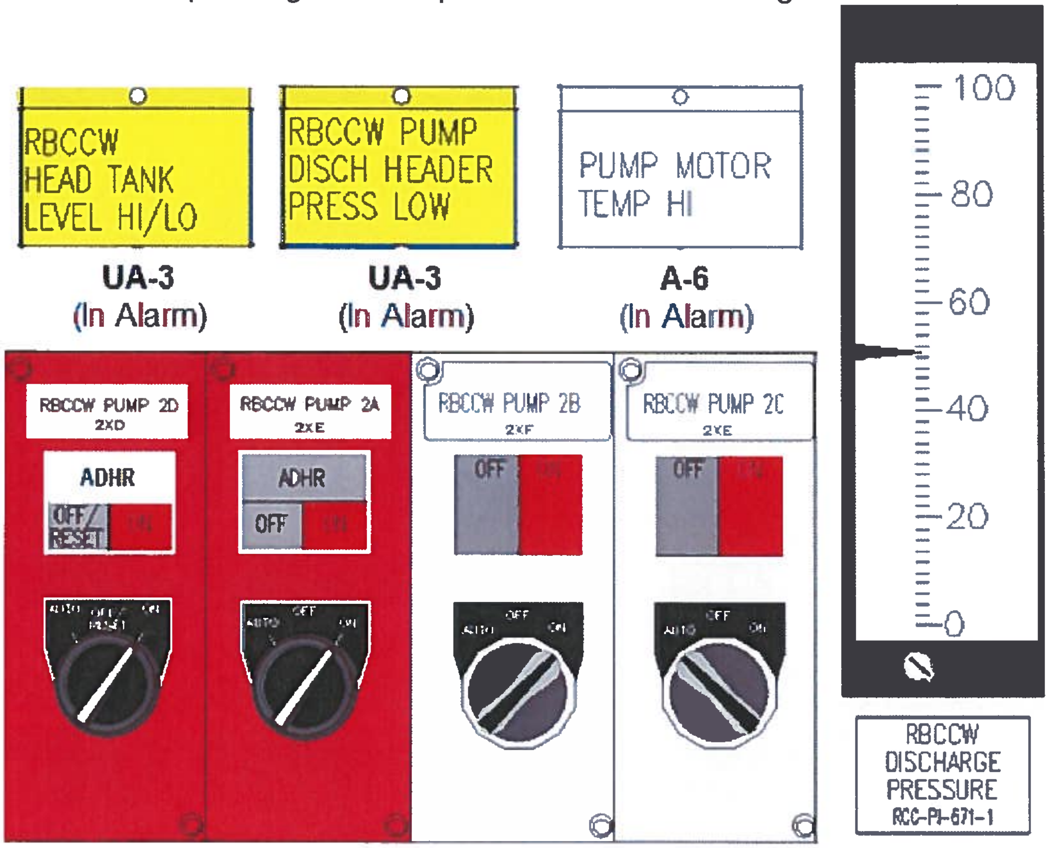
Choice C: Plausible because it uses the total number of fans running vs. the number of exhaust fans.

Choice D: Plausible because it is the correct numerical value but is off by a factor of 10.

SRO Basis: N/A

43. 295018 1

Unit Two is operating at ~65% power when the following are observed:



Which one of the following identifies the operator actions that are required IAW 0AOP-16.0, *RBCCW System Failure*?

A Commence a plant shutdown IAW 0GP-05, *Unit Shutdown*.

B Reduce system heat load by removing RWCU and Fuel Pool Cooling from service.

C Reduce Reactor power as necessary to clear the Recirc Motor high temperature alarm.

D Trip all RBCCW Pumps, insert a manual reactor scram, and trip both recirc pumps.

Answer: D

K/A:

295018 Partial or Complete Loss of Component Cooling Water

AK2 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: (CFR: 41.7 / 45.8)

02 Plant operations

RO/SRO Rating: 3.4/3.6

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the relationship of the loss of RBCCW and the actions required for plant operations

Pedigree: Last used on the 04 NRC Exam

Objective: LOI-CLS-LP-302-H, Obj. 4a
Given plant conditions, determine the required supplementary actions in accordance with the following AOPs: 0AOP-16.0, RBCCW System Failure

Reference: None

Cog Level: High

Explanation: A complete loss of RBCCW is defined as discharge header pressure below 60 psig and all available RBCCW pumps running (AOP-16.0). A complete loss requires a manual scram. FPCCU is removed only for partial loss, power reduction as necessary to maintain drywell temperature is not appropriate since a scram is required.

Distractor Analysis:

Choice A: Plausible because the unit is required to be shutdown but the procedure designates a scram not a normal unit shutdown.

Choice B: Plausible because this is an action if it is not a complete loss of RBCCW

Choice C: Plausible because this is an action if it is not a complete loss of RBCCW.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

4.2 Supplementary Actions (continued)

b. IF ADHR Mode piping is NOT the source of the leakage, THEN **re-align** RBCCW Pumps A and D from ADHR Mode to RBCCW Mode, as necessary ☐

NOTE

A complete loss of RBCCW is defined as discharge header pressure less than 60 psig, high temperature alarms on components supplied by RBCCW, and all available (no more than three) RBCCW pumps operating on the RBCCW header ☐

4. IF there is a complete loss of RBCCW, THEN: ☐

a. **Trip** all RBCCW pumps (including RBCCW Drywell HVAC Cooling Pump if operating on the affected unit and pumps operating in ADHR Mode) ☐

b. **Close** the following valves:

- RCC-V28 (RBCCW To DW Isol Vlvs) ☐
- RCC-V52 (RBCCW TO DW Isol Vlvs) ☐

c. **Trip** RWCU pump(s) ☐

d. **Isolate** RWCU System by closing the following valves:

- G31-F001 (RWCU Inboard Isol Vlv) ☐
- G31-F004 (RWCU Outboard isol Vlv) ☐

e. **Reduce** reactor power with recirc flow in accordance with 0ENP-24.5, Form 2, Immediate Reactor Power Reduction Instructions ☐

f. **Insert** a manual scram ☐

g. **Enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure, AND **perform** concurrently with this procedure ☐

h. **Trip** both reactor recirculation pumps by performing the following:

(1) **Depress** VFD A Emerg Stop ☐

(2) **Depress** VFD B Emerg Stop ☐

44. 295019 1

Unit Two has entered 0AOP-20.0, *Pneumatic (Air/Nitrogen) System Failures*, due to a loss of instrument air pressure.

Which one of the following completes both statements below?

The Diesel Generator starting air ___(1)___ affected.

The VA-2A-BFIV-RB, RB HVAC Butterfly Isolation Valve, fails ___(2)___.

A (1) is
(2) open

B (1) is
(2) as-is

C (1) is NOT
(2) open

D (1) is NOT
(2) as-is

Answer: D

K/A:

295019 Partial or Complete Loss of Instrument Air

AA2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.10 / 43.5 / 45.13)

02 Status of safety-related instrument air system loads

RO/SRO Rating: 3.6/3.7

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing status of equipment on a loss of air

Pedigree: New

Objective: LOI-CLS-LP-302K, Objective 6
Summarize the consequences associated with improper equipment operation specified in 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures

Reference: None

Cog Level: Fundamental

Explanation: A loss of instrument air will not make the DGs inoperable because they have their own dedicated air system. The BFIVs fail as-is.

Distractor Analysis:

Choice A: Plausible because the first part is correct and since air operated valves can be designed to fail open, closed or as-is.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the DGs use air for starting and since air operated valves can be designed to fail open, closed or as-is.

Choice D: Plausible because the DGs use air for starting and since air operated valves can be designed to fail open, closed or as-is.

SRO Basis: N/A

PNEUMATIC (AIR/NITROGEN) SYSTEM FAILURES	0AOP-20.0
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ATTACHMENT 2
Page 2 of 2

Components Required To Perform A Safety-Related Function After Loss of Normal Instrument Air

COMPONENT NUMBER	DESCRIPTION	LOSS OF AIR FAILED POSITION
RNA-V313	Ck Vlv To Iso N2 Backup From RNA	OPEN
RNA-V314	Ck Vlv To Iso N2 Backup From RNA	OPEN
RNA-V315	Ck Vlv To Iso N2 Backup From RNA	CLOSED
RNA-V316	Ck Vlv To Iso N2 Backup From RNA	CLOSED
VA-1(2)A-BFIV-RB	Air Operators For RB HVAC	AS IS
VA-1(2)B-BFIV-RB	Air Operators For RB HVAC	AS IS
VA-1(2)C-BFIV-RB	Air Operators For RB HVAC	AS IS
VA-1(2)D-BFIV-RB	Air Operators For RB HVAC	AS IS
DIESEL ENG 1	Diesel Generator	OPERABLE
DIESEL ENG 2	Diesel Generator	OPERABLE
DIESEL ENG 3	Diesel Generator	OPERABLE
DIESEL ENG 4	Diesel Generator	OPERABLE

45. 295020 1

I&C Techs inadvertently cause a low level 3 (LL3) signal.

Unit Two plant conditions are:

Reactor pressure	930 psig
Drywell pressure	1.7 psig, steady
Drywell temp (average)	140°F, slow rise
Drywell leak calculation	Normal

Which one of the following completes the statement below?

All Drywell Cooler Fans are:

A tripped, but can be overridden on.

B tripped, and cannot be overridden on.

C running, but can be tripped at the RTGB.

D running, and cannot be tripped at the RTGB.

Answer: A

K/A:

295020 Inadvertent Containment Isolation

AA1 Ability to operate and/or monitor the following as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.7 / 45.6)

02 Drywell ventilation/cooling system

RO/SRO Rating: 3.2/3.2

Tier 1 / Group 2

K/A Match: This meets the K/A because it is testing what the DW coolers do on an isolation signal.

Pedigree: Bank

Objective: LOI-CLS-LP-04, Obj. 20
Given plant conditions determine if the drywell coolers should auto start or trip

Reference: None

Cog Level: High

Explanation: LOCA signal on LL3 closes Group 10 which fails dampers open, but also trips fan motors. Override for LOCA trip can be performed as long as a LOCA does not really exist which is overridden in back panels (XU-27/XU-28). The low level condition also is a scram signal which provides an auto start signal for the DW Coolers which is prioritized by the trip signal.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the fans do trip and if the conditions were different they would not be able to be overridden.

Choice C: Plausible because the fans do auto start on a scram signal or usually when the dampers are opened and under different conditions they would be able to be tripped from the RTGB.

Choice D: Plausible because the fans do auto start on a scram signal or usually when the dampers are opened and under different conditions they would not be able to be tripped from the RTGB.

SRO Basis: N/A

Placing a Unit 2 Drywell Cooling Fan control switch in *START* causes the fan's discharge damper to open. WHEN the discharge damper is full open, the fan will start. The control switch should be held in the *START* position until the discharge damper is full open. The RBCCW cooling water valve to the coils will open concurrently with a fan start.

Placing the Drywell Cooling 1B Fan control switch in *START* causes the fan's discharge damper to open. WHEN the discharge damper is full open, the fan will start. The control switch should be held in the *START* position until the fan starts. The common air inlet damper and the RBCCW cooling water valve to the coils will open concurrently with a fan start.

WHEN the control switch for Drywell Cooling Fan 1A, 1C, or 1D is placed in *START*, the associated fan starts and the discharge backdraft damper opens from the fan air flow. The discharge damper position indication does not input to the start logic for Drywell Cooling Fans 1A, 1C, and 1D. The RBCCW cooling water valve to the coils will open concurrently with a fan start.

The Drywell Lower Vent dampers can be positioned to either MIN or MAX position by a two-position control switch on Panel XU-3. Normal plant operating position for these dampers is the MIN position. Placing these dampers to MAX position during plant operation may produce extreme temperature excursions in the upper drywell regions. Low scram air header pressure will reposition these dampers to the MAX position and automatically start any idle drywell cooling fan selected for AUTO.

Drywell Cooler Override Switches, VA-CS-5993/5994, are provided in Panels XU-27/28 to facilitate various modes of Drywell cooler operation as required by the EOPs.

The Pneumatic Nitrogen System or Reactor Building Non-Interruptible Instrument Air pneumatically operates the drywell cooling fans discharge dampers. These dampers will fail open on loss of pneumatics. Unit 2 and 1B drywell cooling fans discharge dampers fail closed on loss of the associated 120 VAC distribution panel.

A contactor in the associated fan's 480 VAC breaker provides drywell cooler FAN ON indication on RTGB Panel XU-3.

The drywell coolers receive a LOCA trip signal from the Core Spray initiation relays.

46. 295021 1
Unit One in MODE 5.
The fuel pool gates are removed.
SDC Loop B is in service.
Fuel pool cooling assist is in operation.

The RHR Loop B pumps tripped and can NOT be restarted.

Which one of the following completes both statements below?
(consider each statement separately)

Fuel pool cooling assist ___(1)___.

Fuel pool cooling assist ___(2)___ capable of being aligned to the SDC Loop A IAW 1OP-17, *Residual Heat Removal System Operating Procedure*.

A (1) remains in service
(2) is

B (1) remains in service
(2) is NOT

C (1) is lost
(2) is

D (1) is lost
(2) is NOT

Answer: D
K/A:
295021 Loss of Shutdown Cooling
AK2 Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following:
(CFR: 41.7 / 45.8)
05 Fuel pool cooling and cleanup system

RO/SRO Rating: 2.7/2.8

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the relationship of using SDC and the Fuel Pool.

Pedigree: New

Objective: LOI-CLS-LP-017, Obj 5
Given a drawing of the RHR system, trace the flow path for all of the six (6) modes of operation.

Reference: None

Cog Level: High

Explanation: Fuel pool cooling assist mode utilizes the B Loop of RHR so that when it is lost so too will the fuel pool cooling assist operations. If the gates were installed then the A Loop of SDC could be used with the B loop discharge flowpath, but with the gates removed this is NOT an option.

Distractor Analysis:

Choice A: Plausible because the students may think that the FPC pumps provide the motive force for this mode of operation and if the gates were installed then this would be correct.

Choice B: Plausible because the students may think that the FPC pumps provide the motive force for this mode of operation and the second part is correct.

Choice C: Plausible because the first part is correct and if the gates were installed then this would be correct.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

8.11 Fuel Pool Cooling Assist Mode With Fuel Pool Gates Removed U:

CAUTION
The following section has the potential to significantly raise area dose rates.

8.11.1 Initial Conditions Date/Time Started ____________

Initials

1. Reactor in Mode 5 with fuel pool gates removed. ____
2. Fuel pool temperature can **NOT** be maintained less than 125°F. ____
3. 0PT-08.0C has been completed satisfactorily within previous 92 days. ____
4. Fuel Pool Cooling system in operation in accordance with 1OP-13 with available fuel pool cooling heat exchangers in operation. ____
5. RHR Loop B is operating in shutdown cooling in accordance with Section 5.7 or 5.8. ____

47. 295023 1

Unit Two is performing refueling operations when the refueling SRO reports that a spent fuel bundle has been dropped in the cattle chute.
The following radiation monitoring alarms are received:

UA-03 (3-7) *Area Rad Refuel Floor High*
UA-03 (4-5) *Process Rx Bldg Vent Rad Hi*

Which one of the following identifies the "Immediate Action" that is required IAW 0AOP-05.0, *Radioactive Spills, High Radiation, and Airborne Activity*?

A Verify Group 6 isolation.

B Evacuate all personnel from the refuel floor.

C Place Control Room Emergency Ventilation System in operation.

D Isolate Reactor Building Ventilation and place Standby Gas Treatment trains in operation.

Answer: C

K/A:
295023 Refueling Accidents
AA1 Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: (CFR: 41.7 / 45.6)
04 Radiation monitoring equipment

RO/SRO Rating: 3.4/3.7

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the immediate operator actions for a radiation event.

Pedigree: Bank

Objective: LOI-CLS-LP-302J, Obj. 5
List the immediate operator actions required to be performed in accordance with 0AOP-05, Radioactive Spills, High Radiation, and Airborne Activity

Reference: None

Cog Level: Fundamental

Explanation: This is an Immediate Action identified in AOP-05.0.

Distractor Analysis:

Choice A: Plausible because this is an auto action not an immediate operator action of the AOP

Choice B: Plausible because IAW 0AOP-05.0 this is the first supplemental action.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because RBHVAC isolation and SBGT start requires *PROCESS RX BLDG VENT RAD HI-HI* (UA-03 3-5) in alarm and these are supplementary actions in the procedure.

SRO Basis: N/A

4.0 OPERATOR ACTIONS

NOTE

The following should be considered for establishment as critical parameters during performance of this procedure: .... ☐

- Area radiation levels
- Personnel habitability in the affected area

4.1 Immediate Actions

1. **IF** a fuel assembly was dropped or damaged,
   THEN ensure the Control Room Emergency Ventilation System (CREVS) is in operation. {7.1.1} .... ☐

48. 295024 1

Unit Two is operating at rated power when high drywell pressure switch C72-PTM-N002A-1 fails high resulting in the annunciation of A-05-(5-6) *Pri Ctmt Press Hi Trip*.

Which one of the following completes the statement below?

RPS high drywell pressure relay C72-K4A will ___(1)___ causing a ___(2)___ scram.

A (1) energize
  (2) half

B (1) energize
  (2) full

C (1) de-energize
  (2) half

D (1) de-energize
  (2) full

Answer: C

K/A:
295024 High Drywell Pressure
EA1 Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.7 / 45.6)
05 RPS

RO/SRO Rating: 3.9/4.0

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the ability to monitor RPS (half scram condition) for a high DW pressure condition

Pedigree: New

Objective: LOI-CLS-LP-003, Objectives:
7.g Given plant conditions state the Normal, Initiation, and Fail position/condition of the following components: (Open/Closed Energized/De Energized) RPS Logic
9. Given any scram signal, describe the logic arrangement for the signal including what combination of signals will cause a Full Scram.

Reference: None

Cog Level: High

Explanation: The RPS relays are de-energize to actuate and a single relay actuates the alarm and will cause a half scram.

Distractor Analysis:

Choice A: Plausible because there are logics that are energize to actuate and the half scram is correct.

Choice B: Plausible because there are logics that are energize to actuate and there are also logics that only require one instrument to actuate (Nuclear instrumentation).

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and some logics do cause a full scram (Nuclear instrumentation).

SRO Basis: N/A

PRI CTMT PRESS HI TRIP

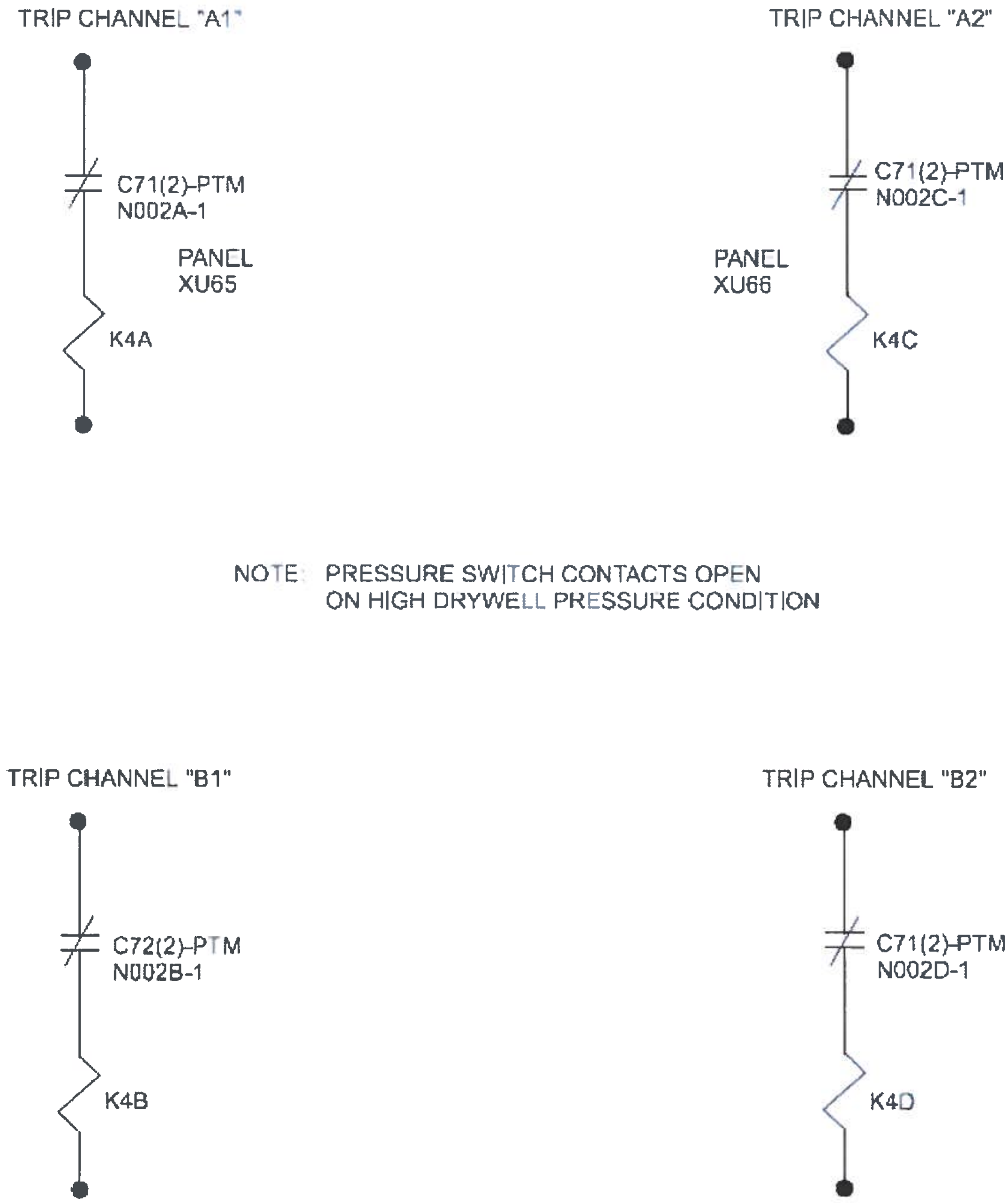
AUTOMATIC ACTIONS

1. If the primary containment pressure high trip signal is received in only one RPS Trip System, a half Scram will occur.
2. If the primary containment pressure high trip signal is received in both RPS Trip Systems, a reactor Scram will occur.

DEVICE/SETPOINTS

Relay C72-K4A-D	Deenergized
Pressure Switch C72-PTM-N002A-1, B-1, C-1, or D-1	1.7 psig

FIGURE 03-15
High Drywell Pressure Trip



49. 295025 1

Unit One was operating at power when a turbine trip occurred.
85 control rods fail to insert.
Reactor pressure peaks at 1145 psig.

Which one of the following completes both statements below?

The reactor recirc pumps ___(1)___ tripped.

Tripping of the reactor recirc pumps results in a rapid decrease in reactor power due to ___(2)___.

A (1) must be manually
  (2) voiding of the moderator

B (1) must be manually
  (2) a reduction in reactor water level

C (1) have automatically
  (2) voiding of the moderator

D (1) have automatically
  (2) a reduction in reactor water level

Answer: C

K/A:
295025 High Reactor Pressure
EK3 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: (CFR: 41.5 / 45.6)
02 Recirculation pump trip

RO/SRO Rating: 3.9/4.1

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the reason the recirc pump is tripped.

Pedigree: Bank

Objective: LOI-CLS-LP-002, Obj. 30
Given Plant conditions determine if the ATWS-RPT protection logic should have actuated

Reference: None

Cog Level: Fundamental

Explanation: The Anticipated Transient Without Scram circuit provides an alternate means of reducing reactor power in the unlikely event that the control rods fail to insert into the core following a Reactor Protection System actuation signal. Tripping of the VFD Input Circuit Breakers (ICB) will rapidly reduce recirculation flow. This results in a rapid decrease in reactor power because of the voiding of the moderator. Setpoints for ATWS trip are high reactor pressure 1137.8 psig and low reactor level LL2 105"

Distractor Analysis:

Choice A: Plausible because the ATWS procedure directs the pumps to be tripped and the second part is correct.

Choice B: Plausible because the ATWS procedure directs the pumps to be tripped and level is reduced in the ATWS procedure which does lower power.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and level is reduced in the ATWS procedure which does lower power.

SRO Basis: N/A

3.2.6 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT)

The Anticipated Transient Without Scram circuit provides an alternate means of reducing reactor power in the unlikely event that the control rods fail to insert into the core following a Reactor Protection System actuation signal. Tripping of the VFD Input Circuit Breakers (ICB) will rapidly reduce recirculation flow. This results in a rapid decrease in reactor power because of the voiding of the moderator.

Two signals are used for the initiation of ATWS-RPT. These signals are LL2 reactor vessel water level and high reactor vessel pressure. Each of these parameters is monitored by four sensors. Two level or pressure instruments in one of two logic trains are required to energize relays which trip both Recirculation Pumps.

50. 295026 1

Unit One failed to scram following a loss of off-site power with the following plant conditions:

Reactor Power	5%
RPV Water Level	-55 inches (N036)
RPV Pressure	850 psig

Which one of the following completes both statements below?

This UA-12 (5-4) alarm is expected to be received when suppression pool water temperature **first** reaches ___(1)___.

The RHR logic requirements to place torus cooling in service under the current plant conditions will require ___(2)___.

A (1) 95°F
(2) placing the "Think Switch" to Manual first and then bypassing the 2/3rd core height interlock

B (1) 95°F
(2) bypassing the 2/3rd core height interlock first and then placing the "Think Switch" to Manual

C (1) 105°F
(2) placing the "Think Switch" to Manual first and then bypassing the 2/3rd core height interlock

D (1) 105°F
(2) bypassing the 2/3rd core height interlock first and then placing the "Think Switch" to Manual

Answer: B

K/A:

295026 Suppression Pool High Water Temperature

G2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

RO/SRO Rating: 4.2/4.0

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing when the torus temperature alarm setpoint and what controls need to be operated to establish cooling.

Pedigree: New

Objective: LOI-CLS-LP-017, Obj 09
Given an RHR pump or valve, list the interlocks, permissives and/or automatic actions associated with the RHR pump or valve, including setpoints.

Reference: None

Cog Level: High

Explanation: LOCA signal is sealed in due to being less than LL3 (45 inches) RPV water level is less than 2/3rd core height (-47 inches) therefore the keylock switch and then the Think switch is required (sequencing is essential). When the torus reaches 95°F this alarm will come in, 105°F is the TMax alarm.

Distractor Analysis:

Choice A: Plausible because the first part is correct and the second part is opposite of the required actions.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the alarm setpoint for the *SPTMS DIV I BULK WTR TEMP SETPT TMAX,* and the second part is opposite of the required actions.

Choice D: Plausible because this is the alarm setpoint for the *SPTMS DIV I BULK WTR TEMP SETPT TMAX,* and the second part is correct

SRO Basis: N/A

SPTMS DIV I BULK WTR TEMP SETPOINT TS1

NOTE:	Inoperability of this annunciator may result in a TRM Required Compensatory Measure.

AUTO ACTIONS

NONE

CAUSE

1. High suppression pool bulk average water temperature.

OBSERVATIONS

1. Recorder Channel 1 on CAC-TR-4426-1A indicates increasing suppression pool temperature.
2. TS1 indicator illuminated (CAC-TY-4426-1).

ACTIONS

1. If suppression pool temperature is approaching 95°F and no testing is in progress that could add heat to the suppression pool, then refer to AOP-14.0, Abnormal Primary Containment Conditions and AOP-30.0, Safety/Relief Valve Failures.
2. If suppression pool temperature is greater than 95°F due to adding heat to the suppression pool from approved testing procedures, then refer to the appropriate test procedure to maintain suppression pool temperature below 105°F.
3. If suppression pool temperature is greater than 95°F and no testing is in progress that could add heat to the suppression pool, then enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions.
4. If a circuit or equipment malfunction is suspected, ensure that a WR/WO is prepared.

DEVICE/SETPOINTS

SPTMS Microprocessor CAC-TY-4426-1	95°F

SPTMS DIV I BULK WTR TEMP SETPT TMAX

AUTO ACTIONS

NONE

CAUSE

1. High suppression pool bulk average water temperature.

OBSERVATIONS

1. Recorder Channel 1 on CAC-TR-4426-1A indicates increasing suppression pool temperature.
2. TMAX indicator illuminated (CAC-TY-4426-1).

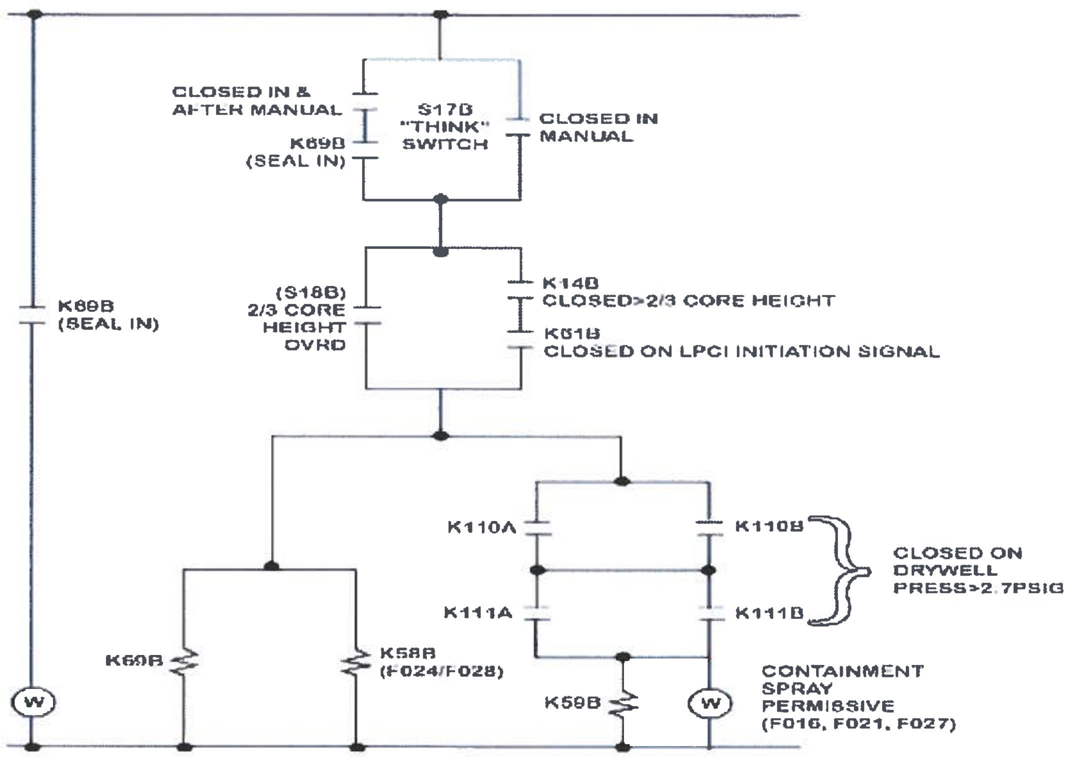
ACTIONS

1. If suppression pool temperature is greater than 95°F and no testing is in progress that could add heat to the suppression pool, then enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions, if not already entered.
2. If suppression pool temperature is approaching 105°F due to adding heat to the suppression pool from approved testing procedures, then refer to the appropriate test procedure to maintain suppression pool temperature below 105°F.
3. If suppression pool temperature is greater than 105°F, then stop all testing and enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions.
4. If a circuit or equipment malfunction is suspected, then ensure that a WR/WO is prepared.

DEVICE/SETPOINTS

SPTMS Microprocessor CAC-TY-4426-1	105°F

FIGURE 17-12
Cooling/Spray Permissive Logic



51. 295028 1

Unit Two is in MODE 3 following a Station Blackout.
IAW 0EOP-01-SBO-01, *Plant Monitoring*, the AO has reported the following temperatures from the RSDP temperature recorder 2CAC-TR-778:

Point 1	290°F
Point 2	118°F
Point 3	255°F
Point 4	230°F
Point 5	191°F
Point 6	117°F

(REFERENCE PROVIDED)

Which one of the following represents the correct calculated Drywell temperature?

A ~205°F

B ~249°F

C ~258°F

D ~267°F

Answer: B

K/A:
295028 High Drywell Temperature
EA2 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13)
01 Drywell temperature

RO/SRO Rating: 4.0/4.1

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the students ability to determine drywell temperature.

Pedigree: Bank

Objective: LOI-CLS-LP-303-B, Obj. 3
Given plant conditions, control room or remote shutdown panel indications, and SBO-04, calculate the following parameters: a. Drywell Temperature

Reference: Attachment 4 of 0EOP-01-SBO-01, Plant Monitoring

Cog Level: Fundamental

Explanation: Attachment 4 of 0EOP-01-SBO-01, Plant Monitoring, has a calculation worksheet for figuring Drywell temperature from RSDP temperature recorder readings.

$$
\begin{aligned}
290 * 0.141 &= 40.89 \\
255 * 0.404 &= 103.02 \\
230 * 0.455 &= \underline{104.65} \\
&\ 248.56
\end{aligned}
$$

Distractor Analysis:

Choice A: Plausible because this is the average of points 1 - 3 used in calculation.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because this is the average of points 1, 3, & 4.

Choice D: Plausible because this is performing the calculation backwards (points 4, 3, 1)

SRO Basis: N/A

ATTACHMENT 4
Page 1 of 1

Drywell Temperature Calculation Using RSDP Recorder Inputs

Values obtained from Recorder CAC-TR-778

Above 70' Elevation

PT 1 ___290___ x 0.141 = ___40.89___ °F

Between 28' and 45' Elevation

PT 3 ___255___ x 0.404 = ___103.02___ °F

Between 10' and 23' Elevation

PT 4 ___230___ x 0.455 = ___104.65___ °F

Average Drywell Temperature ___248.56___ °F
(Sum of 3 Regional Weighted Areas)

52. 295029 1

Unit Two is performing RVCP with HPCI in pressure control.

Subsequently, torus water level reaches -23 inches.

Which one of the following completes both statements below?

The E41-F004, CST Suction Vlv, will ____(1)____.

The E41-F008, Bypass to CST Vlv, will ____(2)____.

A (1) close
(2) close

B (1) close
(2) remain open

C (1) remain open
(2) close

D (1) remain open
(2) remain open

Answer: A

K/A:
295029 High Suppression Pool Water Level
EA1 Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: (CFR: 41.7 / 45.6)
01 HPCI

RO/SRO Rating: 3.4/3.5

Tier 1 / Group 2

K/A Match: This meets the K/A because it is testing operation of HPCI on high torus level

Pedigree: New

Objective: LOI-CLS-LP-019. Obj. 3p
Given plant conditions, predict how the HPCI System will respond to the following events: High/low Suppression Pool water level

Reference: None

Cog Level: High

Explanation: The torus water high level condition (> -25 inches) will cause the torus suction valves to open. When either valve is full open the F008 and F004 will close.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct. The F008 will get a close signal when the F041 or F042 is full open.

Choice C: Plausible because the high level does not directly close the F008 valve. The second part is correct.

Choice D: Plausible because the high level does not directly close the F004 or F008 valve.

SRO Basis: N/A

FIGURE 19-7
CST Suction Valve, E41-F004, Control Logic

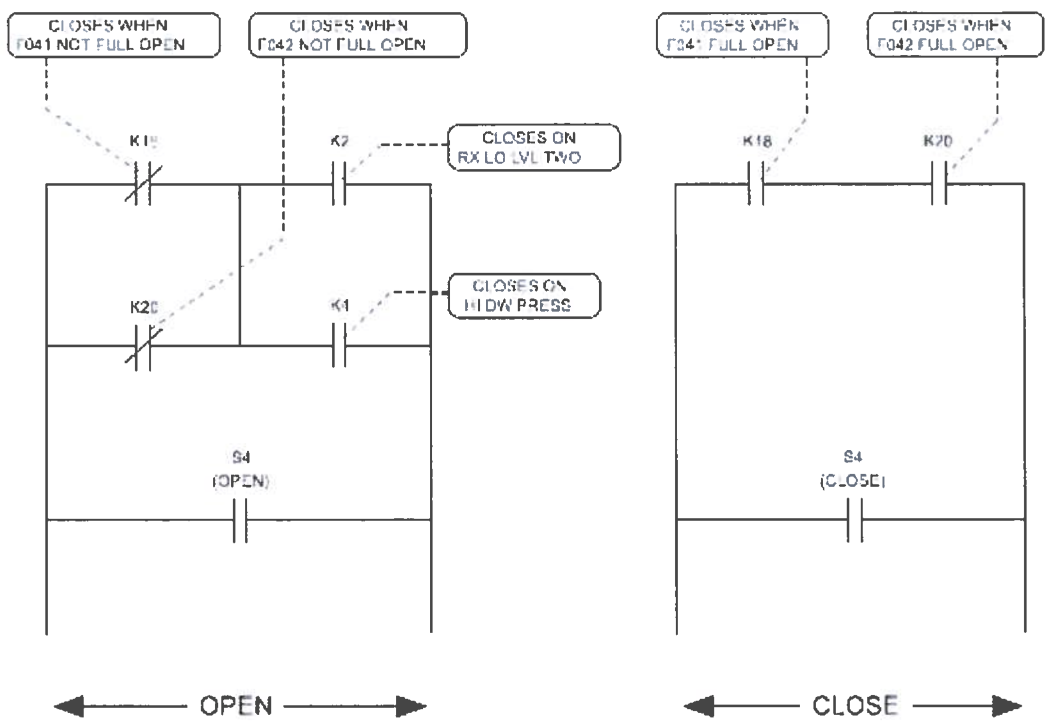
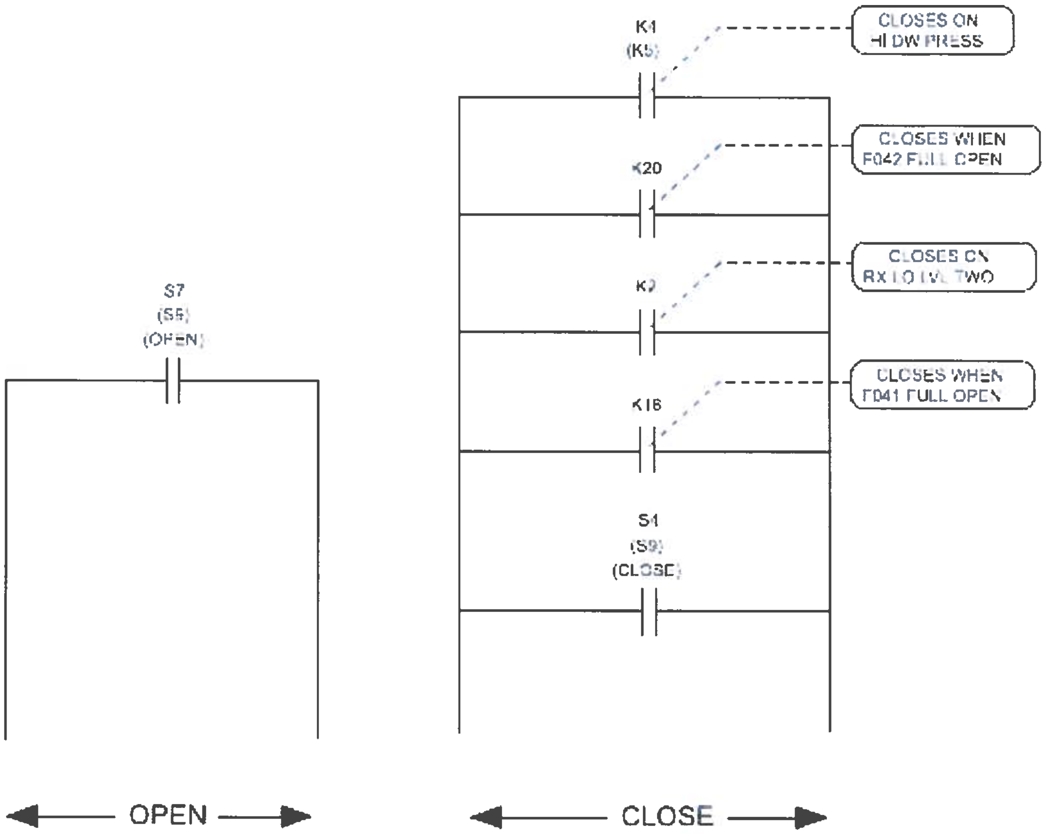
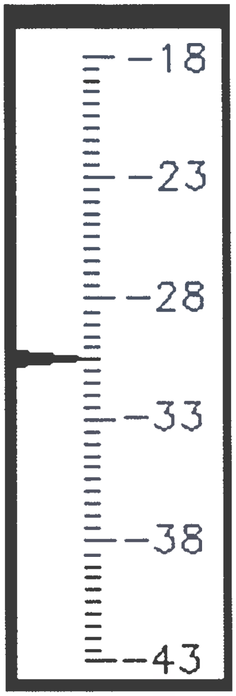


FIGURE 19-15

Test Return Isolation Valve, E41-F008 (E41-F011) Control Logic



53. 295030 1



Unit One is operating at rated power when A-01 (3-7) *Suppression Chamber Lvl Hi/Lo,* is received.

The BOP Operator verifies the alarm using CAC-LI-4177, Supp Pool Level, indicator on Panel XU-51. (indication provided to the left)

Which one of the following identifies the action that is required IAW A-01 (3-7) *Suppression Chamber Lvl Hi/Lo*?

The water level in the Unit One torus must be:

A lowered by using Core Spray and routed to Radwaste.

B lowered using RHR and routed to Radwaste.

C raised by opening the HPCI suction from the CST.

D raised by opening the Core Spray suction from the CST.

Answer: D

K/A:
295030 Low Suppression Pool Water Level
G2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

RO/SRO Rating: 4.2/4.0

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing ability to know whether the alarm is due to high or low level and knowledge of how to correct.

Pedigree: New

Objective: LOI-CLS-LP-302-D, Obj 2
Given plant conditions and AOP-14.0, determine the required supplementary actions.

Reference: None

Cog Level: High

Explanation: The student will verify that level is low using the provided indication and then will determine that the level must be raised IAW the APP. The low level alarm comes in at -30.5 inches and the high level alarm comes in at -27.5 inches. Level can be raised using RHR or the Core Spray systems.

Distractor Analysis:

Choice A: Plausible because it is a combined alarm and if it is assumed that a high water level condition exists the CS system can take a suction from the torus to correct the level condition, but is not allowed in the procedure.

Choice B: Plausible because it is a combined alarm and if it is assumed that a high water level condition exists the RHR system is utilized in the procedure to lower level.

Choice C: Plausible because level is low requiring it to be raised and the HPCI system could gravity drain to the torus, but is not allowed by the procedure.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

SUPPRESSION CHAMBER LVL HI/LO

AUTO ACTIONS

NONE

CAUSE

1. Suppression pool water level high (-27½ inches)
2. Suppression pool water level low (-30½ inches)
3. Circuit malfunction

OBSERVATIONS

1. Suppression pool water level (CAC-LI-2601-1, CAC-LI-4177, CAC-LR-2602)

NOTE:	Rapid changes in suppression pool pressure due to conditions such as inerting or air in-leakage can cause level fluctuations in suppression pool up to 1 inch or more.

ACTIONS

NOTE:	ECCS keepfill stations makeup flow to the suppression pool is approximately 27 gpm.

1. If the cause of the annunciator is a planned evolution, then refer to the appropriate operating procedure to maintain suppression pool water level.
2. If the cause of the annunciator is not a planned evolution, then determine the cause of addition or loss of water to suppression pool and minimize evolutions which add or remove water to or from the suppression pool.
3. If suppression pool water level is high or low, then enter 0AOP-14.0 to drain or fill the suppression pool as necessary.
4. If suppression pool water level is greater than -27 inches or less than -31 inches, then enter 0EOP-02-PCCP.
5. If a circuit malfunction is suspected, ensure a WO is prepared.

4.2.4 Suppression Pool Level High/Low

1. **IF** suppression pool level is approaching -27 inches,
   THEN lower suppression pool level to Radwaste in accordance with 1OP-17(2OP-17), Residual Heat Removal System Operating Procedure. ☐

2. **IF** suppression pool level is approaching -31 inches,
   THEN raise suppression pool level in accordance with the following applicable procedure: ☐

   Unit 1 Only:

   - 1OP-17, Residual Heat Removal System Operating Procedure ☐
   - 1OP-18, Core Spray System Operating Procedure ☐

   Unit 2 Only:

   - 2OP-17, Residual Heat Removal System Operating Procedure ☐
   - 2OP 18, Core Spray System Operating Procedure ☐

54. 295031 1

Unit One is executing the ATWS procedure with the following plant conditions:

Reactor power	12%
Reactor pressure	940 psig, controlled by EHC
Reactor water level	170 inches, controlled by feedwater

Which one of the following identifies the reason the ATWS procedure directs deliberately lowering RPV water level to 90 inches?

A Reduces reactor power so that it will remain below the APRM downscale setpoint.

B Provides heating of the feedwater to reduce potential for high core inlet subcooling.

C Reduces challenges to primary containment if MSIVs close.

D Promotes more efficient boron mixing in the core region.

Answer: B

K/A:

295031 Reactor Low Water Level

EK1 Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.8 to 41.10)

03 Water level effects on reactor power

RO/SRO Rating: 3.7/4.1

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the knowledge of why level is lowered in a ATWS

Pedigree: Bank

Objective: LOI-CLS-LP-300-E, Obj 7
Explain the reason for lowering reactor water level while performing the Anticipated Transient Without Scram Procedure.

Reference: None

Cog Level: fundamental

Explanation: To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, reactor water level is lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, the initiation and growth of oscillations is principally dependent upon the subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude. Twenty-four inches below the lowest nozzle in the feedwater sparger (i.e. 90 inches) has been selected as the upper bound of the reactor water level control band. This water level is sufficiently low that steam heating of the injected water will be at least 65% to 75% effective (i.e., the temperature of the injected water will be increased to 65% to 75% of its equilibrium value in the steam environment). This water level is sufficiently high that even without bypassing the low reactor water level MSIV isolation, reactor water level can be controlled with the feedwater pumps to preclude the isolation.

Distractor Analysis:

Choice A: Plausible because since the operator can re-establish injection at 90 inches irrespective of power level. Power will lower as level is lowered but 90 inches will not guarantee APRMs are downscale

Choice B: Correct Answer, see explanation.

Choice C: Plausible because since there is no current challenge to containment from heat input. If level is lowered due to containment heat input, 90 inches is not specified as the top of the level band. This would be either TAF or the level at which downscales are received

Choice D: Plausible because since lowering level will reduce natural circulation and reduce boron mixing. ATWS procedure directs raising level back to the normal band (170-200 inches) once hot shutdown boron weight is injected

SRO Basis: N/A

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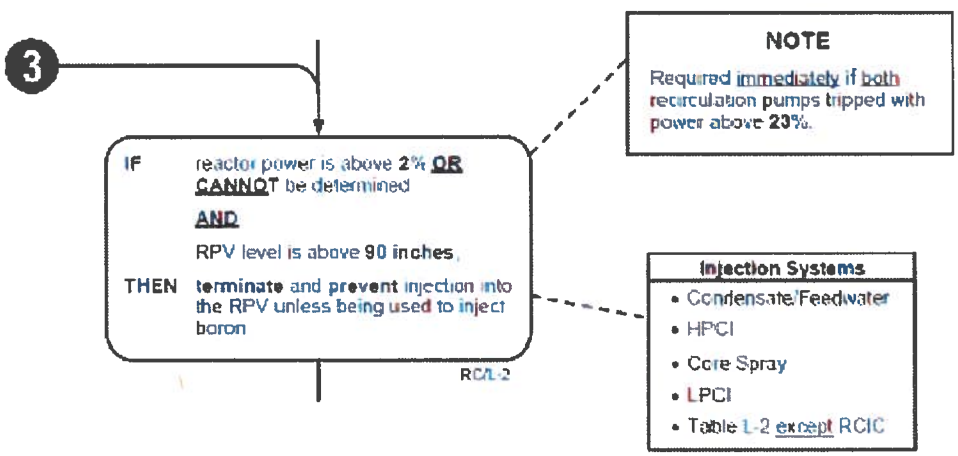
5.4 Step RC/L-2



If reactor power is greater than 23% with both reactor recirculation pumps tripped and RPV level above 90 inches, RPV level needs to be promptly reduced below the feedwater nozzles, to avoid thermal hydraulic instabilities. This is accomplished by termination and prevention of injection systems, from identified systems, particularly feedwater, within 120 seconds.

To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, RPV level is initially lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, initiation and growth of oscillations is principally dependent upon subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude.

55. 295032 1

Which one of the following identifies the reason for performing Emergency Depressurization due to exceeding Maximum Safe Operating Temperatures IAW 0OI-37.9, *Secondary Containment Control Procedure Basis Document*?

A Prevent an unmonitored release.

B Preserve personnel access into the reactor building.

C Prevent damage to equipment required for safe shutdown.

D Ensure ODCM site boundary dose limits are not exceeded.

Answer: C

K/A:
295032 High Secondary Containment Area Temperature
EK3 Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: (CFR: 41.5 / 45.6)
01 Emergency/normal depressurization

RO/SRO Rating: 3.5/3.8

Tier 1 / Group 2

K/A Match: This meets the K/A because it is testing the reason ED is performed for high secondary containment temperatures.

Pedigree: Bank

Objective: LOI-CLS-LP-300-M, Obj 13a
Given plant conditions and the SCCP, determine the required actions if the following limits are exceeded: Maximum Safe operating values with a primary system discharging into secondary containment.

Reference: None

Cog Level: Fundamental

Explanation: The MSOT values are the area temperatures above which equipment necessary for the safe shutdown of the plant will fail. These area temperatures are utilized in establishing the conditions which reactor depressurization is required. The criteria of more than one area specified in this step identifies the rise in reactor building parameters as a wide spread problem which may pose a direct and immediate threat to secondary containment integrity, equipment located in the RB, and continued safe operation of the plant.

Distractor Analysis:

Choice A: Plausible because this is a purpose of SCCP not the reason for ED on Temperature.

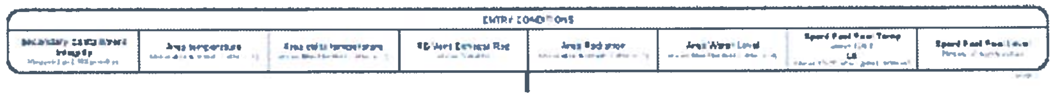
Choice B: Plausible because this is the reason for max safe operating rad levels.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because this is a purpose of SCCP not the reason for ED on Temperature.

SRO Basis: N/A

5.1 Step SCCP-1



The conditions which require entry to SCCP are symptomatic of conditions which, if not corrected, could degrade into an emergency. Adverse effects on the operability of equipment located in the reactor building and conditions directly challenging secondary containment integrity or spent fuel pool cooling were specifically considered in the selection of these entry conditions. In addition, personnel accessibility to some of the areas may be required to perform certain actions specified in the procedure. This was also considered in making these determinations.

An area temperature or area differential temperature above its maximum normal operating level is an indication that steam from a primary system may be discharging into the reactor building. As temperatures continue to increase, the continued operability of equipment needed to carry out EOP actions may be compromised.

56. 295034 1

Which one of the following completes both statements below?

IAW 0AOP-5.4, *Radiological Releases*, RRCP is entered when the Turbine Building Vent Rad Monitor indication exceeds an ___(1)___ EAL.

IAW RRCP, before the radioactivity release rate reaches a ___(2)___ Emergency EAL, Emergency Depressurization is required.

A (1) Unusual Event
(2) Site Area

B (1) Unusual Event
(2) General

C (1) Alert
(2) Site Area

D (1) Alert
(2) General

Answer: D

K/A:

295034 Secondary Containment Ventilation High Radiation

G2.4.08 Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.5

Tier 1 / Group 2

K/A Match: This question matches the KA because it tests the knowledge if the AOP and EOP are performed in conjunction with each other.

Pedigree: new

Objective: LOI-CLS-LP-302-J, Obj. 3c
Given plant conditions, determine the required Supplementary Actions in accordance with: 0AOP-05.4, Radiological Release

Reference: None

Cog Level: Fundamental

Explanation: The AOP states that when an Alert EAL is entered then ENTER RRCP. Before a GE is declared ED is required to be performed. (A scram is required before a SAE is declared)

Distractor Analysis:

Choice A: Plausible because an Unusual Event is the first declaration in the EAL network and a SAE is the criteria for a scram in RRCP.

Choice B: Plausible because an Unusual Event is the first declaration in the EAL network and the second part is correct.

Choice C: Plausible because the first part is correct and the SAE is the criteria for a scram in RRCP.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

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3.0 AUTOMATIC ACTIONS (continued)

- SBGT starts ☐
- Group 6 isolation valves close ☐

4. **IF** UA-03 2-8, Radwaste Effluent Rad Hi Hi, in ALARM, **THEN** D12-V27A(B) (RW Liq Effluent Disch Vlvs) close ☐

5. **IF** UA-23 3-6, Main Steam Line Rad Hi-Hi/Inop, in ALARM, **THEN**:
   - Mechanical vacuum pumps trip ☐
   - OG-V7 (Cndsr Hogging Valve) closes ☐

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Supplementary Actions

1. **IF AT ANY TIME** elevated radiation levels are determined to be from resin injection only, **THEN go to** 0AOP-26.0, High Reactor Coolant or Condensate Conductivity ☐

2. **IF AT ANY TIME** gaseous release rate exceeds an Alert level, **THEN enter** 0EOP-04-RRCP, Radioactivity Release Control Procedure ☐

BEFORE

Offsite release reaches
GE EAL (RG1)
(Table M-4)

RRCP-9

**EMERGENCY DEPRESSURIZATION
REQUIRED**

RRCP-10

57. 295036 1
Following an unisolable RWCU line break in the reactor building the following conditions exist:

South Core Spray Room temperature 155°F
South RHR Room temperature 300°F
UA-12 (2-3) *South Core Spray Room Flood Level Hi,* in alarm
UA-12 (2-4) *South RHR Room Flood Level Hi,* in alarm
UA-12 (1-4) *South RHR Room Flood Level Hi-Hi,* in alarm

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW 0EOP-01-UG, *User's Guide*, ___(1)____equipment required for safe shutdown will fail.

IAW SCCP, Emergency Depressurization ___(1)___ required.

A (1) ONLY the South RHR room
(2) is

B (1) ONLY the South RHR room
(2) is NOT

C (1) the South RHR room AND Core Spray room
(2) is

D (1) the South RHR room AND Core Spray room
(2) is NOT

Answer: B
K/A:
295036 Secondary Containment High Sump / Area Water Level
EK1 Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: (CFR: 41.8 to 41.10)
02 Electrical ground/ circuit malfunction

RO/SRO Rating: 2.6/2.8

Tier 1 / Group 2

K/A Match: This meets the K/A because this is testing the implication of high water level on equipment and whether ED is required.

Pedigree: New

Objective: LOI-CLS-LP-300-M, Obj, 13a
Given plant conditions and the Secondary Containment Control Procedure, determine the required action if the following limits are exceeded: Maximum Safe operating values WITH a primary system discharging into Secondary Containment

Reference: 0EOP-01-NL, *EOP/SAMG Numerical Limits And Values*, Attachment 3, *Containment Parameters*, Table 3-B, *Secondary Containment Area Temperature Limits*

Cog Level: High

Explanation:

Distractor Analysis:

Choice A: Plausible because the first part is correct and for ED two areas in the same parameter must be at max safe conditions, while this question has two parameters in the same area.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because both areas have a max normal condition and for ED two areas in the same parameter must be at max safe conditions, while this question has two parameters in the same area.

Choice D: Plausible because both areas have a max normal condition and the second part is correct.

SRO Basis: N/A

3.0 DEFINITIONS (continued)

- Core Spray Loop A
- Core Spray Loop B
- RHR Loop A (one or two pumps running)
- RHR Loop B (one or two pumps running)

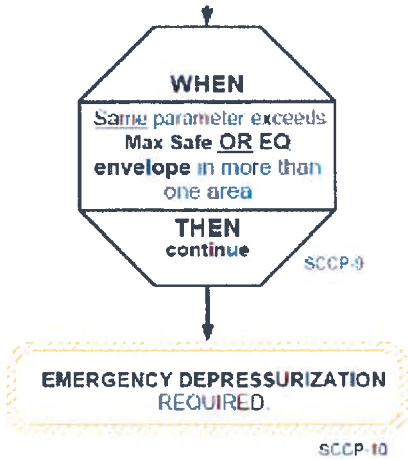
32. **Maximum Normal Operating (Parameter):** The highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

33. **Maximum Pressure Suppression Primary Containment Water Level:** The highest primary containment water level at which the pressure suppression capability of the containment can be maintained. This corresponds to the bottom of the ring header.

34. **Maximum Safe Operating Radiation Level:** The radiation level above which personnel access necessary for the safe shutdown of the plant will be precluded. If the maximum safe operating radiation level is exceeded in an area (but is within the EQ envelope as contained in DR-227, Document Reference for Environmental Qualification Service Conditions) and then later clears and is subsequently followed by another area exceeding maximum safe operating radiation level, action for one area exceeding maximum safe operating radiation level should be taken.

35. **Maximum Safe Operating Temperature:** The temperature above which equipment necessary for the safe shutdown of the plant may fail. This temperature is utilized in establishing the conditions under which RPV depressurization is required. Separate temperatures are provided for each Secondary Containment area. If the maximum safe operating temperature is exceeded in an area and then later clears and is subsequently followed by another area exceeding maximum safe operating temperature, action for two areas exceeding maximum safe operating temperature should be taken.

36. **Maximum Safe Operating Water Level:** The water level above which equipment necessary for the safe shutdown of the plant may fail. This water level is utilized in establishing the conditions under which RPV depressurization is required. Separate water levels are provided for each Secondary Containment area. If the maximum safe operating water level is exceeded in an area and then later clears and is subsequently followed by another area exceeding maximum safe operating water level, action for two areas exceeding maximum safe operating water level should be taken.



ATTACHMENT 3
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Containment Parameters

Secondary Containment Area Temperature Limits

Table 3-B

PLANT AREA	PLANT LOCATION DESCRIPTION	MAX NORM OPERATING VALUE (°F)	MAX SAFE OPERATING VALUE (°F)	AUTO GROUP ISOLATION
N CORE SPRAY	N CORE SPRAY ROOM	120	175	N/A
S CORE SPRAY	S CORE SPRAY ROOM	120	175	N/A
RWCU	PMP ROOM A PMP ROOM B HX ROOM	140	225	3
N RHR	N RHR EQUIP ROOM	175	295	N/A
S RHR	S RHR EQUIP ROOM RCIC EQUIP ROOM	175 165	295 295	N/A 5
HPCI	HPCI EQUIP ROOM	165	165	4
STEAM TUNNEL	RCIC STM TUNNEL HPCI STM TUNNEL	190 190	295 295	5 4
20 FT	2O FT NORTH 20 FT SOUTH	140 140	200 200	N/A N/A
50 FT	50 FT NW 50 FT SE	140 140	200 200	N/A N/A
REACTOR BLDG	MULTIPLE AREAS ANNUN. A-02 5-7	ALARM SETPOINT	N/A	3, 4, AND/OR 5
REACTOR BLDG	MSIV PIT ANNUN. A-06 6-7	ALARM SETPOINT	N/A	1

58. 295037 1

Which one of the following identifies the effect of accomplishing injection of Cold Shutdown Boron Weight during an ATWS?

The reactor is shutdown:

A and will remain shutdown under all conditions.

B and may return to power if a cooldown is initiated.

C and will return to power if RPV water level is raised.

D and may return to power as Xenon depletes during the first 24 hours.

Answer: A

K/A:
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
EK1 Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : (CFR: 41.8 to 41.10)
03 Boron effects on reactor power (SBLC)

RO/SRO Rating: 4.2/4.4

Tier 1 / Group 1

K/A Match: This meets the K/A because the student will have to know the effects that boron has to overcome to have the reactor remain shutdown during an ATWS.

Pedigree: Bank

Objective: LOI-CLS-LP-005, Obj 3
List the positive reactivity effects that must be overcome by SLC injection

Reference: None

Cog Level: Fundamental

Explanation: Injection of the CSBW into the RPV will provide adequate assurance that the reactor is and will remain shutdown. It 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 water temperature

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because a cooldown would add positive reactivity.

Choice C: Plausible because raising level would add positive reactivity

Choice D: Plausible because xenon would add positive reactivity

SRO Basis: N/A

3.0 DEFINITIONS (continued)

12. **Chugging:** An intermittent condensation phenomenon which occurs at the downcomer exit when the drywell is pressurized due to a small high energy (steam) leak inside the drywell. When a steam bubble collapses at the exit of the downcomers, the rush of water filling the void (some of it drawn up into the downcomer pipe) induces severe stress at the junction of the downcomer vent header. Repeated application of this stress can cause these joints to experience fatigue failure (i.e., crack) thereby creating a pathway which bypasses the pressure suppression function of the containment.

13. **Cold Shutdown Boron Weight:** 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 water temperature.

59. 295038 1

A radioactive release has occurred in the Radwaste Building.

Which one of the following completes both statements below?

The Radwaste Building HVAC __(1)__ provide for the adsorption of noble gases.

This discharge will be monitored by the __(2)__.

A (1) will
(2) Main Stack Radiation Monitor

B (1) will
(2) Turbine Building Wide Range Gaseous Monitor (WRGM)

C (1) will NOT
(2) Main Stack Radiation Monitor

D (1) will NOT
(2) Turbine Building Wide Range Gaseous Monitor (WRGM)

Answer: C

K/A:
295038 High Off-Site Release Rate
EA1 Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.7 / 45.6)
01 Stack-gas monitoring system

RO/SRO Rating: 3.9/4.2

Tier 1 / Group 1

K/A Match: This meets the K/A because the student will have to determine the place that the radwaste ventilation system discharges to and the rad monitor that monitors it. (ability to monitor)

Pedigree: New

Objective: LOI-CLS-LP-37.2, Obj 6e
State the interrelationship between the Radwaste Building Ventilation and the following: Process Radiation Monitoring

Reference: None

Cog Level: Fundamental

Explanation: The radwaste vent system does not have any charcoal adsorbers in the system, only HEPA filters. The discharge is to the main stack.

Distractor Analysis:

Choice A: Plausible because other systems provide for the adsorption of noble gases and the second part is correct.

Choice B: Plausible because other systems provide for the adsorption of noble gases and the other systems (Turbine Building Vent) is monitored by the WRGM.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and the other systems (Turbine Building Vent) is monitored by the WRGM.

SRO Basis: N/A

1.1 System Purpose

The purpose of the Radwaste Building Heating, Ventilation, and Air Conditioning System is to maintain areas at a temperature which provides optimal operation of equipment, comfort and safety of personnel, and to control/prevent radiological releases. A negative pressure relative to atmosphere is maintained by this system. Flow through the building is limited to prevent the spread of contamination and to mitigate the consequences of an inadvertent release of radioactive material to the building. All air exiting the Radwaste Building is monitored via the main stack radiation monitor. The Radwaste Building exhaust air volume provides dilution flow to the plant stack for condenser off-gas, condenser air removal, and the Reactor Building standby gas trains.

Several items of personnel safety may be mentioned relative to the building HVAC System. When the Radwaste Building Ventilation system is functioning in concert with the Control Building Ventilation System, major pressure differentials between the volumes are manageable. If the relative pressure between the two volumes is high, personnel access doors may not function as predicted. If the Radwaste Building is under a significant negative pressure and the Control Building pressure is positive in relation to the Radwaste Building, the door will open rapidly.

Certain areas of the Radwaste Building have high noise levels attributed to the ventilation system. Areas close to the supply and exhaust ducts are especially noisy. The upper elevations of the Radwaste Building, especially around the supply and exhaust plenums have very high noise levels. It is imperative that hearing protection is used in these areas. A list of definitions and abbreviations used in this material is in Attachment 1.

60. 300000 1

Unit One is operating at rated power when the following alarms are received:

UA-01 (4-4) *Instr Air Press-Low*
UA-01 (5-1) *Air Dryer 1A Trouble*

The AO reports that the cause of the alarms is due to filter blockage.

Which one of the following completes both statements below?

The Service Air Dryer malfunction will cause SA-PV-5067, Service Air Dryer Bypass Valve, to open when pressure **first** lowers to ___(1)___.

IAW 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures, the required action is to ___(2)___.

A. (1) 105 psig
(2) place the 1B Service Air Dryer in service

B. (1) 105 psig
(2) set the service air dryer maximum sweep value to zero

C. (1) 98 psig
(2) place the 1B Service Air Dryer in service

D. (1) 98 psig
(2) set the service air dryer maximum sweep value to zero

Answer: C

K/A:
300000 Instrument Air System
A2 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: (CFR: 41.5 / 45.6)
01 Air dryer and filter malfunctions

RO/SRO Rating: 2.9/2.8

Tier 2 / Group 1

K/A Match: This meets the KA because it is predicting the response on the system and then using procedure (AOP-20) determine the action required.

Pedigree: New

Objective: LOI-CLS-LP-046, Obj. 6
Given plant conditions, determine if the following automatic actions should occur:
a. Service Air Isolation g. Air Dryer bypass.

Reference: None

Cog Level: High

Explanation: 98 psig is when the bypass valve auto opens, the 105 psig is the isolation setpoint for Service Air. The AOP will direct placing the standby Air Dryer in service.

Distractor Analysis:

Choice A: Plausible because 105 is the isolation setpoint for the service air system and the second part is correct.

Choice B: Plausible because 105 is the isolation setpoint for the service air system and this is an action in the AOP but would not be performed for this failure. If there is a high demand then this is performed to limit the amount of air that is used for the blowdown of the air dryer filter when cycling filters.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the first part is correct and this is an action in the AOP but would not be performed for this failure. If there is a high demand then this is performed to limit the amount of air that is used for the blowdown of the air dryer filter when cycling filters.

SRO Basis: N/A

4.2 Supplementary Actions (continued)

NOTE

- Service Air System pre-filter or after-filter differential pressure should **NOT** exceed 15 psid. ☐
- In service air compressor high discharge pressure (**Unit 1**: greater than or equal to 125 psig, **Unit 2**: greater than or equal to 130 psig) or relief valves lifting could be an indication of air dryer high differential pressure potentially caused by power failures resulting in valves in the flow path failing closed. ☐
- 1(2)SA-PV-5067 [Serv Air Dryer 1(2)A Bypass Pressure Control Valve], is located in the Turbine Building air compressor area. ☐

i. **IF** UA-01 5-3, Air Dryer 1(2)A Trouble, is in alarm, **THEN perform** the following:

(1) **Unit 1 Only**: **Confirm** 1-SA-PV-5067 (Serv Air Dryer 1A Bypass Pressure Control Valve), is OPEN. ☐

CAUTION

The service air dryer provides a low dew point pneumatic source to downstream components. A low dew point is necessary to insure long term reliability of these components. The time the dryer is bypassed should be minimized {7.1.1}. ☐

(2) **Unit 1 Only**: **IF** 1-SA-PV-5067 is **NOT** open, **THEN open** 1-SA-V5089 (Serv Air Dryer Manual Bypass Valve). ☐

(3) **Unit 2 Only**: **Confirm** 2-SA-PV-5067 (Serv Air Dryer 2A Bypass Pressure Control Valve), is OPEN. ☐

(4) **Unit 2 Only**: **IF** 2-SA-PV-5067 is **NOT** open, **THEN open** 2-SA-V5089 (Serv Air Dryer Manual Bypass Valve). ☐

(5) **IF** available, **THEN place** 1B Service Air Dryer in service **AND shutdown** 1(2)A Service Air Dryer in accordance with 0OP-46, Instrument and Service Air System Operating Procedure. ☐

4.2 Supplementary Actions (continued)

c. **IF** air is **NOT** cross-tied,
AND cross-tie operation will **NOT** cause a loss of instrument air on the unaffected unit,
THEN perform the following:

(1) **Obtain** permission from the non-affected unit. .... ☐

(2) **Ensure** 1-SA-PV-5071 (Cross-Tie Valve), located on Unit 1, Panel XU-2, is OPEN. .... ☐

(3) **Ensure** 2-SA-PV-5071 (Cross-Tie Valve), located on Unit 2, Panel XU-2, is OPEN. .... ☐

(4) **IF** opening the cross-tie valve degrades the non-affected unit,
THEN return to Step 1.b(4). .... ☐

d. **IF** the in service air dryer is in sweep mode,
THEN consider securing sweep mode in accordance with Attachment 1, Setting Service Air Dryer(S) Maximum Sweep Value To Zero. .... ☐

61. 300000 2

Unit One is in MODE 3 following a seismic event and reactor scram with the following plant conditions:

Reactor level	55 inches
Reactor pressure	500 psig
Drywell pressure	9 psig
Division I PNS header pressure	93 psig
Division II PNS header pressure	98 psig

Which one of the following completes both statements below?

Div I Backup N2 Rack Isol Vlv, RNA-SV-5482 is ____(1)____.

Div II Backup N2 Rack Isol Vlv, RNA-SV-5481 is ____(2)____.

A. (1) open
   (2) open

B. (1) open
   (2) closed

C. (1) closed
   (2) open

D. (1) closed
   (2) closed

Answer: B

K/A:

300000 Instrument Air System

K3 Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following: (CFR: 41.7 / 45.6)

01 Containment air system

RO/SRO Rating: 2.7/2.9

Tier 2 / Group 1

K/A Match: This meets the KA because it is testing the effect of the low pressure (loss or malfunction) of the air system on containment air (N2 backup).

Pedigree: Last used on 2007 NRC Exam

Objective: LOI-CLS-LP-046-A, Obj. 8
Given plant conditions, determine the effects that the following conditions will have on the Pneumatic System: (LOCT) b. Low Instrument Air/Pneumatic Nitrogen (IAN/RNA/PNS) Header Pressure

Reference: None

Cog Level: High

Explanation: No LOCA signal is present so the Backup N2 valves will not be open on a Core Spray initiation signal. The Backup N2 valves open at 95 psig or lower in the PNS header. This would result in Division I Backup N2 valve (5482) being open and Division II (5481) being closed.

Distractor Analysis:

Choice A: Plausible if the student believes that either division will open both valves.

Choice B: Correct Answer, see explanation.

Choice C: Plausible if the student uses the valves for the division separation.

Choice D: Plausible if the student only checks the LOCA signal and not the low pressure signal.

SRO Basis: N/A

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RB INSTR AIR RECEIVER 1A PRESS LOW

AUTO ACTIONS

1. RNA-SV-5482, High Pressure Bottle Rack Isolation Valve, opens, supplying SRVs and CAC-16 with a pneumatic source.

DEVICE/SETPOINTS

RNA-PSL-3596	95 psig decreasing

Unit 1
APP UA-01 1-2
Page 1 of 2

RB INSTR AIR RECEIVER 1B PRESS LOW

AUTO ACTIONS

1. RNA-SV-5481, High Pressure Bottle Rack Isolation Valve, opens, supplying SRVs and CAC-17 with a pneumatic source.

DEVICE/SETPOINTS

RNA-PSL-3597	95 psig decreasing

62. 400000 1

Unit One is operating at rated power with the following conditions:

CSW Pump 1A trips
Conventional header pressure lowers to 35 psig

Which one of the following completes both statements below?

If CSW header pressure remains at this pressure for ___(1)___ seconds, the SW-V3, SW To TBCCW HXs Otbd Isol Vlv, and SW-V4, SW To TBCCW HXs Inbd Isol Vlv, will close to a throttled position.

IAW 0AOP-19, *Conventional Service Water System Failure*, the SW-V3 and SW-V4 are reopened ___(2)___.

A. (1) 30
(2) ONLY after a reactor Scram is inserted

B. (1) 30
(2) if system pressure is restored by starting the standby CSW pump

C. (1) 70
(2) ONLY after a reactor Scram is inserted

D. (1) 70
(2) if system pressure is restored by starting the standby CSW pump

Answer: D

K/A:

400000 Component Cooling Water System (CCWS)

A2 Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: (CFR: 41.5 / 45.6)

01 Loss of CCW pump

RO/SRO Rating: 3.3/3.4

Tier 2 / Group 1

K/A Match: This meets the KA because it is testing the auto start signal/logic for a cooling water system.

Pedigree: Last used on the 2010 NRC exam

Objective: CLS-LP-302-H, Obj. 4
Given plant conditions and any of the following AOPs, determine the required supplementary actions: d. 0AOP-19.0, Conventional Service Water System Failure

Reference: None

Cog Level: High

Explanation: **IF** conventional service water header pressure remains below 40 psig for 70 seconds,
THEN:
- *SW TO TBCCW HXS OTBD ISOL, SW-V3* closes to a throttled position
- *SW TO TBCCW HXS INBD ISOL, SW-V4* closes to a throttled position

The Standby CSW pump should start and restore CSW header pressure to normal prior to the SW valves throttling closed. If the standby CSW pump fails to auto start, manually starting the pump will restore CSW header pressure. AOP-19 provides guidance to re-open the SW valves only after header pressure has been restored and the cause of low pressure is known (pump trip).

Distractor Analysis:

Choice A: Plausible because 30 seconds is when the DG cooling water valves close and a Scram is inserted only after the SW valves have closed to the throttled position AND CSW header pressure cannot be immediately restored above 40 psig - under this condition all CSW pumps would be shutdown.

Choice B: Plausible because 30 seconds is when the DG cooling water valves close and system pressure restored by the STBY pump start is correct.

Choice C: Plausible because 70 seconds is correct and a Scram is inserted only after the SW valves have closed to the throttled position AND CSW header pressure cannot be immediately restored above 40 psig - under this condition all CSW pumps would be shutdown.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

CONVENTIONAL SERVICE WATER SYSTEM FAILURE	0AOP-19.0
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3.0 AUTOMATIC ACTIONS

1. Standby pump selected to the conventional service water header starts at 40 psig ☐

2. **IF** all conventional service water pumps are tripped,
   THEN:
   - SW-V36 (SW To CW Pumps Inbd Vlv), closes ☐
   - SW-V37 (SW To CW Pumps Otbd Vlv), closes ☐
   - CWIPs trip on low bearing lubricating water flow (5 - 6 gpm, time-delayed 15 minutes), resulting in loss of condenser vacuum ☐

3. **IF** conventional service water header pressure remains less than 40 psig for 70 seconds,
   THEN:
   - SW-V3 (SW To TBCCW HXs Otbd Isol), closes to a throttled position ☐
   - SW-V4 (SW To TBCCW HXs Inbd Isol), closes to a throttled position ☐

4.2 Supplementary Actions (continued)

d. **Attempt** to isolate any source of leakage. ☐

e. **Ensure** discharge valves are CLOSED on shutdown pump(s). ☐

f. **Check** service water traveling screens for excessive build-up **AND wash** if excessive buildup is occurring. ☐

g. **Check** service water trash racks for excessive build-up **AND notify** Maintenance to clean if excessive buildup is occurring. ☐

h. Locally **monitor** each pump discharge strainer differential pressure. ☐

i. **Check** Annunciator Panel UA-01 for lit annunciators. ☐

10. **Refer to** Technical Specification 3.7.2, Service Water (SW) System and Ultimate Heat Sink (UHS) for operability requirements. ☐

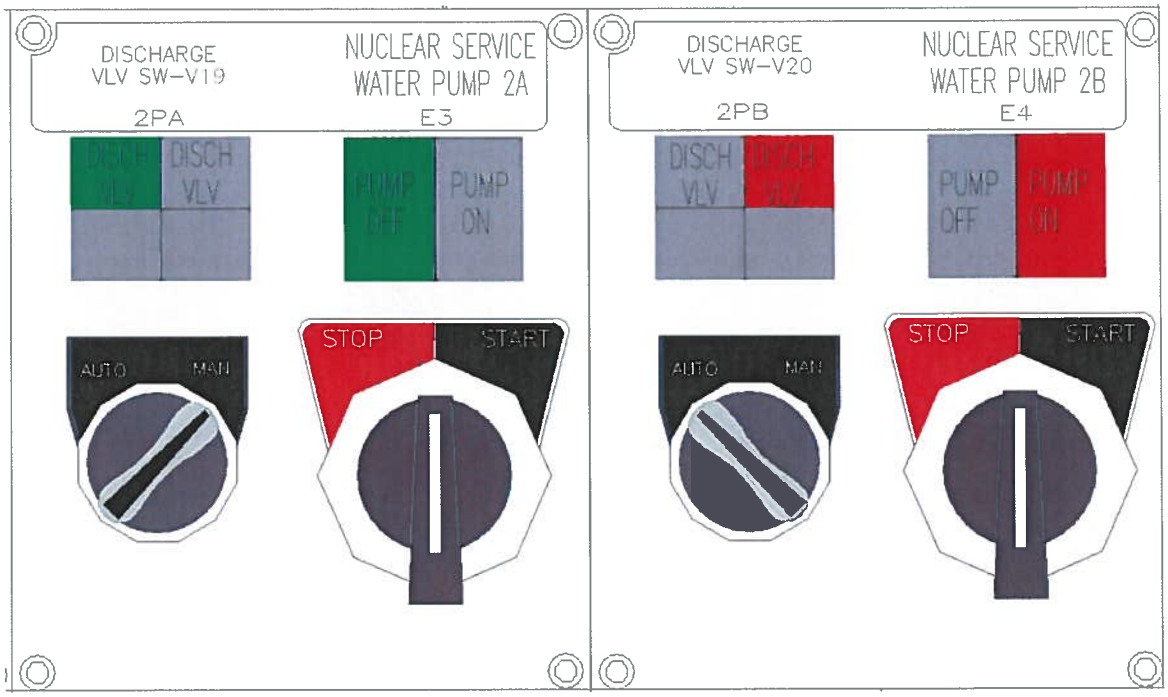
11. **WHEN** conventional service water header pressure is restored to normal
AND the cause of low header pressure has been corrected,
THEN:

a. **Open** SW-V3 (SW To TBCCW HXs Otbd Isol). ☐

b. **Open** SW-V4 (SW To TBCCW HXs Inbd Isol). ☐

63. 400000 2

Unit Two Nuclear Service Water (NSW) pumps are aligned as follows in preparation for equipment realignment:



Subsequently, Off-site power is lost.

Which one of the following identifies how NSW Pumps 2A and 2B will respond when the DGs re-energize their E Buses?

A. Both NSW Pumps auto start immediately when their associated E Bus is energized.

B. Both NSW Pumps auto start five seconds after their associated E Bus is energized.

C. NSW Pump 2B auto starts immediately after the associated E Bus is energized. NSW pump 2A does NOT auto start.

D. NSW Pump 2B auto starts five seconds after the associated E Bus is energized. NSW pump 2A does NOT auto start.

Answer: A

K/A:

400000 Component Cooling Water System (CCWS)

K4 Knowledge of CCWS design feature(s) and or interlocks which provide for the following: (CFR: 41.7)

01 Automatic start of standby pump

RO/SRO Rating: 3.4/3.9

Tier 2 / Group 1

K/A Match: This meets the KA because it is testing the auto start signal/logic for a cooling water system.

Pedigree: Bank

Objective: LOI-CLS-043, Objective 8a
State the power supply (bus and voltage) for the following Service Water System components: Nuclear Service Water Pumps.

Reference: N/A

Cog Level: High

Explanation: NSW pumps auto start immediately after LOOP signal regardless of mode selector switch or discharge valve position. 5 second timer applies only on a LOCA.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this would be the case with a LOCA signal present.

Choice C: Plausible because examinee must know the power supply scheme.

Choice D: Plausible because examinee must know the power supply scheme.

SRO Basis: N/A

Each pump is powered by a 4160 VAC motor supplied from the emergency bus power supplies:

Component	Power Supply
1A CSW pump	E4
1B CSW pump	E1
1C CSW pump	E2
1A NSW pump	E1
1B NSW pump	E2
2A CSW pump	E3
2B CSW pump	E4
2C CSW pump	E1
2A NSW pump	E3
2B NSW pump	E4

In addition to the low header pressure auto start, the NSW pumps will start five seconds after receipt of a LOCA signal, regardless of mode selector switch or discharge valve position. For example, a Division I LOCA signal from either Unit 1 or Unit 2 will auto start the 1A and 2A NSW pumps; the Division II LOCA logic will auto start 1B and 2B NWS pumps.

The NSW pumps, powered through the 4160 VAC emergency buses, will also automatically start immediately after the start of the diesel generators and reenergization of the emergency buses on loss of off-site power (LOOP), regardless of mode selector switch or discharge valve position. For example, a Division I LOOP signal from either Unit 1 or Unit 2 will auto start the 1A and 2A NSW pumps; the Division II LOOP logic will auto start 1B and 2B NSW pumps. If a LOCA signal exists on the division sensing the LOOP, auto start will occur after five seconds, provided that a LOOP signal is not present on the opposite unit. On a dual unit LOOP the NSW pump(s) of the LOCA (and non-LOCA) unit start immediately after the emergency buses are reenergized by their respective diesel generators without the five second delay.

64. 600000 1

Which one of the following identifies the potential consequence of failing to place backup nitrogen in service by placing RNA keylock switches in LOCAL IAW 0ASSD-02, *Control Building*?

RNA keylock switch noun names:
2-RNA-CS-001, Override Switch For Valve RNA-SV-5482
2-RNA-CS-002, Override Switch For Valve RNA-SV-5253

A. Misoperation of RCIC.

B. Loss of drywell cooling.

C. Inability to operate SRVs.

D. Spurious operation of MSIVs.

Answer: C

K/A:
600000 Plant Fire On Site
AK3 Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: (CFR: 41.5 / 45.6)
04 Actions contained in the abnormal procedure for plant fire on site

RO/SRO Rating: 2.8/3.4

Tier 1 / Group 1

K/A Match: This matches the KA because it tests the reason a step in the ASSD procedure is performed. The ASSD procedures are the plant fire procedures.

Pedigree: Bank

Objective: LOI-CLS-LP-304, Obj. 25k
Given ASSD procedures and plant conditions, predict the consequences of FAILURE to perform the following actions: Deenergize RNA-SV-5482 and RNA-SV-5253 via keylock switches RNA-CS-001 and RNA-CS-002.

Reference: None

Cog Level: fundamental

Explanation: The Reactor Building MCC Operator places the key lock switches to the LOCAL position to ensure Nitrogen System is lined up to provide reliable operation of the SRVs.

Distractor Analysis:

Choice A: Plausible because actions for the operation of RCIC are contained in the ASSD procedures.

Choice B: Plausible because a loss of pneumatics would cause the DW cooler dampers to close.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because actions to prevent the spurious operation of the MSIV is contained in the ASSD procedure.

SRO Basis: N/A

1.6.5 **WHEN** directed to start RCIC, **THEN PERFORM** the following at MCC 2XDB:

1. **OPEN** *RCIC TURB TR & THR VLV, E51-V8*, at Compt B37 (**Row C1**). ☐

2. **OPEN** *RCIC TURB STM SPLY VLV, E51-F045*, at Compt B44 (**Row F2**). ☐

3. **INFORM** Unit 2 CRS RCIC should be running. ☐

1.7 **WHEN** directed, **THEN PERFORM** the following at Unit 2 Reactor Building 50 foot elevation:

1.7.1 **PLACE** keylock switch *2-RNA-CS-001* in *LOCAL* for valve *2-RNA-SV-5482*. ☐

1.7.2 **PLACE** keylock switch *2-RNA-CS-002* in *LOCAL* for valve *2-RNA-SV-5253*. ☐

1.7.3 **INFORM** the Unit 2 CRS that backup nitrogen has been made available for SRV operation. ☐

1.8 **IF** directed, **THEN TRANSFER** RCIC suction from CST to suppression pool at MCC 2XDB as follows:

1.8.1 **OPEN** *RCIC SUPP POOL SUCT VLV, E51-F031*, at Compt B45 (**Row G1**). ☐

1.8.2 **OPEN** *RCIC SUPP POOL SUCT VLV, E51-F029*, at Compt B46 (**Row G2**). ☐

1.8.3 **CLOSE** *RCIC CST SUCT VLV, E51-F010*, at Compt B38 (**Row C2**). ☐

65. 700000 1

A grid disturbance occurs with the following Unit One plant parameters:

Generator Load	980 MWe
Generator Reactive Load	160 MVARs, out
Generator Gas Pressure	50 psig

(REFERENCE PROVIDED)

Which one of the following identifies both available options that will place the Unit within the Estimated Capability Curve?

A. Raise gas pressure to 58 psig or lower power to 940 MWe.

B. Raise gas pressure to 58 psig or raise reactive load to 240 MVARs.

C. Raise gas pressure to 58 psig or lower reactive load to 70 MVARs.

D. Lower power to 940 MWe or raise reactive load to 240 MVARs.

Answer: A

K/A:

700000 Generator Voltage and Electric Grid Disturbances

AA2 Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

03 Generator current outside the capability curve

RO/SRO Rating: 3.5/3.6

Tier 1 / Group 1

K/A Match: This meets the K/A because the tests the ability to determine action needed to remain within capability curve.

Pedigree: Last used on 2014 NRC exam

Objective:

CLS-LP-27, Obj. 9 - Given the Generator estimated capability curves, hydrogen pressure and either MVARS, MW, or power factor, determine the limit for MW and MVARS.

Reference: 1OP-27 Attachment 2, Estimated Capability Curves

Cog Level: High

Explanation: Based on the conditions the student should plot the current location on the graph. Plot MWe along the bottom and MVARs up the side. Where these two points intersect, based on 50 psig gas pressure line is outside of the safe area. (Must be inside the curve to be safe) Lowering MWe or raising gas pressure are the only options. For this case lowering or raising MVARs would still be outside the curve.

Distractor Analysis:

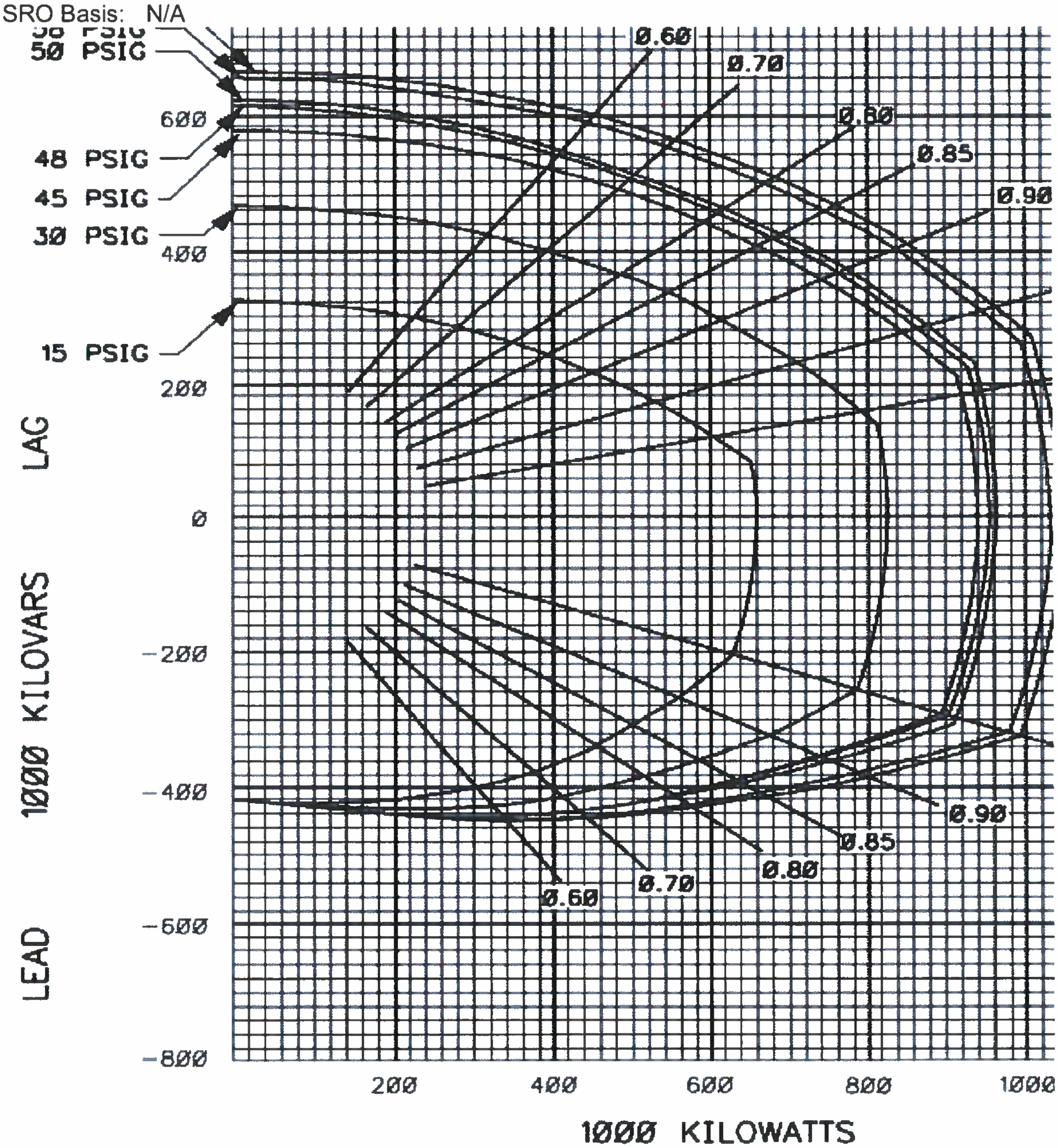
Choice A: Correct Answer, see explanation

Choice B: Plausible because raising pressure will move the plant within the limits of the curve. Raising MVARS will not move the plant within the limits of the curve.

Choice C: Plausible because raising pressure will move the plant within the limits of the curve. Lowering MVARS will not move the plant within the limits of the curve.

Choice D: Plausible because raising MWe will move the plant within the limits of the curve. Raising MVARS will not move the plant within the limits of the curve.

SRO Basis: N/A



66. G2.1.01 1
Which one of the following completes both statements below IAW AD-OP-ALL-1000, *Conduct of Operations*?

With the Unit operating at rated, steady state power, a key parameter that the OATC must monitor to assure a constant awareness of its value and trend is ___(1)___.

An end to end control panel walk down shall be performed every ___(2)___ and documented in the Narrative Logbook.

A. (1) jet pump flow
   (2) one hour

B. (1) jet pump flow
   (2) two hours

C. (1) steam flow / feed flow
   (2) one hour

D. (1) steam flow / feed flow
   (2) two hours

Answer: D
K/A:
G2.1.01 Knowledge of conduct of operations requirements. (CFR: 41.10 / 45.13)

RO/SRO Rating: 3.8/4.2

Tier 3

K/A Match: This meets the K/A because it is testing knowledge of the Conduct of Operations Manual

Pedigree: New

Objective: LOI-CLS-LP-201-D, Obj. 1j - Explain/describe the following IAW AD-OP-ALL-1000, Conduct of Operations, 0OI-01.01, BNP Conduct of Operations Supplement and OPS-NGGC-1314, Communications: Control Board walkdown and monitoring requirements

Reference: None

Cog Level: Fund

Explanation: IAW the conduct of operations document board walk downs must be completed every two hours and section 5.5.6 lists the key parameters to watch.

Distractor Analysis:

Choice A: Plausible because jet pump flow has a daily surveillance requirement and if a watchstander is relieved for greater than one hour it must be entered in narrative logbook.

Choice B: Plausible because jet pump flow has a daily surveillance requirement and part two is correct.

Choice C: Plausible because part one is correct and if a watchstander is relieved for greater than one hour it must be entered in narrative logbook.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

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5.5.6 Control Board Monitoring (continued)

k. Unless involved in activities where Reactor Operator involvement is required by the Conduct of Operations (for example reactivity manipulations, peer checks or detailed panel reviews), the operator shall monitor the following key parameters at a frequency to assure a constant awareness of their value and trend:

- Rx Power
- RPV level (BWR)
- Steam generator pressure (PWR)
- RCS temperature
- Steam generator level (PWR)
- RCS pressure
- Steam flow / feed flow
- Pressurizer level (PWR)

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5.5.6 Control Board Monitoring (continued)

3. The CRS ensures that a licensed operator performs an end to end control panel walk down every two hours. The walk down shall be documented in the Narrative Logbook.

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4. Whenever a watch station is relieved for greater than one hour, this information shall be entered in a Narrative Log Program, a formal turnover and shift turnover sheet will be completed, including the logs signed over.

67. G2.1.32 1

Which one of the following completes the statement below?

1OP-10, *Standby Gas Treatment System Operating Procedure*, prohibits venting the drywell and the suppression pool chamber simultaneously with the reactor at power because this would cause the:

A. unnecessary cycling of reactor building to torus vacuum breakers.

B. unnecessary cycling of torus to drywell vacuum breaker.

C. SBGT Train water seal to blow out of the trough.

D. pressure suppression function to be bypassed.

Answer: D

K/A:

G2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)

RO/SRO Rating: 3.8/4.0

Tier 3

K/A Match: This meets the K/A because it is testing the ability to explain the system precaution.

Pedigree: Last used on 2012 NRC exam

Objective:

Reference: None

Cog Level: Fundamental

Explanation: Per OP-10, torus and drywell cannot be vented at the same time in Modes 1, 2 or 3. per the LER reference, this could result in bypassing pressure suppression function.

Distractor Analysis:

Choice A: Plausible because these vacuum breakers prevent drawing a negative pressure in the suppression pool. Cross connecting the drywell and the suppression pool free air space will not cause a negative pressure in the suppression pool.

Choice B: Plausible because this lineup equalizes pressure between the drywell and the suppression pool free air space since the vacuum breakers operate on a d/p between the spaces this would bypass them, not open them.

Choice C: Plausible because venting containment through large valves with an elevated pressure may blow out the SBGT water seal.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

1.0 PURPOSE

1. This procedure provides instructional guidance for operation of the Standby Gas Treatment System and its associated deluge system.

2.0 SCOPE

1. This procedure provides the prerequisites, precautions, limitations, and instructional guidance for startup, normal operation, shutdown, and infrequent operation of the Standby Gas Treatment System and its associated deluge system.

3.0 PRECAUTIONS AND LIMITATIONS

1. The Standby Gas Treatment System will **NOT** automatically start if the control switch is in STBY. .... ☐

2. Venting the drywell and suppression pool simultaneously is **NOT** performed when the plant is in MODE 1, 2, or 3. {8.1.1} .... ☐

8.0 REFERENCES

8.1 Commitments

1. LER 1-97-011, Drywell and Torus Inerting/Deinerting Lineup Results in Unanalyzed Suppression Pool Bypass Path

68. G2.1.36 1

A core reload is in progress during a refueling outage. The initial loading of fuel bundles around each SRM centered 4-bundle cell was completed with all four SRMs fully inserted and reading 50 cps.

It is now approximately half way through the core loading sequence and SRMs read 80 cps.

Which one of the following completes the statement below IAW 0FH-11, *Refueling*?

Fuel movement must be suspended when any SRM reading **first** rises to _________ upon insertion of the **next** fuel bundle.

A. 100 cps

B. 160 cps

C. 250 cps

D. 400 cps

Answer: B

K/A:

G2.1.36 Knowledge of procedures and limitations involved in core alterations. (CFR: 41.10 / 43.6 / 45.7)

RO/SRO Rating: 3.0/4.1

Tier 3

K/A Match: This meets the K/A because it is testing the fuel movement requirements that an RO would monitor.

Pedigree: New

Objective: LOI-CLS-LP-305, Objectives 18
Given the conditions during a refueling outage state the operator actions required for rising SRM count rates and/or inadvertent criticality.

Reference: None

Cog Level: High

Explanation: An increase in counts by a factor of two during a single bundle insertion is reason to suspend fuel movements. An increase by a factor of five from the baseline is also a reason.

Distractor Analysis:

Choice A: Plausible because this is a doubling of the baseline counts which is used for a different criteria for suspension of fuel movements.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is an increase of the baseline counts by a factor of five which is a reason to suspend fuel movements.

Choice D: Plausible because this is an increase of the counts by a factor of five which is a reason to suspend fuel movements.

SRO Basis: N/A

FH-11:

24. Suspension of fuel movement and notification of the Reactor Engineer is required if either of the following occur:

- An SRM reading rise by a factor of two upon insertion of any single bundle. During a spiral reload, this restriction applies only after the initial loading of fuel bundles around each SRM is complete. During a Core Shuffle, this restriction does **NOT** apply to the SRM that is having an adjacent fuel bundle inserted or removed. .... ☐
- An SRM rise by a factor of five relative to the SRM baseline count rate recorded on Attachment 6, Documentation for SRM Baseline .... ☐

25. SRM count rate may drop to less than 3 cps during either of the following conditions:

- With less than or equal to four fuel assemblies adjacent to the SRM and **NO** other fuel assemblies in the associated core quadrant .... ☐
- During a core spiral offload .... ☐

69. G2.2.02 1

Unit Two is conducting a routine power reduction for rod pattern improvement. The Reactivity Management Plan contains actions for the RO to insert a group of four rods from position 24 to position 12.

Which one of the following completes the statement below IAW AD-OP-ALL-0203, *Reactivity Management*?

The movement of these rods should be:

A. single notched for the entire movement.

B. continuously inserted to the final intended position.

C. continuously inserted and stopped four notches prior to reaching the intended position and then single notched into the final intended position.

D. continuously inserted and stopped one notch prior to reaching the intended position and then single notched into the final intended position.

Answer: D

K/A:

G2.2.02 Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels. (CFR: 41.6 / 41.7 / 45.2)

RO/SRO Rating: 4.6/4.1

Tier 3

K/A Match: This question matches the KA because it tests the generic requirements of control rod movement during any power level.

Pedigree: Bank

Objective: LOI-CLS-LP-201-D, Obj. 22f
Explain the following regarding AD-OP-ALL-0203, Reactivity Management: The procedural requirements for positioning intermediate control rods

Reference: None

Cog Level: High

Explanation: If a rod is to be moved between 46 and 02 three notches or less, it must be single notched the entire move. When moving a control rod four notches or more, the control rod should be stopped one notch prior to reaching the intended position and then single notched into the final intended position.

Distractor Analysis:

Choice A: Plausible because this would apply if the movement was $\leq$ four notches.

Choice B: Plausible because this would apply under emergency conditions.

Choice C: Plausible because the rod does have to be single notched into its final position but the rod can be continuously move if greater than four notches not for four notches.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

5.2.8 [BWR] Single Recirculation Loop Operation

{7.1.5}

1. Standards

   a. Single-Loop operation for extended periods of time is discouraged.

2. Expectations

   a. Plant procedures that address Single Recirculation Loop Operation will identify applicable limitations and trip criteria.

   b. For operations not covered by an approved procedure the Operational Decision Making process will be used to evaluate continued operation in Single-Loop.

   c. The risk associated with single recirculation loop operations shall be carefully considered and appropriate contingencies will be developed.

   d. Operator JITT shall be conducted for planned Single-Loop operations.

5.2.9 Control Rod Manipulations

1. Standards

   a. Ensure all control rod movements are made in a deliberate, carefully controlled manner while constantly monitoring nuclear instrumentation and redundant indications of reactor power and neutron flux.

2. Expectations

   a. [BWR] To minimize the possibility of mispositioning a control rod when inserting or withdrawing to an intermediate position (notch positions '02' through '46'), the following practices shall be followed:

      (1) When moving a control rod four notches or more, the control rod should be stopped one notch prior to reaching the intended position and then single notched into the final intended position. This guidance does not supersede any other requirement to single notch control rods.

      (2) When moving a control rod three notches or less, the control rod should be single notched for the entire move.

70. G2.2.04 1

Which one of the following identifies the Unit Two "Scram Immediate Operator Action" that utilizes a different criteria for performance than on Unit One?

A. Tripping of the main turbine.

B. Tripping one of the running feed pumps.

C. Master level controller setpoint setdown.

D. Placing the reactor mode switch to Shutdown.

Answer: D

K/A:

G2.2.04 Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility. (CFR: 41.6 / 41.7 / 41.10 / 45.1 / 45.13)

RO/SRO Rating: 3.6/3.6

Tier 3

K/A Match: This meets the K/A because it is testing the differences between the Units

Pedigree: Bank

Objective: LOI-CLS-LP-300-C, Obj. 2
List the immediate operator actions for a Reactor Scram. (LOCT)

Reference: None

Cog Level: fund

Explanation: On Unit Two the mode switch cannot be placed to shutdown until MSL flow is less than 3 Mlbms. This restriction does not exist on Unit One.

Distractor Analysis:

Choice A: Plausible because this is an immediate operator action.

Choice B: Plausible because this is an immediate operator action.

Choice C: Plausible because this is an immediate operator action.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

Unit 2 Scram Immediate Actions (0EOP-01-UG)

SCRAM IMMEDIATE ACTIONS

1. **Ensure** SCRAM valves OPEN by manual SCRAM or ARI initiation.
2. **WHEN** steam flow less than 3 x 10^6 lb/hr,
   THEN place reactor mode switch in SHUTDOWN.
3. **IF** reactor power below 2% (APRM downscale trip),
   THEN trip main turbine.
4. **Ensure** master RPV level controller setpoint at +170 inches.
5. **IF:**
   - Two reactor feed pumps running

     AND
   - RPV level above +160 inches

     AND
   - RPV level rising,

   THEN trip one.

Unit 1 Scram Immediate Actions (0EOP-01-UG)

SCRAM IMMEDIATE ACTIONS

1. **Ensure** SCRAM valves OPEN by manual SCRAM or ARI initiation.
2. **Place** reactor mode switch in SHUTDOWN.
3. **IF** reactor power below 2% (APRM downscale trip),
   THEN trip main turbine.
4. **Ensure** master RPV level controller setpoint at +170 inches.
5. **IF:**
   - Two reactor feed pumps running

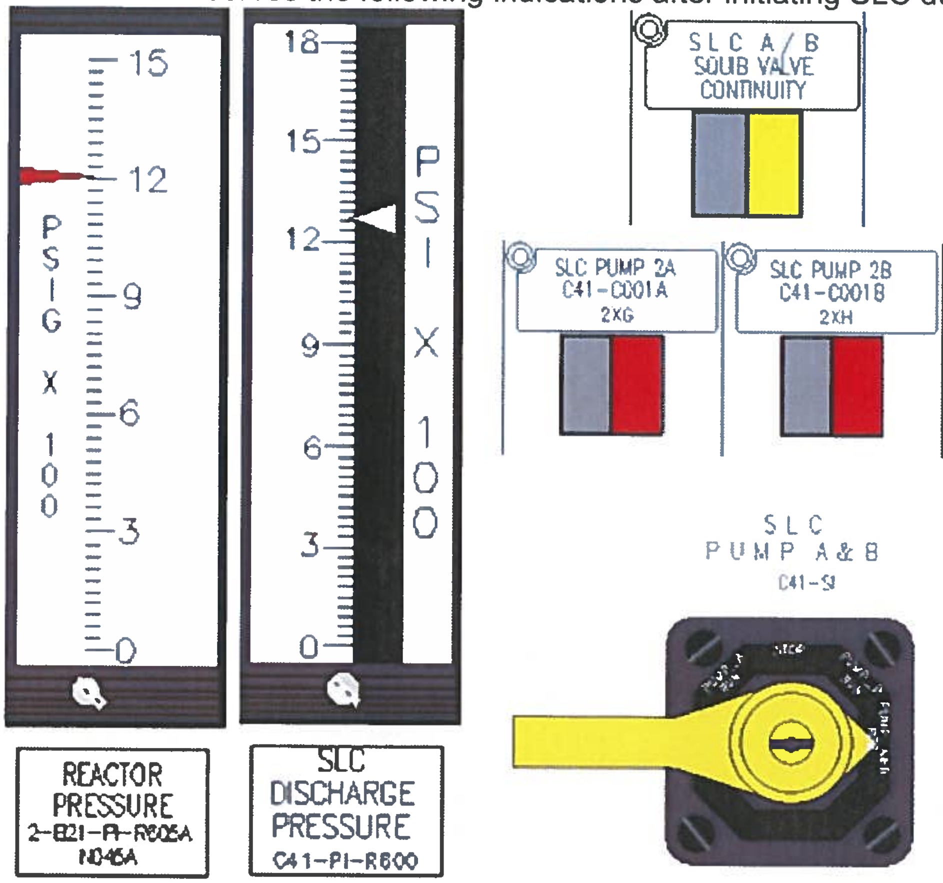
     AND
   - RPV level above +160 inches

     AND
   - RPV level rising,

   THEN trip one.

71. G2.2.44 1

The OATC observes the following indications after initiating SLC during an ATWS.



Which one of the following completes both statements below?

Squib valve ____(1)____ has failed to fire.

IAW 2OP-05, *Standby Liquid System Operating Procedure*, the OATC is required to ____(2)____.

A. (1) A
   (2) place the CS-S1, SLC Pump A & B, in the PUMP A RUN position

B. (1) A
   (2) leave the CS-S1, SLC Pump A & B, in the PUMP A/B RUN position

C. (1) B
   (2) place the CS-S1, SLC Pump A & B, in the PUMP A RUN position

D. (1) B
   (2) leave the CS-S1, SLC Pump A & B, in the PUMP A/B RUN position

Answer: C

K/A:

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.
(CFR: 41.5 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.4

Tier 3

K/A Match: This meets the K/A because it is testing knowledge of the indications and what action is required based on the system lineup.

Pedigree: Previously used on the 2014 NRC exam

Objective: LOI-CLS-LP-005, Obj 13 -
Predict the effect of the following on the Standby Liquid Control System, and based on those predictions use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: a. Failure of one or both squib valves to fire.

Reference: None

Cog Level: Hi

Explanation: The SLC squib valve continuity lights are normally lit and go out when fired on SLC initiation. Per OP-05, if one squib valve fails to fire, two pump SLC operation may still continue provided reactor pressure is below 1184 psig, which it is not.

Distractor Analysis:

Choice A: Plausible because the student may think that the light is illuminated when the squib valve fires and securing 1 pump is correct.

Choice B: Plausible because the student may think that the light is illuminated when the squib valve fires and if reactor pressure was lower this would be correct.

Choice C: Correct Answer, see explanation

Choice D: Plausible because the B squib did not fire and if reactor pressure was lower this would be correct.

SRO Basis: N/A

NOTE:	The SLC pump discharge relief valve should **NOT** actuate with two pumps operating and only one squib valve open unless reactor pressure exceeds 1184 psig, which is possible during an ATWS even with 10 SRVs open.

2. **IF** *SLC A SQUIB VALVE CONTINUITY* **OR** *SLC B SQUIB VALVE CONTINUITY* indicating light on Panel P603 remains on **AND** reactor pressure is greater than or equal to 1184 psig, **THEN PERFORM** the following:

   a. **PLACE** *SLC PUMP A & B* Control Switch, *C41-CS-S1*, to the *SLC PUMP A* **OR** *SLC PUMP B* position. ☐

   b. **ENSURE** the selected SLC pump red indicating light on. ☐

72. G2.3.12 1

Two operators are required to enter a room that is posted as a Locked High Radiation Area (LHRA) to hang a clearance for **scheduled** work.

Which one of the following completes both statements below?

The radiation level at which a LHRA posting is required is ____(1)____ in one hour at 30 centimeters from the radiation source.

The LHRA key is obtained from ____(2)____.

A. (1) >100 mrem
   (2) the Shift Manager

B. (1) >100 mrem
   (2) a RP Technician

C. (1) >1000 mrem
   (2) the Shift Manager

D. (1) >1000 mrem
   (2) a RP Technician

Answer: D

K/A:

G2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)

RO/SRO Rating: 3.2/3.7

Tier 3

K/A Match: This question matches the KA because it is testing the rad requirements for entering a LHRA.

Pedigree: Bank (from Farley)

Objective: LOI-CLS-LP-201-F, Obj. 10
Explain the requirement regarding control of High Radiation Areas per E&RC-0040.

Reference: None

Cog Level: Fundamental

Explanation: Locked High Radiation Area (LHRA) criteria is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1.0 rem (1000 mrem) (10 mSv) in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates or an area accessible to individuals with dose rates in excess of 1.0 rem per hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates but less than 500 rads in one hour at one meter from the radiation source or from any surface penetrated by the radiation. The Shift Manager has a LHRA key for emergency use.

Distractor Analysis:

Choice A: Plausible because this is the limit for a high radiation area not a LHRA. The Shift manager has a key for LHRA but it is for emergency use, not scheduled work.

Choice B: Plausible because this is the limit for a high radiation area not a LHRA. The second part is correct.

Choice C: Plausible because the first part is correct and the Shift manager has a key for LHRA but it is for emergency use, not scheduled work.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

14. **High Radiation Area (HRA):** An area, accessible to individuals in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (100 mrem) (1mSv) in one hour at 30 cm from the radiation source or 30 cm from any surface the radiation penetrates.

15. **Hot Spot (HS):** An accessible, localized source of radiation with a contact dose rate of greater than 100 mrem per hour and greater than five times the general area dose rate at 30 cm.

16. **Licensed Material:** Source material, special nuclear material, or byproduct material received, possessed, used, transferred or disposed of under a general or specific license.

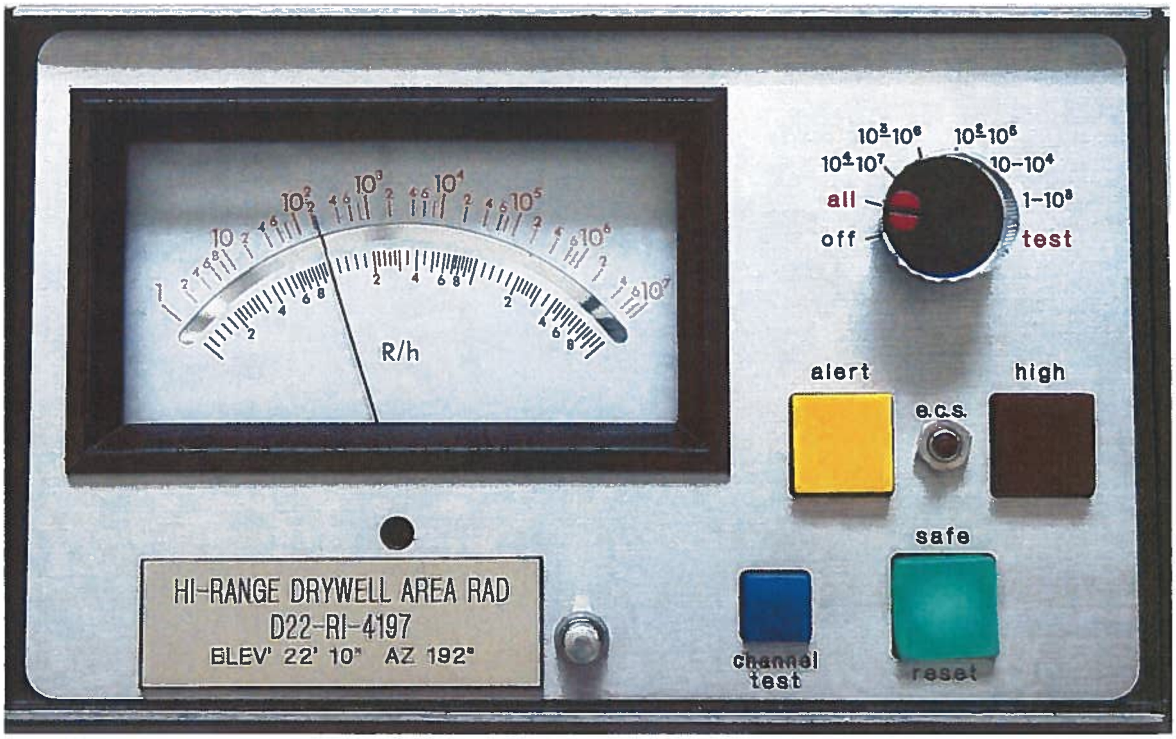
17. **Locked High Radiation Area (LHRA):** An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1.0 rem (1000 mrem) (10 mSv) in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates or an area accessible to individuals with dose rates in excess of 1.0 rem per hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates but less than 500 rads in one hour at one meter from the radiation source or from any surface penetrated by the radiation.

5.1 General Instructions (continued)

12. Entry into HRAs, LHRAs, or VHRAs require a briefing per AD-RP-ALL-2011, Radiation Protection Briefings. {7.1.2}

13. HRA, LHRA less than 10 R/hr, and LHRA greater than or equal to 10 R/hr master keys may be under the control of the Operations Shift Manager for emergency use.

73. G2.3.15 1



Which one of the following identifies the DW radiation value indicated above?

A. ~ 10 R/hr

B. ~ 20 R/hr

C. ~ 100 R/hr

D. ~ 200 R/hr

Answer: D

K/A:

G2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9)

RO/SRO Rating: 2.9/3.1

Tier 3

K/A Match: This question matches the KA as it requires knowledge of the DW rad monitoring system to answer question.

Pedigree: Bank

Objective: CLS-LP-11.1, Obj. 03a
Describe the function/operation of the following: Drywell High Range Radiation Monitors

Reference: None

Cog Level: Fundamental

Explanation: Drywell high range area monitors provide indications of gross fuel failure and are used to determine emergency plan emergency action level associated with abnormal core conditions. With the function switch in the ALL position the upper (red) scale is used, meter readings are taken from the upper scale between 1 - 1,000,000 R/h. Current indication of 200 R/h

Distractor Analysis:

Choice A: Plausible because this is the reading on the bottom scale.

Choice B: Plausible because if function switch is not taken into account the answer could be 20 R/h.

Choice C: Plausible because if the reading on the bottom scale is adjusted by a factor of 10.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

74. G2.4.20 1
A transient has occurred on Unit Two with the following plant conditions:

RPV pressure	1000 psig
Drywell ref leg area temp	197°F
Rx Bldg 50' temp	135°F
Wide Range Level	170 inches (N026A/B)
Shutdown Range Level	160 inches (N027A/B)

(REFERENCE PROVIDED)

Which one of the following completes both statements below concerning the level instruments that can be used to determine reactor water level IAW EOP Caution 1?

Wide Range Level instruments N026A/B ___(1)___ be used.

Shutdown Range Level instruments N0027A/B ___(2)___ be used.

A. (1) can
(2) can

B. (1) can
(2) can NOT

C. (1) can NOT
(2) can

D. (1) can NOT
(2) can NOT

Answer: B
K/A:
G2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.3

Tier 3

K/A Match: The question meets the KA because it is testing the knowledge of EOP Caution 1 which deals with the water level instruments availability to determine level.

Pedigree: New

Objective: LOI-CLS-LP-300-B, Objective 16
Given Plant conditions, determine if the RPV water level instrument is providing valid trending information IAW Caution 1.

Reference: Caution 1 (EOP-01-UG, Att. 19, Att. 22 & Att. 31 pages 1 and 2)

Cog Level: High

Explanation: N026s can be used since reading >20" and RB 50' temp is <140 degrees and N027s cannot be used since in unsafe region for minimum indicated level

Distractor Analysis:

Choice A: Plausible because the first part is correct and if Attachment 19 is only looked at then this is plausible.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because if the temperature was a little higher on the RB 50 foot this would be correct and if Attachment 19 only is looked at for the second part this would be correct.

Choice D: Plausible because if the temperature was a little higher on the RB 50 foot this would be correct and the second part is correct.

SRO Basis: N/A

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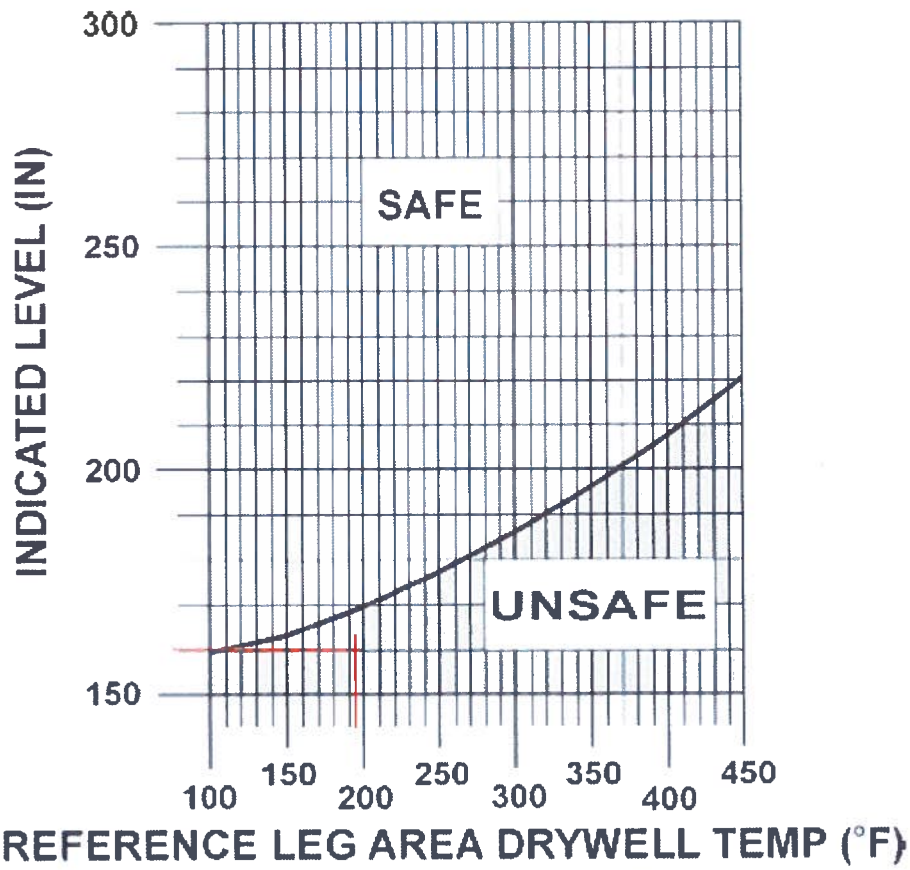
ATTACHMENT 22
Page 1 of 1

<< Shutdown Range Level Instrument (N027A, B) Caution >>



75. G2.4.27 1
A fire has been reported and confirmed in the turbine building breezeway.
A fire hose being used to control/suppress the fire.

Which one of the following completes both statements below IAW 0PFP-013, *General Fire Plan*?

The RO is required to sound the fire alarm and announce the location of the fire ___(1)___.

A call for offsite assistance to the Brunswick County 911 Center ___(2)___ required.

A. (1) ONLY once
(2) is

B. (1) ONLY once
(2) is NOT

C. (1) three times
(2) is

D. (1) three times
(2) is NOT

Answer: C
K/A:
G2.4.27 Knowledge of "fire in the plant" procedures. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.4/3.9

Tier 3

K/A Match: This meets the K/A because it is testing knowledge of the actions contained in the plant fire procedure

Pedigree: New

Objective: FPT-CLS-LP-205
Lesson plan discusses the actions for the control room but no objective is listed.

Reference: None

Cog Level: Fundamental

Explanation: The operator aid (from the General Fire Plan, PFP-013) for the control room operators states to announce the fire location 3 times. The procedure also states to request off site assistance if a fire hose is used for extinguishing the fire.

Distractor Analysis:

Choice A: Plausible because EP announcements are performed once and the second part is correct.

Choice B: Plausible because EP announcements are performed once and the second part because the stem says that the fire is under control.

Choice C: Correct Answer, see explanation

Choice D: Plausible because EP announcements are performed once and the second part because the stem says that the fire is under control.

SRO Basis: N/A

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ATTACHMENT 2
Page 1 of 2

<< (Information Use) - Control Room/Operator Fire Actions >>

1. **Sound** fire alarm, **announce** location of the fire 3 times, then: .... ☐

   - **Announce**: .... ☐

     "Fire brigade muster at the fire house."

   - IF fire is outside the Protected Area,
     THEN announce:
     "All personnel **NOT** involved in fire fighting or direct support activities are to evacuate the involved area immediately." .... ☐

   - IF fire is inside the Protected Area,
     THEN announce:
     "All personnel in the affected area are to evacuate the involved area immediately and report to your normal work location. If your normal work location is inaccessible, report to the O&M lunch room or TAC auditorium as conditions dictate." .... ☐

   - **Announce:**
     "Use of the PA and radio is restricted to emergency fire communications, except as directed by the Unit CRS for operational safety concerns." .... ☐

2. **Announce** the fire over Unit 1 and Unit 2 radio channels .... ☐

c. IF the investigating operator confirms a fire **AND** any of the following conditions exist,
   THEN immediately **request** off site assistance by calling 911: .... ☐

   - Extreme force is necessary to gain entry into fire area .... ☐

76. S209001 1

During a LOCA and LOOP on Unit One, the following plant conditions exist:

An Emergency Depressurization has been performed due to RPV water level

The Reactor Building -17 foot and 20 foot elevations are NOT accessible due to radiation levels.

ALL ECCS pumps are unavailable.

Which one of the following completes the statement below?

The CRS will direct demin water injection to the RPV, IAW 0EOP-01-LEP-01, *Alternate Coolant Injection,* Section:

A. 2.4.3.3a, *Demineralized Water Actions*, Inject demineralized water through Core Spray Loop A

B. 2.4.3.3c, *Demineralized Water Actions*, Inject demineralized water through RHR Loop A

C. 2.4.3.3d, *Demineralized Water Actions*, Inject demineralized water through HPCI

D. 2.4.3.3e, *Demineralized Water Actions,* Inject demineralized water through RCIC

Answer: A

K/A:

209001 Low Pressure Core Spray System

G2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.0

Tier 2 / Group 1

K/A match: This question meets the K/A because an emergency condition exists in the stem (ED followed by no HP injection/high area rad levels). In addition, the SRO is required to have knowledge of the local emergency procedure and that it contains actions the AO must take in the field in order to inject (AO must locally open demin keepfill bypass valves for various injection sources). Contrasting the given conditions, the procedural knowledge of AO field actions, and system knowledge of valve locations will have the operational effect of determining which injection source is viable.

Pedigree: New

Objective: LOI-CLS-LP-300-J Obj 4a Given plant conditions, determine which system should be utilized to restore RPV water level and/or pressure when executing the following:
a. Alternate Cooling Injection Procedure with EOP-01-LEP-01.

Reference: None

Cog Level: High

Explanation: Demineralized water injection requires knowledge from the LEP that the system Keepfill Bypass Valves are to be opened. The only keepfill bypass valve that is not inaccessible due to radiation is the Core Spray Loop A Keepfill Bypass valve. Therefore, Core Spray Loop A is the section to use in order to inject demin water to the RPV.

Distractor Analysis:

Choice A: Correct answer, see explanation.

Choice B: Plausible because it is a section in 2.4.3, it is incorrect because the keepfill bypass valve for RHR loop A is inaccessible due to radiation

Choice C: Plausible because it is a section in 2.4.3, it is incorrect because the keepfill bypass valve for HPCI is inaccessible due to radiation

Choice D: Plausible because it is a section in 2.4.3, it is incorrect because the keepfill bypass valve for RCIC is inaccessible due to radiation

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. Requires the SRO to evaluate the emergency conditions in the stem, and contrast those conditions with the given local emergency procedure sections and then select the appropriate procedure section.

2.4.3 Demineralized Water Actions

NOTE
0E&RC-0040 states that the Shift Manager may grant immediate access to a locked high radiation area to maintain the health and safety of plant personnel or the general public per 10CFR50 54(x) using the keys maintained in the Radiation Protection key locker near the Unit 2 AOG panel ☐

1 **IF AT ANY TIME** a locked high radiation area is entered, without Radiation Protection support, **THEN** promptly notify RP ☐ RO

2 Monitor and control MUD tank level greater than 14 feet ☐ AO

NOTE
Demineralized water transfer pump capacity is 400 gpm ☐

3 Perform for systems **NOT** operating **AND** available to provide injection to RPV ☐ RO

a Inject demineralized water through Core Spray Loop A

(1) Ensure E21-F005A (Inboard Injection Vlv) OPEN ☐ RO

(2) Ensure E21-F004A (Outboard Injection Vlv) OPEN ☐ RO

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2.4.3 Demineralized Water Actions (continued)

NOTE
E21-F028A is located on Reactor Building 50' ☐

(3) Open E21-F028A (Core Spray Loop A Keepfill Station Bypass Valve) ☐ AO

c. Inject demineralized water through RHR Loop A

(1) Ensure E11-F015A (Inboard Injection Vlv) OPEN ☐ RO

(2) Throttle E11-F017A (Outboard Injection Vlv) ☐ RO

NOTE

E11-F082A, E11-F085 and E11-F086A are located on the HPCI mezzanine ☐

(3) Open E11-F082A (RHR Loop A Keepfill Station Bypass Valve) ☐ AO

(4) Open E11-F085 (RHR System Demineralized Water Fill Valve) ☐ AO

AO

d. Inject demineralized water through HPCI

(1) Ensure E41-F012 (HPCI Min Flow Bypass To Torus Vlv) CLOSED ☐ RO

(2) At MCC XDA, Row H1, Compt B24 (HPCI Min Flo BPV To Supp Chamber Valve E41-F012) place breaker OFF ☐ AO

NOTE

E41-V100 is located in NRHR ☐

(3) Open E41-V100 (HPCI Keepfill Station Bypass Valve) ☐ AO

(4) Ensure E41-F007 (Pump Discharge Vlv) OPEN ☐ RO

e. Inject demineralized water through RCIC

(1) Ensure E51-F019 (RCIC Min Flow Bypass To Torus Vlv) CLOSED ☐ RO

2.4.3 Demineralized Water Actions (continued)

(2) At MCC XDB, Row H1, Compt B47 (RCIC Min Flo Bypass To Supp Pool Vlv E51-F019), place breaker OFF ☐ AO

NOTE

E51-V70 is located in SRHR ☐

(3) Open E51-V70 (RCIC Keepfill Station Bypass Valve) ☐ AO

(4) Ensure E51-F012 (Pump Discharge Vlv) OPEN ☐ RO

77. S212000 1

Unit One is at rated power performing 0PT-01.1.6, *Reactor Protection System Manual Scram Test.*

The Reactor Scram System A pushbutton has been depressed.

RPS Trip System A Scram Groups light for groups one, two, three, and four are illuminated

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

The scram pilot valve solenoids associated with these lights are __(1)__.

Tech Spec 3.3.1.1, *Reactor Protection System Instrumentation*, Condition B __(2)__ required to be entered.

A. (1) energized
(2) is

B. (1) energized
(2) is NOT

C. (1) de-energized
(2) is

D. (1) de-energized
(2) is NOT

Answer: B

K/A:

212000 Reactor Protection System

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. (CFR: 41.5 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.4

Tier 2 / Group 1

K/A match: The applicant must interpret the indications (scram group lights) in comparison with the expected conditions for the given action (Sys A pushbutton depressed). The applicant must then use that knowledge to determine whether the system meets the given limited condition for operability as stated in the TS.

Pedigree: New

Objective: LOI-CLS-LP-003 Rev 3 Obj 27 Given plant conditions, determine whether given plant conditions meet minimum Technical Specification requirements associated with the Reactor Protection System.

Reference: TS 3.3.1.1

Cog Level: High

Explanation: **Part 1** When group lights are OFF that is indication that the solenoid valves are de-energized. Groups 2 and 4 remained lit therefore, they are energized. **Part 2** ONLY Condition A and C are required to be entered, due to a failure of the A3 scram channel. Only one required channel is inoperable, but a loss of manual scram function has occurred resulting in RPS trip capability not maintained.

Distractor Analysis:

Choice A: **Part 1** is the correct Answer, see explanation. **Part 2** is plausible because Groups 1 through 4 are also in trip System B (although for the SV-118's), a novice applicant may assume that failure of these solenoids in Trip System A would mean they would not function in Trip System B. In addition, one channel per trip system is required, and since there are only lights for groups 1 through 4 in both trip systems, a novice applicant may assume that all required channels are inop.

Choice B: Correct Answer, see explanation.

Choice C: **Part 1** is plausible because ARI system uses energize to function valves. **Part 2** is plausible because Groups 1 through 4 are also in trip System B (although for the SV-118's), a novice applicant may assume that failure of these solenoids in Trip System A would mean they would not function in Trip System B. In addition, one channel per trip system is required, and since there are only lights for groups 1 through 4 in both trip systems, a novice applicant may assume that all required channels are inop.

Choice D: **Part 1** is plausible because ARI system uses energize to function valves. **Part 2** is the correct Answer, see explanation.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. Requires the SRO to evaluate the failure of PT for RPS, and select the appropriate > 1 hour TS condition.

Table 3.3.1.1-1 (page 3 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
7. Scram Discharge Volume Water Level—High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 108 gallons
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 108 gallons
8. Turbine Stop Valve—Closure	≥ 28% RTP	4	E	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 10% closed
9. Turbine Control Valve Fast Closure, Control Oil Pressure—Low	≥ 28% RTP	2	E	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≥ 500 psig
10. Reactor Mode Switch—Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
	5(a)	1	H	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.9 SR 3.3.1.1.15	NA
	5(a)	1	H	SR 3.3.1.1.9 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

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. OR 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 12 hours

(continued)

ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
B.	————NOTE———— Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. —————— One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
		OR		
		B.2	Place one trip system in trip.	6 hours
C.	One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D.	Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E.	As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 26% RTP.	4 hours

(continued)

DEFINITION OF INSTRUMENT CHANNELS AND TRIP SYSTEMS FOR SELECTED INSTRUMENTS	
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C71A-S3A, B; C72A-S3A, B

INSTRUMENT NUMBER: C71A-S3A, B; C72A-S3A, B
INSTRUMENT NAME: Manual Scram

TS REFERENCE: 3.3.1.1; TRM Table 3.3.1.1-1.11

TRIP CHANNEL: A3 - S3A
B3 - S3B

TRIP SYSTEM: A3 - S3A
B3 - S3B

TRIP LOGIC: A3 and B3 = Reactor scram

Place channel in TRIPPED condition by: Pull fuse

The Reactor Manual Scram relays deenergize the Scram pilot valve solenoids for the RPS Trip System.

- The SV-117 valves are in RPS Trip System A.
- The SV-118 valves are in RPS Trip System B.

Shorting links are normally installed around the auxiliary trip relay contacts in the Manual Scram circuits allowing this Scram signal to be bypassed. These shorting links are located in the back panels and are color coded red for identification.

NOTE:	The shorting links are removed prior to and during the time any control rod is withdrawn (except for control rods removed per Technical Specifications) during operation in refueling mode or during shutdown margin demonstration.

There are two shorting links per RPS Trip System, a total of four for the entire Reactor Protection System.

1.0 PURPOSE

This test is performed to determine the OPERABILITY of the Reactor Protection System Manual Scram function.

2.0 SCOPE

This procedure performs the following:

- A quarterly Channel Functional Test per TS SR 3.3.1.1.9 for Table 3.3.1.1-1 Function 11, Manual Scram.
- Satisfies a portion of the 24 month TS SR 3.3.1.1.15 Logic System Functional Test for Table 3.3.1.1-1 Function 11, Manual Scram.

3.0 PRECAUTIONS AND LIMITATIONS

1. A half-scram signal will exist until RESET .... ☐

4.0 GENERAL INFORMATION

The following annunciators will alarm during the performance of this test:

- A-05, 1-8, Reactor Manual Scram Sys A
- A-05, 2-8, Reactor Manual Scram Sys B

5.0 ACCEPTANCE CRITERIA

1. This test may be considered satisfactory when all of the following criteria are met:

   a. A trip is indicated on RPS A and alarmed on RTGB Panel H12-P603 when C71(C72)-S3A (Manual Reactor Scram System A) push button is depressed.

   b. A trip is indicated on RPS B and alarmed on RTGB Panel H12-P603 when C71(C72)-S3B (Manual Reactor Scram System B) push button is depressed.

   c. The Scram valve solenoids are DE-ENERGIZED when the associated RPS is tripped.

7.0 INSTRUCTIONS

7.1 Test Preparation

1. Obtain Unit CRS permission to perform this test .... ______

2. Ensure all prerequisites listed in Section 6.0, Prerequisites are met .... ______

NOTE

The length of time a half-scram is sealed-in is to be minimized .... ☐

3. **IF** during the performance of this procedure, the expected test results from a half-scram initiation are **NOT** observed, **THEN** immediately reset the half-scram and notify the Unit CRS .... ______

7.2 Manual Scram A Test

1. Depress C71(C72)-S3A (Manual Reactor Scram System A), push button and observe the following actions occur:

   a. Plant Process Computer Event Log displays Manual Scram Channel A Trip (Computer Point D533) .... ______

   b. **IF** the proper Plant Process Computer Event Log (Computer Point D533) was **NOT** received in Step 1.a, **THEN** generate a WO .... ______

   c. Manual Reactor Scram System A push button light comes ON ..... ______

   d. A-05, 1-8, Reactor Manual Scram Sys A, ALARMS .... ______

   e. RPS Trip System A Scram Group lights 1, 2, 3, and 4 located on Panel H12-P603 are OFF, indicating Scram valve solenoids are DE-ENERGIZED .... ______

   f. RPS Trip System A Scram Group lights 1, 2, 3, and 4 located on RPS A Panel H12-P609 are OFF, indicating Scram valve solenoids are DE-ENERGIZED .... ______

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Unit Two is at rated power. A TIP trace is in progress.

TIP D Valve Control Monitor indications are as follows:

Squib Monitor Light	ON
Shear Valve Monitor Light	OFF
TIP Ball Valve OPEN Light	ON
TIP Ball Valve Closed Light	OFF

TIP D Drive Control Unit indications are as follows:

MODE Switch	MANUAL
IN CORE Light	ON

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

Tip Valves are Group __(1)__ PCIVs.

Tech Spec 3.6.1.3, *Primary Containment Isolation Valves (PCIVs)*, Condition __(2)__ is required to be entered.

A. (1) 2
(2) A

B. (1) 2
(2) B

C. (1) 6
(2) A

D. (1) 6
(2) B

Answer: A

K/A:

215001 Traversing In-Core Probe

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.
(CFR: 41.5 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.4

Tier 2 / Group 2

K/A match: The applicant must interpret the given TIP system indications and controls to verify the status of the system. The applicant must then compare those indications and controls with the required conditions for the given mode as stated in the TS to determine the appropriate condition.

Pedigree: New

Objective: LOI-CLS-LP-009.5 Obj 9 Determine whether given plant conditions meet minimum Technical Specification requirements, including the Bases, associated with the Traversing Incore Probe System.

Reference: TS 3.6.1.3

Cog Level: High

Explanation: Part 1: Tip Valves are Group 2 PCIVs. Part 2: Squib Valve Monitor Light On indicates that the squib valve continuity is lost. Therefore, The Shear Valve in the Ball valve and shear valve assembly is inoperable. The ball valve remains operable. Therefore, Condition A is entered for one PCIV inoperable.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because squib Valve Monitor Light On indicates that the squib valve continuity is lost. Therefore, The Shear Valve in the Ball valve and shear valve assembly is inoperable. In addition, a novice applicant may assume that with the MODE switch in MANUAL, the ball valve would be inoperable as well. (However, the only position of the MODE switch that would make the TIP inoperable is OFF.) With Two PCIVS inoperable, condition B would be entered,

Choice C: Part 1 is plausible because group six PCIVs also isolate on LL1 and High DW pressure. Part 2 is plausible because it is correct, see explanation.

Choice D: Part 1 is plausible because group six PCIVs also isolate on LL1 and High DW pressure. Part 2 is plausible because squib Valve Monitor Light On indicates that the squib valve continuity is lost. Therefore, The Shear Valve in the Ball valve and shear valve assembly is inoperable. In addition, a novice applicant may assume that with the MODE switch in MANUAL, the ball valve would be inoperable as well . (However, the only position of the MODE switch that would make the TIP inoperable is OFF.) With Two PCIVS inoperable, condition B would be entered,

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The SRO applicant is required to select the appropriate > 1 hour TS condition based on the status of the TIP system indications.

Table 09.5-2 - Valve Control Monitor Indications

Indication	Comment
Squib Monitor Light	ON indicates that the TIP Shear Valve squib circuit continuity has been lost.
Shear Valve Monitor Light	ON indicates that the squib charge in the TIP Shear Valve has been detonated.
TIP Ball Valve OPEN Light	ON indicates that TIP Ball Valve is OPEN.
TIP Ball Valve CLOSED Light	ON indicates that TIP Ball Valve is CLOSED.
Time Delay Light	ON indicates that the TIP Ball Valve was not OPEN within 6 seconds from when the detector left the in-shield position and the TIP Drive Motor should have stopped.
Purge Light	Indicates that the solenoid for the Indexer Purge System should be energized OPEN.
Fuse F5 Continuity Light	Indicates power to PCIS Group 2 Bus in drawer is available (Fuse F5 is not blown).

Table 09.5-3 - TIP Drive Control Unit Indications

Indication	Comment
DETECTOR POSITION (illuminated digits)	Dynamic digital display of detector position. ("0001" - reference point about one foot behind the Indexer; "9750" - "In Shield position)
CORE LIMIT (illuminated digits)	Static digital display of pre-programmed core top or bottom limits of selected channel.
READY Light	Indicates that Indexer is properly aligned to selected channel.
CORE TOP Light	Detector is at top of core.
IN CORE Light	Detector is above core bottom limit.
IN SHIELD Light	Detector is in Shield Chamber.
SCAN Light	Axial Flux Profile is being recorded.
LOW Speed Light	Detector is being driven at 3 inches per second. (15 feet per minute)
REV (Reverse) Light	Detector moving away from top of core.
FWD (Forward) Light	Detector moving towards top of core.
VALVE Light	ON if TIP Ball Valve is CLOSED.

Table 09.5-4 - P601 TIP Indications

Indication	Comment
TIP Valve Status - Green Light	Green Light ON indicates that each TIP Ball Valve is FULL CLOSED.
TIP Valve Status - Red Light	Red Light ON indicates that a TIP Ball Valve is NOT FULL CLOSED.

Low Speed

OFF Makes low-speed drive a function of detector position

and independent of operator control.

ON Initiates continuous low-speed detector drive.

Core Limit

TOP Permits digital display of selected channel

pre-programmed top-core limit which corresponds to top of active fuel.

BOTTOM As above, except pre-programmed core bottom limit

is displayed. (The core top and bottom numbers are different for each TIP channel because of different lengths of guide tube run).

Mode (Switch S-7)

OFF Deenergizes power supplies in Drive Control Unit.

MANUAL Positions detector in conjunction with the FWD and

REV position of manual switch S-3.

AUTO Permits automatic mode of operation when Auto start

S-2 is pressed.

Manual Valve Control (Spring Return to Closed Position)

CLOSED Permits TIP Ball Valve to open automatically when

mechanism is operated.

OPEN Opens TIP Ball Valve without energizing the Drive

in the TIP Drive Mechanism.

X-Y Recorder (Figure 09.5-14)

Alternate plotting capability via "AFORA" software.

Controls on the X-Y recorder drawer are as follows:

4.0 SYSTEM OPERATION

4.1 Normal Operational Relationships

TIP traces are required to be performed periodically. Technical Specifications require calibration of LPRM detectors at least once every effective full power month (EFPM). Traces are done to obtain new Gain Adjustment Factors (GAF) as calculated by the process computer for each LPRM. These GAFs can be used to adjust the current applied to the LPRM detectors. Current adjustment is required due to loss of detector sensitivity which results from exposure to the fission process in the core. The TIP Detector is run through the dry tube within the LPRM assembly containing the LPRM to be calibrated. A comparison is made between the TIP output and the existing LPRM reading and a GAF is calculated. The current applied to the LPRM detector is adjusted, if necessary.

TIP Detector calibration is also periodically required. Capability for calibration of TIP probes is provided by a common channel which can align each TIP to the center LPRM assembly (28-29) (Figure 09.5-15) and TIP Dry Tube. Each TIP is run through the center LPRM assembly. Readings from the TIPs are compared and the gains are adjusted to meet an average value of the four TIP readings and so that the gains for the TIP channels fall within a specified band.

The TIP System can be operated in an Automatic mode or a Manual mode. OP-09.1 in conjunction with 0ENP-24.15 covers precautions, initial conditions, and specific instructions related to the operation of the TIP System.

Regardless of the mode of operation there are some precautions that must be observed. The TIP Machine should never be turned off with the detector inserted past the TIP Ball Valve. This conditions prevents the isolation logic from retracting the detector and closing the TIP Ball Valve should an isolation signal be received. Also, the TIP Machines should not be left unattended if the detectors are in motion.

4.1.1 Automatic Operation Sequence

The mode switch on the selected Drive Control Unit is placed in AUTO and the manual switch and low speed switch remain in OFF. This gives a permissive to the TIP logic to be run automatically.

The auto start pushbutton (Drive Control Unit) is then pressed. The detector will automatically move from Shield Chamber to the entrance of the Indexer at low speed (3"/sec or 15 ft/min) and will then stop.

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."

ACTIONS

--NOTES--

1. 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.

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 for MSIV leakage not within limit.	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	8 hours

(continued)

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 for MSIV leakage not within limit.	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.	2 hours
C. ------------NOTE------------	C.1 Isolate the affected	8 hours except for

TABLE 12-2
Primary Containment Isolation System Group Isolation
Instrumentation Setpoints

ISOLATION GROUP	ISOLATION SIGNAL	TRIP SETPOINT		NOTES
		Tech Spec. Allowable	Actual (Note 1)	
Group 5	High Steam Flow Low Steam Pressure High Turb Exh Pressure Steam Line Area Hi Temp Steam Line Tunnel High Amb Temp Steam Line Tunnel dT High Equip Area High Temp Equip Area dT High	≤ 275% ≥ 53psig ≤ 6 psig ≤ 175°F ≤ 200°F ≤ 50°F ≤ 175°F ≤ 50°F	220% 70 psig 5 psig 165°F 165°F/190°F 47°F 165°F 47°F	Note 5 Note 4 Note 4 Note 4
Group 6	Low Level #1 High Drywell Pressure Rx Bldg Exhaust Hi Rad Rx Bldg Exhaust Hi Temp High Main Stack Rad	≥ 153" ≤ 1.8 psig ≤ 16 mR/hr N/A ODCM	166" 1.7 psig 4 mR/hr 135°F* ODCM	 Note 6 Note 2
Group 7	Low Steam Pressure AND High Drywell Pressure	≥ 104 psig ≤ 1.8 psig	115 psig 1.7 psig	
Group 8	Low Level #1 High Steam Dome Pressure	≥ 153" ≤ 137 psig	166" 130.8 psig	
Group 9	Low Steam Pressure AND High Drywell Pressure	≥ 53 psig ≤ 1.8 psig	70 psig 1.7 psig	
Group 10	Low Level #3 High Drywell Pressure AND Low Reactor Pressure	N/A	45" 1.7 psig 410 psig	

Note 1: All "Actual" values from TRM
Note 2: Stack radiation high level is calculated in accordance with the Offsite Dose Calculation Manual
Note 3: After a 28.5 minute time delay
Note 4: After 27 minute time delay
Note 5: After a 5 second time delay
*Note 6: Specific "Actual" values from EOP User's Guide, Attachment 1

TABLE 12-2
Primary Containment Isolation System Group Isolation
Instrumentation Setpoints

ISOLATION GROUP	ISOLATION SIGNAL	TRIP SETPOINT		NOTES
		Tech Spec. Allowable Value	Actual (Note 1)	
Group 1	Low Level #3 Main Steam Line High Temp Turbine Bldg Area Hi Temp Main Steam Line High Flow Not in RUN Low Condenser Vacuum Low Steam Pressure	≥ +13" ≤ 197°F N/A ≤ 138% ≤ 33% ≥ 7.5" Hg ≥ 825 psig	45" 190°F 190°F* 137% 30% 10" Hg 835 psig	Note 6 - Unit 2 only
Group 2	Low Level #1 High Drywell Pressure	≥ +153" ≤ 1.8 psig	166" 1.7 psig	
Group 3	Low Level #2 High Diff Flow Area High Temp Area Vent dT High HELB Isolation NRHX Outlet Temp High SLC Initiation	≥ 101" ≤ 73 gpm ≤ 150°F ≤ 50°F ≤ 120°F N/A N/A	105" 43 gpm 140°F 47°F 115°F 135°F* N/A	Note 3 Note 6
Group 4	High Steam Flow Low Steam Pressure High Turb Exh Pressure Steam Line Area Hi Temp Steam Line Tunnel High Amb Temp Steam Line Tunnel dT High Equip Area High Temp	≤ 275% ≥ 104 psig ≤ 9 psig ≤ 200°F ≤ 200°F ≤ 50°F ≤ 175°F	220% 115 psig 7 psig 165°F 165°F/190°F 47°F 165°F	Note 5

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Unit Two is operating at rated power with RHR Loop A operating in suppression pool cooling mode.

A-01 (2-8) *RHR Relay Logic Pwr Failure,* is in alarm due to a blown fuse affecting RHR Logic A ONLY.

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

If a LOCA signal were to occur, 2-E11-F015A, Inboard Injection Vlv, ___(1)___ open automatically on low reactor pressure.

IAW Tech Spec 3.3.5.1, *ECCS Instrumentation*, the required channels ___(2)___ required to be placed in trip within 24 hours.

A. (1) will
(2) are

B. (1) will
(2) are NOT

C. (1) will NOT
(2) are

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

Answer: A

K/A:

219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode

A2 Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

12 Valve logic failure

RO/SRO Rating: 3.0/3.1

Tier 2 / Group 2

K/A Match: This meets the K/A because it is testing whether RHR in suppression pool cooling mode will transfer to LPCI injection mode with a failure of divisional valve logic, and whether the LCO action statement for condition B should be applied.

Pedigree: New

Objective: LOI-CLS-LP-017 Obj 7. Given plant conditions, determine if the RHR system should automatically initiate in the LPCI mode. (LOCT)

Reference: T.S. 3.3.5.1

Cog Level: High

Explanation: Part 1: With the plant at rated power and torus cooling in service on loop A, RHR loop A is not in its normal standby lineup. The loss of Div I RHR relay logic power, will result in the failure of the A loops suppression cooling valves (f024/28) to automatically close on a LPCI initiation. However, the F015A will auto open from the div 2 logic. Part 2: The loss of power to the Div I RHR logic will result in the loss of the functions applicable to condition C.

Distractor Analysis:

Choice A: This is the Correct answer, see explanation.

Choice B: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because a novice candidate might believe the note that required action B.2. is only applicable to functions 3a and 3b also applies to condition B.3.

Choice C: Part 1 is plausible because a candidate might believe a loss of Div I logic would prevent the auto function of the F015a, since the F024 and F028 will not automatically close under these conditions. Part 2 is plausible because it is correct, see explanation.

Choice D: Part 1 is plausible because a candidate might believe a loss of Div I logic would prevent the auto function of the F015a, since the F024 and F028 will not automatically close under these conditions. Part 2 is plausible because a novice candidate might believe the note that required action B.2. is only applicable to functions 3a and 3b also applies to condition B.3.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The SRO applicant is required to select whether an action statement is applicable given a loss of Div I RHR logic.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	----------NOTE---------- Only applicable for Functions 3.a and 3.b. -----------------------------	
		Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	AND		
	B.3	Place channel in trip.	24 hours
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	---------NOTES--------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c, 1.d, 2.c, 2.d, and 2.f. -----------------------------	
		Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	AND		
	C.2	Restore channel to OPERABLE status.	24 hours

(continued)

Table 3.3.5.1-1 (page 1 of 4)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level—Low Level 3	1,2,3, $4^{(a)}$, $5^{(a)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 13 inches
b. Drywell Pressure—High	1,2,3	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.8 psig
c. Reactor Steam Dome Pressure—Low	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
	$4^{(a)}$, $5^{(a)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
d. Core Spray Pump Start—Time Delay Relay	1,2,3, $4^{(a)}$, $5^{(a)}$	2 1 per pump	C	SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 14 seconds and ≤ 16 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level—Low Level 3	1,2,3, $4^{(a)}$, $5^{(a)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 13 inches
b. Drywell Pressure—High	1,2,3	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.8 psig

(continued)

<< Plant Effects from Loss of DC Distribution Panel 3A(4A) >>

RCIC:	Will **NOT** shutdown on reactor high water level, inboard isolation logic INOPERABLE (E51-F007, -F031, and -F062 will **NOT** auto close). Valves E51-F005 and -F025 fail closed.
ADS:	ADS Logic B is INOPERABLE. ADS will initiate from ADS Logic A if Core Spray Pump B or both RHR Loop B pumps are operating.
HPCI:	Will **NOT** auto initiate, outboard isolation logic INOPERABLE (E41-F003, -F041, and -F075 will **NOT** auto close), HPCI flow controller and EGM INOPERABLE (no flow control or indication), HPCI trip logic INOPERABLE, valves E41-F053, -F054, and -F026 fail closed. HPCI isolation is required in accordance with APP 1(2)-A-01 5-5.
CS Loop A:	Will **NOT** auto initiate (manual operation possible but minimum flow valve will **NOT** auto open, and injection valves can **NOT** be opened simultaneously).
RPS Logic A:	Will **NOT** have 10 second time delay prior to reset of full scram, power lost to backup scram valves.
RHR Loop A:	Will auto initiate from RHR Logic B, however the following effects exist: Pumps can **NOT** be restarted if stopped by control switch, pumps will **NOT** trip on No Suction Path Interlock, LOCA interlocks **NOT** functional, min flow valve will **NOT** auto open, Loop A Containment Spray can **NOT** be initiated. If a loss of DC Distribution Panel 3B(4B) has also occurred, RHR Loop A will **NOT** auto initiate (manual operation possible).

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Unit One was operating at power when a Group 1 isolation and reactor scram occurred.

Reactor pressure is 950 psig and being manually controlled by SRVs.

An SRV is stuck open with a stuck open **SRV tailpipe** vacuum breaker.

Torus and Drywell sprays have been initiated IAW PCCP

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

The SRV is discharging through the open vacuum breaker directly into the __(1)__.

The **highest** EAL classification for this event is a(n) __(2)__.

A. (1) drywell
(2) Alert

B. (1) drywell
(2) Site Area Emergency

C. (1) suppression chamber air space
(2) Alert

D. (1) suppression chamber air space
(2) Site Area Emergency

Answer: A

K/A:

239002 Safety Relief Valves

A2 Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

01 Stuck open vacuum breakers

RO/SRO Rating: 3.0/3.3

Tier 2 / Group 1

K/A match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
The applicant must determine that the effect of the stuck open vacuum breaker on the SRV is that it is now bypassing the pressure suppression function and discharging directly into the DW. The applicant must also determine that the ultimate effect of the given conditions is that the RCS barrier has been lost, and that the consequences of this conditions and its potential effects on the health and safety of the public are mitigated by declaring an ALERT IAW with 0PEP-2.1.

Pedigree: New

Objective: LOI-CLS-LP-020 Obj 15e Given plant conditions, predict how ADS/SRVs will be affected by the following: Failure of the SRV tailpipe vacuum breakers.

Reference: 0PEP-02.1

Cog Level: High

Explanation: Part 1: SRV tailpipe vacuum breakers relieve to the drywell, therefore a stuck open tailpipe breaker with a stuck open SRV would discharge steam directly into the drywell. Part 2: With a stuck open relief valve and stuck open vacuum breaker, a LOCA is occurring. With Torus pressure sufficient to warrant drywell sprays (11.5 psig) drywell pressure has more than exceeded 1.7 psig. Therefore the conditions for a loss of the RCS barrier (Primary Containment Pressure >1.7psig due to RCS leakage) are met and an ALERT should be declared based on FA1.1 loss of RCS barrier.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because a group 1 isolation has occurred which is a primary containment isolation signal, an unisolable LOCA is occurring, and a novice candidate might assume that this meets the conditions for a loss of the containment barrier (Unisolable direct downstream pathway to the environment exists after primary containment isolation signal), this loss coupled with the loss in the explanation would meet the criteria for a loss of two barriers, and a declaration of an SAE IAW FS1.1 They also might think that this condition would not be consistent with a LOCA.

Choice C: Part 1 is Plausible because with a failed open Suppression Chamber to Drywell Vacuum Breaker would allow DW steam to go directly to the suppression Chamber Air Space. in addition, a failed open SRV with a broken tailpipe and broken downcomer would directly pressurize the torus air space. Part 2 is plausible because it is correct, see explanation.

Choice D: Part 1 is Plausible because with a failed open Suppression Chamber to Drywell Vacuum Breaker would allow DW steam to go directly to the suppression Chamber Air Space. in addition, a failed open SRV with a broken tailpipe and broken downcomer would directly pressurize the torus air space. Part 2 is plausible because a group 1 isolation has occurred which is a primary containment isolation signal, an unisolable LOCA is occurring, and a novice candidate might assume that this meets the conditions for a loss of the containment barrier (Unisolable direct downstream pathway to the environment exists after primary containment isolation signal), this loss coupled with the loss in the explanation would meet the criteria for a loss of two barriers, and a declaration of an SAE IAW FS1.1. They also might think that this condition would not be consistent with a LOCA.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] The SRO applicant is required to select the appropriate ALERT emergency classification based on the given conditions which equates to a loss of the RCS barrier.

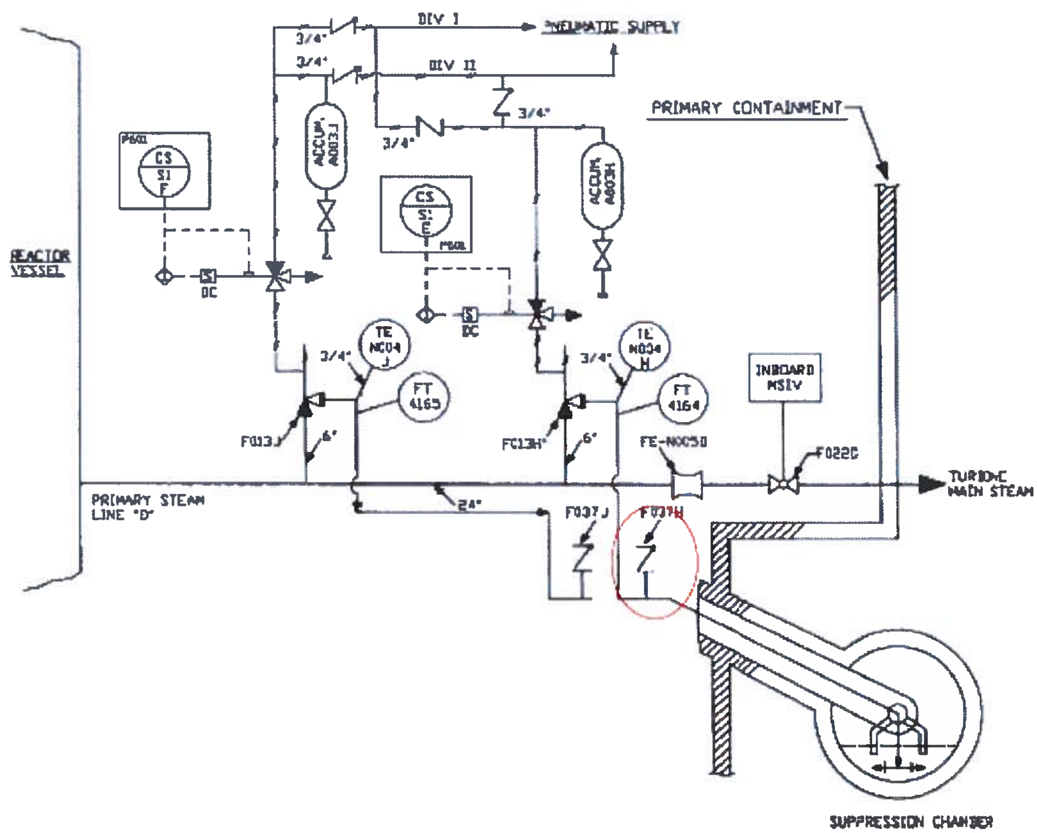

FIGURE 20-2
Typical SRV Piping Arrangement

permissives.

4.2.2 Failure of the SRV Tailpipe Vacuum Breakers

In the event an SRV tailpipe vacuum breaker fails in the open position, the result is a direct path of steam from the reactor to the drywell (i.e., LOCA). In the event an SRV vacuum breaker fails in the closed position, the result is the possible creation of a vacuum in the tailpipe upon closure of an SRV, resulting in the drawing of water into the SRV tailpipe.

FS1.1 1 2 3	FA1.1 1 2 3
Loss or potential loss of any two barriers (Table F-1)	Any loss or any potential loss of either Fuel Clad or RCS (Table F-1)

Table F-1 Fission Product Barrier Threshold Matrix

	Reactor Coolant System Barrier		Containment
ss	Loss	Potential Loss	Loss
ied	1. RPV level cannot be restored and maintained > TAF or cannot be determined	None	None
	1. UNISOLABLE break in any of the following: - Main steam line - HPCI steam line - RCIC steam line - RWCU - Feedwater 2. Emergency Depressurization is required	1. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: • One or more Secondary Containment area radiation Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-3) • One or more Secondary Containment area temperature Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-1)	1. UNISOLABLE primary system leakage that results in exceeding one or more Secondary Containment area temperature Maximum Safe Operating Limits (0EOP-03-SCCP Table SC-1)
	1. Primary Containment pressure > 1.7 psig due to RCS leakage	None	1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise 2. Primary Containment pressure response not consistent with LOCA conditions
	1. Drywell radiation > 27 R/hr with reactor shutdown	None	None
	None	None	1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal

and assumptions consistent with the DBA-LOCA analysis.

4.2.7 Containment Response with Vacuum Breaker Failure

1. **Suppression Chamber to Drywell Vacuum Breakers Failed Open**

   Steam flows from drywell to suppression chamber through the open vacuum breakers equalizing pressure between the two immediately. The steam is not forced through the water of the suppression pool; therefore it will operate only as a surface condenser. As a result, the drywell pressure will probably exceed the design pressure. To prevent this occurrence, light indication is provided for each vacuum breaker. It indicates if the valve is off its seat and is displayed in the Control Room.

2. **Suppression Chamber to Drywell Vacuum Breakers Failed Closed**

   The steam in the drywell will condense and drywell pressure will decrease. With vacuum breakers failed shut, pressure cannot equalize between the suppression pool and the drywell. The pressure in the drywell may decrease such that the suppression chamber to drywell differential pressure limit (10 psid) is exceeded. Vent pipe buckling can occur if suppression chamber pressure is 10 psia greater than drywell pressure. To prevent this situation, the vacuum breakers are operationally checked periodically and 133% capacity is provided with ten vacuum breakers.

81. S261000 1

Unit Two is operating at rated power.

Subsequently, a Div I pneumatic leak occurs causing drywell pressure to rise to 1.9 psig.

Which one of the following completes both statements below?

The SBGT trains ___(1)___ running.

The Div I pneumatics are required be isolated IAW ___(2)___.

A. (1) are NOT
   (2) 0EOP-01-SEP-16, *Drywell Systems Isolation*

B. (1) are NOT
   (2) 0AOP-20.0, *Pneumatic (Air/Nitrogen) System Failures*

C. (1) are
   (2) 0EOP-01-SEP-16, *Drywell Systems Isolation*

D. (1) are
   (2) 0AOP-20.0, *Pneumatic (Air/Nitrogen) System Failures*

Answer: C

K/A:

261000 Standby Gas Treatment System

A2 Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

09 Plant air system failure

RO/SRO Rating: 2.4/2.6

Tier 2 / Group 1

K/A match: A failure of plant air in the drywell would require that the drywell be vented due to increasing DW pressure. In this case, since the impact of the failure is DW pressure increasing > 1.7 psig, the specific impact on SBGT is that it can not be used to vent containment. The procedure to mitigate the consequences of the failure is the leak is required to be isolated using SEP-16. (NOTE: the conditions are NOT entry conditions for the procedures)

Pedigree: New

Objective: LOI-CLS-LP-046A Obj 13. Predict the effect that a loss or malfunction of the Pneumatic System would have on plant operation.

Reference: None

Cog Level: High

Explanation: With drywell pressure >1.7 psig, 2OP-10 cannot be used to vent the drywell. SBGT has automatically started at 1.7 psig. With drywell pressure > 1.7 psig, PCCP directs the use of SEP-16 to isolate containment leaks, and SEP 16 contains the steps to isolate DIV I pneumatics.

Distractor Analysis:

Choice A: Part 1 is plausible because PCCP directs 2OP-10 to control DW pressure <1.7 psig. Part 2 is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible because PCCP directs 2OP-10 to control DW pressure <1.7 psig. Part 2 is plausible because AOP-20 directs the isolation of various pneumatic sources, but while executing the EOPs the EOPs take precedence.

Choice C: Correct Answer, see explanation.

Choice D: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because AOP-20 directs the isolation of various pneumatic sources, but while executing the EOPs the EOPs take precedence.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. The applicant is required to select the appropriate procedure for venting the drywell and isolating pneumatics based on the given conditions.

DRYWELL SYSTEMS ISOLATION	0EOP-01-SEP-16
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2.2 <u>Drywell Pneumatics Isolation</u>

2.2.1 Manpower Required

- 1 Reactor Operator

2.2.2 Special Equipment

None

2.2.3 Operator Actions

NOTE
If <u>both</u> divisions are isolated MSIV and SRV operation will be limited to accumulators. ☐

1. <u>IF</u> Division I pneumatic leakage suspected, <u>THEN:</u>

   a. Notify CRS. ☐ RO

   b. Close RNA-SV-5262 (Div I Non-Intrpt RNA). ☐ RO

   c. Close RNA-SV-5253 (Div I Bu N2 Supp To DW Isol Vlv) ☐ RO

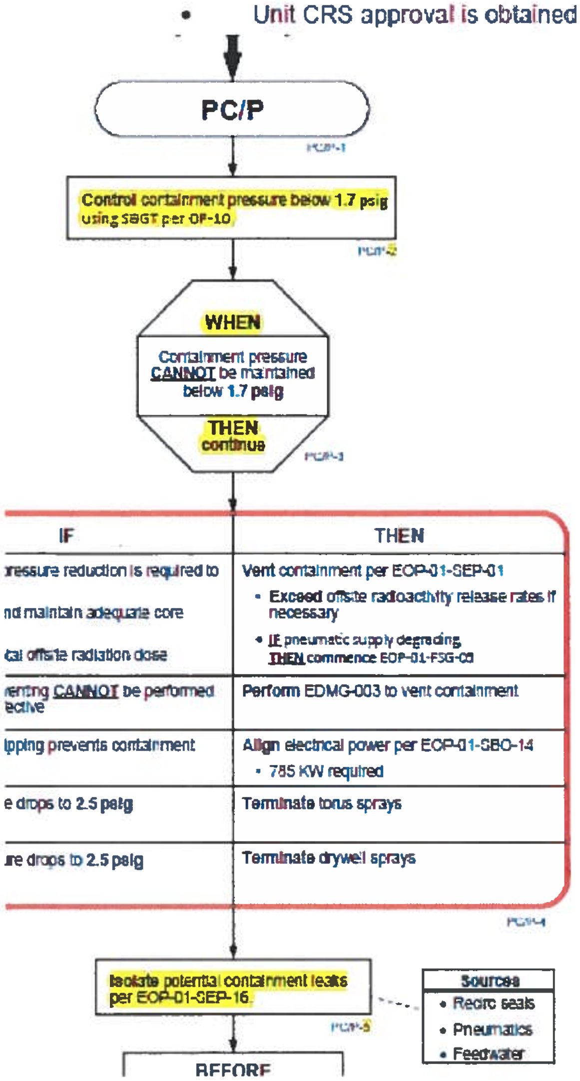
2. IF Division II pneumatic leakage suspected

6.3.2 Venting Containment Via SBGT

Date/Time Started______________

1. Confirm the following Initial Conditions are met:

- Drywell pressure has risen to greater than 0.15 psig. .... ________
- SBGT System is in STANDBY in accordance with Section 6.1.1. .... ________
- One of the following:
  - ◊ Plant stack radiation monitor is in service and CAC-CS-5519 (CAC Purge Vent Isol Ovrd) is in OFF. ..... ________
  - ◊ E&C has sampled the drywell atmosphere and has determined that it is suitable for release. .... ________
- Unit CRS approval is obtained prior to venting. .... ________



82. S262001 1

Unit One is operating at rated power.
Unit Two is in MODE 5 with UAT backfeed established.
A main generator backup lockout occurs on Unit One.

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

All four diesels ___(1)___ automatically start.

IAW Unit One Tech Spec 3.8.1, *AC sources Operating*, Condition E ___(2)___ required to be entered.

A. (1) will
   (2) is

B. (1) will
   (2) is NOT

C. (1) will NOT
   (2) is

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

Answer: D

K/A:

262001 A.C. Electrical Distribution

A2 Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

09 Turbine/generator trip

RO/SRO Rating: 2.9/3.0

Tier 2 / Group 1

K/A Match: This meets the K/A because the candidate is required to predict the status of the EDGs and determine whether the appropriate TS condition is entered.

Pedigree: New

Objective: LOI-CLS-LP-050 Obj 17 Given plant conditions, determine the required action(s) to be taken in accordance with Technical Specifications associated with the 230 KV Electrical Distribution System. **(LOCT)** (SRO Only)

Reference: T.S. 3.8.1 (blank out the LCO statement and Applicability)

Cog Level: High

Explanation: Part 1: DGs do not auto start on a generator backup lockout. Part 2: Condition E is not entered because a loss of two offsite circuits has not occurred. Only one offsite circuit is lost, the Unit One UAT.

Distractor Analysis:

Choice A: Part 1 is plausible because all four DGs start on a generator primary lockout. Part 2 is plausible because if UAT was not in backfeed on Unit Two this would be the case.

Choice B: Part 1 is plausible because all four DGs start on a generator primary lockout. Part 2 is plausible because it is correct, see explanation.

Choice C: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because if UAT was not in backfeed on Unit Two this would be the case.

Choice D: Correct Answer, see explanation.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. Part two requires knowledge of the TS 3.8.1 bases to determine if Unit Two on UAT backfeed can qualify as an offsite source. In addition it requires the appropriate determination of the TS condition application.

3.2.6 Main Transformer, UAT, and Main Generator Protection

1. The Main Generator, MPT, and UAT are all three protected by the Generator/Transformer Primary (86GP) and Backup (86GB) Lockout Relays. The Main Generator is provided additional protection by the Main Generator Differential Lockout Relays (86G).

   a. Generator/Transformer Primary, Backup, and Main Generator Differential Relay trip actions are as follows:

      (1)Main Generator Output breakers trip and lock out.

      (2)Main turbine trips.

      (3)Main generator exciter field breaker trips and locks out.

      (4)UAT 4160 supply breakers to B, C and D buses trip and lock out.

      (5)SAT feeders to C and D buses auto close.

      (6)Four diesel generators auto start for the Main Generator Differential Lockout or the Generator/Transformer Primary Lockout. They do not auto start for a Generator/Transformer Backup Lockout.

Offsite power is supplied to the 230 kV switchyards from the transmission network by eight transmission lines. From the 230 kV switchyards, two qualified electrically and physically separated circuits provide AC power, through either a startup auxiliary transformer (SAT) or backfeeding via a unit auxiliary transformer (UAT), to 4.16 kV BOP buses. A single circuit path (master/slave breakers and interconnecting cables) from each BOP bus provides offsite power to its associated downstream 4.16 kV emergency bus. A detailed description of the offsite power network and circuits to the onsite Class 1E emergency buses is found in the UFSAR, Sections 8.2 and 8.3 (Ref. 2).

A qualified offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from either 230 kV bus (bus A or B) to the onsite Class 1E emergency buses.

The Unit 1 main generator provides the normal source of power to 4.16 kV emergency buses E1 and E2 via its respective UAT. The Unit 2 main generator provides the normal source of power to 4.16 kV emergency buses E3 and E4 via its respective UAT. In the event of a

(continued)

BACKGROUND (continued)

unit trip, an automatic transfer from the normal circuit (main generator output via the UAT) to the respective unit SAT occurs resulting in the SAT supplying power to two 4.16 kV emergency buses. As such, the Unit 1 SAT provides the preferred source of power to emergency buses E1 and E2 and the Unit 1 UAT (backfeed mode) is the alternate source of power to emergency buses E1 and E2. The Unit 2 SAT provides the preferred source of power to emergency buses E3 and E4 and the Unit 2 UAT (backfeed mode) is the alternate source of power to emergency buses E3 and E4. Each UAT can only be considered a qualified offsite source if it is capable of being powered from the 230 kV switchyard (Ref. 3).

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 Unit 1 qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Four diesel generators (DGs); and

c. Two Unit 2 qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable to DGs.

E. Two or more offsite circuits inoperable for reasons other than Condition B.	E.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable. <u>AND</u> E.2 Restore all but one offsite circuit to OPERABLE status.	12 hours from discovery of Condition E concurrent with inoperability of redundant required feature(s) 24 hours

(continued)

83. S271000 1

Unit Two is operating at rated power.

> UA-48 (5-4) *AOG System Bypass,* has been alarming for 1 minute due to High-High off gas flow

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

AOG-XCV-142, Guard Bed Isolation Valve, __(1)__ automatically close.

ODCM 7.3.10, *Gaseous Radwaste Treatment System*, Condition A entry __(2)__ required.

A. (1) will
   (2) is

B. (1) will
   (2) is NOT

C. (1) will NOT
   (2) is

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

Answer: C

K/A:

271000 Offgas System

A2 Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

10 Offgas system high flow

RO/SRO Rating: 3.1/3.3

Tier 2 / Group 2

K/A match: The applicant is required to predict the status of the AOG system (XCV-142) based on high-high offgas flow, and the required procedural actions (i.e ODCM).

Pedigree: New

Objective: LOI-CLS-LP-030 Obj 9 Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM and COLR, determine the required action(s) to be taken in accordance with Technical Specifications, the TRM or ODCM associated with the Condenser Air Removal/Augmented Offgas System.

Reference: ODCM 7.3.10

Cog Level: High

Explanation: Part 1: With XCV-142 remaining open the probable cause for UA-48 (5-4) is off gas flow high, all other conditions (besides circuit failure) for this alarm would cause a closure of the XCV-142. Part 2: with UA-48 (5-4) in alarm, HCV-102 is open. This would bypass the AOG portion of the Gaseous Radwaste Treatment System, leading to reduced hold up times and increased main stack rad levels. ODCM 7.3.10 requires AOG in operation so the comp measure is required.

Distractor Analysis:

Choice A: Part 1 is plausible because high-high cooler condenser condensate level would also cause UA-48 (5-4) to alarm, but it would also close the XCV-142. Part 2 is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible because high-high cooler condenser condensate level would also cause UA-48 (5-4) to alarm, but it would also close the XCV-142. Part 2 is plausible because with the opening of the HCV-102, an additional flowpath is provided around the charcoal adsorbers and the normal flowpath remains in service.

Choice C: This is the Correct answer, see explanation.

Choice D: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because with the opening of the HCV-102, an additional flowpath is provided around the charcoal adsorbers and the normal flowpath remains in service.

SRO Basis: Conditions and limitations in the facility license. [10 CFR 55.43(b)(1). Requires the SRO applicant to have basis knowledge to determine whether ODCM compensatory actions are required based on plant status of the AOG system.

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AOG SYSTEM BYPASS

AUTO ACTIONS

1. *AOG SYSTEM BYPASS VALVE, AOG-HCV-102*, opens

CAUSES

1. High hydrogen - Train A
2. High hydrogen - Train B
3. High-high cooler condenser condensate level
4. High-high off-gas flow
5. Circuit failure

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COOLER CNDSR DRN LEVEL HI AOG SYS BYP

AUTO ACTIONS

1. *GUARD BED ISOLATION VALVE, AOG-XCV-142*, closes
2. *AOG SYSTEM BYPASS VALVE, AOG-HCV-102*, opens

DISCHARGE H2 CONC HIGH

AUTO ACTIONS

1. Isolation to AOG System. (Closes XCV-148, 147, 142, 143, and 141 after a 30 second time delay)
2. Open AOG-HCV-102.

CAUSES

B 7.3.10 GASEOUS RADWASTE TREATMENT SYSTEM

BASES

This requirement provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The GASEOUS RADWASTE TREATMENT SYSTEM refers to the 30-minute offgas holdup line, stack filter house filtration, and the Augmented Off-Gas-Treatment System.

7.3.10 GASEOUS RADWASTE TREATMENT SYSTEM

ODCMS 7.3.10 The GASEOUS RADWASTE TREATMENT SYSTEM shall be in operation.

APPLICABILITY: Whenever the Main Condenser Air Ejector (evacuation) System is in operation.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. GASEOUS RADWASTE TREATMENT SYSTEM not in operation.	A.1 Place GASEOUS RADWASTE TREATMENT SYSTEM in operation.	7 days
B. NOTE Required Compensatory Measure B.1 shall be completed if this Condition is entered. Required Compensatory measure and associated Completion Time not met.	B.1 Submit a Special Report to the NRC that identifies the required inoperable equipment and the reasons for the inoperability, corrective actions taken to restore the required inoperable equipment to OPERABLE status, and a summary description of the corrective actions taken to prevent recurrence.	30 days

Condenser Air Inleakage

CONDENSER AIR INLEAKAGE: MODE 1				
DIAGNOSTIC PARAMETER	**SAMPLE FREQUENCY**	**DIAGNOSTIC LIMIT**	**REMARKS**	**REQUIRED ACTIONS IF LI EXCEEDED**
Air In-leakage, scfm		< 50	Limit applies above 50% power. At 150 scfm the AOG System Bypass valve (AOG-HCV-102) will open. Compensatory measures are required by ODCM 7.3.10.	Develop and implement an in-lea plan.

84. S295001 1

Unit One is operating at 88% power with the following conditions:

Jet Pump Flow Loop A (B21-R611A)	29 Mlbs/hr
Jet Pump Flow Loop B (B21-R611B)	33 Mlbs/hr
Total Core Flow (U1CPWTCF)	62 Mlbs/hr

Which one of the following completes both statements below IAW Tech Spec 3.4.1, *Recirculation Loops Operating,* and Bases? (consider each statement separately)

The current Jet Pump Flow mismatch ____(1)___.

If Jet Pump Flows are **not** matched within limits, then the loop with the ___(2)____ must be considered not in operation.

A. (1) is within limits
   (2) lower flow

B. (1) is within limits
   (2) higher flow

C. (1) is not within limits
   (2) lower flow

D. (1) is not within limits
   (2) higher flow

Answer: C

K/A:

295001 Partial or Complete Loss of Forced Core Flow Circulation

AA2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : (CFR: 41.10 / 43.5 / 45.13)

05 Jet pump operability

RO/SRO Rating: 3.1/3.4

Tier 1 / Group 1

K/A match: This question has the SRO candidate determine jet pump operability based on given core flow conditions.

Pedigree: Bank NRC 10-1

Objective: CLS-LP-002*34
Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM, and COLR determine the required action(s) to be taken in accordance with Technical Specifications associated with the Reactor Recirculation System. (SRO/STA only)

Reference: None

Cog Level: High

Explanation: Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Jet pump loop flow mismatch should be maintained within the following limits:

- jet pump loop flows within 10% (maximum indicated difference 7.5 x10^6 lbs/hr) with total core flow less than 58 x10^6 lbs/hr
- jet pump loop flows within 5% (maximum indicated difference 3.5 x10^6 lbs/hr) with total core flow greater than or equal to 58 x10^6 lbs/hr

Distractor Analysis:

Choice A: Plausible because flow mismatch is within limits for lower reactor power level.

Choice B: Plausible because flow mismatch is within limits for lower reactor power level and because the belief that the higher flow loop will experience excessive vibration could cause them to select the "higher flow" response

Choice C: Correct Answer, see explanation.

Choice D: Plausible because the belief that the higher flow loop will experience excessive vibration could cause them to select the "higher flow" response

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The candidate is required to determine whether jet pump flow is within the limits and then use TS bases information to determine which loop is considered not in operation.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	-------------------NOTE------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation: a. ≤ 10% of rated core flow when operating at < 75% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 75% of rated core flow.	24 hours

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loops are within the allowable limits for mismatch. At low core flow (i.e., < 75% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can, therefore, be allowed when core flow is < 75% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of the percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered not in operation. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

REFERENCES

1. UFSAR, Section 5.4.1.3.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36(c)(2)(ii).

85. S295013 1

Unit Two is performing 2EOP-01-SBO, *Station Blackout*, as the blacked out unit.

The control room instrumentation for torus temperature is unavailable.

Which one of the following completes both statements below?

The CRS will direct torus temperature monitoring locally using ___(1)___.

IAW 0OI-37.8, *Primary Containment Control Procedure Basis Document*, RCIC can operate without equipment damage with a suction temperature up to ___(2)___.

A. (1) 0EOP-01-FSG-08, *Flex Instrumentation*
   (2) 145°F

B. (1) 0EOP-01-FSG-08, *Flex Instrumentation*
   (2) 190°F

C. (1) 0EOP-01-SBO-01, *Plant Monitoring*
   (2) 145°F

D. (1) 0EOP-01-SBO-01, *Plant Monitoring*
   (2) 190°F

Answer: D

K/A:
295013 HIGH SUPPRESSION POOL TEMPERATURE
AA2 Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE : (CFR: 41.10 / 43.5 / 45.13)
02 Localized heating/stratification

RO/SRO Rating: 3.2/3.5

Tier 1 / Group 2

K/A match: This question requires the applicant to determine that with the unavailability of average suppression pool temperature in the MCR during a SBO, individual temperature detectors can be read at the RSDP, by using TR-778 points 6 and 7 and then direct the appropriate procedure or attachment to accomplish this task. In addition, it requires the candidate to know the consequences of localized heating at the RCIC suction.

Pedigree: New

Objective: LOI-CLS-LP-004-A, Obj 19 - Given plant conditions determine the required action(s) to be taken in accordance with Technical Specifications, associated with the Primary Containment System. (SRO/STA only) (LOCT)

Reference: None

Cog Level: Fundamental

Explanation: Part 1: EOP-01-SBO-01 states, "IF Control Room containment parameters NOT available, THEN: Monitor containment parameters at the RSDP, Attachment 3, Instruments Available With Loss of All AC Power and UPS Deenergized." Part 2: 0OI-37.8 states, "RCIC can operate without damage up to 190°F based on EC 96336."

Distractor Analysis:

Choice A: Part 1 is plausible because the use of FSG-08 is used if local monitoring is required and no indication available on the RSDP. Part 2 is plausible because while performing AOP-32, RCIC suction is swapped to the torus at 145°F.

Choice B: Part 1 is plausible because the use of FSG-08 is used if local monitoring is required and no indication available on the RSDP. Part 2 is plausible because it is correct, see explanation.

Choice C: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because while performing AOP-32, RCIC suction is swapped to the torus at 145°F

Choice D: Correct answer, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. This question requires the SRO candidate to select the appropriate procedure to use for suppression pool temperature monitoring when instrumentation is lost in the MCR during a SBO.

2.3 Monitoring Actions (continued)

NOTE
B21-LI-R604B and C32-PR-R609 will **NOT** be available on the RTGB ☐

2. **IF** RPV level will be monitored on B21-LI-R604BX (N026B), at RSDP,
   THEN: ☐ RO

   - Place B21-CS-3345 (Reactor Water Level Normal/Local Switch) to LOCAL ☐ AO

3. **IF AT ANY TIME** local monitoring required,
   THEN obtain as needed per EOP-01-FSG-08: ☐ RO

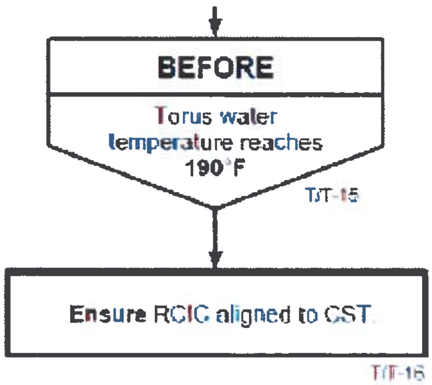
   - RPV Level ☐ AO
   - RPV Pressure ☐ AO
   - Drywell Pressure ☐ AO
   - Torus Pressure and Level ☐ AO
   - Containment Temperature ☐ AO

4. **IF** Control Room containment parameters **NOT** available,
   THEN:

   a. Monitor containment parameters at the RSDP, Attachment 3, Instruments Available With Loss of All AC Power and UPS Deenergized ☐ AO

   b. Periodically notify Control Room of values and trend ☐ AO

5.9 Steps T/T-15 and T/T-16



The normal alignment and preferred suction source for RCIC is normally the CST. This alignment may be altered during SBO conditions, Extended Loss of AC Power (ELAP) conditions or for operational considerations. RCIC is the preferred injection system for ELAP conditions and may be preferred during SBO conditions if lower injection flow and longer system operation time is desired to limit battery depletion. In addition, RCIC may be in service for RPV pressure control.

As stated in Caution 4, operation of RCIC with suction temperature above 190°F may result in equipment damage since the lube oil and control oil is cooled by the water being pumped. Very high lube oil temperatures can result in loss of lubricating qualities in the oil and cause damage to the bearings.

RCIC can operate without damage up to 190°F based on EC 96336. However, if SRVs are being used for RPV pressure control, torus water temperature may eventually become incompatible with long term RCIC operation. Therefore RCIC suction is aligned to the CST before the torus water temperature reaches 190°F.

DRYWELL/SUPP POOL HIGH TEMP

NOTE:	This procedure is only to be used in conjunction with 0AOP-32, Plant Shutdown From Outside Control Room.

AUTO ACTIONS

1. Low RBCCW pressure starts idle RBCCW pump.

CAUSE

1. Insufficient number of drywell coolers in service.
2. Drywell purge exhaust fans not operating properly.
3. Improper operation of RBCCW System.
4. Drywell equipment drain heat exchangers not working properly.
5. Excessive drywell equipment leakage.
6. Circuit malfunction.

OBSERVATIONS

1. Verify proper operation of drywell coolers.
2. Verify proper operation of drywell purge exhaust fans.
3. Verify proper operation of RBCCW System.
4. Verify proper operation of drywell equipment drain heat exchangers.
5. Drywell pressure.

ACTIONS

1. Place RHR Loop B in suppression pool cooling per 0AOP-32.
2. If suppression pool average water temperature exceeds 145°F, then line up the RCIC suction to the CST.
3. Identify and isolate source of drywell leakage.
4. If desired, UA-29 2-5 may be cleared by performing 0OI-63, Section 6.2, Step 6, SV100 Alarm Acknowledge Function, at 2-CAC-TR-778.
5. If a circuit malfunction is suspected ensure that a WR is prepared.

DEVICE/SETPOINTS

CAC-TE-778-1, 3, 4 (Drywell)	200°F ± 8°F
CAC-TE-778-5, 6, 7 (Suppression Pool)	145°F ± 8°F

86. S295015 1

Unit One is performing the ATWS Procedure with the following conditions:

A-05 (2-6) *Reactor Vess Lo Level Trip,* is illuminated
A-06 (1-6) *Reactor Vess Lo Lo Water Level Sys A,* is NOT illuminated
A-06 (2-6) *Reactor Vess Lo Lo Water Level Sys B,* is NOT illuminated
MSIVs are closed
Reactor pressure peaked at 1141 psig and is now being controlled 800-1000 psig.
Torus water temperature is 105°F and rising
Reactor power is 25%

IAW 0OI-37.5, *ATWS Procedure Basis Document*, which one of the following identifies the action that will have the **highest** priority?

A. SLC initiation.

B. Inhibiting ADS.

C. Trip both Reactor Recirc Pumps

D. Termination and prevention RPV injection.

Answer: D

K/A:
295015 Incomplete SCRAM
G2.4.31 Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)

RO/SRO Rating: 4.2/4.1

Tier 1 / Group 2

K/A match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This question requires knowledge of the annunciator response procedure status (where level is at) to determine the appropriate course of action while executing the EOP. Also for prioritization of the action to direct.

Pedigree: New

Objective: 300E-17e Given plant conditions and the Anticipated Transient Without Scram Procedure, determine the following: Priority of execution given to each leg of the procedure.

Reference: None

Cog Level: Hi

Explanation: With Reactor vessel lo lo water level not illuminated, RPV water level is >90". Reactor power is also >23% with the RR pumps tripped. IAW RC/Q-8 and RC/L-2, terminate and prevent is immediately required in order to prevent THI.

Distractor Analysis:

Choice A: Plausible because SLC initiation is required with >2% power and rising torus temperatures in order to not exceed the HCTL. However, it is not the first step required because power is >23%.

Choice B: Plausible because Inhibiting ADS is required, However, it is not the first step required because power is >23%.

Choice C: Plausible because tripping the recirc pumps would be done prior to terminate and prevent but with RPV pressure peaking at > 1138 ARI would have tripped the pumps, Level is not at LL2 yet according to the alarms.

Choice D: Correct Answer, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. This SRO applicant is required to direct the appropriate actions out of the ATWS EOP based on plant conditions. The action is contained in a note in the EOP and not a general strategy of the EOP.

DEVICE/SETPOINTS

N2555 Auxiliary Relay A71-K1A (actuated from B21-LTM-N024A-1-1) OR N2555 Auxiliary Relay A71-K1C (actuated from B21-LTM-N025A-1-1)	De-energized De-energized
Level Transmitter Master Trip Unit B21-LTM-N024A-1-1 and N025A-1-1	105 inches

POSSIBLE PLANT EFFECTS

1. Inoperable equipment may result in a Tech Spec LCO.

REFERENCES

1. LL-93064 - 92
2. 0AOP-39.0, Loss of DC Power

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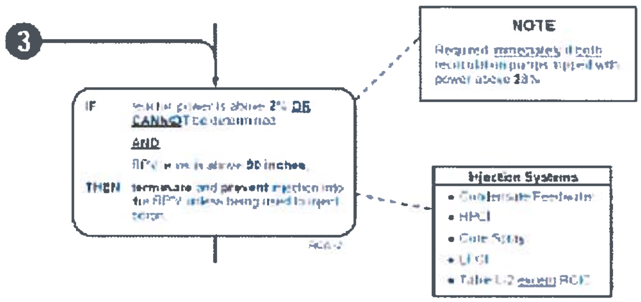
5.4 Step RC/L-2

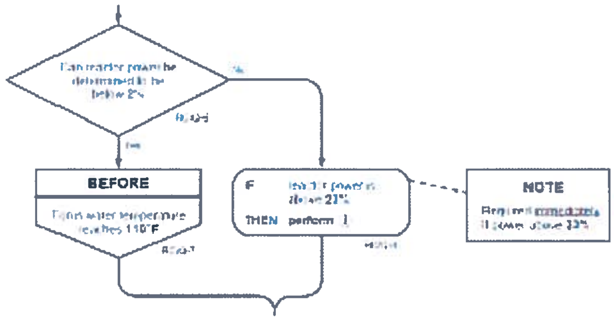


If reactor power is greater than 23% with both reactor recirculation pumps tripped and RPV level above 90 inches, RPV level needs to be promptly reduced below the feedwater nozzles, to avoid thermal hydraulic instabilities. This is accomplished by termination and prevention of injection systems, from identified systems, particularly feedwater, within 120 seconds.

To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, RPV level is initially lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, initiation and growth of oscillations is principally dependent upon subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude.

If reactor power is at or below the APRM downscale trip setpoint (2%), it is highly unlikely that the core bulk boiling boundary would be below that which provides suitable stability margin for operation at high powers and low flows. (A minimum boiling boundary of 4 ft above the bottom of active fuel has been shown to be effective as a stability control because a relatively long two-phase column is required to develop a coupled neutronic/ thermal-hydraulic instability.) Furthermore, flow/density variations would be limited with reactor power this low since the core has a relatively low average void content.

5.33 **Step RC/Q-6 through RC/Q-8**



If reactor power is below 2%, the operator is directed to inject boron before torus water temperature reaches 110°F. This allows sufficient time for Hot Shutdown Boron Weight (HSBW) of boron to be injected.

As long as the core remains submerged (the preferred method of core cooling), fuel integrity and RPV integrity are not directly challenged even under failure-to-scram conditions. A scram failure coupled with an MSIV isolation however, results in rapid heatup of the torus due to the steam discharged from the RPV via SRVs. The challenge to containment thus becomes the limiting factor which defines the requirement for boron injection.

If torus temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit (HCTL), rapid depressurization of the RPV will be required. To avoid depressurizing the RPV with the reactor at power, it is desirable to shut down the reactor prior to reaching HCTL, thus minimizing the quantity of heat rejected to the torus. The Boron Injection Initiation Temperature (BIIT) is defined so as to achieve this when practicable.

5.10 Step RC/L-8 through RC/L-10

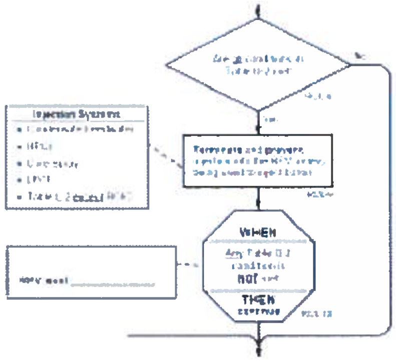


Table Q-2
Terminate and Prevent Determination

	Yes	No
Reactor power above 2% OR CANNOT be determined		
[illegible] above [illegible]°F		
RPV level above TAF		
Notes: • Any SRV OPEN OR being used to control pressure • Drywell pressure above [illegible] psig		

Based on reactor power being above 2%, Step RC/L-2 initially lowered RPV level, to the feedwater sparger, by terminating and preventing injection from identified systems. If all of the conditions in Table Q-2 are met, Step RC/L-9 will lower RPV level further to suppress reactor power. When any condition in Table Q-2 is no longer met the operator is directed to continue to subsequent steps which will establish a new RPV level band.

5.10 Step RC/L-8 through RC/L-10 (continued)

Terminating and preventing injection from:

- Condensate and Feedwater is addressed in 1OP-32 (2OP-32), Condensate And Feedwater System Operating Procedure, and covers terminating and preventing injection by either tripping both Reactor Feed Pumps (RFPs) or by idling one RFP.
- Core Spray is accomplished by tripping the associated Core Spray loop's operating pump.
- HPCI is addressed in 1OP-19 (2OP-19), High Pressure Coolant Injection System Operating Procedure, and covers terminating and preventing injection when HPCI is either operating or not operating.
- RHR is accomplished by tripping the associated RHR loop's operating pump(s).

The Boron Injection Initiation Temperature (BIIT) is a function of reactor power and is the torus temperature before which boron injection must be initiated if a reactor depressurization, due to exceeding the Heat Capacity Temperature Limit (HCTL), is to be precluded. This temperature is 110°F.

The combination of high reactor power (above the APRM downscale trip), high torus temperature (above BIIT), and an open SRV or high drywell pressure (above the scram setpoint), are symptomatic of heat being rejected to the torus at a rate in excess of that which can be removed by the torus cooling system. Unless mitigated, these conditions ultimately result in loss of NPSH for ECCS pumps taking suction on the torus, containment overpressurization, and (ultimately) loss of Primary Containment integrity, which in turn could lead to a loss of adequate core cooling and uncontrolled release of radioactivity to the environment.

The conditions listed in Table Q-2, combined with the inability to shut down the reactor through control rod insertion, dictate a requirement to promptly further reduce reactor power in order to preserve Primary Containment integrity since, as long as these conditions exist, torus heatup will continue.

Since RPV level is only allowed to drop to TAF before injection is restarted, if RPV level is already below TAF, then the objective of the step has been accomplished. Further lowering of RPV level is not necessary, and the steps which deliberately lower RPV level are bypassed.

87. S295021 1

Unit One is in MODE 4, when a loss of SDC occurs due to RCS leakage.

CU1 UNPLANNED loss of RPV inventory for 15 minutes or longer

			4	5	

CU1.1

UNPLANNED loss of reactor coolant results in RPV water level less than a required lower limit for ≥ 15 min. (Note 1)

Which one of the following completes both of the statements below?

The **minimum** required RPV water level to support natural circulation is ___(1)___.

IAW 0PEP-02.2.1, *Emergency Action Level Technical Bases*, the Unusual Event required lower limit is defined as RPV water level less than___(2)___.

A. (1) 200 inches
   (2) 105 inches

B. (1) 200 inches
   (2) 166 inches

C. (1) 254 inches
   (2) 105 inches

D. (1) 254 inches
   (2) 166 inches

Answer: B

K/A:

295021 Loss of Shutdown Cooling

AA2 Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : (CFR: 41.10 / 43.5 / 45.13)

03 Reactor water level

RO/SRO Rating: 3.5/3.5

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the ability of the applicant to determine the water level designated for a UE with a loss of SDC.

Pedigree: Modified from 10-1

Objective: LOI-CLS-LP-301-B Obj. 6 Define the relationship between fission product barrier loss/potential loss and each of the four emergency classifications.

Reference: None

Cog Level: Fundamental

Explanation: Part 1: IAW 0AOP-15.0, the minimum required level for natural circulation is 200 inches. Part 2: IAW 0PEP-02.2.1, the RPV water level lower limit is the low end of any established band.

Distractor Analysis:

Choice A: Part1 is plausible because this is correct, see explanation. Part 2 is plausible because this would be the lower limit if no band was established for SDC in MODE 4.

Choice B: Correct Answer, see explanation

Choice C: Part 1 is plausible because 254 inches is the level of the MSLs and could be confused with Natural Circulation level due to the requirement to be at this level during alternate SDC.

Choice D: Correct Answer, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. The applicant is required to have bases knowledge of the EAL procedure to select the appropriate method of implementing the UE criteria for a lowering RPV level in MODE 4. EAL determination is an SRO only task.

Question from NRC 10.1 exam:

While in Mode 4 a loss of Shutdown Cooling (SDC) occurs.

Which one of the following completes both statements?

The minimum required Reactor Water Level to support Natural Circulation is __(1)__ inches.

An Alert declaration is first required after an unplanned RPV pressure increase greater than __(2)__ psig due to a loss of RCS cooling.

A. (1) 200
(2) 135

B✓ (1) 200
(2) 10

C. (1) 254
(2) 135

D. (1) 254
(2) 10

4.2 Supplementary Actions (continued)

2. IF forced circulation has been lost,
   AND natural circulation has NOT been established,
   THEN ensure reactor vessel water level is being maintained between 200 inches and 220 inches as read on B21-LI-R605A(B) (RPV Water Level),
   OR as directed by the Unit CRS based on plant conditions until forced circulation is restored .... ☐

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RPV level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL recognizes that the minimum required RPV level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP-SAMG NUMERICAL LIMITS AND VALUES, Table 1E
2. SD-01.2 Reactor Vessel Instrumentation Figure 01.2-1 Reactor Water Level Instrument Ranges
3. 1(2) APP A7 2-2 (Reactor Water Level Hi/Low)
4. 0GP-06 Cold Shutdown to Refueling (Head Unbolted) step 5.1.14
5. NEI 99-01 CU1

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Which one of the following completes both statements below?

IAW Tech Spec 3.9.6, *Reactor Pressure Vessel (RPV) Water Level*, the minimum water level over the top of irradiated fuel assemblies seated within the RPV during movement of irradiated fuel assemblies in the RPV is ____(1)____.

The Tech Spec bases for the minimum water level is to provide for ____(2)____ during a fuel handling accident.

A. (1) 19 feet 11 inches
   (2) iodine retention

B. (1) 19 feet 11 inches
   (2) shielding of radioactive decay particles

C. (1) 23 feet
   (2) iodine retention

D. (1) 23 feet
   (2) shielding of radioactive decay particles

Answer: C

K/A:
295023 Refueling Accidents
G2.2.25Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)

RO/SRO Rating: 3.2/4.2

Tier 1 / Group 1

K/A Match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This meets the K/A because it is testing the TS bases for the RPV water level during movement of irradiated fuel assemblies in the RPV.

Pedigree: New

Objective: LOI-CLS-LP-200-B Obj 12. Identify conditions and limitations in the facility license. (SRO/STA only)

Reference: None

Cog Level: Fundamental

Explanation: Part 1: LCO 3.9.6 states, "RPV water level shall be = 23 ft above the top of irradiated fuel assemblies seated within the RPV." " Part 2: The minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the damaged fuel assembly rods is retained by the water.

Distractor Analysis:

Choice A: Part 1 is plausible because IAW TS 3.7.7 the spent fuel storage pool water level shall be = 19 feet 11 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. Part 2 is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible because IAW TS 3.7.7 the spent fuel storage pool water level shall be = 19 feet 11 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. Part 2 is plausible because the water does provide shielding from radioactive decay particles, but that is not the TS bases for the minimum water level.

Choice C: Correct Answer, see explanation

Choice D: Part 2 is plausible because it is correct, see explanation. Part 2 is plausible because the water does provide shielding from radioactive decay particles, but that is not the TS bases for the minimum water level.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. Requires the SRO applicant to know the TS bases for RPV water level.

RPV Water Level
3.9.6

3.9 REFUELING OPERATIONS

3.9.6 Reactor Pressure Vessel (RPV) Water Level

LCO 3.9.6 RPV water level shall be ≥ 23 ft above the top of irradiated fuel assemblies seated within the RPV.

APPLICABILITY: During movement of irradiated fuel assemblies within the RPV,
During movement of new fuel assemblies or handling of control rods within the RPV, when irradiated fuel assemblies are seated within the RPV.

Spent Fuel Storage Pool Water Level
3.7.7

3.7 PLANT SYSTEMS

3.7.7 Spent Fuel Storage Pool Water Level

LCO 3.7.7 The spent fuel storage pool water level shall be ≥ 19 feet 11 inches over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

BACKGROUND	The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level in the reactor vessel. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to well below the 10 CFR 50.67 exposure guidelines (Ref. 3).
APPLICABLE SAFETY ANALYSES	During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for iodine (Ref. 1). This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the damaged fuel assembly rods is retained by the water. Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained well below the allowable limits of Reference 3. RPV water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

(continued)

89. S295026 1

An event on Unit One has resulted in the following plant conditions:

Reactor pressure:	1000 psig
Reactor Water Level	120 inches
Control Rod position	Unknown
APRMs	Downscale
Drywell pressure:	3 psig
Torus pressure:	2 psig
Torus water temp:	152°F
Torus water level:	-36 inches

(REFERENCE PROVIDED)

Which one of the following identifies the required actions for reactor pressure control?

A. Exit the RC/P flowpath of ATWS, and go to 0EOP-01-EDP, *Emergency Depressurization.*

B. Exit the RC/P flowpath of RVCP, and go to 0EOP-01-EDP, *Emergency Depressurization.*

C. Remain in the RC/P flowpath of ATWS, and exceed 100°F/hr cooldown rate if necessary.

D. Remain in the RC/P flowpath of RVCP, and exceed 100°F/hr cooldown rate if necessary.

Answer: C

K/A:

295026 Suppression Pool High Water Temperature

G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)

RO/SRO Rating: 4.3/4.4

Tier 1 / Group 1

K/A match: The applicant is required to determine which procedure to implement based on HCTL

Pedigree: New

Objective: CLS-LP-300L, Obj. 5a Given the Primary Containment Control Procedure, determine the appropriate operator actions if any of the following limits are approached or exceeded: Heat Capacity Temperature Limit

Reference: 0EOP-01-UG, Attachment 7, *Heat Capacity Temperature Limit*

Cog Level: Hi

Explanation: With rods at an unknown position, an ATWS has occurred. Since HCTL is close to the unsafe region (but not violating it), exceeding the cooldown rate in the RC/P flowpath of ATWS is warranted.

Distractor Analysis:

Choice A: Plausible because a novice applicant may misinterpret the graph and believe HCTL is in the unsafe region, and an ED is warranted.

Choice B: Plausible because a novice applicant may believe with APRMs downscale an ATWS has not occurred, and may misinterpret the graph believing an ED is warranted.

Choice C: Correct answer, see explanation.

Choice D: Plausible because a novice applicant may believe with APRMs downscale an ATWS has not occurred. Exceeding the cooldown rate is correct because of the operating point on the HCTL graph.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations (43(b)(5) SRO is required to select the appropriate EOP action based on HCTL.

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ATTACHMENT 7
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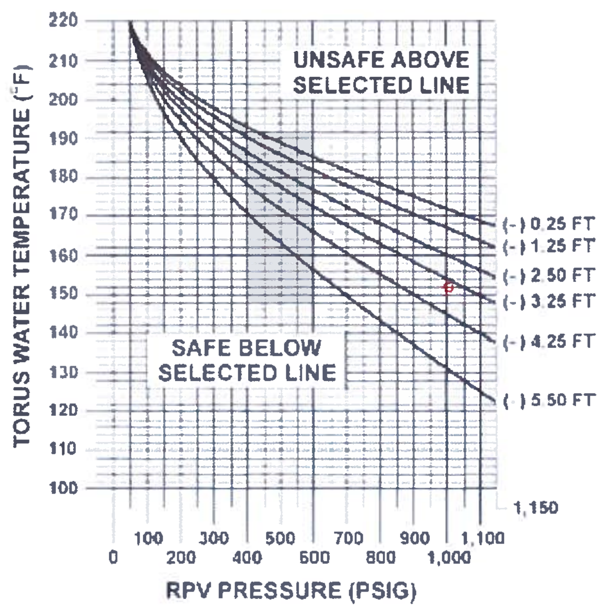
Heat Capacity Temperature Limit



START

ATWS-1

While in this procedure:	
IF	THEN
RPV level CANNOT be determined	Exit RC/L AND RC/P flowpaths and go to EOP-01-RXFP
Emergency depressurization is OR has been required	• Proceed to (5) • Exit RC/P flowpath and go to EOP-01-EDP
Reactor is S/D Without Boron under all conditions (Table Q-1)	1. Terminate boron injection NOT required by other EOPs 2. Exit this flowchart and go to EOP-01-RVCP

ATWS-2

ATWS-2

RC/P

RC/P-1

IF	THEN
RPV pressure approaches **440** psig	Control injection from: 1. Condensate 2. Core Spray 3. LPCI
HCTL **CANNOT** be maintained in safe region, but only if RPV depressurization will **NOT** result in loss of injection required for adequate core cooling	**Maintain RPV pressure below HCTL** • **Exceed 100°F/hr cooldown rate if necessary**
MSIVs are CLOSED **AND** Boron injection is required **AND** Main condenser is available as a heat sink **AND** **NO** indication of a main steam line break exists	**Equalize pressure and open MSIVs** (OP-25): • **Defeat** low RPV level Group 1 isolation per EOP-01-SEP-10
A continuous pneumatic supply is **NOT** available to SRVs	• **IF** stabilizing pressure, **THEN** place control switch to AUTO/CLOSE • **IF** depressurizing RPV, **THEN** minimize cycles

RC/P-2

90. S295035 1

Unit Two is operating at rated power. PCCP has been entered due to high torus water temperature with the following plant conditions:

UA-12 (3-3) *Rx Bldg Diff Press High/Low,* is in alarm
UA-05 (6-10) *Rx Bldg Isolated,* is in alarm.
Reactor Building Pressure (indication on the left)

Which one of the following completes both statements below?

Reactor Building pressure is ___(1)___.

The CRS will direct Reactor Building HVAC restarted IAW ___(2)___.

A. (1) positive
(2) 2OP-37.1, *Reactor Building Heating and Ventilation System Operating Procedure*

B. (1) positive
(2) 0EOP-01-SEP-04, *Reactor Building HVAC Restart Procedure*

C. (1) negative
(2) 2OP-37.1, *Reactor Building Heating and Ventilation System Operating Procedure*

D. (1) negative
(2) 0EOP-01-SEP-04, *Reactor Building HVAC Restart Procedure*

Answer: C

K/A:
295035 Secondary Containment High Differential Pressure
EA2 Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.8 to 41.10)
01 Secondary containment pressure

RO/SRO Rating: 3.8/3.9

Tier 1 / Group 2

K/A match: The applicant is required to interpret secondary containment pressure and select the appropriate procedure based on this interpretation.

Pedigree: New

Objective: LOI-CLS-LP-300-M Obj 11 Given plant conditions involving Reactor Building HVAC system isolation and the Secondary Containment Control Procedure, determine if the Reactor Building HVAC system should be restarted.

Reference: None

Cog Level: Hi

Explanation: Part 1: With indications at upscale > +0.5 inches of h20, reactor building pressure is negative.
Part 2: IAW with UA-5 (6-10) and UA-12 (3-3) RBHVAC is restarted 2OP37.1.

Distractor Analysis:

Choice A: Part 1 is plausible because the reading is +.05 inches of h20 and upscale high, a novice candidate could mistake this for a positive pressure indication. Part 2: is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible because the reading is +0.5 inches of h20 and upscale high, a novice candidate could mistake this for a positive pressure indication. Part 2: is plausible because RBHVAC is restarted using SEP-04, when in SCCP when LL2 and high drywell pressure needs to be defeated. In addition the title is rbhvac restart procedure, and it is a SEP.

Choice C: Correct Answer, see explanation.

Choice D: Part 1 is plausible because it is correct, see explanation. Part 2: is plausible because RBHVAC is restarted using SEP-04, when in SCCP when LL2 and high drywell pressure needs to be defeated. In addition the title is rbhvac restart procedure, and it is a SEP.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] The SRO applicant is required to interpret secondary containment pressure and select the appropriate procedure based on this interpretation.

RX BLDG DIFF PRESS HIGH/LOW
(Reactor Building Differential Pressure High/Low)

AUTO ACTIONS

1. Reactor Building supply and exhaust fans trip.

CAUSE

1. High or low differential pressure between the Reactor Building and atmospheric pressure.
2. Circuit malfunction.

OBSERVATIONS

1. Reactor Building Static Pressure Indicator, 2-VA-PI-1297, on RTGB Panel XU-3.

ACTIONS

1. If secondary containment integrity is required and differential pressure is low, enter 0EOP-03-SCCP, Secondary Containment Control, and execute concurrently with this procedure.
2. Inform E&RC Chemistry Reactor Building Ventilation is not in service.
3. Verify that the valve lineup is correct per 2OP-37.1, Reactor Building Heating and Ventilation System.
4. Start up the system per Section 5.1 of 2OP-37.1.
5. If a circuit malfunction is suspected, ensure that a WR/WO is submitted.

ACTIONS

6. If area radiation levels on Table 5 of 0EOP-03-SCCP exceed maximum normal operating values, enter 0EOP-03-SCCP, Secondary Containment Control.
7. If the Reactor Building HVAC has isolated and it is desired to restart ventilation, enter 2OP-37.1 Reactor Building Heating and Ventilation System.
8. Notify E&RC Counting Room that reactor building ventilation has been secured.

While in this procedure:

IF	THEN
Either: • RB ventilation exhaust radiation exceeds 4 mR/hr • RB ventilation temperature exceeds 135°F (UA-03, 6-2)	• Ensure RB HVAC isolated • Ensure SBGT operating
RB HVAC isolates AND all conditions exist: • RB ventilation exhaust radiation below 4 mR/hr • RB ventilation exhaust radiation monitor has remained on scale • RB ventilation temperature has NOT exceeded 135°F (UA-03, 6-2)	Restart RB HVAC per: • OP-37.1 • EOP-01-SEP-04 if necessary to defeat LL-2 or high drywell pressure isolations

SCCP-2

91. S295037 1

Unit Two is in an ATWS executing RXFP, with the following plant conditions:

Injection to the RPV has been terminated and prevented

The **Minimum Number of SRVs Required for Emergency Depressurization** are open.

Table P-3
Minimum Steam Cooling Pressure

Open SRVs	Pressure (psig)
7 or more	120
6	145
5	175
4	220
3	300
2	455
1	915

IAW RXFP, which one of the following completes the statement below?

The CRS should direct injection to the RPV when EITHER:

___(1)___ SRV remains open

OR

when reactor pressure lowers below the Minimum Steam Cooling Pressure of ___(2)___.

A. (1) NO
(2) 175 psig

B. (1) NO
(2) 455 psig

C. (1) ONLY one
(2) 175 psig

D. (1) ONLY one
(2) 455 psig

Answer: A

K/A:

295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

G2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating: 4.0/4.6

Tier 1 / Group 1

K/A Match: This meets the K/A because it is testing the required steam cooling pressure for adequate core cooling.

Pedigree: New

Objective: LOI-CLS-LP-300-F Obj.3 Given the Reactor Flooding Procedure, which steps have been completed and plant parameter values, determine the required operator actions.

Reference: None

Cog Level: High

Explanation: Part 1: IAW RXFP-9/10, injection is reestablished when either no SRVs are open or Part 2: Reactor is below the MSCP, which in this case for 5 SRVs (MNSRED) is 175 psig.

Distractor Analysis:

Choice A: Correct answer, see explanation.

Choice B: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because this is the minimum steam cooling pressure for having the Minimum Number of SRVs Required for Decay Heat Removal open.

Choice C: Part 1 is plausible because once injection is re-established, RXFP-10 directs injection to continue until at least one SRV is open. In addition, with no SRVs open or below the MSCP adequate core cooling could possibly not exist, so a candidate might assume we would never wait for that condition to reestablish injection. Part 2 is plausible, because it is correct, see explanation.

Choice D: Part 1 is plausible because once injection is re-established, RXFP-10 directs injection to continue until at least one SRV is open. In addition, with no SRVs open or below the MSCP adequate core cooling could possibly not exist, so a candidate might assume we would never wait for that condition to reestablish injection. Part 2 is plausible because this is the minimum steam cooling pressure for having the Minimum Number of SRVs Required for Decay Heat Removal open.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. This requires the SRO to have knowledge of more than the overall sequence of events in the EOPs or mitigating strategy. Since it requires the SRO to know the MNSRED and how to implement Table-P3.

3.0 DEFINITIONS (continued)

37. **Maximum Subcritical Banked Withdrawal Position:** The lowest control rod position to which all controls rods may be withdrawn in bank and the reactor will nonetheless remain shutdown under all conditions. This position is utilized to assure the reactor will remain shutdown irrespective of reactor water temperature.

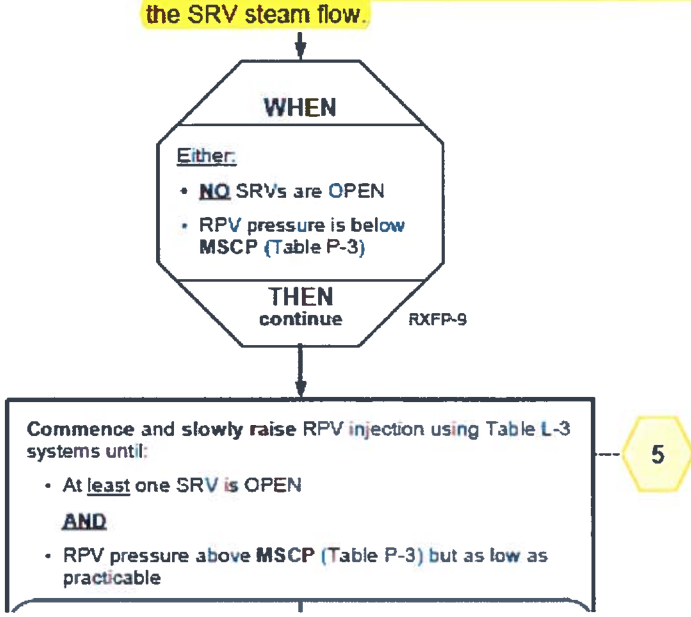
38. **Minimum Core Steam Flow:** The lowest core steam flow which is sufficient to preclude any clad temperature from exceeding 1500°F even if the reactor core is not completely covered.

39. **Minimum Debris Retention Injection Rate:** The lowest RPV injection rate at which it is expected that core debris will be retained in the RPV when RPV level cannot be determined to be above the bottom of active fuel. (Attachment 17)

40. **Minimum Indicated Level:** The highest RPV level instrument indication which results from off-calibration instrument run temperature conditions when RPV level is actually at the elevation of the instrument variable leg tap.

41. **Minimum Number of SRVS Required for Decay Heat Removal:** The least number of SRVs (2) which will remove all decay heat from the core at a pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow.

42. **Minimum Number of SRVS Required for Emergency Depressurization:** The number of SRVs (5) which correspond to a minimum steam cooling pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow.



92. S295038 1
Unit Two is executing RRCP with the following plant conditions:

Main Stack Rad Monitor, D12-RM-23S, is reading 2.3E+08 µCI/sec

Turbine Building Vent Rad Monitor, D12-RM-23, is reading 2.5E+07 µCI/sec

Real-time dose assessment using actual meteorology indicates 0.92 Rem TEDE and 5.1 Rem thyroid CDE at the site boundary

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

IAW RRCP, Unit Two turbine ventilation is required to be in the ___(1)___ ventilation lineup.

The **highest** EAL classification for this event is ___(2)___.

A. (1) recirculation
(2) Site Area Emergency

B. (1) recirculation
(2) General Emergency

C. (1) once through
(2) Site Area Emergency

D. (1) once through
(2) General Emergency

Answer: B
K/A:
295038 High Off-Site Release Rate
EA2 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : (CFR: 41.10 / 43.5 / 45.13)
01 Off-site

RO/SRO Rating: 3.3/4.3

Tier 1 / Group 1

K/A match: The candidate is required to compare the given radiation release values (including site boundary), and compare those to the EALs for rad effluent. Based on this comparison the candidate must make the correct EAL designation.

Pedigree: NEW

Objective: CLS-LP-301-B Obj 9: Given a hypothetical abnormal event and plant operating mode, use 0PEP-02.1 to properly classify or re-classify the event

Reference: 0PEP-02.1

Cog Level: Hi

Explanation: Part 1: IAW the first override in RRCP (RRCP-2) TB ventilation is required to be placed in the recirculation mode of operation. Part 2: Due to Site Area Boundary dose >5000 mrem thyroid CDE, a GE is the highest classification.

Distractor Analysis:

Choice A: Part1 is plausible because it is correct, see explanation. Part 2 is plausible because the main stack and turbine building rad monitors are reading > the SAE setpoint. However, the dose assessments results are above a GE classification.

Choice B: Correct Answer, see explanation.

Choice C: Part1 is plausible because unit two has a once through mode of operation, and a novice applicant might assume that would be used to minimize turbine building dose rates. Part 2 is plausible because the main stack and turbine building rad monitors are reading > the SAE setpoint. However, the dose assessments results are above a GE classification.

Choice D: Part1 is plausible because unit two has a once through mode of operation, and a novice applicant might assume that would be used to minimize turbine building dose rates. Part 2 is plausible because it is correct, see explanation

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] The SRO candidate is required to compare the given radiation release values (including site boundary), and compare those to the EALs for rad effluent. Based on this comparison the candidate must make the correct EAL designation.

5.2 Step RRCP-2

While in this procedure:	
IF	**THEN**
Either: • TB ventilation is SHUTDOWN • TB ventilation is in Once Through Mode	**Perform** per OP-37.3 • **Place** TB HVAC in Recirculation Mode • **Ensure** both TB Air Filter Exhaust Fans operating
Fuel failure indicated by: • Main Steam Line Rad Hi (UA-23, 2-6) • Process Off-Gas Rad Hi (UA-03, 5-2) • Process OG Vent Pipe Rad Hi (UA-03, 6-4)	**Ensure** Control Building Emergency Recirculation operating (OP-37)
Reactor building breached	**Request** ERO evaluate use of mitigating sprays per EDMG-002

RRCP-2

Step RRCP-2 is a procedure override which applies the entire time RRCP is being executed. Each of the three components specify applicable conditions and direct performance of actions as discussed below.

5.2.1 Step RRCP-2 First Override

Continued personnel access to the turbine building may be essential for responding to emergencies or transients which may degrade into emergencies. The turbine building is not an air tight structure, and radioactivity release inside the turbine building would not only limit personnel access but would eventually lead to an unmonitored ground level release, or release via the turbine building ventilation if operating in the once-through lineup.

Operation of the turbine building ventilation in the recirculation lineup helps to improve turbine building accessibility. In addition, since both units share a common turbine building airspace, if the building is intact, removing turbine building ventilation from once through lineup will terminate a large unfiltered volume discharge flow path for a leak on either unit. Due to normal operational requirements when in once through lineup, at least one Air Filter Exhaust Fan and WRGM will be in service providing a monitored and filtered discharge flowpath.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 µCi/sec	2.13E+08 µCi/sec	2.13E+07 µCi/sec	1.80E+05 µCi/sec
Gaseous	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	——	——	——	6.14E+04 cpm
Gaseous	Turbine Building Vent Rad	D12-RM-23	1.07E+08 µCi/sec	1.07E+07 µCi/sec	1.07E+06 µCi/sec	1.13E+04 µCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	——	——	——	2 X hi alarm
Liquid	Radwaste Effluent Rad	D12-RM-K604	——	——	——	2 X hi-hi alarm

GENERAL EMERGENCY	SITE AREA EMERGENCY
RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE	RS1 Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE
1 | 2 | 3 | 4 | 5 | DEF	1 | 2 | 3 | 4 | 5 | DEF
RG1.1 In the absence of real-time dose assessment, reading on any Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4) **RG1.2** Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4) **RG1.3** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation. (Notes 1, 2)	**RS1.1** In the absence of real-time dose assessment, reading on any Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4) **RS1.2** Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4) **RS1.3** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation. (Notes 1, 2)

93. S600000 1

Unit One and Unit Two are executing 0ASSD-01, *Alternative Safe Shutdown Procedure Index*, due to a fire in Main Control Room back panels requiring Main Control Room evacuation. Current plant conditions are:

Unit One and Two have scrammed
All MSIVs are shut

Which one of the following completes both statements below?

The CRS will enter 0ASSD-02, *Control Building,* and ___(1)___ 0ASSD-01.

The CRS will direct actions to achieve a safe shutdown using ___(2)___.

A. (1) exit
(2) HPCI

B. (1) exit
(2) RCIC

C. (1) concurrently perform
(2) HPCI

D. (1) concurrently perform
(2) RCIC

Answer: B

K/A:

600000 Plant Fire On Site

AA2 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: (CFR: 41.10 / 43.5 / 45.13)

07 Whether malfunction is due to common-mode electrical failures

RO/SRO Rating: 2.6/3.0

Tier 1 / Group 1

K/A match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
The applicant is required to determine that based on a fire in the MCR requiring evacuation, common mode failures of electrical equipment could occur and interpret the procedure steps that require exiting ASSD-01 and entering the standalone ASSD-02. In addition, the applicant will determine which train of ASSD equipment (HPCI/RCIC) will remain unaffected by the potential common mode electrical failure.

Pedigree: Modified 14 NRC

Objective: LOI-CLS-LP-301 Obj 20 Given a fire in an ASSD area, describe the potential impact that the fire may have on Safe Shutdown Equipment

Reference: None

Cog Level: Fundamental

Explanation: Part 1: 0ASSD-02, Control Building, is an outside Control Room shutdown procedure for both units. This procedure is a stand-alone post fire shutdown procedure for a Control Room evacuation and requires the reactors to be in hot shutdown/manually scrammed prior to leaving the Control Room. There are parts of the control building that this procedure is not used for, i.e. battery rooms. Part 2: The safe shutdown strategy for control room evacuation requires the B train of ASSD equipment (RCIC).

Distractor Analysis:

Choice A: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because the A train of Safe shutdown equipment (HPCI) is used in other ASSDs.

Choice B: Correct Answer, see explanation

Choice C: Part 1 is plausible because for every other ASSD fire 0ASSD-01 is performed concurrently with specific ASSD sub procedures. Part 2 is plausible because the A train (HPCI) of Safe shutdown equipment is used in other ASSDs.

Choice D: Part 1 is plausible because for every other ASSD fire 0ASSD-01 is performed concurrently with specific ASSD sub procedures. Part 2 is plausible because it is correct, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. requires the SRO applicant to decide the appropriate transition to an event specific subprocedure based on fire in the control building.

2014 Question:

A fire in the control building fire area requires entry into 0ASSD-01, Alternative Safe Shutdown Procedure Index. The CRS has determined that alternate safe shutdown actions are required. Both Unit One and Unit Two have been manually scrammed.

Which one of the following completes the statements below IAW 0ASSD-01?

The next action that is required is to __(1)__.

Following this action both units will __(2)__.

A. (1) place MSIV control switches in close
(2) perform 0ASSD-01, Alternative Safe Shutdown Procedure Index concurrently with 0ASSD-02, Control Building.

B. (1) trip both Reactor Recirc pumps
(2) perform 0ASSD-01, Alternative Safe Shutdown Procedure Index concurrently with 0ASSD-02, Control Building.

C✓ (1) place MSIV control switches in close
(2) exit 0ASSD-01, Alternative Safe Shutdown Procedure Index and enter 0ASSD-02, Control Building

D. (1) trip both Reactor Recirc pumps
(2) exit 0ASSD-01, Alternative Safe Shutdown Procedure Index and enter 0ASSD-02, Control Building

3.5.2 **IF** the fire is in the Control Building fire area, **AND** control room evacuation is required, **THEN PERFORM** the following:

a. **MANUALLY SCRAM** Unit 1 reactor. ☐

b. **PLACE** Unit 1 MSIV control switches in *CLOSE*. ☐

c. **MANUALLY SCRAM** Unit 2 reactor. ☐

d. **PLACE** Unit 2 MSIV control switches in *CLOSE*. ☐

e. Both units **EXIT** this procedure **AND ENTER** 0ASSD-02, Control Building. ☐

3.5.3 **IF** the fire is **NOT** in the Control Building, **THEN ENTER** the applicable ASSD procedure **AND EXECUTE** concurrently with this procedure. ☐

NOTE:	A loss of drywell cooling can be determined using CAC-TR-4426-1A, CAC-TR-4426-2A, in the Control Room or CAC-TR-778, points 1, 3, and 4, at the Remote Shutdown Panel. The time the loss of drywell cooling occurred can be determined from the recorder display information.

3.5.4 **IF** RCIC or HPCI is injecting **AND** drywell cooling has been lost, **THEN START** reactor vessel cooldown at 100°F/hr or greater within 60 minutes of the loss of drywell cooling. ☐

3.5.5 **IF** drywell temperature control is lost, **THEN PERFORM** the following to preserve containment overpressure for RHR pump net positive suction head:

1. **STOP** A, B, C, and D RBCCW pumps for the affected unit. ☐
2. **CLOSE** the following valves for the affected unit:
   - *DW EQUIP DRAIN INBD ISOL VLV, G16-F019* ☐
   - *DW EQUIP DRAIN OTBD ISOL VLV, G16-F020* ☐
   - *DW FLOOR DRAIN INBD ISOL VLV, G16-F003* ☐
   - *DW FLOOR DRAIN OTBD ISOL VLV, G16-F004* ☐

2.1 This procedure is entered from Alternative Safe Shutdown Procedure Index, 0ASSD-01.

AND

2.2 Unit CRS has determined both reactors are to be brought to safe and stable conditions from outside the Control Room using ASSD Train B.

3.0 OPERATOR ACTIONS

3.1 **CONTINUE** implementation of this procedure by performing the steps in Section A. ☐

3.2 **IF** the fire is extinguished while executing this procedure **AND** the Unit CRS determines no action within this procedure is required, **THEN EXIT** this procedure. ☐

4.0 RESTORATION

4.1 **RETURN** plant to general operating condition as directed by plant management. ☐

94. SG2.1.05 1
Which one of the following completes both statements below?
(Consider each statement separately.)

IAW Tech Spec 5.2.2, *Facility Staff*, the shift crew composition may be less than the minimum requirement for a period of time not to exceed ___(1)___ for an unexpected absence of on-duty shift crew members.

IAW 0OI-01.01, *BNP Conduct of Operations Supplement*, the minimum required number of Auxiliary Operators for manning a shift at BNP is ___(2)___.

A. (1) one hour
   (2) three

B. (1) one hour
   (2) nine

C. (1) two hours
   (2) three

D. (1) two hours
   (2) nine

Answer: D
K/A:
G2.1.05 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 2.9/3.9

Tier 3

K/A match: knowledge of required tech spec and conduct of ops shift manning requirements.

Pedigree: 2012 BNP NRC

Objective: LOI-CLS-LP-200-B Obj.12.-Identify conditions and limitations in the facility license.

Reference: None

Cog Level: Low

Explanation: IAW the procedure, 9 AO makeup the minimum shift staffing and two hours is the time to find a replacement. One hour is the time on "stepping out" limitation of the control room personnel. The tech Specs 5.2 only address the number of AOs for the Units which is 3, this does not take into account ASSD and Fire Brigade.

Distractor Analysis:

Choice A: Plausible because TS 5.2.2 requires 3 AOs for both Units which does not take into account ASSD and Fire Brigade requirements. One hour is the "stepping out" time limit for control room personnel.

Choice B: Plausible because nine is correct but one hour is the "stepping out" time limit for control room personnel.

Choice C: Plausible because TS 5.2.2 requires 3 AOs for both Units which does not take into account ASSD and Fire Brigade requirements.

Choice D: Correct Answer, see explanation

SRO Basis: Conditions and limitations in the facility license. (10 CFR 55.43(b)(1)). Requires the SRO applicant to know the limitations for shift staffing in the license.

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5.5 <u>Operations Shift Staffing</u>

5.5.1 General

1. In addition to the requirements of AD-OP-ALL-1000, the following requirements apply:

   a. The following table outlines the administrative guideline for the normal Operations shift complement. Any deviation from the normal shift complement must remain in accordance with Section 5.2.2 of Technical Specifications, applicable sections of 0ASSD-00, User's Guide, 0FPP-031, Fire Brigade Staffing Roster and Equipment Requirements, and 0ERP, Radiological Emergency Response Plan (ERP). (Attachment 13, Operations Staffing Roster contains a listing of required ERO Watch Stations and qualifications for each and ASSD positions.)

BNP Watchstations	BNP Shift Complement	License
Shift Manager (SM)	1 Shift Manager	SRO
Control Room Supervisor (CRS)	2 CRSs (1 for each unit)	SRO
Reactor Operator (RO)	4 Reactor Operators (typically, 2 for each unit)	RO/SRO
Auxiliary Operator (AO)	9 (includes 2 in Radwaste)	NA
Operations Center SRO	1 Operations Center SRO	SRO
STA [Note 1]	1 STA	STA Qualified

Notes:

Organization
5.2

5.2 Organization

5.2.2 Facility Staff (continued)

b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, when either unit is in MODE 1, 2, or 3, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room. With one unit in MODE 1, 2, or 3 and the other unit defueled, the minimum shift crew shall include a total of two SROs and two ROs.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Deleted.

95. SG2.1.43 1

Following the bypass of Unit Two feedwater heaters 4A and 5A, the following plant conditions exist:

Reactor Power is 60%
Feedwater Temperature is 330°F

Final Feedwater Temperature vs Power

RX PWR	Nominal FW Temp	Nominal FW Temp Reduced 10°F	110.3°F Reduced FFWT
65%	394.4	384.4	296.4
64%	393.1	383.1	295.5
63%	391.7	381.7	294.6
62%	390.4	380.4	293.7
61%	389.0	379.0	292.8
60%	387.6	377.6	291.9
[illegible]	[illegible]	[illegible]	[illegible]

IAW 0OI-01.01, *BNP Conduct of Operations Supplement,* which one of the following completes both statements below? (consider each statement separately)

The CRS ___(1)___ required to implement the thermal limit penalties for FHOOS (feedwater heater out of service).

Entry into **Tech Spec 3.0.3** ___(2)___ required if final feedwater temperature is less than the 110.3°F reduced final feedwater temperature value.

A. (1) is
   (2) is

B. (1) is
   (2) is NOT

C. (1) is NOT
   (2) is

D. (1) is NOT
   (2) is NOT

Answer: B

K/A:

G2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc. (CFR: 41.10 / 43.6 / 45.6)

RO/SRO Rating: 4.1/4.3

Tier 3

K/A match: The applicant is required to use the final feedwater temperature reduction attachment to determine if the effect of the feedwater reduction is severe enough on reactivity to require implementation of thermal limit penalties.

Pedigree: New

Objective: CLS-LP-032 obj 27 Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM, and COLR, determine whether given plant conditions meet minimum Technical Specifications requirements associated with the Condensate and Feedwater System.

Reference: NONE

Cog Level: High

Explanation: : Part1: A final fw temp of 385°F is less than the nominal FW temp for 60% power, but >10°F reduced from nominal. therefore the thermal limit penalties for FHOOS do not need to be implemented. Part 2:There are **NO** core operating limits specified in the COLR for operation beyond 110.3°F Final Feedwater Temperature. Thermal limits **CANNOT** be verified to be within the limits specified in the COLR, which requires entry into the Actions of LCO 3.2.1, 3.2.2, and 3.2.3. These LCOs require thermal limits be restored within 4 hours, LCO 3.0.3 is not entered,

Distractor Analysis:

Choice A: Part 1 is correct, see explanation. Part2 is plausible because a candidate may believe since there are no thermal limits specified in the COLR for this condition, LCO 3.0.3 would be applicable.

Choice B: Correct Answer, see explanation

Choice C: Part 1 is plausible because a final fw temp of 330°F is less than the nominal FW temp reduced by 10°F for 60% power, but greater than the 110.3°F reduced FFWT, a novice applicant may believe thermal limit penalties are only applied at the 110.3°F value. Part2 is plausible because a candidate may believe since there are no thermal limits specified in the COLR for this condition, LCO 3.0.3 would be applicable.

Choice D: Part 1 is plausible because a final fw temp of 330°F is less than the nominal FW temp reduced by 10°F for 60% power, but greater than the 110.3°F reduced FFWT, a novice applicant may believe thermal limit penalties are only applied at the 110.3°F value. Part 2 is correct, see explanation.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)] This question requires that the applicant determines whether the TS thermal limits should incur a penalty. In addition, it also requires that the candidate determines whether LCO 3.0.3 applies for a given condition.

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 7 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.

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<< Equipment Out Of Service Contingencies >>

EOOS Condition	Power	Required Action (Note 1, 2)
SLO	Any	• **Reduce** reactor power to ≤ 50% • Implement 0GP-14. • **Implement** applicable SLO power to flow map. • **IF** ≥ 23% RTP, **THEN implement** thermal limit penalty. • Reference TS 3.4.1.
TBVOOS	≥ 23% RTP	• **Implement** thermal limit penalty. • **Reference** TS 3.7.6.
FHOOS (FWTR) (FFTR)	≥ 23% RTP	• **IF** < the value in the "110.3°F Reduced FW Temp" column of 1(2)OP-32, Att 6, **THEN enter** LCO 3.2.1, 3.2.2 and 3.2.3.
	≥ 23% RTP	• **IF** > 10°F below nominal FW temperature, **THEN**: ◊ **Implement** applicable FWTR power to flow map. ◊ **Implement** thermal limit penalty.
		• **Implement** applicable FWTR power to flow map

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ATTACHMENT 6
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Final Feedwater Temperature vs Power

RX PWR	Nominal FW Temp	Nominal FW Temp Reduced 10°F	110.3°F Reduced FFWT
65%	394.4	384.4	296.4
64%	393.1	383.1	295.5
63%	391.7	381.7	294.6
62%	390.4	380.4	293.7
61%	389.0	379.0	292.8
60%	387.6	377.6	291.9
59%	386.2	376.2	290.9

96. SG2.2.15 1

Unit One is operating at rated power.

A-03 (2-2) *Auto Depress Control Pwr Failure*, is in alarm due to Fuse F5 being blown.

(REFERENCE PROVIDED)

Which one of the following completes both statements below?

Fuse F5 is located on ADS Logic ___(1)___.

ADS ___(2)___ operable.

A. (1) A
(2) is

B. (1) A
(2) is NOT

C. (1) B
(2) is

D. (1) B
(2) is NOT

Answer: C

K/A:

G2.2.15 Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc. (CFR: 41.10 / 43.3 / 45.13)

RO/SRO Rating: 3.9/4.3

Tier 3

K/A match: THIS QUESTION WAS PRE-SUBMITTED FOR APPROVAL.
This question requires the candidate to use a drawing to determine operability of ADS.

Pedigree: New

Objective: LOI-CLS-LP-020 OBJ 15d. Given plant conditions, predict how ADS/SRVs will be affected by the following: Loss of DC power

Reference: 1-FP-05887 (Block out references to which logic string is logic A and B)

Cog Level: High

Explanation: Part 1: Fuse F5 is located on the alternate power source, only logic B has an alternate power source. Therefore, the fuse is on logic B, Part2: Since the drawing is shown in the de-energized state, fuse 5 being blown will have no impact on ADS instrumentation. ADS remains on its normal power source.

Distractor Analysis:

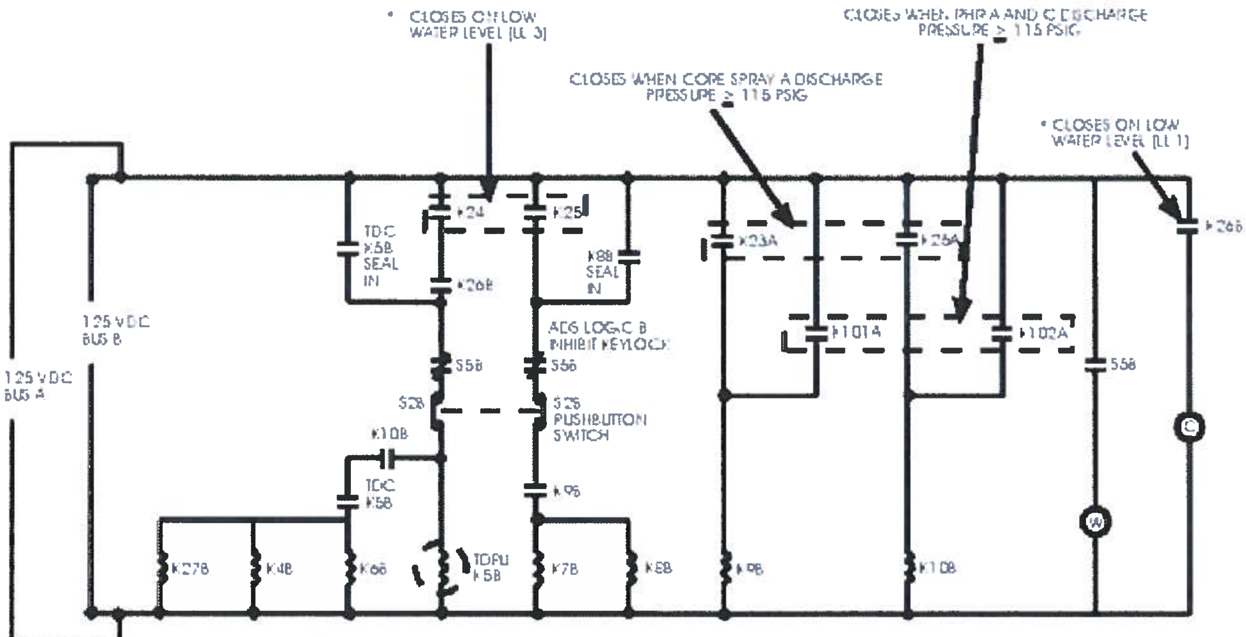
Choice A: Part 1 is plausible the fuse is located on 125V DC 3A power or the operator might forget which train of logic has two power supplies. Part 2 is plausible because it is correct, see explanation.

Choice B: Part 1 is plausible the fuse is located on 125V DC 3A power or the operator might forget which train of logic has two power supplies. Part 2 is plausible because if the drawing was shown in the energized state, this would be correct.

Choice C: Correct Answer, see explanation.

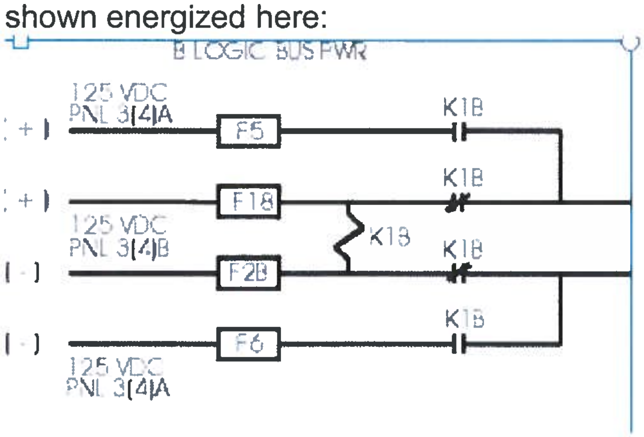
Choice D: Part 1 is plausible because it is correct, see explanation. Part 2 is plausible because if the drawing was shown in the energized state, this would be correct.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)] Requires the SRO candidate to use a drawing to determine the status of ADS power and to know whether that loss effects ADS operability.



* SEE FIGURE 11 FOR LOW WATER LEVEL INSTRUMENTS/RELAYS.

shown energized here:



97. SG2.2.22 1
Unit Two is operating at rated power.

While performing 0PT-07.2.4A, *Core Spray Loop A Operability*, Core Spray Room Cooler A fails to start when Core Spray Pump A is started.

The reactor building AO reports that the room cooler tripped on thermal overload.

IAW AD-OP-ALL-1000, *Conduct of Operations*, which one of the following completes both statements below? (consider each statement separately)

Core Spray Loop A is ____(1)____.

A one time reset of the thermal overload ____(2)____ allowed before a Maintenance and Engineering evaluation.

A. (1) OPERABLE
(2) is

B. (1) OPERABLE
(2) is NOT

C. (1) INOPERABLE
(2) is

D. (1) INOPERABLE
(2) is NOT

Answer: D
K/A:
G2.2.22 Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

RO/SRO Rating: 4.0/4.7

Tier 3

K/A match: Requires knowledge of conduct of ops procedure to determine whether Core Spray A meets the conditions for operability in the tech specs based on cooler operation.

Pedigree: Bank NRC 08

Objective: CLS-LP-18, Obj. 18. Given plant conditions and TS, including bases, TRM, ODCM, and COLR, determine the required actions to be taken in accordance the TS associated with the Core Spray System.

Reference: None

Cog Level: Hi

Explanation: Part 1: Per 0OI-01.01, When any ECCS Room Cooler is determined to be INOPERABLE, then the ECCS equipment associated with that room cooler is to be declared INOPERABLE per the applicable Technical Specifications. Part 2: Per AD-OP-ALL-1000, the breaker should only be reset once the condition is identified and corrected, **and** plant conditions dictate the reset before maint and eng personnel are available.

Distractor Analysis:

Choice A: Part 1 is plausible, because the room cooler is not part of the Core Spray system listed in the tech spec bases. Part 2 is plausible because during transient conditions the breaker could be reset, however, the plant is in a stable condition.

Choice B: Part 1 is plausible, because the room cooler is not part of the Core Spray system listed in the tech spec bases. Part 2 is correct, see explanation.

Choice C: Part 1 is correct, see explanation. Part 2 is plausible because during transient conditions the breaker could be reset, however, the plant is in a stable condition.

Choice D: Correct Answer, see explanation.

SRO Basis: Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)] Requires the SRO candidate to have knowledge of conduct of ops procedure to determine whether Core Spray A meets the conditions for operability in the tech specs based on cooler operation.

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5.19 Resetting Protective Devices

{7.1.4}

5.19.1 Standards

1. Protective devices should not be reset without a clear understanding of the reason for the protective device trip.

2. The overriding priority for the operating crew upon the trip of any protective device is to stabilize the plant and restore the systems to the safest possible condition.

5.19.2 Expectations

1. Protection devices which have actuated (breakers, fuses, bistables, MOV thermal overloads, lockouts, etc.) should only be restored with shift supervision approval, under the following conditions. The following conditions do not apply to 120 volt breakers that only supply lighting or receptacles.

   a. The cause of the actuation has been identified and corrected.

   b. Restoring the protective device is not recommended unless plant conditions dictate that the component repositioning must be completed before Maintenance and Engineering personnel are available. Remote operation of the component with no personnel in the immediate area after resetting the protective device is recommended if repositioning is required prior to completion of the evaluation by Maintenance and Engineering.

2. The SM may approve additional protective device resetting after consultation with Engineering.

5.16.2 Degraded Equipment Controls— System/Component Related Guidance (continued)

(1) Reference TRM Appendix F, Safety Function Determination Program (SFDP), Attachments 1 and 2 to assist with determination of Technical Specification 3.8.1 and 3.8.7 requirements and to assess the possible impact on supported systems.

(2) If an evaluation of the SFDP is performed, then document the evaluation and the results in the the narrative log or on Attachment 26, if the narrative log is not available.

4. ECCS Room Coolers {7.1.3}

NOTE

- The following step is not required to be performed if the ECCS Room Cooler is INOPERABLE due to the loss of a 4160V or 480V E-Bus. E-Bus INOPERABILITY impacts the OPERABILITY of ECCS subsystems. Technical Specifications and the SFDP will provide Required Actions to be taken for the loss of the E-Bus.
- In Mode 4 and Mode 5, ECCS Room Coolers are not required to be OPERABLE to support OPERABILITY of the associated ECCS Systems.

a. When any ECCS Room Cooler is determined to be INOPERABLE, then the ECCS equipment associated with that room cooler is to be declared INOPERABLE per the applicable Technical Specifications.

EXAMPLE

The RHR Room Coolers are to be considered redundant components required to support the operation of RHR. Therefore, should a room cooler be found or made INOPERABLE, a 7 day Active LCO is required to be established on the RHR system. Likewise, should both room coolers be found INOPERABLE, the action required is the same as if both RHR loops and HPCI were INOPERABLE. Should it be identified that one RHR Room Cooler is INOPERABLE and one RHR Loop is also INOPERABLE (specific combinations do not matter), the action is as if only one RHR Loop is INOPERABLE (7 days).

5. Control Building HVAC Air Compressors

98. SG2.3.11 1
Following a small steam line break in the drywell plant conditions are as follows:

Drywell pressure:	25 psig and rising
Drywell hydrogen:	1.3%
Suppression Chamber hydrogen:	1.2%
Torus level:	42 inches

Which one of the following completes both statements below?

The CRS is required to direct venting containment IAW 0EOP-01-SEP-01, *Primary Containment Venting*, using ___(1)___.

Venting of the ___(2)___ will be directed **first**.

A. (1) Section 2.1, *Containment Pressure Control*
   (2) drywell

B. (1) Section 2.1, *Containment Pressure Control*
   (2) torus

C. (1) Section 2.2, *Containment Hydrogen Control*
   (2) drywell

D. (1) Section 2.2, *Containment Hydrogen Control*
   (2) torus

Answer: D
K/A:
G2.3.11 Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

RO/SRO Rating: 3.8/4.3

Tier 3

K/A match: Requires the ability to determine the procedure section for venting, and the correct sequence of termination of venting.

Pedigree: New

Objective: CLS-LP-300-L*08d
Given the Primary Containment Control Procedure and plant conditions, determine if the following actions are required: Venting the primary containment IRRESPECTIVE of radioactivity release rate limits

Reference: None

Cog Level: High

Explanation: Part 1: Following the H2 leg of the PCCP with the given conditions will drive you to step PC/G-9 which directs you to "Vent Containment per EOP-01-SEP-01, since H2 is the driving condition for venting, then section 2.2 is the appropriate section to implement. Part 2: IAW SEP-01, the torus is vented first as long as the torus water level is less than 6 feet.

Distractor Analysis:

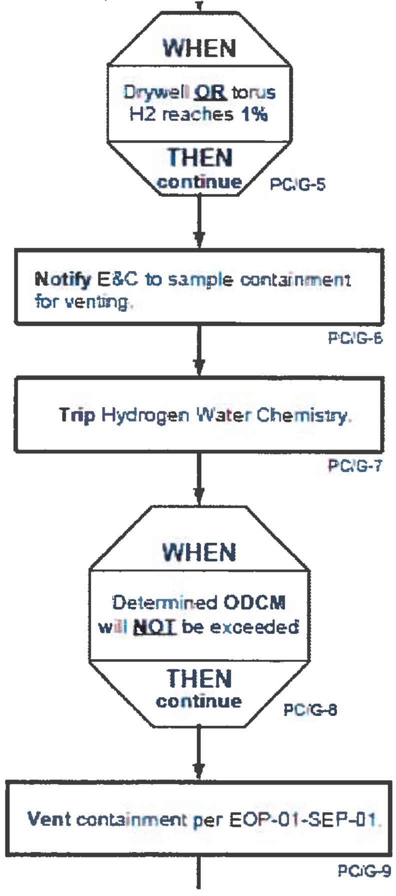
Choice A: Part 1 is plausible because H2 concentration is less than the entry limit into PCCP (entry at 1.5%), and Containment pressure is >11.5 psig (pressure for DW sprays), therefore a novice applicant might believe that the appropriate procedure section required is for containment pressure control. Part 2 is plausible since venting of the drywell is performed first if torus water level is >6 feet.

Choice B: Part 1 is plausible because H2 concentration is less than the entry limit into PCCP, and Containment pressure is >11.5 psig (pressure for DW sprays), therefore a novice applicant might believe that the appropriate procedure section required is for containment pressure control. Part 2 is correct.

Choice C: Part 1 is plausible since it is correct, see explanation. Part 2 is plausible since venting of the drywell is performed first if torus water level is >6 feet.

Choice D: Correct Answer, see explanation..

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] Requires knowledge of diagnostic step in EOP, and selection of appropriate emergency contingency procedure.



2.2.3 Containment Hydrogen Control Actions (continued)

e. Close VA-D-BFV-RB (SBGT A Isol Damper).............................. ☐ RO

f. Close VA-H-BFV-RB (SBGT B Isol Damper).............................. ☐ RO

g. IF venting the torus,
THEN open:

(1) CAC-V7 (Torus Purge Exh Vlv).. ☐ RO

(2) CAC-V8 (Torus Purge Exh Vlv).. ☐ RO

h. IF venting the drywell,
THEN open:

(1) CAC-V9 (Drywell Purge Exh Vlv)...................................... ☐ RO

(2) CAC-V10 (Drywell Purge Exh Vlv)..................................... ☐ RO

i. Open VA-F-BFV-RB (SBGT DW Suct Damper).............................. ☐ RO

13. IF directed to terminate torus venting,
THEN:

a. Ensure primary containment purging terminated per EOP-01-SEP-05.. ☐ RO

99. SG2.4.30 1

Unit Two is operating at rated power with LPCI A inoperable and the following sequence of events occurs:

0000	7 day completion time for LCO 3.5.1, *ECCS Operating*, Condition A expires and Condition C is entered requiring that the Unit be placed in MODE 3 in 12 hours.
0030	Plant shutdown is commenced per LCO 3.5.1, Condition C.
0050	LPCI A is repaired and declared operable; LCO 3.5.1 Conditions A and C are exited.
0100	Management decides to continue the plant shutdown as planned to complete other maintenance items.
0230	Unit Two in MODE 3

(REFERENCE PROVIDED)

Which one of the following identifies the reportability requirements, if any, for this event?

A report to the NRC:

A. is not required.

B. would be submitted no later than 0400.

C. would be submitted no later than 0430.

D. would be submitted no later than 0630.

Answer: C

K/A:

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. (CFR: 41.10 / 43.5 / 45.11)

RO/SRO Rating: 2.7/4.1

Tier 3

K/A match: Applicant required to determine report status for the given condition.

Pedigree: Bank

Objective: LOI-CLS-LP-201-D, Obj 11
Explain the following regarding NRC Reporting requirements per AD-LS-ALL-0006, Notification/Reportability Evaluation: d. Determination of "clock start" time for reportable events (LOCT)

Reference: 0OI-01.07 Attachment 1

Cog Level: High

Explanation: A TS required shutdown requires a 4 hour NRC report. The time starts when the shutdown is started, Completion of the shutdown required by TS is an LER.

Distractor Analysis:

Choice A: Plausible because the plant was not shutdown due to the TS as LPCI was repaired and the TS exited.

Choice B: Plausible because this would be 4 hours from when the TS shutdown condition was entered.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because an LER is required after completing a TS required shutdown, but in this case the shutdown was not TS required.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]. Requires SRO administration procedure knowledge of reportability requirements based on plant conditions.

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Reportability Evaluation Checklist

NOTE

- If the answer to any of the following questions is YES, the event is reportable within 4 hours.
- If all answers to the following questions are NO, the event is not reportable within 4 hours.

4 HOUR REPORTABILITY			
ITEM #	**YES**	**NO**	**DESCRIPTIVE QUESTION**
2.1			**NOTE** Includes any Safety Limit violation (Tech Spec 2.2). Is plant shutdown required by technical specifications being initiated? [10 CFR 50.72(b)(2)(i)]

Plant Shutdown Required by Technical Specifications (See Section 3.2.1 of this report)	
§ 50.72(b)(2)(i) "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications."	**§ 50.73(a)(2)(i)(A)** "The completion of any nuclear plant shutdown required by the plant's Technical Specifications."

Discussion

The 10 CFR 50.72 reporting requirement is intended to capture those events for which TS require the initiation of reactor shutdown to provide the NRC with early warning of safety-significant conditions serious enough to warrant that the plant be shut down. For 10 CFR 50.72 reporting purposes, the phrase "initiation of any nuclear plant shutdown" includes action to start reducing reactor power; i.e., adding negative reactivity to achieve a nuclear plant shutdown required by TS. This includes initiation of any shutdown due to expected inability to restore equipment prior to exceeding the LCO action time. As a practical matter, in order to meet the time limits for reporting under 10 CFR 50.72, the reporting decision should sometimes be based on such expectations. (See Example 4.)

The "initiation of any nuclear plant shutdown" does not include mode changes required by TS if they are initiated after the plant is already in a shutdown condition.

A reduction in power for some other purpose, not constituting initiation of a shutdown required by TS, is not reportable under this criterion.

For 10 CFR 50.73 reporting purposes, the phrase "completion of any nuclear plant shutdown" is defined as the point in time during a TS-required shutdown when the plant enters the first shutdown condition required by an LCO (e.g., hot standby (Mode 3) for PWRs with the Standard Technical Specifications (STS)). For example, if at 0200 hours a plant enters an LCO action statement that states, "restore the inoperable channel to operable status within 12 hours or be in at least Hot Standby within the next 6 hours," the plant must be shut down (i.e., at least in hot standby) by 2000 hours. An LER is required if the inoperable channel is not returned to operable status by 2000 hours and the plant enters hot standby.

An LER is not required if a failure was or could have been corrected before a plant has completed shutdown (as discussed above) and no other criteria in 10 CFR 50.73 apply.

5.4 Making Emergency Notification System and LER Reports (continued)

Table 1. Emergency Notification System Reporting Overview

Event or Condition	ENS notification within 1 hour	ENS notification within 4 hours	ENS notification within 8 hours	60-day LER	Job Aid Section
Plant shutdown (S/D) required by Tech Specs		Initiation of S/D required by Tech Specs [50.72 (b)(2)(i)]		Completion of a S/D required by Tech Specs [50.73 (a)(2)(i)(A)]	A.3, A.4

100. SG2.4.35 1

Unit One and Unit Two have entered SBO procedures at time 1300 due to a loss of all onsite and offsite power.

Which one of the following completes both statements below?

IAW 1EOP-01-SBO, *Station Blackout*, opening the reactor building roof hatch is required to be performed **before** ___(1)___.

IAW 0OI-37.14, *Station Blackout Procedure Basis Document*, the reactor building doors and roof hatch are opened to ensure ___(2)___.

A. (1) 1330
(2) equipment availability

B. (1) 1330
(2) habitability

C. (1) 1500
(2) equipment availability

D. (1) 1500
(2) habitability

Answer: D

K/A:

G2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.8/4.0

Tier 3

K/A match: Requires the applicant to have knowledge of when to implement the AO task in a EOP sub procedure (opening RB roof hatch during SBO), and the operational effects if not completed (jeopardized habitability).

Pedigree: New

Objective: LOI-CLS-LP-303-B Obj 2 Given plant conditions, EOP-01-SBO Flowchart, and SBO Support Procedures, determine the required operator actions. Temperature analysis states that access would be prohibited in the RB building due to 117' elevation ceiling temperature if the hatch was not opened.

Reference: None

Cog Level: Fundamental

Explanation: This is a **time sensitive action** from the SBO procedure that directs the RB roof hatch to be opened within 2 hours of the start of the SBO.

Distractor Analysis:

Choice A: Part 1 is plausible because this is the time critical action time limit in SBO procedure for opening control panel doors. Part 2 is plausible because high temperatures could be thought to jeopardize equipment availability, however the hatch and doors are opened to ensure habitability.

Choice B: Part 1 is plausible because this is the time critical action time limit in SBO procedure for opening control panel doors. Part 2 is correct, see explanation.

Choice C: Part 1 is correct, see explanation. Part 2 is plausible because high temperatures could be thought to jeopardize equipment availability, however the hatch and doors are opened to ensure habitability.

Choice D: Correct, see explanation.

SRO Basis: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] Knowledge of when to implement attachments and the basis for the step.

5.25 Steps SBO-27 and SBO-28

Open reactor building doors per EOP-01-SBO-04.

SBO-27

Time Sensitive

RB roof hatch required open within **2 hours**.

Time ______________

Stage alternate fuel pool makeup/spray equipment per EOP-01-SEP-12

SBO-28

If ELAP conditions exist, reactor building temperatures will rise rapidly due to the loss of building ventilation. The refuel floor roof hatch and 20' elevation personnel access doors are blocked opened to provide alternate ventilation. The refuel floor roof hatch should be opened as soon as resources are available and is required to be open within 2 hours of the SBO start time recorded at Step SBO-1. The 20' elevation personnel access doors should be opened as soon as resources are available and are required to be open within 6 hours of the SBO start time recorded at Step SBO-1 by the text procedure. The reactor building temperature analysis (BNP-MECH-FLEX-0001) shows 117' elevation ceiling temperature will reach 114°F at time 2 hours, which is approaching the temperature that access would be prohibited. Alternate ventilation should be established as early as possible based on priorities and available resources.

Facility: Brunswick Scenario No.: 1 Op-Test No.: 2016 Draft

Examiners: ______________ Operators: ______________

Initial Conditions:

Unit Two is operating at rated power.
1A NSW Pump is under clearance for planned maintenance.
2C TCC pump is aligned to Unit One.
APRM 2 failed downscale and bypassed

Turnover

Start the 2C Condensate Booster Pump and Secure the 2A Condensate Booster Pump.
2OP-32, Section 6.3.6 is completed up to step 6.3.6.11.

Event No.	Malf. No.	Event Type*	Event Description
1		N - BOP	Swap Condensate Booster Pumps
2	ES014F	C - ATC C - CRS	Inadvertent HPCI Initiation w/ failure to trip (TS)(AOP-03.0)
3	ZUA343	C - BOP C - CRS	Off Gas Filter Hi d/P
4	RC055 D	C - ATC C - CRS	Recirc Pump Trip (TS)(AOP-04.0)
5		R - ATC	Reduce reactor power for Single Loop Operation
6	CF039F	C - BOP C - CRS	Heater Drain Deaerator Controller Failure (AOP-23.0)
7	NB005F	M	Fuel Failure (RSP)(AOP-05.4)
8	K2501A K2503A	C	Manual Scram Failure – Alternate Rod Insertion Close Group I Valves
9	RH020F	C	Torus cooling Valve Failure (PCCP)

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

Event	Event Description Summary
1	The Unit will be operating at rated power and after taking the shift turnover will swap condensate booster pumps IAW 2OP-32. (Start 2C, secure 2A)
2	After the condensate booster pumps have been swapped, an inadvertent HPCI initiation will occur. The HPCI system will not trip but can be manually isolated. AOP-03.0 and Tech Specs will be entered.
3	After Tech Specs are addressed the Off Gas Filter Hi d/P alarm will be received. The standby Off Gas filter will be placed in service.
4	VFD coolant leakage will occur that will cause a trip of the 2B Recirc pump. The supply breaker does not trip and must be opened manually. 0AOP-04.0 will be entered. Technical Specifications will be addressed. .
5	The plant will be greater than the allowable for single loop operation. Recirc flow must be lowered and/or control rods must be inserted to reduce power. With only one recirc pump in operation reactor power must be less than 50% and core flow must be greater than 30.8 Mlbs but less than 45 Mlbs.
6	After the power reduction the Heater Drain Deaerator controller will fail. AOP-23.0 will be entered.
7	Fuel failure will begin requiring entry into AOP-05.4. When Process Off gas Hi Hi alarm with the Main Stack rising a scram will be inserted and Group I valves closed.
8	The crew will be required to (1) Scram, (2) Close Group I valves. (Each of which are critical tasks) The manual scram pushbuttons will not work, Alternate Rod Insertion (ARI) will be initiated to scram the rods.
9	Torus cooling valve will trip on thermals (will reset if attempted).

Facility: Brunswick Scenario No.: 2 Op-Test No.: 2016 Draft

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions:

Unit Two is at 95% power.
1A NSW Pump is under clearance for planned maintenance.
APRM 2 has failed downscale and is bypassed.
2C TCC Pump is aligned to Unit One.

Turnover

The OATC will reduce power to ~850 MWe Gross (reactivity plan is to use recirc flow)
The BOP operator will then Isolate 230 kV Delco West (Line 30) IAW the marked up of 2OP-50, Section 6.2.6.

Event No.	Malf. No.	Event Type*	Event Description
1		R - ATC	Lower power to 850 MWe to remove 230 kV Line 30
2		N - BOP	Remove 230 kV Line 30 from service
3	RD001M (26-11)	C - ATC C - CRS	Rod Drift (TS)
4	K4526A	C - BOP C - CRS	ADHR primary pump trip (AOP)
5	NI063F	C - ATC C - CRS	Recirc Loop B Flow transmitter Failure (TS)
6	CF089F	C - BOP C - CRS	Heater Drain Deaerator Pump Trip (AOP)
7	CA008F	M	Small steam leak in DW results in an ATWS requiring terminate and prevent actions (RSP)(ATWS)
8	K2119A	C	SLC Mode Switch Failure
9	K2624A	C	Alternate Rod Insertion reset failure

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

Event	Event Description Summary
1	After taking the watch the CRS will direct power reduced to 850 MWe.
2	The BOP will isolate 230 kV Line 30.
3	Control Rod 26-11 will start to drift in. The crew will enter 0AOP-02.0 and take action IAW 2APP-A-05 (3-2). When the high temperature alarm is received Engineering will report that scram times cannot be assured based on past history of the control rod. Determine TS 3.1.3 condition C1 to insert the control rod in 3 hours and C2 to disarm the control rod within 4 hours.
4	After Tech Specs are addressed the Alternate Decay Heat Removal (ADHR) primary pump will trip. AOP-38.0 will be entered
5	The Recirc Loop B flow transmitter to APRM Channel 4 will fail downscale resulting in a rod block and a trip input to each voter. The crew will respond per APPs and bypass APRM 4. The APRM will be declared Inoperable per TS 3.3.1.1, Condition A and placed in trip within 12 hours. APRM TS Actions to be taken requires the APRM mode selector switch to be place in INOP IAW 0OI-18
6	A motor overload will occur on Heater Drain Pump 2A. The crew will reference APP UA-06 1-7, Bus 2D 4KV Motor Ovld and determine which pump has the overload condition. The crew should start HDP 2C and secure HDP 2A. The crew may reference AOP-23.0.
7	A small steam leak in the DW results in rising Drywell pressure requiring a reactor scram. An ATWS will occur, conditions will require terminate and prevent actions to be performed.
8	When SLC is initiated, the mode switch will fail and the pumps will not start. LEP-03 will be executed to inject the boron into the core.
9	Alternate Rod Insertion (ARI) will not reset, the crew will perform LEP-02 to drive control rods into the core. When level is stabilized after terminating and preventing ARI will be repaired to allow the rods to be manually scrammed.

Facility: Brunswick Scenario No.: 3 Op-Test No.: 2016 Draft

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions:

Unit Two is at 100% power.
1A NSW Pump is under clearance for planned maintenance.
2C TCC Pump is aligned to Unit One.

Turnover

The BOP will perform PT-40.2.11, Main Generator Voltage Regulator Manual And Automatic Operational Check.

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP	Perform PT-40.2.11, Main Generator Voltage Regulator Manual And Automatic Operational Check
2	ZA411	C-ATC C-CRS	DWEDT Pump failure
3	RC053F	C-ATC C-CRS	VFD Cell Failure (TS)(AOP)
4		R-ATC	Power maneuver
5	CW019F	C-BOP C-CRS	NSW Pump 2B Trip (failure of standby to start) (TS)(AOP)
6	CW039F	C-BOP C-CRS	CWIP Trip (AOP)
7	RW013F	M C	RWCU leak / Scram SBGT Fails to start (AOP)(RSP)(SCCP)
8	K1507A	M C	ED Failure of 2 ADS valves to open (EDP)

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

Event	Event Description Summary
1	Perform PT-40.2.11, Main Generator Voltage Regulator Manual And Automatic Operational Check.
2	Annunciator A-04 1-1, Drywell Equip Drain Sump Lvl Hi, will annunciate and the sumps will not auto start. One of the sump pumps will need to be manually started
3	A power cell in VFD A will fail. Recirc Pump 2A speed will lower and a speed hold will initiate. Loop flows will be outside mismatch limits.
4	The crew will reset the speed hold and match loop flows.
5	NSW Pump B will trip and the crew will start NSW Pump A. Since 1A NSW Pump is out of service, Tech Specs will apply. Crew will enter 0AOP-18.0, Nuclear Service Water System failure, and carry out appropriate actions.
6	Circulating Water Pump 2A will trip on motor winding fault, and the standby Circulating Water Intake Pump will be started. 0AOP-37.0 will be entered due to lowering vacuum.
7	A large un-isolable RWCU leak will occur. Crew will enter AOP-5.0 and SCCP. The CRS should direct a SCRAM. SBGT will fail to auto start and should be manually started.
8	Secondary containment conditions will worsen, forcing the CRS to direct an Emergency Depressurization (or Anticipation of Emergency Depressurization) due to high water levels. If Anticipation is performed, the second area high water level will annunciate requiring the emergency depressurization. Two ADS SRV's will fail to manually open. The CRS should direct opening two additional SRV's.

Facility: Brunswick Scenario No.: 4 Op-Test No.: 2016 Draft

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions:

Unit Two is at 100% power.
1A NSW Pump is under clearance for planned maintenance.
2C TCC Pump is aligned to Unit One.

Turnover

The BOP will start CREV in the area high radiation mode for inspection testing IAW 0OP-37, Section 6.1.3.

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP	Manual start of CREV in area high radiation mode.
2	NB007F	C-ATC C-CRS	C32-N004A Fails High (TS)
3	EE030M-2TD	C-BOP C-CRS	MCC 2TD trip / Standby Stator Water Cooling Pump fails to auto start
4	ES025F	C-ATC C-CRS	RCIC steam leak (AOP)(TS)
5	K4516A	C-BOP C-CRS	TCC Pump Failure (AOP)
6		R-ATC	Power Reduction
7	EE009F	M C	Loss of Off-Site Power / Scram DG3 failure to auto start / DG4 Diff O/C (RSP)(PCCP)(AOP)
8	ES004F CA020F	C M	SRV Failure / Tailpipe Break / DW Spray Logic Failure ED on PSP (AOP)(EDP)

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

Event	Event Description Summary
1	The BOP will start CREV in the area high radiation mode IAW 2OP-37.
2	After CREV is started, C32-LT-N004A will fail high. The crew will reference Tech Spec 3.3.2.2 and determine a 7 day LCO exists to place the failed channel in the tripped condition. The crew should select level B per OP-32.
3	MCC 2TD will trip and the standby stator cooling water pump will fail to auto start. The standby stator cooling water pump can be manually started. The 2D air compressor will also be lost and 0AOP-20.0 may be entered. Unit One may be contacted to place the 1D Air Compressor in lead.
4	A break in the RCIC steam line in the south RHR room will occur. The break can be isolated by closing either the E51-F007 or the E51-F008. The crew will respond to the steam leak IAW AOP-05.0.
5	TBCCW Pump 2B will trip and TBCCW low header pressure will alarm. The crew will respond per 0AOP-17.0. TBCCW pressure will recover and actions for partial loss of TBCCW will be performed.
6	A power reduction will be required IAW AOP-17.0.
7	A Loss of Offsite Power will occur. The crew will respond per 0AOP-36.1. DG3 will fail to auto start and DG4 will trip on Diff O/C. The BOP operator will start DG3 to energize bus E3 and perform 0AOP-36.1.
8	SRV F will fail open. AOP-30 will be entered. The SRV will not reset using the control switch. Pulling fuses IAW AOP-30 results in loss of indication but the SRV remains open. SRV F tailpipe will rupture, pressurizing containment. The DW Spray logic (think switch) will fail causing an inability to spray the torus or drywell. Emergency Depressurization is required when PSP is violated.

Facility: Brunswick   Scenario No.: 5   Op-Test No.: 2016 Draft

Examiners: ____________   Operators: ____________

____________   ____________

____________   ____________

Initial Conditions:

A2X sequence at step 166.
Core Spray Loop B under clearance, remaining ECCS LP systems protected.
Permission for continuous withdrawal has been granted for the rods going from 12-48.
IRM A was bypassed due to spiking and the paperwork is being evaluated by the WCC SRO for its return to service.

Turnover

BOP OP complete Step 6.3.46 of 0GP-02, Approach to Criticality and Pressurization of the Reactor.
IAW the reactivity plan the ATC operator is to raise power to 6-10% using rods.

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP	Place 2A RFPT level control in automatic
2		R-ATC	Raise reactor power using control rods
3	RD032M	C-ATC C-CRS	Difficult to move control rod (AOP)
4	K4510C	C-BOP C-CRS	Steam Packing Exhauster Trip
5	NI018F	C-ATC C-CRS	IRM Failure (TS)
6	ED_IAD CGJ6	C-BOP C-CRS	DG3 / E3 / E7 control Power loss (AOP)(TS)
7	CA002F	M C	Lowering Torus Level / RHR F028A mech trip / RHR F024B thermal trip / CS F020A broke (PCCP)
8	RP008F	M	Scram / Emergency Depressurization (RSP)(ATWS)(EDP)

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

Event	Event Description Summary
1	0GP-02, Approach to Criticality and Pressurizations of the Reactor will be completed starting at Step 6.3.46 to place the RFP master controller into Auto.
2	The crew will raise power by pulling control rods in preparation for placing the Mode switch to RUN. Rod pulls will commence at Step 161 (42-39 @ 12) of the A2X sequence.
3	Control rods will continue to be withdrawn raising power. When control rod 42-23 is selected for withdrawal, it will be stuck at position 12. AOP-02 may be entered and 2OP-07, Section 8.2 actions are required to withdraw a difficult intermediate control rod.
4	SPE 2A will trip causing a loss of gland sealing header pressure. SPE 2B will be placed in service
5	While withdrawing control rods, IRM C will fail upscale causing a rod block and half scram. SRO will address IRM A and C inoperability IAW TS 3.3.1.1. Once addressed, I&C will report IRM A is ready to be returned to service following proper channel check. The crew will take the actions of the APP and bypass IRM C and reset the half scram.
6	DC Panel 2A will trip resulting in loss of control power to DG 3, Bus E3 and Bus E7. The crew will respond per AOP-39.0 and transfer the control power to alternate. DG 3, Bus E3 and Bus E7 are inoperable until transferred to alternate supply. Once control power is transferred, a 7 day action is required to restore to the normal source. The BOP operator will return DG 3 to AUTO IAW AOP-39.0.
7	Torus level will begin to lower due to an unisolable leak on RHR suction. If attempted to raise torus water level, on RHR A loop the E11-F028A (Torus Discharge Isol Vlv) will trip when opened, on RHR B loop the E11-F024B (Torus Cooling Isol Vlv) will thermal trip when opened, and on Core Spray the E21-F002A (Core Spray Pump A Suction Valve From The Condensate Storage Tank) handwheel will be broke.
8	Before level reaches -5.5 feet in the torus a reactor scram is required. When torus water level reaches -5.5 feet emergency depressurization is required. The crew can anticipate emergency depressurization.

DUKE ENERGY

BRUNSWICK TRAINING SECTION

OPERATIONS TRAINING

INITIAL LICENSED OPERATOR

SIMULATOR EVALUATION GUIDE

2016 NRC SCENARIO 1

SWAP CBP, HPCI INITIATION, OFFGAS FILTER DP, RR PP TRIP, HDD CONT FAILURE, FUEL FAILURE, SCRAM FAILURE, SPC VLV FAILURE

REVISION 0

Developer: Bob Bolin	**Date:** 07/07/2016
Technical Review: Dan Hulgin	**Date:** 9/12/2016
Validators: Dwayne Wolf Shawn Zander Grant Newton	**Date:** 09/06/16
Facility Representative: Craig Oliver	**Date:** 09/22/2016

REVISION SUMMARY	
0	Scenario developed for 2016 NRC Exam.

TABLE OF CONTENTS

1.0 SCENARIO OUTLINE

Event	Malf. No.	Type*	Event Description
1		N - BOP	Swap Condensate Booster Pumps
2	ES014F	C - ATC C - CRS	Inadvertent HPCI Initiation w/ failure to trip (TS)(AOP-03.0)
3	ZUA343	C - BOP C - CRS	Off Gas Filter Hi d/P
4	RC055D	C - ATC C - CRS	Recirc Pump Trip (TS)(AOP-04.0)
5		R - ATC	Reduce reactor power for Single Loop Operation
6	CF039F	C - BOP C - CRS	Heater Drain Deaerator Controller Failure (AOP-23.0)
7	NB005F	M	Fuel Failure (RSP)(AOP-05.4)
8	K2501A K2503A	C	Manual Scram Failure – Alternate Rod Insertion Close Group I Valves
9	RH020F	C	Torus cooling Valve Failure (PCCP)
*(N)ormal, (R)eactivity, (C)omponent or Instrument, (M)ajor			

2.0 SCENARIO DESCRIPTION SUMMARY

Event	Description
1	The Unit will be operating at rated power and after taking the shift turnover will swap condensate booster pumps IAW 2OP-32. (Start 2C, secure 2A)
2	After the condensate booster pumps have been swapped, an inadvertent HPCI initiation will occur. The HPCI system will not trip but can be manually isolated. AOP-03.0 and Tech Specs will be entered.
3	After Tech Specs are addressed the Off Gas Filter Hi d/P alarm will be received. The standby Off Gas filter will be placed in service.
4	VFD coolant leakage will occur that will cause a trip of the 2B Recirc pump. The supply breaker does not trip and must be opened manually. 0AOP-04.0 will be entered. Technical Specifications will be addressed. .
5	The plant will be greater than the allowable for single loop operation. Recirc flow must be lowered and/or control rods must be inserted to reduce power. With only one recirc pump in operation reactor power must be less than 50% and core flow must be greater than 30.8 Mlbs but less than 45 Mlbs.
6	After the power reduction the Heater Drain Deaerator controller will fail. AOP-23.0 will be entered.
7	Fuel failure will begin requiring entry into AOP-05.4. When Process Off gas Hi Hi alarm with the Main Stack rising a scram will be inserted and Group I valves closed.
8	The crew will be required to (1) Scram, (2) Close Group I valves. (Each of which are critical tasks) The manual scram pushbuttons will not work, Alternate Rod Insertion (ARI) will be initiated to scram the rods.
9	Torus cooling valve will trip on thermals (will reset if attempted).

3.0 CREW CRITICAL TASKS

Critical Task #1
Insert control rods with ARI IAW the Scram hard Card or LPC when Process Off gas Hi Hi annunciator is received and Main Stack Rad is rising.
Critical Task #2
Close Group 1 isolation valves when Process Off gas Hi Hi annunciator is received and Main Stack Rad is rising.

4.0 TERMINATION CRITERIA

When all control rods are inserted and suppression pool cooling is being placed in service (F024A valve failure recognized) the scenario may be terminated.

5.0 IMPLEMENTING REFERENCES

NOTE: Refer to the most current revision of each Implementing Reference.

Number	Title
A-01 (1-4)	HPCI LOW FLOW
A-01 (3-5)	HPCI ISOLATION TRIP SIG A INITIATED
A-01 (4-4)	HPCI SYS PRESS LO
2APP-UA-03, 5-3	OFFGAS STBY FILTER DIFF-HIGH
2APP-A-07, 2-3	RECIRC VFD B ALARM UNACK
2APP-A-07, 3-3	RECIRC VFD B ALARM
2APP-A-07, 3-4	RECIRC VFD B COOLING SYS TROUBLE
2APP-A-07, 4-3	RECIRC VFD B TRIP WARNING
2APP-A-07, 4-6	RECIRC LOOP B ONLY OUT OF SERVICE
2APP-A-07, 5-3	RECIRC VFD B TRIPPED
2APP-A-07, 5-5	PUMP B SEAL STAGING FLOW HI/LO
2APP-A-05, 4-8	OPRM TRIP ENABLED
2APP-UA-23, 4-4	E REHEATER FIRST STAGE LEVEL HI-LO
2APP-UA-23, 4-5	W REHEATER FIRST STAGE LEVEL HI-LO
2OP-30	Condenser Air Removal And Off-Gas Recombiner System
2AOP-04	Low Core Flow
2APP-UA-04, 2-10	HD DEAERATOR LEVEL HIGH-LOW.
2APP-UA-03, 4-6	AREA RAD RADWASTE BLDG HIGH
2APP-UA-03, 5-2	PROCESS OFF-GAS RAD HIGH
2APP-UA-03, 5-7	AREA RAD TURBINE BLDG HIGH
2APP-UA-23, 2-6	MAIN STEAM LINE RAD HI
2APP-UA-23, 3-6	MAIN STEAM LINE RAD HI-HI/INOP
0AOP-05.4	Radiological Release
2EOP-01-RSP	Reactor Scram Procedure
2EOP-01-RVCP	Reactor Vessel Control Procedure
0EOP-02-PCCP	Primary Containment Control Procedure
0EOP-04-RRCP	Radioactivity Release Control Procedure

6.0 SETUP INSTRUCTIONS

1. **PERFORM** TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 5, Checklist for Simulator Exam Security.
2. **RESET** the Simulator to IC-11.
3. **ENSURE** the RWM is set up as required for the selected IC.
4. **ENSURE** appropriate keys have blanks in switches.
5. **RESET** alarms on SJAE, MSL, and RWM NUMACs.
6. **ENSURE** no rods are bypassed in the RWM.
7. **PLACE** all SPDS displays to the Critical Plant Variable display (#100).
8. **ENSURE** hard cards and flow charts are cleaned up
9. **TAKE** the SIMULATOR OUT OF FREEZE
10. **LOAD** Scenario File.
11. **ALIGN** the plant as follows:

Manipulation
1. Ensure 2C TCC Pump is in service on Unit One. 2. Bypass APRM 2

12. **IF desired**, take a **SNAPSHOT** and save into an available IC for later use.
13. **PLACE** a clearance on the following equipment.

Component	Position
APRM 2	Blue Tag

14. **INSTALL** Protected Equipment signage and **UPDATE** RTGB placard as follows:

Protected Equipment
1. 2A and 2B NSW pumps 2. 2A FPC Pump/Hx, 2A RCC Pump, and 2C Demin Transfer Pump.

15. **VERIFY** 0ENP 24.5 Form 2 (Immediate Power Reduction Form) for IC-11 is in place.

16. **ENSURE** each Implementing References listed in Section 7 is intact and free of marks.

17. **ENSURE** all materials in the table below are in place and marked-up to the step identified.

Required Materials
2OP-32, Section 6.3.6 has been completed up to step 6.3.6.11

18. **ADVANCE** the recorders to prevent examinees from seeing relevant scenario details.

19. **PROVIDE** Shift Briefing sheet for the CRS.

20. **VERIFY** all actions contained in TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 4, Simulator Training Instructor Checklist, are complete.

7.0 INTERVENTIONS

TRIGGERS

Trig	Type	ID
1	Remote Function	EE_UTSHED4 - [UNIT TRIP LOAD SHED SEL SW, 2A COND BOOSTER PUMP]
1	Remote Function	EE_LSHED4 - [LOCA LOAD SHED SEL SW, 2A CONDENSATE BOOSTER PUMP]
2	Remote Function	EE_LSHED6 - [LOCA LOAD SHED SEL SW, 2C CONDENSATE BOOSTER PUMP]
2	Remote Function	EE_UTSHED6 - [UNIT TRIP LOAD SHED SEL SW, 2C COND BOOSTER PUMP]
3	Malfunction	ES014F - [INADVERTANT HPCI SYS INITIATION]
4	Annunciator	ZUA343 - [OFF GAS FILTER DIFF-HIGH]
5	Malfunction	RC055D - [VFD COOLING SYSTEM LEAKAGE]
6	Trigger Command	DOD:Q2735RRH
7	Trigger Command	DOD:Q2735LGH
8	Malfunction	CF039F - [HTR DRN DEAER LVL CNTRLR FAILURE]
9	Malfunction	NB005F - [FUEL FAILURE]
10	Malfunction	NB010F - [GROSS FUEL FAILURE]
11	Remote Function	MI_IACBLRM1 - [UNIT 1 CABLE SPREAD ROOM VENT FANS]
12	Remote Function	MI_ZVACS918_1 - [UNIT 1 CB MECHANICAL EQUIP ROOM VENT FANS CS]
13	Remote Function	CF_ZVCF5719 - [HWC-SV-5719 - H2 INJECTION OPEN VLV]
14	Remote Function	CF_ZXCF5717 - [HWC-SV-5717 - H2 INJECTION CLOSE VLV]
15	Trigger Command	and:zua343

Trig #	Trigger Text
6	!K2735NWD - Not [2B RECIRC STOP] delete Q2735RRH
7	K2735BPH - [2B RECIRC STOP] delete Q2735LGH
15	QK214RRZ - [AOG SYS ISOL HCV-101 RED]

MALFUNCTIONS

Malf ID	Mult ID	Description	Current Value	Target Value	Rmp time	Actime	Dactime	Trig
ES014F		INADVERTANT HPCI SYS INITIATION	False	True				3
RC055D	VFD B	VFD COOLING SYSTEM LEAKAGE	0.00	100.00	00:01:00			5
CF039F		HTR DRN DEAER LVL CNTRLR FAILURE	False	True				8
NB005F		FUEL FAILURE	0.00	100.00	00:10:00			9
NB010F		GROSS FUEL FAILURE	0.00	100.00	00:10:00			10
RH020F	E11-F024B	FULL FLOW * VLV E11-F024B	True	True				
RP006F		MANUAL SCRAM DEFEAT	True	True				
NI032F	APRM 2	APRM FAILS LO	True	True				

REMOTES

Remf Id	Mult Id	Description	Current Value	Target Value	Rmp time	Actime	Trig
CC_IACW4518		2C TBCCW PUMP UNIT ALIGNMENT	1	1			
EE_LSHED4		LOCA LOAD SHED SEL SW, 2A CONDENSATE BOOSTER PUMP	DISABLE	ENABLE			1
EE_LSHED6		LOCA LOAD SHED SEL SW, 2C CONDENSATE BOOSTER PUMP	ENABLE	DISABLE			2
EE_UTSHED4		UNIT TRIP LOAD SHED SEL SW, 2A COND BOOSTER PUMP	DISABLE	ENABLE			1
EE_UTSHED6		UNIT TRIP LOAD SHED SEL SW, 2C COND BOOSTER PUMP	ENABLE	DISABLE			2
MI_IACBLRM1		UNIT 1 CABLE SPREAD ROOM VENT FANS	AUTO	OFF			11
MI_ZVACS918_1		UNIT 1 CB MECHANICAL EQUIP ROOM VENT FANS CS	NEUT	STOP			12
CF_ZVCF5719		HWC-SV-5719 - H2 INJECTION OPEN VLV	NORMAL	OPEN			13
CF_ZXCF5717		HWC-SV-5717 - H2 INJECTION CLOSE VLV	NORMAL	CLOSE			14

PANEL OVERRIDES

Tag ID	Description	Position / Target	Actual Value	Override Value	Rmp time	Actime	Dactime	Trig
K1C08A	REMOTE TURBINE	TURB_TRIP	OFF	OFF				
Q2735RRH	2B RECIRC RED	ON/OFF	ON	ON				
Q2735LGH	2B RECIRC GREEN	ON/OFF	OFF	OFF				

ANNUNCIATORS

Window	Description	Tagname	Override Type	OVal	AVal	Actime	Dactime	Trig
4-3	OFF GAS FILTER DIFF-HIGH	ZUA343	ON	ON	OFF			4

8.0 OPERATOR RESPONSE AND INSTRUCTIONAL STRATEGIES

EVENT 1: SWAP CBPs	
Simulator Operator Actions	
	Ensure Monitored Parameters is open and Scenario Based Testing Variables are loaded.
	Initiate Trigger 1 to ENABLE 2A CBP Unit Trip and LOCA Load Shed Selector Switches.
	Initiate Trigger 2 to DISABLE 2C CBP Unit Trip and LOCA Load Shed Selector Switches
	Initiate Trigger 13 to place HWCH-SV-5719 to open.
	Initiate Trigger 14 to place HWCH-SV-5717 to close.

Simulator Operator Role Play	
	Acknowledge request to enable 2A CBP Unit Trip and LOCA Load Shed selector switches, after Sim Operator activates the trigger report that the action is complete.
	Acknowledge request to disable 2C CBP Unit Trip and LOCA Load Shed selector switches, after Sim Operator activates the trigger report that the action is complete.
	Acknowledge request to place HWCH-SV-5719 (Condensate Booster Pump C H2 Injection Isolation Valve) in AUTO, have SIM OP initiate trigger 13 and report valve is open.
	Acknowledge request to place HWCH-SV-5717 (Condensate Booster Pump B H2 Injection Isolation Valve) in CLOSE and then report that the action is complete
	Acknowledge any requests for Radwaste actions.
	Acknowledge request to perform 2OP-32, Attachment 10. Report that you will co-ordinate the performance of the attachment with the WCC.

Evaluator Notes	
Plant Response:	2C CBP is started and 2A CBP is secured.
Objectives:	SRO - Directs BOP to swap Condensate Booster Pumps BOP – Swap Condensate Booster Pumps RO – Monitors the plant
Success Path:	Condensate Booster Pumps are swapped
Event Termination:	When directed by the Lead Evaluator, go to Event 2.

EVENT 1: SWAP CBPs

Time	Pos	EXPECTED Operator Response	NOTES
	SRO	Conduct shift turnover shift briefing.	
		Direct CBPs to be swapped. (2OP-32, Section 6.3.6)	
		May conduct a brief (see Enclosure 1 on page 43 for format)	
	RO	Monitors the plant	
	BOP	Swap Condensate Booster Pumps IAW 2OP-32, Section 6.3.6: Make a PA announcement for starting 2C Condensate Booster Pump and Securing 2A Condensate Booster Pump. Should also check that Bus C is clear of personnel, may state that they would use plant cameras. Performs 2OP-32, Section 6.3.6. Per the turnover, Steps 1-10 are complete. Directs an AO to perform steps 12a, 12g, and 12h. Directs Radwaste to perform steps 21 and 22. Either directs an AO or the WCC SRO to perform Attachment 10 of 2OP-32.	

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

9. **IF** 2B Condensate Booster Pump is to be started, **THEN perform** the following: .... N/A
   a. **Open** COD-V5016 (Condensate Booster Pump B Inboard Mechanical Seal Vent Valve). .... N/A
   b. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5016 (Condensate Booster Pump B Inboard Mechanical Seal Vent Valve). .... N/A
   c. **Open** COD-V5017 (Condensate Booster Pump B Outboard Mechanical Seal Vent Valve). .... N/A
   d. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5017 (Condensate Booster Pump B Outboard Mechanical Seal Vent Valve). .... N/A
10. **IF** 2C Condensate Booster Pump is to be started, **THEN perform** the following: .... RO
    a. **Open** COD-V5018 (Condensate Booster Pump C Inboard Mechanical Seal Vent Valve). .... AO
    b. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5018 (Condensate Booster Pump C Inboard Mechanical Seal Vent Valve). .... AO
    c. **Open** COD-V5019 (Condensate Booster Pump C Outboard Mechanical Seal Vent Valve). .... AO
    d. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5019 (Condensate Booster Pump C Outboard Mechanical Seal Vent Valve). .... AO
11. **Note** status of the following alarms:
    - UA-13, 1-1, Gen-Xfmr Primary L/O Unit Trip .... ______
    - UA-13, 1-2, Generator Diff L/O Unit Trip .... ______
    - UA-13, 1-3, Gen-Xfmr Backup L/O Unit Trip .... ______

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

12. IF all the alarms in Step 11 are CLEAR,
THEN **perform** the following:[8.7.5] .... ______

a. **Place** the following switches in ENABLED for the off going condensate booster pump:

- Unit Trip Load Shed Selector Switch .... ______
- LOCA Load Shed Selector Switch............................ ______

b. **Ensure** the oncoming condensate booster pump mode selector switch in MAN.. ______

c. **Confirm** oncoming condensate pump discharge valve closes, if OPEN:

- COD-V4 (Condensate Booster Pump A Disch Vlv) .... ______
- COD-V5 (Condensate Booster Pump B Disch Vlv) .... ______
- COD-V6 (Condensate Booster Pump C Disch Vlv) .... ______

NOTE
The start/stop control switch for a Condensate Booster Pump must be held in START until the recirculation valve is full OPEN. The recirculation valve does **NOT** stroke OPEN until all other condensate booster pump permissives are met. While opening the recirc valve, condensate booster pump suction pressure will lower as additional flow up to 2000 gpm is routed to the condenser. .... ☐

d. **Start** the oncoming condensate booster pump. .... ______

e. **Confirm** oncoming condensate booster pump discharge valve opens:

- COD-V4 (Condensate Booster Pump A Disch Vlv) .... ______
- COD-V5 (Condensate Booster Pump B Disch Vlv) .... ______
- COD-V6 (Condensate Booster Pump C Disch Vlv) .... ______

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

f. **WHEN** condensate booster pump discharge pressure stabilizes,
THEN perform the following: .... ________

(1) **Stop** off going condensate booster pump. .... ________

(2) **IF** B21-F032A (Feedwater Isol Vlv) **AND** B21-F032B (Feedwater Isol Vlv) are OPEN,
THEN place the off going condensate booster pump mode switch in AUTO. .... ________

g. **Place** the following switches in DISABLED for oncoming condensate booster pump:

- Unit Trip Load Shed Selector Switch .... ________
- LOCA Load Shed Selector Switch .... ________

h. **IF** HWC is in service,
THEN perform the following: .... ________

(1) **Place** HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, for oncoming condensate booster pump, in AUTO:

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve) .... ________

(2) **Confirm** HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, for oncoming condensate booster pump, is OPEN:

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve) .... ________

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

(3) **Place** the HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, for off going condensate booster pump in CLOSED.

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve) .... _____
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve) .... _____
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve) .... _____

i. **Go to** Step 20. .... _____

13. **Place** the following switches in DISABLED for the oncoming Condensate Booster Pump:

- Unit Trip Load Shed Selector Switch .... N/A
- LOCA Load Shed Selector Switch .... N/A

14. **Place** the oncoming condensate booster pump mode selector switch in MAN. .... N/A

15. **Confirm** oncoming condensate booster pump discharge valve closes.

- COD-V4 (Condensate Booster Pump A Disch Vlv) .... N/A
- COD-V5 (Condensate Booster Pump B Disch Vlv) .... N/A
- COD-V6 (Condensate Booster Pump C Disch Vlv) .... N/A

NOTE

The start/stop control switch for a condensate booster pump must be held in START <u>until</u> the recirculation valve is full OPEN. The recirculation valve does **NOT** stroke OPEN <u>until</u> all other condensate booster pump permissives are met. While opening the recirc valve, condensate booster pump suction pressure will lower as additional flow up to 2000 gpm is routed to the condenser. .... N/A

16. **Start** oncoming condensate booster pump .... N/A

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

17. **Confirm** oncoming condensate booster pump discharge valve opens:

- COD-V4 (Condensate Booster Pump A Disch Vlv) .... N/A
- COD-V5 (Condensate Booster Pump B Disch Vlv) .... N/A
- COD-V6 (Condensate Booster Pump C Disch Vlv) .... N/A

18. **WHEN** condensate booster pump discharge pressure stabilizes, **THEN perform** the following: .... N/A

a. **Stop** off going condensate booster pump. .... N/A

b. **IF** B21-F032A (Feedwater Isol Vlv) **AND** B21-F032B (Feedwater Isol Vlv) are OPEN, **THEN place** the off going condensate booster pump mode switch in AUTO. .... N/A

c. **Place** the following switches in ENABLED for the off going condensate booster pump:

- Unit Trip Load Shed Selector Switch .... N/A
- LOCA Load Shed Selector Switch .... N/A

19. **IF** HWC is in service, **THEN perform** the following: .... N/A

a. **Place** oncoming condensate booster pump, HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, in AUTO:

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve) .... N/A
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve) .... N/A
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve) .... N/A

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

b. **Confirm** oncoming condensate booster pump, HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, OPEN:

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve)........ N/A
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve)........ N/A
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve)........ N/A

c. **Place** off going condensate booster pump, HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, in CLOSED:

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve)........ N/A
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve)........ N/A
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve)........ N/A

20. <u>IF</u> three condensate pumps are in service per Step 5.b(2), <u>THEN</u> **perform** Section 6.3.23, Securing from Three Condensate Pump Operation........ ____

21. **Direct** Radwaste Operator to remove the additional CFD or CDD placed in service in Step 5.a........ ____

22. **Direct** Radwaste Operator to monitor CDD effluent conductivity for each demineralizer in service. .... ____

23. <u>IF</u> any condensate pump Unit Trip Load Shed Selector Switch <u>AND</u> LOCA Load Shed Selector Switch was manipulated, <u>THEN</u> **complete** Attachment 9, Condensate Pump Unit Trip Load Shed/LOCA Load Shed Selector Switch Alignment Documentation........ ____

24. **Complete** Attachment 10, Condensate Booster Pump Unit Trip Load Shed/LOCA Load Shed Selector Switch Alignment Documentation........ ____

ATTACHMENT 10
Page 1 of 1

Condensate Booster Pump Unit Trip Load Shed/LOCA Load Shed Selector Switch Alignment Documentation

NOTE

- When performing this attachment, pump selector switches, for pump(s) which have **NOT** been realigned may be marked NA. ☐
- The condensate booster pump selected for standby operation (Mode Selector Switch in AUTO) is required to have both the LOCA Load Shed Selector Switch and Unit Trip Load Shed Selector Switch in ENABLED. ☐

1. Circle the required selector switch position.

Number	Description	Position/ Indication [Note 1]	Checked	Verified
Turbine Building - 4160V Switchgear Bus 2D - Elev 20 ft				
AD9	2B Condensate Booster Pump Unit Trip Load Shed Selector Switch	ENABLED/ DISABLED		
AD9	2B Condensate Booster Pump LOCA Load Shed Selector Switch	ENABLED/ DISABLED		
Turbine Building - 4160V Switchgear Bus 2C - Elev 20 ft				
AC1	2A Condensate Booster Pump Unit Trip Load Shed Selector Switch	ENABLED/ DISABLED		
AC1	2A Condensate Booster Pump LOCA Load Shed Selector Switch	ENABLED/ DISABLED		
AC2	2C Condensate Booster Pump Unit Trip Load Shed Selector Switch	ENABLED/ DISABLED		
AC2	2C Condensate Booster Pump LOCA Load Shed Selector Switch	ENABLED/ DISABLED		

EVENT 2: HPCI INITIATION	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 3** to initiate HPCI.
	After isolated, **delete** HPCI Inadvertent Initiation.

Simulator Operator Role Play	
	If contacted as Reactor Engineer to look at thermal limits due to HPCI injection, report that he will evaluate and monitor
	If contacted as I&C to assist with troubleshooting, after HPCI is isolated and Tech Spec are addressed, remove the HPCI initiation and trip failures and then report that I/C found a relay failure that caused the initiation signal. It will take about 4 hours to replace the relay but the initiation signal is now clear. The trip pushbutton had a loose wire, which has been re-attached.
	If contacted as chemistry acknowledge request for coolant samples for indications of fuel failure.

Evaluator Notes	
Plant Response:	HPCI will inadvertently initiate. The crew will verify level and then secure HPCI. HPCI manual isolation pushbutton will fail. If injection occurs, the crew will enter AOP-03. Technical Specifications will be addressed.
Objectives:	SRO - Direct actions in response to an inadvertent HPCI initiation and potential positive reactivity addition Determine actions required for LCO per Technical Specifications RO - Respond to an inadvertent HPCI initiation and potential positive reactivity addition
Success Path:	Verify HPCI initiation signal not present and isolate HPCI
Event Termination:	Go to Event 3 at the direction of the Lead Evaluator.

EVENT 2: HPCI INITIATION

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct crew to trip HPCI following verification of false initiation	
	SRO	Direct crew to isolate HPCI on failure of trip pushbutton (May use isolation pushbutton or direct steam supply valves to be closed).	
	SRO	Direct crew to enter and execute 0AOP-3.0 Positive Reactivity Addition, if injection has occurred.	
	SRO	Contact maintenance to look at the HPCI Initiation signal. May also contact Reactor Engineer to look at thermal limits.	
	SRO	Evaluate Tech Spec 3.5.1 ECCS - Operating Condition D1 - Verify RCIC is OPERABLE Condition D2 - Restore HPCI in 14 days	
	SRO	Direct HPCI shutdown IAW 2OP-19 after I/C confirms signal has cleared.	
	SRO	May direct RCIC and ADS to be protected.	
	SRO	May conduct a brief (see Enclosure 1 on page 43 for format)	
	BOP	Monitor reactor plant parameters during evolution	

EVENT 2: HPCI INITIATION

Time	Pos	EXPECTED Operator Response	Comments
	ATC	Verify false HPCI initiation signal (No LL2 signal present or high drywell pressure)	
	ATC	Trip HPCI by pushing the HPCI trip pushbutton, recognize failure of trip	
	ATC	May depress Manual Isolation System A pushbutton to isolate HPCI. OR May isolate the steam supply valves (E41-F002 and F003) to HPCI to isolate system.	
	ATC	Enter and execute AOP-3.0 Positive Reactivity Addition	
	ATC	Respond to the following A-01 alarms: 1-4, HPCI LOW FLOW 3-5, HPCI ISOLATION TRIP SIG A INITIATED 4-4, HPCI SYS PRESS LO	
	ATC	Perform 2OP-19 Section 6.2 to shutdown HPCI.	

6.2 Shutdown

6.2.1 HPCI System Shutdown

NOTE

HPCI running in Auto or Manual is an R2 Reactivity Management Evolution .... ☐

1. **Confirm** either Step 1.a or Step 1.b of the following initial conditions are met:

   a. **IF** the HPCI System has automatically initiated, **THEN confirm** one of the following exists: .... __________

      - The system is **NO** longer required to maintain reactor water level .... __________
      - The automatic initiation signal is **NOT** valid .... __________

   b. **IF** the HPCI System was manually started, **THEN confirm** both of the following exists: .... __________

      - The system is **NO** longer required to maintain reactor water level .... __________
      - The system is **NO** longer required to maintain reactor pressure .... __________

2. **IF** HPCI automatically initiated, **THEN perform** the following: .... __________

   a. **Confirm** from at least two independent indications at least one of the following conditions exist: .... __________

      - Adequate core cooling is ensured .... __________
      - The initiation signal was **NOT** valid .... __________
      - HPCI is **NOT** functioning properly in the AUTOMATIC mode .... __________

   b. **Place** E41-S21 (Vacuum Pump) control switch in START .... __________

   c. **Depress** E41-S17 (Initiation Signal/Reset) pushbutton .... __________

6.2.1 HPCI System Shutdown (continued)

NOTE

Operation of the HPCI System under minimum flow conditions is minimized. Section 6.2.1 Step 3 through Section 6.2.1 Step 11 are performed as expeditiously as possible. {8.1.2} ☐

3. **Ensure** E41-F006 (HPCI Injection Vlv) is CLOSED. ________
4. **Ensure** E41-F008 (Bypass To CST Vlv) is CLOSED. ________
5. **IF** OPEN **AND NOT** in use for RCIC operation, **THEN close** E41-F011 (Redundant Isol To CST Vlv). ________
6. **Ensure** E41-F012 (Min Flow Bypass To Torus Vlv) opens. ________

NOTE

If Turbine Trip pushbutton is released before E41-F001 (Turbine Steam Supply Vlv) is fully closed, the turbine will attempt to restart. ☐

7. **Close** E41-F001 (Turbine Steam Supply Vlv) and immediately **depress and hold** Turbine Trip pushbutton until E41-F001 is fully CLOSED. ________

END R.M. LEVEL R2 Reactivity Evolution

NOTE

E41-V8 (Turbine Stop Valve) closes while the Turbine Trip pushbutton is depressed. However, when the Turbine Trip pushbutton is released, E41-V8 will reopen until the HPCI oil pressure source is removed which will allow spring pressure to close the valve. ☐

8. **Ensure** E41-V8 (Turbine Stop Valve) closes. ________
9. **Ensure** E41-S20 (Auxiliary Oil Pump) auto starts as the turbine speed lowers. ________
10. **Close** E41-F059 (Cooling Water Supply Vlv). ________
11. **Ensure** E41-F012 (Min Flow Bypass To Torus Vlv) closes. ________
12. **Ensure** HPCI E41-FIC-R600 (Flow Control) is in AUTO (A). ________
13. **Adjust** HPCI E41-FIC-R600 (Flow Control) setpoint to 4300 gpm. ________

6.2.1 HPCI System Shutdown (continued)

NOTE

- Technical Specification 3.6.1.6.1 (MODE 1, 2, or 3) requires completion of 0PT-02.3.1B, Suppression Pool to Drywell Vacuum Breaker Position Check, within 6 hours after any discharge of steam to the suppression chamber from any source and within 6 hours following an operation that causes any of the vacuum breakers to open. .... ☐
- Section 6.2.1 Step 14 ensures compliance with Technical Specifications. .... ☐

14. **__IF AT ANY TIME__** in MODE 1, 2, or 3, after any discharge of steam to the suppression chamber from any source **__THEN__ ensure** 0PT-02.3.1B, Suppression Pool to Drywell Vacuum Breaker Position Indication Check, is completed within 6 hours. [8.2.1] .... ________

15. **__WHEN__** 15 minutes have elapsed after tripping the HPCI turbine, **__THEN__ stop** E41-S21 (Vacuum Pump). .... ________

16. **Place** E41-S21 (Vacuum Pump) control switch in AUTO. .... ________

17. **__WHEN__** differential temperature across the HPCI turbine bearings reduces to approximately 0°F, as indicated on E41-TR-R605, **__THEN__ stop** E41-S20 (Auxiliary Oil Pump). .... ________

18. **Place** E41-S20 (Auxiliary Oil Pump) in AUTO. .... ________

19. **Place** E41-S22 (Barometric Cndsr Condensate Pump) control switch in AUTO. .... ________

20. **Ensure** E41-F025 (Cond Pump Disch Otbd Drain Vlv) is OPEN. .... ________

21. **Shut down** Standby Gas Treatment System per 2OP-10, Standby Gas Treatment System Operating Procedure, Section for Control Room Manual Shutdown, and **return** to Section 6.2.1 Step 22 .... ________

22. **Complete** Section 6.1.1 to return the system to STANDBY. .... ________

EVENT 3: OFF GAS FILTER HI D/P	
Simulator Operator Actions	
	At the direction of the lead evaluator, **Initiate Trigger 4** to bring in Off-Gas Filter Diff-Hi annunciator

Simulator Operator Role Play	
	IF contacted as Outside AO to verify Off-gas filter Diff pressure, report local DP indication is reading 13 inches water.
	IF contacted as Unit One report steps 6.3.3.2 and 3 are complete. (1-OG-FV-244-4, 1-OG-FV-244-5, and 1-AOG-HCV-101 are closed)
	IF contacted as AO report 1-OG-CD-V7 is CLOSED (Step 4)
	IF contacted as AO report 2-OG-CD-V7 is OPEN (Step 5)
	IF contacted as Outside AO to verify Off-gas filter Diff pressure after filter swap, report local DP indication is reading 3 inches water.
	When contacted as AO report 2AOG-HCV-101 is in OPEN position.

Evaluator Notes	
Plant Response:	Off-Gas Filter Diff-Hi alarm annunciates.
Objectives:	SRO -Direct actions in response to a Off-gas Filter Diff-Hi alarm. BOP – Respond to clogged off-gas filter IAW APP and 2OP-30.
Success Path:	Swap Off-gas filters per OP-30.
Event Termination:	Go to Event 4 at the direction of the Lead Evaluator.

EVENT 3: OFF GAS FILTER HI D/P

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct crew to perform the actions of UA-3, 4-3, OFF GAS FILTER DIFF-HIGH alarm	
	SRO	Direct crew to swap Off-gas filters per OP-30 Section 6.3.3. Should identify that a PRR needs to be written for the APP (or verified that one has already been written) to reference the correct procedure section to swap off-gas filters (Section 6.3.3)	
	SRO	May conduct a brief (see Enclosure 1 on page 43 for format)	
	ATC	Monitors the plant.	
	BOP	Respond to UA-3, 4-3, OFF GAS FILTER DIFF-HIGH alarm Dispatch Outside AO to verify Off-gas filter Diff Hi locally	
	BOP	Place Off-gas Standby filter in service per OP-30 Section 6.3.3	

6.3.3 Placing Off Gas Standby Filter In Service

1. **Ensure** the following Initial Conditions are met:
   a. Off Gas filter is in service. .... _________
   b. Off Gas standby filter is available for operation. .... _________
2. **Ensure** the following Unit 1 valves are CLOSED:
   - 1-OG-FV-244-4 (OG Stby Filt Inlet Isol Vlv) .... _________
   - 1-OG-FV-244-5 (OG Stby Filt Inlet Isol Vlv) .... _________
3. **Ensure** 1-AOG-HCV-101 (Standby Off Gas Filter Outlet Isolation Valve) is CLOSED. .... _________
4. **Ensure** 1-OG-CD-V7 (Standby Filter Drain Valve) is CLOSED to prevent a cross-connect of Unit 1 and Unit 2 Off Gas Systems. .... _________
5. **Open** 2-OG-CD-V7 (Standby Filter Drain Valve). .... _________
6. **Open** 2-OG-FV-244-4 and 2-OG-FV-244-5 (OG Stby Filt Inl Vlvs). .... _________

NOTE

If 2-AOG-CS-3161 (AOG Sys Vlv Cont Sel Sw) on Panel XU-80 is **NOT** in CENT, 2-AOG-HCV-101 (Standby Off Gas Filter Outlet Isolation Valve) **CANNOT** be operated from the Control Room. .... ☐

7. **Open** 2-AOG-HCV-101 (Standby Off Gas Filter Outlet Isolation Valve). .... _________
8. **IF** Off Gas filter is to be taken out of service, **THEN close** 2-OG-FV-244-1, 2-OG-FV-244-2 and 2-OG-FV-244-3 (OG Filter Isol Vlvs). .... _________
9. **Ensure** Off Gas standby filter pressure differential is less than 10 inches of water. .... _________
10. At Local Control Panel H2E, **place** 2-AOG-HCV-101 (AOG System Isolation Valve) control switch to OPEN. .... _________

EVENT 4/5: 2B REACTOR RECIRC PUMP TRIP / EXIT SCRAM AVOIDANCE REGION	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 5,** VFD B Coolant Leakage.
	Note: Short time delay before first alarm (1.5 min), followed by pump trip.
	Malfunctions required: VFD B Coolant Leakage / 4 kV breaker override

Simulator Operator Role Play	
	If contacted as the AO to investigate VFD Alarms, wait until Recirc pump has tripped and report that the coolant pumps are tripped due to coolant leakage.
	If contacted as reactor engineer, report you will monitor thermal limits and use 0ENP-24.5 rods as needed to get below MELLL line.
	If contacted as NIT to backup OPRM data acknowledge the request.
	If requested as RBAO to reduce Recirc purge flow, wait 3 minutes and report actions complete.
	If contacted as I/C, acknowledge report of VFD coolant leak.
	If contacted as chemistry, acknowledge request for coolant sample for indications of fuel failure. If asked the results of the previous sample are still be processed.

Evaluator Notes	
Plant Response:	A coolant leak develops on VFD B that will cause the Recirc pump to trip. The supply breaker does not trip and must be opened manually. 2AOP-04.0 will be entered. Technical Specifications will be addressed. The plant will be in the Scram Avoidance Region of the power to flow map. Recirc flow must be increased or control rods must be inserted to exit this region. With only one Recirc pump in operation reactor power must be less than 50% and core flow must be greater than 30.8 Mlbs but less than 45 Mlbs.
Objectives:	SRO - Direct Shift Response To Recirculation Pump Trip Per 2AOP-04.0 ATC: Respond to a Recirc Pump trip IAW 2AOP-04.0, Reduce reactor power.
Success Path:	Identifies that the Recirc supply breaker did not trip and manually trips the breaker. Manipulates reactor power to exit the scram avoidance region using rods and / or Recirc flow
Event Termination:	Go to Event 6 at the direction of the Lead Evaluator.

EVENT 4/5: 2B REACTOR RECIRC PUMP TRIP / EXIT SCRAM AVOIDANCE REGION

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct response to Annunciators.	
	SRO	Direct entry into 2AOP-04.0, Low Core Flow	
	SRO	Determine region of operation on power/flow map (computer display 806 may be used for reference)	
	SRO	Direct actions to reduce power to ≤50% and Insert rods if greater than MELLL line. (Flow must be maintained >30.8 but <45 Mlbs/hr and power must be <50% for single recirc pup operation)	
	SRO	TS 3.4.1 Recirculation Loops Operating Determine Condition A applies Required Action A.1: APLHGR limits and APRM setpoints must be adjusted within 6 hours	
	SRO	May conduct a brief (see Enclosure 1 on page 43 for format)	
	BOP	Plant Monitoring	
	BOP	May determine region of operation on power/flow map (computer display 806 may be used for reference)	
	BOP	May monitor for THI Check the VFD HMI screens, report loss of cooling.	

EVENT 4/5: 2B REACTOR RECIRC PUMP TRIP / EXIT SCRAM AVOIDANCE REGION			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	ATC	Respond to the following alarms during this event: A-07: 2-3, RECIRC VFD B ALARM UNACK 3-3, RECIRC VFD B ALARM 3-4, RECIRC VFD B COOLING SYS TROUBLE 4-3, RECIRC VFD B TRIP WARNING 4-6, RECIRC LOOP B ONLY OUT OF SERVICE 5-3, RECIRC VFD B TRIPPED 5-5, PUMP B SEAL STAGING FLOW HI/LO A-05: 4-8 OPRM TRIP ENABLED UA-23: 4-4, E REHEATER FIRST STAGE LEVEL HI-LO 4-5, W REHEATER FIRST STAGE LEVEL HI-LO	
		Observes indications on the VFD B HMI on Panel XU-4	
		Dispatch AO to investigate	
		Diagnose and report supply breaker not tripped. IAW APP A-7, 5-3, Recirc VFD B Tripped and confirmation on XU-4, 2-B32-YFD-VDT-002B, Recirc VFD 2B identifies that both VFD coolant pumps are tripped, this indicates that the supply breaker should have tripped.	
		Opens Recirc Pump B - 4 kV Supply Breaker	
		Determine if valid core flow indication exists on process computer (WTCF).	

EVENT 4/5: 2B REACTOR RECIRC PUMP TRIP / EXIT SCRAM AVOIDANCE REGION

Time	Pos	EXPECTED Operator Response	Comments
		Determines region of operation on power/flow map (computer display 806 may be used for reference)	
		May do one or both of the following to lower power to less than 50%: 1. Insert control rods Turns select power on Selects a control rod from the ENP-24.5, Immediate Power Reduction sheet. (see page 35) Drive the selected rod as specified for each group of rods as needed.. 2. Lower core flow using the running Recirc Pump	
		Reduce CRD flow to 30 gpm per 2OP-08.	
		If charging pressure high alarms, may request CRD Pump A discharge valve throttled closed per the APP	
		Maintain core flow >30.8 E6 lb/hr to prevent excessive cooldown of idle loop. or If Core Flow is <30.8 E6 lb/hr log bottom head and loop temperature every 15 minutes	
		Monitor for THI	
		Notify Reactor Engineer	
		Notify chemistry	

FOR SIMULATOR USE ONLY
IC–11 Revised 3-22-2015

FORM 2
Page 2 of 3
Immediate Reactor Power Reduction Instructions

Sheet 1 of 2 (*Control Rod Insertions ONLY*) CRS ______

Control Rod	Correct Rod Selected and Verified*	Control Rod Position	Licensed Operator	Second Licensed Operator	Projected Power Reduction (% CTP)
18-19		08 To 00			
18-35		08 To 00			
34-35		08 To 00			
34-19		08 To 00			2% ▼
26-11		36 To 00			
10-27		36 To 00			
26-43		36 To 00			
42-27		36 To 00			18% ▼
26-27		48 To 00			2% ▼
18-11		48 To 00			
18-43		48 To 00			
34-43		48 To 00			
34-11		48 To 00			13% ▼
10-19		48 To 00			
10-35		48 To 00			
42-35		48 To 00			
42-19		48 To 00			8% ▼

*Concurrent Verification of rod selection required prior to rod movement.

EVENT 6: HEATER DRAIN DEAERATOR CONTROLLER FAILURE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 8** to fail Heater Drain Controller.
	If directed to place controller in Manual or to swap master controllers, Delete CF039F.

Simulator Operator Role Play	
	If contacted as TBAO to investigate, report LC-91 is in master and is sending a full open signal.
	If asked by I&C to investigate controller failure, acknowledge the request.
	When HDD level is stabilized and if directed to place controller in Manual or to swap master controllers, have Sim Operator delete CF039F and report controller in manual maintaining level

Evaluator Notes	
Plant Response:	Heater Drain Tank Lowers Low level alarm at 32” Both Heater Drain pumps trip at 24” Condensate Booster Pump C auto start if power is not sufficiently reduced
Objectives:	Enter 0AOP-23.0 May reduce power Control level using HD-57
Success Path:	Manually control level in Heater Drain Tank using the HD-57
Event Termination:	Go to Event 7 at the direction of the Lead Evaluator.

EVENT 6: HEATER DRAIN DEAERATOR CONTROLLER FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct entry in 0AOP-23.0, Condensate/Feedwater System Failure	
	SRO	May direct a power reduction to stabilize Condensate/Feedwater	
	SRO	Directs manual control with HD-57 to stabilize HD Tank level	
	SRO	Directs I&C to investigate	
	SRO	May contact Shift Manager	
	SRO	May conduct a brief (See Enclosure 1, page 43 for format of the brief.	
	ATC	Monitor plant	
	ATC	Announce entry into 0AOP-23.0, Condensate/Feedwater System Failure	
	ATC	May reduce Reactor power IAW 0ENP-24.5 as directed by CRS.	

EVENT 6: HEATER DRAIN DEAERATOR CONTROLLER FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Acknowledge and report alarm: UA-4 2-10 HD DEAERATOR LEVEL HIGH-LOW. Alarm at 30 inches and lowering. Pump trip at 24 inches and lowering.	
	BOP	Diagnose HD Pump discharge valves full open	
	BOP	Enter and announce 0AOP-23.0, Condensate/Feedwater System Failure	
	BOP	Trips one of the operating Heater Drain pump	
	BOP	Maintains heater drain deaerator level less than 60 inches indicated on HEATER DRAIN DEAERATOR LEVEL, HD-LI-97 If level reaches 60 inches UA-4, 3-10 may alarm and the HDD Moisture removal valves will open. After level is stabilized the APP has direction for re-closing the moisture removal valves.	Move to the next event when level is being controlled with the HD-V57.
	BOP	May dispatch TBAO to check HD Pump Air-Operated Discharge Level Control Valves, HD-LV-91-1, 2, & 3.	
	BOP	May direct TBAO to place HDD level control in Manual IAW 2OP-35 Section 6.3.8. or swap controller IAW 2OP-35, Section 6.3.8	
	BOP	Monitors main condenser vacuum and condensate parameters	
	BOP	May have to secure a CBP if one auto started during the evolution.	

EVENT 7/8/9: FUEL FAILURE / SCRAM / SPC F024B VALVE FAILURE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 9** to activate the Fuel Failure
	Monitor MSL Rad and as power is reduced, **adjust** Fuel Failure and Gross Fuel Failure to ensure MSL Hi-HI is received. (If NB005F reaches 100, Manually **initiate Trigger 10**).
	When requested as Unit 1 to stop Unit 1 Cable Spread Vent Fans, **Initiate Trigger 11**.
	When requested as Unit 1 to stop CB Mech Equipment Room Vent Fan, **Initiate Trigger 12.**
	When requested delete malfunction to reset thermal overload on torus cooling water valve.
	When directed by the Lead Evaluator, place the simulator in FREEZE
	Do not reset the simulator until receipt of concurrence to do so from the lead examiner

Simulator Operator Role Play	
	If PEP-3.4.7 is requested, acknowledge the request.
	If asked as E&RC, acknowledge the request to obtain actual/projected off site dose rates.
	If asked as E&RC, acknowledge the request to sample off gas and reactor coolant for potential fuel damage.
	If asked as E&RC, acknowledge the request to obtain noble gas dose rates.
	If requested to investigate breaker for the Torus cooling valve (F024B) report that it has tripped on thermals. If asked to reset have Sim Op delete malfunction and report breaker is reset.

Evaluator Notes	
Plant Response:	Fuel failure will occur. When the Main Stack Rad indication is rising then a reactor scram and Group I isolation is inserted. Reactor Scram Pushbuttons are failed, other methods are available (i.e. ARI). Torus cooling valve will thermal out, can be reset.
Objectives:	SRO - Directs actions for a Reactor Scram, and RRCP. ROs - Insert a scram IAW RRCP and close Group I valves.
Success Path:	Rods inserted with ARI or mode switch.
Scenario Termination:	When rods are inserted and SPC is being placed in service the scenario may be terminated.
Remind students not to erase any charts and not to discuss the scenario until told to do so by the evaluator/instructor.	

EVENT 7/8/9: FUEL FAILURE / SCRAM / SPC FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct entry in to AOP-5.4, Radiological Releases.	
	SRO	Directs reactor power to be reduced to clear UA-03 5-2	
	SRO	Asks Unit One to calculate site boundary dose per PEP-3.4.7. Asks HPs for field surveys	
	SRO	Ensure AOG in service	
	SRO	Notify E&RC to sample off- gas and reactor coolant for potential fuel damage.	
	SRO	When UA-03 4-2 is received: Verifies Main Stack Rad levels rising. Ensures AOG bypass valve is closed. May direct a power reduction ***When UA-23 3-6 is received, Directs the insertion of a manual scram***. (May direct sooner as a conservative decision) ***Directs the closure of Group I valves*** (MSIVs, Steam Line Drains (B21-F016 and F019) and Recirc Sample Valves (B32-F019 and F020)	***Critical Task #1*** ***Critical Task #2***
	SRO	When Torus temperature is greater than 95° F, directs placing Torus Cooling in service. Gives permission for a reset of the torus cooling water valve thermals.	

EVENT 7/8/9: FUEL FAILURE / SCRAM / SPC FAILURE			
	ATC	Inserts a manual reactor scram and recognizes failure of manual scram channel B. See Enclosure 2 (page 44) for Scram Hard Card	
	ATC	***Initiates ARI per the Scram Hard Card.***	***Critical Task #1***
	ATC	Perform immediate actions for Reactor scram After steam flow is less than 3 x 10^6 lb/hr, place the reactor mode switch to shutdown When APRM downscale trip, then Trip the main turbine Ensure the master reactor level controller setpoint is +170 inches When RPV water level is above 160 inches and rising then trips one reactor feed pump.	
	ATC	***Close Group I Isolation valves.*** Closes MSIVs Closes Main Steam Line Drain valves B21-F016 and F019. Closes Reactor Recirc Sample Valves B32-F019 and F020	***Critical Task #2***
	ATC	May operate SRVs as directed to stabilize pressure and control pressure as directed by the CRS.	

EVENT 7/8/9: FUEL FAILURE / SCRAM / SPC FAILURE

	BOP	Report Annunciators when received UA-03 (5-7) Area Rad Turbine Bldg High UA-03 (5-2) Process Off-Gas Rad High UA-23 (2-6) Main Steam Line Rad Hi UA-23 (3-6) Main Steam Line Rad Hi-Hi/Inop	
		Announces and enters 0AOP-5.4, Radiological Releases	
		May operate SRVs as directed to stabilize pressure and control pressure as directed by the CRS.	
		When Torus temperature is greater than 95° F, places Torus Cooling in service. (Enclosure 3, page 45)	
		Recognize failure of torus cooling water valve and dispatch an AO to investigate breaker.	
		Maintains level as directed by the CRS	

ENCLOSURE 1

Page 1 of 1

CONDUCT OF OPERATIONS	AD-OP-ALL-1000
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ATTACHMENT 8
Page 1 of 1

<< Crew Brief Template >>

Begin Brief	☐ Announce "Crew Brief" ☐ All crew members acknowledge announcement

Recap	**(As Required)**
	☐ Update the crew as needed:
	☐ Describe what happened and major actions taken
	☐ Procedures in-progress
	☐ Notifications:
	    ☐ Maintenance
	    ☐ Engineering
	    ☐ Others (Dispatcher, Station Management, etc.)
	☐ Future Direction and priorities
	☐ Discuss any contingency plans

Input	**(As Required)**
	☐ Solicit questions/concerns from each crew member:
	    ☐ ROs
	    ☐ CRS
	    ☐ STA
	☐ Are there any alarms unexpected for the plant conditions?
	☐ What is the status of Critical Parameters?

EAL	**(As Required)** ☐ Provide EAL and potential escalation criteria
Finish Brief	☐ Restore normal alarm announcement? (Yes/No) ☐ Announce "End of Brief"

ENCLOSURE 2

5.4 SCRAM Card

Enter applicable leg ☐

Scram	**ATWS**
All Control Rods FULL-IN ☐	Indications of Hydraulic/Electrical ATWS ☐
RPV Water Level ☐ __________ inches	**Ensure** ARI initiated ☐
RPV Pressure ☐ __________ psig	Reactor Power ☐ __________ %
Communicate scram report to CRS ☐	**Communicate** ATWS report to CRS ☐
Place SULCV in service ☐	**IF** enabled, **THEN initiate** a recirc pump manual runback ☐
Insert Nuclear Instrumentation ☐	**IF** reactor power above **2%** **OR** **CANNOT** be determined, **THEN trip** both recirc pumps ☐
Ensure Turbine Oil System Operating ☐	**Report** reactor power to CRS ☐
Ensure Reactor Recirculation Pump speed at 34% ☐	**Exit** scram card and **perform** EOP-01-LEP-02 ☐
Ensure Heater Drain Pumps tripped ☐	
Exit scram card ☐	

ENCLOSURE 3 Page 1 of 2

ATTACHMENT 8A
Page 1 of 1
Emergency Suppression Pool Cooling Using Loop A (2OP-17)

NOTE:	This attachment is **NOT** to be used for normal system operations.

START RHR SW A LOOP (CONV)

- **OPEN** *SW-V101* ☐
- **CLOSE** *SW-V143* ☐
- **START** CSW PUMPS AS NEEDED ☐
- **IF** LOCA SIGNAL IS PRESENT **THEN** PLACE RHR SW BOOSTER PUMPS A & C LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- **START** RHR SW PMP ☐
- **ADJUST** *E11-PDV-F068A* ☐
- **ESTABLISH** CLG WTR TO VITAL HDR ☐
- **START** ADDITIONAL RHR SW PUMP **AND ADJUST** FLOW AS NEEDED ☐

START RHR SW A LOOP (NUC)

- **OPEN** *SW-V105* ☐
- **OPEN** *SW-V102* ☐
- **CLOSE** *SW-V143* ☐
- **START** PUMPS ON NSW HDR AS NEEDED ☐
- **IF** LOCA SIGNAL IS PRESENT **THEN PLACE** RHR SW BOOSTER PUMPS A & C LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- **START** RHR SW PMP ☐
- **ADJUST** *E11-PDV-F068A* ☐
- **ESTABLISH** CLG WTR TO VITAL HDR ☐
- **START** ADDITIONAL RHR SW PUMP **AND ADJUST** FLOW AS NEEDED ☐

START RHR LOOP A

- **IF** LOCA SIGNAL IS PRESENT, **THEN VERIFY** COOLING LOGIC IS MADE UP ☐
- **IF** *E11-F015A* IS OPEN, **THEN CLOSE** *E11-F017A* ☐
- **START** LOOP A RHR PMP ☐
- **OPEN** *E11-F028A* ☐
- **THROTTLE** *E11-F024A* ☐
- **THROTTLE** *E11-F048A* ☐
- **START** ADDITIONAL LOOP A RHR PMP **AND ADJUST** FLOW AS NEEDED ☐

ENCLOSURE 3 Page 2 of 2

ATTACHMENT 8B
Page 1 of 1
Emergency Suppression Pool Cooling Using Loop B (2OP-17)

NOTE:	This attachment is **NOT** to be used for normal system operations.

START RHR SW B LOOP (NUC)

- **OPEN** *SW-V105* ☐
- **CLOSE** *SW-V143* ☐
- **START** PMPS ON NSW HDR AS NEEDED ☐
- **IF** LOCA SIGNAL IS PRESENT **THEN** PLACE RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- **START** RHR SW PMP ☐
- **ADJUST** *E11-PDV-F068B* ☐
- **ESTABLISH** CLG WTR TO VITAL HDR ☐
- **START** ADDITIONAL RHR SW PUMP **AND ADJUST** FLOW AS NEEDED ☐

START RHR SW B LOOP (CONV)

- **OPEN** *SW-V101* ☐
- **OPEN** *SW-V102* ☐
- **CLOSE** *SW-V143* ☐
- **START** CSW PUMPS AS NEEDED ☐
- **IF** LOCA SIGNAL IS PRESENT **THEN PLACE** RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- **START** RHR SW PMP ☐
- **ADJUST** *E11-PDV-F068B* ☐
- **ESTABLISH** CLG WTR TO VITAL HDR ☐
- **START** ADDITIONAL RHR SW PUMP **AND ADJUST** FLOW AS NEEDED ☐

START RHR LOOP B

- **IF** LOCA SIGNAL IS PRESENT, **THEN VERIFY** COOLING LOGIC IS MADE UP ☐
- **IF** *E11-F015B* IS OPEN, **THEN CLOSE** *E11-F017B* ☐
- **START** LOOP B RHR PMP ☐
- **OPEN** *E11-F028B* ☐
- **THROTTLE** *E11-F024B* ☐
- **THROTTLE** *E11-F048B* ☐
- **START** ADDITIONAL LOOP B RHR PMP **AND ADJUST** FLOW AS NEEDED ☐

ATTACHMENT 1 - Scenario Quantitative Attribute Assessment

Category	NUREG 1021 Rev. 2 Supp. 1 Req.	Scenario Content
Total Malfunctions	5-8	8
Malfunctions after EOP Entry	1-2	2
Abnormal Events	2-4	4
Major Transients	1-2	1
EOPs Used	1-2	2
EOP Contingency	0-2	1
Run Time	60-90 min	90
Crew Critical Tasks	2-3	2
Tech Specs	2	2
Instrument / Component Failures before Major	2 – OATC 2 - BOP	4
Instrument / Component Failures after Major	2	2
Normal Operations	1	1
Reactivity manipulation	1	1

ATTACHMENT 2 – Shift Turnover

Brunswick Unit 2 Plant Status					
Station Duty Manager:	E. Neal			Workweek Manager:	B. Craig
Mode:	1	Rx Power:	100%	Gross*/Net MWe*:	977 / 951
Plant Risk: Current EOOS Risk Assessment is:				Green	
SFP Time to 200 Deg F:	49.7 hrs			Days Online:	80 days
Turnover:	2OP-32, Section 6.3.6 has been completed up to step 6.3.6.11.				
Protected Equipment:	2A FPC Pump/Hx, 2A RCC Pump, and 2C Demin Transfer Pump for Fuel Pool Decay Heat Removal and inventory makeup. 2A/B NSW Pumps due to 1A NSW pump maintenance				
Comments:	1A NSW Pump is under clearance for planned maintenance. 2C TCC Pump is in service on Unit One. APRM 2 has failed downscale and is bypassed. The BOP operator will continue procedure to swap Condensate Booster Pumps (place CBP 2C in service and remove CBP 2A from service for maintenance).				

6.3.6 Transferring to Standby Condensate Booster Pump

1. **Confirm** at least one condensate booster pump in operation................ RO

NOTE

- Lowering CFD and CDD system differential pressure will raise Condensate Booster Pump suction pressure. This may also minimize the potential of an override or bypass condition of the CDD and CFD's. .... ☒
- If HWC is in service, from operating experience, UA-23, 2-6 Main Steam Line Rad Hi may alarm when starting a condensate booster pump. [8.7.17]. .... ☒

2. **Notify** Chemistry performing this section.. RO

   J. Johnson
   Person Notified

3. **Designate** the oncoming and off going condensate booster pumps below: .... RO

   Oncoming Condensate Booster Pump C

   Off going Condensate Booster Pump A

4. **Confirm** Condensate Booster Pump suction pressure is greater than 85 psig on either COD-PI-30-1 or PPC point U2CODL071........... RO

5. **IF** Condensate Booster Pump suction pressure is less than or equal to 85 psig,
   THEN perform the following:... N/A

   a. **Direct** Radwaste operator to perform the following to lower CFD and CDD system differential pressure:

      (1) **IF** available,
      THEN place an additional CFD in service....................... N/A

      (2) **IF** available,
      THEN place an additional CDD in service .... N/A

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

b. **IF** after performing Step 5.a, Condensate Booster Pump suction pressure is still less than or equal to 85 psig, **THEN perform** the following: .... N/A

(1) **Record** the following in the Operator's Log:

- Values from both suction pressure indicators. .... N/A
- Reactor power level. .... N/A

(2) **Perform** Section 6.3.22, Starting a Third Condensate Pump for Operation During Abnormal Plant Conditions to place the third Condensate Pump in service. .... N/A

(3) **Record** third Condensate Pump start to support Condensate Booster Pump start in Operator log. .... N/A

6. **Direct** Radwaste operator to monitor for proper operation of hotwell level control. .... RO

7. **Ensure** proper oil level in oncoming condensate pump oil reservoir. .... RO

CAUTION

Proper venting of the condensate booster pump mechanical seal chamber, prior to starting the pump is critical to seal performance and longevity. .... ☒

8. **IF** 2A Condensate Booster Pump is to be started, **THEN perform** the following: .... N/A

a. **Open** COD-V5014 (Condensate Booster Pump A Inboard Mechanical Seal Vent Valve). .... N/A

b. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5014 (Condensate Booster Pump A Inboard Mechanical Seal Vent Valve). .... N/A

c. **Open** COD-V5015 (Condensate Booster Pump A Outboard Mechanical Seal Vent Valve). .... N/A

d. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5015 (Condensate Booster Pump A Outboard Mechanical Seal Vent Valve). .... N/A

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

9. IF 2B Condensate Booster Pump is to be started, THEN **perform** the following: .... N/A

   a. **Open** COD-V5016 (Condensate Booster Pump B Inboard Mechanical Seal Vent Valve). .... N/A

   b. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5016 (Condensate Booster Pump B Inboard Mechanical Seal Vent Valve). .... N/A

   c. **Open** COD-V5017 (Condensate Booster Pump B Outboard Mechanical Seal Vent Valve). .... N/A

   d. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5017 (Condensate Booster Pump B Outboard Mechanical Seal Vent Valve). .... N/A

10. IF 2C Condensate Booster Pump is to be started, THEN **perform** the following: .... RO

   a. **Open** COD-V5018 (Condensate Booster Pump C Inboard Mechanical Seal Vent Valve). .... AO

   b. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5018 (Condensate Booster Pump C Inboard Mechanical Seal Vent Valve). .... AO

   c. **Open** COD-V5019 (Condensate Booster Pump C Outboard Mechanical Seal Vent Valve). .... AO

   d. **WHEN** a continuous solid stream of water has been observed for at least 2 minutes, **THEN close** COD-V5019 (Condensate Booster Pump C Outboard Mechanical Seal Vent Valve). .... AO

11. **Note** status of the following alarms:

   - UA-13, 1-1, Gen-Xfmr Primary L/O Unit Trip .... ______
   - UA-13, 1-2, Generator Diff L/O Unit Trip .... ______
   - UA-13, 1-3, Gen-Xfmr Backup L/O Unit Trip .... ______

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

12. **IF** all the alarms in Step 11 are CLEAR, **THEN perform** the following:[8.7.5] .... _________

    a. **Place** the following switches in ENABLED for the off going condensate booster pump:

    - Unit Trip Load Shed Selector Switch .... _________
    - LOCA Load Shed Selector Switch.................................. _________

    b. **Ensure** the oncoming condensate booster pump mode selector switch in MAN. .... _________

    c. **Confirm** oncoming condensate pump discharge valve closes, if OPEN:

    - COD-V4 (Condensate Booster Pump A Disch Vlv) .... _________
    - COD-V5 (Condensate Booster Pump B Disch Vlv) .... _________
    - COD-V6 (Condensate Booster Pump C Disch Vlv) .... _________

NOTE

The start/stop control switch for a Condensate Booster Pump must be held in START until the recirculation valve is full OPEN. The recirculation valve does **NOT** stroke OPEN until all other condensate booster pump permissives are met. While opening the recirc valve, condensate booster pump suction pressure will lower as additional flow up to 2000 gpm is routed to the condenser. .... ☐

    d. **Start** the oncoming condensate booster pump. .... _________

    e. **Confirm** oncoming condensate booster pump discharge valve opens:

    - COD-V4 (Condensate Booster Pump A Disch Vlv) .... _________
    - COD-V5 (Condensate Booster Pump B Disch Vlv) .... _________
    - COD-V6 (Condensate Booster Pump C Disch Vlv) .... _________

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

f. **WHEN** condensate booster pump discharge pressure stabilizes,
THEN perform the following: .... ________

(1) **Stop** off going condensate booster pump. .... ________

(2) **IF** B21-F032A (Feedwater Isol Vlv) **AND** B21-F032B (Feedwater Isol Vlv) are OPEN,
THEN place the off going condensate booster pump mode switch in AUTO. .... ________

g. **Place** the following switches in DISABLED for oncoming condensate booster pump:

- Unit Trip Load Shed Selector Switch .... ________
- LOCA Load Shed Selector Switch .... ________

h. **IF** HWC is in service,
THEN perform the following: .... ________

(1) **Place** HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, for oncoming condensate booster pump, in AUTO:

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve) .... ________

(2) **Confirm** HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, for oncoming condensate booster pump, is OPEN:

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve) .... ________

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

(3) **Place** the HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, for off going condensate booster pump in CLOSED.

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve) .... ________
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve) .... ________

i. **Go to** Step 20. .... ________

13. **Place** the following switches in DISABLED for the oncoming Condensate Booster Pump:
    - Unit Trip Load Shed Selector Switch.. ________
    - LOCA Load Shed Selector Switch .... ________
14. **Place** the oncoming condensate booster pump mode selector switch in MAN. .... ________
15. **Confirm** oncoming condensate booster pump discharge valve closes.
    - COD-V4 (Condensate Booster Pump A Disch Vlv) .... ________
    - COD-V5 (Condensate Booster Pump B Disch Vlv) .... ________
    - COD-V6 (Condensate Booster Pump C Disch Vlv) .... ________

NOTE

The start/stop control switch for a condensate booster pump must be held in START <u>until</u> the recirculation valve is full OPEN. The recirculation valve does **<u>NOT</u>** stroke OPEN <u>until</u> all other condensate booster pump permissives are met. While opening the recirc valve, condensate booster pump suction pressure will lower as additional flow up to 2000 gpm is routed to the condenser. .... ☐

16. **Start** oncoming condensate booster pump .... ________

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

17. **Confirm** oncoming condensate booster pump discharge valve opens:

    - COD-V4 (Condensate Booster Pump A Disch Vlv) .... ________
    - COD-V5 (Condensate Booster Pump B Disch Vlv) .... ________
    - COD-V6 (Condensate Booster Pump C Disch Vlv) .... ________

18. **<u>WHEN</u>** condensate booster pump discharge pressure stabilizes, **<u>THEN</u> perform** the following: .... ________

    a. **Stop** off going condensate booster pump. .... ________

    b. **<u>IF</u>** B21-F032A (Feedwater Isol Vlv) **<u>AND</u>** B21-F032B (Feedwater Isol Vlv) are OPEN, **<u>THEN</u> place** the off going condensate booster pump mode switch in AUTO. .... ________

    c. **Place** the following switches in ENABLED for the off going condensate booster pump:

       - Unit Trip Load Shed Selector Switch .... ________
       - LOCA Load Shed Selector Switch .... ________

19. **<u>IF</u>** HWC is in service, **<u>THEN</u> perform** the following: .... ________

    a. **Place** oncoming condensate booster pump, HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, in AUTO:

       - HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve) .... ________
       - HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve) .... ________
       - HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve) .... ________

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

b. **Confirm** oncoming condensate booster pump, HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, OPEN:

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve).. _________
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve).. _________
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve).. _________

c. **Place** off going condensate booster pump, HWC condensate H2 injection isolation solenoid valve, at HWC local panel H2J, in CLOSED:

- HWCH-SV-5717 (Condensate Booster Pump A H_2 Injection Isolation Valve).. _________
- HWCH-SV-5718 (Condensate Booster Pump B H_2 Injection Isolation Valve).. _________
- HWCH-SV-5719 (Condensate Booster Pump C H_2 Injection Isolation Valve).. _________

20. **IF** three condensate pumps are in service per Step 5.b(2) , **THEN perform** Section 6.3.23, Securing from Three Condensate Pump Operation.. _________

21. **Direct** Radwaste Operator to remove the additional CFD or CDD placed in service in Step 5.a. .... _________

22. **Direct** Radwaste Operator to monitor CDD effluent conductivity for each demineralizer in service. .... _________

23. **IF** any condensate pump Unit Trip Load Shed Selector Switch **AND** LOCA Load Shed Selector Switch was manipulated, **THEN complete** Attachment 9, Condensate Pump Unit Trip Load Shed/LOCA Load Shed Selector Switch Alignment Documentation...... _________

24. **Complete** Attachment 10, Condensate Booster Pump Unit Trip Load Shed/LOCA Load Shed Selector Switch Alignment Documentation.. _________

6.3.6 Transferring to Standby Condensate Booster Pump (continued)

Date/Time Completed ____________

Performed By (Print) Initials

____________________ ________

____________________ ________

____________________ ________

____________________ ________

Reviewed By ____________________________
Unit CRS/SRO

Condensate Booster Pump Unit Trip Load Shed/LOCA Load Shed Selector Switch Alignment Documentation

NOTE

- When performing this attachment, pump selector switches, for pump(s) which have **NOT** been realigned may be marked NA. ☐
- The condensate booster pump selected for standby operation (Mode Selector Switch in AUTO) is required to have both the LOCA Load Shed Selector Switch and Unit Trip Load Shed Selector Switch in ENABLED. ☐

1. Circle the required selector switch position.

Number	Description	Position/ Indication [Note 1]	Checked	Verified
Turbine Building - 4160V Switchgear Bus 2D - Elev 20 ft				
AD9	2B Condensate Booster Pump Unit Trip Load Shed Selector Switch	ENABLED/ DISABLED		
AD9	2B Condensate Booster Pump LOCA Load Shed Selector Switch	ENABLED/ DISABLED		
Turbine Building - 4160V Switchgear Bus 2C - Elev 20 ft				
AC1	2A Condensate Booster Pump Unit Trip Load Shed Selector Switch	ENABLED/ DISABLED		
AC1	2A Condensate Booster Pump LOCA Load Shed Selector Switch	ENABLED/ DISABLED		
AC2	2C Condensate Booster Pump Unit Trip Load Shed Selector Switch	ENABLED/ DISABLED		
AC2	2C Condensate Booster Pump LOCA Load Shed Selector Switch	ENABLED/ DISABLED		

BRUNSWICK TRAINING SECTION

OPERATIONS TRAINING

INITIAL LICENSED OPERATOR

SIMULATOR EVALUATION GUIDE

2016 NRC SCENARIO 2

LOWER POWER, REMOVE 230KV LINE FROM SERVICE, ROD DRIFT, ADHR PP TRIP, RECIRC LOOP FLOW FAILURE, HDD PP TRIP, ATWS, SLC MODE SWITCH FAILURE, ARI FAIL TO RESET

REVISION 0

Developer: Bob Bolin	**Date:** 07/07/2016
Technical Review: Dan Hulgin	**Date:** 9/12/2016
Validators: Kyle Cooper Grant Newton Hunter Morris	**Date:** 09/06/16
Facility Representative: Greg Oliver	**Date:** 09/22/2016

REVISION SUMMARY	
0	Scenario developed for 2016 NRC Exam.

TABLE OF CONTENTS

1.0 SCENARIO OUTLINE

Event	Malf. No.	Type*	Event Description
1		R - ATC	Lower power to 850 MWe to remove 230 kV Line 30
2		N - BOP	Remove 230 kV Line 30 from service
3	RD001M (26-11)	C - ATC C - CRS	Rod Drift (TS)
4	K4526A	C - BOP C - CRS	ADHR Secondary pump trip (AOP)
5	NI063F	C - ATC C - CRS	Recirc Loop B Flow transmitter Failure (TS)
6	CF089F	C - BOP C - CRS	Heater Drain Deaerator Pump Trip (AOP)
7	CA008F	M	Small steam leak in DW results in an ATWS requiring terminate and prevent actions (RSP)(ATWS)
8	K2119A	C	SLC Mode Switch Failure
9	K2624A	C	Alternate Rod Insertion reset failure
*(N)ormal, (R)eactivity, (C)omponent or Instrument, (M)ajor			

2.0 SCENARIO DESCRIPTION SUMMARY

Event	Description
1	After taking the watch the CRS will direct power reduced to 850 MWe.
2	The BOP will isolate 230 kV Line 30.
3	Control Rod 26-11 will start to drift in. The crew will enter 0AOP-02.0 and take action IAW 2APP-A-05 (3-2). When the high temperature alarm is received Engineering will report that scram times cannot be assured based on past history of the control rod. Determine TS 3.1.3 condition C1 to insert the control rod in 3 hours and C2 to disarm the control rod within 4 hours.
4	After Tech Specs are addressed the Alternate Decay Heat Removal (ADHR) Seconday pump will trip. AOP-38.0 will be entered
5	The Recirc Loop B flow transmitter to APRM Channel 4 will fail downscale resulting in a rod block and a trip input to each voter. The crew will respond per APPs and bypass APRM 4. The APRM will be declared Inoperable per TS 3.3.1.1, Condition A and placed in trip within 12 hours. APRM TS Actions to be taken requires the APRM mode selector switch to be place in INOP IAW 0OI-18
6	A motor overload will occur on Heater Drain Pump 2A. The crew will reference APP UA-06 1-7, Bus 2D 4KV Motor Ovld and determine which pump has the overload condition. The crew should start HDP 2C and secure HDP 2A. The crew may reference AOP-23.0.
7	A small steam leak in the DW results in rising Drywell pressure requiring a reactor scram. An ATWS will occur, conditions will require terminate and prevent actions to be performed.
8	When SLC is initiated, the mode switch will fail and the pumps will not start. LEP-03 will be executed to inject the boron into the core.
9	Alternate Rod Insertion (ARI) will not reset, the crew will perform LEP-02 to drive control rods into the core. When level is stabilized after terminating and preventing ARI will be repaired to allow the rods to be manually scrammed.

3.0 CREW CRITICAL TASKS

Critical Task #1
Insert control rods IAW LEP-02
Critical Task #2
Direct Alternate Boron Injection IAW LEP-03
Critical Task #3
Terminate and prevent injection from HPCI/Condensate and Feedwater/LP ECCS

4.0 TERMINATION CRITERIA

When all rods are inserted and level is being controlled above TAF the scenario may be terminated.

5.0 IMPLEMENTING REFERENCES

NOTE: Refer to the most current revision of each Implementing Reference.

Number	Title
A-05, 3-2	ROD DRIFT
0AOP-02.0	CONTROL ROD MALFUNCTION/MISPOSITION
UA-18, 6-1	BUS E4 4KV MOTOR OVLD.
UA-01, 2-3	ADHR PRIMARY LOOP TROUBLE
UA-01, 3-3	ADHR SECONDARY LOOP TROUBLE
0AOP-38.0	LOSS OF FUEL POOL COOLING
A-06, 2-8	APRM UPSCALE
A-06, 3-8	APRM UPSCALE TRIP/INOP
A-06, 5-7	FLOW REF OFF NORMAL
A-05, 2-2	ROD OUT BLOCK
A-05, 4-8	OPRM TRIP ENABLED
UA-5, 3-5	SBGT SYS B FAILURE
UA-5, 4-6	SBGT SYS A FAILURE

6.0 SETUP INSTRUCTIONS

1. **PERFORM** TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 5, Checklist for Simulator Exam Security.
2. **RESET** the Simulator to IC-11.
3. **ENSURE** the RWM is set up as required for the selected IC.
4. **ENSURE** appropriate keys have blanks in switches.
5. **RESET** alarms on SJAE, MSL, and RWM NUMACs.
6. **ENSURE** no rods are bypassed in the RWM.
7. **PLACE** all SPDS displays to the Critical Plant Variable display (#100).
8. **ENSURE** hard cards and flow charts are cleaned up
9. **TAKE** the SIMULATOR OUT OF FREEZE
10. **LOAD** Scenario File.
11. **ALIGN** the plant as follows:

Manipulation
Ensure 2C TCC pump is in service on Unit One.
Bypass APRM 2
RCC Pump D in service for ADHR
RCC Pump A in service for RBCCW

12. **IF desired**, take a **SNAPSHOT** and save into an available IC for later use.
13. **PLACE** a clearance on the following equipment.

Component	Position
APRM 2	Blue tag

14. **INSTALL** Protected Equipment signage and **UPDATE** RTGB placard as follows:

Protected Equipment
1. 2A and 2B NSW pumps
2. 2A FPC Pump/Hx, 2A RCC Pump, and 2C Demin Transfer Pump.

15. **VERIFY** 0ENP 24.5 Form 2 (Immediate Power Reduction Form) for IC-11 is in place.

16. **ENSURE** each Implementing References listed in Section 7 is intact and free of marks.

17. **ENSURE** all materials in the table below are in place and marked-up to the step identified.

Required Materials
Marked up of 2OP-50, Section 6.2.6

18. **ADVANCE** the recorders to prevent examinees from seeing relevant scenario details.

19. **PROVIDE** Shift Briefing sheet for the CRS.

20. **VERIFY** all actions contained in TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 4, Simulator Training Instructor Checklist, are complete.

7.0 INTERVENTIONS

TRIGGERS

Trig	Type	ID
1	Malfunction	RD001M - [CONTROL ROD SLOW INSERTION DRIFT]
2	Annunciator	ZA512 - [CRD HYD TEMP HIGH]
3	Trigger Command	MFD:RD001M,26-11
4	Remote Function	RD_RDELDIS - [ELECTRICAL DISARM OF ROD]
5	DI Override	K4526A - [RBCCW PMP D AUTO]
5	DI Override	K4526A - [RBCCW PMP D AUTO]
5	DI Override	K4526A - [RBCCW PMP D AUTO]
5	DO Override	Q4526AMW - [RBCCW PMP D ADHR MODE]
5	DO Override	Q4526LG4 - [RBCCW PMP D OFF G]
5	Malfunction	RP011F - [ATWS 4]
6	Remote Function	CC_MODE - [RBCCW/ADHR VALVE LINEUPS]
6	Remote Function	CC_MSS - [RBCCW/ADHR PUMP MODE SELECTOR SWITCH]
7	Remote Function	CC_PDV - [RBCCW PUMP DISCHARGE VALVE]
8	Malfunction	NI063F - [RECIRC LOOP B XMITTER FAILURE]
9	Malfunction	CF089F - [HEATER DRAIN PUMP MOTOR WINDING FAULT]
10	Malfunction	NB006F - [MSL BRK BEFORE FLOW RESTRICTOR]
11	Remote Function	EP_IAEOPJP1 - [BYPASS LL-3 GROUP I ISOL (SEP-10)]

Trig #	Trigger Text
3	KM118EDN - [SCRAM TEST SWITCH 26-11] true deletes RD001M

ANNUNCIATORS

Window	Description	Tagname	Override Type	OVal	AVal	Actime	Dactime	Trig
1-2	CRD HYD TEMP HIGH	ZA512	ON	ON	OFF			2

MALFUNCTIONS

Malf ID	Mult ID	Description	Current Value	Target Value	Rmp time	Actime	Dactime	Trig
RD001M	26-11	CONTROL ROD SLOW INSERTION DRIFT	False	True				1
NI063F	APRM 4	RECIRC LOOP B XMITTER FAILURE	0.00	125.00				8
NB006F	A	MSL BRK BEFORE FLOW RESTRICTOR	0.00	1.0e-1	0:03:00			10
CF089F	A	HEATER DRAIN PUMP MOTOR WINDING FAULT	False	True				9
RP011F		ATWS 4	False	True				5
RP005F		AUTO SCRAM DEFEAT	True	True				
NI032F	APRM 2	APRM FAILS LO	True	True				

REMOTES

Remf Id	Mult Id	Description	Current Value	Target Value	Rmp time	Actime	Trig
RD_RDELDIS	26-11	ELECTRICAL DISARM OF ROD	ARM	DISARM			4
CC_MODE	PUMP-A	RBCCW/ADHR VALVE LINEUPS	RBCCW	ADHR			6
CC_MSS	A	RBCCW/ADHR PUMP MODE SELECTOR SWITCH	RBCCW	ADHR			6
CC_PDV	A_V38_V5114	RBCCW PUMP DISCHARGE VALVE	1.0000	1.0e-01			7
CC_IACW4518		2C TBCCW PUMP UNIT ALIGNMENT	1	1			
EP_IAEOPJP1		BYPASS LL-3 GROUP I ISOL (SEP-10)	OFF	ON			11

PANEL OVERRIDES

Tag ID	Description	Position / Target	Actual Value	Override Value	Rmp time	Actime	Dactime	Trig
K4526A	RBCCW PUMP D OFF	OFF/RESEST	OFF	ON				5
K4526A	RBCCW PMP D AUTO	AUTO	OFF	OFF				5
K4526A	RBCCW PMP D ON	ON	ON	OFF				5
Q4526LG4	RBCCW PMP D OFF G	ON/OFF	OFF	OFF				5
Q4526AMW	RBCCW PMP D ADHR MODE	ON/OFF	ON	OFF				5
K2119A	S/B LIQ PUMP A & B	PUMP_A	OFF	OFF				
K2119A	S/B LIQ PUMP A & B	PUMP_A&B	OFF	OFF				
K2119A	S/B LIQ PUMP A & B	PUMP_B	OFF	OFF				
K2624A	CS-5562 ARI	RESET	OFF	OFF				
K2625A	CS-5560 ARI	INOP	OFF	OFF				

8.0 OPERATOR RESPONSE AND INSTRUCTIONAL STRATEGIES

EVENT 1: SWAP CBPs	
Simulator Operator Actions	
	Ensure Monitored Parameters is open and Scenario Based Testing Variables are loaded.
	Initiate Trigger 1 to ENABLE 2A CBP Unit Trip and LOCA Load Shed Selector Switches.
	Initiate Trigger 2 to DISABLE 2C CBP Unit Trip and LOCA Load Shed Selector Switches
	Initiate Trigger 13 to place HWCH-SV-5719 to open.
	Initiate Trigger 14 to place HWCH-SV-5717 to close.

Simulator Operator Role Play	
	Acknowledge request to enable 2A CBP Unit Trip and LOCA Load Shed selector switches, after Sim Operator activates the trigger report that the action is complete.
	Acknowledge request to disable 2C CBP Unit Trip and LOCA Load Shed selector switches, after Sim Operator activates the trigger report that the action is complete.
	Acknowledge request to place HWCH-SV-5719 (Condensate Booster Pump C H2 Injection Isolation Valve) in AUTO, have SIM OP initiate trigger 13 and report valve is open.
	Acknowledge request to place HWCH-SV-5717 (Condensate Booster Pump B H2 Injection Isolation Valve) in CLOSE and then report that the action is complete
	Acknowledge any requests for Radwaste actions.
	Acknowledge request to perform 2OP-32, Attachment 10. Report that you will co-ordinate the performance of the attachment with the WCC.
	Bus C Area is reported as clear if requested or if camera is checked.

Evaluator Notes	
Plant Response:	2C CBP is started and 2A CBP is secured.
Objectives:	SRO - Directs BOP to swap Condensate Booster Pumps BOP – Swap Condensate Booster Pumps RO – Monitors the plant
Success Path:	Condensate Booster Pumps are swapped
Event Termination:	When directed by the Lead Evaluator, go to Event 2.

EVENT 1: LOWER POWER TO 850 MWE

Time	Pos	EXPECTED Operator Response	NOTES
	SRO	Conduct shift turnover shift briefing.	
		Direct power to be reduced using recirc flow to ~850 MWe. (2OP-02, Section 6.2.1)	
		Contacts chemistry for samples due to 15% power change. May contact Load dispatcher to inform of power decrease. May conduct a brief (See Enclosure 1, page 62 for format of the brief.	
	RO	Reduces reactor power using recirc IAW 2OP-02 Section 6.2.1	
		May null the DVM meter.	
	BOP	Monitors the plant	

6.2 Shutdown

6.2.1 Lowering Speed/Power Using Individual Recirculation Pump Control Or Recirc Master Control

1. **Confirm** reactor recirculation pump in operation in accordance with Section 6.1.2 .... ________

NOTE

- Recirculation Pump speed changes are performed when directed by 0GP-05, Unit Shutdown, and 0GP-12, Power Changes. Other operating procedures are used simultaneously with this procedure as directed by 0GP-05, Unit Shutdown, and 0GP-12, Power Changes .... ☐
- Speed changes are accomplished by depressing Lower Slow, Lower Medium, or Lower Fast pushbuttons. The Lower Slow pushbutton changes Recirc pump speed at 0.06%/decrement at 1 rpm/second. The Lower Medium pushbutton changes Recirc pump speed at 0.28%/decrement at 5 rpm/second. The Lower Fast pushbutton changes Recirc pump speed at 2.8%/decrement at 100 rpm/second .... ☐

2. **IF AT ANY TIME** any of the following conditions exist, **THEN enter** 1AOP-04.0, Low Core Flow.{8.1.9} .... ________
   - Entry into Region A of Power to Flow Map
   - OPRM INOPERABLE **AND** any of the following
     - ◊ Entry into Region B of Power to Flow Map
     - ◊ Entry into 5% Buffer Region of Power to Flow Map
     - ◊ Entry into OPRM Enabled Region and indications of THI (Thermal Hydraulic Instability) exist

6.2.1 Lowering Speed/Power Using Individual Recirculation Pump Control Or Recirc Master Control (continued)

CAUTION

- The OPRM System monitors LPRMs for indication of thermal hydraulic instability (THI). When greater than or equal to 25% power and less than or equal to 60% recirculation flow, alarms and automatic trips are initiated upon detection of THI. Pump operations are governed by the limits of the applicable Power Flow Map, as specified in the COLR. {8.1.9} .... ☐
- Entry into the 5% Buffer Region warrants increased monitoring of reactor instrumentation for signs of Thermal Hydraulic Instability. Time in the 5% Buffer Region presents additional risk and is minimized.{8.1.9} .... ☐
- With core flow less than 57.5 x 10^6 lbs/hr, jet pump loop flows are required within 10% (maximum indicated difference 6.0 x 10^6 lbs/hr). With core flow greater than or equal to 57.5 x 10^6 lbs/hr, jet pump loop flows are required within 5% (maximum indicated difference 3.0 x 10^6 lbs/hr). .... ☐
- When Recirc Pump speeds are less than 40%, decreasing speed using a Lower Fast pushbutton can result in a Speed Hold condition due to exceeding the regen torque limit. .... ☐

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

3. IF desired to lower the speed of both recirculation pumps simultaneously, **THEN depress** Recirc Master Control Lower (Slow Medium Fast) pushbutton. .... ______

4. IF desired to lower the speed of an individual recirculation pump, **THEN depress** the Recirc VFD A(B) Lower (Slow Medium Fast) pushbutton. .... ______

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6.2.1 Lowering Speed/Power Using Individual Recirculation Pump Control Or Recirc Master Control (continued)

5. **Confirm** the following, as applicable:

- Recirc Pump A(B) Speed Demand, Calculated Speed, and Actual Speed have lowered .... ________
- Reactor power lowers .... ________
- B32-R617(R613) [Recirc Pump A(B) Discharge Flow] lowers .... ________
- B32-VFD-IDS-003A(B) [Recirc VFD 2A(B) Output Wattmeter] lowers .... ________
- B32-VFD-IDS-001A(B)]Recirc VFD 2A(B) Output Frequency Meter] lowers .... ________

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

Date/Time Completed ________________

Performed By (Print) Initials

________________________ ______

________________________ ______

________________________ ______

________________________ ______

Reviewed By: ________________________
Unit CRS/SRO

EVENT 2: ISOLATE 230 KV DELCO WEST LINE 30	
Simulator Operator Actions	

Simulator Operator Role Play	

Evaluator Notes	
Plant Response:	230 kV Delco West line is isolated
Objectives:	SRO - Direct 230kV Delco West Line isolated ATC – Plant monitoring BOP – Performs 2OP-50 Section 6.2.6 for isolating ONLY the Delco West Line
Success Path:	230 kV Delco West (Line 30) isolated
Event Termination:	Go to Event 3 at the direction of the Lead Evaluator.

EVENT 2: ISOLATE 230 KV DELCO WEST LINE 30

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Directs 230kV Delco West Line isolated IAW marked up version of 2OP-50, Section 6.2.6.	
	BOP	Performs 2OP-50, Section 6.2.6	
	RO	Monitors the plant.	

6.2.6 De-energizing The 230 kV Switchyard

1. **Ensure** the Unit 2 230 kV switchyard is ENERGIZED. .... AD
2. **Ensure** the 4kV Auxiliary Electrical Systems are DE-ENERGIZED in accordance with Section 6.2.3. .... N-1 SRO
3. **Ensure** the SAT is DE-ENERGIZED in accordance with Section 6.2.4 .... N-1 SRO
4. **Ensure** Caswell Beach Pumping Station is DE-ENERGIZED in accordance with Section 6.2.5 .... N-1 SRO
5. **Ensure** required LCOs for Technical Specification Sections 3.8.1, 3.8.2, 3.8.7 and 3.8.8 are initiated .... SRO
6. **Obtain** Load Dispatcher's permission to de-energize the 230 kV switchyard. .... AD

   A Powers - the Delco West Line ONLY
   Person Contacted

7. **Place** Auto Reclose switches for the following PCBs in MAN:
   - 31B (Bus 2B Whiteville 230 kV Breaker) .... N-1 SRO
   - 31A (Bus 2A Whiteville 230 kV Breaker) .... N-1 SRO
   - 30B (Bus 2B Delco West Line 230 kV Breaker) .... ______
   - 30A (Bus 2A Delco West Line 230 kV Breaker) .... ______
   - 28B (Bus 2B Wallace 230 kV Breaker) .... N-1 SRO
   - 28A (Bus 2A Wallace 230 kV Breaker) .... N-1 SRO
   - 27B (Bus 2B Town Creek 230 kV Breaker) .... N-1 SRO
   - 27A (Bus 2A Town Creek 230 kV Breaker) .... N-1 SRO

6.2.6 De-energizing The 230 kV Switchyard (continued)

CAUTION

PCB Supervisory switch must be in LOCAL before the associated PCB is operated from Panel XU-5. ☐

8. **Place** Supervisory switches for the following PCBs in LOCAL:
   - 31B (Bus 2B Whiteville 230 kV Breaker) N-1 SRO
   - 30B (Bus 2B Delco West Line 230 kV Breaker) ______
   - 28B (Bus 2B Wallace 230 kV Breaker) N-1 SRO
   - 27B (Bus 2B Town Creek 230 kV Breaker) N-1 SRO
   - 31A (Bus 2A Whiteville 230 kV Breaker) N-1 SRO
   - 30A (Bus 2A Delco West Line 230 kV Breaker) ______
   - 28A (Bus 2A Wallace 230 kV Breaker) N-1 SRO
   - 27A (Bus 2A Town Creek 230 kV Breaker) N-1 SRO
9. **Open** 31B (Bus 2B Whiteville 230 kV PCB) N-1 SRO
10. **Confirm** 31B (Bus 2B Whiteville 230 kV PCB) is OPEN by observing the indicating lights N-1 SRO
11. **Open** 31A (Bus 2A Whiteville 230 kV PCB) N-1 SRO
12. **Confirm** 31A (Bus 2A Whiteville 230 kV PCB) is OPEN by observing the indicating lights N-1 SRO
13. **Open** 30B (Bus 2B Delco West Line 230 kV PCB) ______
14. **Confirm** 30B (Bus 2B Delco West Line 230 kV PCB) is OPEN by observing the indicating lights ______
15. **Open** 30A (Bus 2A Delco West Line 230 kV PCB) ______
16. **Confirm** 30A (Bus 2A Delco West Line 230 kV PCB) is OPEN by observing the indicating lights ______
17. **Open** 28B (Bus 2B Wallace 230 kV PCB) N-1 SRO

6.2.6 De-energizing The 230 kV Switchyard (continued)

18. **Confirm** 28B (Bus 2B Wallace 230 kV PCB) is OPEN by observing the indicating lights .... N-1 SRO
19. **Open** 28A (Bus 2A Wallace 230 kV PCB) .... N-1 SRO
20. **Confirm** 28A (Bus 2A Wallace 230 kV PCB) is OPEN by observing the indicating lights .... N-1 SRO
21. **Open** 27B (Bus 2B Town Creek 230 kV PCB) .... N-1 SRO
22. **Confirm** 27B (Bus 2B Town Creek 230 kV PCB) is OPEN by observing the indicating lights .... N-1 SRO
23. **Open** 27A (Bus 2A Town Creek 230 kV PCB) .... N-1 SRO
24. **Confirm** 27A (Bus 2A Town Creek 230 kV PCB) is OPEN by observing the indicating lights .... N-1 SRO

NOTE

If work is to be performed on a 230 kV bus, the manual disconnects are to be opened .... ☐

25. **Place** Supervisory switches for the following PCBs in REMOTE:
    - 31B (Bus 2B Whiteville 230 kV Breaker) .... N-1 SRO
    - 30B (Bus 2B Delco West Line 230 kV Breaker) .... ______
    - 28B (Bus 2B Wallace 230 kV Breaker) .... N-1 SRO
    - 27B (Bus 2B Town Creek 230 kV Breaker .... N-1 SRO
    - 31A (Bus 2A Whiteville 230 kV Breaker) .... N-1 SRO
    - 30A (Bus 2A Delco West Line 230 kV Breaker) .... ______
    - 28A (Bus 2A Wallace 230 kV Breaker) .... N-1 SRO
    - 27A (Bus 2A Town Creek 230 kV Breaker) .... N-1 SRO

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6.2.6 De-energizing The 230 kV Switchyard (continued)

Date/Time Completed ________

Performed By (Print) Initials

________________ ________

________________ ________

________________ ________

________________ ________

Reviewed By ________________________

Unit CRS/SRO

N-1, Partial usage to isolate only the Delco West 230 kV Line (Line 30)

EVENT 3: ROD DRIFT

Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 1** to drift CR 26-11 into the core.
	When CR 26-11 is inserted to 00, **Initiate Trigger 2** to activate CRD High Temperature alarm.
	If control rod is scrammed, verify the rod drift malfunction deletes.
	Two minutes after control rod is disarmed or scrammed, delete CRD HYD TEMP HIGH alarm.
	If asked to disarm CRD 26-11 **Initiate Trigger 4**.

Simulator Operator Role Play	
	If contacted as the RE to address thermal limits, acknowledge the request. When contacted for scramming control rod 26-11, report that Thermal Limits will NOT be exceeded by this single rod scram.
	If asked as the RBAO to investigate HCU for control 26-11, report that the HCU scram outlet riser is hot to the touch.
	When contacted as the RBAO and after high temperature alarm has been actuated, report that the CRD temperature is 390°F and slowly rising.
	When contacted as the System Engineer report that based on past history of this rod (26-11) scram times cannot be guaranteed.
	If asked as the RBAO to disarm control rod, coordinate with Sim Operator after 5 minutes.
	If requested, close/reopen the 113 valve (Charging Header Isolation Valve) as necessary
	As RBAO, Report Accumulator pressure 980# after rod has been scrammed.

Evaluator Notes	
Plant Response:	Control Rod 26-11 will drift full in. Crew should enter AOP-02.0 and take action IAW 2APP-A-05 (3-2). When the high temperature alarm is received, Engineering will report that scram times cannot be assured based on past history of the control rod. Determine TS 3.1.3 condition C1 in 3 hours <u>and</u> C2 within 4 hours.
Objectives:	SRO - Direct actions in response to a drifting control rod and evaluate Tech Specs. RO - Respond to a drifting control rod.
Success Path:	The drifting control rod is fully inserted, determined that the control rod must be placed under clearance and electrically disarmed.
Event Termination:	Go to Event 4 at the direction of the Lead Evaluator.

EVENT 3: ROD DRIFT			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Direct actions of 2APP-A-05 (3-2) *ROD DRIFT*	
	SRO	Direct entry into 0AOP-02.0, Control Rod Malfunction/Misposition.	
	SRO	After System Engineer reports that the scram times cannot be guaranteed, according to Note 2 in TS Table 3.1.4-1 the rod must be declared inoperable. Tech Spec 3.1.3 Control Rod Operability Condition C. One or more control rods inoperable for reasons other than Condition A or B <u>Required Action</u> C.1 Fully insert inoperable control rod (3 hrs) C.2 Disarm the associated CRD (4 hrs)	
	SRO	Contact System Engineer on high temperature condition of control rod. Contact RE to inform of rod drift and to evaluate thermal limits	
	SRO	May direct the control rod to be scrammed to attempt to reseat the leaking outlet valve IAW A-05 (3-2) *ROD DRIFT* May conduct a brief (See Enclosure 1, page 62 for format of the brief.	
	BOP	Monitor reactor plant parameters during evolution. May read APP actions for the OATC to perform	

EVENT 3: ROD DRIFT

Time	Pos	EXPECTED Operator Response	Comments
	ATC	Acknowledge alarms: A-05 (2-2) ROD OUT BLOCK A-05 (3-2) Rod Drift Announce and enter 0AOP-02.0, Control Rod Malfunction/Misposition.	
	ATC	Perform the actions of APP-A-05 (3-2) ROD DRIFT as follows: ☐ Determine which control rod is drifting. ☐ Select the drifting control rod and determine direction of drift. ☐ Attempt to arrest the drift by giving a withdraw signal. ☐ If rod continues to drift in, apply an RMCS insert signal and fully insert to position 00. ☐ Attempt to locate and correct the cause of the rod malfunction as follows: ☐ Check and adjust cooling water header pressure if required. ☐ Direct AO to check for leaking scram valve. ☐ May direct an AO to check HCU temperature on RO18 temperature recorder (in the Rx Bldg.)	
	ATC	Monitor core parameters, main steam line radiation and off-gas activity.	
	ATC	Perform 2OP-07 Section 6.3.17, Single Rod Scram from RPS Test Panel. CRS will NA appropriate steps.	The examiner will **prompt** the performer that the "Green light is ON" and the control rod is fully inserted when step 6.3.17.11 is performed.

6.3.17 Single Rod Scram From RPS Test Panel

1. **Confirm** the following initial conditions are met:

- All applicable prerequisites in Section 5.0 are met. .... ________
- Attachment 1 has been reviewed. .... ________
- Communications are established between RPS Test Panel and the Control Room. .... ________
- Reactor Engineer recommends performance of this section and has determined Technical Specification Thermal Limits will **NOT** be exceeded by this single rod scram......................... ________

____

Reactor Engineer

2. **IF AT ANY TIME** it becomes necessary to scram a single control rod for operability concerns **THEN perform** 0PT-14.2.1, Single Rod Scram Insertion Times Test for that control rod. .... ________

3. **Obtain** permission from the Unit CRS to perform this section. .... ________
CRS

4. **Document** applicable control rod to be scrammed in the space provided: .... ________

_____-_____

Control Rod

5. **IF** recommended by Reactor Engineering to support diagnostic data, **THEN record** the following: .... ________

- Reactor pressure: .... ________

  __________ psig

- Applicable accumulator pressure: .... ________

  __________ psig

6.3.17 Single Rod Scram From RPS Test Panel (continued)

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

6. **Select** applicable control rod at P603. .... ____/____ CV

7. **Close** C12-113 (Charging Water Riser Isolation Valve) for the applicable control rod. .... ________

8. **IF** RWM scram time recording is recommended by Reactor Engineering, **THEN perform** the following: .... ________

   a. **Have** Reactor Engineering connect temporary scram time test cable to single rod scram interface box (located on terminal strip GM in P616-RMCS cabinet) and route cable up to RPS Test Panel P610 in accordance with Attachment 12, (Reference Use) - Test Cable Arrangement For RWM Scram Recording. .... ________

      ____
      Reactor Engineer

      (1) **Insert** black lead into NEUTRAL socket on the P610 test panel. .... ____/____ IV

      (2) **Insert** red lead into socket corresponding to control rod to be tested at P610. .... ____/____ IV

9. **Monitor** control rod position. .... ________

10. **IF AT ANY TIME** the control rod does **NOT** fully scram after lowering the scram test switch, **THEN** immediately **notify** the Unit CRS to determine operability of the rod (Technical Specification 3.1.3). .... ________

11. Using a currently licensed RO/SRO, **perform** the following:

    a. **Scram** the applicable control rod by lowering the scram test switch on RPS Test Panel P610 to the scram (down) position. .... ____/____ CV

6.3.17 Single Rod Scram From RPS Test Panel (continued)

b. **WHEN** the scrammed control rod is fully inserted **OR** 10 seconds have elapsed (whichever occurs first), **THEN return** applicable scram test switch to the normal (up) position. .... ____/____
IV

12. **Confirm** rod position display indicates "00" for scrammed rod and the GREEN "Full In" light is ON. .... ________

13. **IF** control rod did **NOT** fully insert, **THEN reference** Technical Specifications for OPERABILITY. .... ________
CRS

NOTE

Holding Emergency Rod In Notch Override switch in EMERGENCY ROD IN position for a period of time will flush any ingested crud from the drive to help prevent double notching. .... ☐

14. **Hold** the Emergency Rod In Notch Override switch in EMERGENCY ROD IN position for at least 15 seconds and **record** insert stall flow.

______ stall flow	______ stall flow	______ stall flow
1:____	2:____	3:____

15. **Repeat** Section 6.3.17 Step 14 two additional times. .... ________

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

16. Slowly **open** applicable C12-113 (Charging Water Riser Isolation Valve). .... ____/____
IV

17. **Confirm** associated accumulator pressure is greater than 955 psig. .... ________

6.3.17 Single Rod Scram From RPS Test Panel (continued)

18. **IF** RWM scram time was recorded,
THEN perform the following: .... ________

a. **Contact** Reactor Engineering to upload data .... ________

____
Reactor Engineer

b. **Remove** temporary scram timing cables from P616 and P610.. ____/____
IV

c. **Perform** the following to delete RWM scram data buffers:

(1) **Select** SCRAM DATA screen on RWM Operator Display in the Control Room.. ________

(2) **Press** DELETE softkey to delete scram data.................. ________

(3) **Confirm** SCRAM DATA screen displays:........................ ________

- ROD SCRAM TIMING FUNCTION: READY
- ROD SCRAM TIMING DATA: NOT TRANSFERRED

EVENT 4: ADHR SECONDARY PUMP TRIP	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 5** to trip the running ADHR Pump.
	When informed to align 2A RCC pump to ADHR mode **Initiate Trigger 6**
	If asked to throttle closed the RCC-V5114, **Initiate Trigger 7**. When asked to re-open the RCC-V5114, then adjust the remote to 1.0.

Simulator Operator Role Play	
	If directed to investigate the trip of RCC Pump D, report the pump is tripped on overcurrent.
	When directed to align RBCCW Pump 2A to ADHR mode IAW 2OP-21 Section 6.3.16 (steps 2b through 2i) have Sim Op align pump to ADHR mode and inform BOP Op that the steps are complete.
	When contacted as RBAO report radiation monitor is aligned per 2OP-21 Section 6.3.18 step 4. RCC-V5154 (Rad Monitor Bypass Standby Isolation Valve) is CLOSED RCC-V5116 (Rad Monitor Bypass ADHR Isolation Valve) is OPEN RCC-V5115 (Rad Monitor Bypass Common Mode Isolation Valve) is OPEN
	When contacted report RCC-V5114 (RBCCW Pump 2A ADHR Mode Discharge Valve) is throttled 90% closed. (2OP-21 Section 6.3.18 Step 5a)
	When contacted report RCC-V5114 (RBCCW Pump 2A ADHR Mode Discharge Valve) is full open. (2OP-21 Section 6.3.18 Step 5c)

Evaluator Notes	
Plant Response:	The running ADHR Secondary Loop Pump (RCC Pump D) will trip. The crew will have to start RCC Pump C. Shutdown RCC Pump A. Re-align RCC Pump A for ADHR mode and then start the pump for ADHR. (AOP-38.0 will be entered).
Objectives:	SRO – Direct swapping of RCC pumps and then direct starting of RCC Pump in ADHR Mode. RO – Swap RCC pumps, Place RCC Pump in ADHR Mode.
Success Path:	Standby ADHR Pump placed in service.
Event Termination:	Go to Event 5 at the direction of the Lead Evaluator.

EVENT 4: ADHR SECONDARY PUMP TRIP

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct entry into 0AOP-38.0, Loss of Fuel Pool Cooling	
		Direct swapping of RBCCW pumps Start RBCCW Pump C, secure A.	
		Direct alignment of RBCCW Pump A to ADHR Mode.	
		Direct starting RBCCW Pump 2A in ADHR Mode.	
		Direct I/C to investigate trip of RBCCW Pump 2D.	
		May conduct a brief (see Enclosure 1 on page 62 for format)	

EVENT 4: ADHR SECONDARY PUMP TRIP			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	RO	Plant Monitoring	
	BOP	Report trip of RBCCW Pump 2D (running in ADHR Mode) UA-01 3-3, ADHR SECONDARY LOOP TROUBLE	
		Announce and enter AOP-38.0, Loss of Fuel Pool Cooling	
		Perform 2OP-21, Section 6.3.10 (page 33) to swap RBCCW pumps. (Start C and secure A) Plant announcement for the start of 2C RCC Pump and securing of 2A RCC Pump.	
		Perform 2OP-21, Section 6.3.16 (page 34) to align RBCCW Pump A into ADHR Mode. Direct RB AO to perform steps 2b through 2i. Step 3 is N/A	
		Perform 2OP-21, Section 6.3.18 (page 37) to start RBCCW Pump A in ADHR Mode. Notifies E&C, starting ADHR pump Step 2 is N/A Step 3 is N/A Direct the RB AO to perform step 4 and 5a. Announce starting of RCC Pump 2A. Direct the RB AO to perform step 5c. May direct AO to ensure primary loop is operating IAW 2OP-13.1.	

6.3.10 Transferring to the Standby RBCCW Pump - RBCCW Mode

1. **Ensure** the following initial conditions are met:
   - Applicable prerequisites listed in Section 5.0, Prerequisites are met .... ________
   - RBCCW System in operation with two pumps aligned for RBCCW Mode in service .... ________
2. **Start** the standby RBCCW pump by placing the associated pump control switch in ON:
   - RBCCW PUMP 2A .... ________
   - RBCCW PUMP 2B .... ________
   - RBCCW PUMP 2C .... ________
   - RBCCW PUMP 2D .... ________
3. **Secure** the desired RBCCW pump by placing the associated pump control switch in OFF:
   - RBCCW PUMP 2A .... ________
   - RBCCW PUMP 2B .... ________
   - RBCCW PUMP 2C .... ________
   - RBCCW PUMP 2D .... ________
4. <u>IF</u> a third RBCCW pump is aligned to RBCCW Mode, <u>**AND**</u> RBCCW discharge header pressure has stabilized,
   <u>**THEN**</u> **place** the pump control switch in AUTO .... ________

6.3.16 Alignment of RBCCW Pump from RBCCW Mode to ADHR Mode

1. **Ensure** the following initial condition is met:

   - One RBCCW Heat Exchanger is aligned to ADHR Mode per Section 6.3.14 .... ________

   - Key for RBCCW/ADHR Mode Selector Switch has been obtained from one of the following:

     ◊ Control Rm Key Locker - key 98 .... ________

     ◊ WCC Key Locker - key 167 or 168 .... ________

NOTE

RBCCW Pump 2A and RBCCW Pump 2D can support either RBCCW Mode or ADHR Mode. A Mode Selector Switch is located on the pump breaker and a white ADHR Mode indicating light is on the RTGB. This switch determines which of the two header pressures (RBCCW or ADHR) will be monitored for the pump auto start on low header pressure when the pump control switch is placed in AUTO. When the Mode Selector Switch is placed in the ADHR Mode position, the white light is ON on the RTGB .... ☐

2. IF aligning RBCCW Pump 2A to ADHR Mode, **THEN perform** the following .... ________

   a. **Ensure** RBCCW Pump 2A control switch is in OFF .... ________

   b. **Close** RCC-V32 (RBCCW Pump 2A RBCCW Suction) .... ________

   c. **Close** RCC-V38 (RBCCW Pump 2A RBCCW Mode Discharge Valve) .... ________

   d. **Open** RCC-V5105 (RBCCW Pump 2A ADHR Mode Suction Valve) .... ________

   e. **Open** RCC-V5114 (RBCCW Pump 2A ADHR Mode Discharge Valve) .... ________

   f. **Open** RCC-V303 (RBCCW Pump 2A Casing Vent Valve) .... ________

   g. WHEN a steady stream of water is present, **THEN close** RCC-V303 (RBCCW Pump 2A Casing Vent Valve) .... ________

   h. **Ensure** 2-RCC-SS-7667 (Pump 2A RBCCW/ADHR Mode Selector Switch) located at MCC 2XE, in ADHR .... ________

6.3.16 Alignment of RBCCW Pump from RBCCW Mode to ADHR Mode (continued)

i. **Remove** key from 2-RCC-SS-7667 (Pump 2A RBCCW/ADHR Mode Selector Switch) ____

j. **Confirm** the ADHR white indicating light on the RTGB for RBCCW Pump 2A is ON. ____

3. IF aligning RBCCW Pump 2D to ADHR Mode, THEN **perform** the following: ____

a. **Ensure** RBCCW Pump 2D control switch is in OFF. ____

b. **Close** RCC-V5107 (RBCCW Pump 2D RBCCW Mode Suction Valve) ____

c. **Close** RCC-V5111 (RBCCW Pump 2D RBCCW Mode Discharge Valve) ____

d. **Open** RCC-V5104 (RBCCW Pump 2D ADHR Mode Suction Valve) ____

e. **Open** RCC-V5113 (RBCCW Pump 2D ADHR Mode Discharge Valve) ____

f. **Open** RCC-V5139 (RBCCW Pump 2D Casing Vent Valve) ____

g. WHEN a steady stream of water is present, THEN **close** RCC-V5139 (RBCCW Pump 2D Casing Vent Valve) ____

h. **Ensure** 2-RCC-SS-7668 (Pump 2D RBCCW/ADHR Mode Selector Switch) located at MCC 2XD, in ADHR ____

i. **Remove** key from 2-RCC-SS-7668 (Pump 2D RBCCW/ADHR Mode Selector Switch) ____

j. **Confirm** the ADHR white indicating light on the RTGB for RBCCW Pump 2D is ON. ____

6.3.16 Alignment of RBCCW Pump from RBCCW Mode to ADHR Mode (continued)

NOTE

ADHR Mode piping is placed either in Standby Mode or in service to ensure RBCCW circulation and proper chemistry control when NOT undergoing maintenance. ☐

4. **Place** ADHR in service per Section 6.3.18, Starting an RBCCW Pump - ADHR Mode ________

Date/Time Completed ________

Performed By (Print)	Initials
____________	______
____________	______
____________	______
____________	______

Reviewed By ____________________

Unit CRS/SRO

6.3.18 Starting an RBCCW Pump - ADHR Mode

1. **Ensure** the following initial conditions are met:

   - Designated RBCCW Pump is aligned to ADHR Mode per Section 6.3.16. .... ________

   - **IF NO** RBCCW pump is operating in ADHR Mode, **THEN** E&C notified commencing startup of ADHR Mode. .... ________

     ____
     Person Notified

NOTE

- RBCCW Pump 2A and RBCCW Pump 2D can support either RBCCW Mode or ADHR Mode. A Mode Selector Switch is located on the pump breaker and a white ADHR Mode indicating light is on the RTGB. This switch determines which of the two header pressures (RBCCW or ADHR) will be monitored for the pump auto start on low header pressure when the pump control switch is placed in AUTO. When the Mode Selector Switch is placed in the ADHR Mode position, the white light is ON on the RTGB. .... ☐
- RBCCW Pump 2D will **NOT** auto re-start when power returns after a LOOP or bus under voltage condition with the control switch in ON or AUTO. The control switch must be placed in OFF/RESET prior to restarting the pump. .... ☐

CAUTION

Two pump operation in ADHR Mode subjects RCC-V37 (RBCCW Pump 1A Discharge Check Valve) and RCC-V5110 (RBCCW Pump 1D Discharge Check Valve) to accelerated wear. This lineup is expected to be utilized only when maximum ADHR capacity is required. [8.7.2] .... ☐

2. **IF** desired to start a second pump aligned to ADHR Mode, **THEN perform** the following: .... ________

   a. **Obtain** concurrence from Engineering to start a second pump in the ADHR Mode. .... ________

      ____
      Person Contacted

   b. **Go to** Step 5.b. .... ________

6.3.18 Starting an RBCCW Pump - ADHR Mode (continued)

3. **IF** the ADHR Mode has been shutdown for greater than 72 hours
OR maintenance has been performed,
THEN fill and **vent** the ADHR piping per Section 6.3.13 .... ________

4. **Ensure** the following valve alignment for system radiation monitoring:

   - RCC-V5154 (Rad Monitor Bypass Standby Isolation Valve) is CLOSED .... ________

   - RCC-V5116 (Rad Monitor Bypass ADHR Isolation Valve) is OPEN .... ________

   - RCC-V5115 (Rad Monitor Bypass Common Mode Isolation Valve) is OPEN .... ________

5. For the RBCCW pump aligned to ADHR Mode to be started, **perform** the following:

   a. **Throttle** 80% to 95% closed the associated pump discharge valve:

      - RCC-V5114 (RBCCW Pump 2A ADHR Mode Discharge Valve) .... ________

      - RCC-V5113 (RBCCW Pump 2D ADHR Mode Discharge Valve) .... ________

   b. **Start** an RBCCW pump aligned to ADHR Mode by placing the associated pump control switch in ON:

      - RBCCW PUMP 2A .... ________

      - RBCCW PUMP 2D .... ________

   c. **IF** throttled in Step 5.a,
   THEN slowly **open** the associated pump discharge valve: .... ________

      - RCC-V5114 (RBCCW Pump 2A ADHR Mode Discharge Valve) .... ________

      - RCC-V5113 (RBCCW Pump 2D ADHR Mode Discharge Valve) .... ________

6.3.18 Starting an RBCCW Pump - ADHR Mode (continued)

d. **Ensure** a log entry is made stating two RBCCW pumps are in service in the ADHR Mode and Engineering has been notified. ________

NOTE

The normal parameters for Supplemental Spent Fuel Pool Cooling are provided in Attachment 1, Normal System Operation Parameters. Equipment manipulations to maintain these parameters are performed per 2OP-13.1, Alternate Decay Heat Removal System Primary Loop Operating Procedure. ☐

6. **IF** a Primary Loop pump is operating,
THEN maintain Primary Loop flow per 2OP-13.1, Alternate Decay Heat Removal System Primary Loop Operating Procedure. ________

7. **Ensure** Plant Process Computer setup as follows per 0OP-55, Plant Process and ERFIS Computer Systems Operating Procedure: ________

- PPC U2RCCA111 point ENABLED. ________
- PPC U2RCCA095 Value Monitoring setup with the nominal flow values per Attachment 1 Section 2.5 for the number of RBCCW Pumps in ADHR Mode to provide audible alarms for ADHR secondary flow changes. ________

Date/Time Completed ________

Performed By (Print)	Initials
________	________
________	________
________	________
________	________

Reviewed By ________________

Unit CRS/SRO

EVENT 5: RECIRC LOOP B FLOW TRANSMITTER FAILURE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 8** to activate Recirc Loop B Flow failure.

Simulator Operator Role Play	
	If contacted as I&C to investigate, acknowledge the request.
	After LCO entries have been determined and SRO is waiting for I&C, call as WCCSRO and request APRM 4 be placed in tripped condition to support I&C trouble shooting. The WCC will hang the status control tag paperwork.
	If asked to pull fuses (for TRM 3.3 actions, 2-C12A-F1 Labeled ROD OUT BLOCK RELAYS C12A in P616 panel) acknowledge the request

Evaluator Notes	
Plant Response:	Flow reference off normal alarm, rod block and scram signal to all 4 voters Flow transmitter signals are displayed on PC display 845, and on individual NUMACs by selecting Input Status.
Objectives:	SRO - Determine LCO for APRM 4 inoperability and direct placing channel in trip. RO - Respond To A Flow Unit/Transmitter Failure Per APP A-06 5-7.
Success Path:	ARPM 4 TS 3.3.1.1 declaration and placed in trip condition IAW 0OI-18.
Event Termination:	Go to Event 6 at the direction of the Lead Evaluator.

EVENT 5: RECIRC LOOP B FLOW TRANSMITTER FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of APPs	
		Direct I&C to investigate	
		Evaluate Tech Spec 3.3.1.1 Reactor Protection System Instrumentation TS 3.3.1.1, Function 2b, Required Action A1. With one or more required channels inoperable, place in trip condition in **12 hours** Evaluate TRM 3.3 Control Rod Block Instrumentation TRM 3.3, Function 1a, Required Condition A1. With one of the required channels not operable - **24 hours** to restore to operable.	
		Refers to 0OI-18 for actions to place APRM 4 in a tripped condition.	
		Direct APRM 4 mode selector switch placed in INOP to allow I&C troubleshooting.	
		May conduct a brief (see Enclosure 1 on page 62 for format)	

EVENT 5: RECIRC LOOP B FLOW TRANSMITTER FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Monitors the plant.	
		May check back panel APRM indications.	
	ATC	Acknowledges, refers to & reports annunciators A-6 2-8 *APRM UPSCALE* 3-8 *APRM UPSCALE TRIP/INOP* 5-7 *FLOW REF OFF NORMAL* A-5 2-2 *ROD OUT BLOCK* 4-8 *OPRM TRIP ENABLED*	
		Diagnose and report failure of APRM 4 Flow Transmitter	
		Obtains key number 114 from the SRO key locker to place APRM 4 in trip.	
		Places APRM mode selector switch in INOP IAW 0OI-18.	

EVENT 6: HEATER DRAIN DEAERATOR PUMP TRIP	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 9** to trip a Heater Drain Pump.
	.

Simulator Operator Role Play	
	If contacted as AO to investigate, wait until pump is tripped and report over-current flags for all phases of 2A HDP 4KV breaker on Bus 2D
	If contacted as RE for reduced FW Temp, acknowledge any requests.
	If asked as I&C to investigate, acknowledge the request

Evaluator Notes	
Plant Response:	Heater Drain Pump 2A shaft seizes and trips on overcurrent. Heater Drain tank level will rise and the crew will throttle HD-V57 to stabilize HD Tank level. If the standby HDP is not started, RFP suction pressure will lower during the transient requiring power reduction to stabilize Condensate/feedwater.
Objectives:	SRO - Directs 0AOP-23, Condensate/Feedwater System Failures, and possible 0AOP-03.0, Positive Reactivity Addition, entry. RO - Diagnose HDP pump trip and start the standby HDP.
Success Path:	2C HDP started with HDD level recovered in the normal band.
Event Termination:	Go to Event 7 at the direction of the Lead Evaluator.

EVENT 6: HEATER DRAIN DEAERATOR PUMP TRIP

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct annunciator response for UA-04: 4-10 HD PUMP A TRIP 2-10 HD DEAERATOR LEVEL HIGH-LOW 3-10 HD DEAERATOR LEVEL HIGH TRIP Direct entry into 0AOP-23, Condensate/Feedwater System Failures Direct starting standby HDP.	
		May direct 2AOP-3.0, Positive Reactivity Addition, entry if power rises due to the HDD Ext Trip.	
		May direct monitoring of final feedwater temperature.	
		May direct maintenance to investigate trip	
		May conduct a brief (see Enclosure 1 on page 62 for format)	
	RO	Plant Monitoring	
		May reduce power IAW 0AOP-23 to stabilize reactor water level.	

EVENT 6: HEATER DRAIN DEAERATOR PUMP TRIP			
	BOP	Recognize and report annunciators: UA-04 4-10 *HD PUMP A TRIP* 2-10 *HD DEAERATOR LEVEL HIGH-LOW* 3-10 *HD DEAERATOR LEVEL HIGH TRIP* UA-06 1-7 *BUS 2D 4 KV MOTOR OVLD*	
		Manually starts 2C HDP IAW APP or AOP.	
		Enter and announce 0AOP-23, Condensate/Feedwater System Failures. Monitors final feedwater temperature (FFWT) IAW 2OI-03.2	
		May open the HD-V57 to assist in HDD level recovery.	
		Directs an AO to 4.16 KV Switchgear Bus 2D to investigate 2A HDP trip	
		Verifies auto actions for *HD DEAERATOR LEVEL HIGH TRIP*, if it occurs. 1. Non-return valves (EX-V11 and EX-V12) to deaerator close. (Only close if turbine load is below 500 MWe) 2. HDD Extraction Line B moisture removal valves (MVD-LV-266 and MVD-LV-267) open. May reference 2OP-35 to recover MRVs following HDD level restoration.	

EVENTS 7/8/9: STEAM LEAK IN DW - ATWS / SLC SWITCH FAILURE / ARI RESET FAILURE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 10** to activate small steam leak in DW.
	If requested to defeat Group I LL3, wait 2 minutes, and install jumpers **(Trigger 11)**
	If requested to install LEP-02, Section 2.3 jumpers, wait 5 minutes, and inform the SRO that the jumpers are installed (RP005F already active).

Simulator Operator Role Play	
	Acknowledge request as I&C to investigate failure of SLC switch.
	If requested as I&C to investigate the failure of the ARI reset failure, acknowledge the request.

Evaluator Notes	
Plant Response:	Most control rods will fail to insert on the scram. The crew will respond to the ATWS per EOP-01-ATWS. When SLC initiation is attempted, the switch positions will not work. The crew will enter LEP-03 and align for alternate boron injection using CRD. The scram cannot be reset due to failure of the ARI to reset.
Objectives:	SRO - Direct actions to control reactor power per EOP-01-ATWS.. RO - Perform actions for an ATWS per EOP-01-ATWS.
Success Path:	Lower level to control power, inject SLC, insert control rods.

EVENTS 7/8/9: STEAM LEAK IN DW - ATWS / SLC SWITCH FAILURE / ARI RESET FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Enter RSP and transition to ATWS. Direct mode switch to shutdown when steam flow < 3 Mlbs/hr. Direct ARI initiation. Direct Recirc Pumps Tripped. ***Direct SLC initiation, then LEP-03, Alternate Boron Injection.*** Direct ADS inhibited. Direct RWCU isolation verification. ***Direct LEP-02, Alternate Rod Insertion***	***CRITICAL TASK #2*** ***CRITICAL TASK #1***
		Direct Group 10 switches to override reset.	
		Direct terminate and prevent HPCI/Feedwater (CS/RHR when LOCA signal received) to lower level to 90 inches.	***CRITICAL TASK #3***
		When level reaches 90 inches, evaluate Table Q-2: If not met, establishes a level band of LL4 to +90 inches. If met, directs RPV injection to remain terminated.	
		When Torus temperature is greater than 95° F, enters PCCP and directs Torus Cooling. (See Enclosure 5, page 68)	
		Directs Drywell cooling restored per SEP-10.	
		Direct injection established to maintain RPV level LL4 to TAF (or the level at which APRMs indicate downscale)	

EVENTS 7/8/9: STEAM LEAK IN DW - ATWS / SLC SWITCH FAILURE / ARI RESET FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	RO	Place mode switch to shutdown when steam flow < $3x10^6$ lb/hr.	
		Initiates ARI.	
		Trips Recirc Pumps.	
		Initiates SLC. ***Determines SLC switch failure.*** ***Directs LEP-03, Alternate Boron Injection***	*CRITICAL TASK #2*
		Recognizes failure of SLC switch and reports to CRS.	
		Monitor APRMs for downscale.	
		Performs LEP-02, Alternate Rod Insertion. ***Section 2.1, Initial Actions*** (see page 48) ***Section 2.3, Reset RPS and Initiate a Manual Scram*** (see page 51) ***Section 2.4, Reactor Manual Control System (RMCS)*** (see page 54) May also perform Section 2.5, Increasing Cooling Water Header Pressure (see page 56).	*CRITICAL TASK #1*
		Recognizes failure of ARI to reset, informs CRS	

EVENTS 7/8/9: STEAM LEAK IN DW - ATWS / SLC SWITCH FAILURE / ARI RESET FAILURE			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	BOP	Places ADS in inhibit.	
		Places Group 10 switches to override / reset	
		Terminate and prevent injection to RPV. Terminates and prevents HPCI IAW Hard Card. (Enclosure 2, page 63) Terminates and Prevents Feedwater IAW Hard Card. (Enclosure 3, page 64)	***CRITICAL TASK #3***
		May place HPCI in service for level control during ATWS when directed by the SRO. (Enclosure 6, page 70)	
		Restart RFP to maintain level as directed by SRO. (Enclosure 4, page 65)	
		When Torus temperature is greater than 95° F, places Torus Cooling in service. (Enclosure 5, page 68)	

ALTERNATE CONTROL ROD INSERTION	0EOP-01-LEP-02
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1.0 ENTRY CONDITIONS

- As directed by Emergency Operating Procedures (EOPs)
- As directed by Severe Accident Management Guideline (SAMGs)

2.0 OPERATOR ACTIONS

2.1 <u>Initial Actions</u>

2.1.1 Manpower Required

- 1 Reactor Operator
- 1 Auxiliary Operator

2.1.2 Special Equipment

None

NOTE

- Two-handed operation is allowed during implementation of this procedure in order to minimize delay in control rod insertion. ☐
- Section 2.1.3 Step 1 through Step 6 may be performed concurrently with the rest of this procedure. ☐
- The system designation C11 is for Unit 1 and C12 for Unit 2. ☐

2.1.3 Operator Actions

1. **Monitor** reactor power on APRMs until IRM recorders on scale. ☐ RO

2. **Insert** IRMs and **monitor** reactor power on IRM recorders. ☐ RO

3. **Downrange** IRMs to bring them on scale. ☐ RO

4. <u>**WHEN**</u> IRMs on Range 3 <u>**OR**</u> below, <u>**THEN**</u> **insert** SRMs. ☐ RO

5. **Monitor** reactor period. ☐ RO

2.1.3 Operator Actions (continued)

6. **Monitor** control rod position using:

- Process computer .... ☐ RO
- SPDS .... ☐ RO
- RWM .... ☐ RO
- Four rod .... ☐ RO
- Full core display .... ☐ RO

7. **WHEN** either:

- All control rods in .... ☐ RO
- Only one control rod **NOT** fully inserted .... ☐ RO
- **NO** more than 10 control rods withdrawn to position 02 **AND** **NO** control rod withdrawn beyond position 02 .... ☐ RO
- Reactor engineering has determined the reactor will remain shutdown under all conditions without boron .... ☐ RO

THEN perform Section 2.2, Control Rod Insertion Verification on Page 7. .... ☐ RO

2.1.3 Operator Actions (continued)

8. **Insert** control rods by one or more methods:

- Section 2.3, Reset RPS and Initiate a Manual Scram on Page 15 ☐ RO
- Section 2.4, Reactor Manual Control System (RMCS) on Page 18 ☐ RO
- Section 2.5, Increasing Cooling Water Header Pressure on Page 20 ☐ RO
- Section 2.6, Scram Individual Control Rods on Page 22 ☐ RO
- Section 2.7, De-energize Scram Solenoids and Vent Scram Air Header on Page 26 ☐ RO
- Section 2.8, Venting Over Piston Area on Page 32 ☐ RO

2.3 Reset RPS and Initiate a Manual Scram

2.3.1 Manpower Required

- 1 Reactor Operator

2.3.2 Special Equipment

- RO Desk Locked Drawer
  - ◊ 4 jumpers (15, 16, 17 and 18) ☐

2.3.3 Manual Scram Actions

NOTE

Section 2.3.3 Step 1 and Step 2 may be performed concurrently. ☐

1. **IF** an automatic scram signal present **AND** power available to RPS bus,
   THEN install jumpers to bypass reactor scram:

   - Jumper 15 in Panel H12-P609, Terminal Board DD, from right side of Fuse C71A(C72A)-F14A to Terminal 4 of Relay C71A(C72A)-K12E ☐ RO

   - Jumper 16 in Panel H12-P609, Terminal Board BB, from left side of Fuse C71A(C72A)-F14C to Terminal 4 of Relay C71A(C72A)-K12G ☐ RO

   - Jumper 17 in Panel H12-P611, Terminal Board DD, from right side of Fuse C71A(C72A)-F14B to Terminal 4 of Relay C71A(C72A)-K12F ☐ RO

   - Jumper 18 in Panel H12-P611, Terminal Board BB, from left side of Fuse C71A(C72A)-F14D to Terminal 4 of Relay C71A(C72A)-K12H ☐ RO

2. **Inhibit** ARI:

   a. **Place** C11(C12)-CS-5560 (ARI Auto/Manual Initiation Switch) to INOP. ☐ RO

2.3.3 Manual Scram Actions (continued)

b. **Place and hold** C11(C12)-CS-5562 (ARI Reset) switch in RESET .... ☐ RO

c. **WHEN** 5 seconds have elapsed, **THEN release** .... ☐ RO

d. **Confirm** red TRIP light located above C11(C12)-CS-5561 (ARI Initiation) OFF .... ☐ RO

3. **Ensure** Disch Vol Vent & Drain Test switch in ISOLATE .... ☐ RO

4. **Confirm** CLOSED:

- C11(C12)-V139 (Disch Vol Vent Vlv) .... ☐ RO
- C11(C12)-CV-F010 (Disch Vol Vent Vlv) .... ☐ RO
- C11(C12)-V140 (Disch Vol Drain Vlv) .... ☐ RO
- C11(C12)-CV-F011 (Disch Vol Drain Vlv) .... ☐ RO

5. **Reset** RPS .... ☐ RO

6. **IF** either RPS A **OR** B can be RESET, **THEN go to** Section 2.3.3 Step 8 .... ☐ RO

7. **IF** RPS **CANNOT** be RESET, **THEN return to** Section 2.1.3 Step 7 .... ☐ RO

8. **Place** Disch Vol Vent & Drain Test switch to NORMAL .... ☐ RO

2.3.3 Manual Scram Actions (continued)

9. **Confirm** OPEN:

   - C11(C12)-V139 (Disch Vol Vent Vlv) ☐ RO
   - C11(C12)-CV-F010 (Disch Vol Vent Vlv) ☐ RO
   - C11(C12)-V140 (Disch Vol Drain Vlv) ☐ RO
   - C11(C12)-CV-F011 (Disch Vol Drain Vlv) ☐ RO

10. **WHEN** the scram discharge volume has drained for approximately 2 minutes **OR** A-05 1-6, SDV Hi-Hi Level RPS Trip clears, **THEN continue** ☐ RO

11. **IF** venting control rod over piston area per Section 2.8, **THEN notify** AO to secure venting prior to inserting a manual scram ☐ RO

12. Manually **scram** the reactor ☐ RO

13. **IF** control rods moved inward **AND** all control rods **NOT** inserted to **OR** beyond Position 00, **THEN return to** Section 2.3.3 Step 3 ☐ RO

14. **IF** all control rods inserted to **OR** beyond Position 00 **OR** control rods did **NOT** move inward, **THEN return to** Section 2.1.3 Step 7 ☐ RO

2.4 Reactor Manual Control System (RMCS)

2.4.1 Manpower Required

- 1 Reactor Operator

2.4.2 Special Equipment

- RO Desk Locked Drawer
  - ◊ **Unit 1 Only:** 1 5450 key for RWM .... ☐
  - ◊ **Unit 2 Only:** 1 5451 key for RWM .... ☐

2.4.3 RMCS Actions

1. **IF** a reactor scram sealed in,
   THEN ensure available CRD pumps operating .... ☐ RO

2. **Ensure** C11(C12)-FC-R600 (CRD Flow Control) in MAN .... ☐ RO

3. **IF** a CRD pump **NOT** operating,
   THEN:

   a. **Close** the in-service C11(C12)-F002A(F002B) (Flow Control Vlv) .... ☐ RO

   b. **Start** one CRD pump .... ☐ RO

   c. **Adjust** C11(C12)-FC-R600 (CRD Flow Control) to greater than or equal to 30 gpm. .... ☐ RO

   d. **IF** available,
      THEN start the second CRD pump. .... ☐ RO

4. **IF NO** CRD pump can be started,
   THEN return to Section 2.1.3 Step 7. .... ☐ RO

2.4.3 RMCS Actions (continued)

5. **Insert** control rods with RMCS:

   a. **Throttle open** C11(C12)-F002A(F002B) (Flow Control Vlv) until drive water differential pressure greater than or equal to 260 psid. ☐ RO

   b. **IF** drive water differential pressure less than 260 psid, **THEN throttle closed** C11(C12)-PCV-F003 (Drive Pressure Vlv) until drive water differential pressure greater than or equal to 260 psid. ☐ RO

   c. **Bypass** RWM. ☐ RO

   d. **Insert** control rods with Emergency Rod In Notch Override switch. ☐ RO

6. **WHEN** all control rods inserted to **OR** beyond Position 00 **OR** **CANNOT** be inserted with RMCS, **THEN return to** Section 2.1.3 Step 7 ☐ RO

2.5 Increasing Cooling Water Header Pressure

2.5.1 Manpower Required

- 1 Reactor Operator

2.5.2 Special Equipment

None

2.5.3 Cooling Water Header Actions

1. **IF** a reactor scram sealed in,
   THEN ensure available CRD pumps operating. ☐ RO

2. **IF** a CRD pump **NOT** operating,
   THEN:

   a. **Ensure** C11(C12)-FC-R600 (CRD Flow Control) in MAN. ☐ RO

   b. **Close** the in-service C11(C12)-F002A(F002B) (Flow Control Vlv) ☐ RO

   c. **Start** one CRD pump ☐ RO

   d. **Adjust** C11(C12)-FC-R600 (CRD Flow Control) to greater than or equal to 30 gpm. ☐ RO

   e. **IF** available,
      THEN start the second CRD pump. ☐ RO

3. **IF NO** CRD pump can be started,
   THEN return to Section 2.1.3 Step 7. ☐ RO

2.5.3 Cooling Water Header Actions (continued)

4. IF a reactor scram NOT sealed in,
   THEN **maximize** cooling water header pressure:

   a. **Ensure** C11(C12)-FC-R600 (CRD Flow Control) in MAN and fully **open** the in service C11(C12)-F002A(F002B) (Flow Control Vlv)............ ☐ RO

   b. Fully **open** C11(C12)-PCV-F003 (Drive Pressure Vlv)............ ☐ RO

5. WHEN all control rods inserted to OR beyond Position 00 OR control rods NOT moving inward,
   THEN **return to** Section 2.1.3 Step 7............ ☐ RO

TERMINATION	
	Simulator Operator Actions
	When directed by the Lead Evaluator, delete the following commands: Malfunction - K2624A, ARI Reset Malfunction - K2625A, ARI INOP Malfunction – RP011F, ATWS 4 (Make sure RPS is reset and scram air header pressurized before deleting)
	When directed by the Lead Evaluator, place the simulator in FREEZE
	DO NOT RESET THE SIMULATOR PRIOR TO RECEIPT OF CONCURRENCE TO DO SO FROM THE LEAD EXAMINER

	Simulator Operator Role Play
	After Sim Operator has deleted SDV malfunction, Inform the CRS that a loose wire was found on ARI switch and it has been repaired.

Evaluator Notes	
Plant Response:	When actions are taken to control reactor water level during the ATWS after terminating and preventing, ARI will be repaired and rods can be inserted.
Objectives:	SRO - Directs actions for an ATWS. RO - Insert control rods IAW LEP-02.
Success Path:	Rods inserted with LEP-02, Alternate Rod Insertion.
Scenario Termination:	***When all rods are inserted and level is being controlled above TAF with injection established, the scenario may be terminated.***

Remind students not to erase any charts and not to discuss the scenario until told to do so by the evaluator/instructor.

TERMINATION

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Exit ATWS and enter RVCP when all rods are in.	
		Direct level restored to 170 – 200 inches after rods are all in.	
	RO	Confirms ARI reset when reported fixed.	
		Inserts a scram after discharge volume has drained for ~2 minutes.	***CRITICAL TASK #1***
		Reports all rods in.	
	BOP	Maintains reactor pressure as determined by the CRS.	
		Maintains level as directed by the SCO.	
		Restores level to 170 – 200 inches after all rod inserted. (Enclosure 4, page 65, contains actions for restart of Condensate and Feedwater)	

ENCLOSURE 1

ATTACHMENT 8

<< Crew Brief Template >>

Begin Brief	☐ Announce "Crew Brief" ☐ All crew members acknowledge announcement

Recap	**(As Required)**
	☐ Update the crew as needed:
	☐ Describe what happened and major actions taken
	☐ Procedures in-progress
	☐ Notifications:
	    ☐ Maintenance
	    ☐ Engineering
	    ☐ Others (Dispatcher, Station Management, etc.)
	☐ Future Direction and priorities
	☐ Discuss any contingency plans

Input	**(As Required)**
	☐ Solicit questions/concerns from each crew member:
	    ☐ ROs
	    ☐ CRS
	    ☐ STA
	☐ Are there any alarms unexpected for the plant conditions?
	☐ What is the status of Critical Parameters?

EAL	**(As Required)** ☐ Provide EAL and potential escalation criteria
Finish Brief	☐ Restore normal alarm announcement? (Yes/No) ☐ Announce "End of Brief"

ENCLOSURE 2

SECURING HPCI INJECTION

1.0 INITIAL CONDITIONS

1. **WHEN DIRECTED** BY 2EOP-01-LPC TO "TERMINATE AND PREVENT" HPCI INJECTION, **OR** ☐
2. **WHEN DIRECTED** BY 0EOP-01-RXFP TO "TERMINATE AND PREVENT" HPCI INJECTION, **OR** ☐
3. **WHEN** PERMISSION GIVEN BY THE UNIT CRS TO SECURE HPCI INJECTION WITH A HPCI AUTO START SIGNAL PRESENT. ☐

2.0 PROCEDURAL STEPS

1. **IF** HPCI IS **NOT** OPERATING, **PERFORM** THE FOLLOWING:
   a. **PLACE** HPCI AUXILIARY OIL PUMP CONTROL SWITCH IN PULL-TO-LOCK. ☐
2. **IF** HPCI IS OPERATING, **PERFORM** THE FOLLOWING:
   b. **DEPRESS AND HOLD** THE HPCI TURBINE TRIP PUSHBUTTON. ☐
   c. **WHEN** HPCI TURBINE SPEED IS 0 RPM, **AND** HPCI TURBINE CONTROL VALVE, E41-V9 IS CLOSED, **THEN PLACE** HPCI AUXILIARY OIL PUMP CONTROL SWITCH IN PULL-TO-LOCK. ☐
   d. **WHEN** HPCI TURB BRG OIL PRESS LO, A-01 4- 2, IS SEALED IN, **THEN RELEASE** THE HPCI TURBINE TRIP PUSHBUTTON. ☐
   e. **ENSURE** HPCI TURBINE STOP VALVE, E41-V8, **AND** HPCI TURBINE CONTROL VALVE, E41-V9, REMAIN CLOSED, **AND** HPCI DOES **NOT** RESTART. ☐

ENCLOSURE 3

Terminating and Preventing Injection From Condensate and Feedwater During EOP's (2OP-32)

1. **IF** desired **TRIP** all operating RFPs. ☐
2. **IF** one or more RFPs are in service **IDLE** one RFP as follows:
   a. **IF** two RFPs are operating **THEN TRIP** one. ☐
   b. **PERFORM** either of the following for the operating RFP:
      1. **PLACE** MAN/DFCS control switch to MAN. ☐
      2. **RAPIDLY REDUCE** speed to approximately 1000 rpm with the LOWER/RAISE speed control switch. ☐

      OR

      1. **PLACE** RFPT Speed Control in M (MANUAL) ☐
      2. **SELECT** DEM and **RAPIDLY REDUCE** speed to approximately 2550 rpm. ☐
3. **CLOSE** the following valves:
   – FW HTR 5A OUTLET VLVS, FW-V6 ☐
   – FW HTR 5B OUTLET VLVS, FW-V8 ☐

   OR

   – FW HTR 4A INLET VLV, FW-V118 ☐
   – FW HTR 4B INLET VLV, FW-V119 ☐
4. **ENSURE** the SULCV is closed by performing the following:
   a. **PLACE** SULCV, in M (Manual). ☐
   b. **SELECT** DEM and **DECREASE** signal until VALVE DEM indicates 0%. ☐
5. **ENSURE** FW-V120, is closed. ☐

ENCLOSURE 4

Aligning Condensate and Feedwater After Terminating and Preventing

1. **Ensure** FW-FV-177 (Feedwater Recirc to Condenser Vlv) CLOSED .... ☐
2. **Ensure** FW Control Mode Select in 1 ELEM .... ☐
3. **Ensure** at least one valve OPEN: .... ☐
   - B21-F032A (Feedwater Isol Vlv) .... ☐
   - B21-F032B (Feedwater Isol Vlv) .... ☐
4. **IF NO** RFP operating,
   THEN: .... ☐
   a. **Ensure** RFPT A(B) Sp Ctl:
      (1) In M (manual) .... ☐
      (2) Pmp A(B) Dem at 0.0 PCT .... ☐
   b. **Place** FW-FV-46(47) [RFP (A/B) Recirc Vlv] in OPEN .... ☐
   c. **Ensure:** .... ☐
      - FW-V3(V4) [RFP (A/B) Disch Vlv] OPEN .... ☐
      - RFP A(B) Manual/DFCS control switch in MANUAL .... ☐
   d. **Depress:** .... ☐
      (1) Reactor Water Level High Reset A .... ☐
      (2) Reactor Water Level High Reset B .... ☐
      (3) Reactor Water Level High Reset C .... ☐
      (4) RFP A(B) Reset .... ☐
   e. **Confirm** OPEN: .... ☐
      - RFP A(B) LP Stop Vlvs .... ☐
      - RFP A(B) HP Stop Vlvs .... ☐
   f. **Depress** RFP A(B) RFPT Start .... ☐
   g. **WHEN** at 1000 rpm,
      THEN raise RFP A(B) to at least 2550 rpm .... ☐

ENCLOSURE 4

Aligning Condensate and Feedwater After Terminating and Preventing (continued)

5. <u>IF</u> desired to transfer RFP A(B) to DFCS,
   <u>THEN</u>: ☐
   a. **Ensure** speed at least 2550 rpm ☐
   b. **Depress** DFCS Ctrl Reset ☐
   c. **Place** Manual/DFCS control switch in DFCS ☐
6. **Raise** RFP A(B) speed until discharge pressure approximately 100 psig above RPV pressure band ☐

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ENCLOSURE 4

Injection After Terminating and Preventing Condensate and Feedwater

1. **WHEN** RPV injection directed,
   THEN as needed: ☐
   - **Adjust** SULCV Valve Dem ☐
   - **Throttle** FW-V120 (FW Htrs 4&5 Byp Vlv) ☐

2. **WHEN** automatic control desired,
   THEN: ☐

   a. **Confirm** RPV level greater than +170 inches ☐

   b. **Ensure** FW-V120 (FW Htrs 4&5 Byp Vlv) CLOSED ☐

   c. **Open** FW-V10 (FW Recirc To Cond Isol Vlv) ☐

   d. **Adjust** SULCV to between 25 PCT and 55 PCT using: ☐
   - SULCV Valve Dem ☐
   - FW-FV-177 (Feedwater Recirc To Condenser Vlv) ☐

   e. **Ensure** Mstr RFPT Sp/Rx Lvl Ctl: ☐

   (1) In M (manual) ☐

   (2) Level Setpoint at current RPV level ☐

   f. **Place** SULCV in A (automatic) ☐

   g. **Adjust** as needed to control RPV level: ☐
   - Mstr RFPT Sp/Rx Lvl Ctl Level Setpoint ☐
   - FW-FV-177 (Feedwater Recirc To Condenser Vlv) ☐

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ENCLOSURE 5

ATTACHMENT 8A

Emergency Suppression Pool Cooling Using Loop A (2OP-17)

NOTE:	This attachment is **NOT** to be used for normal system operations.

START RHR SW A LOOP (CONV)

- OPEN *SW-V101* ☐
- CLOSE *SW-V143* ☐
- START CSW PUMPS AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN PLACE RHR SW BOOSTER PUMPS A & C LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068A* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR SW A LOOP (NUC)

- OPEN *SW-V105* ☐
- OPEN *SW-V102* ☐
- CLOSE *SW-V143* ☐
- START PUMPS ON NSW HDR AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN PLACE RHR SW BOOSTER PUMPS A & C LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068A* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR LOOP A

- IF LOCA SIGNAL IS PRESENT, THEN VERIFY COOLING LOGIC IS MADE UP ☐
- IF *E11-F015A* IS OPEN, THEN CLOSE *E11-F017A* ☐
- START LOOP A RHR PMP ☐
- OPEN *E11-F028A* ☐
- THROTTLE *E11-F024A* ☐
- THROTTLE *E11-F048A* ☐
- START ADDITIONAL LOOP A RHR PMP AND ADJUST FLOW AS NEEDED ☐

ENCLOSURE 5

ATTACHMENT 8B
Page 1 of 1
Emergency Suppression Pool Cooling Using Loop B (2OP-17)

> **NOTE:** This attachment is NOT to be used for normal system operations.

START RHR SW B LOOP (NUC)

- OPEN *SW-V105* ☐
- CLOSE *SW-V143* ☐
- START PMPS ON NSW HDR AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN PLACE RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068B* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR SW B LOOP (CONV)

- OPEN *SW-V101* ☐
- OPEN *SW-V102* ☐
- CLOSE *SW-V143* ☐
- START CSW PUMPS AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT THEN PLACE RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- START RHR SW PMP ☐
- ADJUST *E11-PDV-F068B* ☐
- ESTABLISH CLG WTR TO VITAL HDR ☐
- START ADDITIONAL RHR SW PUMP AND ADJUST FLOW AS NEEDED ☐

START RHR LOOP B

- IF LOCA SIGNAL IS PRESENT, THEN VERIFY COOLING LOGIC IS MADE UP ☐
- IF *E11-F015B* IS OPEN, THEN CLOSE *E11-F017B* ☐
- START LOOP B RHR PMP ☐
- OPEN *E11-F028B* ☐
- THROTTLE *E11-F024B* ☐
- THROTTLE *E11-F048B* ☐
- START ADDITIONAL LOOP B RHR PMP AND ADJUST FLOW AS NEEDED ☐

ENCLOSURE 6

HPCI INJECTION IN EOPs

1.	**IF** HPCI IS TRIPPED ON HIGH WATER LEVEL, **DEPRESS** HIGH WATER LEVEL SIGNAL RESET, E41-S25, PUSH BUTTON, **AND ENSURE** THE INDICATING LIGHT IS OFF.	☐
2.	**ENSURE** AUXILIARY OIL PUMP IS **NOT** RUNNING	☐
3.	**ENSURE** E41-V9 AND E41-V8 ARE CLOSED	☐
4.	**OPEN** E41-F059	☐
5.	**PLACE** HPCI FLOW CONTROL, E41-FIC-R600, IN MANUAL (M), **AND ADJUST** OUTPUT DEMAND TO APPROXIMATELY MIDSCALE, USING THE MANUAL LEVER.	☐
6.	**START** VACUUM PUMP **AND LEAVE** IN START	☐
7.	**OPEN** E41-F001	☐
8.	**START** AUXILIARY OIL PUMP **AND LEAVE** IN START	☐
9.	**OPEN** E41-F006, IMMEDIATELY AFTER E41-V8 HAS DUAL INDICATION	☐
10.	**ENSURE** E41-V9 AND E41-V8 ARE OPEN	☐
11.	**WHEN** SPEED STOPS INCREASING, **THEN ADJUST** SPEED TO APPROXIMATELY 2100 RPM	☐
12.	**ADJUST** HPCI FLOW CONTROL, E41-FIC-R600, TO OBTAIN DESIRED FLOW RATE	☐
13.	**ENSURE** E41-F012 IS CLOSED WHEN FLOW IS GREATER THAN 1400 GPM	☐
14.	**ADJUST** HPCI FLOW CONTROL, E41-FIC-R600, SETPOINT TO MATCH SYSTEM FLOW, AND THEN PLACE E41-FIC-R600 IN AUTO (A)	☐
15.	**ENSURE** E41-F025 AND E41-F026 ARE CLOSED	☐
16.	**START** SBGT (OP-10)	☐
17.	**ENSURE** BAROMETRIC CNDSR CONDENSATE PUMP IS OPERATING	☐

ATTACHMENT 1 - Scenario Quantitative Attribute Assessment

Category	NUREG 1021 Rev. 2 Supp. 1 Req.	Scenario Content
Total Malfunctions	5-8	7
Malfunctions after EOP Entry	1-2	2
Abnormal Events	2-4	2
Major Transients	1-2	1
EOPs Used	1-2	2
EOP Contingency	0-2	2
Run Time	60-90 min	90
Crew Critical Tasks	2-3	3
Tech Specs	2	2
Instrument / Component Failures before Major	2 – OATC 2 - BOP	4
Instrument / Component Failures after Major	2	2
Normal Operations	1	1
Reactivity manipulation	1	1

ATTACHMENT 2 – Shift Turnover

Brunswick Unit 2 Plant Status					
Station Duty Manager:	E. Neal			Workweek Manager:	B. Craig
Mode:	1	Rx Power:	100%	Gross*/Net MWe*:	977 / 951
Plant Risk: Current EOOS Risk Assessment is:				Green	
SFP Time to 200 Deg F:	49.7 hrs			Days Online:	82 days
Turnover:					
Protected Equipment:	2A FPC Pump/Hx, 2A RCC Pump, and 2C Demin Transfer Pump for Fuel Pool Decay Heat Removal and inventory makeup. 2A/B NSW Pumps due to 1A NSW pump maintenance.				
Comments:	1A NSW Pump is under clearance for planned maintenance. APRM 2 has failed downscale and is bypassed. 2C TCC Pump is in service on Unit One. The OATC reduce power to ~850 MWe Gross (reactivity plan is to use recirc flow) The BOP operator will then Isolate 230 kV Delco West (Line 30) IAW the marked up of 2OP-50, Section 6.2.6.				

6.2.6 De-energizing The 230 kV Switchyard

1. **Ensure** the Unit 2 230 kV switchyard is ENERGIZED. .... AD

2. **Ensure** the 4kV Auxiliary Electrical Systems are DE-ENERGIZED in accordance with Section 6.2.3. .... N-1 SRO

3. **Ensure** the SAT is DE-ENERGIZED in accordance with Section 6.2.4 .... N-1 SRO

4. **Ensure** Caswell Beach Pumping Station is DE-ENERGIZED in accordance with Section 6.2.5 .... N-1 SRO

5. **Ensure** required LCOs for Technical Specification Sections 3.8.1, 3.8.2, 3.8.7 and 3.8.8 are initiated.......................... SRO

6. **Obtain** Load Dispatcher's permission to de-energize the 230 kV switchyard. .... AD

   A Powers - the Delco West Line ONLY
   Person Contacted

7. **Place** Auto Reclose switches for the following PCBs in MAN:

   - 31B (Bus 2B Whiteville 230 kV Breaker).......................... N-1 SRO
   - 31A (Bus 2A Whiteville 230 kV Breaker).......................... N-1 SRO
   - 30B (Bus 2B Delco West Line 230 kV Breaker) .... ________
   - 30A (Bus 2A Delco West Line 230 kV Breaker) .... ________
   - 28B (Bus 2B Wallace 230 kV Breaker) .... N-1 SRO
   - 28A (Bus 2A Wallace 230 kV Breaker) .... N-1 SRO
   - 27B (Bus 2B Town Creek 230 kV Breaker).......................... N-1 SRO
   - 27A (Bus 2A Town Creek 230 kV Breaker).......................... N-1 SRO

6.2.6 De-energizing The 230 kV Switchyard (continued)

CAUTION

PCB Supervisory switch must be in LOCAL before the associated PCB is operated from Panel XU-5. ☐

8. **Place** Supervisory switches for the following PCBs in LOCAL:
   - 31B (Bus 2B Whiteville 230 kV Breaker) N-1 SRO
   - 30B (Bus 2B Delco West Line 230 kV Breaker) ________
   - 28B (Bus 2B Wallace 230 kV Breaker) N-1 SRO
   - 27B (Bus 2B Town Creek 230 kV Breaker) N-1 SRO
   - 31A (Bus 2A Whiteville 230 kV Breaker) N-1 SRO
   - 30A (Bus 2A Delco West Line 230 kV Breaker) ________
   - 28A (Bus 2A Wallace 230 kV Breaker) N-1 SRO
   - 27A (Bus 2A Town Creek 230 kV Breaker) N-1 SRO
9. **Open** 31B (Bus 2B Whiteville 230 kV PCB). N-1 SRO
10. **Confirm** 31B (Bus 2B Whiteville 230 kV PCB) is OPEN by observing the indicating lights. N-1 SRO
11. **Open** 31A (Bus 2A Whiteville 230 kV PCB). N-1 SRO
12. **Confirm** 31A (Bus 2A Whiteville 230 kV PCB) is OPEN by observing the indicating lights. N-1 SRO
13. **Open** 30B (Bus 2B Delco West Line 230 kV PCB). ________
14. **Confirm** 30B (Bus 2B Delco West Line 230 kV PCB) is OPEN by observing the indicating lights. ________
15. **Open** 30A (Bus 2A Delco West Line 230 kV PCB). ________
16. **Confirm** 30A (Bus 2A Delco West Line 230 kV PCB) is OPEN by observing the indicating lights. ________
17. **Open** 28B (Bus 2B Wallace 230 kV PCB). N-1 SRO

6.2.6 De-energizing The 230 kV Switchyard (continued)

18. **Confirm** 28B (Bus 2B Wallace 230 kV PCB) is OPEN by observing the indicating lights. .... N-1 SRO

19. **Open** 28A (Bus 2A Wallace 230 kV PCB). .... N-1 SRO

20. **Confirm** 28A (Bus 2A Wallace 230 kV PCB) is OPEN by observing the indicating lights. .... N-1 SRO

21. **Open** 27B (Bus 2B Town Creek 230 kV PCB). .... N-1 SRO

22. **Confirm** 27B (Bus 2B Town Creek 230 kV PCB) is OPEN by observing the indicating lights. .... N-1 SRO

23. **Open** 27A (Bus 2A Town Creek 230 kV PCB). .... N-1 SRO

24. **Confirm** 27A (Bus 2A Town Creek 230 kV PCB) is OPEN by observing the indicating lights. .... N-1 SRO

NOTE

If work is to be performed on a 230 kV bus, the manual disconnects are to be opened. .... ☐

25. **Place** Supervisory switches for the following PCBs in REMOTE:
    - 31B (Bus 2B Whiteville 230 kV Breaker) .... N-1 SRO
    - 30B (Bus 2B Delco West Line 230 kV Breaker) .... ______
    - 28B (Bus 2B Wallace 230 kV Breaker) .... N-1 SRO
    - 27B (Bus 2B Town Creek 230 kV Breaker .... N-1 SRO
    - 31A (Bus 2A Whiteville 230 kV Breaker) .... N-1 SRO
    - 30A (Bus 2A Delco West Line 230 kV Breaker) .... ______
    - 28A (Bus 2A Wallace 230 kV Breaker) .... N-1 SRO
    - 27A (Bus 2A Town Creek 230 kV Breaker) .... N-1 SRO

6.2.6 De-energizing The 230 kV Switchyard (continued)

Date/Time Completed ____________

Performed By (Print) Initials

______________________ ________

______________________ ________

______________________ ________

______________________ ________

Reviewed By ______________________________

Unit CRS/SRO

N-1, Partial usage to isolate only the Delco West 230 kV Line (Line 30)

DUKE ENERGY

BRUNSWICK TRAINING SECTION

OPERATIONS TRAINING

INITIAL LICENSED OPERATOR

SIMULATOR EVALUATION GUIDE

2016 NRC SCENARIO 3

PT-40.2.11, DWEDT FAILURE, VFD CELL BYPASS, NSW PUMP TRIP, CWIP PUMP TRIP, RWCU LEAK, SBGT START FAILURE, ED, ADS VLV FAILURE

REVISION 0

Developer: Bob Bolin	**Date:** 07/07/2016
Technical Review: Dan Hulgin	**Date:** 9/12/2016
Validators: Dwayne Wolf Shawn Zander Grant Newton	**Date:** 09/07/16
Facility Representative: Craig Oliver	**Date:** 09/22/2016

REVISION SUMMARY	
0	Scenario developed for 2016 NRC Exam.

TABLE OF CONTENTS

1.0 SCENARIO OUTLINE

Event	Malf. No.	Type*	Event Description
1		N-BOP	Perform PT-40.2.11, Main Generator Voltage Regulator Manual And Automatic Operational Check
2	ZA411	C-ATC C-CRS	DWEDT Pump failure
3	RC053F	C-ATC C-CRS	VFD Cell Failure (TS)(AOP)
4		R-ATC	Power maneuver
5	CW019F	C-BOP C-CRS	NSW Pump 2B Trip (failure of standby to start) (TS)(AOP)
6	CW039F	C-BOP C-CRS	CWIP Trip (AOP)
7	RW013F	M C	RWCU leak / Scram SBGT Fails to start (AOP)(RSP)(SCCP)
8	K1507A	M C	ED Failure of 2 ADS valves to open (EDP)
*(N)ormal, (R)eactivity, (C)omponent or Instrument, (M)ajor			

2.0 SCENARIO DESCRIPTION SUMMARY

Event	Description
1	Perform PT-40.2.11, Main Generator Voltage Regulator Manual And Automatic Operational Check.
2	Annunciator A-04 1-1, Drywell Equip Drain Sump Lvl Hi, will annunciate and the sumps will not auto start. One of the sump pumps will need to be manually started
3	A power cell in VFD A will fail. Recirc Pump 2A speed will lower and a speed hold will initiate. Loop flows will be outside mismatch limits.
4	The crew will reset the speed hold and match loop flows.
5	NSW Pump B will trip and the crew will start NSW Pump A. Since 1A NSW Pump is out of service, Tech Specs will apply. Crew will enter 0AOP-18.0, Nuclear Service Water System failure, and carry out appropriate actions.
6	Circulating Water Pump 2A will trip on motor winding fault, and the standby Circulating Water Intake Pump will be started. 0AOP-37.0 will be entered due to lowering vacuum.
7	A large un-isolable RWCU leak will occur. Crew will enter AOP-5.0 and SCCP. The CRS should direct a SCRAM. SBGT train A will fail to auto start and should be manually started.
8	Secondary containment conditions will worsen, forcing the CRS to direct an Emergency Depressurization (or Anticipation of Emergency Depressurization) due to high water levels. If Anticipation is performed, the second area high water level will annunciate requiring the emergency depressurization. Two ADS SRV's will fail to manually open. The CRS should direct opening two additional SRV's.

3.0 CREW CRITICAL TASKS

Critical Task #1
Insert a reactor scram prior to any area reaching its Max Safe Operating Value
Critical Task #2
Perform Emergency Depressurization when two plant areas exceed max safe operating water level.

4.0 TERMINATION CRITERIA

When emergency depressurization has been performed and the reactor has been depressurized to <100 psig the scenario may be terminated.

5.0 IMPLEMENTING REFERENCES

NOTE: Refer to the most current revision of each Implementing Reference.

Number	Title
A-04, 1-1	DRYWELL EQUIP DRAIN SUMP LVL HI
A-06, 3-1	RECIRC VFD A ALARM UNACK
A-06, 4-5	RECIRC LOOP A ONLY OUT OF SERV
0AOP-04.0	Low Core Flow
2OP-02, 6.1.3 6.2.1 6.3.4	Reactor Recirculation System Operating Procedure
UA-01, 1-10	NUCLEAR HEADER SERV WTR PRESS-LOW
UA-01, 4-10	NUCLEAR HDR SW PUMP B TRIP
UA-18, 6-1	BUS E4 4KV MOTOR OVLD.
0AOP-18.0	NUCLEAR SERVICE WATER SYSTEM FAILURES
UA-01, 1-7	CIRC WATER PUMP A TRIP
0AOP-37.0	Low Condenser Vacuum
UA-03, 2-7	AREA RAD RX BLDG HIGH
0AOP-05.0	Radioactive Spills, High Radiation, And Airborne Activity
UA-5, 4-6	SBGT SYS A FAILURE
	, ,

6.0 SETUP INSTRUCTIONS

1. **PERFORM** TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 5, Checklist for Simulator Exam Security.
2. **RESET** the Simulator to IC-11.
3. **ENSURE** the RWM is set up as required for the selected IC.
4. **ENSURE** appropriate keys have blanks in switches.
5. **RESET** alarms on SJAE, MSL, and RWM NUMACs.
6. **ENSURE** no rods are bypassed in the RWM.
7. **PLACE** all SPDS displays to the Critical Plant Variable display (#100).
8. **ENSURE** hard cards and flow charts are cleaned up
9. **TAKE** the SIMULATOR OUT OF FREEZE
10. **LOAD** Scenario File.
11. **ALIGN** the plant as follows:

Manipulation
Ensure 2C TCC pump is in service on Unit One. Loaded in Scenario File
Ensure 2B NSW pump is running, 2A in standby

12. **IF desired**, take a **SNAPSHOT** and save into an available IC for later use.
13. **PLACE** a clearance on the following equipment.

Component	Position

14. **INSTALL** Protected Equipment signage and **UPDATE** RTGB placard as follows:

Protected Equipment
1. 2A and 2B NSW pumps 2. 2A FPC Pump/Hx, 2A RCC Pump, and 2C Demin Transfer Pump.

15. **VERIFY** 0ENP 24.5 Form 2 (Immediate Power Reduction Form) for IC-11 is in place.

16. **ENSURE** each Implementing References listed in Section 7 is intact and free of marks.

17. **ENSURE** all materials in the table below are in place and marked-up to the step identified.

Required Materials
0PT-40.2.11

18. **ADVANCE** the recorders to prevent examinees from seeing relevant scenario details.

19. **PROVIDE** Shift Briefing sheet for the CRS.

20. **VERIFY** all actions contained in TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 4, Simulator Training Instructor Checklist, are complete.

7.0 INTERVENTIONS

TRIGGERS

Trig	Type	ID
1	Annunciator	ZA411 - [DRYWELL EQUIP DRAIN SUMP LVL HI]
2	Trigger Command	did:k2115a
3	Trigger Command	did:k2116a
4	DI Override	K2721K - [VFD A LOWER FAST]
4	Malfunction	RC053F - [VFD A POWER CELL COMMUNICATION FAILURE]
5	Malfunction	CW019F - [NUC SERVICE WATER PUMP MOTOR WINDING FAULT]
6	Malfunction	CW039F - [CIRC WATER INTAKE PUMP MOTOR WINDING FAULT]
7	Malfunction	RW013F - [RWCU BRK IN TRIANGLE ROOM 77`]
9	Remote Function	RW_ZVRW004M - [G31-F004 OUTBOARD ISOLATION VALVE]
10	Annunciator	ZUA1214 - [SOUTH RHR RM FLOOD LEVEL HI-HI]

Trig #	Trigger Text
2	K2115JBU - [DRYWELL EQUIP DR PUMP A]
3	K2116JBU - [DRYWELL EQUIP DR PUMP B]
9	K1410JCK - [RWCU VLV G31-F004]
11	K6101WOV - [SBGT SYS A]

MALFUNCTIONS

Malf ID	Mult ID	Description	Current Value	Target Value	Rmp time	Actime	Dactime	Trig
RW015F		G31-F001 FAILURE TO AUTO CLOSE	True	True				
RW016F		G31-F004 FAILURE TO AUTO CLOSE	True	True				
RC053F	CELL B1	VFD A POWER CELL COMMUNICATION FAILURE	False	True		00:00:01		4
CW019F	B	NUC SERVICE WATER PUMP MOTOR WINDING FAULT	False	True				5
CW039F	A	CIRC WATER INTAKE PUMP MOTOR WINDING FAULT	False	True				6
RW013F		RWCU BRK IN TRIANGLE ROOM 77`	0.00	100.00	00:10:00			7
RW017F	G31-F001	REACTOR WTR CLEANUP * VLV G31-F001	True	True				

REMOTES

Remf Id	Mult Id	Description	Current Value	Target Value	Rmp time	Actime	Trig
CC_IACW4518		2C TBCCW PUMP UNIT ALIGNMENT	1	1			
RW_ZVRW004M		G31-F004 OUTBOARD ISOLATION VALVE	ON	OFF			9

PANEL OVERRIDES

Tag ID	Description	Position / Target	Actual Value	Override Value	Rmp time	Actime	Dactime	Trig
K6101B	SBGT SYS A	PREF	ON	OFF				
Q6101ARV	SBGT SYS A CONT PREF R 4	ON/OFF	OFF	ON				
K2115A	DRYWELL EQUIP DR PUMP A	OUT	OFF	ON				
K2115A	DRYWELL EQUIP DR PUMP A	NORM	ON	OFF				
K2116A	DRYWELL EQUIP DR PUMP B	OUT	OFF	ON				
K2116A	DRYWELL EQUIP DR PUMP B	NORM	ON	OFF				
K1505A	AUTO DEPRESS VLV B21-F013D	OPEN	OFF	OFF				
K1511A	AUTO DEPRESS VLV B21-F013A	OPEN	OFF	OFF				
K4B20A	NUC HDR SW PMP A DISCH VLV	AUTO	ON	OFF				
K2721K	VFD A LOWER FAST	LOWER FAST	OFF	ON			00:00:01	4
Q2721LWF	VFD A LOWER FAST	ON/OFF	OFF	OFF				

ANNUNCIATORS

Window	Description	Tagname	Override Type	OVal	AVal	Actime	Dactime	Trig
3-1	DRYWELL EQUIP DRAIN SUMP LEAK HI	ZA431	OFF	OFF	OFF			
1-1	DRYWELL EQUIP DRAIN SUMP LVL HI	ZA411	ON	ON	OFF			1
1-4	SOUTH RHR RM FLOOD LEVEL HI-HI	ZUA1214	ON	ON	OFF			10

8.0 OPERATOR RESPONSE AND INSTRUCTIONAL STRATEGIES

EVENT 1: PT-40.2.11	
Simulator Operator Actions	
	Ensure Monitored Parameters is open and Scenario Based Testing Variables are loaded.

Simulator Operator Role Play	
	Acknowledge any requests for the Load Dispatcher.
	When asked the voltage regulator operation was smooth and in the same direction of of the rheostat.

Evaluator Notes	
Plant Response:	
Objectives:	SRO - Directs BOP to perform PT-40.2.11 BOP – Performs PT-40.2.11 RO – Monitor Balance of Plant
Success Path:	PT-40.2.11 is completed.
Event Termination:	When directed by the Lead Evaluator, go to Event 2.

EVENT 1: PT-40.2.11

Time	Pos	EXPECTED Operator Response	NOTES
	SRO	Conduct shift turnover shift briefing.	
		Direct performance of PT-40.2.11	
		May conduct a brief (see Enclosure 1 on page 45 for format)	
	RO	Monitors the plant	
	BOP	Performs PT-40.2.11 See attached procedure.	

1.0 PURPOSE

The purpose of this test is to demonstrate the OPERABILITY of the voltage regulator transfer circuitry and exercise the regulator potentiometers.

2.0 SCOPE

1. This test is performed once every 92 days and demonstrates OPERABILITY of voltage regulator transfer circuitry and exercises the regulator potentiometers.

2. This test may also be used to demonstrate proper operation of the voltage regulator potentiometer and transfer circuitry, after completion of maintenance.

3.0 PRECAUTIONS AND LIMITATIONS

1. Main generator loading is within the limits of the Generator Reactive Capability Curve shown on Attachment 1, Estimated Capability Curve, and with a minimum of 20 MVAR (positive). .... ☐

2. This test is **NOT** performed if erratic operation of the voltage regulator is noted immediately prior to the performance of this test. .... ☐

3. The Load Dispatcher is to be informed when the main generator automatic voltage regulator is **NOT** in service. Log entries are made documenting the notification. {9.1.1} .... ☐

4.0 GENERAL INFORMATION

None

5.0 ACCEPTANCE CRITERIA

1. This test may be considered satisfactory when the following criteria are met:

   a. DC regulator output variation is smooth and in the same direction as the rheostat movement.

   b. AC regulator output variation is smooth and in the same direction as the rheostat movement.

6.0 PREREQUISITES

1. **Confirm** Generator and Exciter System in operation in accordance with 1(2)OP-27, Generator and Exciter System Operating Procedure .... _______

2. **Confirm** Plant Electrical System in operation in accordance with 1(2)OP-50, Plant Electric System Operating Procedure .... _______

3. **Confirm** DC Electrical System in operation in accordance with 1(2)OP-51, DC Electrical System Operating Procedure .... _______

4. **Confirm** 120 Volt AC UPS, Emergency, and Conventional Electrical Systems in operation in accordance with 1(2)OP-52, 120 Volt AC UPS, Emergency, and Conventional Electrical Systems Operating Procedure .... _______

5. **Confirm** NO system load changes are anticipated .... _______

7.0 INSTRUCTIONS

7.1 General

1. **Obtain** permission from Unit CRS to perform this test .... _______
2. **Ensure** all Prerequisites listed in Section 5.0 are met .... _______

7.2 Operate 70CS (Gen Manual Volt Adj Rheo)

1. **Ensure** 43CS (Regulator Mode Selector) in AUTO .... _______
2. **Station** an operator at the Excitation Regulator and Control cubicle in the Turbine Building on the 70 ft elevation west to monitor regulator output during the following steps .... _______

NOTE

- Section 7.2 Step 3 and Section 7.2 Step 4 are repeated as necessary to ensure proper operation/indication of the manual rheostat .... ☐
- DC regulator output is locally monitored using D1VM (D.C. Reg. Output)........................ ☐

3. **Raise** 70CS (Gen Manual Volt Adj Rheo) until the Upper Limit light comes ON .... _______

NOTE

The Intermed light will come ON during lowering of 70CS (Gen Manual Volt Adj Rheo) and will remain ON after the Low Limit light is ON .... ☐

4. **Lower** 70CS (Gen Manual Volt Adj Rheo) until the Low Limit light comes ON .... _______
5. Using 70CS (Gen Manual Volt Adj Rheo) on the RTGB, **null** Gen Volt Reg Diff Volt meter .... _______
6. **IF** D1VM (D.C. Reg. Output) variation was **NOT** smooth **AND** in the same direction as rheostat movement,
THEN go to Section 7.3 Step 7 .... _______

7.2 Operate 70CS (Gen Manual Volt Adj Rheo) (continued)

7. **IF** D1VM (D.C. Reg. Output) variation was smooth **AND** in the same direction as rheostat movement,
THEN perform the following: {9.1.1}

   a. **Notify** the Load Dispatcher the main generator voltage regulator is being placed in MANUAL. .... _______

   ____
   Person Notified

   b. **Document** the Load Dispatcher notification in the log. .... _______

   c. **Place** 43CS (Regulator Mode Selector) in MAN. .... _______

7.3 Operate 90CS (Gen Auto Volt Adj Rheo)

NOTE

- Section 7.3 Step 1 and Section 7.3 Step 2 may be repeated as necessary to ensure proper operation/indication of the automatic rheostat. ☐
- AC regulator output may be locally monitored using A1VM (A.C. Reg. Output). ☐

1. **Raise** 90CS (Gen Auto Volt Adj Rheo) until the Upper Limit light comes ON. _______
2. **Lower** 90CS (Gen Auto Volt Adj Rheo) until the Low Limit light comes ON. _______
3. **Null** Gen Volt Reg Diff Volt meter on the RTGB using 90CS (Gen Auto Volt Adj Rheo). _______
4. **IF** A1VM (A.C. Reg. Output) variation was **NOT** smooth **AND** in the same direction as rheostat movement, **THEN go to** Section 7.3 Step 6. _______
5. **IF** A1VM (A.C. Reg. Output) variation was smooth **AND** in the same direction as rheostat movement, **THEN perform** the following: {9.1.1}
   a. **Place** 43CS (Regulator Mode Selector) in AUTO. _______
   b. **Notify** the Load Dispatcher the main generator voltage regulator is in AUTOMATIC. _______

      ____
      Person Notified

   c. **Document** Load Dispatcher notification in the log. _______
6. **IF** extended manual voltage regulator operation becomes necessary, **THEN coordinate** with the Load Dispatcher to maintain minimum generator MVAR load and generator voltage in accordance with the System Operation section of 1(2)OP-27, Generator and Exciter System Operating Procedure. _______
7. **IF** either regulator output variation was **NOT** smooth **AND** in the same direction as the rheostat, **THEN prepare** a W/R for the regulator. _______

MAIN GENERATOR VOLTAGE REGULATOR MANUAL AND AUTOMATIC OPERATIONAL CHECK	0PT-40.2.11
	Rev. 6
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7.4 Restoration

1. **Perform** review of completed procedure sections to verify Section 5.0, Acceptance Criteria, for tests performed, have been met. .... _______

   ____
   IV

2. **IF** Acceptance Criteria is **NOT** met, **THEN perform** following:

   a. **Report** any equipment found INOPERABLE or **NOT** meeting Acceptance Criteria to Supervisor. .... _______

   b. **Ensure** CR has been initiated. .... _______

3. **Ensure** required information has been recorded on Attachment 2, Certification and Review Form. .... _______

4. **Notify** Unit CRS when this procedure is complete or found to be unsatisfactory. .... _______

EVENT 2: DWEDT PUMP FAILURE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 1** to activate the DWED Sump Lvl Hi Annunciator.
NOTE	If the simulator is left in run the DWED Sump Lvl Hi Alarm will annunciate on its own after approximately 50 minutes. (The malfunctions will still work if it is allowed to annunciate)
	When either sump pump has been running for ~30 seconds delete malfunction for the DWED Sump Lvl Hi Annunciator.

Simulator Operator Role Play	
	Acknowledge requests as I&C for troubleshooting DWED Sump Pump auto start failure.
	If asked, the last time the sumps were pump was ~4 hours ago.

Evaluator Notes	
Plant Response:	Annunciator A-04 (1-1), Drywell Equip Drain Sump Lvl Hi.
Objectives:	RO - Pump the DWEDT
Success Path:	Pumps the DWEDT.
Event Termination:	Go to Event 3 at the direction of the Lead Evaluator.

EVENT 2: DWEDT PUMP FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of APPs Direct RO to start DWEDS Pump, if asked. Contact I/C for troubleshooting the failure of the DWEDS to auto start.	
	RO	Refer to APP: A-04 (1-1), Drywell Equip Drain Sump Lvl Hi	
		Diagnose failure of DWEDS Pump	
		Start a DWEDS Pump (may use 0OP-47 Section 5.3.5) Verifies pump shuts off after a period of time.	
	BOP	Monitors the plant	

5.3.5 Manually Pumping Drywell Floor Or Equipment Drain Sumps

1. **Ensure** the following:

   a. Drywell Floor or Equipment Drain sump needs to be manually pumped to determine in-leakage rates .... ________

   OR

   b. Drywell Floor or Equipment Drain sump needs to be manually pumped as determined by the Unit CRS. .... ________

2. On Panel P603, **place** control switches for the applicable sump pump(s) in START **AND** then in AUTO:

   - G16-C001A (Drywell Floor Drain Pump 1(2)A) .... ________
   - G16-C001B (Drywell Floor Drain Pump 1(2)B) .... ________
   - G16-C006A (Drywell Equip Drain Pump 6A)................................ ________
   - G16-C006B (Drywell Equip Drain Pump 6B)................................ ________

Date/Time Completed ____________________

Performed By (Print) Initials

__________________________ _____

__________________________ _____

__________________________ _____

__________________________ _____

Reviewed By ________________________
Unit CRS/SRO

EVENT 3/4: VFD A CELL FAILURE / MANEUVER POWER	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 4** to activate VFD A Cell failure.

Simulator Operator Role Play	
	If contacted as I&C to investigate, acknowledge the request.
	If asked as Reactor Engineer for guidance on restoring Loop flow limits, ask the CRS for their recommendations, then concur with that recommendation.

Evaluator Notes	
Plant Response:	A power cell in VFD A will fail. Recirc Pump 2B speed will lower and a speed hold will initiate. Loop flows will be outside mismatch limits. The crew will respond per AOP-04.0, reset the speed hold and match loop flows or lower the speed of 2B to get within Tech Spec limits.
Objectives:	SRO - Direct Shift Response To A Recirculation Flow Control Failure Causing A Decreasing Flow Per AOP-04.0 RO - Respond To A Recirc Flow Control Failure Decreasing Per AOP-04.0
Success Path:	Reset the speed hold condition and match recirc loop flows.
Event Termination:	Go to Event 5 at the direction of the Lead Evaluator.

EVENT 3/4: VFD A CELL FAILURE / MANEUVER POWER

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct entry into AOP-04.0	
		With recirculation loops flows mismatched, enter LCO 3.4.1 Condition A. NOTE: May balance loops and not enter Tech. Specs. Question examinee about Tech Spec actions if not entered. <u>TS 3.4.1 Condition A.1.</u> Satisfy the requirements of the LCO within 6 hours by restoring matched flows or impose limits specified by the LCO. NOTE: Declare the loop with lower flow not in operation.	
		Direct speed hold reset on VFD A	
		Direct loop flow mismatch restored to within limit	
		Direct I&C to investigate cell failure	
		May conduct a brief (see Enclosure 1 on page 45 for format)	
	BOP	Monitors the plant.	
		Determine cause to be cell failure at HMI	

EVENT 3/4: VFD A CELL FAILURE / MANEUVER POWER			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	RO	Reference applicable APPs: A-06, 3-1, Recirc VFD A Alarm Unack A-06, 4-5, Recirc Loop A Only Out Of Serv	
		Recognize/report lowering Recirc A speed/speed hold	
		Enter/announce 2AOP-04.0, Low Core Flow	
		Determine Loop flow outside mismatch limits Core flow >57.5 Mlbs, Jet Pump flows must be within 3 Mlbs.	
		Reset speed hold on VFD A IAW 2OP-02 Section 6.3.4. (see page 26)	
		Restore loop flows to within limits as directed by CRS. Lower the B Recirc Pump Speed IAW 2OP-02 Section 6.2.1. (see page 27) Raise the A Recirc Pump Speed IAW 2OP-02 Section 6.1.3. (see page 28)	

6.3.4 Recovery From Recirc VFD Speed Hold Condition

1. **Confirm** Recirc VFD A(B) Speed Hold yellow light ON at Panel P603. ________
2. **Ensure** the cause of the Speed Hold condition has been identified ________
3. **Ensure** Plant conditions have stabilized ________
4. **Check** the following parameters are approximately the same:
   - Recirc Pump A(B) Speed Demand ________
   - Recirc Pump A(B) Actual Speed ________
   - Recirc Pump A(B) Calculated Speed ________
5. **Depress** Recirc VFD A(B) SP Hold Reset to reset the speed hold condition ________
6. **Confirm** Recirc VFD A(B) Speed Hold yellow status light is OFF ________
7. **Check** flow conditions stable ________

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

8. **Adjust** Recirc VFD speed and Recirc flow as directed by the Unit CRS ________

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

Date/Time Completed ________________

Performed By (Print)	Initials
________________	____
________________	____
________________	____
________________	____

Reviewed By: ________________
Unit CRS/SRO

6.2 Shutdown

6.2.1 Lowering Speed/Power Using Individual Recirculation Pump Control Or Recirc Master Control

1. **Confirm** reactor recirculation pump in operation in accordance with Section 6.1.2. ____________

NOTE

- Recirculation Pump speed changes are performed when directed by 0GP-05, Unit Shutdown, and 0GP-12, Power Changes. Other operating procedures are used simultaneously with this procedure as directed by 0GP-05, Unit Shutdown, and 0GP-12, Power Changes. ☐
- Speed changes are accomplished by depressing Lower Slow, Lower Medium, or Lower Fast pushbuttons. The Lower Slow pushbutton changes Recirc pump speed at 0.06%/decrement at 1 rpm/second. The Lower Medium pushbutton changes Recirc pump speed at 0.28%/decrement at 5 rpm/second. The Lower Fast pushbutton changes Recirc pump speed at 2.8%/decrement at 100 rpm/second. ☐

2. **IF AT ANY TIME** any of the following conditions exist, **THEN enter** 1AOP-04.0, Low Core Flow.{8.1.9} ____________

   - Entry into Region A of Power to Flow Map
   - OPRM INOPERABLE **AND** any of the following
     - ◊ Entry into Region B of Power to Flow Map
     - ◊ Entry into 5% Buffer Region of Power to Flow Map
     - ◊ Entry into OPRM Enabled Region and indications of THI (Thermal Hydraulic Instability) exist

6.2.1 Lowering Speed/Power Using Individual Recirculation Pump Control Or Recirc Master Control (continued)

CAUTION

- The OPRM System monitors LPRMs for indication of thermal hydraulic instability (THI). When greater than or equal to 25% power and less than or equal to 60% recirculation flow, alarms and automatic trips are initiated upon detection of THI. Pump operations are governed by the limits of the applicable Power Flow Map, as specified in the COLR. {8.1.9} .... ☐
- Entry into the 5% Buffer Region warrants increased monitoring of reactor instrumentation for signs of Thermal Hydraulic Instability. Time in the 5% Buffer Region presents additional risk and is minimized.{8.1.9} .... ☐
- With core flow less than 57.5 x 10^6 lbs/hr, jet pump loop flows are required within 10% (maximum indicated difference 6.0 x 10^6 lbs/hr). With core flow greater than or equal to 57.5 x 10^6 lbs/hr, jet pump loop flows are required within 5% (maximum indicated difference 3.0 x 10^6 lbs/hr). .... ☐
- When Recirc Pump speeds are less than 40%, decreasing speed using a Lower Fast pushbutton can result in a Speed Hold condition due to exceeding the regen torque limit. .... ☐

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

3. IF desired to lower the speed of both recirculation pumps simultaneously,
   THEN depress Recirc Master Control Lower (Slow Medium Fast) pushbutton. .... ______

4. IF desired to lower the speed of an individual recirculation pump,
   THEN depress the Recirc VFD A(B) Lower (Slow Medium Fast) pushbutton. .... ______

6.2.1 Lowering Speed/Power Using Individual Recirculation Pump Control Or Recirc Master Control (continued)

5. **Confirm** the following, as applicable:

- Recirc Pump A(B) Speed Demand, Calculated Speed, and Actual Speed have lowered .... ______
- Reactor power lowers .... ______
- B32-R617(R613) [Recirc Pump A(B) Discharge Flow] lowers .... ______
- B32-VFD-IDS-003A(B) [Recirc VFD 2A(B) Output Wattmeter] lowers .... ______
- B32-VFD-IDS-001A(B)]Recirc VFD 2A(B) Output Frequency Meter] lowers .... ______

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

Date/Time Completed ______________

Performed By (Print)	Initials
______________	______
______________	______
______________	______
______________	______

Reviewed By: ______________
Unit CRS/SRO

6.1.3 Raising Speed/Power Using Individual Recirculation Pump Control or Recirc Master Control

1. **Ensure** the following Initial Conditions are met:

   a. Reactor Recirculation Pumps in operation in accordance with Section 6.1.2. .... ______

   b. Recirculation Pump flow limits are CLEAR .... ______

NOTE

- Recirculation Pump speed changes are performed when directed by 0GP-04, Increasing Turbine Load to Rated Power, and 0GP-12, Power Changes. Other operating procedures are used simultaneously with this procedure as directed by 0GP-04, Increasing Turbine Load to Rated Power, 0GP-12, Power Changes, or the Unit CRS. .... ☐
- Speed changes are accomplished by depressing Raise Slow or Raise Medium pushbuttons. The Raise Slow pushbutton changes Recirc pump speed at 0.06%/increment at 1 rpm/second. The Raise Medium pushbutton changes Recirc pump speed at 0.28%/increment at 5 rpm/second. .... ☐

CAUTION

The OPRM System monitors LPRMs for indication of thermal hydraulic instability (THI). When greater than or equal to 25% power and less than or equal to 60% recirculation flow, alarms and automatic trips are initiated upon detection of THI. Pump operations are governed by the limits of the applicable Power Flow Map, as specified in the COLR. {8.1.9} .... ☐

2. **IF AT ANY TIME** any of the following conditions exist, **THEN enter** 2AOP-04.0, Low Core Flow. {8.1.9} .... ______

   - Entry into Region A of Power to Flow Map
   - OPRM INOPERABLE **AND** any of the following
     - ◊ Entry into Region B of Power to Flow Map
     - ◊ Entry into 5% Buffer Region of Power to Flow Map
     - ◊ Entry into OPRM Enabled Region and indications of THI (Thermal Hydraulic Instability) exist

6.1.3 Raising Speed/Power Using Individual Recirculation Pump Control or Recirc Master Control (continued)

CAUTION

- The OPRM System monitors LPRMs for indication of thermal hydraulic instability (THI). When greater than or equal to 25% power and less than or equal to 60% recirculation flow, alarms and automatic trips are initiated upon detection of THI. Pump operations is be within the limits of the applicable Power-Flow Map, as specified in the COLR. The Scram Avoidance Region is avoided. {8.1.9} .... ☐
- With core flow less than 57.5 x 10^6 lbs/hr, jet pump loop flows are required within 10% (maximum indicated difference 6.0 x 10^6 lbs/hr). With core flow greater than or equal to 57.5 x 10^6 lbs/hr, jet pump loop flows are required within 5% (maximum indicated difference 3.0 x 10^6 lbs/hr). .... ☐
- If total reactor feedwater flow lowers to less than 16.4% of rated flow, Speed Limiter Number 1 will cause the Recirculation Pumps to run back to 34% speed. This signal must be manually reset in accordance with Section 6.3.3. .... ☐
- When total core flow is greater than 43 mlb/hr, Speed Limiter Number 2 will cause a runback to approximately 48% speed if reactor water level is less than 182 inches and either reactor feed pump A or B suction flow is less than 14.9% of individual RFP rated suction flow. This signal must be manually reset using Section 6.3.3. .... ☐

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

3. <u>IF</u> desired to raise the speed of both Recirc Pumps simultaneously, as directed by the Unit CRS,
<u>**THEN**</u> **depress** Recirc Master Control Raise Slow or Raise Medium pushbutton. .... ________

4. <u>IF</u> desired to raise the speed of an individual Recirc Pump, as directed by the Unit CRS,
<u>**THEN**</u> **depress** the VFD A(B) Raise Slow or Raise Medium pushbutton for the Recirc Pump. .... ________

5. **Confirm** the following, as applicable:

   - A rise in Recirc Pump A(B) Speed Demand, Calculated Speed, and a rise in Actual Speed. .... ________
   - A rise in Reactor power. .... ________
   - A rise in B32-R617(R613) [Recirc Pump A(B) Discharge Flow] .... ________

6.1.3 Raising Speed/Power Using Individual Recirculation Pump Control or Recirc Master Control (continued)

- A rise in B32-VFD-IDS-003A(B) [Recirc VFD 2A(B) Output Wattmeter] .... ______
- A rise in B32-VFD-IDS-001A(B) [Recirc VFD 2A(B) Output Frequency Meter] .... ______

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

Date/Time Completed ______________

Performed By (Print) Initials

______________________ ______

______________________ ______

______________________ ______

______________________ ______

Reviewed By: ______________________
Unit CRS/SRO

EVENT 5: NSW PUMP B TRIP (FAILURE OF STANDBY TO START)	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 5** to trip the 2B NSW Pump.

Simulator Operator Role Play	
	If contacted as OAO to investigate NSW pump and breaker, After the pump has tripped report 51 devices on all three phases are tripped at the breaker on E4
	If contacted as I&C to investigate, acknowledge the request.

Evaluator Notes	
Plant Response:	The running NSW pump will TRIP on motor overload. The STBY NSW pump will fail to AUTO start. The BOP operator should recognize the failure and manually start the STBY NSW pump. With a U1 NSW pump under clearance will require entry into TS.
Objectives:	SRO - Direct actions for loss of NSW Determine actions required for LCO per Technical Specifications RO - Respond to the failure of an automatic start of the A NSW pump
Success Path:	Determine TS required actions and Start 2A NSW Pump.
Event Termination:	Go to Event 6 at the direction of the Lead Evaluator.

EVENT 5: NSW PUMP B TRIP (FAILURE OF STANDBY TO START)

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct entry into 0AOP-18.0, NSW System Failure.	
		Contact maintenance to investigate trip of 2B NSW Pump. May also report to I/C that 2A NSW Pump did not auto start.	
		Evaluate Tech Spec 3.7.2 Service Water System and Ultimate Heat Sink. • Determine 2B NSW pump inoperable • Determine 1A NSW Pump inoperable due to clearance. • Per the Bases, 3 NSW pumps required site wide. • 3.7.2 Condition B. One required NSW pump inoperable for reasons other than condition A. Required Action B.1 Restore required NSW pump to Operable status in 7 days	
		May direct 2C CSW pump to be placed on the NSW header.	
		May conduct a brief (see Enclosure 1 on page 45 for format)	

EVENT 5: NSW PUMP B TRIP (FAILURE OF STANDBY TO START)

Time	Pos	EXPECTED Operator Response	Comments
	ATC	Monitor reactor plant parameters during evolution.	
	BOP	Acknowledge / reference UA-18 (6-1) BUS E4 4KV MOTOR OVLD	
		Recognize trip of 2B NSW pump and lowering NSW system pressure.	
		Announce and execute 0AOP-18.0, NSW System Failure.	
		Recognize the failure of the STBY NSW pump to start and starts standby pump. • Places 2A NSW pump in Manual. • Starts 2A NSW Pump.	
		Refer to alarms. • UA-01 (1-10) NUCLEAR HEADER SERV WTR PRESS-LOW • UA-01 (4-10) NUCLEAR HDR SW PUMP B TRIP • UA-05 (1-9) FAN CLG UNIT CS PUMP RM A INL PRESS LO • UA-05 (2-9) FAN CLG UNIT CS PUMP RM B INL PRESS LO	
		May align the 2C CSW pump to the NSW header.	

EVENT 6: CWIP TRIP	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 6** to activate CW Pump A trip

Simulator Operator Role Play	
	If asked as Outside AO, acknowledge request to check pump. After 2-3 minutes, call back and report that shear pin on the traveling screens for CW Pump A broke.
	If asked as TBAO, identify that breaker AB8 on 4160 V Switchgear 2C is tripped on overcurrent. No other abnormalities.
	If asked as I&C to investigate, acknowledge the request
	If asked for prestart checks for the 2C CWIP, report prestart checks are SAT.
	If asked to verify no personnel are around the 2C Bus, report all clear.

Evaluator Notes	
Plant Response:	Circ Water Pump A will trip and annunciator UA-01, 1-7, CIRC WATER PUMP A TRIP, will alarm. After investigating the cause of the alarm, another Circ Water Pump should be started IAW the APP.
Objectives:	SRO - Direct actions of APP-UA-01, 1-7, CIRC WATER PUMP A TRIP Direct Emergency Depressurization BOP – Perform action of APP UA-01, 1-7, CIRC WATER PUMP A TRIP RO – Monitor plant parameters
Success Path:	Another Circ Water pump is be started.
Event Termination:	Go to Event 7 at the direction of the Lead Evaluator

EVENT 6: CWIP TRIP

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of APP-UA-01, 1-7, CIRC WATER PUMP A TRIP.	
		May direct entry into enter 0AOP-37.0, Loss Of Condenser Vacuum	
		May direct power lowered to 90%	
		May conduct a brief on when Reactor Scram is required (see Enclosure 1 on page 45 for format)	
	ATC	Plant Monitoring	
		May lower power as directed by the CRS. (See page 27)	
	BOP	Take actions IAW APP-UA-01, 1-7, CIRC WATER PUMP A TRIP (see page 38) NOTE: CW ISOL VALVES MODE SELECTOR SWITCH will need to be placed into position D to start C CWIP May announce and enter 0AOP-37.0, Loss Of Condenser Vacuum	
		Direct AOs to investigate pump and pump breaker to determine cause of pump trip.	

CW PUMP A TRIP

AUTO ACTIONS

1. CW Pump A trips

CAUSE

1. Instantaneous overcurrent
2. Time overcurrent
3. Phase overcurrent
4. Differential overcurrent or phase angle (lockout relay)
5. Condenser pit flood level hi-hi
6. Low lube water flow
7. High traveling screen A dP (48 in. water) **AND** screen A stopped
8. High traveling screen A dP (48 in. water) **AND** high screen B, C or D dP (18 in. water)
9. LOCA Load Shed
10. Unit Trip Load Shed
11. Circuit malfunction

OBSERVATIONS

1. Condenser vacuum decreasing (process computer points T000, T001, Recorder OG-PR-23 on XU-2, and 1-OG-PI-23-1A, -2A on XU-80)
2. Generator output decreasing
3. Local relay indication at the breaker compartment
4. Circulating water discharge temperature increasing (BOP typer)
5. CW PUMP LUBE WATER FLOW-LOW (UA-01 5-7) alarm
6. TURB BLDG NW CNDSR PIT FLOOD LVL HI (UA-28 6-6) alarm
7. TURB BLDG E CNDSR PIT FLOOD LVL HI (UA-28 6-5) alarm
8. TURB BLDG SW CNDSR PIT FLOOD LVL HI (UA-28 6-7) alarm
9. CW SCREEN DIFF HI - HI (UA-01 1-4) alarm
10. CW SCREEN A DIFF HIGH OR STOPPED (UA-01 1-5) alarm
11. CW SCREEN B DIFF HIGH OR STOPPED (UA-01 2-5) alarm
12. CW SCREEN C DIFF HIGH OR STOPPED (UA-01 3-5) alarm
13. CW SCREEN D DIFF HIGH OR STOPPED (UA-01 4-5) alarm

ACTIONS

1. If a radioactive liquid release is in progress, terminate the release.
2. If reactor power is less than 90% **OR** a CWIP pump can be started within 5 minutes, **THEN START** an available CWIP.
3. If reactor power is greater than 90% **AND** an available CWIP pump was **NOT** started within 5 minutes, then power must be reduced to 90 to 92% prior to starting a CWIP.

EVENT 7: RWCU LEAK / SBGT FAILS TO START	
Simulator Operator Actions	
	At the discretion of the lead evaluator, **Initiate Trigger 7** to activate the RWCU Leak.

Simulator Operator Role Play	
	If contacted as engineering, acknowledge request for EQ envelopes for the U2 Reactor Building.
	If HP's contacted to perform field surveys acknowledge the request.
	If directed to reset breakers for the RWCU isolation valves, wait 2 minutes and report HP has restricted access to the reactor building. If directed to co-ordinate entry with the HP's, wait 15 minutes and report the breakers will not reset.

Evaluator Notes	
Plant Response:	A large un-isolable RWCU leak will occur. Crew will enter AOP-5.0 and SCCP. SRO should direct a SCRAM.
Objectives:	SRO - Direct response to un-isolable primary system breach in secondary containment. RO - Respond to un-isolable primary system breach in secondary containment. Perform SCRAM actions.
Success Path:	Reactor scram is inserted before max norm operating value is exceeded.
Event Termination:	When a reactor scram is inserted and SCCP entered.

EVENT 7: RWCU LEAK / SBGT FAILS TO START			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Direct entry into 0AOP-5.0, Radioactive Spills, High Radiation, And Airborne Activity	
		Direct RO to trip and isolate RWCU.	
		Announce and enter SCCP procedure	
		Direct a reactor manual scram prior to any area reaching its Max Safe Operating Value	***Critical Task #1***
		May direct a cool down at normal cool down rates (<100°F/hr).	
		Request EQ envelopes for the U2 Rx Bldg	
		Enter and execute RVCP. ☐ Direct RO/BOP to stabilize reactor pressure below 1050 psig. ☐ Verify Instrument operability per Caution 1. ☐ Direct crew to not use N026A/B due to 50' temperatures after 50' alarm reported. ☐ Direct verification of group isolations, ECCS initiations and DG starts as appropriate. ☐ Direct RO/BOP to restore and maintain reactor water level 170"-200"	
		Recognize when alarm A-2 6-8, RB 20/50 FT ELEV TEMP HI, is reported that if 50' elevation is greater than 140°F that the Wide Range (N026) level indicators are inaccurate.	
		Contact I/C for assistance with RWCU isolation valve failures	

EVENT 7: RWCU LEAK / SBGT FAILS TO START

Time	Pos	EXPECTED Operator Response	Comments
	ATC	***Insert Reactor scram as directed by CRS*** Depresses both of the manual scram pushbuttons. Place mode switch to shutdown when steam flow < $3x10^6$ lb/hr. **IF** reactor power is below 2% (APRM downscale trip), **THEN** TRIP the main turbine. ENSURE the master reactor level controller setpoint is +170". **IF** two reactor feed pumps are running, **AND** reactor vessel level is above 160" **AND** rising, **THEN** TRIP one.	***Critical Task #1***
	ATC/ BOP	Maintain reactor pressure as directed by CRS.	
	ATC/ BOP	Maintain reactor water level as directed by SRO.	

EVENT 7: RWCU LEAK / SBGT FAILS TO START

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Respond to UA-03 2-7, AREA RAD RX BLDG HI. Enter and execute 0AOP-5.0, Radioactive Spills, High Radiation, And Airborne Activity. ☐ Evacuate Unit 2 Reactor Bldg. ☐ Direct AO to close PIV-33 RB Sprinkler Shutoff Valve. ☐ Direct E&RC to take applicable 0AOP-5.0 actions. ☐ Check area radiation readings at back panels. ☐ Diagnose source of radiation as RWCU leak.	
		Recognize and report to CRS alarm A-2 6-8, RB 20/50 FT ELEV TEMP HI.	
		Responds to UA-5, 4-6, SBGT SYSA Failure Recognize failure of SBGT to start, places SBGT train A switches to start	
	ATC/ BOP	Maintain reactor pressure as directed by CRS.	
	ATC/ BOP	Maintain reactor water level as directed by SRO.	

EVENT 8: EMERG DEPRESS / ADS VALVE FAILURE / TERMINATION	
Simulator Operator Actions	
	2 minutes after receiving Annunciator UA-12 (2-4) SOUTH RHR RM FLOOD HI, or when anticipation of emergency depressurization is performed, **Initiate TRIGGER 10** (South RHR RM Flood HI-HI)
	When directed by the lead evaluator, place the simulator in FREEZE
	DO NOT RESET THE SIMULATOR PRIOR TO RECEIPT OF CONCURRENCE TO DO SO FROM THE LEAD EXAMINER

Simulator Operator Role Play	

Evaluator Notes	
Plant Response:	Secondary containment conditions will worsen, forcing the SRO to direct an Emergency Depressurization due to high water levels. Two ADS SRV's will fail to manually open. SRO should direct opening two additional SRV's. Scenario will end when reactor pressure reaches 100#.
Objectives:	SRO - Evaluate plant conditions and direct an Emergency Depressurization. RO - Performs actions for Emergency Depressurization.
Success Path:	ED has been performed.
Scenario Termination:	***When emergency depressurization has been performed and the reactor has been depressurized to <100 psig the scenario may be terminated.*** Remind students not to erase any charts and not to discuss the scenario until told to do so by the evaluator/instructor.

EVENT 8: EMERG DEPRESS / ADS VALVE FAILURE / TERMINATION

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Continue reactor cooldown per SCCP direction.	
		Direct Emergency Depressurization when RHR RM FLOOD LEVEL HI-HI alarm (Two plant areas with water levels above Max Safe – South CS and RHR)	***CRITICAL TASK #2***
		Direct RO/BOP to open 7 ADS valves.	
		If informed by RO/BOP that 2 SRVs failed to open, direct opening additional SRVs until 7 SRVs are open.	
		Enter PCCP when torus temperature exceeds 95°F. Directs all available loops to be placed in suppression pool cooling.	
	ATC/ BOP	Recognize and report South CS and South RHR Room Flood Hi-Hi alarms.	
		Open seven ADS valves as directed by SRO.	***CRITICAL TASK #2***
		Recognize failure of 2 ADS valves to OPEN and report to SRO.	
		Open 2 additional SRVs as directed by SRO.	
		Maintain reactor water level as directed by SRO.	
		Place available loops in suppression Pool Cooling IAW hard card. (see page 40)	

ENCLOSURE 1

ATTACHMENT 8

<< Crew Brief Template >>

Begin Brief	☐ Announce "Crew Brief" ☐ All crew members acknowledge announcement

Recap	**(As Required)**
	☐ Update the crew as needed:
	☐ Describe what happened and major actions taken
	☐ Procedures in-progress
	☐ Notifications:
	    ☐ Maintenance
	    ☐ Engineering
	    ☐ Others (Dispatcher, Station Management, etc.)
	☐ Future Direction and priorities
	☐ Discuss any contingency plans

Input	**(As Required)**
	☐ Solicit questions/concerns from each crew member:
	    ☐ ROs
	    ☐ CRS
	    ☐ STA
	☐ Are there any alarms unexpected for the plant conditions?
	☐ What is the status of Critical Parameters?

EAL	**(As Required)** ☐ Provide EAL and potential escalation criteria
Finish Brief	☐ Restore normal alarm announcement? (Yes/No) ☐ Announce "End of Brief"

ATTACHMENT 1 - Scenario Quantitative Attribute Assessment

Category	NUREG 1021 Rev. 2 Supp. 1 Req.	Scenario Content
Total Malfunctions	5-8	7
Malfunctions after EOP Entry	1-2	2
Abnormal Events	2-4	4
Major Transients	1-2	2
EOPs Used	1-2	2
EOP Contingency	0-2	1
Run Time	60-90 min	90
Crew Critical Tasks	2-3	2
Tech Specs	2	2
Instrument / Component Failures before Major	2 – OATC 2 - BOP	4
Instrument / Component Failures after Major	2	2
Normal Operations	1	1
Reactivity manipulation	1	1

ATTACHMENT 2 – Shift Turnover

Brunswick Unit 2 Plant Status					
Station Duty Manager:	E. Neal			Workweek Manager:	B. Craig
Mode:	1	Rx Power:	100%	Mode:	1
Plant Risk: Current EOOS Risk Assessment is:				Green	
SFP Time to 200 Deg F:	49.7 hrs			Days Online:	80 days
Turnover:					
Protected Equipment:	2A FPC Pump/Hx, 2A RCC Pump, and 2C Demin Transfer Pump for Fuel Pool Decay Heat Removal and inventory makeup. 2A/B NSW Pumps due to 1A NSW pump maintenance				
Comments:	1A NSW Pump is under clearance for planned maintenance. 2C TCC Pump is in service on Unit One. The BOP will perform PT-40.2.11, Main Generator Voltage Regulator Manual And Automatic Operational Check.				

Continuous Use

BRUNSWICK UNIT 0

SURVEILLANCE TEST PROCEDURE

0PT-40.2.11

MAIN GENERATOR VOLTAGE REGULATOR MANUAL AND AUTOMATIC OPERATIONAL CHECK

REVISION 6

REVISION SUMMARY
PRR 00652903 **DESCRIPTION**
This procedure revision upgraded the procedure to the new PAS template and AD-DC-ALL-0202, Writer's Manual for Controlled Procedures Manual. Statements for new procedure Scope, General Information and Records sections were added in accordance with the Writer's guide format for test procedures. Added Estimated Capability Curve graphic from 1(2)OP-27 for operator efficiency. Addressed PRR 00686591 to reorder steps Section 7.3 Step 5 to place the voltage regulator in automatic then perform notifications. No technical changes were made to the procedure. Revised by E.R. Sessoms.

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this test is to demonstrate the OPERABILITY of the voltage regulator transfer circuitry and exercise the regulator potentiometers.

2.0 SCOPE

1. This test is performed once every 92 days and demonstrates OPERABILITY of voltage regulator transfer circuitry and exercises the regulator potentiometers.

2. This test may also be used to demonstrate proper operation of the voltage regulator potentiometer and transfer circuitry, after completion of maintenance.

3.0 PRECAUTIONS AND LIMITATIONS

1. Main generator loading is within the limits of the Generator Reactive Capability Curve shown on Attachment 1, Estimated Capability Curve, and with a minimum of 20 MVAR (positive). .... ☐

2. This test is **NOT** performed if erratic operation of the voltage regulator is noted immediately prior to the performance of this test. .... ☐

3. The Load Dispatcher is to be informed when the main generator automatic voltage regulator is **NOT** in service. Log entries are made documenting the notification. {9.1.1} .... ☐

4.0 GENERAL INFORMATION

None

5.0 ACCEPTANCE CRITERIA

1. This test may be considered satisfactory when the following criteria are met:

   a. DC regulator output variation is smooth and in the same direction as the rheostat movement.

   b. AC regulator output variation is smooth and in the same direction as the rheostat movement.

6.0 PREREQUISITES

1. **Confirm** Generator and Exciter System in operation in accordance with 1(2)OP-27, Generator and Exciter System Operating Procedure....... _______

2. **Confirm** Plant Electrical System in operation in accordance with 1(2)OP-50, Plant Electric System Operating Procedure .... _______

3. **Confirm** DC Electrical System in operation in accordance with 1(2)OP-51, DC Electrical System Operating Procedure .... _______

4. **Confirm** 120 Volt AC UPS, Emergency, and Conventional Electrical Systems in operation in accordance with 1(2)OP-52, 120 Volt AC UPS, Emergency, and Conventional Electrical Systems Operating Procedure .... _______

5. **Confirm** <u>NO</u> system load changes are anticipated.................................. _______

7.0 INSTRUCTIONS

7.1 General

1. **Obtain** permission from Unit CRS to perform this test. .... _______

2. **Ensure** all Prerequisites listed in Section 5.0 are met. .... _______

7.2 Operate 70CS (Gen Manual Volt Adj Rheo)

1. **Ensure** 43CS (Regulator Mode Selector) in AUTO. .... _______

2. **Station** an operator at the Excitation Regulator and Control cubicle in the Turbine Building on the 70 ft elevation west to monitor regulator output during the following steps. .... _______

NOTE

- Section 7.2 Step 3 and Section 7.2 Step 4 are repeated as necessary to ensure proper operation/indication of the manual rheostat. .... ☐
- DC regulator output is locally monitored using D1VM (D.C. Reg. Output). .... ☐

3. **Raise** 70CS (Gen Manual Volt Adj Rheo) until the Upper Limit light comes ON. .... _______

NOTE

The Intermed light will come ON during lowering of 70CS (Gen Manual Volt Adj Rheo) and will remain ON after the Low Limit light is ON. .... ☐

4. **Lower** 70CS (Gen Manual Volt Adj Rheo) until the Low Limit light comes ON. .... _______

5. Using 70CS (Gen Manual Volt Adj Rheo) on the RTGB, **null** Gen Volt Reg Diff Volt meter. .... _______

6. **IF** D1VM (D.C. Reg. Output) variation was **NOT** smooth **AND** in the same direction as rheostat movement,
THEN go to Section 7.3 Step 7. .... _______

7.2 Operate 70CS (Gen Manual Volt Adj Rheo) (continued)

7. **IF** D1VM (D.C. Reg. Output) variation was smooth **AND** in the same direction as rheostat movement,
THEN perform the following: {9.1.1}

a. **Notify** the Load Dispatcher the main generator voltage regulator is being placed in MANUAL. .... _______

____
Person Notified

b. **Document** the Load Dispatcher notification in the log. .... _______

c. **Place** 43CS (Regulator Mode Selector) in MAN. .... _______

7.3 Operate 90CS (Gen Auto Volt Adj Rheo)

NOTE
• Section 7.3 Step 1 and Section 7.3 Step 2 may be repeated as necessary to ensure proper operation/indication of the automatic rheostat. .... ☐
• AC regulator output may be locally monitored using A1VM (A.C. Reg. Output). .... ☐

1. **Raise** 90CS (Gen Auto Volt Adj Rheo) until the Upper Limit light comes ON. .... _______

2. **Lower** 90CS (Gen Auto Volt Adj Rheo) until the Low Limit light comes ON. .... _______

3. **Null** Gen Volt Reg Diff Volt meter on the RTGB using 90CS (Gen Auto Volt Adj Rheo). .... _______

4. **IF** A1VM (A.C. Reg. Output) variation was **NOT** smooth **AND** in the same direction as rheostat movement,
   THEN go to Section 7.3 Step 6. .... _______

5. **IF** A1VM (A.C. Reg. Output) variation was smooth **AND** in the same direction as rheostat movement,
   THEN perform the following: {9.1.1}

   a. **Place** 43CS (Regulator Mode Selector) in AUTO. .... _______

   b. **Notify** the Load Dispatcher the main generator voltage regulator is in AUTOMATIC. .... _______

   ____
   Person Notified

   c. **Document** Load Dispatcher notification in the log. .... _______

6. **IF** extended manual voltage regulator operation becomes necessary,
   THEN coordinate with the Load Dispatcher to maintain minimum generator MVAR load and generator voltage in accordance with the System Operation section of 1(2)OP-27, Generator and Exciter System Operating Procedure. .... _______

7. **IF** either regulator output variation was **NOT** smooth **AND** in the same direction as the rheostat,
   THEN prepare a W/R for the regulator. .... _______

7.4 Restoration

1. **Perform** review of completed procedure sections to verify Section 5.0, Acceptance Criteria, for tests performed, have been met. .... _______

   ____
   IV

2. **IF** Acceptance Criteria is **NOT** met, **THEN perform** following:

   a. **Report** any equipment found INOPERABLE or **NOT** meeting Acceptance Criteria to Supervisor. .... _______

   b. **Ensure** CR has been initiated. .... _______

3. **Ensure** required information has been recorded on Attachment 2, Certification and Review Form. .... _______

4. **Notify** Unit CRS when this procedure is complete or found to be unsatisfactory. .... _______

8.0 RECORDS

Completed portions of this procedure are transmitted to QA records for retention per Quality Assurance Program requirements.

9.0 REFERENCES

9.1 Commitments

1. VAR-002-1.1b, Voltage and Reactive (VAR) Reliability Standard, Federal Energy Regulatory Commission

9.2 Technical Specifications

None

9.3 Updated Final Safety Analysis Report

None

9.4 Drawings

None

9.5 Procedures

1. 1OP-27, Generator and Exciter System Operating Procedure
2. 1OP-50, Plant Electric System Operating Procedure
3. 1OP-51, DC Electrical System Operating Procedure
4. 1OP-52, 120 Volt AC UPS, Emergency, and Conventional Electrical Systems Operating Procedure
5. 2OP-27, Generator and Exciter System Operating Procedure
6. 2OP-50, Plant Electric System Operating Procedure
7. 2OP-51, DC Electrical System Operating Procedure
8. 2OP-52, 120 Volt AC UPS, Emergency, and Conventional Electrical Systems Operating Procedure

9.6 Vendor/Technical Manuals

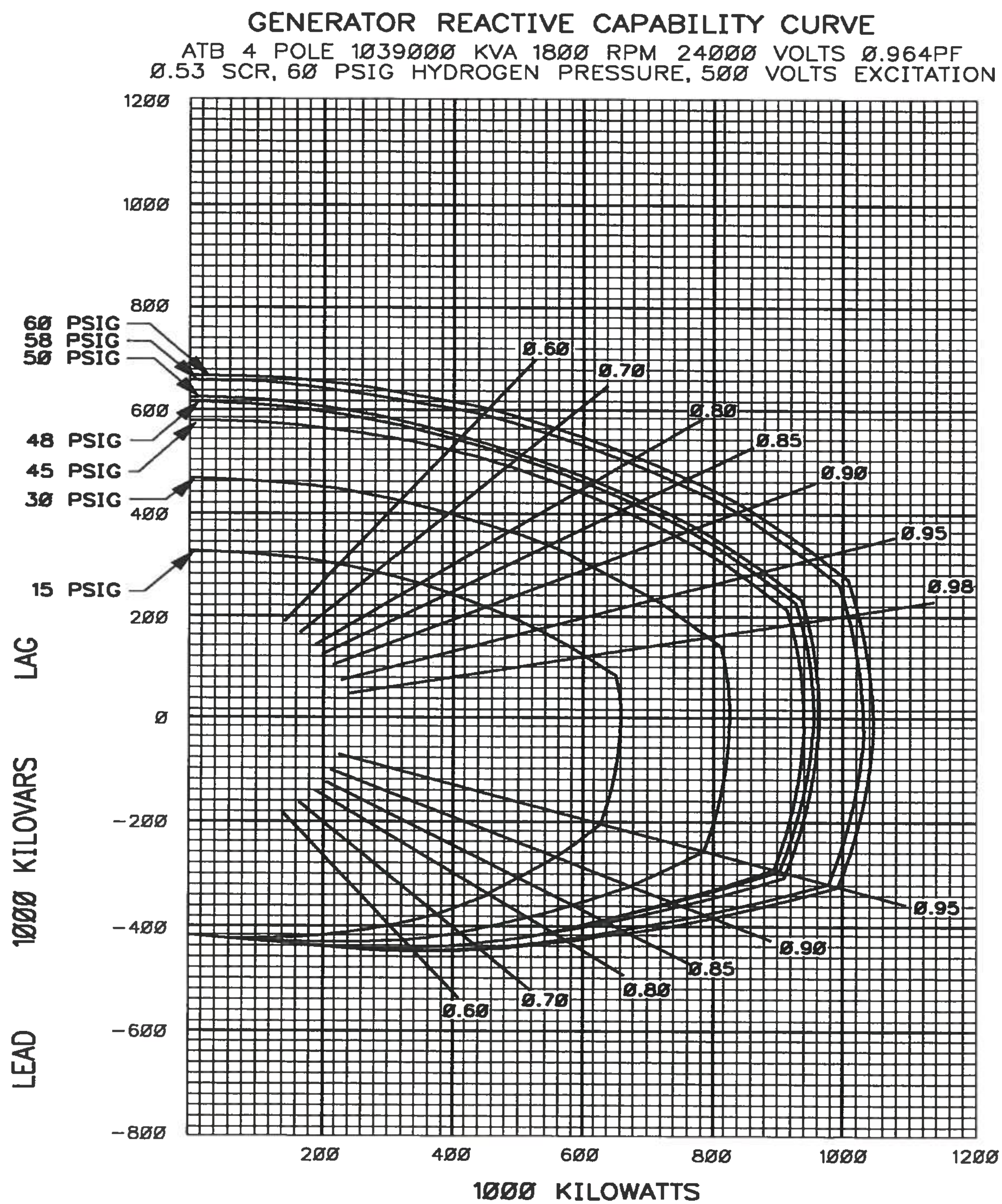
1. FP-85650, Power System Stabilizer Equipment
2. FP-20183, Steam Turbine-Generator, Generator Section (GEK-14870)

9.7 Miscellaneous Documents

1. SD-27, Main Generator and Exciter System

ATTACHMENT 1
Page 1 of 1

Estimated Capability Curve



Certification and Review Form

Date Completed ____________ Time Completed ____________ Unit _____ % Pwr ____________ GMWE _________	Reason For Test ☐ Routine Surveillance ☐ WO # ______________________ ☐ Other (explain) ______________ ______________________________

General Comments and Recommendations: ______________________________

____

____

	Initials	Name (Print)
Test procedure performed by:	______	______________________
	______	______________________
	______	______________________
	______	______________________
	______	______________________

Exceptions to satisfactory performance: ______________________________

____

Corrective action required: ______________________________________

____

Test Procedure has been completed SAT or UNSAT (circle as appropriate):

Unit CRS/SRO ______________________________ ____________
Signature Date

Test procedure has been reviewed by:

Shift Manager ______________________________ ____________
Signature Date

DUKE ENERGY

BRUNSWICK TRAINING SECTION

OPERATIONS TRAINING

INITIAL LICENSED OPERATOR

SIMULATOR EVALUATION GUIDE

2016 NRC SCENARIO 4

START CREV, N004A FAILURE, STATOR COOLING TRIP, RCIC STEAM LEAK, TCC FAILURE, LOOP, DG3 FAILURE, SRV TAILPIPE, ED

REVISION 0

Developer: Bob Bolin	**Date:** 07/07/2016
Technical Review: Dan Hulgin	**Date:** 9/12/2016
Validators: Dwayne Wolf Hunter Morris Grant Newton	**Date:** 09/08/16
Facility Representative: Craig Oliver	**Date:** 09/22/2016

REVISION SUMMARY	
0	Scenario developed for 2016 NRC Exam.

TABLE OF CONTENTS

1.0 SCENARIO OUTLINE

Event	Malf. No.	Type*	Event Description
1		N-BOP	Manual start of CREV in area high radiation mode.
2	NB007F	C-ATC C-CRS	C32-LT-N004A Fails high (TS)
3	EE030M-2TD	C-BOP C-CRS	MCC 2TD trip / Standby Stator Water Cooling Pump fails to auto start
4	ES025F	C-ATC C-CRS	RCIC steam leak (AOP)(TS)
5	K4516A	C-BOP C-CRS	TCC Pump Failure (AOP)
6		R-ATC	Power Reduction
7	EE009F	M C	Loss of Off-Site Power / Scram DG3 Diff O/C / DG4 failure of output breaker to close (RSP)(PCCP)(AOP)
8	ES004F CA020F	C M	SRV Failure / Tailpipe Break / DW Spray Logic Failure ED on PSP (AOP)(EDP)
*(N)ormal, (R)eactivity, (C)omponent or Instrument, (M)ajor			

2.0 SCENARIO DESCRIPTION SUMMARY

Event	Description
1	The BOP will start CREV in the area high radiation mode IAW 0OP-37, Section 6.1.3.
2	After CREV is started, C32-LT-N004A will fail high. The crew will reference Tech Spec 3.3.2.2 and determine a 7 day LCO exists to place the failed channel in the tripped condition. The crew should select level B per OP-32.
3	MCC 2TD will trip and the standby stator cooling water pump will fail to auto start. The standby stator cooling water pump can be manually started. The 2D air compressor will also be lost and 0AOP-20.0 may be entered. Unit One may be contacted to place the 1D Air Compressor in lead.
4	A break in the RCIC steam line in the south RHR room will occur. The break can be isolated by closing either the E51-F007 or the E51-F008. The crew will respond to the steam leak IAW AOP-05.0.
5	TBCCW Pump 2B will trip and TBCCW low header pressure will alarm. The crew will respond per 0AOP-17.0. TBCCW pressure will recover and actions for partial loss of TBCCW will be performed.
6	A power reduction will be required IAW AOP-17.0.
7	A Loss of Offsite Power will occur. The crew will respond per 0AOP-36.1. DG3 will trip on Diff O/C and DG4 output breaker will fail to close, can be closed manually.
8	SRV F will fail open. AOP-30 will be entered. The SRV will not reset using the control switch. Pulling fuses IAW AOP-30 results in loss of indication but the SRV remains open. SRV F tailpipe will rupture, pressurizing containment. The DW Spray logic (think switch) will fail causing an inability to spray the torus or drywell. Emergency Depressurization is required when PSP is violated.

3.0 CREW CRITICAL TASKS

Critical Task #1
Close DG4 output breaker
Critical Task #2
Emergency Depressurize when violating PSP

4.0 TERMINATION CRITERIA

When all rods are inserted and level is being controlled above TAF the scenario may be terminated.

5.0 IMPLEMENTING REFERENCES

NOTE: Refer to the most current revision of each Implementing Reference.

Number	Title
UA-14 (4-2)	CB MACH ROOM VENT FAN TRIP
A-07 (4-2)	FW CTL SYS TROUBLE
0AOP-23.0	CONDENSATE/FEEDWATER SYSTEM FAILURE
UA-06 (2-5)	SUB 2F 480V FEEDER BKR TRIP
UA-13 (6-6)	RFP B CONTROL TROUBLE
UA-02 (1-8)	STAT COOLANT INLET FLOW-LOW
UA-02 (1-9)	LOSS OF STAT COOLANT TRIP CKT ENER
UA-02 (2-8)	STAT COOLANT PRESS-LOW
UA-02 (6-9)	EXCITER COOLANT FLOW-LOW
UA-03 (2-4)	TBCCW PUMP DISCH HEADER PRESS LOW
0AOP-17.0	TURBINE BUILDING CLOSED COOLING WATER SYSTEM FAILURE
A-03 (4-8)	OPRM TRIP ENABLED
0AOP-36.1	LOSS OF ANY 4160V BUSES OR 480V E-BUSES

6.0 SETUP INSTRUCTIONS

1. **PERFORM** TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 5, Checklist for Simulator Exam Security.
2. **RESET** the Simulator to IC-11.
3. **ENSURE** the RWM is set up as required for the selected IC.
4. **ENSURE** appropriate keys have blanks in switches.
5. **RESET** alarms on SJAE, MSL, and RWM NUMACs.
6. **ENSURE** no rods are bypassed in the RWM.
7. **PLACE** all SPDS displays to the Critical Plant Variable display (#100).
8. **ENSURE** hard cards and flow charts are cleaned up
9. **TAKE** the SIMULATOR OUT OF FREEZE
10. **LOAD** Scenario File.
11. **ALIGN** the plant as follows:

Manipulation
Ensure 2C TCC pump is in service on Unit One. Loaded in Scenario File
Ensure 2B Stator Cooling Pump running and 2A in standby

12. **IF desired**, take a **SNAPSHOT** and save into an available IC for later use.
13. **PLACE** a clearance on the following equipment.

Component	Position

14. **INSTALL** Protected Equipment signage and **UPDATE** RTGB placard as follows:

Protected Equipment
1. 2A and 2B NSW pumps 2. 2A FPC Pump/Hx, 2A RCC Pump, and 2C Demin Transfer Pump.

15. **VERIFY** 0ENP 24.5 Form 2 (Immediate Power Reduction Form) for IC-11 is in place.

16. **ENSURE** each Implementing References listed in Section 7 is intact and free of marks.

17. **ENSURE** all materials in the table below are in place and marked-up to the step identified.

Required Materials

18. **ADVANCE** the recorders to prevent examinees from seeing relevant scenario details.

19. **PROVIDE** Shift Briefing sheet for the CRS.

20. **VERIFY** all actions contained in TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 4, Simulator Training Instructor Checklist, are complete.

7.0 INTERVENTIONS

TRIGGERS

Trig	Type	ID
1	Malfunction	NB007F - [RX LVL TRANSMITTER C32-N004A FAILS]
2	Malfunction	EE030M - [INDIVIDUAL BUS FAILURE]
4	Malfunction	ES025F - [RCIC STM BRK - SOUTH RHR]
5	Annunciator	ZUA324 - [TBCCW PUMP DISCH HEADER PRESS LOW]
5	AO Override	G4H11G14 - [TBCCW DISCHARGE PRESS TOC-PI-556]
5	DI Override	K4517A - [TB CCW PMP B ON]
5	DI Override	K4517A - [TB CCW PMP B ON]
5	DO Override	Q4517LG4 - [TB CCW PMP B OFF G]
6	Malfunction	EE009F - [LOSS OF OFF-SITE POWER]
7	Remote Function	SW_IAVSW193 - [SW-V193 MAN ISOL NSW TO RBCCW]
7	Remote Function	SW_VHSW146L - [CONV SW TO RBCCW HXS V146]
8	Remote Function	RP_IAEPAMGA - [RPS M-G SET A EPA BKRS]
8	Remote Function	RP_IARPSA - [RESTART RPS MG SET A]
9	Remote Function	RP_IAEPAMGB - [RPS M-G SET B EPA BKRS]
9	Remote Function	RP_IARPSB - [RESTART RPS MG SET B]
10	Remote Function	ED_ZIEDH11 - [PNL 2AB-RX PWR (E7=NORM/E8=ALT)]
10	Remote Function	ED_ZIEDHX0 - [PNL 32AB PWR (E7=NORM/E8=ALT)]
10	Remote Function	ED_ZIEDH08 - [PNL 2AB PWR (E7=NORM/E8=ALT)]
11	Malfunction	ES004F - [ADS VALVE F FAILS OPEN]
12	DO Override	Q1508LGJ - [SRV VLV B21-F013F GREEN]
12	DO Override	Q1508RRJ - [SRV VLV B21-F013F RED]
12	DO Override	Q1520SA9 - [AMBER LED +5V]
12	Malfunction	CA020F - [SRV F TAIL PIPE RUPTURE]
13	Remote Function	MI_ZVACS918_1 - [UNIT 1 CB MECHANICAL EQUIP ROOM VENT FANS CS]
14	Remote Function	MI_IACBLRM1 - [UNIT 1 CABLE SPREAD ROOM VENT FANS]

Trig #	Trigger Text

MALFUNCTIONS

Malf ID	Mult ID	Description	Current Value	Target Value	Rmp time	Actime	Dactime	Trig
DG006F	DG 4	DG OUTPUT BREAKER FAIL TO AUTO CLOSE	False	True				
DG026F		DG3 DIFFERENTIAL FAULT	False	True				
NB007F		RX LVL TRANSMITTER C32-N004A FAILS	0.00	100.0	00:02:00			1
EE030M	2TD	INDIVIDUAL BUS FAILURE	False	True				2
ES025F		RCIC STM BRK - SOUTH RHR	0.00	5.0	00:10:00			4
EE009F		LOSS OF OFF-SITE POWER	False	True				6
ES004F		ADS VALVE F FAILS OPEN	False	True				11
CA020F		SRV F TAIL PIPE RUPTURE	False	True		00:01:00		12

REMOTES

Remf Id	Mult Id	Description	Current Value	Target Value	Rmp time	Actime	Trig
CC_IACW4518		2C TBCCW PUMP UNIT ALIGNMENT	1	1			
SW_VHSW146L		CONV SW TO RBCCW HXS V146	SHUT	OPEN			7
SW_IAVSW193		SW-V193 MAN ISOL NSW TO RBCCW	OPEN	CLOSE			7
RP_IARPSA		RESTART RPS MG SET A	NORMAL	RESET			8
RP_IAEPAMGA		RPS M-G SET A EPA BKRS	SET	SET		00:00:05	8
RP_IARPSB		RESTART RPS MG SET B	NORMAL	RESET			9
RP_IAEPAMGB		RPS M-G SET B EPA BKRS	SET	SET		00:00:05	9
ED_ZIEDH08		PNL 2AB PWR (E7=NORM/E8=ALT)	NORMAL	ALT		00:00:30	10
ED_ZIEDH11		PNL 2AB-RX PWR (E7=NORM/E8=ALT)	NORMAL	ALT		00:02:30	10
ED_ZIEDHX0		PNL 32AB PWR (E7=NORM/E8=ALT)	NORMAL	ALT		00:04:30	10
MI_ZVACS918_1		UNIT 1 CB MECHANICAL EQUIP ROOM VENT FANS CS	NEUT	STOP			13
MI_IACBLRM1		UNIT 1 CABLE SPREAD ROOM VENT FANS	AUTO	OFF			14
ED_IARKAX5		X-TIE BKR E7-E8 (AX5) RACK STATUS	OUT	IN		00:05:00	15
ED_IARKAI0		X-TIE BKR E8-E7 (AI0) RACK STATUS	OUT	IN		00:02:30	15

PANEL OVERRIDES

Tag ID	Description	Position / Target	Actual Value	Override Value	Rmp time	Actime	Dactime	Trig
K4517A	TB CCW PMP B ON	OFF	OFF	ON				5
K4517A	TB CCW PMP B ON	ON	ON	OFF				5
Q4517LG4	TB CCW PMP B OFF G	ON/OFF	OFF	OFF				5
Q1508LGJ	SRV VLV B21-F013F GREEN	ON/OFF	ON	OFF				12
Q1508RRJ	SRV VLV B21-F013F RED	ON/OFF	OFF	OFF				12
Q1520SA9	AMBER LED +5V	ON/OFF	OFF	OFF				12
K5412A	STAT COOLANT PMP A	AUTO	OFF	OFF				
G4H11G14	TBCCW DISCHARGE PRESS TOC-PI-556	39	80.2739	39				5
K1727A	CONT SPRAY VLV CONTROL	NORMAL	ON	OFF				
K1727A	CONT SPRAY VLV CONTROL	MANUAL	OFF	OFF				
K1727A	CONT SPRAY VLV CONTROL	RESET	OFF	OFF				
K1227A	CONT SPRAY VLV CONTROL	NORMAL	ON	OFF				
K1227A	CONT SPRAY VLV CONTROL	MANUAL	OFF	OFF				
K1227A	CONT SPRAY VLV CONTROL	RESET	OFF	OFF				

ANNUNCIATORS

Window	Description	Tagname	Override Type	OVal	AVal	Actime	Dactime	Trig
2-4	TBCCW PUMP DISCH HEADER PRESS LOW	ZUA324	ON	ON	OFF			5

8.0 OPERATOR RESPONSE AND INSTRUCTIONAL STRATEGIES

EVENT 1: Manual Start of CREV	
Simulator Operator Actions	
	Ensure Monitored Parameters is open and Scenario Based Testing Variables are loaded.
	When contacted to secure the U1 CB Mech Equipment Room Vent Fans **Initiate Trigger 13**
	When contacted to stop the Cable Spread Room 1 Vent Fans **Initiate Trigger 14**

Simulator Operator Role Play	

Evaluator Notes	
Plant Response:	
Objectives:	SRO - Directs BOP to manually start CREV BOP – Manual Start of CREV RO – Monitors the plant
Success Path:	CREV manually started
Event Termination:	When directed by the Lead Evaluator, go to Event 2.

EVENT 1: Manual Start of CREV			
Time	**Pos**	**EXPECTED Operator Response**	**NOTES**
	SRO	Conduct shift turnover shift briefing.	
		Direct CREV to be started in the area high radiation mode IAW OP-37.	
		May conduct a brief (see Enclosure 1 on page 44 for format)	
	RO	Monitors the plant	
	BOP	Manually starts CREV in the area high radiation mode IAW 0OP-37, Section 6.1.3.	

6.1.3 Manual Startup of the Control Building Emergency Recirculation System

1. **Confirm** the following initial conditions are met:

   - All applicable prerequisites as listed in Section 5.0 are met .... ________
   - The Control Building Emergency Recirculation System has failed to start after an initiation signal, ... ________

     or

   - A manual start in accordance with 0AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity, is required {8.1.1} .... ________

     or

   - Surveillance or inspection tests are required .... ________

NOTE

- Indications for the Control Building Ventilation System are located on Panel XU-3 on both units .... ☐
- Controls for the Mechanical Equipment Room Ventilation Fans and the Control Building Wash Room Exhaust Fan are on XU-3 on Units 1 and 2 .... ☐
- Controls for the Cable Spread Room ventilation fans are on Panel XU-3 for the respective unit .... ☐

2. **Perform** the following to place the Control Building Emergency Recirculation System in the area high radiation mode (includes Secondary Containment Isolation):

NOTE

- Placing one of the 2A(B)-ERF-CB (CB Emerg Recirc Fans) in ON will INOP the automatic start function of the non-operating fan. .... ☐
- Controls for the Control Building Emergency Recirculation Fans are on Panel XU-3 on Unit 2 .... ☐

   a. **Place** one of the 2A(B)-ERF-CB (CB Emerg Recirc Fans) in ON .... ________

6.1.3 Manual Startup of the Control Building Emergency Recirculation System (continued)

CAUTION

Detection of heat in the charcoal bed, detectors 2-FP-CB-4-20 and 2-FP-CB-4-21 for A or detectors 2-FP-CB-4-14 and 2-FP-CB-4-15 for B, will trip the associated Emergency Recirculation Fan. .... ☐

b. **Confirm** 2L-D-CB (Ctl RM Norm Mu Air Dmpr) closes. .... _____

c. **Confirm** VA-2J-D-CB (CB Emerg Recirc Damper) opens. .... _____

d. **Stop** 2D-EF-CB (CB Washroom Exhaust Fan) and **confirm** associated damper closes. .... _____

NOTE

The Control Building Mechanical Equipment Room Vent Fans can be stopped only by simultaneously placing both Units' control switches in OFF. .... ☐

e. Simultaneously **place** both Units' control switches in OFF, for 2F-SF-CB and 2E-EF-CB (CB Mechanical Equip Room Vent Fans) to stop the fans and **confirm** associated supply and exhaust dampers close. .... _____

f. **Stop** 2A-SF-CB and 2A-EF-CB (Cable Spread Room 2 Vent Fans) and **confirm** associated supply and exhaust dampers close. .... _____

g. **Stop** 1A-SF-CB and 1A-EF-CB (Cable Spread Room 1 Vent Fans) and **confirm** associated supply and exhaust dampers close. .... _____

6.1.3 Manual Startup of the Control Building Emergency Recirculation System (continued)

NOTE

The Control Building Emergency Recirculation System is now in operation for high radiation conditions. ☐

3. **Perform** the following to place the Control Building Emergency Recirculation System in the fire mode:

NOTE

Placing one of the 2A(B)-ERF-CB (CB Emerg Recirc Fans) in ON will INOP the automatic start function of the non-operating fan. ☐

a. **Place** one of the 2A(B)-ERF-CB (CB Emerg Recirc Fans) in ON. ______

CAUTION

Detection of heat in the charcoal bed, detectors 2-FP-CB-4-20 and 2-FP-CB-4-21 for A or detectors 2-FP-CB-4-14 and 2-FP-CB-4-15 for B, will trip the associated Emergency Recirculation Fan. ☐

b. **Confirm** 2L-D-CB (Ctl RM Norm Mu Air Dmpr) closes. ______

c. **Confirm** VA-2J-D-CB (CB Emerg Recirc Damper) opens. ______

d. **Stop** 2D-EF-CB (CB Washroom Exhaust Fan) and **confirm** associated damper closes. ______

NOTE

The Control Building Emergency Recirculation System is now in operation for fire conditions. ☐

4. **WHEN** the initiating conditions have cleared, **THEN place** Control Building Ventilation System in operation in accordance with Section 6.1.4. ______

EVENT 2: C32-LT-N004A FAILS HIGH	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 1** to fail C32-LT-N004A upscale.

Simulator Operator Role Play	
	If contacted as TB AO to check UPS Panel V10, Ckt #3, acknowledge request, then report no tripped breakers on UPS Panel V10. If asked, the inverters on the trip cabinets are energized.
	If contacted as maintenance or I&C to investigate trip, acknowledge request

Evaluator Notes	
Plant Response:	C32-LT-N004A will fail high. The crew will reference Tech Spec 3.3.2.2 and determine a 7 day LCO exists to place the failed channel in the tripped condition. The crew should select level B per OP-32.
Objectives:	SRO - Determine TS LCO for C32-LT-N004A failing high RO - Transfer DFCS to control to B
Success Path:	TS LCO 3.3.2.2, Condition A One feedwater and main turbine high water level trip channel inoperable. Required Action A.1 Place channel in trip within 7 days. DFCS Feedwater Level Select transferred to B
Event Termination:	Go to Event 3 at the direction of the Lead Evaluator.

EVENT 2: C32-LT-N004A FAILS HIGH

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Acknowledges annunciator report A-07 4-2 *FW CTL SYS TROUBLE*	
		Contacts I&C to investigate.	
		May direct FWCS Level control to be selected to Level B.	
		Determines TS 3.3.2.2 Condition A A.1 Place channel in trip in 7 days.	
		May conduct a brief (see Enclosure 1 on page 44 for format)	
	RO	Acknowledges and reports annunciator report of A-07 4-2 *FW CTL SYS TROUBLE*	
		Diagnose failure of the C32-N004A	
		If directed by the CRS, shifts LEVEL A/B select switch to Position B.	
	BOP	Monitors the plant	

EVENT 3: MCC 2TD LOSS / STATOR COOLING STANDBY PUMP FAILURE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 2** to trip the feeder breaker to MCC 2TD.
	If requested to place the 1D Air Compressor in lead **Activate Remote AI_2DLEAD**, DELTA SA-CS-7892 (LEAD/LAG SWITCH) to 1D LEAD, 2D LAG

Simulator Operator Role Play	
	When asked as the TB AO to investigate the 2F feeder breaker trip, report a trip of the feeder breaker to MCC 2TD, (ATO) on 480V Substation 2F is tripped with the white overcurrent indicating flag protruding from the breaker.
	If asked as I&C to investigate, acknowledge any requests for MCC trip / Auto start failure. If asked do not recommend re-energizing 2TD until an investigation can be completed.
	If asked to investigate/acknowledge the 2B RFP alarm, acknowledge the local panel alarm and report that the alarm on the local panel is “HPU Pump 2 Running in Stby”. If asked the standby pump is operating with no problems noted.
	If dispatched to verify proper operation of the standby Stator Water Cooling Water Pump or the 2B air compressor, report no problems with the operation of the pump/compressor are noted.
	If contacted as U1, report that the 1D air compressor is running as lag compressor. If asked to place the 1D Air Compressor in lead, after SIM OP activates remote, report 1D air compressor has been placed in lead.

Evaluator Notes	
Plant Response:	The crew will respond to a trip of MCC 2TD with the standby stator cooling water pump failure to auto start. The standby stator cooling water pump can be manually started. The 2D air compressor will also be lost (loss of controls) and 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures, may be entered.
Objectives:	SRO - Direct the standby Stator Cooling Water pump to be started. RO - Start the standby Stator Water Cooling pump identify 2D air compressor failure.
Success Path:	Standby Stator Cooling Water Pump started and actions of 0AOP-20.0 Pneumatic (Air/Nitrogen) System Failures, addressed.
Event Termination:	Go to Event 4 at the direction of the Lead Evaluator.

EVENT 3: MCC 2TD LOSS / STATOR COOLING STANDBY PUMP FAILURE			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Acknowledges report of alarms received/cleared for the BOP/RO.	
		Directs BOP operator to start the standby stator water cooling pump.	
		May ask for I&C to investigate 1) The trip of the feeder breaker to 2TD 2) The failure of the standby Stator Water Cooling pump to auto-start.	
		May direct entry into 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures,.	
		May review the load list for MCC 2TD (0OI-50.11).	
		May conduct a brief (see Enclosure 1 on page 44 for format)	
	ATC	Monitors the plant.	

EVENT 3: MCC 2TD LOSS / STATOR COOLING STANDBY PUMP FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Report alarms to the CRS. UA-6, 2-5 – Sub 2F 480V Feeder Bkr Trip UA-13, 6-6 – RFP B Control Trouble UA-2, 1-8 – Stat coolant Inlet Flow-Low UA-2, 1-9 – Loss of Stat Coolant Trip Ckt Ener UA-2, 2-8 – Stat Coolant Press-Low UA-2, 6-9 – Exciter Coolant Flow-Low	
		Start the standby Stator Water Cooling Pump. UA-2, 4-9 – Stator Cool Reserve Pump Running will annunciate on starting of the standby pump and then will clear when the 2B pump is placed in off.	
		Dispatch an AO to investigate the Sub 2F Feeder Breaker Trip.	
		May dispatch an AO to verify proper operation of the Stator Water Cooling pump that was started.	
		May Dispatch an AO to investigate the alarm on the 2B RFP.	
		May enter and announce 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures, for the trip of 2D Air Compressor. May ask Unit One to place the 1D Air Compressor in the lead position. May place the 2D A/C in Stop.	

EVENT 4: RCIC STEAM LEAK	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 4** to initiate a RCIC steam leak. (Increase as necessary to have room temperatures slowly rising until system is isolated)

Simulator Operator Role Play	
	After the initial alarms for the steam leak are received, report as RB AO steam is blowing out of the -17 foot in South RHR room and you are leaving the building.
	If contacted as I&C to investigate, acknowledge the request.
	If contacted to close 2-FP-PIV33, Unit 2 Reactor Building Sprinkler Shutoff Valve, wait three minutes and report that the valve is closed.

Evaluator Notes	
Plant Response:	A break in the RCIC steam line in the south RHR room will occur. The break can be isolated. If the system is delayed from being isolated, observe temperatures in the Reactor Building (specifically South RHR Room temperature), before any area exceeds MSOTL, a Reactor Manual Scram should be inserted. The RCIC system should be declared inoperable and Tech Specs entered.
Objectives:	SRO - Determine RCIC should be isolated and actions required for LCO per Technical Specifications RO - Respond to an isolable RCIC steam line break.
Success Path:	Evaluate Tech Specs to determine required actions as outlined in SRO actions below.
Event Termination:	Go to Event 5 at the direction of the Lead Evaluator.

EVENT 4: RCIC STEAM LEAK

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Diagnose RCIC leak and direct RCIC isolation	
		May direct entry into 0AOP-05.0	
		May direct reactor building evacuated	
		Contact maintenance about the RCIC steam leak	
		Refers to Tech Spec 3.5.3 RCIC System and determines: CONDITION A. RCIC System inoperable. REQUIRED ACTION: A.1 Verify by administrative means HPCI System is OPERABLE. Immediately AND A.2. Restore RCIC System to OPERABLE status. 14 days	
		May conduct a brief (see Enclosure 1 on page 44 for format)	

EVENT 4: RCIC STEAM LEAK

Time	Pos	EXPECTED Operator Response	Comments
	RO	Respond to alarms: UA-03 3-5, PROCESS RX BLDG VENT RAD HI-HI UA-03 2-7, AREA RAD RX BLDG HIGH UA-05 6-10, RX BLDG ISOLATED	
		Diagnose RCIC steam line leak	
		Isolate RCIC by closing either isolation valve: E51-F007 (Steam Supply Inboard Isol Vlv) and/or E51-F008 (Steam Supply Outboard Isol Vlv)	
		May reference procedure 2OP-16, Section 6.3.4. (see page 25)	
	BOP	Monitors the plant.	
		May announce and enter AOP-05 May direct AO to close 2-FP-PIV33	

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6.3.4 Isolating the RCIC System Steam Supply

1. **Confirm** all applicable prerequisites listed in Section 5.0 are met. .... _______

2. <u>IF</u> rapid isolation of RCIC steam line is desired,
   <u>**THEN**</u> **perform** the following: .... _______

   a. **Close** E51-F007 (Steam Supply Inboard Isol Vlv). .... _______

   b. **Close** E51-F008 (Steam Supply Outboard Isol Vlv). .... _______

CAUTION

Opening the E51-F045 (Turbine Steam Supply Vlv) to de-pressurize the RCIC steam line will roll the RCIC turbine. .... ☐

3. <u>IF</u> rapid isolation is <u>**NOT**</u> desired,
   <u>**THEN**</u> **perform** the following to isolate and de-pressurize the RCIC steam supply line: .... _______

   a. **Close** E51-F007 (Steam Supply Inboard Isol Vlv). .... _______

   b. **Open** MVD-V5002 (HPCI/RCIC Cond Drn Line Back Press Orifice Bypass Valve). .... _______

   c. **Open** E51-F045 (Turbine Steam Supply Vlv) and **monitor** turbine response. .... _______

   d. **Close** E51-F025 (Supply Drain Pot Inbd Drain Vlv) .... _______

   e. **Close** E51-F026 (Supply Drain Pot Otbd Drain Vlv) .... _______

   f. <u>**WHEN**</u> RCIC steam line has been de-pressurized for approximately 2 minutes,
      <u>**THEN**</u> **close** E51-F008 (Steam Supply Outboard Isol Vlv). .... _______

   g. **Close** E51-F045 (Turbine Steam Supply Vlv). .... _______

   h. **Close** MVD-V5002 (HPCI/RCIC Cond Drn Line Back Press Orifice Bypass Valve). .... _____ / _____
      IV

6.3.4 Isolating the RCIC System Steam Supply (continued)

NOTE

- Technical Specification 3.6.1.6.1 (MODES 1, 2, or 3) requires completion of 0PT-02.3.1B, Suppression Pool to Drywell Vacuum Breaker Position Check, within 6 hours after any discharge of steam to the suppression chamber from any source and within 6 hours following an operation that causes any of the vacuum breakers to open. .... ☐
- Section 6.3.4 Step 3.i ensures compliance with Technical Specifications and may be completed as required during the performance of the procedure. .... ☐

i. **IF** in MODES 1, 2, or 3,
THEN ensure 0PT-02.3.1B, Suppression Pool to Drywell Vacuum Breaker Position Check, is completed within 6 hours after any discharge of steam to the suppression chamber from any source. {8.1.7} .... ______

Date/Time Completed ____________________

Performed By (Print) Initials

____________________ ______

____________________ ______

____________________ ______

____________________ ______

Reviewed By ____________________
Unit CRS/SRO

EVENT 5/6: TCC PUMP B TRIP / POWER REDUCTION	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 5** to trip the 2B TCC Pump.
	When power is reduced change TCC pressure override to current value and activate over one minute then delete annunciator override, when pressure is at current value delete override.

Simulator Operator Role Play	
	If contacted as the TB AO, wait one minute and report that 2B TCC pump is hot to the touch and the breaker is tripped (magnetic).
	If contacted as Unit One CRS, report Unit One is using the 2C TCC Pump and cannot be released to Unit Two operation
	If contacted as I&C to investigate 2B TCC Pump, acknowledge request.
	If contacted as Unit One to start the 1D air compressor, report 1D Air Compressor is running.
	If contacted as RE for power reduction or Reactivity Plan, ask the CRS what their recommendation is, then concur with that recommendation.
	If contacted as chemistry acknowledge request for sample due to a 15% power change.

Evaluator Notes	
Plant Response:	TBCCW Pump 2B will trip and TBCCW low header pressure will alarm. The crew will respond per 0AOP-17.0. With 2C TBCCW Pump supplying Unit 1, a power reduction will be required. TBCCW pressure will recover and actions for partial loss of TBCCW will be performed.
Objectives:	SRO - Direct entry into 0AOP-17.0 RO - Power reduction with Recirc flow and restoration of TCC pressure Perform actions for a partial loss of TCC.
Success Path:	TCC pressures restored to normal with reactor power reduced to the recirc flow limit.
Event Termination:	Go to Event 7 at the direction of the Lead Evaluator.

EVENT 5/6: TCC PUMP B TRIP / POWER REDUCTION

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Acknowledge report of annunciator UA-03 2-4 *TBCCW PUMP DISCH HEADER PRESS LOW*	
		Direct entry into 0AOP-17.0, Turbine Building Closed Cooling Water System Failure.	
		Direct power reduction IAW 0ENP-24.5 to the Recirc flow limit.	
		Directs I&C to investigate loss of 2B TCC pump.	
		Acknowledge report of annunciator A-03 4-8 *OPRM TRIP ENABLED*	
		Contact chemistry to sample coolant because of the power reduction (>15%)	
		Briefs crew on reactor scram if TCC pressure is not restored above 42 psig within 4 minutes of reaching 47Mlbm/hr.	
		May conduct a brief (see Enclosure 1 on page 44 for format)	

EVENT 5/6: TCC PUMP B TRIP / POWER REDUCTION			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	RO	Plant Monitoring.	
		Reduces reactor power with Recirc Flow to ENP-24.5 flow limit of 47 Mlbm/hr. May use the Manual Runback for flow (see page 34)	
		Acknowledge and report annunciator A-03 4-8 *OPRM TRIP ENABLED*	
	BOP	Acknowledge and Report annunciator UA-03 2-4 *TBCCW PUMP DISCH HEADER PRESS LOW*	
		Diagnose loss of 2B TCC Pump. Announce and enter 0AOP-17.0, Turbine Building Closed Cooling Water System Failure. (see page 30) Performs step 4.2.3 (page 30) Performs Step 4.2.6 (page 31)	
		Report annunciator UA-03 2-4 *TBCCW PUMP DISCH HEADER PRESS LOW* clear.	

4.2 Supplementary Actions (continued)

NOTE

TBCCW pump power supplies are as follows: ☐

- TBCCW Pump 1A, MCC 1TJ
- TBCCW Pump 1B, MCC 1TM
- TBCCW Pump 2A, MCC 2TJ
- TBCCW Pump 2B, MCC 2TM
- TBCCW Pump 2C, MCC 2TH, with an automatic transfer switch to select MCC 1TH as the power supply on loss of power to 2TH

NOTE

In accordance with 0AP-013, Plant Equipment Control, tripped breakers (thermally or magnetically) should NOT be reset except in an emergency situation until an evaluation of the circuit condition has been performed. Breakers that have tripped thermally may be reset as deemed necessary by the Unit CRS for continued reliable operation of the plant ☐

b. IF TBCCW pump breakers local thermal or magnetic trips have activated,
THEN **perform** the following:

(1) **Initiate** a WO for evaluation of the affected circuit ☐

(2) WHEN directed by the Unit CRS,
THEN **reset** tripped breakers ☐

3. IF only one TBCCW pump is in service
AND TBCCW pressure is less than 42 psig,
THEN **perform** the following:

a. **Reduce** reactor power with recirc flow in accordance with 0ENP-24.5, Form 2, Immediate Reactor Power Reduction Instructions ☐

b. IF TBCCW pressure is greater than 42 psig within 4 minutes,
THEN **perform** Section 4.2 Step 6, on page 7 ☐

c. IF TBCCW pressure is NOT greater than 42 psig within 4 minutes,
THEN **perform** Section 4.2 Step 7, on page 9 ☐

4.2 Supplementary Actions (continued)

4. IF a TBCCW system leak is suspected,
THEN:

   a. **Monitor** TBCCW Head Tank level. ☐

   b. **Maintain** TBCCW Head Tank level in accordance with 2OP-44, Turbine Building Closed Cooling Water System Operating Procedure. ☐

   c. **Check** system piping to locate leakage. ☐

   d. **Isolate** any leakage found. ☐

   e. **Monitor** temperatures on equipment cooled by TBCCW. ☐

5. IF TBCCW heat exchanger outlet temperature is greater than 110°F
OR component temperatures are rising,
THEN **reduce** reactor power as necessary to reduce TBCCW temperature. ☐

NOTE

A partial loss of TBCCW or service water is defined as reduced cooling available with the expectation that normal cooling can be quickly re-established. ☐

6. IF there is a partial loss of TBCCW or service water,
THEN **perform** the following:

   a. **Ensure** all available TBCCW pumps are operating. ☐

NOTE

High temperature indications on equipment cooled by TBCCW in conjunction with CSW header pressure approaching 90 psig are indications of a Conventional Service Water System failure. {7.1.1} ☐

   b. IF a failure of CSW is indicated,
   THEN **enter** 0AOP-19.0, Conventional Service Water System Failure, AND **perform** concurrently with this procedure. ☐

4.2 Supplementary Actions (continued)

c. **Reduce** system heat load by removing the following loads as plant conditions permit:

(1) Out-of-service equipment ☐

(2) Sample coolers ☐

(3) Bus-duct cooling ☐

NOTE

- If only the main or standby compressor is operating on the unaffected unit and the idle compressor is available, there should be sufficient compressed air capacity to support air demand. ☐
- Service Air Compressors 1B and 2B are **NOT** designed to individually carry the full system demand of both units when the cross tie valves are open. ☐

d. **IF** air pressure can **NOT** be maintained,
THEN enter 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures **AND perform** concurrently with this procedure ☐

e. **IF** Unit 1 and Unit 2 Service Air Systems are cross-tied,
THEN:

(1) **Ensure** the unaffected unit's air compressors have sufficient capacity to support air demand. ☐

(2) **Ensure** the unaffected unit's Service Air Compressor D is operating. ☐

f. **IF** Unit 1 and Unit 2 Service Air Systems are **NOT** cross-tied **AND** it is possible to cross-tie,
THEN:

NOTE

- If only the main or standby compressor is operating on the unaffected unit and the idle compressor is available, there should be sufficient compressed air capacity to support air demand. ☐
- Service Air Compressors 1B and 2B are **NOT** designed to individually carry the full system demand of both units when the cross tie valves are open. ☐

(1) **Obtain** permission from the unaffected unit's CRS to cross-tie the Service Air Systems. ☐

4.2 Supplementary Actions (continued)

(2) **Ensure** 2-SA-PV-5071 (Cross-Tie Valve) on Unit 2 Panel XU-2, is OPEN. ☐

(3) **Ensure** 1-SA-PV-5071 (Cross-Tie Valve) on Unit 1 Panel XU-2, is OPEN. ☐

(4) IF the uninvolved unit's air systems are adversely affected,
THEN perform the following at the direction of the Unit CRS or Reactor Operator:

- IF Service Air Dryer 1B is in standby, OR in service on Unit 1, THEN **close** 2-SA-PV-5071 (Cross-Tie Valve), using the control switch located on Unit 2 Panel XU-2 ☐

OR

- IF Service Air Dryer 1B is in service on Unit 2, THEN **close** 1-SA-PV-5071 (Cross Tie Valve), using the control switch located on Unit 1 Panel XU-2. ☐

g. **Trip** the affected unit's air compressors. ☐

6.3.16 Initiation Of A Manual Runback

NOTE

The Man Runback feature is enabled only when both Recirc Pumps are operating at greater than 54.8% speed ☐

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

1. **Confirm** the following Initial Conditions are met:
   - Manual Runback Enabled white light on Panel P603 is ON ________
   - Immediate power reduction is required, which reduces total core flow to 47 mlb/hr, or ________
   - The Unit CRS directs initiation of a Manual Runback ________

NOTE

A Manual Runback lowers both Recirc Pump speeds at 100 rpm/second to 53.6% speed, which results in approximately 47mlb/hr core flow. The Manual Runback can be reset by depressing the Man Runback pushbutton a second time ☐

2. **Depress** the Man Runback pushbutton ________
3. **Confirm** the following:
   a. Both Recirc Pump speeds are lowering ________
   b. The Manual Runback Enabled light is flashing ________
   c. 2-A-06, 3-2 (2-A-07, 2-4), Recirc Flow A(B) Limit, annunciator is ON ________
   d. Resultant Core Flow is approximately 47 mlb/hr, unless manually RESET ________

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

4. **Go to** 2AOP-04.0, Low Core Flow ________

EVENT 7: LOOP / SCRAM / DG FAILURES	
	Simulator Operator Actions
	At the discretion of the lead evaluator, **Initiate Trigger 6** to active the LOOP.
	Acknowledge and silence Fireworks alarms Acknowledge Unit 1 alarms as needed.
	If directed to align RBCCW to CSW cooling, wait 4 minutes and **Initiate Trigger 7.**
	If directed to restart RPS MG sets, wait 3 minutes and insert the following as requested: For RPS A **Initiate Trigger 8** and/or for RPS B **Initiate Trigger 9**.
	If directed to swap AB panels **Initiate Trigger 10** and inform Sim Role Player when timed out.
	If directed to rack in 480V cross-tie breakers **Initiate Trigger 15**

	Simulator Operator Role Play
	If requested to monitor DGs, acknowledge alarms on DG local Alarm Panel (Instructor Aids/Panels) and report alarms if requested
	If directed to align RBCCW to CSW cooling, wait 4 minutes and inform Sim Operator to align RBCCW to CSW cooling then report valve open.
	If directed to restart RPS MG sets, wait 3 minutes and inform Sim Operator to restart RPS then report actions complete.
	If directed as RBAO to ensure BFIV latching mechanisms are disengaged, wait two minutes, then report latches are disengaged.
	If requested to transfer 2AB, 32AB, 2AB-RX, acknowledge request, inform Sim operator and when the remotes are timed out inform the control room the action is complete.
	If directed to cross-tie 480V after remote timers time out report breakers racked in.

	Evaluator Notes
Plant Response:	The crew will respond to a Loss of Offsite Power. The reactor will scram on MSIV closure on the LOOP. All Diesel Generators will start on the LOOP signal. DG3 will trip on Diff O/C. DG 4 output breaker will fail to auto close. The BOP operator will close DG 4 output breaker to energize Bus E4.
Objectives:	SRO - Direct actions of AOP-36.1 RO - Close DG4 output breaker. Perform scram immediate operator actions.
Success Path:	Scram immediate operator actions are complete and DG4 output breaker is closed.

EVENT 7: LOOP / SCRAM / DG FAILURES

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct AOP-36.1 entry.	
		Direct DG4 output breaker closed.	***CRITICAL TASK #1***
		Contacts Maintenance for failure of DG3 and DG4 output breaker.	
		Enters and directs actions of RVCP: ☐ Direct control of reactor pressure using SRVs (establishes pressure band 800 – 1000 psig) ☐ Direct water level band of 170 – 200 inches	
		Enters and directs actions of PCCP: ☐ Monitor and control Suppression Pool temperature below 95 deg F. ☐ Direct starting available RHR Loops in Suppression pool Cooling as necessary to maintain temp below 95 F. ☐ Monitor HCTL ☐ Direct operation of available drywell coolers ☐ Verify RCC operation and alignment to the drywell	

EVENT 7: LOOP / SCRAM / DG FAILURES

Time	Pos	EXPECTED Operator Response	Comments
	ATC	**Unit 2 SCRAM Immediate Actions** 1. **Ensure** SCRAM valves OPEN by manual SCRAM or ARI initiation. 2. **WHEN** steam flow less than 3.0 Mlb/hr, **THEN place** reactor mode switch in SHUTDOWN. 3. **IF** reactor power below 2% (APRM downscale trip), **THEN trip** main turbine. 4. **Ensure** master RPV level controller setpoint at +170 inches. 5. **IF** Two reactor feed pumps running **AND** • RPV level above +160 inches **AND** • RPV level rising, **THEN trip** one.	
		Communicate scram report to CRS Place SULCV in service (See Enclosure 4 page 48) Insert Nuclear Instrumentation Ensure Turbine Oil System Operating Ensure Reactor Recirculation Pumps at 34% Ensure Heater Drain Pumps tripped Maintain reactor water level between 170 – 200 inches	
		Place RHR Loops in Suppression pool Cooling as necessary (see Enclosure 3 page 46)	
		Control reactor pressure 800 – 1000 psig	

EVENT 7: LOOP / SCRAM / DG FAILURES			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	BOP	Diagnose failure of DG4 output breaker	
		Manually close DG4 Output Breaker	***CRITICAL TASK #1***
		Diagnose and report to the SRO DG3 tripped and Locked out.	
		Perform the following 0AOP-36.1 actions:	
		Dispatch AO to monitor DGs	
		Momentarily place DIV I NON-INTRPT RNA, SV-5262 control switch to OVERRIDE/RESET, then to OPEN, and ensure DIV I NON-INTRPT RNA, SV-5262 opens.	
		May start the CRD system in accordance with OP-08, Section 8.17, or it may be started IAW SEP-09.	
		Ensure the associated NSW and CSW pumps are operating.	
		Direct an AO to swap the AB panels to their alternate source.	
		Ensure 125V and 24V DC battery chargers return to service for each energized 480V E Bus.	
		Perform the following to transfer RBCCW HXs from the NSW header to the CSW header: ☐ Confirm CSW system available. Ensure at least one of the following is closed: ☐ RBCCW HX SERVICE WATER INLET VALVE,SW-V103 ☐ RBCCW HX SERVICE WATER INLET VALVE, SW-V106 ☐ Direct an AO to open CONVENTIONAL HEADER TO RBCCW HEAT EXCHANGERS SUPPLY VALVE,SW-V146.	

EVENT 7: LOOP / SCRAM / DG FAILURES			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	BOP	Continue 0AOP-36.1 actions:	
		Ensure Control Building Ventilation started on the affected unit:	
		Perform the following to restore drywell cooling: ☐ If three RBCCW pumps are running, then STOP one RBCCW pump, and place its control switch in AUTO. ☐ If only one RBCCW pump is running, then START a second pump, if available. ☐ If no RBCCW pump is running, then place all RBCCW pump control switches in OFF, and perform one of the following:	
		IF any local drywell temperature is currently greater than the starting temperature limit OR has exceed the starting temperature limit since the initiation of the event, then perform 2OP-21, Section 8.6. IF all local drywell temperatures have remained less than the starting temperature limit since the initiation of the event, then perform 2OP-21, Section 5.2.	
		ENSURE all available drywell coolers on the affected unit are operating.	
		IF HPCI is running with suction from the CST AND CST level indication is NOT available in the Control Room or Radwaste, then monitor CST level locally and report level every hour.	
		Start RPS MG Sets A(B) in accordance with OP-03, Section 5.2	
		May direct for 480V busses to be cross-tied.	

EVENT 7: LOOP / SCRAM / DG FAILURES

Time	Pos	EXPECTED Operator Response	Comments
	BOP	Continue 0AOP-36.1 actions:	
		Perform the following to start the Reactor Building HVAC: ☐ If PROCESS OG VENT PIPE RAD HI-HI (UA-03, 5-4) is in alarm, and is NOT the result of a valid high radiation signal, then place CAC PURGE VENT ISOL OVRD, CAC-CS-5519, in OVERRIDE Reset the following Reactor Building Ventilation Radiation Monitors on Panel H12-P606: ☐ PROCESS REACTOR BLDG VENTILATION RADIATION MONITOR A, D12-RM-K609A ☐ PROCESS REACTOR BLDG VENTILATION RADIATION MONITOR B, D12-RM-K609B. Depress the following Isolation Reset Groups push buttons: ☐ ISOLATION RESET GROUPS 1, 2, 3, 6, 8, A71-S32 ☐ ISOLATION RESET GROUPS 1, 2, 3, 6, 8, A71-S33. ☐ Ensure Instrument Air header pressure is greater than 95 psig. ☐ Ensure BFIV latching mechanisms are disengaged. (Local). ☐ Open RB VENT INBD ISOL VALVES, A-BFIV-RB and C-BFIV-RB. ☐ Open RB VENT OTBD ISOL VALVES, B-BFIV-RB and D-BFIV-RB. Start three sets of Reactor Building Ventilation Fans in accordance with OP-37.1, Section 8.8 to maintain Reactor Building static pressure negative.	

EVENT 8: SRV FAILURE /TAILPIPE BREAK / ED - PSP / TERMINATION	
	Simulator Operator Actions
	Initiate Trigger 11, to fail the SRV.
	When contacted to pull SRV F fuses **Initiate Trigger 12**. (this also fails the downcomer)
	When directed by the Lead Evaluator, place the simulator in FREEZE
	DO NOT RESET THE SIMULATOR PRIOR TO RECEIPT OF CONCURRENCE TO DO SO FROM THE LEAD EXAMINER

	Simulator Operator Role Play
	If contacted to pull fuses for SRV F IAW AOP-30.0, wait 2 minutes have SIM OP **Initiate Trigger 12** and report that the Fuses for SRV F have been pulled.

Evaluator Notes	
Plant Response:	An SRV will fail open and then the tailpipe will break causing a violation of PSP requiring the plant to be emergency depressurized.
Objectives:	SRO - Directs actions for Emergency Depressurization. RO - Perform Emergency depressurization.
Success Path:	Emergency depressurization performed.
Scenario Termination:	***When emergency depressurization has been performed and RPV pressure is less than 100 psig, the scenario may be terminated.***

Remind students not to erase any charts and not to discuss the scenario until told to do so by the evaluator/instructor.

EVENT 8: SRV FAILURE /TAILPIPE BREAK / ED - PSP / TERMINATION			
Time	**Pos**	**EXPECTED Operator Response**	**Comments**
	SRO	Directs announcement of 0AOP-30.0.	
		Direct level maintained 170 – 200 inches.	
		Directs Emergency Depressurization when PSP is violated. (see Enclosure 2 on page 45)	***CRITICAL TASK #2***
	ATC	Maintains level as directed by the CRS.	
		Maintains reactor pressure as determined by the CRS. Informs CRS of SRV F failure to close.	
		Announces and enters 0AOP-30.0. Attempts to cycle control switch for stuck open SRV. Directs WCCSRO to pull fuses for SRV F	
		Performs Emergency Depressurization when directed by the CRS.	***CRITICAL TASK #2***
	BOP	Continues 0AOP-36.1 actions. (see actions in event 7)	

ENCLOSURE 1

ATTACHMENT 8

<< Crew Brief Template >>

Begin Brief	☐ Announce "Crew Brief" ☐ All crew members acknowledge announcement

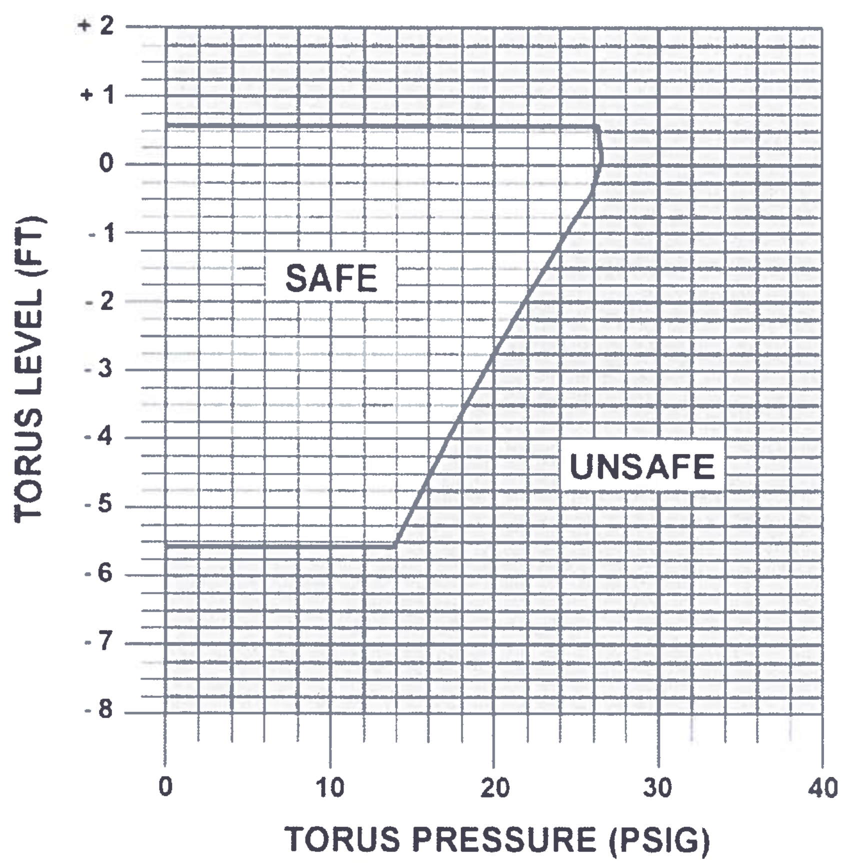
Recap	**(As Required)**
	☐ Update the crew as needed:
	☐ Describe what happened and major actions taken
	☐ Procedures in-progress
	☐ Notifications:
	    ☐ Maintenance
	    ☐ Engineering
	    ☐ Others (Dispatcher, Station Management, etc.)
	☐ Future Direction and priorities
	☐ Discuss any contingency plans

Input	**(As Required)**
	☐ Solicit questions/concerns from each crew member:
	    ☐ ROs
	    ☐ CRS
	    ☐ STA
	☐ Are there any alarms unexpected for the plant conditions?
	☐ What is the status of Critical Parameters?

EAL	**(As Required)** ☐ Provide EAL and potential escalation criteria
Finish Brief	☐ Restore normal alarm announcement? (Yes/No) ☐ Announce "End of Brief"

ENCLOSURE 2

<< Pressure Suppression Pressure >>



ENCLOSURE 3

<< Emergency Suppression Pool Cooling Using Loop A (2OP-17) >>

NOTE

This attachment is not to be used for normal system operations. .... ☐

<u>Start RHR SW A LOOP (CONV)</u>

- **Open** SW-V101 ☐
- **Close** SW-V143 ☐
- **Start** CSW PUMPS AS NEEDED ☐
- **<u>IF</u>** LOCA SIGNAL IS PRESENT, **<u>THEN</u> place** RHR SW BOOSTER PUMPS A & C LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- **Start** RHR SW PMP ☐
- **Adjust** E11-PDV-F068A ☐
- **Establish** CLG WTR TO VITAL HDR ☐
- **Start** ADDITIONAL RHR SW PUMP and **adjust** FLOW AS NEEDED ☐

<u>Start RHR SW A LOOP (NUC)</u>

- **Open** SW-V105 ☐
- **Open** SW-V102 ☐
- **Close** SW-V143 ☐
- **Start** PUMPS ON NSW HDR AS NEEDED ☐
- **<u>IF</u>** LOCA SIGNAL IS PRESENT, **<u>THEN</u> place** RHR SW BOOSTER PUMPS A & C LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- **Start** RHR SW PMP ☐
- **Adjust** E11-PDV-F068A ☐
- **Establish** CLG WTR TO VITAL HDR ☐
- **Start** ADDITIONAL RHR SW PUMP and **adjust** FLOW AS NEEDED ☐

<u>Start RHR LOOP A</u>

- **<u>IF</u>** LOCA SIGNAL IS PRESENT, **<u>THEN</u>** ☐ **Verify** COOLING LOGIC IS MADE UP
- **<u>IF</u>** E11-F015A IS OPEN, **<u>THEN</u> close** E11-F017A ☐
- **Start** LOOP A RHR PMP ☐
- **Open** E11-F028A ☐
- **Throttle** E11-F024A ☐
- **Throttle** E11-F048A ☐
- **Start** ADDITIONAL LOOP A RHR PMP and **adjust** FLOW AS NEEDED ☐

2 2/1061
2 S/1062

ENCLOSURE 3

<< Emergency Suppression Pool Cooling Using Loop B (2OP-17) >>

NOTE

This attachment is not to be used for normal system operations. .... ☐

Start RHR SW B LOOP (NUC)

Open SW-V105 ☐

Close SW-V143 ☐

Start CSW PUMPS AS NEEDED ☐

IF LOCA SIGNAL IS PRESENT,
THEN place RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐

Start RHR SW PMP ☐

Adjust E11-PDV-F068B ☐

Establish CLG WTR TO VITAL HDR ☐

Start ADDITIONAL RHR SW PUMP and **adjust** FLOW AS NEEDED ☐

Start RHR SW B LOOP (CONV)

Open SW-V101 ☐

Open SW-V102 ☐

Close SW-V143 ☐

Start PUMPS ON NSW HDR AS NEEDED ☐

IF LOCA SIGNAL IS PRESENT,
THEN place RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐

Start RHR SW PMP ☐

Adjust E11-PDV-F068B ☐

Establish CLG WTR TO VITAL HDR ☐

Start ADDITIONAL RHR SW PUMP and **adjust** FLOW AS NEEDED ☐

Start RHR LOOP B

IF LOCA SIGNAL IS PRESENT,
THEN ☐
Verify COOLING LOGIC IS MADE UP

IF E11-F015B IS OPEN,
THEN close E11-F017B ☐

Start LOOP B RHR PMP ☐

Open E11-F028B ☐

Throttle E11-F024B ☐

Throttle E11-F048B ☐

Start ADDITIONAL LOOP A RHR PMP and **adjust** FLOW AS NEEDED ☐

ENCLOSURE 4

Feedwater Level Control Following a Reactor Scram (EOP)

NOTE: This attachment is not to be used for routine system operation

1. **ENSURE** the following:
   - FW-V6 **AND** FW-V8 **OR** FW-V118 **AND** FW-V119 closed .... ☐
   - FW-FV-177 closed .... ☐
   - FW-V120 closed .... ☐
   - FW control MODE SELECT in 1 ELEM .... ☐
   - SULCV in M (MANUAL) closed .... ☐
   - B21-F032A **AND/OR** B21-F032B open .... ☐
2. **PLACE** the MSTR RFPT SP/RX LVL CTL in M (MANUAL), THEN: .... ☐
   - **ADJUST** to 187" .... ☐
3. **IF** any RFP is running, **THEN**:
   a. **PLACE** RFP A(B) Recirc Vlv, control switch to open .... ☐
   b. **PLACE** RFPT A(B) SP CTL in M (MANUAL) .... ☐
4. **IF** no RFP is running, **THEN**:
   a. **PLACE** RFP A(B) RECIRC VLV, control switch to open .... ☐
   b. **ENSURE** the following:
      - FW-V3(V4) [RFP A(B) Disch Vlv] open .... ☐
      - RFPT A(B) SP CTL in M (MANUAL) at lower limit .... ☐
      - RFPT A(B) Man/DFCS control switch in MAN .... ☐
      - Reactor water level is less than +206 inches **AND** RFPT A&B HIGH LEVEL TRIP reset .... ☐
   c. **DEPRESS** RFPT A(B) RESET .... ☐

ENCLOSURE 4 Page 2 of 2

Feedwater Level Control Following a Reactor Scram (EOP)

d. **ENSURE** RFPT A(B) LP **AND** HP STOP VLVS open .... ☐

e. **ROLL** RFPT A(B) to 1000 rpm by depressing RFP A(B) START .... ☐

f. **RAISE** RFPT A(B) to at least 2550 rpm using the LOWER/RAISE control switch .... ☐

g. **DEPRESS** RFPT A(B) DFCS CTRL RESET .... ☐

5. **ENSURE** MAN/DFCS control switch in DFCS .... ☐

6. **RAISE** RFPT A(B) SP CTL speed until discharge pressure is greater than or equal to 100 psig above reactor pressure .... ☐

7. **ADJUST** SULCV to establish desired injection .... ☐

8. **IF** desired, **THEN PLACE** SULCV in A (AUTO) .... ☐

9. **IF** needed, **THEN THROTTLE** FW-V120 .... ☐

10. **IF** needed, **THEN GO TO** 2OP-32, Condensate And Feedwater System Operating Procedure, for level control .... ☐

ATTACHMENT 1 - Scenario Quantitative Attribute Assessment

Category	NUREG 1021 Rev. 2 Supp. 1 Req.	Scenario Content
Total Malfunctions	5-8	8
Malfunctions after EOP Entry	1-2	2
Abnormal Events	2-4	4
Major Transients	1-2	2
EOPs Used	1-2	2
EOP Contingency	0-2	1
Run Time	60-90 min	90
Crew Critical Tasks	2-3	2
Tech Specs	2	2
Instrument / Component Failures before Major	2 – OATC 2 - BOP	4
Instrument / Component Failures after Major	2	2
Normal Operations	1	1
Reactivity manipulation	1	1

ATTACHMENT 2 – Shift Turnover

Brunswick Unit 2 Plant Status					
Station Duty Manager:	E. Neal			Workweek Manager:	B. Craig
Mode:	1	Rx Power:	100%	Mode:	1
Plant Risk: Current EOOS Risk Assessment is:				Green	
SFP Time to 200 Deg F:	49.7 hrs			Days Online:	80 days
Turnover:	Feedwater Temperature Reduction will be implemented this weekend				
Protected Equipment:	2A FPC Pump/Hx, 2A RCC Pump, and 2C Demin Transfer Pump for Fuel Pool Decay Heat Removal and inventory makeup. 2A/B NSW Pumps due to 1A NSW pump maintenance				
Comments:	1A NSW Pump is under clearance for planned maintenance. 2C TCC Pump is in service on Unit One. The BOP is to start CREV in the area high radiation mode for inspection testing IAW 0OP-37, Section 6.1.3. (The inspection is scheduled to take three hours)				

6.1.3 Manual Startup of the Control Building Emergency Recirculation System

1. **Confirm** the following initial conditions are met:

   - All applicable prerequisites as listed in Section 5.0 are met........________
   - The Control Building Emergency Recirculation System has failed to start after an initiation signal, ...________

   or

   - A manual start in accordance with 0AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity, is required {8.1.1}..________

   or

   - Surveillance or inspection tests are required.............................________

NOTE

- Indications for the Control Building Ventilation System are located on Panel XU-3 on both units.. ☐
- Controls for the Mechanical Equipment Room Ventilation Fans and the Control Building Wash Room Exhaust Fan are on XU-3 on Units 1 and 2............................... ☐
- Controls for the Cable Spread Room ventilation fans are on Panel XU-3 for the respective unit. .... ☐

2. **Perform** the following to place the Control Building Emergency Recirculation System in the area high radiation mode (includes Secondary Containment Isolation):

NOTE

- Placing <u>one</u> of the 2A(B)-ERF-CB (CB Emerg Recirc Fans) in ON will INOP the automatic start function of the non-operating fan. .... ☐
- Controls for the Control Building Emergency Recirculation Fans are on Panel XU-3 on Unit 2.. ☐

   a. **Place** <u>one</u> of the 2A(B)-ERF-CB (CB Emerg Recirc Fans) in ON...________

6.1.3 Manual Startup of the Control Building Emergency Recirculation System (continued)

CAUTION

Detection of heat in the charcoal bed, detectors 2-FP-CB-4-20 and 2-FP-CB-4-21 for A or detectors 2-FP-CB-4-14 and 2-FP-CB-4-15 for B, will trip the associated Emergency Recirculation Fan. .... ☐

b. **Confirm** 2L-D-CB (Ctl RM Norm Mu Air Dmpr) closes. .... ________

c. **Confirm** VA-2J-D-CB (CB Emerg Recirc Damper) opens. .... ________

d. **Stop** 2D-EF-CB (CB Washroom Exhaust Fan) and **confirm** associated damper closes. .... ________

NOTE

The Control Building Mechanical Equipment Room Vent Fans can be stopped only by simultaneously placing both Units' control switches in OFF. .... ☐

e. Simultaneously **place** both Units' control switches in OFF, for 2F-SF-CB and 2E-EF-CB (CB Mechanical Equip Room Vent Fans) to stop the fans and **confirm** associated supply and exhaust dampers close. .... ________

f. **Stop** 2A-SF-CB and 2A-EF-CB (Cable Spread Room 2 Vent Fans) and **confirm** associated supply and exhaust dampers close. .... ________

g. **Stop** 1A-SF-CB and 1A-EF-CB (Cable Spread Room 1 Vent Fans) and **confirm** associated supply and exhaust dampers close. .... ________

6.1.3 Manual Startup of the Control Building Emergency Recirculation System (continued)

NOTE
The Control Building Emergency Recirculation System is now in operation for high radiation conditions. .... ☐

3. **Perform** the following to place the Control Building Emergency Recirculation System in the fire mode:

NOTE
Placing <u>one</u> of the 2A(B)-ERF-CB (CB Emerg Recirc Fans) in ON will INOP the automatic start function of the non-operating fan. .... ☐

a. **Place** <u>one</u> of the 2A(B)-ERF-CB (CB Emerg Recirc Fans) in ON. .... _________

CAUTION
Detection of heat in the charcoal bed, detectors 2-FP-CB-4-20 and 2-FP-CB-4-21 for A or detectors 2-FP-CB-4-14 and 2-FP-CB-4-15 for B, will trip the associated Emergency Recirculation Fan. .... ☐

b. **Confirm** 2L-D-CB (Ctl RM Norm Mu Air Dmpr) closes. .... _________

c. **Confirm** VA-2J-D-CB (CB Emerg Recirc Damper) opens. .... _________

d. **Stop** 2D-EF-CB (CB Washroom Exhaust Fan) and **confirm** associated damper closes. .... _________

NOTE
The Control Building Emergency Recirculation System is now in operation for fire conditions. .... ☐

4. **<u>WHEN</u>** the initiating conditions have cleared,
<u>THEN</u> place Control Building Ventilation System in operation in accordance with Section 6.1.4. .... _________

6.1.3 Manual Startup of the Control Building Emergency Recirculation System (continued)

Date/Time Completed ____________________

Performed By (Print)	Initials

Reviewed By ____________________

Unit CRS/SRO

BRUNSWICK TRAINING SECTION

OPERATIONS TRAINING

INITIAL LICENSED OPERATOR

SIMULATOR EVALUATION GUIDE

2016 NRC SCENARIO 5

PLACE RFP IN AUTO, DIFF TO MOVE ROD, SPE TRIP, IRM FAILURE, DG3/E3/E7 CP LOSS, LOWERING TORUS LEVEL, RHR/CS FAILURES, ED (TORUS LVL)

REVISION 0

Developer: Bob Bolin	**Date:** 07/07/2016
Technical Review: Dan Hulgin	**Date:** 9/12/2016
Validators: Dwayne Wolf Shawn Zander Hunter Morris	**Date:** 09/08/16
Facility Representative: Craig Oliver	**Date:** 09/22/2016

REVISION SUMMARY	
0	Scenario developed for 2016 NRC Exam.

TABLE OF CONTENTS

1.0 SCENARIO OUTLINE

Event	Malf. No.	Type*	Event Description
1		N-BOP	Place 2A RFPT level control in automatic
2		R-ATC	Raise reactor power using control rods
3	RD032M	C-ATC C-CRS	Difficult to move control rod (AOP)
4	K4510C	C-BOP C-CRS	Steam Packing Exhauster Trip
5	NI018F	C-ATC C-CRS	IRM Failure (TS)
6	ED_IADCGJ6	C-BOP C-CRS	DG3 / E3 / E7 Control Power loss (AOP)(TS)
7	CA002F	M C	Lowering Torus Level / RHR F028A mech trip / RHR F024B thermal trip / CS F020A Handwheel broke (PCCP)
8	RP008F	M	Scram / Emergency Depressurization (RSP)(ATWS)(EDP)
*(N)ormal, (R)eactivity, (C)omponent or Instrument, (M)ajor			

2.0 SCENARIO DESCRIPTION SUMMARY

Event	Description
1	Step 6.3.46 of 0GP-02, Approach to Criticality and Pressurizations of the Reactor will be completed starting at Step 6.3.46.
2	The crew will raise power by pulling control rods in preparation for placing the Mode switch to RUN. Rod pulls will commence at Step 161 (42-39 @ 12) of the A2X sequence.
3	Control rods will continue to be withdrawn raising power. When control rod 42-23 is selected for withdrawal, it will be stuck at position 12. AOP-02 may be entered and 2OP-07, Section 8.2 actions are required to withdraw a difficult intermediate control rod.
4	SPE 2A will trip causing a loss of gland sealing header pressure. SPE 2B will be placed in service
5	While withdrawing control rods, IRM C will fail upscale causing a rod block and half scram. SRO will address IRM A and C inoperability IAW TS 3.3.1.1. Once addressed, I&C will report IRM A is ready to be returned to service following proper channel check. The crew will take the actions of the APP and bypass IRM C and reset the half scram.
6	DC Panel 2A will trip resulting in loss of control power to DG3, Bus E3 and Bus E7. The crew will respond per 0AOP-39.0 and transfer the control power to alternate. DG3, Bus E3 and Bus E7 are inoperable until transferred to alternate supply. Once control power is transferred, a 7 day action is required to restore to the normal source. The BOP operator will return DG3 to AUTO IAW AOP-39.0.
7	Torus level will begin to lower due to an unisolable leak on RHR suction. If attempted to raise torus water level, on RHR A loop the E11-F028A (Torus Discharge Isol Vlv) will trip when opened, on RHR B loop the E11-F024B (Torus Cooling Isol Vlv) will thermal trip when opened, and on Core Spray the E21-F002A (Core Spray Pump A Suction Valve From The Condensate Storage Tank) handwheel will be broke.
8	Before level reaches -5.5 feet in the torus a reactor scram is required. When torus water level reaches -5.5 feet emergency depressurization is required. The crew can anticipate emergency depressurization.

3.0 CREW CRITICAL TASKS

Critical Task #1
Scram the reactor before torus water level drops below -5.5 feet.
Critical Task #2
Emergency Depressurize the reactor when torus water level reaches -5.5 feet. OR Anticipate Emergency Depressurization of the reactor when all attempts to fill the torus have failed.

4.0 TERMINATION CRITERIA

When all rods are inserted and the reactor has been depressurized to less than 100 psig the scenario may be terminated.

5.0 IMPLEMENTING REFERENCES

NOTE: Refer to the most current revision of each Implementing Reference.

Number	Title
UA-02, 4-5	GLAND SEAL VACUUM LOSS
2OP-26.1, Section 8.1	SHIFTING STEAM PACKING EXHAUSTERS
A-05, 2-4	IRM UPSCALE
A-05, 3-4	IRM A UPSCALE/INOP
A-05, 1-7	REACTOR AUTO SCRAM SYS A
A-05, 4-7	NEUT MON SYS TRIP
A-05, 2-2	ROD OUT BLOCK
UA-17, 2-3	DG-3/E3 ESS LOSS OF NORM POWER
UA-19, 6-3	DG-1 CTL PWR SUPPLY LOST
UA-21, 6-2	DG-3 LO START AIR PRESS
UA-21, 6-3	DG-3 CTL POWER SUPPLY LOST
0AOP-39.0	LOSS OF DC POWER
A-01, 3-7	SUPPRESSION CHAMBER LVL HI/LO
A-05, 5-5	PRI CMT HI/LO PRESS
0EOP-01-SEP-18	FILLING THE TORUS
0EOP-01-SEP-15	ANTICIPATE EMERGENCY DEPRESSURIZATION

6.0 SETUP INSTRUCTIONS

1. **PERFORM** TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 5, Checklist for Simulator Exam Security.
2. **RESET** the Simulator to IC-06.
3. **ENSURE** the RWM is set up as required for the selected IC.
4. **ENSURE** appropriate keys have blanks in switches.
5. **RESET** alarms on SJAE, MSL, and RWM NUMACs.
6. **ENSURE** no rods are bypassed in the RWM.
7. **PLACE** all SPDS displays to the Critical Plant Variable display (#100).
8. **ENSURE** hard cards and flow charts are cleaned up
9. **TAKE** the SIMULATOR OUT OF FREEZE,
10. **CLOSE** the CS B Loop valves
11. **LOAD** Scenario File.
12. **ALIGN** the plant as follows:

Manipulation
Insert control rods up to Step 160 of GP-10, Sequence A2X is completed. Raise pressure set to 900 psig Verify level is stable Verify drive water pressure is at 260 psid

13. **IF desired**, take a **SNAPSHOT** and save into an available IC for later use.
14. **PLACE** a clearance on the following equipment.

Component	Position
IRM A (Blue Tag)	Bypassed
Core Spray Loop B	Red Tag

15. **INSTALL** Protected Equipment signage and **UPDATE** RTGB placard as follows:
    a. ADHR / FPC/ Demin Transfer Pump
    b. All remaining LP ECCS systems

16. **ENSURE** each Implementing References listed in Section 7 is intact and free of marks.

17. **ENSURE** all materials in the table below are in place and marked-up to the step identified.

Required Materials
0GP-02 up to Step 6.3.46
0GP-10 up to step 161

18. **ADVANCE** the recorders to prevent examinees from seeing relevant scenario details.

19. **PROVIDE** Shift Briefing sheet for the CRS.

20. **VERIFY** all actions contained in TAP-409, Miscellaneous Simulator Training Guidelines, Attachment 4, Simulator Training Instructor Checklist, are complete.

7.0 INTERVENTIONS

TRIGGERS

Trig	Type	ID
4	DI Override	K4510C - [STM PACKING EXHAUSTER A CLOSE DI]
4	DI Override	K4510C - [STM PACKING EXHAUSTER A CLOSE DI]
4	DI Override	K4510C - [STM PACKING EXHAUSTER A CLOSE DI]
5	Malfunction	NI018F - [IRM C FAILS HI]
6	Remote Function	ED_IADCGJ6 - [LOAD BKR GJ6 SBD 2A TO 125V P 2A (DG)]
7	Remote Function	ED_IADCAPD3 - [DG-3 DC BKR CTL PWR ON/OFF]
7	Remote Function	EG_0003 - [DG-3 LOCKOUT RESET]
7	Remote Function	ED_IADCADG3 - [DG-3 DC BKR CTL PWR (NML=2A ALT=U1)]
8	Remote Function	ED_IADCABE3 - [SWGR E3 DC BKR CTL PWR (NML=2A ALT=U1)]
10	Malfunction	CA002F - [TORUS WATER LEAK]
11	Trigger Command	DOD:Q1217LGN
12	Trigger Command	DOD:Q1707LGN

Trig #	Trigger Text
11	K1217ENN - [TORUS ISO VLV E11-F028A]
12	K1707JNN - [FULL FLOW VLV E11-F024B]

MALFUNCTIONS

Malf ID	Mult ID	Description	Current Value	Target Value	Rmp time	Actime	Dactime	Trig
RD032M	42-23	CONTROL ROD WITHDRAWAL SLUGGISH	True	True				
NI018F		IRM C FAILS HI	False	True				5
CA002F		TORUS WATER LEAK	False	True				10
RD012M	42-23	STUCK CONTROL ROD	True	True				

REMOTES

Remf Id	Mult Id	Description	Current Value	Target Value	Rmp time	Actime	Trig
_IABKCF06		BKR CTL DC FUSES CORE SPRAY PUMP 2B	OUT	OUT			
CS_ZVCS31BT		E21-F031B MIN FLOW	OFF	OFF			
CS_ZVCS15BT		E21-F015B FULL FLOW TEST	OFF	OFF			
CS_ZVCS05BT		E21-F005B INBD INJ VLV	OFF	OFF			
CS_ZVCS04BM		E21-F004B OTBD INJ VLV	OFF	OFF			
CS_ZVCS01BT		E21-F001B TORUS SUCTION	OFF	OFF			
CS_VHCS10B		E21-F010B OPEN/CLOSE	CLOSE	CLOSE			
ED_IADCGJ6		LOAD BKR GJ6 SBD 2A TO 125V P 2A (DG)	CLOSE	OPEN			6
ED_IADCADG3		DG-3 DC BKR CTL PWR (NML=2A ALT=U1)	NORMAL	ALT			7
ED_IADCABE3		SWGR E3 DC BKR CTL PWR (NML=2A ALT=U1)	NORMAL	ALT			8
EG_0003	DG-3	DG-3 LOCKOUT RESET	NORMAL	RESET		00:00:02	7
RH_ZVRH24BT		E11-F024B FULL FLOW TEST	OFF	OFF			
RH_ZVRH28AM		E11-F028A TORUS ISOLATION	OFF	OFF			
ED_IADCAPD3		DG-3 DC BKR CTL PWR ON/OFF	ON	ON		00:00:01	7

PANEL OVERRIDES

Tag ID	Description	Position / Target	Actual Value	Override Value	Rmp time	Actime	Dactime	Trig
K4510C	STM PACKING EXHAUSTER A CLOSE DI	NORMAL	ON	OFF				4
K4510C	STM PACKING EXHAUSTER A CLOSE DI	START	OFF	OFF				4
K4510C	STM PACKING EXHAUSTER A CLOSE DI	STOP	OFF	ON				4
Q1217LGN	TORUS ISO VLV E11-F028A GREEN	ON/OFF	OFF	ON				
Q1707LGN	FULL FLOW E11-FO24B GREEN	ON/OFF	OFF	ON				

OPERATOR RESPONSE AND INSTRUCTIONAL STRATEGIES

EVENT 1: PLACING 2A RFPT CONTROLLER IN AUTOMATIC	
Simulator Operator Actions	
	Ensure Monitored Parameters is open and Scenario Based Testing Variables are loaded.

Simulator Operator Role Play	

Evaluator Notes	
Plant Response:	Place RFPT Master Controller in Automatic IAW 0GP-02, Step 6.3.46
Objectives:	SRO – Direct BOP to perform Step 6.3.46 of 0GP-02 BOP – Place RFPT Level Controller is placed in Automatic ATC – Monitors plant
Success Path:	RFPT Master Level Controller is in Automatic and Reactor water level is controlled in band.
Event Termination:	Go to Event 2 at the direction of the Lead Evaluator.

EVENT 1: PLACING 2A RFPT CONTROLLER IN AUTOMATIC

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct BOP to perform Step 6.3.46 of 0GP-02	
	RO	Monitors the plant	
	BOP	Place RFPT Master Controller in Automatic IAW 0GP-02, Step 6.3.46.	

6.3 Heating And Pressurization Of The Reactor (continued)

e. B21-F019 (Main Steam Line Drain Otbd Isol Vlv) .... ____/____
IV

f. B21-F016 (Main Steam Line Drain inbd Isol Vlv) .... ____/____
IV

46. **WHEN** reactor feed pump discharge pressure is greater than 900 psig, **THEN place** C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl) in A (automatic) as follows: .... ____

a. **Ensure** C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl), in M (manual) .... ____

b. **Ensure** Feedwater Control Mode Select in 1 ELEM .... ____

c. **Depress** SEL pushbutton on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] until A(B) BIAS is indicated and **ensure** bias is set to 0% .... ____

d. **Depress** SEL pushbutton on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] until PMP A(B) DEM is displayed .... ____

e. **Depress** SEL pushbutton on C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl), until MASTR DEM is displayed .... ____

f. Using the raise and lower pushbuttons on C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl), **set** MASTR DEM to equal the PMP A(B) DEM value displayed on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] .... ____

g. **Depress** A/M pushbutton on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] and **confirm** the following:

- Indicator on control station changes to A (automatic) .... ____
- PMP DEM signal remains unchanged .... ____

h. **Depress** SEL pushbutton on the out-of-service C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] until LVL ERROR is indicated and **confirm** LVL ERROR is approximately 0 inches .... ____

i. **Depress** A/M pushbutton on C32-SIC-R600 (Mstr RFPT Sp/Rx Lvl Ctl) and **confirm** the indicator on the control station changes to A (automatic) .... ____

6.3 Heating And Pressurization Of The Reactor (continued)

j. **Confirm** signals for PMP A(B) DEM on C32-SIC-R601A(B) [RFPT A(B) Sp Ctl] and VALVE DEM on FW-LIC-3269 (SULCV Ctl) remain unchanged .... ________

k. **Depress** A/M pushbutton on FW-LIC-3269 (SULCV Ctl) and **confirm** the indicator on the control station changes to M (manual) .... ________

CAUTION

Momentarily depressing the raise or lower pushbuttons on FW-LIC-3269 (SULCV Ctl) will cause valve demand to change in increments of 0.1%. Continually depressing the raise or lower pushbuttons will cause valve demand to change at an exponential rate. .... ☐

l. Using raise pushbutton on FW-LIC-3269 (SULCV Ctl), slowly **open** the SULCV until VALVE DEM is 100% .... ________

m. **Confirm** reactor water level is being maintained between 182 and 192 inches .... ________

EVENTS 2/3: RAISE REACTOR POWER / DIFF TO MOVE ROD	
Simulator Operator Actions	

Simulator Operator Role Play	
	If asked as the RE, continuous rod withdrawal is allowed.

Evaluator Notes	
Plant Response:	Control rods will continue to be withdrawn until control rod 42-23 which is difficult to move, requires OP-07 actions to move.
Objectives:	SRO - Directs and monitor reactor power ascension with control rods Direct actions for a difficult to move control rod. RO - Withdraw control rods to raise reactor power Perform 2OP-07 actions for difficult to move control rod
Success Path:	Control rod 42-23 withdrawn to position 48 by use of increase drive water DP.
Event Termination:	Go to Event 4 at the direction of the Lead Evaluator.

EVENTS 2/3: RAISE REACTOR POWER / DIFF TO MOVE ROD

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Directs RO to continue to raise reactor power by withdrawing control rods. (Continuous withdrawal allowed).	
		Directs RO to perform 2OP-07.	
		May direct AOP-02 (Control Rod malfunction) – Provides notifying RE and Using 2OP-07 to move rod.	
		May conduct a brief (see Enclosure 1 on page 56 for format)	
	BOP	Monitor reactor plant parameters during evolution.	
	ATC	Continues rod withdrawal per GP-10 (see page 18) IAW guidance of 2OP-07 (see page 20).	
		Report A-6 2-7 *APRM DOWNSCALE* annunciator clears.	
		Recognizes control rod 42-23 will not move.	
		Notifies SRO control rod 42-23 will not move.	
		Identifies 2OP-07, Reactor Manual Control System Operating Procedure, Section 6.3.2 (Control Rod Difficult to Withdraw, Control Rod NOT at Position 00) is required. (page 23)	
		Continues rod withdrawal per GP-10 IAW guidance of 2OP-07.	

ATTACHMENT 3A
Page 2 of 26
Rod Sequence A2X Withdraw Check Off Sheet (Expanded Group A2)

NOTES:

1. Concurrent verification of rod selection is required PRIOR to rod movement.

2. The initials in "O. T." column verify and document the following control rod coupling integrity checks have been performed:

   WHEN a control rod is withdrawn to the FULL OUT position, a continuous withdraw signal has been maintained for at least 3 to 5 seconds, **OR** a separate notch out signal has been applied, **AND**

   - *ROD OVER TRAVEL (A-05, 4-2)* annunciator does **NOT** alarm
   - *ROD DRIFT (A-05 3-2)* annunciator does **NOT** alarm
   - The Full Out light indication for the selected control rod is not lost
   - The four-rod display indicates 48 for the selected control rod

3. Initials in the "Rod P. I." column confirm that the rod position indications for those positions covered by that item of the check off sheet are operable. The RWM Inferred Rod Position capability may be used as an alternate method to determine Rod Position.

4. During manipulation of control rods a second Licensed Operator shall monitor control rod selection and movement. This individual shall ensure correct placement of control rods, and document these verifications by initialing the Rod Sequence Check Off Sheet. **IF** inoperable Rod Position indication necessitates inserting the rod in question one notch further than its insert/withdraw limit and bypassing the rod on the RWM, **THEN** the second Licensed Operator's initials also documents verification of this action.

R8 5. Any deviation from the original rod move sequence should be reviewed by the Reactor Engineer, authorized by the Unit CRS, and documented on the proper rod sequence check off sheet. For changes in direction or control rod double notches, the affected page(s) of the sequence pull sheet must be copied, rod move documented, then the documentation must be included with the original rod sequence attachment.

ATTACHMENT 3A
Page 10 of 26
Rod Sequence A2X Withdraw Check Off Sheet (Expanded Group A2)

Item	Rod Number	Correct Rod Selected And Verified [Note 1]	Position From/To	Actual Position	Initials	Over Travel [Note 2]	Rod P.I. [Note 3]	Initials [Note 4]	Comments
STEP 9 (BPWS 4)									
137	50-31	/	08 to 12			N/A			
138	42-07	/	08 to 12			N/A			
139	10-07	/	08 to 12			N/A			
140	02-31	/	08 to 12			N/A			
141	10-39	/	08 to 12			N/A			
142	18-47	/	08 to 12			N/A			
143	34-47	/	08 to 12			N/A			
144	42-39	/	08 to 12			N/A			
145	42-23	/	08 to 12			N/A			
146	34-15	/	08 to 12			N/A			
147	26-07	/	08 to 12			N/A			
148	18-15	/	08 to 12			N/A			
149	10-23	/	08 to 12			N/A			
150	18-31	/	08 to 12			N/A			
151	26-39	/	08 to 12			N/A			
152	34-31	/	08 to 12			N/A			
153	26-23	/	08 to 12			N/A			
STEP 10 (BPWS 4)									
154	50-31	/	12 to 48						
155	42-07	/	12 to 48						
156	10-07	/	12 to 48						
157	02-31	/	12 to 48						
158	10-39	/	12 to 48						
159	18-47	/	12 to 48						
160	34-47	/	12 to 48						
161	42-39	/	12 to 48						
162	42-23	/	12 to 48						
163	34-15	/	12 to 48						
164	26-07	/	12 to 48						
165	18-15	/	12 to 48						
166	10-23	/	12 to 48						
167	18-31	/	12 to 48						
168	26-39	/	12 to 48						
169	34-31	/	12 to 48						
170	26-23	/	12 to 48						

Notes (for further details, see Page 2 of this attachment):

1. Concurrent Verification of rod selection is required PRIOR to rod movement.
2. Initials in the "Over Travel" column signify completion of control rod coupling integrity checks for fully withdrawn control rods.
3. Initials in the "Rod P. I." column confirm that the rod position indications for those positions covered by that item of the check off sheet are operable.
4. Column used by a second Licensed Operator to document monitoring of control rod selection and movement to ensure correct placement of control rods.

<< (Reference Use) - Section 6.1.1 Continuous Control Rod Withdrawal >>

NOTE

The purpose of this attachment is to provide the Reactor Operator with guidance for control rod movement and use 0ENP-24.5, Reactivity Control Planning, and General Operating Procedure pull sheets as the place keeping tool for execution of steps. .... ☐

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

1. **Select** control rod by depressing its Control Rod Select pushbutton.

2. **Confirm** the following:

   - The backlighted Control Rod Select pushbutton is brightly ILLUMINATED.
   - The white indicating light on the full core display is ON.
   - Rod Withdrawal Permissive indication is ON

3. Continuously **withdraw** control rod to position designated on General Operating Procedure or 0ENP-24.5, Reactivity Control Planning, pull sheets by holding Emergency Rod In Notch Override switch to OVERRIDE, while simultaneously holding Rod Movement switch to NOTCH OUT.{8.1.2}

4. **Monitor** control rod position and nuclear instrumentation while withdrawing the control rod.

5. <u>**IF**</u> control rod fails to withdraw,
   <u>**THEN**</u> **go to** Section 6.3.1, Section 6.3.2, Section 6.3.7, or Section 6.3.8 to free the control rod and **return to** Attachment 15 Step 6.

<< (Reference Use) - Section 6.1.1 Continuous Control Rod Withdrawal >>

6. **IF** the control rod is being withdrawn to an intermediate position **THEN perform** the following:

   a. Before control rod reaches the position designated on General Operating Procedure or 0ENP-24.5, Reactivity Control Planning, pull sheets, **release** Rod Movement and Emergency Rod In Notch Override control switches.{8.1.2}.

   b. **Ensure** control rod settles into desired position.

   c. **Confirm** rod settle light is OFF.

7. **IF** the control rod is being fully withdrawn to position "48" **THEN perform** the following:

NOTE

A continuous withdraw signal of approximately 3 to 5 seconds is sufficient time to ensure the control rod remains coupled. Longer continuous withdraw signals may be utilized if a control rod flush is desired. .... ☐

   a. **WHEN** control rod reaches position "48", **THEN perform** either of the following:

      - **Maintain** a continuous withdraw signal for the desired time.

      - **Apply** a separate notch withdraw signal.

   b. **Confirm** control rod does **NOT** retract beyond position "48" (Technical Specification SR 3.1.3.4).

   c. **Release** Rod Movement and Emergency Rod In Notch Override switches, if used.

   d. **Ensure** control rod settles at position "48".

   e. **Confirm** rod settle light is OFF.

   f. **Confirm** control rod reed switch position indicators agree with FULL OUT indication on full core display.

<< (Reference Use) - Section 6.1.1 Continuous Control Rod Withdrawal >>

8. **Repeat** Attachment 15 Step 1 through Attachment 15 Step 7.f. of this Attachment, for the remainder of the control rods requiring movement, using General Operating Procedure or 0ENP-24.5, Reactivity Control Planning, pull sheets.{8.1.2}.

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

9. **WHEN** control rod movement is **NO** longer required **THEN go to** Section 6.1.1 Step 7.

6.3.2 Control Rod Difficult To Withdraw And Control Rod NOT At Position 00

Control Rod ________________

1. **Record** Control Rod Number above. .... ________

2. **Confirm** the following initial conditions are met:

   - All applicable prerequisites in Section 5.0 are met. .... ________

   - Control rod will **NOT** withdraw in accordance with Section 6.1.1. .... ________

   - Control rod is **NOT** at position "00". .... ________

   - Unit CRS has consulted Technical Specifications 3.1.3, Control Operability, and 3.3.2.1 Control Rod Block Instrumentation for the required actions prior to the performance of Section 6.3.2, Control Rod Difficult To Withdraw And Control Rod NOT At Position 00 .... ________
     CRS

3. **Ensure** failure of the control rod to withdraw is **NOT** the result of a rod block from the RWM or RBM. .... ________

4. Notify the Reactor Engineer. .... ________

____
Reactor Engineer

CAUTION

If reactor pressure is less than or equal to 800 psig and higher than normal CRD drive water pressure is used to withdraw a control rod, then the latching function of the CRD may be lost. .... ☐

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

5. **Attempt** to withdraw the control rod using 300 psid drive header differential pressure as follows:

   a. **Raise** CRD drive differential pressure to 300 psid .... ________

   b. **Attempt** to withdraw control rod .... ________

6.3.2 Control Rod Difficult To Withdraw And Control Rod NOT At Position 00 (continued)

c. **IF** control rod moves,
THEN immediately **restore** drive pressure to 260 to 275 psid and **attempt** to withdraw rod in accordance with Section 6.1.1. ________

- **IF** control rod will **NOT** continue to withdraw at normal drive pressure,
THEN return drive differential pressure to 300 psid and **withdraw** rod in accordance with Section 6.1.1. ________

d. **IF** control rod withdraws,
THEN go to Section 6.3.2 Step 16. ________

e. **Repeat** Section 6.3.2 Step 5.b and Section 6.3.2 Step 5.c, as necessary. ________

6. **Attempt** to withdraw control rod using 350 psid drive header differential pressure as follows:

a. **Raise** CRD drive differential pressure to 350 psid ________

b. **Attempt** to withdraw control rod ________

c. **IF** control rod moves,
THEN immediately **restore** drive pressure to 260 to 275 psid and **attempt** to withdraw rod in accordance with Section 6.1.1. ________

- **IF** control rod will **NOT** continue to withdraw at normal drive pressure,
THEN return drive differential pressure to 350 psid and **withdraw** the rod in accordance with Section 6.1.1. ________

d. **IF** control rod withdraws,
THEN go to Section 6.3.2 Step 16. ________

16. **Lower** control rod drive differential pressure to between 260 and 275 psid ________

EVENT 4: STEAM PACKING EXHAUSTER (SPE) TRIP	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 4** to trip the A SPE.

Simulator Operator Role Play	
	If contacted as I&C to investigate, acknowledge the request.
	If asked to investigate MCC 2TA for the SPE, report that compartment CA6, OG-SPEM-A (Steam Seal SPE Motor 2A) is tripped.
	If asked as AO to Open 2-MVD-V52 float trap outlet valve for 2B SPE report that the valve is Open.
	If asked as AO to Close 2-MVD-V51 float trap outlet valve for 2A SPE report that the valve is Closed.

Evaluator Notes	
Plant Response:	The SPE trips and the exhauster valves close. APP UA-2 4-5 Gland Seal Vacuum Loss annunciates. The BOP will start the B SPE and place in service to maintain vacuum.
Objectives:	SRO - Direct B SPE started. RO - Diagnose A SPE failure and Starts B SPE.
Success Path:	SPE B is started and vacuum returned to normal.
Event Termination:	Go to Event 5 at the direction of the Lead Evaluator.

EVENT 4: STEAM PACKING EXHAUSTER (SPE) TRIP

Time	Pos	EXPECTED Operator Response	Comments
	CRS	Direct I&C to investigate	
		May direct entry into 0AOP-37.0, Loss of Condenser Vacuum.	
		Direct the B SPE to be started	
		May conduct a brief (see Enclosure 1 on page 56 for format)	
	ATC	Monitors the plant.	
	BOP	Acknowledges, refers to & reports annunciator UA-2 4-5 *GLAND SEAL VACUUM LOSS* May announce and enter 0AOP-37.0, Loss of Condenser Vacuum. Performs actions of APP (page 27) Starts standby SPE IAW 2OP-26.1 Section 6.3.1 (See page 28) Closes OG-MOV-D1 (Steam Seal SPE 2A MO Disch Vlv)	

Unit 2
APP-UA-02 4-5
Page 1 of 2

GLAND SEAL VACUUM LOSS

AUTO ACTIONS

NONE

CAUSE

1. Steam packing exhauster not operating.
2. Steam packing exhauster suction/discharge valve throttled closed, so gland seal vacuum cannot be maintained.
3. Tube leaks in gland exhauster condenser.
4. Steam Seal Feed Valve or the Steam Packing Unloading valves are not controlling steam seal header pressure correctly.
5. Insufficient condensate flow through the steam packing exhauster.
6. Circuit malfunction.

OBSERVATIONS

1. Gland seal vacuum on OG-PI-EPT9-SPE below 5 inches of water.
2. Main condenser vacuum decreasing.
3. Increased off-gas flow.
4. Steam seal header pressure high.

ACTIONS

1. Start standby exhauster and adjust its discharge valve to maintain vacuum between 10 and 20 inches H_2O per 2OP-26.1.
2. If gland seal regulator is not operating properly, refer to 2APP-UA-02 3-5, STEAM SEAL REGULATOR PRESS-LOW.
3. Drain the loop seals on the in-service steam packing exhauster.
4. If steam seal header pressure is above 7 psig, check the Steam Seal Feed Valve MS-SSFV operating correctly as follows:
   a. Throttle closed, Mn Steam to Seals Vlv MVD-S1 to restore steam seal header pressure between 1.5 and 4.0 psig.
   b. If throttling the MVD-S1 was successful in restoring steam seal header pressure, then bypass the Steam Seal Feed Valve per 2OP-26.1 Section 8.3.
   c. If throttling the MVD-S1 was not successful then reopen MVD-S1 and bypass the Steam Seal Unloading Valve by throttling open Steam Seal Bypass Unload Vlv, MVD-B.
5. If moisture is suspected in Steam Packing Exhauster instrument lines, drain per 2OP-26.1 Section 8.2.

DEVICE/SETPOINTS

Pressure Switch OG-PS-VS-SPE	5 inches of water

6.3 Infrequent Operation

6.3.1 Shifting Steam Packing Exhausters

1. **Confirm** the following initial conditions are met:

   a. Gland Sealing Steam System is in operation per Section 6.1 ..... ________

   b. Condensate System is in service and is aligned to supply adequate flow to the SPE per 2OP-30 section for Swapping Off-Gas Trains During Normal Conditions. .... ________

2. **IF** Steam Packing Exhauster SPE 2A is operating, **THEN perform** the following: .... ________

   a. **Open** 2-MVD-V52 (Float Trap Outlet Valve) for SEP 2B. .... ________

   b. **Start** Steam Packing Exhauster SPE 2B. .... ________

   c. **Ensure** OG-MOV-E2 (Steam Seal SPE 2B MO Inlet Vlv) is OPEN. .... ________

   d. **Throttle closed** OG-MOV-D1 (Steam Seal SPE 2A MO Disch Vlv) and **throttle open** OG-MOV-D2 (Steam Seal SPE 2B MO Disch Vlv) while maintaining OG-PI-EPT-9 (Steam Packing Exhauster Vacuum) located on Panel XU-2, between 10 and 20 inches water vacuum. .... ________

   e. **Ensure** OG-MOV-D1 (Steam Seal SPE 2A MO Disch Vlv) is CLOSED. .... ________

   f. **Stop** Steam Packing Exhauster SPE 2A. .... ________

   g. **Close** 2-MVD-V51 (Float Trap Outlet Valve) for SPE 2A. .... ________

   h. **Ensure** OG-MOV-E1 (Steam Seal SPE 2A MO Inlet Vlv) is CLOSED. .... ________

EVENT 5: IRM C FAILURE	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 5**, to fail IRM C upscale.

Simulator Operator Role Play	
	If contacted as the RE for IRM C inoperability, acknowledge request.
	When IRM C inoperability has been addressed and by Lead Examiners direction, contact the control room as WCC SRO and report IRM A can be declared Operable following a satisfactory channel check. Once declared operable the off normal tag can be removed and the WCC will follow up with the paperwork.

Evaluator Notes
Plant Response: The crew will continue raising power by pulling control rods in preparation for placing the Mode switch to RUN. IRM C will fail upscale causing a rod block and half scram.
Objectives: SRO - Determine Technical Specification application. RO - Perform actions for IRM C failure
Success Path: Declare IRM A operable by channel check and bypass IRM C.
Event Termination: Go to Event 5 at the direction of the Lead Evaluator.

EVENT 5: IRM C FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Directs APP reference.	
		Contacts I&C for IRM C failure. May contact Shift Manager also.	
		References TS 3.3.1.1 and determines with IRMs A & C inoperable: Condition A is applicable for Function 1a Required Action A.1 Place channel in trip is required within 12 hours or A.2 Place associated trip system in trip is required in 12 hours. May enter TRM 3.3 (Control Rod Block Instrumentation) Function 3 Condition A, Tracking LCO	
		May conduct a brief (see Enclosure 1 on page 56 for format)	
		Evaluates IRM A operability following satisfactory channel check . 2OP-09, Attachment 4, 2.3.4 (Operability Guidance). **NOTE:** WCC provides cue that IRM A can be declared operable after channel check is SAT. Channel Check definition in the RO DSR. Channel Checks are a sufficient WO PMT for SRMs and IRMs at power unless a component failure is suspected in which case an I/V curve and TDR trace is desirable	
		Directs IRM A channel check be performed.	
	BOP	Plant Monitoring:	

EVENT 5: IRM C FAILURE

Time	Pos	EXPECTED Operator Response	Comments
	ATC	Determines IRM C failed upscale.	
		Responds and reports applicable alarms for IRM C failing upscale. *A-05* *1-7 REACTOR AUTO SCRAM SYS A* *4-7 NEUT MON SYS TRIP* *2-4 IRM UPSCALE* *2-2 ROD OUT BLOCK* *3-4 IRM A UPSCALE/INOP*	
		A-5 *IRM A UPSCALE/INOP* actions: May Reposition range switch for IRM C to bring indicated power to between 15 and 50 on the 0-125 scale. May verify IRM C Drawer Selector switch (Control Panel H12-P606) is in OPERATE. May notify CRS of Tech Spec applicability	
		May inform CRS IRM C cannot be bypassed and half scram cannot be reset due to IRM A being bypassed.	
		Performs channel check of IRM A for operability. RO DSR Item # 9 (IRM channel check) 2OI-03.2, Definition 5.1.	
		Removes IRM A from Bypass	
		Bypasses IRM C per APP guidance.	
		Resets half scram per APP guidance.	

IRM A UPSCALE/INOP

AUTO ACTIONS

1. Rod withdrawal block (bypassed when reactor mode switch is in RUN).
2. Reactor half-Scram (bypassed when reactor mode switch is in RUN).

CAUSE

1. IRM Channel(s) A, C, E, or G indicating greater than or equal to 117 on the 0-125 scale.
2. IRM Channel(s) A, C, E, or G inoperative signals:
   a. IRM drawer selector switch not in operate.
   b. IRM drawer module unplugged.
   c. IRM detector high voltage power supply low voltage.
3. IRM A, C, E, or G detector failure.
4. Improper ranging of IRM A, C, E, or G range switches during reactor startup or shutdown.
5. Circuit malfunction.

OBSERVATIONS

1. IRM Channel A, C, E, or G indicating greater than or equal to 117 on the 0-125 scale.
2. REACTOR AUTO SCRAM SYS A (A-05 1-7) alarm.
3. ROD OUT BLOCK (A-05 2-2) alarm.
4. NEUT MON SYS TRIP (A-05 4-7) alarm.
5. IRM UPSCALE (A-05 2-4) alarm.
6. IRM Channel A, C, E, or G upscale trip or inop (UPSC TR OR INOP) red indicating light is on.
7. The rod withdrawal permissive indicating light will be off.

ACTIONS

1. Monitor IRM Channels A, C, E, and G to determine affected channel(s).
2. If a sudden rise in indicated reactor power occurred in more than one channel, insert in sequence control rods as necessary to turn the power increase and verify that the correct rod withdrawal sequence is being used.
3. Reposition range switch for IRM A, C, E, or G to bring indicated power to between 15 and 50 on the 0-125 scale.
4. Verify that IRM A, C, E, and G Drawer Selector switches (Control Panel H12-P606) are in OPERATE.

ACTIONS (Continued)

5. If the alarm still exists and one channel is affected, perform the following:
   a. Refer to Technical Specifications and TRM for IRM channel operability requirements.
   b. Notify the Unit CRS.
   c. Bypass the affected channel using the IRM bypass switch
   d. Reset half Scram using the Reactor Scram Reset Switch (C72-S5).
6. If IRM detector failure or circuit malfunction is suspected, ensure that a W/R is prepared.

DEVICE/SETPOINTS

Relay C51-Z20-K90	Deenergized
IRM A, C, E, or G upscale trip unit	More than or equal to 117/125
IRM A, C, E, or G inop trip unit	a. IRM drawer selector switch not in operate b. IRM drawer module unplugged c. High voltage power supply less than or equal to 80 VDC

POSSIBLE PLANT EFFECTS

1. Reactor Scram if RPS Trip System B is tripped.
2. If an IRM channel is bypassed or inoperable, a Technical Specification LCO or TRM Compensatory Measure may result.

REFERENCES

1. LL-9364 - 78
2. FP-5352 - 8
3. Technical Specification 3.3.1.1, TRM 3.3
4. APP A-05 1-7, REACTOR AUTO SCRAM SYS A
5. APP A-05 2-2, ROD OUT BLOCK
6. APP A-05 2-4, IRM UPSCALE
7. APP A-05 4-7, NEUT MON SYS TRIP

EVENT 6: DG3 / E3 / E7 LOSS OF CONTROL POWER	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 6** to trip 125 VDC Panel 2A.
	When requested to align alternate control power: **Initiate Trigger 7**, to align alternate control power to the DG3 and to reset DG3 local DG engine control panel lockout. **Initiate Trigger 8**, to align alternate control power to E3/E7.

Simulator Operator Role Play	
	Acknowledge/reset Unit One alarms, as necessary
	If contacted as TBAO, report Switchboard 2A load breaker GJ6, Feed to Panel 2A, is tripped.
	If contacted as I&C, report problem is a due to GJ6 breaker failure, not a fault on the system.
	If contacted as I&C to verify alternate power to ESS cabinet, report ESS cabinet has transferred to alternate power.

Evaluator Notes	
Plant Response:	DC Panel 2A will trip resulting in loss of control power to DG 3, Bus E3 and Bus E7. The crew will respond per 0AOP-39.0 and transfer the control power to alternate. DG3, Bus E3 and Bus E7 are inoperable until transferred to alternate supply. Once control power is transferred, a 7 day action is required to restore to the normal source. The BOP operator will return DG 3 to AUTO IAW AOP-39.0.
Objectives:	SRO - Directs AOP-39 and APP actions Evaluate TS 3.8.1 and 3.8.7. RO - Perform AOP-39 actions.
Success Path:	Restore DG3 control power and then return DG3 to Auto.
Event Termination:	Go to Event 7 at the direction of the Lead Evaluator.

EVENT 6: DG3 / E3 / E7 LOSS OF CONTROL POWER

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of APPs: *UA-21 6-2, DG-3 LO START AIR PRESS* *UA-21 6-3, DG-3 CTL POWER SUPPLY LOST* *UA-19 6-3, DG-1 CTL PWR SUPPLY LOST* *UA-17, 2-3, DG-3/E3 ESS LOSS OF NORM POWER*	
		Direct actions of 0AOP-39.0, Loss Of DC Power	
		Contact I&C to verify ESS cabinets have transferred to alternate power.	
		Direct transfer of control power to alternate source	
		Direct returning DG3 to Auto	
		Determine Tech Specs **3.8.1 AC Sources - Operating, Condition D applies**. (until alt power established) D.1 Perform SR 3.8.1.1 – within 2 hours and once per 12 hours <u>AND</u> D.3.1 Determine OPERABLE DGs not inoperable due to common cause failure – 24 hours <u>OR</u> D.3.2 Perform SR 3.8.1.2 for OPERABLE DGs – 24 hours <u>AND</u> Restore DG to OPERABLE status – 7 days **3.8.7 Distribution Systems - Operating, Condition C applies**. C.1 Declare required features supported by the inoperable DC electrical power distribution system inoperable - Immediately. <u>AND</u> C.2 Initiate action to transfer DC electrical power distribution subsystem to its alternate DC source - Immediately <u>AND</u> C.3 Declare required features supported by the inoperable DC electrical power distribution subsystem OPERABLE – Upon completion of transfer of the required feature's DC electrical power distribution subsystem to its OPERABLE DC source. <u>AND</u> C.4 Restore DC electrical power distribution subsystem to OPERABLE status – 7 Days.	
		May conduct a brief (see Enclosure 1 on page 56 for format)	

EVENT 6: DG3 / E3 / E7 LOSS OF CONTROL POWER

Time	Pos	EXPECTED Operator Response	Comments
	ATC	Monitors the plant.	
	BOP	Report annunciators and review APPs: *UA-21 6-2, DG-3 LO START AIR PRESS* *UA-21 6-3, DG-3 CTL POWER SUPPLY LOST* *UA-19 6-3, DG-1 CTL PWR SUPPLY LOST* *UA-17, 2-3, DG-3/E3 ESS LOSS OF NORM POWER*	
		Announce and enter 0AOP-39.0, Loss of DC Power. (see page 37)	

4.2 Supplementary Actions

1. Loss of Battery Chargers:

   a. **Monitor** 125V and 24V DC battery voltages. ☐

   b. **IF** power has been removed from the battery chargers for greater than 1 hour,
   THEN remove selected loads from the battery based on 0OI-50, 125/250 and 24/48 VDC Electrical Load List and Unit CRS direction. ☐

   c. Before 125V DC battery voltage reaches the low voltage limit of 105 volts, **remove** loads as directed by the Unit CRS as necessary to maintain battery voltage greater than 105 volts. ☐

   d. Before 24V battery voltage reaches the low voltage limit of 21 volts, **remove** loads as directed by the Unit CRS as necessary to maintain battery voltage greater than 21 volts. ☐

   e. **IF** battery charger AC power has been lost due to Station Blackout,
   THEN enter 1EOP-01-SBO(2EOP-01-SBO), Station Blackout. ☐

2. Loss of Any DC Panel:

   a. **Determine** which panel has been lost using Attachment 3, Annunciators Associated with Losses of Various DC Panels, if necessary. ☐

   b. **Dispatch** an operator to investigate the cause of the loss of DC power. ☐

   c. **Contact** Duty I&C to determine actual electrical system ground conditions prior to transferring any panel to alternate source or reenergizing from the normal source. ☐

   d. **IF** I&C determines a panel is faulted,
   THEN DO NOT reenergize the panel until the fault is isolated. ☐

   e. **Refer to** 0OI-50, 125/250 and 24/48 VDC Electrical Load List, for specific load information. ☐

   f. **IF** switchyard control power
   OR 4KV bus control power is lost,
   THEN request the Load Dispatcher to minimize grid operations. ☐

4.2 Supplementary Actions (continued)

g. Using the following table, **determine** the appropriate section based on the DC panel lost. .... ☐

(1) **Go to** the appropriate section for additional actions. .... ☐

Unit	Div.	DC Panel Lost	Normal Power Supply to Panel	Procedure Section
1	I	3A, 5A, 11A	1A-1	Section 4.2 Step 3 on page 8
		1A, 7A	1A-2	Section 4.2 Step 4 on page 10
	II	1B, 7B, 3AB	1B-1	Section 4.2 Step 5 on page 14
		3B, 11B, 9A	1B-2	Section 4.2 Step 6 on page 20
2	I	4A, 6A, 12A, 17	2A-1	Section 4.2 Step 3 on page 8
		2A, 8A	2A-2	Section 4.2 Step 4 on page 10
	II	2B, 8B, 4AB, 13, MWT	2B-1	Section 4.2 Step 5 on page 14
		4B, 12B, 10A	2B-2	Section 4.2 Step 6 on page 20

4.2 Supplementary Actions (continued)

(3) WHEN directed,
THEN **perform** the following at Sub E5: ☐

(a) **Open** Sub E5 1-E5-FM9-72-NORM (Normal Control Power Circuit Breaker), inside Compt. FM9. ☐

(b) **Close** Sub E5 1-E5-FM9-72-ALT (Alternate Control Power Circuit Breaker), inside Compt. FM9. ☐

b. IF loss of DC Distribution Panel 2A has occurred,
THEN **dispatch** an operator to the Diesel Generator Building. ☐

(1) WHEN directed,
THEN **perform** the following for DG3: ☐

(a) **Open** DG3 Normal Feed 8, normal diesel generator control power breaker, in the rear upper right inside of the Excitation Control Panel. ☐

(b) **Close** DG3 Alternate Feed 8A, alternate diesel generator control power breaker, in the rear upper right inside of the Excitation Control Panel. ☐

(c) **Confirm** the Governor Control At Setpoint light is LIT within 10 seconds after control power has been restored. ☐

(d) IF the Governor Control At Setpoint light does NOT light
THEN **initiate** a WO/WR ☐

(e) **Depress** Lockout Reset pushbutton, on the local diesel generator engine control panel ☐

(f) **Confirm** diesel generator Avail light on Panel XU-2 is ON. ☐

(g) **Depress** DG3 Auto Switch push button on RTGB Panel XU-2. ☐

LOSS OF DC POWER	0AOP-39.0
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4.2 Supplementary Actions (continued)

(2) **WHEN** directed,
THEN perform the following at Bus E3: ☐

(a) **Open** Bus E3 125 Volt E3 Bus Normal Control Power breaker inside Compt. AI4. ☐

(b) **Close** Bus E3 125 Volt E3 Bus Alternate Control Power breaker inside Compt. AI4 ☐

(3) **WHEN** directed,
THEN perform the following at Sub E7: ☐

(a) **Open** Sub E7 2-E7-FN1-72-NORM (Swgr 125V DC Normal Control Power Circuit Breaker), inside Compt. FN1 ☐

(b) **Close** Sub E7 2-E7-FN1-72-ALT (Swgr 125V DC Alternate Control Power Circuit Breaker), inside Compt. FN1 ☐

4.2 Supplementary Actions (continued)

d. **IF** loss of DC Distribution Panel 2A has occurred, **THEN confirm** ESS Panel H60 is OPERABLE by performing the following: ☐

(1) Alternate source from Battery Bus 1A-1, Panel 3A, is OPERABLE. ☐

NOTE

- Loadside is the right side of the terminal strip. ☐
- Drawing F-09118-1 is the interconnection wiring diagram for ESS Panel H60. ☐

(2) **Request** I&C to determine power is available indicated by measurement of 125 VDC system voltage between the following points in ESS Panel H60:

- Loadside of FU-2 to loadside of FU-4 ☐
- Loadside of FU-6 to loadside of FU-8 ☐
- Loadside of FU-10 to loadside of FU-12 ☐
- Loadside of FU-14 to loadside of FU-16 ☐
- Loadside of FU-18 to loadside of FU-20 ☐

EVENT 7: LOWERING TORUS WATER LEVEL / ATTEMPT TO FILL TORUS	
Simulator Operator Actions	
	At the direction of the Lead Evaluator, **Initiate Trigger 10** to start Torus Water Leak
	NOTE: It will take ~24 minutes to reach -5.5 feet in the torus.

Simulator Operator Role Play	
	If contacted to look for leaks in the RB -17' elevation, after 5 minutes report none found.
	When directed to open E21-F002A (Core Spray Pump A Suction Valve From The Condensate Storage Tank), wait 3 minutes and report that the handwheel is broken, the valve cannot be opened.
	When directed to align RHR Loop A, wait 3 minutes and report SEP-18 Section 2.2.3 Steps 5a-c are complete. If directed to investigate F028A breaker, report overcurrent trip, will not reset if asked.
	When directed to align RHR Loop B, wait 3 minutes and report SEP-18 Section 2.2.3 Steps 6a-c are complete. If directed to investigate F024B breaker, report thermal trip, will not reset if asked.

Evaluator Notes	
Plant Response:	Torus level will begin to lower due to an unisolable leak on RHR suction. If attempted to raise torus water level, on RHR A loop the E11-F028A (Torus Discharge Isol Vlv) will trip when opened, on RHR B loop the E11-F024B (Torus Cooling Isol Vlv) will will trip when opened, and on Core Spray the E21-F002A (Core Spray Pump A Suction Valve From The Condensate Storage Tank) handwheel will be broke.
Objectives:	SRO -Direct actions for a lowering torus water level IAW PCCP RO - Respond to a lowering torus water level IAW PCCP.
Success Path:	Attempts to add water to torus through RHR and Core Spray systems.
Event Termination:	When torus fill through RHR / CS has been attempted or a reactor scram inserted.

EVENT 7: LOWERING TORUS WATER LEVEL / ATTEMPT TO FILL TORUS

Time	Pos	EXPECTED Operator Response	Comments
	SRO	Direct actions of PCCP.	
		Direct torus fill IAW 0EOP-01-SEP-18	
		May conduct a brief on when Reactor Scram is required (see Enclosure 1 on page 56 for format)	
	ATC	Report annunciator A-01 3-7, Suppression Chamber Lvl Hi/Lo	
		Diagnose lowering torus water level.	
		When directed by the CRS, perform 0EOP-01-SEP-18, Filling the Torus. (page 45) If RHR Loop A is selected, report unable to fill due to E11-F028A (Torus Discharge Isol Vlv) breaker magnetic trip. If RHR Loop B is selected, report unable to fill due to E11-F024B (Torus Cooling Isol Vlv) breaker thermal trip. If CS Loop A is selected, report unable to fill due to E21-F002A (Core Spray Pump A Suction Valve From The Condensate Storage Tank) handwheel broken.	

EVENT 7: LOWERING TORUS WATER LEVEL / ATTEMPT TO FILL TORUS			
	BOP	Monitors the plant	
		Report A-05 5-5, Pri Cmt Hi/Lo Press	
		Dispatch AO to look for the leak.	

1.0 ENTRY CONDITION

- As directed by Emergency Operating Procedures (EOPs)
- As directed by Severe Accident Management Guidelines (SAMGs)

2.0 INSTRUCTIONS

2.1 <u>Core Spray Torus Fill</u>

2.1.1 Manpower Required

- 1 Reactor Operator
- 2 Auxiliary Operators

2.1.2 Special Equipment

None

2.1.3 Core Spray Torus Fill Actions

1. **Select** Core Spray loop to be used: .... ☐ RO

   A B

2. **Confirm** Core Spray loop to be used:
   - <u>**NOT**</u> in operation .... ☐ RO
   - Suction aligned to torus .... ☐ RO

3. **Monitor** and **control** CST level greater than 11 feet. .... ☐ AO

4. **Monitor** torus level. .... ☐ RO

2.1.3 Core Spray Torus Fill Actions (continued)

NOTE

Normal torus level is -27 to -31 inches. ☐

5. <u>IF</u> Core Spray Loop A selected,
<u>**THEN:**</u>

a. **Unlock** and slowly **throttle open** E21-F002A (Core Spray Pump A Suction Valve From The Condensate Storage Tank). ☐ AO

b. <u>**WHEN**</u> at desired torus level,
<u>**THEN:**</u> ☐ RO

- **Close** E21-F002A (Core Spray Pump A Suction Valve From The Condensate Storage Tank). ☐ AO

FILLING THE TORUS	0EOP-01-SEP-18
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2.2 **RHR Torus Fill**

2.2.1 **Manpower Required**

- 1 Reactor Operator
- 2 Auxiliary Operators

2.2.2 **Special Equipment**

None

2.2.3 **RHR Torus Fill Actions**

1. **Select** RHR loop to be used: .... ☐ RO

   A B

2. **Confirm** RHR loop to be used **NOT** in operation .... ☐ RO

3. **Monitor and control** MUD tank level greater than 14 feet .... ☐ AO

4. **Monitor** torus level. .... ☐ RO

NOTE

Normal torus level is -27 to -31 inches. .... ☐

5. **IF** RHR Loop A selected,
   THEN:

NOTE

Valves located on HPCI mezzanine. .... ☐

a. **Close** E11-V195 (RHR Keepfill Station Outlet Isolation Valve) .... ☐ AO

b. **Close** E11-V194 (RHR Keepfill Station Inlet Isolation Valve) .... ☐ AO

c. **Open** E11-F082A (RHR Loop A Keepfill Station Bypass Valve) .... ☐ AO

2.2.3 RHR Torus Fill Actions (continued)

d. **Place** E11-CS-S18A (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD .... ☐ RO

e. Momentarily **place** E11-CS-S17A (Containment Spray Valve Control Switch) to MANUAL .... ☐ RO

f. **Open** E11-F028A (Torus Discharge Isol Vlv) .... ☐ RO

g. Slowly **throttle open** E11-F024A (Torus Cooling Isol Vlv) .... ☐ RO

h. <u>**WHEN**</u> at desired torus level, <u>**THEN**</u> **close** E11-F024A (Torus Cooling Isol Vlv) .... ☐ RO

i. **Close** E11-F028A (Torus Discharge Isol Vlv) .... ☐ RO

j. **Close** E11-F082A (RHR Loop A Keepfill Station Bypass Valve) .... ☐ AO

6. <u>**IF**</u> RHR Loop B selected, <u>**THEN:**</u>

NOTE
Valves located on Reactor Building 50' west. .... ☐

a. **Close** E11-F098 (RHR Keepfill Station Outlet Isolation Valve) .... ☐ AO

b. **Close** E11-F099 (RHR Keepfill Station Inlet Isolation Valve) .... ☐ AO

c. **Open** E11-F088 (RHR Loop B Keepfill Station Bypass Valve) .... ☐ AO

d. **Place** E11-CS-S18B (2/3 Core Height LPCI Initiation Override Switch) to MANUAL OVERRD .... ☐ RO

2.2.3 RHR Torus Fill Actions (continued)

e. Momentarily **place** E11-CS-S17B (Containment Spray Valve Control Switch) to MANUAL ☐ RO

f. **Open** E11-F028B (Torus Discharge Isol Vlv) ☐ RO

g. Slowly **throttle open** E11-F024B (Torus Cooling Isol Vlv) ☐ RO

h. **WHEN** at desired torus level, **THEN close** E11-F024B (Torus Cooling Isol Vlv) ☐ RO

i. **Close** E11-F028B (Torus Discharge Isol Vlv) ☐ RO

j. **Close** E11-F088 (RHR Loop B Keepfill Station Bypass Valve) ☐ AO

7. **Exit** this section and **go to** Section 2.4 ☐ RO

EVENT 8: SCRAM / EMERGENCY DEPRESSURIZATION	
Simulator Operator Actions	
	When directed by the Lead Evaluator, place the simulator in FREEZE
	DO NOT RESET THE SIMULATOR PRIOR TO RECEIPT OF CONCURRENCE TO DO SO FROM THE LEAD EXAMINER

Simulator Operator Role Play	

Evaluator Notes	
Plant Response:	Before level reaches -5.5 feet in the torus a reactor scram is required. When torus water level reaches -5.5 feet emergency depressurization is required. The crew can anticipate emergency depressurization.
Objectives:	SRO - Direct ED or Anticipate ED based on torus water level. RO - Perform ED or Anticipate ED.
Success Path:	Reactor depressurized.
Scenario Termination:	***When all rods are inserted and RPV pressure is less than 100 psig, the scenario may be terminated.***
Remind students not to erase any charts and not to discuss the scenario until told to do so by the evaluator/instructor.	

EVENT 8: SCRAM / EMERGENCY DEPRESSURIZATION

Time	Pos	EXPECTED Operator Response	Comments
	CRS	***Direct a reactor scram before torus level reaches -5.5 feet.***	***CRITICAL TASK #1***
		Direct 0EOP-01-SEP-15, Anticipate Emergency Depressurization. ***OR*** ***Direct Emergency Depressurization***	***CRITICAL TASK #2***
	ATC	***When directed to scram performs scram immediate actions*** (see page 52) Performs Scram Hard Card (see page 53)	***CRITICAL TASK #1***
		Reports all rods in.	
	BOP	Maintains reactor pressure as determined by the CRS.	
		Maintains level as directed by the CRS. May align condensate and feedwater IAW hard card. (See Enclosure 2 page 57)	
		If directed performs 0EOP-01-SEP-15, Anticipate Emergency Depressurization. (see page 56)	***CRITICAL TASK #2***
		If directed opens 7 ADS valves.	

Unit 2 SCRAM Immediate Actions

1. **Ensure** SCRAM valves OPEN by manual SCRAM or ARI initiation.

2. **WHEN** steam flow less than 3.0 Mlb/hr,
   THEN place reactor mode switch in SHUTDOWN.

3. **IF** reactor power below 2% (APRM downscale trip),
   THEN trip main turbine.

4. **Ensure** master RPV level controller setpoint at +170 inches.

5. **IF:**

   - Two reactor feed pumps running

     AND

   - RPV level above +160 inches

     AND

   - RPV level rising,

   THEN trip one.

SCRAM Card

Enter applicable leg ☐

Scram	ATWS
All Control Rods FULL-IN ☐	Indications of Hydraulic/Electrical ATWS ☐
RPV Water Level ☐	**Ensure** ARI initiated ☐
________________ inches	Reactor Power ☐
RPV Pressure ☐	________________ %
________________ psig	**Communicate** ATWS report to CRS ☐
Communicate scram report to CRS ☐	**IF** enabled, **THEN initiate** a recirc pump manual runback ☐
Place SULCV in service ☐	**IF** reactor power above **2% OR CANNOT** be determined, **THEN trip** both recirc pumps ☐
Insert Nuclear Instrumentation ☐	**Report** reactor power to CRS ☐
Ensure Turbine Oil System Operating ☐	**Exit** scram card and **perform** EOP-01-LEP-02 ☐
Ensure Reactor Recirculation Pump speed at 34% ☐	
Ensure Heater Drain Pumps tripped ☐	
Exit scram card ☐	

8

1/905
2/906
S/907

1.0 ENTRY CONDITION

- As directed by Emergency Operating Procedures (EOPs)

2.0 INSTRUCTIONS

2.1 Reactor Vessel Depressurization

2.1.1 Manpower Required

- 1 Reactor Operator

2.1.2 Special Equipment

None

2.1.3 Operator Actions

1. **Ensure:**
   - Flow path available from RPV to condenser .... ☐ RO
   - EHC System in operation .... ☐ RO
   - Circulating water in operation .... ☐ RO
   - Vacuum System in operation .... ☐ RO
   - Turbine Shaft Sealing System in operation .... ☐ RO
2. **IF AT ANY TIME** Main Steam Line Break indicated by:
   - A-06 3-6, Stm Tunnel Hi Temp Sys A .... ☐ RO
   - A-06 4-6, Stm Tunnel Hi Temp Sys B .... ☐ RO

2.1.3 Operator Actions (continued)

- A-06 6-7, MSIV Pit/TB/TB Tunnel Hi Temp .... ☐ RO
- A-06 5-6, Mn Stm Line Hi Flow Sys A .... ☐ RO
- A-06 6-6, Mn Stm Line Hi Flow Sys B .... ☐ RO

THEN terminate RPV depressurization .... ☐ RO

3. **IF AT ANY TIME** fuel failure indicated by:
   - UA-23 2-6, Main Steam Line Rad Hi .... ☐ RO
   - UA-03 5-2, Process Off-Gas Rad Hi .... ☐ RO
   - UA-03 6-4, Process OG Vent Pipe Rad Hi .... ☐ RO

   THEN terminate RPV depressurization .... ☐ RO

4. **IF AT ANY TIME** RPV pressure reduction will result in loss of injection required for adequate core cooling, **THEN terminate** RPV depressurization .... ☐ RO

5. **IF** MSIV's CLOSED, **THEN equalize** pressure and **open** MSIV's (OP-25) .... ☐ RO

6. **Unit 2 only: Maintain** main steam line flow less than $3x10^6$ lbm/hr while performing Step 7 .... ☐ RO

7. Rapidly **depressurize** RPV with Main Turbine Bypass valves irrespective of cooldown rate .... ☐ RO

8. **Exit** this procedure and **continue** in procedure(s) in effect .... ☐ RO

ENCLOSURE 1 Page 1 of 1

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ATTACHMENT 8
Page 1 of 1

<< Crew Brief Template >>

Begin Brief	☐ Announce "Crew Brief" ☐ All crew members acknowledge announcement

Recap	**(As Required)**
	☐ Update the crew as needed:
	☐ Describe what happened and major actions taken
	☐ Procedures in-progress
	☐ Notifications:
	    ☐ Maintenance
	    ☐ Engineering
	    ☐ Others (Dispatcher, Station Management, etc.)
	☐ Future Direction and priorities
	☐ Discuss any contingency plans

Input	**(As Required)**
	☐ Solicit questions/concerns from each crew member:
	    ☐ ROs
	    ☐ CRS
	    ☐ STA
	☐ Are there any alarms unexpected for the plant conditions?
	☐ What is the status of Critical Parameters?

EAL	**(As Required)** ☐ Provide EAL and potential escalation criteria
Finish Brief	☐ Restore normal alarm announcement? (Yes/No) ☐ Announce "End of Brief"

ENCLOSURE 2

Feedwater Level Control Following a Reactor Scram

NOTE	This attachment is **NOT** to be used for routine system operation.

1. **ENSURE** the following:
   - FW-V6 **AND** FW-V8 **OR** FW-V118 **AND** FW-V119 closed ☐
   - FW-FV-177 closed ☐
   - FW-V120 closed ☐
   - FW control MODE SELECT in 1 ELEM ☐
   - SULCV in M (MANUAL) closed ☐
   - B21-F032A **AND/OR** B21-F032B open ☐
2. **PLACE** the MSTR RFPT SP/RX LVL CTL in M (MANUAL), **THEN:** ☐
   - **ADJUST** to 187" ☐
3. **IF** any RFP is running, **THEN**:
   a. **PLACE** RFP A(B) RECIRC VLV, control switch to open ☐
   b. **PLACE** RFPT A(B) SP CTL in M (MANUAL) ☐
4. **IF** no RFP is running, **THEN**:
   a. **PLACE** RFP A(B) RECIRC VLV, control switch to open ☐
   b. **ENSURE** the following:
      - RFP A(B) DISCH VLV, FW-V3(V4) open ☐
      - RFPT A(B) SP CTL in M (MANUAL) at lower limit ☐
      - RFPT A(B) MAN/DFCS control switch in MAN ☐
      - Reactor water level is less than +206 inches **AND** RFPT A&B HIGH LEVEL TRIP reset ☐
   c. **DEPRESS** RFPT A(B) RESET ☐

ENCLOSURE 2

Feedwater Level Control Following a Reactor Scram

d. **ENSURE** RFPT A(B) LP **AND** HP STOP VLVS open ☐

e. **ROLL** RFPT A(B) to 1000 rpm by depressing RFP A(B) START ☐

f. **RAISE** RFPT A(B) to approximately 2550 rpm using the LOWER/RAISE control switch ☐

g. **DEPRESS** RFPT A(B) DFCS CTRL RESET ☐

5. **ENSURE** MAN/DFCS control switch in DFCS ☐
6. **RAISE** RFPT A(B) SP CTL speed until discharge pressure is greater than or equal to 100 psig above reactor pressure ☐
7. **ADJUST** SULCV to establish desired injection ☐
8. **IF** desired, **THEN PLACE** SULCV in A (AUTO) ☐
9. **IF** needed, **THEN THROTTLE** FW-V120 ☐
10. **IF** needed, **THEN GO TO** 2OP-32 Section 8.17 for level control ☐

ATTACHMENT 1 - Scenario Quantitative Attribute Assessment

Category	NUREG 1021 Rev. 2 Supp. 1 Req.	Scenario Content
Total Malfunctions	5-8	7
Malfunctions after EOP Entry	1-2	2
Abnormal Events	2-4	2
Major Transients	1-2	2
EOPs Used	1-2	2
EOP Contingency	0-2	1
Run Time	60-90 min	90
Crew Critical Tasks	2-3	2
Tech Specs	2	2
Instrument / Component Failures before Major	2 – OATC 2 - BOP	4
Instrument / Component Failures after Major	2	2
Normal Operations	1	1
Reactivity manipulation	1	1

ATTACHMENT 5
Page 1 of 1

<< Neutron Monitoring Spiking Troubleshooting Form >>

1. Initiator's name Unit Two SRO
2. **Check** all instruments that are spiking and the associated Unit:

☐ Unit 1	☐ SRM A	☒ IRM A	☐ IRM E
☒ Unit 2	☐ SRM B	☐ IRM B	☐ IRM F
	☐ SRM C	☐ IRM C	☐ IRM G
	☐ SRM D	☐ IRM D	☐ IRM H

3. Time and date of event Today - previous shift
4. What is the duration of the spiking (duration of individual spike)? Add additional information below to characterize spiking event.

☐ Seconds ☒ Minutes ☐ Hours

1. **Ensure** all required observations to support operability are appropriately documented.

5. Has a WO or AR been initiated? If yes, list number(s): 00345765	☒ Yes	☐ No
6. Has a log entry been made?	☒ Yes	☐ No
7. Is there any welding occurring in the plant?	☒ Yes	☐ No
8. Are there any personnel under-vessel?	☐ Yes	☒ No
9. Are there any plant evolutions in progress?	☒ Yes	☐ No
10. Is there any electrical switching occurring?	☐ Yes	☒ No
11. Are any control rods being moved or selected?	☒ Yes	☐ No
12. Has there been a recent change in the mode switch?	☐ Yes	☒ No
13. Is there any major equipment being started?	☒ Yes	☐ No
14. Has there been any observed relay chatter?	☐ Yes	☒ No
15. Is there any refuel bridge movement?	☐ Yes	☒ No
16. Are the rod interlocks being affected?	☒ Yes	☐ No
17. Completed copy of this attachment sent to engineer	☒ Yes	☐ No

Note below any additional information that may aid troubleshooting (such as 2 instruments spiking but **NOT** in the same manner):

Multiple upscale and downscale alarms during startup over a 15 minute period
All other IRMs responded normally

ATTACHMENT 2 – Shift Turnover

Brunswick Unit 2 Plant Status					
Station Duty Manager:	E. Neal			Workweek Manager:	B. Craig
Mode:	2	Rx Power:	2%	Gross*/Net MWe*:	N/A
Plant Risk: Current EOOS Risk Assessment is:				Green	
SFP Time to 200 Deg F:	45.7 hrs			Days Online:	0 days
Turnover:	IAW the reactivity plan the OATC is to raise power to 6-10%. A2X sequence at step 161. Permission for continuous withdrawal has been granted for the rods going from 12-48.				
Protected Equipment:	ADHR / FPC Loop A / Demin Transfer Pump All remaining ECCS LP systems				
Comments:	IRM A was bypassed due to spiking and the paperwork is being evaluated by the WCC SRO for its return to service. Core Spray Loop B under clearance, expected return in 4 hours. The BOP operator is to complete Step 6.3.46 of 0GP-02, Approach to Criticality and Pressurization of the Reactor.				

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

SIM JPM A - 2016 NRC INITIAL EXAM - RO/ISRO/USRO

LESSON TITLE: Reset Recirc Pump Runback – Both Recirc Pumps Trip

LESSON NUMBER: LOT-SIM-JP-002-A09

REVISION NO: 0

Dan Hulgin 8/18/16
PREPARER / DATE

Bob Bolin 9/06/16
TECHNICAL REVIEWER / DATE

Grant Newton
Shawn Zander 9/06/16
VALIDATOR / DATE

Craig Oliver 09/22/2016
LINE SUPERVISOR / DATE

9-27-16
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

202201B401
Recover from a Reactor Recirculation Pump Runback Per OP-02

K/A REFERENCE AND IMPORTANCE RATING:

202002 A2.01 3.4/3.4
Ability to predict the impacts of recirculation pump trip and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations.

REFERENCES:

2OP-02, Section 6.3.3, Recovery From Reactor Recirculation Pump Runback
2OP-02, Section 6.1.3, Raising Speed Using Individual Recirculation Pump Control
2AOP-04.0, Low Core Flow

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

1 - Reactivity Control (Recirculation Flow Control System)

SETUP INSTRUCTIONS

SIMULATOR SETUP

Initial Conditions

1. Recommended Initial Conditions

   IC-10

2. Required Plant Conditions

   Recirculation Pump A at approximately 70% flow, Recirculation Pump B at Limiter #1. OPRM Trip Enabled annunciator in ALARM.

Malfunctions:

Insert the following malfunctions:

Malf ID	Mult ID	Description	Current Value	Target Value	Rmptime	Actime	Deactime	Trig
RC024F	VFD B	VFD B RUNBACK #1ACTUATES	False	True				
EE026F		Loss of 4160V Bus B	False	True				1

Set Trigger 1, Q2722RSM, VFD B Raise Medium pushbutton to TRUE.

Special Instructions

Initiate VFD B runback and allow plant conditions to stabilize. Once runback is complete, delete malfunction RC024F. If necessary, insert control rods to get below the MELLL Line on the Power-Flow map. Verify VFD B RUNBACK #1 ACTUATES is clear. Reset Speed Hold, if amber light is illuminated.

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM will be performed in the simulator on Unit Two.

Read the following to the JPM performer.

TASK CONDITIONS:

1. Reactor Recirculation Pump operation was previously in accordance with 2OP-02, Section 6.1.2.
2. Recirculation Pump 2B has run back to limiter number 1, and the cause has been corrected.
3. A reactivity management briefing is complete, and your reactivity management team is available in the Control Room
4. Another operator is monitoring Nuclear Instrumentation.

INITIATING CUE:

You are directed by the Unit CRS to reset the Recirculation Pump runback signal and raise flow of Reactor Recirculation Pump 2B to match flow of Recirculation Pump 2A.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 0 - May perform take a minute at job site prior to beginning task.
Examinee may cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Am I using appropriate gloves? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

Step 1 - Obtain current revision of 2OP-02 Sections 6.3.3 may also get Section 6.1.3.
Provide current revision of 2OP-02 Section 6.3.3, and if asked Section 6.1.3.

SAT/UNSAT

TIME START: ____________

Step 2 - Verify the conditions that caused the runback have cleared, or Recirc Pump speed has been lowered below the runback setpoint.
Verifies Recirc Pump speed below the runback setpoint, and condition is clear as part of Task Conditions.

SAT/UNSAT

Step 3 – **ENSURE** *RECIRC PUMP B SPEED DEMAND* signal is approximately the same as the following:

- *RECIRC PUMP B CALCULATED SPEED*
- *RECIRC PUMP B ACTUAL SPEED*

Ensures Calculated Speed and Actual Speed approximately the same.

SAT/UNSAT

Step 4 - **RESET** the Recirc Pump runback for Reactor Recirculation Pump B as follows:

a. **DEPRESS** Recirc VFD B *RUNBACK RESET* push button.
Runback Rest push button is depressed.

****CRITICAL STEP**** **SAT/UNSAT**

b. **CONFIRM** yellow *AUTOMATIC RUNBACK* light extinguished.
Yellow Automatic Runback light is confirmed extinguished

SAT/UNSAT

c. **CONFIRM** annunciator *RECIRC FLOW B LIMIT (A-07 2-4) is clear.*
Annunciator confirmed clear.

SAT/UNSAT

PROMPT:	If asked which VFD RAISE pushbutton to use, or if the VFD RAISE SLOW pushbutton is depressed, direct Examinee as the CRS to raise speed of the Recirculation Pump using the VFD RAISE MEDIUM pushbutton.

Step 5 - ADJUST flow as directed by the Unit CRS.
VFD RAISE MEDIUM push button is depressed in accordance with 2OP-02, Section 6.1.3, pages 38-40.

**** CRITICAL STEP **   SAT/UNSAT**

NOTE:	4160 V Bus B will trip when the VFD RAISE MEDIUM push button is depressed.

PROMPT:	If asked, as CRS, direct Examinee to enter and announce the appropriate AOP.

Step 6 – Determines that no Recirc Pumps are running.
Diagnosis failure of Bus B which causes both Recirc Pumps to lose power.

SAT/UNSAT

Step 7: May enter and announce entry into 2AOP-04.0, Low Core Flow
2AOP-04.0 announced and entered.

SAT/UNSAT

Step 8 – Inserts a manual reactor scram

IAW 2AOP-04.0, Immediate Operator Action, depresses both RPS Channel Manual Pushbuttons.

Performs the following scram immediate actions:

1. *Ensure SCRAM valves OPEN by manual SCRAM or ARI initiation.*
2. *WHEN steam flow less than 3.0 Mlb/hr,*
   THEN place reactor mode switch in SHUTDOWN.
3. *IF reactor power below 2% (APRM downscale trip),*
   THEN trip main turbine.
4. *Ensure master RPV level controller setpoint at +170 inches.*
5. *IF two reactor feed pumps running and RPV level is greater than 160 inches and rising,*
   THEN trip one reactor feed pump.

**** CRITICAL STEP **   SAT/UNSAT**

Step 9 – **Informs** CRS that all rods are in, RPV water level and RPV pressure.

Acknowledge scram report as the CRS.

SAT/UNSAT

TERMINATING CUE: Once a manual reactor scram is inserted and scram immediate actions are complete, the JPM can be terminated.

TIME COMPLETED: ______________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Not Critical	Given as initial conditions
3	Not Critical	Observation of indications
4a	Critical	Action required to reset the runback
4b	Not Critical	Observation of indications
4c	Not Critical	Observation of indications
5	Critical	Action required to meet Initiating Cue
6	Not Critical	Observation of indications
7	Not Critical	Observation of indications
8	Critical	Immediate Operator action from 2AOP-04.0
9	Not Critical	Communication

REVISION SUMMARY

0	New JPM written for 2016 Initial NRC exam.

Validation Time: 20 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	X	No	____		

EVALUATION

Performer: ______________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: ____________

TASK CONDITIONS:

1. Reactor Recirculation Pump operation was previously in accordance with 2OP-02, Section 6.1.2.

2. Recirculation Pump 2B has run back to limiter number 1, and the cause has been corrected.

3. A reactivity management briefing is complete, and your reactivity management team is available in the Control Room

4. Another operator is monitoring Nuclear Instrumentation.

INITIATING CUE:

You are directed by the Unit CRS to reset the Recirculation Pump runback signal and raise flow of Reactor Recirculation Pump 2B to match flow of Recirculation Pump 2A.

6.1.3 Raising Speed/Power Using Individual Recirculation Pump Control or Recirc Master Control

1. **Ensure** the following Initial Conditions are met:

   a. Reactor Recirculation Pumps in operation in accordance with Section 6.1.2. .... _________

   b. Recirculation Pump flow limits are CLEAR................................ _________

NOTE

- Recirculation Pump speed changes are performed when directed by 0GP-04, Increasing Turbine Load to Rated Power, and 0GP-12, Power Changes. Other operating procedures are used simultaneously with this procedure as directed by 0GP-04, Increasing Turbine Load to Rated Power, 0GP-12, Power Changes, or the Unit CRS. .... ☐
- Speed changes are accomplished by depressing Raise Slow or Raise Medium pushbuttons. The Raise Slow pushbutton changes Recirc pump speed at 0.06%/increment at 1 rpm/second. The Raise Medium pushbutton changes Recirc pump speed at 0.28%/increment at 5 rpm/second. .... ☐

CAUTION

The OPRM System monitors LPRMs for indication of thermal hydraulic instability (THI). When greater than or equal to 25% power and less than or equal to 60% recirculation flow, alarms and automatic trips are initiated upon detection of THI. Pump operations are governed by the limits of the applicable Power Flow Map, as specified in the COLR. {8.1.9}.. ☐

2. **<u>IF AT ANY TIME</u>** any of the following conditions exist, **<u>THEN</u> enter** 2AOP-04.0, Low Core Flow. {8.1.9} .... _________

   - Entry into Region A of Power to Flow Map
   - OPRM INOPERABLE **<u>AND</u>** any of the following
     - ◊ Entry into Region B of Power to Flow Map
     - ◊ Entry into 5% Buffer Region of Power to Flow Map
     - ◊ Entry into OPRM Enabled Region and indications of THI (Thermal Hydraulic Instability) exist

6.1.3 Raising Speed/Power Using Individual Recirculation Pump Control or Recirc Master Control (continued)

CAUTION

- The OPRM System monitors LPRMs for indication of thermal hydraulic instability (THI). When greater than or equal to 25% power and less than or equal to 60% recirculation flow, alarms and automatic trips are initiated upon detection of THI. Pump operations is be within the limits of the applicable Power-Flow Map, as specified in the COLR. The Scram Avoidance Region is avoided. {8.1.9} .... ☐
- With core flow less than 57.5 x 10^6 lbs/hr, jet pump loop flows are required within 10% (maximum indicated difference 6.0 x 10^6 lbs/hr). With core flow greater than or equal to 57.5 x 10^6 lbs/hr, jet pump loop flows are required within 5% (maximum indicated difference 3.0 x 10^6 lbs/hr). .... ☐
- If total reactor feedwater flow lowers to less than 16.4% of rated flow, Speed Limiter Number 1 will cause the Recirculation Pumps to run back to 34% speed. This signal must be manually reset in accordance with Section 6.3.3. .... ☐
- When total core flow is greater than 43 mlb/hr, Speed Limiter Number 2 will cause a runback to approximately 48% speed if reactor water level is less than 182 inches and either reactor feed pump A or B suction flow is less than 14.9% of individual RFP rated suction flow. This signal must be manually reset using Section 6.3.3. .... ☐

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

3. IF desired to raise the speed of both Recirc Pumps simultaneously, as directed by the Unit CRS,
   THEN depress Recirc Master Control Raise Slow or Raise Medium pushbutton. .... ______
4. IF desired to raise the speed of an individual Recirc Pump, as directed by the Unit CRS,
   THEN depress the VFD A(B) Raise Slow or Raise Medium pushbutton for the Recirc Pump. .... ______
5. **Confirm** the following, as applicable:
   - A rise in Recirc Pump A(B) Speed Demand, Calculated Speed, and a rise in Actual Speed. .... ______
   - A rise in Reactor power. .... ______
   - A rise in B32-R617(R613) [Recirc Pump A(B) Discharge Flow] .... ______

6.1.3 Raising Speed/Power Using Individual Recirculation Pump Control or Recirc Master Control (continued)

- A rise in B32-VFD-IDS-003A(B) [Recirc VFD 2A(B) Output Wattmeter]... ________
- A rise in B32-VFD-IDS-001A(B) [Recirc VFD 2A(B) Output Frequency Meter]... ________

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

Date/Time Completed ____________________

Performed By (Print) Initials

____________________________ ______

____________________________ ______

____________________________ ______

____________________________ ______

Reviewed By: ____________________________
Unit CRS/SRO

6.3.3 Recovery From Reactor Recirculation Pump Runback

1. **Ensure** Reactor Recirculation Pump operation was previously in accordance with Section 6.1.2. .... _________

NOTE

Recirculation pump runback to approximately 48% speed occurs when reactor water level is less than or equal to 182 inches and reactor feed pump A or B suction flow is less than or equal to 14.9% of individual RFP rated suction flow. A recirculation pump speed runback to 34% will occur when the recirculation pump discharge valve is **NOT** fully OPEN or total feedwater flow is less than 16.4% of the rated flow. Both of these conditions will require a manual reset of the runback. .... ☐

2. **Ensure** the conditions that caused the runback have cleared, or recirc pump speed has been lowered below the runback setpoint......... _________

3. **Ensure** the system operation has stabilized. .... _________

CAUTION

- The OPRM System monitors LPRMs for indication of thermal hydraulic instability (THI). When greater than or equal to 25% power and less than or equal to 60% recirculation flow, alarms and automatic trips are initiated upon detection of THI. Pump operations are within the limits of the applicable Power-Flow Map, as specified in the COLR. The Scram Avoidance Region is avoided. {8.1.9}. .... ☐

4. **IF AT ANY TIME** any of the following conditions exist, **THEN enter** 2AOP-04.0, Low Core Flow.{8.1.9} .... _________
   - Entry into Region A of Power to Flow Map
   - OPRM INOPERABLE **AND** any of the following
     - ◊ Entry into Region B of Power to Flow Map
     - ◊ Entry into 5% Buffer Region of Power to Flow Map
     - ◊ Entry into OPRM Enabled Region and indications of THI (Thermal Hydraulic Instability) exist

5. **Ensure** Recirc Pump A(B) Speed Demand signal is approximately the same as the following:
   - Recirc Pump A(B) Calculated Speed .... _________
   - Recirc Pump A(B) Actual Speed .... _________

6.3.3 Recovery From Reactor Recirculation Pump Runback (continued)

6. **Reset** the Recirc Pump runback for Reactor Recirculation Pump A(B) as follows:

   a. **Depress** Recirc VFD A(B) Runback Reset pushbutton. .... _________

   b. **Confirm** yellow Automatic Runback light is OFF. .... _________

   c. **Confirm** 2-A-06, 3-2 (2-A-07, 2-4), Recirc Flow A(B) Limit, is CLEAR. .... _________

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

7. **Adjust** flow as directed by the Unit CRS. .... _________

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

Date/Time Completed ____________________

Performed By (Print)	Initials
______________________________	______
______________________________	______
______________________________	______
______________________________	______

Reviewed By: ______________________________
Unit CRS/SRO

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

SIM JPM B - 2016 NRC INITIAL EXAM - RO/ISRO/USRO

LESSON TITLE: **MECHANICAL TRIP VALVE OIL TRIP TEST**

LESSON NUMBER: **LOT-SIM-JP-026.2-01**

REVISION NO: **0**

Dan Hulgin 8/18/16
PREPARER / DATE

Bob Bolin 9/06/16
TECHNICAL REVIEWER / DATE

Kyle Cooper
Shawn Zander 9/06/16
VALIDATOR / DATE

Craig Olivera 09/22/2016
LINE SUPERVISOR / DATE

9-27-16
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

245202B101
Perform Mechanical Trip Valve Oil Trip Test Per OP-26

K/A REFERENCE AND IMPORTANCE RATING:

245000 A3.01 Ability to manually operate and/or monitor in the control room: Turbine Trip

REFERENCES:

2OP-26, Section 6.3.8, Mechanical Trip Valve Oil Trip Test

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

4 - Heat Removal (main Turbine Generator and Auxiliary Systems)

SETUP INSTRUCTIONS

SIMULATOR SETUP

Initial Conditions

1. Recommended Initial Conditions

   IC-11

2. Required Plant Conditions

   Turbine is at 1800 rpm.

Malfunctions:

None

Special Instructions

None.

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM will be performed in the simulator on Unit Two.

Read the following to the JPM performer.

TASK CONDITIONS:

1. All applicable prerequisites listed in Section 5 of 2OP-26 are met.
2. Last performance of 2OP-26 Section 6.3.15 was successful.

INITIATING CUE:

You are directed by the Unit CRS to perform 2OP-26, Section 6.3.8, Mechanical Trip Valve Oil Trip Test.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 0 - May perform take a minute at job site prior to beginning task.

Examinee may cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Am I using appropriate gloves? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

Step 1 - Obtain current revision of 2OP-26 Sections 6.3.8.

Provide current revision of 2OP-26 Section 6.3.8.

SAT/UNSAT

TIME START: ____________

Step 2 - Depress the Locked out pushbutton.

Depresses the Locked Out pushbutton on the XU-1 panel.

****CRITICAL STEP**   SAT/UNSAT**

Step 3 – **Confirms** Locked Out light illuminates and UA-23, 3-3, Overspeed Trip Locked annunciator is On

Confirms Locked Out white light illuminates and UA-23, 3-3, Overspeed Trip Locked annunciator is acknowledged and reported to Unit CRS.

SAT/UNSAT

Step 4 - **Depress and hold** the Oil Trip Pushbutton until Tripped light comes On, then release the Oil Trip pushbutton.

Oil Trip pushbutton is depressed and held until the Tripped light is illuminated.

****CRITICAL STEP**   SAT/UNSAT**

Step 5 - Depress and hold the Push to Reset pushbutton and confirm the Resetting light comes on.

Push to Reset pushbutton is depressed and the Resetting light illuminates.

**** CRITICAL STEP ** SAT/UNSAT**

Step 6 - When ~5 seconds have elapsed confirm the Reset light comes On and UA-23, 4-3 Turbine Overspeed Trip Reset, annunciator is received.

Confirms the Reset light illuminates and Turbine Overspeed Trip Reset annunciator is acknowledged and reported to Unit CRS.

SAT/UNSAT

Step 7 - When the Reset light comes On then release the Push to Reset pushbutton.

The Push to Reset pushbutton is released and annunciator UA-23, 4-3, Turbine Overspeed Trip Reset annunciator is reset and reported as cleared to the Unit CRS.

**** CRITICAL STEP ** SAT/UNSAT**

Step 8 - When ~10 seconds have elapsed then depress the Normal pushbutton.

The Normal pushbutton is depressed and the Normal light is confirmed On and the Locked Out Light extinguishes. Annunciator UA-23, 3-3, Overspeed Trip Locked annunciator is reset and reported as cleared to the Unit CRS.

**** CRITICAL STEP ** SAT/UNSAT**

Step 9 – **Informs** CRS Mechanical Trip Valve Oil Trip is complete.

Acknowledge report as the CRS.

SAT/UNSAT

TERMINATING CUE: All actions in 2OP-26, Section 6.3.8 have been completed.

TIME COMPLETED: ______________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Action required to complete the procedure without tripping the main turbine.
3	Not Critical	Observation of indications as a result of the previous step
4	Critical	Action required to complete the procedure
5	Critical	Action required to complete the procedure
6	Not Critical	Observation of indications as a result of the previous step
7	Critical	If released prior to the Reset light illuminating the trip will trip.
8	Critical	Required to restore the system to normal alignment and remove the overspeed trip locked out.
9	Not Critical	Communication

REVISION SUMMARY

0	New JPM written for 2016 Initial NRC exam.

Mechanical Trip Valve Oil Trip Test

Validation Time: 10 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ______________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

TASK CONDITIONS:

1. All applicable prerequisites listed in Section 5 of 2OP-26 are met.
2. Last performance of 2OP-26 Section 6.3.15 was successful.

INITIATING CUE:

You are directed by the Unit CRS to perform 2OP-26, Section 6.3.8, Mechanical Trip Valve Oil Trip Test.

6.3.8 Mechanical Trip Valve Oil Trip Test

1. **Confirm** the following initial conditions are met:

   - All applicable prerequisites listed in Section 5.0 are met. .... ________
   - Turbine is at 1800 rpm, on-line or off-line................................. ________
   - **IF** last performance of Section 6.3.15 was unsuccessful, **THEN** Unit CRS permission is obtained to perform this test. ...... ________

NOTE

During performance of this test, annunciators UA-23, 3-3, Overspeed Trip Locked, and UA-23, 4-3, Turbine Overspeed Trip Reset, are expected alarms. .... ☐

2. **IF AT ANY TIME** during the performance of this test the expected indications are **NOT** observed, **THEN perform** the following:... ________

   a. Immediately **notify** the System Engineer. .... ________

      ____
      Person Notified

   b. **Reference** EC-293249 for expected position of linkages........... ________

3. **Depress** the Locked Out pushbutton.. ________

4. **Confirm** the following:

   a. The Locked Out light comes ON. .... ________

   b. UA-23, 3-3, Overspeed Trip Locked, annunciator is ON. .... ________

5. **Depress** and **hold** the Oil Trip pushbutton until the Tripped light comes ON, then **release** the Oil Trip pushbutton. .... ________

6.3.8 Mechanical Trip Valve Oil Trip Test (continued)

NOTE

- Steps that require holding a pushbutton IN and then confirming required actions occur may be performed and then signed off after completion of the confirmation steps. .... ☐
- EC-293249 (NCR-741915, WO-13511606) document the Resetting Light and UA-23, 4-3, Turbine Overspeed Trip Reset, may **NOT** respond as indicated in Step 6.a and Step 6.c. These are indications only and do **NOT** impact the ability to test and reset the turbine trip logic. Therefore these steps may be marked NA. .... ☐

6. **Depress** and **hold** the Push To Reset pushbutton and **confirm** the following: .... ______
   a. The Resetting light comes ON. .... ______
   b. **WHEN** approximately 5 seconds have elapsed, **THEN** the Reset light comes ON. .... ______
   c. UA-23, 4-3, Turbine Overspeed Trip Reset, annunciator is ON. .... ______
7. **WHEN** the Reset light comes ON, **THEN** **release** the Push To Reset pushbutton and **confirm** the UA-23, 4-3, Turbine Overspeed Trip Reset, annunciator is CLEAR. .... ______
8. **WHEN** at least 10 seconds have elapsed, **THEN** **depress** the Normal pushbutton and **confirm** the following: .... ______
   a. The Normal light comes ON and the Locked Out light goes OFF. .... ______
   b. The UA-23, 3-3, Overspeed Trip Locked, annunciator CLEARS. .... ______

6.3.8 Mechanical Trip Valve Oil Trip Test (continued)

Date/Time Completed ____________________

Performed By (Print) Initials

________________________ ____

________________________ ____

________________________ ____

________________________ ____

Reviewed By ________________________
Unit CRS/SRO

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

SIM JPM C - 2016 NRC INITIAL EXAM - RO/ISRO/USRO

LESSON TITLE: RCIC Start – Steam Line Ruptures and RCIC Fails to Isolate

LESSON NUMBER: LOT-SIM-JP-016-A05

REVISION NO: 04

Lou Sosler 9/10/2015
PREPARER / DATE

John Biggs 9/15/2015
TECHNICAL REVIEWER / DATE

Derek Pickett 9/10/2015
VALIDATOR / DATE

Jerry Pierce 9/24/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

217003B101, Manually Startup The RCIC System Per OP-16

K/A REFERENCE AND IMPORTANCE RATING:

217000A4.08 3.7 3.6

Ability to manually operate and/or monitor RCIC system flow

REFERENCES:

S/969 (RCIC Hard Card)

OP-16, Section 6.1.3

TOOLS AND EQUIPMENT:

None.

SAFETY FUNCTION (from NUREG 1123, Rev 2.):

2 - Inventory Control

SIMULATOR SETUP

Recommended Initial Conditions

Any 100% IC

Required Plant Conditions:

- RPV level <166 inches
- Inhibit ADS
- Place HPCI in PTL
- Trip RFPs

Triggers:

Auto: Q1619RRM, E51-F013 Red Light Equal to TRUE.

Malfunctions:

Event	System	Tag	Title	Value/ Ramp Rate	Activate Time (sec)	Deactivate Time (sec)
N/A	ES	ES055F	E51-F007, Failure to Auto Close	N/A	N/A	N/A
N/A	ES	ES056F	E51-F008, Failure to Auto Close	N/A	N/A	N/A
1	ES	ES025F	RCIC Stm Brk – S RHR Room	20%/ 0 sec.	40 sec	Trigger 1
N/A	ES	ES041F	RCIC Failure to Auto Start	N/A	N/A	N/A

Overrides:

None

Remotes

None

SAFETY CONSIDERATIONS:

None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 2.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the performer.

TASK CONDITIONS:

1. Both Reactor Feed Pumps have tripped and are not available.
2. Reactor level is below 166 inches.
3. HPCI is not available.

INITIATING CUE:

You are directed by the Unit CRS to place RCIC in service per the Hard Card and restore Reactor level to 166 to 206 inches. Notify the Unit CRS when all required actions are complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

Step 2 - Ensure the following valves are open: Turbine Trip & Throttle Valve, E51-V8, and Turbine Trip & Throttle Valve Actuator, E51-V8, and Turbine Governor Valve, E51-V9.

E51-V8 (valve position) E51-V8 (actuator position) and E51-V9 are open.

SAT/UNSAT

Step 3 – Open Cooling Water Supply Valve, E51-F046.

E51-F046 is full open.

**** CRITICAL STEP ** SAT/UNSAT**

Step 4 - Start Vacuum Pump and leave switch in START.

Vacuum Pump running with switch in Start.

SAT/UNSAT

Step 5 - Open Turbine Steam Supply Valve, E51-F045.

E51-F045 is full open.

**** CRITICAL STEP ** SAT/UNSAT**

Step 6 – Open RCIC Injection Valve, E51-F013.

E51-F013 is full open.

**** CRITICAL STEP ** SAT/UNSAT**

Step 7 – Ensure that the RCIC turbine starts and comes up to speed as directed by RCIC FLOW CONTROL.

RCIC Turbine speed observed to come up to speed.

SAT/UNSAT

Step 7a – Raises flow to 500 gpm.

Raises RCIC FLOW CONTROLLER to500 gpm.

SAT/UNSAT

<u>NOTE:</u> When Reactor water level has started to rise and when directed by the evaluator activate Trigger 1 to initiate steam line break.

Step 8 – Recognize the RCIC isolation and trip signal.

RCIC isolation and trip is recognized.

SAT/UNSAT

Step 9 – Recognize the failure of the RCIC Steam Supply Valves, E51-F007 and E51-F008, to close.

Failure of E51-F007 and E51-F008 to close is recognized. Operator refers to 2APP-A-03 (5-2 and 6-2).

SAT/UNSAT

Step 10 – Manually close RCIC Steam Supply Valve, E51-F007 <u>OR</u> RCIC Steam Supply Valve, E51-F008 <u>OR</u> both.

E51-F007 or E51-F008 or both are closed.

**** CRITICAL STEP ** SAT/UNSAT**

Step 11 – Notify the Unit CRS that the RCIC Steam Pipe has ruptured and that E51-F007 and/or E51-F008 were manually closed to isolate the leak.

Unit CRS is notified

SAT/UNSAT

<u>TERMINATING CUE:</u> When the RCIC Steam Line rupture is isolated and the Unit CRS is notified, this JPM is complete.

Time Completed: ___________

NOTE: Comments required for any step evaluated as UNSAT.

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Not Critical	Not required to complete task.
3	Critical	Pump will be damaged without cooling water.
4	Not Critical	Not required to complete task.
5-6	Critical	Required to complete task.
7-9	Not Critical	Ensure and Recognize steps.
10	Critical	Actions required to complete task.
11	Not Critical	Informing CRS of results.

REVISION SUMMARY

5	ALL NON-TECHNICAL CHANGES: • Corrected revision numbers in revision summary. • Corrected OP-16 section in references from 5.3 to 6.1.3. • Corrected year of signature from 205 to 2015
4	New JPM format. Added Critical/Non Critical step explanation.
3	New JPM format.

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	X	No	____		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

Read the following to the performer.

TASK CONDITIONS:

1. Both Reactor Feed Pumps have tripped and are not available.
2. Reactor level is below 166 inches.
3. HPCI is not available.

INITIATING CUE:

You are directed by the Unit CRS to place RCIC in service per the Hard Card and restore Reactor level to 166 to 206 inches. Notify the Unit CRS when all required actions are complete.

<< RCIC Instructional Aid for EOPs >>

MANUAL RCIC INJECTION
(2OP-16 Section 5.3)

1. **ENSURE** THE FOLLOWING VALVES ARE OPEN: E51-V8 (VALVE POSITION), E51-V8 (ACTUATOR POSITION), AND E51-V9. ☐
2. **OPEN** E51-F046 ☐
3. **START** VACUUM PUMP AND LEAVE SWITCH IN START. ☐
4. **OPEN** E51-F045 ☐
5. **OPEN** E51-F013 ☐
6. **ENSURE** RCIC TURBINE STARTS AND COMES UP TO SPEED AS DIRECTED BY RCIC FLOW CONTROL ☐
7. **ADJUST** RCIC FLOW CONTROLLER TO OBTAIN DESIRED FLOW RATE. ☐
8. **ENSURE** E51-F019 IS CLOSED WITH FLOW GREATER THAN 80 GPM. ☐
9. **ENSURE** THE FOLLOWING VALVES ARE CLOSED: E51-F025, E51-F026, E51 F004, AND E51-F005 ☐
10. **START** SBGT (1OP-10) ☐
11. **ENSURE** BAROMETRIC CNDSR CONDENSATE PUMP OPERATES ☐

RCIC PRESSURE CONTROL
(2OP-16 SECTION 8.2)

1. **ENSURE** THE FOLLOWING VALVES ARE OPEN: E51-V8 (VALVE POSITION), E51-V8 (ACTUATOR POSITION), AND E51-V9. ☐
2. **OPEN** E51-F046 ☐
3. **START** VACUUM PUMP AND LEAVE SWITCH IN START. ☐
4. **ENSURE** E51-F013 IS CLOSED ☐
5. **ENSURE** E41-F011 IS OPEN ☐
6. **THROTTLE OPEN** E51-F022 UNTIL DUAL INDICATION IS OBTAINED ☐
7. **OPEN** E51-F045 ☐
8. **THROTTLE OPEN** E51-F022 OR **ADJUST** RCIC FLOW CONTROL, E51 FIC R600, TO OBTAIN DESIRED SYSTEM PARAMETERS AND REACTOR PRESSURE. ☐
9. **ENSURE** E51-F019 IS CLOSED WITH FLOW GREATER THAN 80 GPM. ☐
10. **ENSURE** THE FOLLOWING VALVES ARE CLOSED: E51-F025, E51-F026, E51 F004, AND E51-F005. ☐
11. **START** SBGT (2OP-10) ☐
12. **ENSURE** BAROMETRIC CNDSR CONDENSATE PUMP OPERATES ☐

FOR SHUTDOWN: REFER TO 2OP-16

FOR TRANSFER BETWEEN PRESSURE AND LEVEL CONTROL: REFER TO 2OP-16

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

SIM JPM D - 2016 NRC INITIAL EXAM - RO/ISRO

LESSON TITLE: **SUPPRESSION POOL COOLING PER HARD CARD - SW RELEASE**

LESSON NUMBER: **LOT-SIM-JP-017-A12**

REVISION NO: **2**

Dan Hulgin 8/18/16
PREPARER / DATE

Bob Bolin 9/06/16
TECHNICAL REVIEWER / DATE

Shawn Zander 9/06/16
VALIDATOR / DATE

Craig Olver 09/22/2016
LINE SUPERVISOR / DATE

9-27-16
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

205014B101
Start Up RHR In Suppression Pool Cooling Mode Per OP-17

K/A REFERENCE AND IMPORTANCE RATING:

219000 A4.01
Ability to manually operate and/or monitor in the control room: Pumps

REFERENCES:

Hard Card- Emergency Suppression Pool Cooling Using Loop B (2OP-17) S/1064

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

5 Containment Integrity

SAFETY CONSIDERATIONS

None

SETUP INSTRUCTIONS

Recommended Initial Conditions

IC-11, 100% Power, BOC

Required Plant Conditions

1. Activate the fuel failure and Stuck SRV.
2. When MSL Rad Hi-Hi is in alarm and Suppression Pool Temperature is >95° F, scram the reactor, perform scram immediate actions, close the MSIV's, and allow plant conditions to stabilize.
3. Place RHR Loop A in Suppression Pool Cooling, open CSW to vital header.
4. Start both NSW pumps.
5. Place a red cap on RHR and RHR SW Pump 2D.

Triggers

Trigger 1 - K1708JCN, E11-F048B to close.

Malfunctions

CW013F, RHR B HX Tube Leak to 100% on trigger 1.

EL_IALSRB2B, Mechanical Trip RHRSW Booster Pump 2D to Trip on trigger 1

Overrides

ZUA355 Service WTR Effluent Rad High to ON with 10 sec TD on trigger 1

Remotes

RS_IARHBYPB, E11-F068B Auto Close Bypass Switch, to BYPASS.

ED_IABKCJ16, Bkr Ctl DC Fuses RHR SW 2D OUT

ED_IABKCF03, Bkr Ctl DC Fuses RHR 2D OUT

Special Instructions

None

SAFETY CONSIDERATIONS:

1. None.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM will be performed in the simulator on Unit Two.

Read the following to the JPM performer.

TASK CONDITIONS:

1. A Resin Injection has occurred, currently raising power to burn off the resin.
2. RWCU has been isolated.
3. SRV C was not properly seated, subsequently it has been closed.
4. Torus temperature is >95 deg. F.
5. The A Loop of RHR has been started in Suppression Pool Cooling mode using CSW.
6. 2D RHR and 2D RHR SW Pumps are under clearance.

INITIATING CUE:

You are directed to place the B Loop of RHR in Suppression Pool Cooling in accordance with the hard card using NSW and inform the Unit CRS when the required actions are complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Obtain Hard Card for Emergency Suppression Pool Cooling Using Loop B.
Hard Card obtained.

SAT/UNSAT

TIME START: ___________

Step 2 – **Open** SW-V105, Nuc Sw Supply Vlv.
Places control switch for the SW-V105 to OPEN and verifies green light out/red light on. SW-V105 is open (**critical***).*

****CRITICAL STEP**SAT/UNSAT**

Step 3 – **Close** SW-V143, Well Water Supply Vlv.
Verifies control switch for the SW-V143 to CLOSE and verifies green light on/red light off. SW-V143 is closed.

SAT/UNSAT

NOTE:	The performer should recognize that a LOCA signal does not exist and N/A the step for using the manual override switch. RHR SW Pump 2D is under clearance and is not available .

Step 4 – **Start** RHR SW PMP.
Places control switch for RHR SW Booster Pump 2B to START. Verifies green light off/red light on, and verifies RHR SW Booster Pump 2B discharge pressure is rising on SW-PI-1155-1. RHR SW Booster Pump 2B is running (**critical***)*

****CRITICAL STEP**SAT/UNSAT**

Step 5 – **Adjust** E11-F068B, HX 2B Sw Discharge Vlv.
Throttles control switch for E11-F068B to obtain a RHR SW flow of 2000-4000 gpm on E11-FI-R602B.

****CRITICAL STEP**SAT/UNSAT**

NOTE:	SW-V117, NSW to Vital Header, should not be opened since CSW is already supplying the vital header. Opening SW-V117 would cross-tie the CSW and NSW headers. Although an analysis has been performed for this alignment, it is not preferred.

Step 6 – **Establish** CLG WTR TO VITAL HDR.

Verifies that SW-V111, Conventional SW to Vital Header Vlv, is supplying the vital header on A Loop (Red light on/Green Light off).

SAT/UNSAT

NOTE:	The performer should recognize that a LOCA signal does not exist and that the F015B is closed and N/A these two steps on the hard card.

Step 7 – **Start** LOOP B RHR PMP.

Places control switch for RHR Pump 2B to START. Verifies red light on/green light off. Verifies RHR system pressure rises. Verifies RHR Pump pressure rises. 2B RHR Pump running. (**critical***)*

****CRITICAL STEP**SAT/UNSAT**

Step 8 – **Open** E11-F028B, Torus Discharge Isol Vlv.

Places control switch for E11-F028B to OPEN. Verifies red light on/green light off.

****CRITICAL STEP**SAT/UNSAT**

Step 9 – **Throttle** E11-F024B, Torus Cooling Isol Vlv.

Throttles control switch for E11-F024B to obtain 6,000 to 10,000 gpm for 1 RHR pump in operation.

****CRITICAL STEP**SAT/UNSAT**

NOTE:	**ALTERNATE PATH BEGINS AT STEP 10 and 11:** When the performer begins to throttle the F048B the RHR HX tubes will rupture resulting initially in a RHR Hi Conductivity alarm followed by a trip of the RHR SW Pump and the F068B will fail to close resulting in high service water radiation alarm. There are several ways to terminate the release, closing the F068B, stopping the running RHR pumps, or closing F047B and F003B. The critical action is to terminate the release. F068B should be closed as this is an automatic action that failed to occur.

Step 10 – **Throttle** E11-F048B, HX 2B Byp Vlv.

Throttles control switch in CLOSE for E11-F048B.

****CRITICAL STEP**SAT/UNSAT**

Step 11 – **Terminates** the Service Water radiation release.

Closes the F068B and/or stops the running RHR pump(s) and/or closes F047B and F003B.

****CRITICAL STEP**SAT/UNSAT**

PROMPT:	If informed as Unit SRO that B Loop cannot be placed in SPC, inform the performer that another operator will be assigned to complete isolation of RHR Loop B.

Step 12 – **Informs** Unit CRS RHR B Loop cannot be placed in SPC.

CRS informed that RHR Loop B cannot be placed in SPC.

***SAT/UNSAT**

TERMINATING CUE:	When the service water radiation release has been terminated and the Unit CRS is informed this JPM is complete.

TIME COMPLETED: ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Necessary for RHR B loop SPC with Nuc SW.
3	Not Critical	Unnecessary alignment.
4	Critical	Necessary for RHR B loop SPC HX cooling.
5	Critical	Necessary for RHR B loop SPC HX cooling.
6	Not Critical	Unnecessary alignment.
7	Critical	Necessary for RHR flow
8	Critical	Necessary for RHR B loop SPC
9	Critical	Necessary for RHR B loop SPC
10	Critical	Necessary for RHR B loop SPC
11	Critical	Necessary to terminate release
12	Not Critical	Communication

REVISION SUMMARY

2	New template. Enhanced Standards for JPM steps Changed SCO to CRS Added Critical Step delineation
1	Converted format to Word using current JPM template, updated to match the current revision of the procedure. Changed title to LOT from LOR.
0	Initial Revision.

(Provide sufficient detail for reviewers and evaluators to understand the scope of any technical and/or administrative changes).

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	X	No	____		

EVALUATION

Performer: ____

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ____

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

TASK CONDITIONS:

1. A Resin Injection has occurred, currently raising power to burn off the resin.
2. RWCU has been isolated.
3. SRV C was not properly seated, subsequently it has been closed.
4. Torus temperature is >95 deg. F.
5. The A Loop of RHR has been started in Suppression Pool Cooling mode using CSW.
6. 2D RHR and 2D RHR SW Pumps are under clearance.

INITIATING CUE:

You are directed to place the B Loop of RHR in Suppression Pool Cooling in accordance with the hard card using NSW and inform the Unit CRS when the required actions are complete.

<< Emergency Suppression Pool Cooling Using Loop B (2OP-17) >>

NOTE
This attachment is not to be used for normal system operations. ☐

Start RHR SW B LOOP (NUC)

- **Open** SW-V105 ☐
- **Close** SW-V143 ☐
- **Start** CSW PUMPS AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT, **THEN place** RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- **Start** RHR SW PMP ☐
- **Adjust** E11-PDV-F068B ☐
- **Establish** CLG WTR TO VITAL HDR ☐
- **Start** ADDITIONAL RHR SW PUMP and **adjust** FLOW AS NEEDED ☐

Start RHR SW B LOOP (CONV)

- **Open** SW-V101 ☐
- **Open** SW-V102 ☐
- **Close** SW-V143 ☐
- **Start** PUMPS ON NSW HDR AS NEEDED ☐
- IF LOCA SIGNAL IS PRESENT, **THEN place** RHR SW BOOSTER PUMPS B & D LOCA OVERRIDE SWITCH TO MANUAL OVERRIDE ☐
- **Start** RHR SW PMP ☐
- **Adjust** E11-PDV-F068B ☐
- **Establish** CLG WTR TO VITAL HDR ☐
- **Start** ADDITIONAL RHR SW PUMP and **adjust** FLOW AS NEEDED ☐

Start RHR LOOP B

- IF LOCA SIGNAL IS PRESENT, **THEN** ☐
- **Verify** COOLING LOGIC IS MADE UP
- IF E11-F015B IS OPEN, **THEN close** E11-F017B ☐
- **Start** LOOP B RHR PMP ☐
- **Open** E11-F028B ☐
- **Throttle** E11-F024B ☐
- **Throttle** E11-F048B ☐
- **Start** ADDITIONAL LOOP A RHR PMP and **adjust** FLOW AS NEEDED ☐

2 2/1063
2 S/1064

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

SIM JPM E - 2016 NRC INITIAL EXAM - RO/ISRO

LESSON TITLE: **VENT THE DRYWELL PER OP-10 W/ STACK RAD MONITOR INCREASE >50%**

LESSON NUMBER: **LOT-SIM-JP-010-A02**

REVISION NO: **8**

Dan Hulgin 8/18/16
PREPARER / DATE

Bob Bolin 9/06/16
TECHNICAL REVIEWER / DATE

Grant Newton 9/06/16
VALIDATOR / DATE

Craig Oliver 09/27/2016
LINE SUPERVISOR / DATE

9-27-16
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

261008B101
Perform Normal Primary Containment Venting

K/A REFERENCE AND IMPORTANCE RATING:

261000 A4.01
Ability to manually operate and/or monitor in the Control Room Off site Release Rate

REFERENCES:

2OP-10,Section 6.3.2-Venting Containment Via SBGT

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

9 Radioactivity Release

SAFETY CONSIDERATIONS

None

SETUP INSTRUCTIONS

Recommended Initial Conditions

IC-11, 100% Power, BOC

Required Plant Conditions

1. Drywell Pressure above 0.5 psig SLOWLY rising or stable, AND below 1.8 psig.

Triggers

Trigger 1 Q6225LGT CAC-V23 Green Lamp = False.

Malfunctions

None

Overrides

Event	Panel	Tag	Title	Value (ramp rate)	Activate Time (sec)	Deactivate Time (sec)
E1	XU-3	G5B02G15	Main Stack Radiation	2.48 / 2 min	0 SEC	N/A

Remotes

None

Special Instructions

1. Secure Drywell Coolers 2C and 2D Fans 1 and 2
2. Allow drywell pressure to rise to 0.6 psig as indicated on CAC-PI-2685-1 on XU-51.
3. Restart Drywell Coolers 2D Fan 2 and allow Drywell pressure to stabilize.
4. Override Drywell Cooler 2C Fans 1 and 2 and Drywell Cooler 2D Fan 1 control switches OFF

SAFETY CONSIDERATIONS:

1. None.

EVALUATOR NOTES: (Do not read to performer)

1. A marked up copy of 2OP-10, Section 6.3.2 **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM will be performed in the simulator on Unit Two.

Read the following to the JPM performer.

TASK CONDITIONS:

1. Drywell pressure is above normal due to a partial loss of Drywell Cooling.
2. Standby Gas Treatment System is in the Standby Alignment.
3. The plant stack radiation monitor is in service and CAC-CS-5519, CAC Purge Vent Isolation Override is in OFF.
4. ERFIS is unavailable.

INITIATING CUE:

The Unit CRS directs you to vent the Drywell via Standby Gas Treatment, and inform him (her) when drywell pressure has been reduced below 0.5 psig.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Obtain copy of 2OP-10 Standby Gas Treatment System Operating Procedure, Section 6.3.2.

Copy of 2OP-10 Standby Gas Treatment System Operating Procedure, Section 6.3.2 is obtained.

SAT/UNSAT

TIME START: ___________

Step 2 – **Record** 2-D12-RR-R600B (Stack Rad Monitor) digital point display.

Records the 2-D12-RR-R600B (Stack Rad Monitor) digital point display in the space provided on step 6.3.2.2a (value of ~1.12 E1 or 11.2).

SAT/UNSAT

Step 3 – **Multiply** the value obtained in Step 2.a by 1.5 to obtain the value for a 50% rise in stack radiation monitor reading.

Records the value obtained in step 6.3.2.2a (~1.12 E1 or 11.2) in the space provided on step 6.3.2.2b. Multiplies the value by 1.5, and records the product (~1.68 E1 or 16.8) in the space provided in step 6.3.2.2b.

SAT/UNSAT

PROMPT: If asked, sign the procedure as the IV for the calculation. Do not change the value if calculated incorrectly.

Step 4 – **Close** 2-VA-2D-BFV-RB (Reactor Building SBGT Train 2A Inlet Valve).

Reactor Building SBGT Train 2A Inlet Valve 2-VA-2D-BFV-RB switch is rotated counterclockwise to the close position and held (throttle valve) for 10 seconds after the red light is extinguished and the green light is illuminated at which time it can be released to the neutral position. 2-VA-2D-BFV-RB is closed (****critical***).*

****CRITICAL STEP**SAT/UNSAT**

Step 5 – **Close** 2-VA-2H-BFV-RB (Reactor Building SBGT Train 2B Inlet Valve).

Reactor Building SBGT Train 2B Inlet Valve 2-VA-2H-BFV-RB switch is rotated counterclockwise to the close position and held (throttle valve) for 10 seconds after the red light is extinguished and the green light is illuminated at which time it can be released to the neutral position. 2-VA-2H-BFV-RB is closed (**critical**).*

****CRITICAL STEP**SAT/UNSAT**

Step 6 – **Open** 2-VA-2F-BFV-RB (SBGT DW Suct Damper).

SBGT DW Suction Damper 2-VA-2F-BFV-RB control switch is rotated clockwise to the open position and then released. Observes red light illuminates and the green light goes out. 2-VA-2F-BFV-RB is open (**critical**).*

****CRITICAL STEP**SAT/UNSAT**

PROMPT:	If asked as CRS, direct performer to vent the drywell only.

Step 7 – **Open** 2-CAC-V9 (DW Purge Exh Vlv).

DW Purge Exh Vlv 2-CAC-V9 switch is rotated clockwise from the close position to the open position and then released to the neutral position. Observes the red light illuminates and the green light goes out. 2-CAC-V9 is open (**critical**).*

****CRITICAL STEP**SAT/UNSAT**

SIM OP:	When CAC-V23 is opened, verify Trigger 1 initiates to ramp Main Stack Rad Monitor value.

Step 8 – **Open** 2-CAC-V23 (DW Purge Exh Vlv).

DW Purge Exh Vlv 2-CAC-V23 switch is rotated clockwise from the close position to the open position and then released to the neutral position. Observes the red light illuminates and the green light goes out. 2-CAC-V23 is open (**critical**).*

****CRITICAL STEP**SAT/UNSAT**

PROMPT: If requested as CRS, inform performer that it is desired to vent from the drywell head (additional vent capacity is desired).

Step 9 – **IF** additional vent capacity is desired, **THEN open** 2-CAC-V49 (DW Head Purge Exh Vlv).

2-CAC-V49 switch is rotated clockwise from the close position to the open position and then released to the neutral position. Observes the red light illuminates and the green light goes out. 2-CAC-V49 is open.

SAT/UNSAT

Step 10 – **IF** additional vent capacity is desired, **THEN open** 2-CAC-V50 (DW Head Purge Exh Vlv).

2-CAC-V50 switch is rotated clockwise from the close position to the open position and then released to the neutral position. Observes the red light illuminates and the green light goes out. 2-CAC-V50 is open.

SAT/UNSAT

Step 11 – On Panel XU-3, **monitor** 2-D12-RR-R600B (Stack Rad Monitor) for a rise in activity during the performance of this procedure.

Monitors 2-D12-RR-R600B for a rise in activity, and determines Stack Rad Monitor reading has risen by 50%.

SAT/UNSAT

NOTE:	It is critical for at least one valve to be closed in each vent path that is open, i.e., CAC-V23 or CAC-V9, AND, CAC-V49 or CAC-V50, or that the primary containment suction valve VA-2F-BFV-RB is closed to isolate the release path.
SIM OP:	When the vent path has been isolated, delete the meter override on the Main Stack Rad Monitor.
PROMPT:	If the examinee informs the Unit CRS that the Main Stack has risen by >50%, direct examinee as Unit CRS to perform required actions for the increase.
NOTE:	Either Step 12 OR Step 13 is CRITICAL.
PROMPT:	Another operator is available to perform Independent Verifications.
	****ALTERNATE PATH BEGINS AT STEP 12****

Step 12 – **IF** stack radiation rises to greater than the value determined in Section 6.3.2 Step 2.b, **THEN perform** the following to secure venting the drywell: **Close** 2-CAC-V23 (DW Purge Exh Vlv).

2-CAC-V23 control switch is rotated counterclockwise to the close position. Observes the red light goes out and the green light illuminates. 2-CAC-V23 is closed (****critical***).*

****CRITICAL STEP**SAT/UNSAT**

Step 13 – **IF** stack radiation rises to greater than the value determined in Section 6.3.2 Step 2.b, **THEN perform** the following to secure venting the drywell **Close** 2-CAC-V9 (Drywell Purge Exh Vlv).

2-CAC-V9 control switch is rotated counterclockwise to the close position. Observes the red light goes out and the green light illuminates. 2-CAC-V9 is closed (****critical***).*

****CRITICAL STEP**SAT/UNSAT**

NOTE:	Either Step 14 OR Step 15 is **CRITICAL** if the 2-CAC-49 and V50 were opened.

Step 14 – **IF** stack radiation rises to greater than the value determined in Section 6.3.2 Step 2.b, **THEN perform** the following to secure venting the drywell **Ensure** 2-CAC-V49 (DW Head Purge Exh Vlv) is CLOSED.

2-CAC-V49 control switch is rotated counterclockwise to the close position. Observes the red light goes out and the green light illuminates. 2-CAC-V49 is closed (****critical***).*

****CRITICAL STEP**SAT/UNSAT**

Step 15 – **IF** stack radiation rises to greater than the value determined in Section 6.3.2 Step 2.b, **THEN perform** the following to secure venting the drywell **Ensure** 2-CAC-V50 (DW Head Purge Exh Vlv) is CLOSED.

2-CAC-V50 control switch is rotated counterclockwise to the close position. Observes the red light goes out and the green light illuminates. 2-CAC-V50 is closed (****critical***).*

****CRITICAL STEP**SAT/UNSAT**

PROMPT:	If asked, inform examinee as Unit CRS that E&RC has been notified to sample primary containment, and to reference E&RC 2020 Setpoint Determinations for Gaseous Radiation Monitors (Noble Gas Instantaneous Release Rate Determination).

NOTE:	Step 16 is not critical if either step 12 or 13 AND either 14 or 15 was completed SAT. Release path may be isolated by closing 1 valve in each vent path OR by closing the common isolation in Step 16.

Step 16 – **CLOSE** SBGT DW SUCT DAMPER, 2-VA-2F-BFV-RB.

SBGT DW Suction Damper 2-VA-2F-BFV-RB control switch is rotated counter-clockwise to the closed position and then released. Observes green light illuminates and the red light goes out. 2-VA-2F-BFV-RB is closed (****critical***).*

****CRITICAL STEP**SAT/UNSAT**

NOTE:	The following valves would auto open on SBGT Initiation therefore Steps 17 and 18 are NOT critical.

Step 17 – **Open** 2-VA-2H-BFV-RB (SBGT Train 2B Reactor Building Suction Valve).

Verifies 2-VA-2H-BFV-RB indicates full open. Observes green light out and the red light lit. 2-VA-2H-BFV-RB is full open.

SAT/UNSAT

Step 18 – **Open** 2-VA-2D-BFV-RB (SBGT Train 2A Reactor Building Suction Valve).

Verifies 2-VA-2D-BFV-RB indicates full open. Observes green light out and the red light lit. 2-VA-2D-BFV-RB is full open.

SAT/UNSAT

PROMT:	If asked, inform examinee as Unit CRS that another operator is standing by to perform 0PT-02.3.1b, Suppression Pool to Drywell Vacuum Breaker Position Check.

Step 19 – **Inform** Unit CRS that venting is secured due to increase of 50% in Main Stack Rad Monitor reading.

Unit CRS is informed venting is secured due to increase of 50% in Main Stack Rad Monitor reading.

SAT/UNSAT

TERMINATING CUE:	When Primary containment Venting has been secured and the Unit CRS is notified, this JPM is complete.

TIME COMPLETED: ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Not Critical	Readings
3	Not Critical	Readings
4	Critical	Necessary for venting alignment
5	Critical	Necessary for venting alignment
6	Critical	Necessary for venting alignment
7	Critical	Necessary for venting alignment
8	Critical	Necessary for venting alignment
9	Not Critical	Additional venting alignment
10	Not Critical	Additional venting alignment
11	Not Critical	Monitoring
12	Critical	Communication
13	Critical	Necessary for termination of release
14	Critical	Necessary for termination of release
15	Critical	Necessary for termination of release
16	Critical	Necessary for termination of release
17	Not Critical	Auto-action for securing vent
18	Not Critical	Auto-action for securing vent
19	Not Critical	Communication

REVISION SUMMARY

8	Enhanced Standards for JPM steps Added Critical Step delineation Fixed numbering
7	Updated to the new JPM template.

(Provide sufficient detail for reviewers and evaluators to understand the scope of any technical and/or administrative changes).

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	X	No	____		

EVALUATION

Performer: ______________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: __________

TASK CONDITIONS:

1. Drywell pressure is above normal due to a partial loss of Drywell Cooling.
2. Standby Gas Treatment System is in the Standby Alignment.
3. The plant stack radiation monitor is in service and CAC-CS-5519, CAC Purge Vent Isolation Override is in OFF.
4. ERFIS is unavailable.

INITIATING CUE:

The Unit CRS directs you to vent the Drywell via Standby Gas Treatment, and inform him (her) when drywell pressure has been reduced below 0.5 psig.

6.3.2 Venting Containment Via SBGT

Date/Time Started_______________

1. **Confirm** the following Initial Conditions are met:
   - Drywell pressure has risen to greater than 0.15 psig. .... ________
   - SBGT System is in STANDBY in accordance with Section 6.1.1. .... ________
   - One of the following:
     - ◊ Plant stack radiation monitor is in service and CAC-CS-5519 (CAC Purge Vent Isol Ovrd) is in OFF. ..... ________
     - ◊ E&C has sampled the drywell atmosphere and has determined that it is suitable for release. .... ________
   - Unit CRS approval is obtained prior to venting. .... ________

NOTE

- Backwashing an RWCU filter or AOG sampling may cause stack radiation to rise. Venting primary containment and RWCU backwashing or AOG effluent sampling are **NOT** done concurrently. .... ☐
- To aid in monitoring for a 50% rise in stack radiation level, value monitoring alarm may be set up to monitor ERFIS point. .... ☐

2. **IF** 2-D12-RR-R600B (Stack Rad Monitor) is OPERABLE **THEN perform** the following: .... ________
   a. **Record** 2-D12-RR-R600B (Stack Rad Monitor) digital point display. .... ________

      2-D12-RR-R600B: ____________________

   b. **Multiply** the value obtained in Step 2.a by 1.5 to obtain the value for a 50% rise in stack radiation monitor reading. .... ____/____ IV

      (1.5) X (_____________) = ____________
      (Section 6.3.2 Step 2.a)

3. **Close** 2-VA-2D-BFV-RB (Reactor Building SBGT Train 2A Inlet Valve). .... ________

6.3.2 Venting Containment Via SBGT (continued)

4. **Close** 2-VA-2H-BFV-RB (Reactor Building SBGT Train 2B Inlet Valve)........ _________

5. **Open** 2-VA-2F-BFV-RB (SBGT DW Suct Damper)........ _________

CAUTION
Simultaneous venting of the drywell and the suppression pool is **<u>NOT</u>** performed when the plant is in MODE 1, 2, or 3. {8.1.1}........ ☐

6. **<u>IF</u>** venting the suppression chamber, **<u>THEN</u> perform** the following:........ _________

   a. **Open** 2-CAC-V172 (Supp Pool Purge Exh Vlv)........ _________

   b. **Open** 2-CAC-V22 (Torus Purge Exhaust Vlv)........ _________

   c. **<u>IF</u>** 2-D12-RR-R600B (Stack Rad Monitor) is OPERABLE **<u>THEN</u> perform** the following:........ _________

      (1) On Panel XU-3, **monitor** 2-D12-RR-R600B (Stack Rad Monitor) for a rise in activity during the performance of this procedure........ _________

      (2) **<u>IF</u>** stack radiation rises to greater than the value determined in Step 2.b, **<u>THEN</u> perform** the following to secure venting the suppression pool:........ _________

         - **Close** 2-CAC-V172 (Supp Pool Purge Exh Vlv)..... ____/____ IV

         - **Close** 2-CAC-V22 (Torus Purge Exhaust Vlv)...... ____/____ IV

6.3.2 Venting Containment Via SBGT (continued)

- **Notify** E&C to perform the following:
  - ◊ **Sample** primary containment. .... _________
  - ◊ **Refer to** 0E&RC-2020, Setpoint Determinations for Gaseous Radiation Monitors (Noble Gas Instantaneous Release Rate Determination). .... _________

____
Person Notified

d. **WHEN** the desired suppression chamber pressure is reached, as indicated on Computer Point L128, **THEN close** the following valves: .... _________

- 2-CAC-V172 (Supp Pool Purge Exh Vlv)........................ ____/____ IV
- 2-CAC-V22 (Torus Purge Exhaust Vlv) .... ____/____ IV

7. **IF** venting the drywell, **THEN perform** the following:.. _________

a. **Open** 2-CAC-V9 (DW Purge Exh Vlv).. _________

b. **Open** 2-CAC-V23 (DW Purge Exh Vlv)...................................... _________

NOTE

CAC-V49 (DW Head Purge Exh Vlv) and CAC-V50 (DW Head Purge Exh Vlv) are **NOT** normally opened, but are available if additional vent capacity is desired..................... ☐

c. **IF** additional vent capacity is desired, **THEN open** 2-CAC-V49 (DW Head Purge Exh Vlv). .... _________

d. **IF** additional vent capacity is desired, **THEN open** 2-CAC-V50 (DW Head Purge Exh Vlv). .... _________

e. On Panel XU-3, **monitor** 2-D12-RR-R600B (Stack Rad Monitor) for a rise in activity during the performance of this procedure. .... _________

6.3.2 Venting Containment Via SBGT (continued)

f. **IF** 2-D12-RR-R600B (Stack Rad Monitor) is OPERABLE **THEN perform** the following: .... ________

(1) On Panel XU-3, **monitor** 2-D12-RR-R600B (Stack Rad Monitor) for a rise in activity during the performance of this procedure. .... ________

(2) **IF** stack radiation rises to greater than the value determined in Section 6.3.2 Step 2.b, **THEN perform** the following to secure venting the drywell: .... ________

- **Close** 2-CAC-V23 (DW Purge Exh Vlv)............... ____/____ IV

- **Close** 2-CAC-V9 (Drywell Purge Exh Vlv). .... ____/____ IV

- **Ensure** 2-CAC-V49 (DW Head Purge Exh Vlv) is CLOSED. .... ____/____ IV

- **Ensure** 2-CAC-V50 (DW Head Purge Exh Vlv) is CLOSED. .... ____/____ IV

- **Notify** E&C to perform the following:

  ◊ **Sample** primary containment..................... ________

  ◊ **Refer to** 0E&RC-2020, Setpoint Determinations for Gaseous Radiation Monitors (Noble Gas Instantaneous Release Rate Determination)..................... ________

____
Person Notified

g. **WHEN** the desired drywell pressure is reached as indicated on CAC-PI-2685-1 (Drywell Pressure) on Panel XU-51, **THEN perform** the following: .... ________

- **Close** 2-CAC-V23 (DW Purge Exh Vlv). .... ____/____ IV

6.3.2 Venting Containment Via SBGT (continued)

- **Close** 2-CAC-V9 (Drywell Purge Exh Vlv)...................... ____/____ IV

- **Ensure** 2-CAC-V49 (DW Head Purge Exh Vlv) is CLOSED.. ____/____ IV

- **Ensure** 2-CAC-V50 (DW Head Purge Exh Vlv) is CLOSED.. ____/____ IV

8. **Close** 2-VA-2F-BFV-RB (SBGT DW Suct Damper)............................ ____/____ IV

9. **Open** 2-VA-2H-BFV-RB (SBGT Train 2B Reactor Building Suction Valve).. ____/____ IV

10. **Open** 2-VA-2D-BFV-RB (SBGT Train 2A Reactor Building Suction Valve).. ____/____ IV

NOTE

Technical Specification 3.6.1.6.1 (MODES 1, 2, or 3) requires completion of 0PT-02.3.1B, Suppression Pool To Drywell Vacuum Breaker Position Check, within 6 hours following an operation that causes any of the vacuum breakers to open. Due to the inability to detect a vacuum breaker opening and subsequently reclosing, other than observation from a dedicated Operator, the conservative approach is to perform 0PT-02.3.1B, Suppression Pool To Drywell Vacuum Breaker Position Check following venting activities. .... ☐

11. <u>IF</u> in MODES 1, 2, or 3,
<u>THEN</u> perform 0PT-02.3.1B, Suppression Pool To Drywell Vacuum Breaker Position Check, within 6 hours following an operation that causes any of the vacuum breakers to open. .... ________

6.3.2 Venting Containment Via SBGT (continued)

Date/Time Completed ____________

Performed By (Print) Initials

____________________ ______

____________________ ______

____________________ ______

____________________ ______

Reviewed By ____________________________

Unit CRS/SRO

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

SIM JPM F - 2016 NRC INITIAL EXAM - RO/ISRO

LESSON TITLE: **Shifting Caswell Beach Lube Water Pumps From The RTGB**

LESSON NUMBER: **LOT-SIM-JP-029-01**

REVISION NO: **0**

Dan Hulgin 8/18/16
PREPARER / DATE

Bob Bolin 9/06/16
TECHNICAL REVIEWER / DATE

Dwayne Wolf
Grant Newton 9/06/16
VALIDATOR / DATE

Craig Olive 09/22/2016
LINE SUPERVISOR / DATE

9-26-16
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

275002B101 Startup The Circulating Water System Per OP-29

K/A REFERENCE AND IMPORTANCE RATING:

400000 A4.01 Ability to manually operate and/or monitor in the control room: CCW indications and control.

REFERENCES:

2OP-29, Section 6.3.27, Shifting Caswell Beach Lube Water Pumps From The RTGB

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

8 - Plant Service Systems (Component Cooling Water System)

SETUP INSTRUCTIONS

SIMULATOR SETUP

Initial Conditions

1. Recommended Initial Conditions

   IC-11

2. Required Plant Conditions

   None.

Malfunctions:

None.

Special Instructions

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM will be performed in the simulator on Unit Two.

Read the following to the JPM performer.

TASK CONDITIONS:

1. An Auxiliary operator is stationed at Caswell Beach.
2. All Section 5.0 prerequisites of 2OP-29, *Circulating Water System* are met.

INITIATING CUE:

You are directed by the Unit CRS to place Caswell Beach Bearing Lube Water pump 2B in service, and secure the Caswell Beach Bearing Lube Water pump 2A IAW 2OP-29 Section 6.3.27, *Shifting Caswell Beach Lube Water Pumps From The RTGB.* Inform the CRS when complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 0 - May perform take a minute at job site prior to beginning task.
Examinee may cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Am I using appropriate gloves? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

Step 1 - Obtain current revision of 2OP-29 Section 6.3.27.
Provide current revision of 2OP-29 Section 6.3.27.

SAT/UNSAT

TIME START: ____________

PROMPT: When contacted as the Auxiliary Operator at Caswell Beach, acknowledge the communication, and report that you are standing by for the pump shift.

Step 2 - **Establish** communication with Auxiliary Operator at Caswell Beach.
Contacts the AO at Caswell Beach.

SAT/UNSAT

NOTE: When the Point Select push button is depressed it initiates a 10 second window for starting or stopping Lube Water pumps. Place keeping of the pump start steps and pump stop steps may be deferred until after the Lube Water pump is running or stopped.

Step 3 – **Start** the non-operating Bearing Lube Water pump as follows:

a. **Depress** Point Select push button for the selected Bearing Lube Water pump.
Depresses Point Select push button for the Caswell Beach Bearing Lube Water pump 2B on Panel XU-2 and the checkback lamp comes ON.

****CRITICAL STEP**** **SAT/UNSAT**

Step 4 - **Start** the non-operating Bearing Lube Water pump as follows:

b. **WHEN** checkback lamp in the selected point push button comes ON, **THEN place** the Bearing Lube Water pump control switch to START.

Places the control switch for the Caswell Beach Bearing Lube Water pump 2B to START, within 10 seconds of the checkback lamp for the Caswell Beach Bearing Lube Water pump 2B point push button coming ON and the Caswell Beach Bearing Lube Water pump 2B starts.

****CRITICAL STEP** SAT/UNSAT**

PROMPT:	When contacted as the Auxiliary Operator at Caswell Beach, acknowledge the communication , and report a good start on the Caswell Beach Bearing Lube Water pump 2B.

Step 5 - **Start** the non-operating Bearing Lube Water pump as follows:

c. **Confirm** with the Auxiliary Operator that the pump starts without unusual noise or cavitation.

Confirms with the Auxiliary Operator that the Caswell Beach Bearing Lube Water pump 2B has started without unusual noise or cavitation.

SAT/UNSAT

Step 6 - **Stop** the previously operating Bearing Lube Water pump as follows:

a. **Depress** Point Select push button for the selected Bearing Lube Water pump.

Depresses Point Select push button for the Caswell Beach Bearing Lube Water pump 2A on Panel XU-2 and the checkback lamp comes ON.

**** CRITICAL STEP ** SAT/UNSAT**

Step 7 - **Stop** the previously operating Bearing Lube Water pump as follows:

b. **WHEN** checkback lamp in the selected point push button comes ON, **THEN place** the Bearing Lube Water pump control switch to STOP.

Places the control switch for the Caswell Beach Bearing Lube Water pump 2A to STOP, within 10 seconds of the checkback lamp for the Caswell Beach Bearing Lube Water pump 2A point push button coming ON and the Caswell Beach Bearing Lube Water pump 2A stops.

**** CRITICAL STEP ** SAT/UNSAT**

PROMPT:	When contacted as the Auxiliary Operator at Caswell Beach, acknowledge the communication , and report that the Caswell Beach Bearing Lube Water pump 2A has stopped, and that lube water flow is adequate for all CWOD pumps.

Step 8 – **Stop** the non-operating Bearing Lube Water pump as follows:

c. **Confirm** with the Auxiliary Operator that the pump stops.

Confirms with the Auxiliary Operator that the Caswell Beach Bearing Lube Water pump 2A has stopped.

SAT/UNSAT

Step 9 – **Informs** CRS that the Caswell Beach Lube Water Pumps have been shifted with the 2B pump now running and the 2A pump secured.

Informs CRS that the pump shift is complete.

SAT/UNSAT

TERMINATING CUE:	Once Caswell Beach Bearing Lube Water pump 2B is running ,pump 2A is secured, and the CRS has been notified , the JPM can be terminated.

TIME COMPLETED: ______________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Not Critical	Communication
3	Critical	Action required to start pump
4	Critical	Action required to start pump
5	Not Critical	Communication
6	Critical	Action required to stop pump
7	Critical	Action required to stop pump
8	Not Critical	Communication
9	Not Critical	Communication

REVISION SUMMARY

0	New JPM written for 2016 Initial NRC exam.

Validation Time: 10 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

TASK CONDITIONS:

1. An Auxiliary operator is stationed at Caswell Beach.

2. All Section 5.0 prerequisites of 2OP-29, *Circulating Water System* are met.

INITIATING CUE:

You are directed by the Unit CRS to place Caswell Beach Bearing Lube Water pump 2B in service, and secure the Caswell Beach Bearing Lube Water pump 2A IAW 2OP-29 Section 6.3.27, *Shifting Caswell Beach Lube Water Pumps From The RTGB.* Inform the CRS when complete.

6.3.27 Shifting Caswell Beach Lube Water Pumps From The RTGB

1. **Confirm** the following initial Conditions are met:

   - All applicable prerequisites listed in Section 5.0, Prerequisites are met. .... _________
   - An Auxiliary Operator is available at Caswell Beach .... _________

2. **Establish** communication with Auxiliary Operator at Caswell Beach. ... _________

NOTE
When the Point Select push button is depressed it initiates a 10 second window for starting or stopping Lube Water pumps. Place keeping of the pump start steps and pump stop steps may be differed until after the Lube Water pump is running or stopped.. .... ☐

3. **Start** the non-operating Bearing Lube Water pump as follows:

   a. **Depress** Point Select push button for the selected Bearing Lube Water pump.. _________

   b. <u>**WHEN**</u> checkback lamp in the selected point push button comes ON,
      <u>**THEN**</u> **place** the Bearing Lube Water pump control switch to START. .... _________

   c. **Confirm** with the Auxiliary Operator that the pump starts without unusual noise or cavitation. .... _________

4. **Stop** the previously operating Bearing Lube Water pump as follows:

   a. **Depress** Point Select push button for the selected Bearing Lube Water pump.. _________

   b. <u>**WHEN**</u> checkback lamp in the selected point push button comes ON,
      <u>**THEN**</u> **place** the Bearing Lube Water pump control switch to STOP. .... _________

   c. **Confirm** with the Auxiliary Operator that the pump stops. .... _________

5. **Ensure** lube water flow is adequate for all CWOD pumps per 0OI-03.11, Auxiliary Operator U0 Outside Electronic Rounds (DSR & CS) .... _________

6.3.27 Shifting Caswell Beach Lube Water Pumps From The RTGB (continued)

Date/Time Completed ____________

Performed By (Print)	Initials

Reviewed By ________________________
Unit CRS/SRO

DUKE ENERGY

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

SIM JPM G - 2016 NRC INITIAL EXAM - RO/ISRO

LESSON TITLE: **SUBSTITUTING A CONTROL ROD POSITION INTO THE RWM**

LESSON NUMBER: **LOT-SIM-JP-007-B02**

REVISION NO: **3**

Dan Hulgin 08/18/16
PREPARER / DATE

Bob Bolin 09/06/16
TECHNICAL REVIEWER / DATE

Dwayne Wolf 09/06/16
VALIDATOR / DATE

Craig Oliver 09/22/2016
LINE SUPERVISOR / DATE

9-28-16
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

214202B101
Determine The RWM Substitute Rod Position For A Failed Reed Switch Position Indicator Per OP-07

K/A REFERENCE AND IMPORTANCE RATING:

201006 A4.06 3.2/3.2
Ability to manually operate and/or monitor in the control room: Selected rod position indication

REFERENCES:

2OP-07, Section 6.3.11 Determination Of The RWM Substitute Position
0OI-53, Rod Worth Minimizer (NUMAC-RWM)

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

7 Instrumentation

SAFETY CONSIDERATIONS

None

SETUP INSTRUCTIONS

Recommended Initial Conditions

IC-11, 100% Power, BOC

Required Plant Conditions

Fail the Reed switch for Control Rod 22-03 position 48.

Triggers

None

Malfunctions

System	Tag	Title	Value
RD	RD179M	Reed Switch Failure Rod 22-03	48

Overrides

None

Remotes

None

Special Instructions

None

SAFETY CONSIDERATIONS:

1. None.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable copy of 2OP-07, Section 6.3.11 WILL be provided to the performer.
2. If requested, a copy of 0OI-53, WILL be provided to the performer.
3. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
4. This JPM will be performed in the simulator on Unit Two.

Read the following to the JPM performer.

TASK CONDITIONS:

1. Control rod 22-03 has just been positioned at position 48.
2. Prerequisites listed in section 5.0 of 2OP-07 are met.
3. The on duty Reactor Engineer has been notified.
4. The Unit CRS has reviewed Technical Specifications for applicability and has given permission to perform this procedure.

INITIATING CUE:

You are directed to enter a substitute value for Control Rod 22 03 into the RWM, and inform the CRS when complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - **Obtain** a copy of 2OP-07, Reactor Manual Control System Operating Procedure, Section 6.3.11.

Copy of 2OP-07, Reactor Manual Control System Operating Procedure, Section 6.3.11 is obtained.

SAT/UNSAT

TIME START: ___________

PROMPT:	Role play as a Concurrent Verification, if requested. DO NOT correct the performer.
PROMPT:	Role play as CRS, to initial Steps 6.3.11.2, 4 & 5 when asked.

Step 2 – Using Concurrent Verification, **Insert** or **withdraw** control rod one additional notch to an operable control rod reed switch position indicator.

- *Turns on Rod Select Power by placing the control switch to ON.*
- *Selects Control Rod 22-03 on the RTGB select matrix by depressing its* Control Rod Select pushbutton
- *Inserts Control Rod 22-03 to position 46 using the Rod Movement control switch in the IN position for one notch.*
- *Control Rod 22-03 is at position 46* (**critical***).*
- *Control Rod 22-03 is selected* (**critical***).*

****CRITICAL STEP**SAT/UNSAT**

Step 3 – **Record** the OPERABLE control rod reed switch position below.

Records Control Rod 22-03 at position 46 for step 6.3.11.6.c in the space labeled "Operable Control Rod Reed Switch Position".

SAT/UNSAT

Step 4 – **Restore** control rod to the position of the failed control rod reed switch position indicator.

- *Withdraws Control Rod 22-03 to position 48 using the Rod Movement control switch.*
- *May use continuous withdraw and perform an over travel check -OR- May single notch out to 48 and then attempt to notch out past position 48.*
- *Control Rod 22-03 is at position 48* (**critical***).*

****CRITICAL STEP**SAT/UNSAT**

Step 5 – **Ensure** inferred position offered by the RWM deviates by one notch and in the correct direction from the position determined in Section 6.3.11 Step 6.c.

Verifies RWM infers substitute value of 48.

SAT/UNSAT

Step 6 – **Record** the inferred rod position below.

Records 48 for step 6.3.11.6.g in the space labeled "Inferred Position".

SAT/UNSAT

NOTE: 0OI-53 may be used for guidance in performance of step 7.

Step 7 – **IF** a valid inferred position from the RWM **OR** valid rod position determined by appropriate methods (as identified in Section 6.1.1) has been obtained, **THEN perform** the following: Substitute the valid rod position into RWM.

- *At the RWM Operator's Console on the RTGB, depresses the ETC softkey to obtain the SUBSTITUTE OPTIONS softkey.*
- *Depresses the SUBSTITUTE OPTIONS softkey to change to the SUBSTITUTE OPTION screen.*
- *Verifies RWM offers an inferred position of 48 as the substitute position or depresses the increment/decrement softkey to adjust the substitute position to 48.*
- *Depresses the ENTER SUBSTITUTE softkey.*
- *Depresses the EXIT softkey to return to the main menu screen*
- *48 substituted as the position for Control Rod 22-03* (**critical***).*

****CRITICAL STEP**SAT/UNSAT**

PROMPT:	Inform trainee as the Unit CRS that Reactor Engineer will enter the substitute value in the PPC. Also another operator will execute a CORE MON, and make the Log entries.

Step 8 – **Notify** Unit CRS that a substitute value of 48 has been entered for Control Rod 22-03 per OP-7.0.

Unit CRS notified.

SAT/UNSAT

TERMINATING CUE:	When Control rod 22-03 has been given a substitute position of 48 in the RWM, and the CRS has been notified, this JPM is complete.

TIME COMPLETED: ______________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Rod selected is necessary to substitute rod position in step 7. Rod at position 46 is necessary for RWM to infer correct position.
3	Not Critical	Recording value
4	Critical	Rod at position 48 is necessary for RWM to infer correct position.
5	Not Critical	Verification
6	Not Critical	Recording value
7	Critical	Necessary to substitute value.
8	Not Critical	Communication

REVISION SUMMARY

3	New Format. Critical Steps added based on necessity for RWM inferred position. Standards enhanced Critical Step delineation table added. Changed SCO to CRS. Corrected procedure section. Added additional Notes and Prompts Validation time changed to 15 minutes based on validators' time.
2	Convert to Word, changed title from LOR to LOT

(Provide sufficient detail for reviewers and evaluators to understand the scope of any technical and/or administrative changes).

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ______________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: __________

ATTACHMENT 5
Page 1 of 4
Control Rod Movement

C
Continuous
Use

The purpose of this attachment is to document the rod pattern prior to power change. Enter the rod position information or attach Display 810 Edit.

Unit: 2

51					48	48	48	48	48				
47			48	48	48	48	48	48	48	48	48		
43		48	48	48	48	48	36	48	48	48	48	48	
39		48	48	48	48	48	48	48	48	48	48	48	
35	48	48	48	48	08	48	00	48	08	48	48	48	48
31	48	48	48	48	48	48	48	48	48	48	48	48	48
27	48	48	36	48	00	48	48	48	00	48	36	48	48
23	48	48	48	48	48	48	48	48	48	48	48	48	48
19	48	48	48	48	08	48	00	48	08	48	48	48	48
15		48	48	48	48	48	48	48	48	48	48	48	
11		48	48	48	48	48	36	48	48	48	48	48	
07			48	48	48	48	48	48	48	48	48		
03					48	48	48	48	48				
	02	06	10	14	18	22	26	30	34	38	42	46	50

Prepared by: *Bryan Wester* Date Today
Reactor Engineer

Verified by: *John Miller* Date Today
Reactor Engineer or SRO

Approved by: *Jake Beamer* Date Today
Unit CRS

ATTACHMENT 5
Page 2 of 4
Control Rod Movement

Individual Rod Movement Instructions

Sheet 1 of 1

SRO Initials: JB

Control Rod	Correct Rod Selected/Verified (CV) NOTE 3			Control Rod Position To	Licensed Operator	Overtravel Check NOTE 1	Full Out Position Check NOTE 2	Second Licensed Operator	Comments
22-03	/	/	/	46					
22-03	/	/	/	48					
-	/	/	/						
-	/	/	/						
-	/	/	/						
-	/	/	/						
-	/	/	/						
-	/	/	/						
-	/	/	/						
-	/	/	/						
-	/	/	/						
-	/	/	/						

Control Rod Movement

NOTE 1: **WHEN** a control rod is withdrawn to the Full Out position, either **MAINTAIN** the continuous withdrawal signal for at least 3 to 5 seconds **OR APPLY** a separate notch withdrawal signal, **AND PERFORM** the following rod coupling integrity check:

- **CONFIRM** *ROD OVER TRAVEL (A-05 4-2)* annunciator does **NOT** alarm. (SR 3.1.3.4)
- **CONFIRM** rod full out light is not lost.
- **CONFIRM** rod position indication on the four-rod display indicates position 48.
- **CONFIRM** *ROD DRIFT (A-05 3-2)* annunciator does **NOT** alarm.

NOTE 2: **VERIFY** the rod reed switch position indicator corresponds to the control rod position indicated by the Full Out reed switch.

NOTE 3: Concurrent Verification (CV) of rod selection required prior to rod movement.
Additional (CV) signoffs for subsequent rod selection following a deselect.

Control Rod Movement

Other Instructions: Performing movement of Control Rod 22-03 to determine RWM substitute position.

Date/Time Completed ______________

Performed By (Print)	Initials

Reviewed By: ______________________ Unit CRS

TASK CONDITIONS:

1. Control rod 22-03 has just been positioned at position 48.
2. Prerequisites listed in section 5.0 of 2OP-07 are met.
3. The on duty Reactor Engineer has been notified.
4. The Unit CRS has reviewed Technical Specifications for applicability and has given permission to perform this procedure.

INITIATING CUE:

You are directed to enter a substitute value for Control Rod 22-03 into the RWM, and inform the CRS when complete.

6.3.11 Determination Of The RWM Substitute Position

Control Rod ___________________

1. **Record** Control Rod Number above. .... ________

2. **Confirm** the following initial conditions are met:

   - All applicable prerequisites in Section 5.0 are met. .... ________
   - Attachment 1 has been reviewed. .... ________
   - RWM requires a position to be substituted. .... ________
   - Unit CRS has referenced Technical Specification SR 3.1.3.1. .... ________ CRS

3. **IF** control rod movement is **NOT** documented by an approved procedure
   THEN obtain copy of 0GP-11, Second Operator Rod Sequence Checkoff Sheets to record rod movements. .... ________

4. **IF AT ANY TIME** a control rod position must be substituted in the RWM or a control rod bypassed in the RWM,
   THEN reference Technical Specification 3.3.2.1. .... ________ CRS

5. **Obtain** permission from the Unit CRS to perform this procedure section. .... ________ CRS

NOTE

- 0OI-53, Rod Worth Minimizer, may be of assistance in performance of this procedure. .... ☐
- The RWM inferred position is **NOT** valid for conditions of three or more failed reed switch positions which include two failed even positions (e.g, RWM inferred position is **NOT** valid for failed positions "06", "07", "08"). If below the RWM LPSP, then a one notch withdrawal/insert error may be created to determine a valid inferred position. If below the RWM LPSP, and at the Insert/Withdrawal limit, then a one notch withdrawal/insert error may be created to determine a valid inferred position. .... ☐

6. **IF** a valid rod position has **NOT** been determined by appropriate methods (as identified in Section 6.1.1),
   THEN determine a valid RWM inferred position by performing the following: .... ________

6.3.11 Determination Of The RWM Substitute Position (continued)

a. **IF** less than or equal to LPSP **AND** at a Withdraw Limit, **THEN bypass** the affected control rod using Section 6.3.14. ......________

BEGIN R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

b. Using Concurrent Verification, **Insert** or **withdraw** control rod one additional notch to an operable control rod reed switch position indicator ....________

c. **Record** the OPERABLE control rod reed switch position below:...________

____

Operable Control Rod Reed Switch Position

d. **Restore** control rod to the position of the failed control rod reed switch position indicator. ....________

e. **IF** control rod was bypassed in the RWM, **THEN unbypass** affected control rod using Section 6.3.15........________

f. **Ensure** inferred position offered by the RWM deviates by one notch and in the correct direction from the position determined in Section 6.3.11 Step 6.c. ....________

g. **Record** the inferred rod position below:...................................________

____

Inferred Position

h. **IF** necessary to ensure a valid inferred position from the RWM, **THEN repeat** Section 6.3.11 Step 6.a through Section 6.3.11 Step 6.f. ....________

6.3.11 Determination Of The RWM Substitute Position (continued)

7. **IF** a valid inferred position from the RWM **OR** valid rod position determined by appropriate methods (as identified in Section 6.1.1) has been obtained,
   THEN perform the following: .... ________

   a. **Substitute** the valid rod position into RWM. .... ________

   b. **Obtain** concurrence of Reactor Engineer and **substitute** affected control rod position in PPC in accordance with 0OP-55, Plant Process and ERFIS Computer Systems Operating Procedure .... ________

   c. **Execute** a CORE MON and **confirm** the correct control rod position substitution was made. .... ________

   d. **Record** the following in the Operator's log:

      - Control rod number.. ________
      - Failed position switch .... ________
      - Substituted position .... ________

   e. **Continue** rod motion as directed by the Unit CRS..................... ________

8. **IF** a valid rod position can **NOT** be determined **AND** reactor power is less than or equal to LPSP,
   THEN perform the following: .... ________

   a. **Refer to** Technical Specification Section 3.1.3. .... ________
   CRS

   b. **Bypass** affected control rod in the RWM using Section 6.3.14. .... ________

   c. **Record** control rod bypassed in the Operator's log. .... ________

   d. Fully **insert** affected control rod until reactor power is greater than LPSP. .... ________

6.3.11 Determination Of The RWM Substitute Position (continued)

9. **IF** a valid control rod position can **NOT** be determined **AND** reactor power is greater than LPSP,
THEN **position** control rod to a valid control rod position indication in accordance with the Reactor Engineer. .... _________

END R.M. LEVEL R2/R3 REACTIVITY EVOLUTION

10. **WHEN** affected control rod is moved to a position with an OPERABLE reed switch position indicator **OR** a valid rod position can be determined,
THEN **perform** the following: .... _________

    a. **Obtain** concurrence of Reactor Engineer and **remove** the PPC control rod position substituted in Section 6.3.11 Step 7.b, in accordance with 0OP-55, Plant Process and ERFIS Computer Systems Operating Procedure. .... _________

    b. **IF** the control rod was bypassed in Section 6.3.11 Step 8.a, **THEN** **unbypass** affected control rod using Section 6.3.15. .... _________

    c. **Record** the control rod returned to normal in the Operator's log. .... _________

Date/Time Completed _________

Performed By (Print)	Initials

Reviewed By _________________________

Unit CRS/SRO

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

SIM JPM H - 2016 NRC INITIAL EXAM – RO

LESSON TITLE: **Test the Main Steam Isolation Valves**

LESSON NUMBER: **LOT-SIM-JP-025-A04**

REVISION NO: **0**

Lou Sosler 9/11/2015
PREPARER / DATE

John Biggs 9/15/2015
TECHNICAL REVIEWER / DATE

Brian Moschet 9/11/2015

Derek Pickett 9/10/2015
VALIDATOR / DATE

Jerry Pierce 9/24/2015
LINE SUPERVISOR / DATE

Jim Barry 9/25/2015
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

239201B201, Test Main Steam Isolation Valves per 0PT-40.2.7

K/A REFERENCE AND IMPORTANCE RATING:

239001 A4.01 4.2/4.0
Ability to manually operate and/or monitor the MSIVs in the Control Room

REFERENCES:

0PT-40.2.7, Testing of Main Steam Line Isolation Valves After Maintenance
0PT-40.2.8, Main Steam Isolation Valve Closure Test

TOOLS AND EQUIPMENT:

Stop Watch

SAFETY FUNCTION (from NUREG 1123):

3 – Pressure Control

SIMULATOR SETUP

Initial Conditions: Reactor power $\leq$50 RTP%

Place Feedwater Control Mode Select switch in 1-ELEM per 2OP-32.

SAFETY CONSIDERATIONS:

None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 2.
4. Critical Step Basis
    1. Prevents Task Completion
    2. May Result in Equipment Damage
    3. Affects Public Health and Safety
    4. Could Result in Personal Injury
5. **Provide copy of 0PT-40.2.7, Acceptance Criteria, Prerequisites, Section 6.2, and Attachment 2, Post Maintenance Testing B21-F022A (Inboard MSIV A Vlv)**

Read the following to the JPM performer.

TASK CONDITIONS:

1. Unit Two startup is in progress following a forced outage to repair MSIV 2B21-F022A, Inboard MSIV A Valve.

2. Conditions are such that steam flow can be stopped in the main steam line of the MSIVs being tested.

3. It is not required to stop steam flow in MSL A to perform the slow closure test of B21-F022A, Inboard MSIV A Valve.

4. No other tests or maintenance activities are in progress that could provide a half scram signal to the RPS logic.

5. Another operator has placed Feedwater Control Mode Select switch in 1-ELEM per 2OP-32, Condensate and Feedwater System Operating Procedure.

INITIATING CUE:

You are directed by the Unit CRS to perform 0PT-40.2.7, Testing of Main Steam Isolation Valve after Maintenance, for MSIV 2B21-F022A, Inboard MSIV A Valve ONLY and inform the CRS if the stroke time meets the acceptance criteria.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT/UNSAT

TIME START: ___________

NOTE: The examinee should be provided a copy of 0PT-40.2.7, Testing of Main Steam Isolation Valve after Maintenance, and given time to review and pre-mark appropriate sections.

PROMPT If asked, a Reactivity Management Team is in place for this test.

Step 2 – Confirm Reactor power is less than 55% RTP

Confirmed power less than 55% RTP.

SAT/UNSAT

Step 2a – Confirm conditions are such that steam flow can be stopped in the main steam line of the MSIV being tested

Confirms steam flow can be stopped in the A Main Steam Line.

SAT/UNSAT

PROMPT If asked, No other tests or maintenance activities are in progress that could provide a half scram signal to the RPS logic.

Step 3 – Confirm all MSIVs are open.

Confirmed all MSIVs are open.

SAT/UNSAT

Step 4 – Confirm Reactor Recirculation system is **NOT** in single loop operation (SLO)

Confirmed Reactor Recirculation system not in single loop.

SAT/UNSAT

NOTE: Have stop watch ready to give to Examinee.

Step 5 – Obtain a stopwatch and record calibration information.
Stop watch obtained and calibration information recorded.

SAT/UNSAT

PROMPT If asked, As the CRS grant permission to perform the test.

Step 5a – Ensures all prerequisites are met.
Verifies all steps in Section 5.0 are met.

SAT/UNSAT

Step 5b – Ensures Feedwater Control Mode Select switch, in 1-ELEM per 2OP-32, Condensate and Feedwater System Operating Procedure.
Verifies Feedwater Control Mode Select control switch is in 1 ELEM.

SAT/UNSAT

NOTE: IF AT ANY TIME while performing this test in MODE 1, annunciator A-05, 4-6, Main Steam Isol Vlv Not Full Open, is received, THEN suspend this test and determine its cause.

Step 6 – Ensure the following annunciators are clear:

- A-05, 4-6, Main Steam Isol Vlv Not Full Open
- A-05, 1-7, Reactor Auto Scram Sys A
- A-05, 2-7, Reactor Auto Scram Sys B

Annunciators confirmed to be clear.

SAT/UNSAT

NOTE: When this test is performed in MODE 1, reactor pressure, power level, and steam flow are monitored while closing the MSIVs. Any deviation from expected plant response is cause for suspension of this test and notification of the Unit CRS prior to proceeding.

PROMPT It is NOT required to stop steam flow in Main Steam Line A.

NOTE: Performer should NA step 6.2.2.

PROMPT It IS required to perform slow closure (spring closure) test of B21-F022A.

Step 7 - **Depress** and **hold** B21-F022A (Inboard MSIV A Test) pushbutton until the valve goes CLOSED, approximately 45-60 seconds.

B21-F022A (Inboard MSIV A Test) pushbutton depressed and held until the valve is CLOSED, green light on, red light off.

****CRITICAL STEP** SAT/UNSAT**

Step 8 - **Release** B21-F022A (Inboard MSIV A Test) pushbutton and **confirm** the valve goes OPEN

Pushbutton for B21-F022A released and valve open confirmed.

****CRITICAL STEP** SAT/UNSAT**

PROMPT If asked, stroke time testing is required.

NOTE: Operation with both MSIVs closed in a main steam line is minimized to reduce the severity of differential pressure transients when reopening the Outboard MSIV.

Step 9 - **Perform** stroke time test as follows:

a. **Ensure** B21-F022A (Inboard MSIV A Vlv) OPEN.

B21-F022A verified open.

SAT/UNSAT

b. **Close** B21-F022A (Inboard MSIV A Vlv) utilizing the pistol grip switch.

B21-F022A pistol grip switch taken to close.

****CRITICAL STEP** SAT/UNSAT**

c. **Record** stroke time:

Stroke time recorded.

SAT/UNSAT

d. **Enter** the measured stroke time from Section 6.2 Step 4.c and **calculate** the corrected stroke time (Stroke Time from Section 6.2, Step 4.c X 1.1 = Corrected Stroke Time)

Corrected stroke time calculated

****CRITICAL STEP** SAT/UNSAT**

e. **Record** corrected stroke time on Attachment 1 or Attachment 2
Corrected Stroke Time recorded on Attachment 2

SAT/UNSAT

NOTE: Step 6.2.5 is NA, as the B21-F028A was not closed previously.

PROMPT If asked, it is required by plant conditions to open B21-F022A.

Step 10 – **IF** required by plant conditions, **THEN open** B21-F022A (Inboard MSIV A Vlv).
B21-F022A pistol grip switch taken to open.

SAT/UNSAT

NOTE: Step 6.2.7 is N/A

NOTE: Annunciator A-7, 4-2, FW Sys Ctrl Trbl, may alarm.

Step 11 – **Informs** CRS that the stroke time for the Inboard MSIV A is SAT
Determines from Attachment 2 that the stroke time for A MSIV is within the Acceptance Criteria.

SAT/UNSAT

PROMPT Inform Examinee that another operator will complete the Restoration section of the PT.

TERMINATING CUE: When the 2B21-F022A, Inboard MSIV A Valve, has been re-opened after testing and the CRS is notified that the stroke time meets the Acceptance Criteria of the PT this JPM is complete.

TIME COMPLETED: ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2-6	Not Critical	Verification of initial conditions and pre-requisites.
7-8	Critical	Required actions to complete the test.
9a	Not Critical	Verification step.
9b	Critical	Action required to complete the test.
9c	Not Critical	Recording time not critical to test completion.
9d	Critical	Calculation of Corrected Stroke Time required to complete task.
9e	Not Critical	Recording required information.
10	Not Critical	Re-opening valve not required to obtain results.

REVISION SUMMARY

0	New JPM.

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	X	Admin	____
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ______________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: ____________

TASK CONDITIONS:

1. Unit Two startup is in progress following a forced outage to repair MSIV 2B21-F022A, Inboard MSIV A Valve.

2. Conditions are such that steam flow can be stopped in the main steam line of the MSIVs being tested.

3. It is not required to stop steam flow in MSL A to perform the slow closure test of B21-F022A, Inboard MSIV A Valve.

4. No other tests or maintenance activities are in progress that could provide a half scram signal to the RPS logic.

5. Another operator has placed Feedwater Control Mode Select switch in 1-ELEM per 2OP-32, Condensate and Feedwater System Operating Procedure.

INITIATING CUE:

You are directed by the Unit CRS to perform 0PT-40.2.7, Testing of Main Steam Isolation Valve after Maintenance, for MSIV 2B21-F022A, Inboard MSIV A Valve ONLY and inform the CRS if the stroke time meets the acceptance criteria.

Continuous Use

BRUNSWICK UNIT 0

SURVEILLANCE TEST PROCEDURE

0PT-40.2.7

TESTING OF MAIN STEAM ISOLATION VALVES AFTER MAINTENANCE

REVISION 16

Special Considerations:

This is a Reactivity Management Procedure.

REVISION SUMMARY
PRR 635473 **DESCRIPTION**
Revision 16 - Editorial Revision to capitalize component positions, make consistent use of emphasis techniques, make component descriptions title case, update placekeeping aids, revise Steps in Section 6.5 for Acceptance Criteria review and update Attachment 4 to latest version of the Certification/Review form. Revised by Jim McCrary

TABLE OF CONTENTS

1.0 PURPOSE

This test demonstrates the OPERABILITY of the main steam isolation valves, MSIVs, after maintenance and provides direction for the slow closure test of MSIVs.

2.0 SCOPE

1. This test demonstrates each MSIVs ability to full stroke within the stroke times specified in Unit 1(Unit 2) Technical Specifications SR 3.6.1.3.5. This satisfies the IST requirement in Technical Specification 5.5.6.

2. This test checks the MSIV Slow Closure function described in UFSAR Sections 5.4.5 and 7.3.1.1.5.

3. This test does **NOT** provide instructions for stroke adjustments subsequent to testing.

3.0 PRECAUTIONS AND LIMITATIONS

1. When isolating and unisolating a steam line in MODE 1 or 2 a small pressure change may occur causing a reactivity change. This reactivity change is classified as a Reactivity Manipulation (R2) per OPS-NGGC-1306, Reactivity Management Program.

2. If this test is being performed in MODE 1, the RPS System will receive a partial trip signal that will **NOT** be annunciated as long as the remaining MSIVs are in the open position.

3. Annunciator A-05, 4-6, Main Steam Isol Vlv Not Full Open, may alarm when a main steam line is isolated. This annunciator is received only when two or more MSIVs are closed.

4. Operation with both MSIVs closed in a main steam line is minimized to reduce the severity of differential pressure transients when reopening the Outboard MSIV. The section of pipe between the inboard and outboard MSIV will depressurize as it cools down or if any steam leaks are present (such as stem packing leak).

3.0 PRECAUTIONS AND LIMITATIONS (continued)

5. An administrative band of 3.6 seconds to 4.4 seconds is applied when in MODE 2 or 3 due to temperature affects on stroke time. This is an administrative limit, and is **NOT** controlled by the IST program. More detail concerning these administrative limits is available in Section 8.7 Miscellaneous Document 3, EC# 86807, Evaluation of MSIV Stroke Time Criteria. If the corrected stroke time is satisfactory, but outside the Administrative range, it is to be adjusted to within the Administrative range per the applicable CHANNEL CALIBRATION.

4.0 ACCEPTANCE CRITERIA

This test may be considered satisfactory with the successful completion of this procedure.

NOTE

This test demonstrates the slow closure design base function of the MSIVs described in UFSAR Sections 5.4.5 and 7.3.1.1.5. This function requires MSIVs to close on spring pressure alone. Slow closure is **NOT** a safety function. .... ☐

1. Slow Closure Test

   a. When an MSIV is given a close signal from the Control Room test pushbutton, the valve goes to the CLOSED position.

NOTE

- The Valve Stroke time test satisfies Unit 1(Unit 2) Technical Specifications SR 3.6.1.3.5 and partially satisfies the IST requirement in Technical Specification 5.5.6. .... ☐
- Stroke time is measured from the time the control switch is repositioned to the time the valve is fully stroked by light indication.. .... ☐
- For MSIVs, the measured stroke times are multiplied by a correction factor of 1.1 to compensate for the position settings of the indicating light sensors of 10% and 100% open. .... ☐

2. Valve Stroke Time

   a. Corrected stroke times are within the Limiting range as specified by the minimum and maximum stroke times shown on Attachment 1, Unit 1 Nuclear Steam Supply System Valves Data or Attachment 2, Unit 2 Nuclear Steam Supply System Valves Data

4.0 ACCEPTANCE CRITERIA (continued)

b. For tests where the corrected stroke time of the valve is less than the minimum or greater than the maximum Limiting stroke time or the valve disc or stem fail to exhibit the required change of position, the valve shall immediately be declared INOPERABLE.

NOTE

This test partially satisfies the IST requirement in Technical Specification 5.5.6. .... ☐

3. Valve Fail-Safe Testing

a. The fail-safe test is considered satisfactory when the control switch is placed in the CLOSED position for fail-closed valves or OPEN position for fail-open valves, and the valve changes position in response to control switch movement.

5.0 PREREQUISITES

1. **Confirm** Reactor power is less than 55% RTP.......................................______

2. **Confirm** conditions are such that steam flow can be stopped in the main steam line of the MSIV being tested or **NO** steam flow exists..........______

3. IF unit is in MODE 1,
THEN confirm the following:

   - **NO** other tests or maintenance activities are in progress that could provide a half scram signal to the RPS logic. ....______

   - All main steam isolation valves are OPEN. ....______

4. **Confirm** the Reactor Recirculation system is **NOT** in single loop operation (SLO). ....______

5. **Obtain** a stopwatch and **record** information:..______

TEST EQUIPMENT			
Item	**ID No.**	**Cal Date**	**Cal Due Date**
Stopwatch			

6.0 INSTRUCTIONS

6.1 General

1. **Request** permission from the Unit CRS to perform this test.....................______

2. **Ensure** all prerequisites in Section 5.0 are met.......................................______

3. **Ensure** Feedwater Control Mode Select switch, in 1-ELEM per 1OP-32 (2OP-32) Condensate and Feedwater System Operating Procedure. ....______

4. **IF AT ANY TIME** while performing this test in MODE 1, annunciator A-05, 4-6, Main Steam Isol Vlv Not Full Open, is received,
THEN suspend this test and **determine** its cause.................................______

NOTE

When isolating and unisolating a steam line in MODE 1 or 2 a small pressure change may occur causing a reactivity change. This reactivity change is classified as a Reactivity Manipulation (R2) per OPS-NGGC-1306, Reactivity Management Program. .... ☐

6.2 Post Maintenance Testing B21-F022A (Inboard MSIV A Vlv)

1. **IF** unit is in MODE 1,
   THEN **ensure** the following annunciators CLEAR:
   - A-05, 4-6, Main Steam Isol Vlv Not Full Open .... _______
   - A-05, 1-7, Reactor Auto Scram Sys A .... _______
   - A-05, 2-7, Reactor Auto Scram Sys B .... _______

CAUTION

When this test is performed in MODE 1, reactor pressure, power level, and steam flow are monitored while closing the MSIVs. Any deviation from expected plant response is cause for suspension of this test and notification of the Unit CRS prior to proceeding. .... ☐

BEGIN R.M. LEVEL R2 REACTIVITY EVOLUTION

2. **IF** it is required to stop steam flow in Main Steam Line A,
   THEN **perform** the following:

   a. **Depress** and **hold** B21-F028A (Outboard MSIV A Test) pushbutton until the valve is CLOSED. .... _______

   b. **Place** pistol grip switch for B21-F028A (Outboard MSIV A Vlv) in CLOSE. .... _______

3. **IF** performing slow closure (spring closure) test of B21-F022A (Inboard MSIV A Vlv),
   THEN **perform** the following:

   a. **Depress** and **hold** B21-F022A (Inboard MSIV A Test) pushbutton until the valve goes CLOSED, approximately 45-60 seconds. .... _______

   b. **Release** B21-F022A (Inboard MSIV A Test) pushbutton and **confirm** the valve goes OPEN. .... _______

6.2 Post Maintenance Testing B21-F022A (Inboard MSIV A Vlv) (continued)

CAUTION

Operation with both MSIVs closed in a main steam line is minimized to reduce the severity of differential pressure transients when reopening the Outboard MSIV.................. ☐

4. **Perform** stroke time test as follows:

   a. **Ensure** B21-F022A (Inboard MSIV A Vlv) OPEN. ...._______

   b. **Close** B21-F022A (Inboard MSIV A Vlv) utilizing the pistol grip switch. ...._______

   c. **Record** stroke time: ...._______

   Stroke Time __________ Seconds

   d. **Enter** the measured stroke time from Section 6.2 Step 4.c and **calculate** the corrected stroke time. ...._______

   _______ IV

   __________ **seconds** **X 1.1 =** __________ **seconds**

   Stroke Time from Section 6.2 Step 4.c — Corrected Stroke Time

   e. **Record** corrected stroke time on Attachment 1 or Attachment 2. ..._______

5. **IF** B21-F028A (Outboard MSIV A Vlv) pistol grip switch was placed in CLOSE in Section 6.2 Step 2,
   THEN place pistol grip switch in OPEN and **confirm** the valve goes OPEN.._______

6. **IF** required by plant conditions,
   THEN open B21-F022A (Inboard MSIV A Vlv).._______

END R.M. LEVEL R2 REACTIVITY EVOLUTION

6.2 Post Maintenance Testing B21-F022A (Inboard MSIV A Vlv) (continued)

7. **IF** all the following conditions are met:

   - Corrected stroke time is within the Limiting range
   - Corrected stroke time is outside the Administrative range
   - The unit is in MODE 2 or 3 with the Drywell/MSIV Pit **NOT** accessible,

   THEN generate an CR to adjust the valve stroke time to within the Administrative range during the next outage.._______

ATTACHMENT 2
Page 1 of 1

Unit 2 Nuclear Steam Supply System Valves Data

VALVE NUMBER	STROKE DIRECTION	PROC SECTION	STEP #	REMOTE POSITION INDICATION (INITIALS)		STROKE TIME TEST (SEC)	ADMINISTRATIVE RANGE (MODE 2 & 3 ONLY)		ACCEPTANCE CRITERIA LIMITING VALUE		REF. STROKE TIME	FAIL-SAFE TEST (INITIALS)	VALVE SAT/ UNSAT
				STEM	IND. LIGHTS		MINIMUM (≥)	MAXIMUM (≤)	MINIMUM (≥)	MAXIMUM (≤)			
INBOARD													
2-B21-F022A	OPEN	Section 6.2	Step 4										
2-B21-F022A	CLOSED						3.6 *	4.4 *	3.0	5.0	4.0		
2-B21-F022B	OPEN	Section 6.4	Step 4										
2-B21-F022B	CLOSED						3.6 *	4.4 *	3.0	5.0	4.0		
2-B21-F022C	OPEN	Section 6.6	Step 4										
2-B21-F022C	CLOSED						3.6 *	4.4 *	3.0	5.0	4.0		
2-B21-F022D	OPEN	Section 6.8	Step 4										
2-B21-F022D	CLOSED						3.6 *	4.4 *	3.0	5.0	4.0		
OUTBOARD													
2-B21-F028A	OPEN	Section 6.3	Step 4										
2-B21-F028A	CLOSED						3.6 *	4.4 *	3.0	5.0	4.0		
2-B21-F028B	OPEN	Section 6.5	Step 4										
2-B21-F028B	CLOSED						3.6 *	4.4 *	3.0	5.0	4.0		
2-B21-F028C	OPEN	Section 6.7	Step 4										
2-B21-F028C	CLOSED						3.6 *	4.4 *	3.0	5.0	4.0		
2-B21-F028D	OPEN	Section 6.9	Step 4										
2-B21-F028D	CLOSED						3.6 *	4.4 *	3.0	5.0	4.0		

* If corrected stroke time is within the Limiting range and outside the Administrative range and the unit is in MODE 2 or 3, an CR shall be generated to schedule and track performance of 2MST-RPS22(A-H)R to be completed during the next outage to ensure the valve stroke time is adjusted to within the Administrative range.

Performed by (Signature) ______________________________ Date______________

Performed by (Signature) ______________________________ Date______________

Reviewed, IST Group (Signature) _________________________ Date ______________

DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

IP JPM I - 2016 NRC INITIAL EXAM - RO/ISRO

LESSON TITLE: **ALTERNATE COOLANT INJECTION LEP-01 – HEATER DRAIN PUMPS**

LESSON NUMBER: **AOT-OJT-JP-300-J13**

REVISION NO: **6**

Dan Hulgin 8/18/16
PREPARER / DATE

Bob Bolin 9/07/16
TECHNICAL REVIEWER / DATE

Hunter Morris 9/07/16
VALIDATOR / DATE

Craig Oliver 09/22/2016
LINE SUPERVISOR / DATE

9-28-2016
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

200072B504
Perform Alternate Coolant Injection With Heater Drain Pumps Per LEP-01.

K/A REFERENCE AND IMPORTANCE RATING:

295031 AA1.08 3.8/3.9
Ability to operate alternate injection system systems as they apply to Reactor Water Level Low.

REFERENCES:

0EOP-01-LEP-01, ALTERNATE COOLANT INJECTION

TOOLS AND EQUIPMENT:

CR104P key for Unit Trip Load Shed Selector switch.

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

2 Inventory Control

SAFETY CONSIDERATIONS:

1. Notify SM/CRS of JPM performance prior to commencing In-plant JPM.
2. Determine actual radiological conditions and potentially contaminated areas to achieve ALARA.
3. Ensure all electrical safety requirements are observed.
4. Review Work Practices section prior to conduct of the JPM.
5. DO NOT OPERATE any plant equipment during performance of this JPM.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee. 0EOP-01-LEP-01, will be provided to the examinee when asked for.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM will be performed on Unit 1or Unit 2.
4. Consider starting this JPM in the Control Room due to the need to obtain a CR104P key for Unit Trip Load Shed Selector Switch as well as for obtaining permission to enter the 4 KV Switchgear area in the Turbine Building
5. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. A low Reactor Water level condition exists on Unit ____.
2. The CRS is executing the Reactor Vessel Control Procedure (EOP-01-RVCP)
3. RVCP directs use of Alternate Coolant Injection per EOP-01-LEP-01.
4. RPV Pressure is 450 psig.
5. The main condenser is under vacuum.

INITIATING CUE:

You are directed to perform the Auxiliary Operator actions for Alternate Coolant Injection, Heater Drain Pump Injection per EOP-01-LEP-01, Section 2.2, and inform the Control Room when all required Auxiliary Operator actions are complete.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT / UNSAT

TIME START ___________

PROMPT:	Inform examinee that LEP-01, Section 2.2 Steps 1 through 3 have been completed.

Step 2 – **Maintain** level in the heater drain tank: Place feedwater heater level controllers to MAN and decrease the air signal to 0% to open the associated feedwater heater level control valves: HD-LC-75 (Feedwater Heater 4A Level Controller).

Places HD-LC-75 Auto/Manual Selector to MAN .
Adjust Controller output to 0% using Manual Control Unit Thumbwheel.
HD-LC-75 in MAN with air signal at 0% (**critical***).*

**** CRITICAL STEP ** SAT / UNSAT**

Step 3 – **Maintain** level in the heater drain tank: Place feedwater heater level controllers to MAN and decrease the air signal to 0% to open the associated feedwater heater level control valves: HD-LC-83 (Feedwater Heater 5A Level Controller).

Places HD-LC-83 Auto/Manual Selector to MAN .
Adjust Controller output to 0% using Manual Control Unit Thumbwheel.
HD-LC-83 in MAN with air signal at 0% (**critical***).*

**** CRITICAL STEP ** SAT / UNSAT**

Step 4 – **Maintain** level in the heater drain tank: Place feedwater heater level controllers to MAN and decrease the air signal to 0% to open the associated feedwater heater level control valves: HD-LC-79 (Feedwater Heater 4B Level Controller).

Places HD-LC-79 Auto/Manual Selector to MAN .
Adjust Controller output to 0% using Manual Control Unit Thumbwheel.
HD-LC-79 in MAN with air signal at 0% (**critical***).*

**** CRITICAL STEP ** SAT / UNSAT**

Step 5 – **Maintain** level in the heater drain tank: Place feedwater heater level controllers to MAN and decrease the air signal to 0% to open the associated feedwater heater level control valves: HD-LC-87 (Feedwater Heater 5B Level Controller).

Places HD-LC-87 Auto/Manual Selector to MAN .
Adjust Controller output to 0% using Manual Control Unit Thumbwheel.
HD-LC-87 in MAN with air signal at 0% (**critical***).*

**** CRITICAL STEP **   SAT / UNSAT**

Step 6 - **Ensure** HD-LC-91 (Heater Drain Deaerator Level Controller) in AUTO.

Verifies HD-LC-91 in AUTO (Auto/Manual Selector in AUTO).
HD-LC-91 in AUTO.

SAT / UNSAT

Step 7 - **Unit 2 Only**: **Ensure** HD-LC-97 (Heater Drain Deaerator Level Controller) in AUTO.

Verifies HD-LC-97 in AUTO (Controller Mode (Manual or Auto) –A displayed on the controller).
HD-LC-97 in AUTO

SAT / UNSAT

PROMPT:	When informed that AO actions for step 4 are complete, inform examinee that LEP-01, Section 2 Step 5 through 6 have been completed. Inform Examinee that Heater Drain Pump 1(2)A is to be started for alternate coolant injection.

NOTE:	A CR104P key for Unit Trip Load Shed Selector Switch is located in the RO Desk locked drawer. A key can also be found in the Control room or WCC key lockers. Heater Drain Pump 1(2)A Unit Trip Load Selector Switch is on BOP Bus 1(2)D. Permission is required to enter the 4 KV Switchgear area in the Turbine Building. STEP 8a is to be performed if this JPM is performed on Unit 1. STEP 8b is to be performed if this JPM is performed on Unit 2.

Step 8a - **Place** Unit Trip Load Shed Selector Switch for heater drain pump to be started in DISABLED: At Bus 1D, Row H1, Compt AD8 (Htr Drain Pump 1A).

Heater Drain Pump 1A Unit Trip Load Selector Switch is placed in DISABLED.

**** CRITICAL STEP **   SAT / UNSAT**

Step 8b - **Place** Unit Trip Load Shed Selector Switch for heater drain pump to be started in DISABLED: At Bus 2D, Row I1, Compt AD8 (Htr Drain Pump 2A).

Heater Drain Pump 2A Unit Trip Load Selector Switch is placed in DISABLED.

**** CRITICAL STEP **   SAT / UNSAT**

Step 10 - **Inform** Control Room AO Actions for Alternate Coolant Injection using Heater Drain Pump Injection are complete.

Control Room informed AO actions per LEP-01, Section 2.2 are complete.

SAT / UNSAT

TERMINATING CUE: When AO Actions for Alternate Coolant Injection using Heater Drain Pump Injection are complete, this JPM is complete.

TIME COMPLETED ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Critical	Necessary for HDP alternate coolant injection
3	Critical	Necessary for HDP alternate coolant injection
4	Critical	Necessary for HDP alternate coolant injection
5	Critical	Necessary for HDP alternate coolant injection
6	Non Critical	Verification
7	Non Critical	Verification
8a	Critical	Necessary for HDP alternate coolant injection (UNIT 1)
8b	Critical	Necessary for HDP alternate coolant injection (UNIT 2)
9	Critical	Necessary for HDP alternate coolant injection
10	Critical	Communication

REVISION SUMMARY

6	New JPM template Critical Step Delineation table added Renumbered steps Corrected procedure section Enhanced standards
5	Added basis for critical steps Minor format changes to cover/signature page Change SCO title to CRS

Validation Time: 10 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	_____	Unit:	1/ 2
Setting:	In-Plant	X	Simulator	_____	Admin	___
Time Critical:	Yes	_____	No	X	Time Limit	N/A
Alternate Path:	Yes	_____	No	X		

EVALUATION

Performer: ____

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ____

____

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☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

TASK CONDITIONS:

1. A low Reactor Water level condition exists on Unit _____.
2. The CRS is executing the Reactor Vessel Control Procedure (EOP-01-RVCP)
3. RVCP directs use of Alternate Coolant Injection per EOP-01-LEP-01.
4. RPV Pressure is 450 psig.
5. The main condenser is under vacuum.

INITIATING CUE:

You are directed to perform the Auxiliary Operator actions for Alternate Coolant Injection, Heater Drain Pump Injection per EOP-01-LEP-01, Section 2.2, and inform the Control Room when all required Auxiliary Operator actions are complete.

BRUNSWICK UNIT 0

EMERGENCY OPERATING PROCEDURE

0EOP-01-LEP-01

ALTERNATE COOLANT INJECTION

REVISION 34

Special Considerations:

EOP-Protected procedure – Any revision to this procedure should be reviewed by an EOP writer.

REVISION SUMMARY
PRR 756734 **DESCRIPTION**
Revision 34 adds use of the torus suction to Section 2.5, RCIC Injection Using Manual Valve Operations and revises Section 2.10, RCIC Restoration to reflect same. Revised by Mike English.

TABLE OF CONTENTS

TABLE OF CONTENTS
(continued)

1.0 ENTRY CONDITIONS

- As directed by Emergency Operating Procedures (EOPs)
- As directed by Severe Accident Management Guidelines (SAMGs)
- As directed by Security Events, 0AOP-40.0

2.0 INSTRUCTIONS

2.1 SLC Pump Demineralized/Fire Water Injection

2.1.1 Manpower Required

- 1 Reactor Operator
- 1 Auxiliary Operator

2.1.2 Special Equipment

- RO Desk Locked Drawer
  - ◊ 1 LEP toolbox key (LSV-1) .... ☐
- Reactor Building 80' LEP Toolbox
  - ◊ 1 pipe wrench .... ☐
  - ◊ 1 Demineralized Water to SLC Jumper Hose .... ☐
  - ◊ 1 Fire Hose to SLC Connector .... ☐

2.1.3 SLC Pump Actions

1. IF it has been determined the reactor will remain shutdown under all conditions without boron,
   THEN continue in this procedure. .... ☐
   RO

2. IF directed to inject demineralized water,
   THEN:

   a. **Unlock** and **close** C41-F001 (SLC Storage Tank Outlet Isolation Valve). .... ☐
   AO

2.1.3 SLC Pump Actions (continued)

b. **Remove** cap and **connect** one end of the Demineralized Water to SLC Jumper Hose to the threaded connection at C41-V5000 (SLC Demineralized Water Supply Isolation Valve)........ ☐ AO

c. **Remove** cap and **connect** the other end of the Demineralized Water to SLC Jumper Hose to the threaded connection upstream of C41-F014 (SLC Test Tank Outlet Demineralized Water Supply Isolation Valve)........ ☐ AO

d. **Unlock** and **open** C41-V5000 (SLC Demineralized Water Supply Isolation Valve). .... ☐ AO

e. **Unlock** and **open** C41-F014 (SLC Test Tank Outlet Demineralized Water Supply Isolation Valve)........ ☐ AO

3. **IF** demineralized water **NOT** available **AND** fire protection water available,
THEN:

a. **Unlock** and **close** C41-F001 (SLC Storage Tank Outlet Isolation Valve)........ ☐ AO

NOTE

Fire hose stations are located on north wall of CRD rebuild room and south wall of Reactor Building 80'. .... ☐

b. **Disconnect** the nozzle from the fire hose selected for supplying SLC. .... ☐ AO

2.1.3 SLC Pump Actions (continued)

Figure 1, Fire Hose to SLC Connector

c. **Remove** cap and **connect** one end of the Fire Hose to SLC Connector to the threaded connection upstream of C41-F014 (SLC Test Tank Outlet Demineralized Water Supply Isolation Valve) Figure 1, Fire Hose to SLC Connector. .... ☐ AO

d. **Connect** the other end of the Fire Hose to SLC Connector to the fire hose selected in Section 2.1.3 Step 3.b. .... ☐ AO

e. **Ensure** vent valve on the Fire Hose to SLC Connector CLOSED. .... ☐ AO

f. **Open** the hose reel angle valve for the fire hose selected in Section 2.1.3 Step 3.b. .... ☐ AO

g. **Open** vent valve on the Fire Hose to SLC Connector. .... ☐ AO

h. **WHEN** a stream of water issues from the valve,
THEN close vent valve on the Fire Hose to SLC Connector. .... ☐ AO

2.1.3 SLC Pump Actions (continued)

i. **Unlock** and **open** C41-F014 (SLC Test Tank Outlet Demineralized Water Supply Isolation Valve)........ ☐ AO

4. **Start** SLC Pumps A and B from the RTGB........ ☐ RO

5. **<u>WHEN</u>** SLC pumps **<u>NOT</u>** required for RPV injection, **<u>THEN:</u>**

a. **Stop** SLC Pumps A and B........ ☐ RO

b. **Close** C41-F014 (SLC Test Tank Outlet Demineralized Water Supply Isolation Valve)........ ☐ AO

c. **Exit** this section and **go to** Section 2.6........ ☐ RO

2.2 Heater Drain Pump Injection

2.2.1 Manpower Required

- 1 Reactor Operator
- 1 Auxiliary Operator

2.2.2 Special Equipment

- RO Desk Locked Drawer
  - ◊ 1 CR104P key for Unit Trip Load Shed Selector Switch .... ☐

2.2.3 Heater Drain Pump Actions

1. **Ensure** FW-FV-177 (Feedwater Recirc To Condenser Vlv) CLOSED. .... ☐ RO

2. **Open** FW-V13 (RFP Bypass Vlv). .... ☐ RO

3. **Close:**
   - COD-V49 (RFP A Suction Vlv). .... ☐ RO
   - COD-V50 (RFP B Suction Vlv). .... ☐ RO

4. **Maintain** level in the heater drain tank:

   a. **Place** feedwater heater level controllers to MAN and **decrease** the air signal to 0% to open the associated feedwater heater level control valves:

      (1) HD-LC-75 (Feedwater Heater 4A Level Controller). .... ☐ AO

      (2) HD-LC-83 (Feedwater Heater 5A Level Controller). .... ☐ AO

      (3) HD-LC-79 (Feedwater Heater 4B Level Controller). .... ☐ AO

      (4) HD-LC-87 (Feedwater Heater 5B Level Controller). .... ☐ AO

2.2.3 Heater Drain Pump Actions (continued)

b. **Ensure** HD-LC-91 (Heater Drain Deaerator Level Controller) in AUTO. ☐ AO

c. **Unit 2 Only:** **Ensure** HD-LC-97 (Heater Drain Deaerator Level Controller) in AUTO. ☐ AO

d. **IF** main condenser under vacuum,
THEN continue in this section at Section 2.2.3 Step 5. ☐ RO

e. **IF** Condensate Transfer System available **AND** condenser **NOT** under vacuum,
THEN:

NOTE

RFA-LI-56 (RFB-LI-57) (RFP Cond Level) indicator on XU-2 reading on-scale indicates level in the hotwell is sufficient to drain to the heater drain tank. ☐

(1) **Notify** Radwaste to make up to the hotwell to maintain hotwell level above +16 inches. ☐ RO

(2) **WHEN** hotwell level above +16 inches,
THEN open HD-V57 (Deaerator Fill & Drain Vlv) to drain the hotwell to the heater drain tank. ☐ RO

5. **Ensure** CLOSED:

- FW-V118 (FW Htr 4A Inlet Vlv). ☐ RO
- FW-V119 (FW Htr 4B Inlet Vlv). ☐ RO
- FW-V6 (FW Htr 5A Outlet Vlv). ☐ RO

2.2.3 Heater Drain Pump Actions (continued)

- FW-V8 (FW Htr 5B Outlet Vlv). .... ☐ RO
- FW-V120 (FW Htrs 4 & 5 Byp Vlv). .... ☐ RO

> **NOTE**
>
> Heater drain deaerator level must be greater than or equal to 48 inches to start a heater drain pump. .... ☐

6. **Circle** heater drain pump to be started. .... ☐ RO

   A B C

7. **<u>Unit 1 Only:</u> Place** Unit Trip Load Shed Selector Switch for heater drain pump to be started in DISABLED:

   - At Bus 1D, Row H1, Compt AD8 (Htr Drain Pump 1A). .... ☐ AO
   - At Bus 1C, Row II1, Compt AC3 (Htr Drain Pump 1B). .... ☐ AO
   - At Bus 1D, Row D1, Compt AD4 (Htr Drain Pump 1C). .... ☐ AO

8. **<u>Unit 2 Only:</u> Place** Unit Trip Load Shed Selector Switch for heater drain pump to be started in DISABLED:

   - At Bus 2D, Row I1, Compt AD8 (Htr Drain Pump 2A). .... ☐ AO
   - At Bus 2C, Row MM1, Compt AC3 (Htr Drain Pump 2B). .... ☐ AO
   - At Bus 2D, Row H1, Compt AD7 (Htr Drain Pump 2C). .... ☐ AO

2.2.3 Heater Drain Pump Actions (continued)

9. **Start** selected heater drain pump. .... ☐
RO

10. **Place** FW-LIC-3269 (Startup Level Control Valve) in M (Manual) and **open**. .... ☐
RO

11. **WHEN** heater drain pump injection **NO** longer required, **THEN** **exit** this section and **go to** Section 2.7. .... ☐
RO

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

IP JPM J - 2016 NRC INITIAL EXAM - RO/ISRO/USRO

LESSON TITLE: **REMOTE SHUTDOWN PANEL SRV OPERATION**

LESSON NUMBER: **LOT-OJT-JP-300-J25**

REVISION NO: **0**

Dan Hulgin 8/18/16
PREPARER / DATE

Bob Bolin 9/07/16
TECHNICAL REVIEWER / DATE

Grant Newton
Hunter Morris 9/07/16
VALIDATOR / DATE

Craig Oliver 09/27/2016
LINE SUPERVISOR / DATE

9-28-2016
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

239006B501 - Perform Remote Shutdown Panel SRV Operation per 0EOP-01-LEP-05.

K/A REFERENCE AND IMPORTANCE RATING:

295016 AA1.08 4.0/4.0
Ability to operate and/or monitor Reactor Pressure as it applies to Control Room Abandonment.

REFERENCES:

0EOP-01-LEP-05, REMOTE SHUTDOWN PANEL SRV OPERATION

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

7 Instrumentation

SAFETY CONSIDERATIONS:

1. Notify SM/CRS of JPM performance prior to commencing In-plant JPM.
2. Determine actual radiological conditions and potentially contaminated areas to achieve ALARA.
3. Ensure all electrical safety requirements are observed.
4. Review Work Practices section prior to conduct of the JPM.
5. DO NOT OPERATE any plant equipment during performance of this JPM.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee. 0EOP-01-LEP-05, will be provided to the examinee when asked for.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM will be performed on Unit 1or Unit 2.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. Unit _____ is executing 0EOP-01-EDP.
2. Reactor Pressure is 500 psig
3. SRV operation from the Control Room is not successful.

INITIATING CUE:

You are directed by the Control Room Supervisor to rapidly depressurize the RPV by opening SRVs B, E, and G from the Unit _____ Remote Shutdown Panel (RSDP) IAW 0EOP-01-LEP-05.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT / UNSAT

TIME START ___________

NOTE:	Special Equipment is located in the following areas: **RO Desk Locked Drawer:** 1 LEP toolbox key (LSV-1) **ASSD Toolbox in Control Room:** 1 sound-powered phone with extension cord for RO **Reactor Building 20' LEP Toolbox** 4 T112 keys 1 sound-powered phone with extension cord for RO

Step 2 – **Notify** CRS: RTGB level indicators B21-LI-R604B and C32-PR-R609 (N026B) will be lost.

Notifies the CRS that level indicators B21-LI-R604B and C32-PR-R609 (N026B) will be lost .

SAT / UNSAT

Step 3 – **Notify** CRS: Control of SRV B, E and G from RTGB will be lost.

Notifies the CRS that Control of SRV B, E and G from RTGB will be lost.

SAT / UNSAT

Step 4 - **Establish** communication between RSDP and Control Room.

Establishes communication with the Control Room.

SAT / UNSAT

Step 5 - At the Remote Shutdown Panel: **Ensure** B21-F013E (Manual Relief E Vlv Close/Open) control switch in CLOSE.

Verifies B21-F013E Close/Open *control switch is in CLOSE .*

SAT / UNSAT

Step 6 - At the Remote Shutdown Panel: **Place** B21-F013E (Manual Relief E Vlv Normal/Local) control switch in LOCAL.

Inserts Key and places B21-F013E Normal/Local *control switch in LOCAL. B21-F013E* Normal/Local *control switch in LOCAL* (**critical***).*

**** CRITICAL STEP ** SAT / UNSAT**

Step 7 - At the Remote Shutdown Panel: **Ensure** B21-F013G (Manual Relief E Vlv Close/Open) control switch in CLOSE.

Verifies B21-F013G Close/Open *control switch is in CLOSE .*

SAT / UNSAT

Step 8 - At the Remote Shutdown Panel: **Place** B21-F013G (Manual Relief E Vlv Normal/Local) control switch in LOCAL.

Inserts Key and places B21-F013G Normal/Local *control switch in LOCAL. B21-F013G* Normal/Local *control switch in LOCAL* (**critical***).*

**** CRITICAL STEP ** SAT / UNSAT**

Step 9 - At the Remote Shutdown Panel: **Ensure** B21-F013B (Manual Relief B Vlv Close/Open) control switch in CLOSE.

Verifies B21-F013B Close/Open *control switch is in CLOSE .*

SAT / UNSAT

Step 10 - At the Remote Shutdown Panel: **Place** B21-F013B (Manual Relief E Vlv Normal/Local) control switch in LOCAL.

Inserts Key and places Normal/Local *control switch in LOCAL. B21-F013B* Normal/Local *control switch in LOCAL* (**critical***).*

**** CRITICAL STEP ** SAT / UNSAT**

Step 11 - At the Remote Shutdown Panel: **Place** B21-CS-3345 (Reactor Water Level Normal/Local Switch) in LOCAL to transfer level transmitter B21-LT-N026B output to B21-LI-R604BX.

Inserts Key and places B21-CS-3345 Normal/Local *control switch in LOCAL.*
B21-F013B Normal/Local *control switch in LOCAL.*

SAT / UNSAT

PROMPT:	If asked, CAC-LI-3342 (Supp Pool Level) is -2 ft.
PROMPT:	If asked, *B21-LI-R604BX (Reactor Water Level) is* 150 inches and reading is valid IAW Caution 1.

Step 12 - **Confirm** torus water level is greater than -8 feet.

Verifies torus water level greater than -8 feet on CAC-LI-3342 (Supp Pool Level)

SAT / UNSAT

Step 13 - **Monitor** RPV level.

Monitors RPV level on B21-LI-R604BX (Reactor Water Level).

SAT / UNSAT

NOTE:	SRV B, E, and G can be opened in any sequence. Each SRV being opened is a critical step.
PROMPT:	Once SRVs are opened and if asked the status of pressure on C32-PI-3332 (Reactor Pressure). State pressure is lowering on C32-PI-3332 (Reactor Pressure) and is currently 350 psig.

Step 14 - **Monitor** and **control** RPV pressure using SRVs B, E and G as directed by Control Room.

Places OPEN/CLOSE control switches to OPEN for:

SRV	***SAT***	***UNSAT***
B		
E		
G		

Verifies Reactor Pressure is lowering on C32-PI-3332 (Reactor Pressure)

**** CRITICAL STEP **   SAT / UNSAT**

Step 15 - **Notify** the Control Room SRVs B, E, and G are open and reactor pressure is lowering.

Notifies the control room that SRVs B, E, and G are open, and RPV pressure is 350 psig and lowering.:

SAT / UNSAT

TERMINATING CUE: When SRVs B, E, and G have been opened, and the control room has been notified this JPM is complete.

TIME COMPLETED ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Non Critical	Communication related
3	Non Critical	Communication related
4	Non Critical	Communication related
5	Non Critical	Verification
6	Critical	Necessary to open SRV
7	Non Critical	Verification
8	Critical	Necessary to open SRV
9	Non Critical	Verification
10	Critical	Necessary to open SRV
11	Non Critical	Line up for parameter monitoring
12	Non Critical	Parameter monitoring
13	Non Critical	Parameter monitoring
14	Critical	Necessary to depressurize RPV
15	Non Critical	Communication related

REVISION SUMMARY

0	New JPM

Validation Time: 10 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	____	Unit:	1/ 2
Setting:	In-Plant	X	Simulator	____	Admin	___
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

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☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

TASK CONDITIONS:

1. Unit _____ is executing 0EOP-01-EDP.
2. Reactor Pressure is 500 psig
3. SRV operation from the Control Room is not successful.

INITIATING CUE:

You are directed by the Control Room Supervisor to rapidly depressurize the RPV by opening SRVs B, E, and G from the Unit _____ Remote Shutdown Panel (RSDP) IAW 0EOP-01-LEP-05.

DUKE ENERGY

BRUNSWICK UNIT 0

EMERGENCY OPERATING PROCEDURE

0EOP-01-LEP-05

REMOTE SHUTDOWN PANEL SRV OPERATION

REVISION 0

Special Considerations:

EOP-Protected procedure – Any revision to this procedure should be reviewed by an EOP writer.

REVISION SUMMARY
PRR 709061 **DESCRIPTION**
New procedure for operation of SRVs at the Remote Shutdown Panel (RSDP) and EC 99559. Written by Mike English.

TABLE OF CONTENTS

1.0 ENTRY CONDITIONS

- As directed by Emergency Operating Procedures (EOPs)

2.0 INSTRUCTIONS

2.1 SRV Operation

2.1.1 Manpower Required

- 1 Reactor Operator

2.1.2 Special Equipment

- RO Desk Locked Drawer
  - ◊ 1 LEP toolbox key (LSV-1) .... ☐
- ASSD Toolbox in Control Room
  - ◊ 1 sound-powered phone with extension cord for RO........................ ☐
- Reactor Building 20' LEP Toolbox
  - ◊ 4 T112 keys .... ☐
  - ◊ 1 sound-powered phone with extension cord for RO........................ ☐

2.1.3 Operator Actions

1. **Notify** CRS:
   - RTGB level indicators B21-LI-R604B and C32-PR-R609 (N026B) will be lost. .... ☐ RO
   - Control of SRV B, E and G from RTGB will be lost. .... ☐ RO

> **NOTE**
>
> ASSD Unit 1(2) Train B sound powered phone circuit is located on the RSDP. .... ☐

2. **Establish** communication between RSDP and Control Room. .... ☐ RO

2.1.3 Operator Actions (continued)

3. At the Remote Shutdown Panel:

   a. **Ensure** B21-F013E (Manual Relief E Vlv Close/Open) control switch in CLOSE. .... ☐ RO

   b. **Place** B21-F013E (Manual Relief E Vlv Normal/Local) control switch in LOCAL. .... ☐ RO

   c. **Ensure** B21-F013G (Manual Relief G Vlv Close/Open) control switch in CLOSE. .... ☐ RO

   d. **Place** B21-F013G (Manual Relief G Vlv Normal/Local) control switch in LOCAL. .... ☐ RO

   e. **Ensure** B21-F013B (Manual Relief B Vlv Close/Open) control switch in CLOSE. .... ☐ RO

   f. **Place** B21-F013B (Manual Relief B Vlv Normal/Local) control switch in LOCAL. .... ☐ RO

   g. **Place** B21-CS-3345 (Reactor Water Level Normal/Local Switch) in LOCAL to transfer level transmitter B21-LT-N026B output to B21-LI-R604BX. .... ☐ RO

NOTE

CAC-LI-3342 (Supp Pool Level) is on the RSDP. .... ☐

4. **Confirm** torus water level is greater than -8 feet.. .... ☐ RO

2.1.3 Operator Actions (continued)

NOTE

- B21-LI-R604BX (Reactor Water Level) is on the RSDP. ☐
- Attachment 1 provides Caution 1 requirements for B21-LI-R604BX. ☐

CAUTION

RSDP RPV level indicator B21-LI-5977 is calibrated for hot conditions and includes significant loop uncertainties. Therefore it is not expected to be consistent with Control Room indication and B21-LI-R604BX should be used. ☐

5. **Monitor** RPV level.. ☐
RO

NOTE

C32-PI-3332 (Reactor Pressure) is on the RSDP. ☐

CAUTION

- SRV indication on the Remote Shutdown Panel is control switch position **NOT** valve position. ☐
- Power to the SRV acoustic monitors comes from Sub E6(E8). If DC power is lost to the RTGB SRV control circuits, SRV position indication lights on Panel XU-73 should be available. ☐

6. **Monitor** and **control** RPV pressure using SRVs B, E and G as directed by Control Room. ☐
RO

7. **WHEN** SRV operation, from RSDP, is **NO** longer required, **THEN exit** this section and **go to** Section 2.2 ☐
RO

2.2 <u>Restoration</u>

2.2.1 Manpower Required

- 2 Reactor Operators

2.2.2 Special Equipment

- <u>RO Desk Locked Drawer</u>
  - ◊ 1 LEP toolbox key (LSV-1) .... ☐

2.2.3 Operator Actions

1. **Record** Unit number. .... ________

   Unit Number ____________

2. At the RTGB:

   a. **Ensure** B21-F013B (Manual Relief B Vlv) control switch in CLOSE.. ____/____ CV

   b. **Ensure** B21-F013E (Manual Relief E Vlv) control switch in CLOSE.. ____/____ CV

   c. **Ensure** B21-F013G (Manual Relief G Vlv) control switch in CLOSE.. ____/____ CV

3. At the Remote Shutdown Panel:

   a. **Ensure** B21-F013E (Manual Relief E Vlv Close/Open) control switch in CLOSE. .... ____/____ IV

   b. **Place** B21-F013E (Manual Relief E Vlv Normal/Local) control switch in NORMAL and **remove** key. .... ____/____ IV

   c. **Ensure** B21-F013G (Manual Relief G Vlv Close/Open Control Switch) in CLOSE. .... ____/____ IV

   d. **Place** B21-F013G (Manual Relief G Vlv Normal/Local) control switch in NORMAL and **remove** key.. ____/____ IV

2.2.3 Operator Actions (continued)

e. **Ensure** B21-F013B (Manual Relief B Vlv Close/Open) control switch in CLOSE. .... ____/____ IV

f. **Place** B21-F013B (Manual Relief B Vlv Normal/Local) control switch in NORMAL and **remove** key. .... ____/____ IV

g. **IF** EOP-01-LEP-04 **NOT** in progress,
THEN:

(1) **Place** B21-CS-3345 (Reactor Water Level Normal/Local Switch) in NORMAL and **remove** key to transfer level transmitter B21-LT-N026B output to the RTGB. .... ____/____ IV

(2) **Confirm** RTGB level indicators consistent with plant conditions:

- B21-LI-R604B .... ____/____ IV
- C32-PR-R609 (N026B) .... ____/____ IV

h. **Confirm** CLOSED at RTGB:

- B21-F013B (Manual Relief B Vlv). .... ____/____ IV
- B21-F013E (Manual Relief E Vlv). .... ____/____ IV
- B21-F013G (Manual Relief G Vlv) .... ____/____ IV

4. **Restore** headset to Control Room ASSD equipment tool box .... ________
5. **Perform** a Control Room ASSD tool box inventory. .... ________
6. **Restore** EOP equipment to RO desk locked drawer .... ________
7. **Perform** a Control Room EOP inventory. .... ________
8. **Restore** EOP equipment to the tool boxes .... ________

2.2.3 Operator Actions (continued)

9. **Perform** a LEP tool box inventory. .... _________

10. **Exit** this procedure and **continue** in procedure(s) in effect. .... _________

Date/Time Completed ______________________

Performed By (Print) Initials

______________________________ _______

______________________________ _______

______________________________ _______

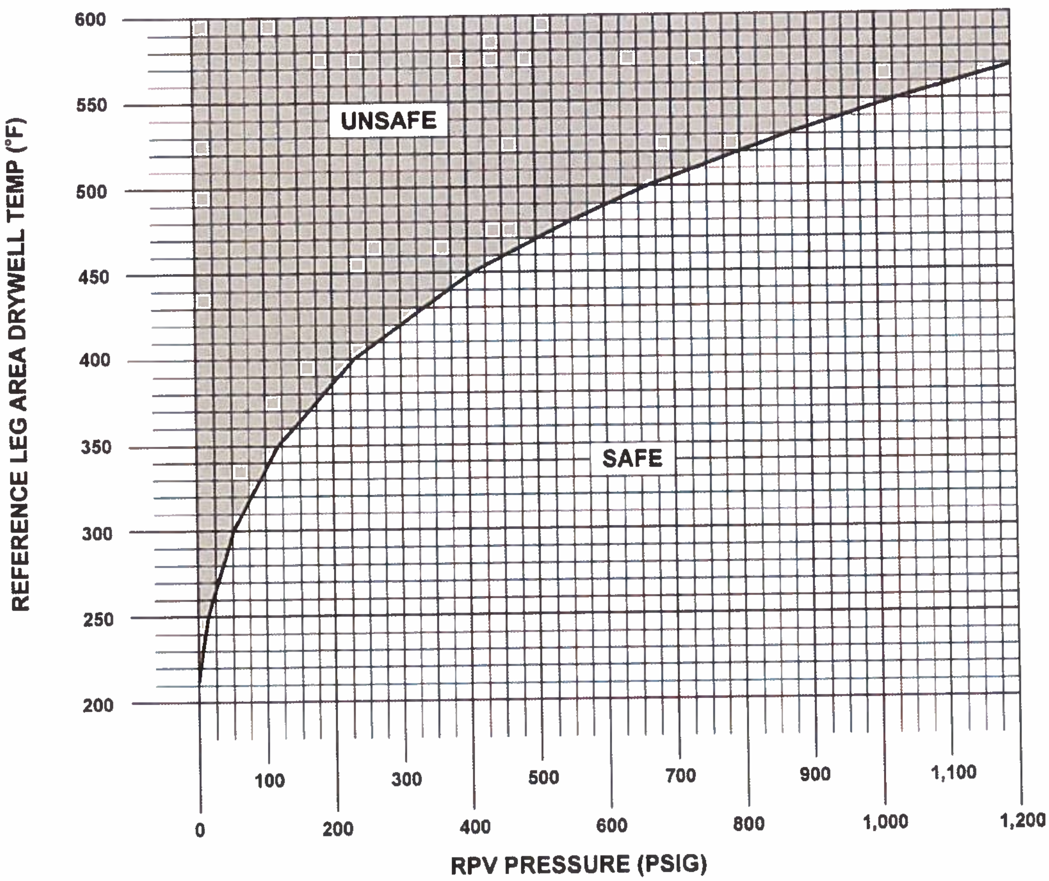
______________________________ _______

Reviewed By: ____________________________
CRS

ATTACHMENT 1
Page 1 of 1

Caution 1 Information for B21-LI-R604BX

Reactor Saturation Limit



INSTRUMENT:
Wide Range Level Instruments
B21-LI-R604BX (B21-LT-N026B)
Indicating Range 0-210 Inches
Cold Reference Leg

CONDITIONS FOR USE:

1. Temperature on Reactor Building 50' is below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102 or ERFIS Computer Point B21TA103)
   AND .... ☐
2. If reference leg area drywell temperature (RSDP CAC-TR-778, Point 1 or RTGB) in UNSAFE region of Reactor Saturation Limit, indicated level is greater than 20 inches.......... ☐

   OR

   If reference leg area drywell temperature (RSDP CAC-TR-778, Point 1 or RTGB) in SAFE region of Reactor Saturation Limit, indicated level is greater than 10 inches.......... ☐

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

IP JPM K - 2016 NRC INITIAL EXAM - RO/ISRO/USRO

LESSON TITLE: **Racking In E6 Cross-Tie with Breaker Charging Spring Failure**

LESSON NUMBER: **AOT-OJT-JP-303-13**

REVISION NO: **6**

Dan Hulgin 8/18/16
PREPARER / DATE

Bob Bolin 9/07/16
TECHNICAL REVIEWER / DATE

Hunter Morris 9/07/16
VALIDATOR / DATE

Craig Oliver 09/27/2016
LINE SUPERVISOR / DATE

9-28-2016
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

262605B104 - Rack in a 480 Volt Electrically Operated Breaker (K-3000) per 1(2)OP-50.

K/A REFERENCE AND IMPORTANCE RATING:

295003 AA1.01 3.7 / 3.8
Ability to Operate and/or Monitor AC Electrical Distribution System as it applies to a partial or complete loss of A.C. power.

REFERENCES:

0EOP-01-SBO-07, 480V E-bus Crosstie

TOOLS AND EQUIPMENT:

Racking tool for 480V Breakers
Manual charging handle for 480V Breaker

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

6 (Electrical Distribution)

SAFETY CONSIDERATIONS:

1. Notify SM/CRS of JPM performance prior to commencing In-plant JPM.
2. Determine actual radiological conditions and potentially contaminated areas to achieve ALARA.
3. Ensure all electrical safety requirements are observed.
4. Review Work Practices section prior to conduct of the JPM.
5. DO NOT OPERATE any plant equipment during performance of this JPM.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee. 0EOP-01-SBO-07, Attachment 1, will be provided to the examinee when asked for.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM will be performed on Unit 1.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. A complete Loss of Offsite Power has occurred on both Units.
2. 0EOP-01-SBO-07 is being executed, and Step 2.1.3.11 is ready to be performed.
3. A Flex DG is NOT supplying E6.
4. 480v Crosstie breaker on E5 has been racked in.

INITIATING CUE:

You are directed by the Reactor Operator to complete the Auxiliary Operator actions associated with cross-tying 480V Substation E5 to E6 IAW 0EOP-01-SBO-07, Step 2.1.3.11, and inform the Control Room when the E5 to E6 cross-tie breakers are ready to be closed.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

Step 1 - Perform take a minute at job site prior to beginning task.

Examinee should cover the following questions, as deemed necessary. What are the hazards in the area? What PPE is required? Tools/PPE inspected prior to use? Energy sources secured/isolated? Is Clearance/Tag Out sufficient? What's the worst that can happen? Any ALARA concerns? Will I affect plant status? HU Tools needed?

SAT / UNSAT

TIME START __________

<u>NOTE</u>: A 480 V racking tool is contained in the DG Building 23' LEP Toolbox.

<u>PROMPT:</u> Inform Examinee that use of electrical safety equipment may be simulated, but that the examinee should state the location of this equipment. Inform Examinee that electrical equipment compartments are NOT to be breached.

<u>NOTE</u>: If requested, pictures will be provided of the internals of the 480 V breaker.

Step 2 – At E6, Row F1, rack in Compt AX1 (Tie Breaker To E5): **Confirm** locally breaker OPEN.

(Tie breaker to E5) Compt AX1 on Bus E6 verified open as indicated by the green open flag.

SAT / UNSAT

<u>PROMPT</u>: If asked, inform the examinee that the locking hasp position is as seen.

Step 3 - At E6, Row F1, rack in Compt AX1 (Tie Breaker To E5): **<u>IF</u>** necessary, **<u>THEN</u> depress** locking hasp to allow opening of racking shutter.

Locking hasp DEPRESSED or verified to already be depressed on E6 Compt. AX1.

****CRITICAL STEP**** **SAT / UNSAT**

Step 4 - At E6, Row F1, rack in Compt AX1 (Tie Breaker To E5):**Rotate** racking crank clockwise until breaker stops.

Breaker Compt AX1 on Bus E6 stops in the CONNECT position (racked in and shutter window closes when the racking tool is removed).

**** CRITICAL STEP **   SAT / UNSAT**

PROMPT:	As requested, inform the examinee that the closing springs failed to charge as indicated by lack of charging noise when toggle switch turned on and/or lack of spring charged indicator at front of breaker. ****ALTERNATE PATH BEGINS AT STEP 5****

Step 5 - At E6, Row F1, rack in Compt AX1 (Tie Breaker To E5): Place Charging Power toggle switch to ON, determine springs failed to charge, and Attachment 1, Manually Charging 480v Breaker Charging Springs is required.

Charging power switch for E6 Compt AX1 placed to the ON position, springs determined not charged, and Attachment 1 determined to be used.

SAT / UNSAT

NOTE:	A manual charging handle for the 480 VAC cross-tie breaker springs is located in the DG Building 23' LEP Toolbox.

Step 6 – **Place** charging power toggle switch to OFF. (attachment 1)

Charging power toggle switch is OFF (down position).

SAT / UNSAT

PROMPT:	Provide a picture to the trainee to identify the location of the manual charging lever. The manual charging lever is located at the bottom middle of the 480 VAC breaker. The equipment enclosure should NOT be breached.

Step 7 - ***Describe*** the action to open breaker door, and insert manual charging handle behind the breaker compartment door, using 480v breaker pictures.

Manual charging handle is inserted in the breaker.

**** CRITICAL STEP **   SAT / UNSAT**

Step 8 – (Simulate) **Pump** manual charging handle until closing springs are charged (clicks into position) and **confirm** charge is satisfactory by Springs Charged indicator.

Closing springs are fully charged, as indicated by the yellow springs charged indication.

**** CRITICAL STEP **   SAT / UNSAT**

Step 9 – (Simulate) **Remove** manual charging handle and close compartment door.

Manual charging handle removed, compartment door closed.

SAT / UNSAT

Step 10 – **Place** charging power toggle switch to ON.

Charging power toggle switch is ON (up position).

SAT / UNSAT

NOTE: Step 2.1.3.12 is N/A, a Flex DG is not supplying E6.

Step 11 – **Inform** control room that E5-E6 tie breakers are ready to be closed.

Control room contacted and told E5-E6 tie breakers are ready to be closed.

SAT / UNSAT

PROMPT: When contacted as control room that E5-E6 tie breakers are ready to be closed, inform examinee to stand clear so the breakers can be closed.

TERMINATING CUE: Substation E6 crosstie breaker is racked in, closing springs charged and is ready to be closed then this JPM is complete.

TIME COMPLETED ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Administrative
2	Non Critical	Verify only. No action required.
3	Critical	Required to complete task.
4	Critical	Required to complete task.
5	Non Critical	Since charging springs are not charged, turning power on accomplishes nothing.
6	Non Critical	Places system in original configuration, but does completes action.
7	Critical	Action required to complete task.
8	Critical	Action required to complete task.
9	Non Critical	Actions not required to accomplish task.
10	Non Critical	Actions not required to accomplish task.
11	Non Critical	Communicates results of actions.

REVISION SUMMARY

6	Changed 2.1.3.10 to 2.1.3.11 due to procedure numbering change (non-technical change). Changed wording on steps to match procedure verbiage (non-technical change).
5	Changed Duke logo. Revised from 0AOP-36.2 to 0EOP-01-SBO-07 Added pictures of the 480v Breaker

Validation Time: 12 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	____	Unit:	1
Setting:	In-Plant	X	Simulator	____	Admin	___
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	X	No	____		

EVALUATION

Performer: ________________________________

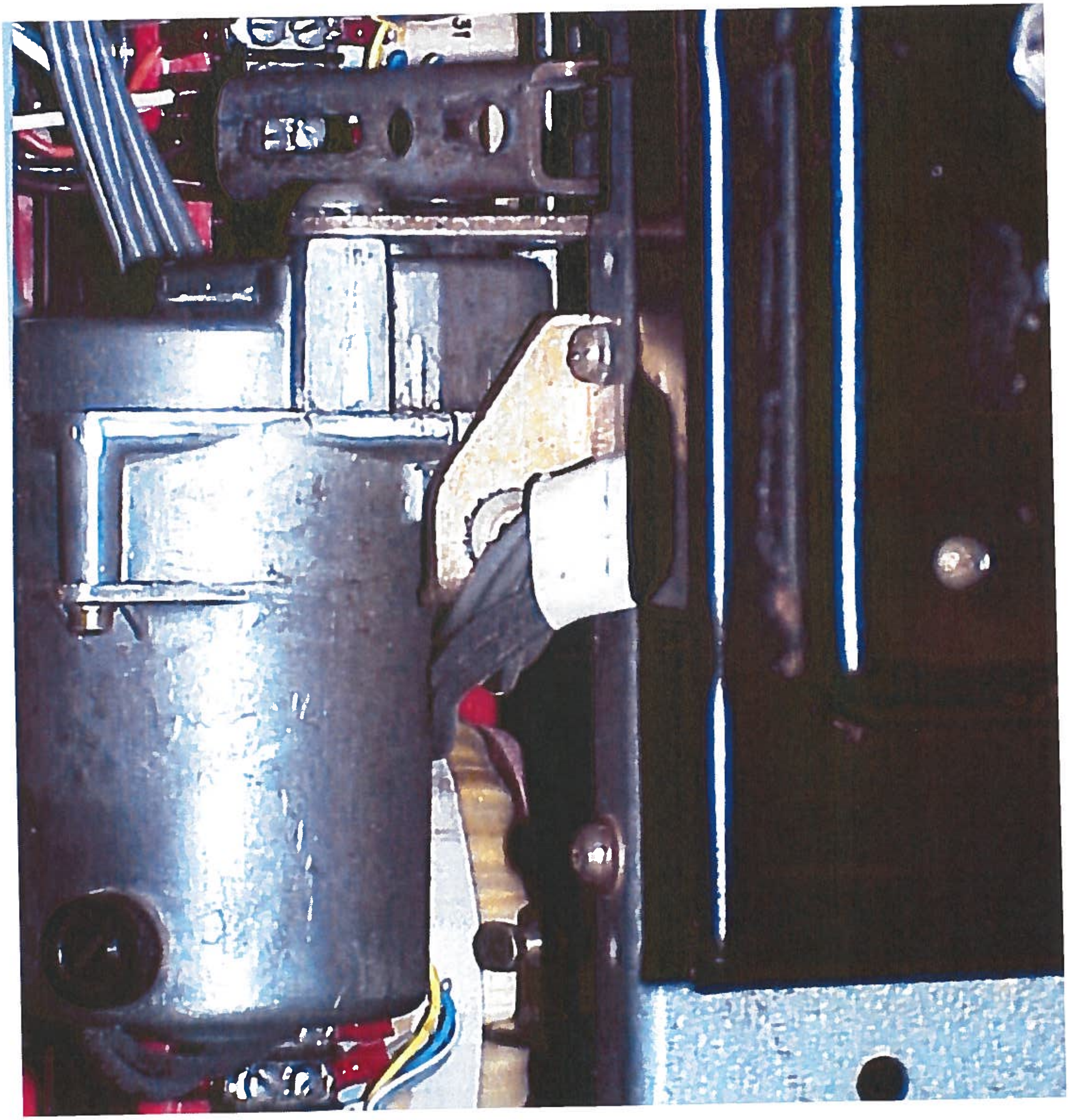
JPM: Pass _____ Fail _____

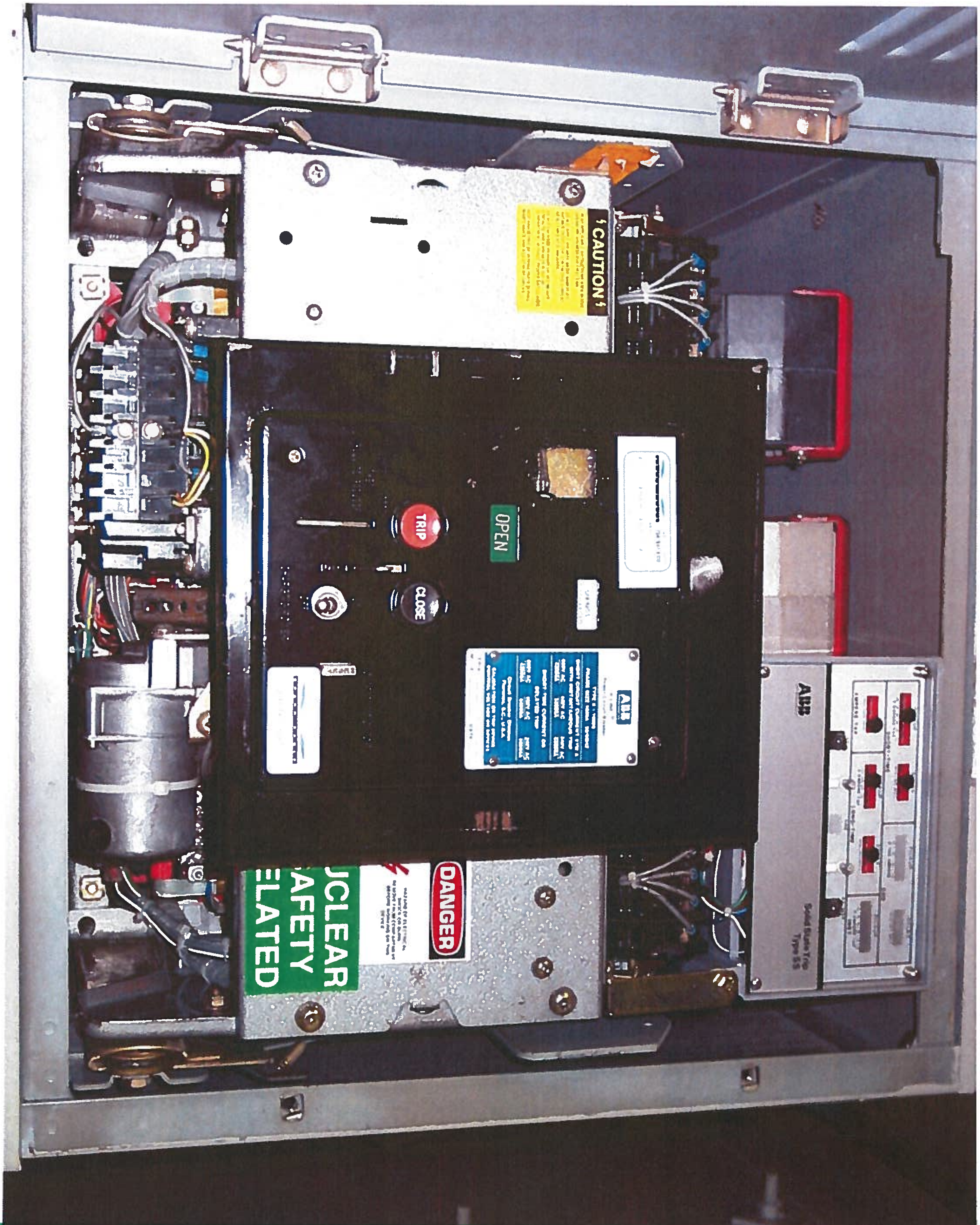
Remedial Training Required: Yes _____ No _____

Comments: ____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: ___________




CAUTION
TRIP
OPEN
CLOSE
DANGER
ABB
Solid State Trip
Type SS

TASK CONDITIONS:

1. A complete Loss of Offsite Power has occurred on both Units.
2. 0EOP-01-SBO-07 is being executed, and Step 2.1.3.11 is ready to be performed.
3. A Flex DG is NOT supplying E6.
4. 480v Crosstie breaker on E5 has been racked in.

INITIATING CUE:

You are directed by the Reactor Operator to complete the Auxiliary Operator actions associated with cross-tying 480V Substation E5 to E6 IAW 0EOP-01-SBO-07, Step 2.1.3.11, and inform the Control Room when the E5 to E6 cross-tie breakers are ready to be closed.

1.0 ENTRY CONDITIONS

- As directed by Emergency Operating Procedures (EOPs)

2.0 INSTRUCTIONS

2.1 Energizing E5 From E6

2.1.1 Manpower Required

- 1 Reactor Operator
- 1 Auxiliary Operator

2.1.2 Special Equipment

- RO Desk Locked Drawer
  - ◊ 1 LEP toolbox key (LSV-1) .... ☑
- 1 flashlight .... ☑
- Diesel Building 23' LEP Toolbox
  - ◊ 1 Racking tool for 480V breakers .... ☑
- Unit 1 Control Building 23' Stairwell LEP Toolbox
  - ◊ 1 rope 10 ft. to tie open battery room door .... ☑

2.1.3 Energizing E5 From E6 Actions

1. IF a FLEX DG supplying E6,
   THEN at MCC 1CA **ensure** OFF:
   - Row A1, Compt C03 (Ventilation A/C Cndsr 1D 1-VA-1D-CU-CB) .... ☐N/A AO
   - Row A2, Compt C17 (Control Bldg Floor Drain Sump Pmp 1-CB-1E-1) .... ☐N/A AO
   - Row A3, Compt C16 (CB HVAC Ctl Sys Instr Air Compr 2-VA-2B-AC-CB) .... ☐N/A AO

2.1.3 Energizing E5 From E6 Actions (continued)

- Row B1, Compt C08 (Control Bldg Floor Drain Sump Pmp 1-CB-1E-2) .... ☐ N/A AO
- Row B4, Compt C28 (FP Filt/Demin 1A Holding Pump 1-G41-Z001-7A) .... ☐ N/A AO
- Row C1, Compt C04 (Ventilation A/C Supply Fan 1D 1-VA-1D-SF-CB) .... ☐ N/A AO
- Row C2, Compt C18 (Ventilation Exhaust Fan 1A 1-VA-1A-EF-CB) .... ☐ N/A AO
- Row C3, Compt C19 (Ventilation Supply Fan 1A 1-VA-1A-SF-CB) .... ☐ N/A AO
- Row C4, Compt C20 (Ventilation Exhaust Fan 1C 1-VA-1C-EF-CB) .... ☐ N/A AO
- Row D2, Compt C21 (Ventilation Supply Fan 1C 1-VA-1C-SF-CB) .... ☐ N/A AO
- Row E2, Compt C24, (SAMA Diesel Generator #2 Supply) .... ☐ N/A AO
- Row E4, Compt C12 (Fdr Bkr For Ltg Distr Xfmr 1CE1 1-1CE1-XFMR) .... ☐ N/A AO
- Row F1, Compt C11 (Fdr Bkr For Conv Distr Pnl Xfmr 1G-CB 1-1G-CB-XFMR) .... ☐ N/A AO
- Row F2, Compt C09 (Control Room Heating Coil 1-VA-1A-EHE-CB) .... ☐ N/A AO

2.1.3 Energizing E5 From E6 Actions (continued)

- Row F4, Compt C10 (RPS MG Set 1A Control Panel 1-C71-S001A) .... ☐ N/A AO
- Row F5, Compt C06 (Batt Charger 1A-2 AC Input Normal Feed 1-1A-2-125VDC-CHRGR) .... ☐ N/A AO
- Row F6, Compt C07 (Primary UPS 1A, 1-UPS-1A) .... ☐ N/A AO
- Row F7, Compt C05 (Batt Charger 1A-1 AC Input Normal Feed 1-1A-1-125VDC-CHRGR) .... ☐ N/A AO
- Row G2, Compt C29 (Primary UPS 1A Heater) .... ☐ N/A AO
- Row G4, Compt C27 (CB Electric Unit Heater 1-VA-1A-UH-CB) .... ☐ N/A AO
- Row G5, Compt C32 (CB Cndsr Unit Exh Booster Fan 1-VA-1A-BF-CB) .... ☐ N/A AO
- Row H3, Compt XX5 (Transformer 480/208/120V 9KVA 3Ø 1-1CA-DIST-PNL-XFMR) .... ☐ N/A AO

2. At E1, Row N1, Compt AF9 (Nuc Serv Wtr Pmp 1A):

   a. **Ensure** breaker OPEN .... ☑ AO

   b. **Remove** normal control power fuses .... ☑ AO

   c. **Remove** alternate control power fuses .... ☑ AO

3. At E5 **open:**

   - Row A2, Compt AU6 (Feeder To MCC DGA) .... ☑ AO

2.1.3 Energizing E5 From E6 Actions (continued)

- Row B1, Compt AT7 (MCC 1OG) .... ☑ AO
- Row B2, Compt AT8 (Feeder To MCC 2XA-2)................................ ☑ AO
- Row B3, Compt AT9 (MCC 1XC) .... ☑ AO
- Row C1, Compt AU4 (Feeder Breaker MCC 1XA)........................... ☑ AO
- Row C2, Compt AU1 (MCC 2XJ) .... ☑ AO
- Row C3, Compt AU2 (MCC 1XE).. ☑ AO
- Row C4, Compt AU3 (MCC 1PA).. ☑ AO
- Row E3, Compt AV0 (MCC 1XG).. ☑ AO

4. <u>**IF**</u> DG 2 <u>**NOT**</u> available,
   <u>**THEN**</u> at E5, Row D2, **open** Compt AU5 (MCC 1XL) .... ☑ AO

5. <u>**IF**</u> a FLEX DG supplying E6,
   <u>**THEN**</u> **go to** Section 2.1.3 Step 8 .... ☐ N/A RO

6. At E6 **open**:

   - Row B3, Compt AV5 (MCC 1XH).. ☑ AO
   - Row C1, Compt AV6 (Distr Pnl E10).. ☑ AO
   - Row C2, Compt AV7 (Distr Pnl E9).. ☑ AO
   - Row C3, Compt AV8 (Feeder To MCC 2XB-2) .... ☑ AO

2.1.3 Energizing E5 From E6 Actions (continued)

- Row D1, Compt AW0 (Feeder To MCC 1XB) .... ☑ AO
- Row D2, Compt AW1 (MCC 2XK) .... ☑ AO
- Row E1, Compt AW4 (MCC 2OG) .... ☑ AO
- Row F2, Compt AW6 (MCC 1XD) .... ☑ AO

7. **<u>IF</u>** DG 2 **<u>NOT</u>** available,
   <u>THEN</u> at E6 **open:**

- Row C4, Compt AV9 (MCC DGB) .... ☑ AO
- Row D3, Compt AW2 (MCC 1XF) .... ☑ AO
- Row D4, Compt AW3 (MCC 1PB) .... ☑ AO
- Row E2, Compt AW5 (MCC 1XM) .... ☑ AO

8. **Place** control switch for E5 feeder breakers to TRIP and **confirm** OPEN: .... ☑ RO

- Breaker AU9 (Sub E5 480V Main Breaker) .... ☑ RO
- Breaker AF8 (Bus E1 To Sub E5) .... ☑ RO

9. **<u>IF</u>** breaker AU9 (Sub E5 480V Main Breaker) will **<u>NOT</u>** open electrically,
   <u>THEN</u>: .... ☐ N/A RO

   At E5, Row E2, Compt AU9 (Main Breaker):

   a. **Depress** TRIP pushbutton on Compt AU9 .... ☐ N/A AO

2.1.3 Energizing E5 From E6 Actions (continued)

b. **Confirm** Compt AU9 OPEN. ☐ N/A
AO

10. At E5, Row A1, **rack in** Compt AT4 (Tie Breaker To E6):

a. **Confirm** locally breaker OPEN. ☑
AO

b. <u>**IF**</u> necessary,
<u>**THEN**</u> **depress** locking hasp to allow opening of racking shutter. ☑
AO

NOTE
Breaker CONNECTED when racking shutter drops when racking tool removed. ☑

CAUTION
Crank <u>**NOT**</u> to be forced after breaker stops ☑

c. **Rotate** racking crank clockwise until breaker stops. ☑
AO

d. **Place** Charging Power toggle switch to ON ☑
AO

e. **Confirm** charge satisfactory by SPRINGS CHARGED indicator. ☑
AO

f. <u>**IF**</u> closing springs fail to charge,
<u>**THEN**</u> manually **charge** per Attachment 1 and **return**. ☐ N/A
AO

11. At E6, Row F1, **rack in** Compt AX1 (Tie Breaker To E5):

a. **Confirm** locally breaker OPEN. ☐
AO

b. <u>**IF**</u> necessary,
<u>**THEN**</u> **depress** locking hasp to allow opening of racking shutter. ☐
AO

2.1.3 Energizing E5 From E6 Actions (continued)

NOTE

Breaker CONNECTED when racking shutter drops when racking tool removed. .... ☐

CAUTION

Crank **NOT** to be forced after breaker stops .... ☐

c. **Rotate** racking crank clockwise until breaker stops. .... ☐ AO

d. **Place** Charging Power toggle switch to ON .... ☐ AO

e. **Confirm** charge satisfactory by SPRINGS CHARGED indicator.. ☐ AO

f. IF closing springs fail to charge,
THEN manually **charge** per Attachment 1 and **return**...................... ☐ AO

NOTE

Crosstie breakers will **NOT** close electrically if a FLEX DG supplying E6. .... ☐

12. IF a FLEX DG supplying E6,
THEN go to Section 2.1.3 Step 14. .... ☐ AO

13. **Place** and **hold** control switch for Bus E5 Tie To Bus E6 cross-tie breakers in CLOSE until both AT4 (Mstr) and AX1 (Slave) indicate CLOSED. .... ☐ RO

14. IF Breaker AT4 (Tie Breaker To E6) will **NOT** close electrically,
THEN:.. ☐ RO

At E5, Row A1, Compt AT4 (Tie Breaker To E6):

a. **Lift** Manual Close lever on Compt AT4. .... ☐ AO

b. **Confirm** Compt AT4 CLOSED.. ☐ AO

2.1.3 Energizing E5 From E6 Actions (continued)

15. **IF** breaker AX1 (Tie Breaker To E5) will **NOT** close electrically,
THEN: .... ☐
RO

At E6, Row F1, Compt AX1 (Tie Breaker To E5):

a. **Lift** Manual Close lever on Compt AX1 .... ☐
AO

b. **Confirm** Compt AX1 CLOSED. .... ☐
AO

16. **IF** either Tie Breaker was manually closed,
THEN notify Control Room E5 ENERGIZED from E6 .... ☐
AO

17. **IF** a FLEX DG supplying E6,
THEN: .... ☐
RO

a. At MCC 1CA **place** ON:

- Row F5, Compt C06 (Batt Charger 1A-2 AC Input Normal Feed 1-1A-2-125VDC-CHRGR) .... ☐
AO

- Row F7, Compt C05 (Batt Charger 1A-1 AC Input Normal Feed 1-1A-1-125VDC-CHRGR) .... ☐
AO

- Row A3, Compt C16 (CB HVAC Ctl Sys Instr Air Compr 2-VA-2B-AC-CB) .... ☐
AO

- Row C4, Compt C20 (Ventilation Exhaust Fan 1C 1-VA-1C-EF-CB) .... ☐
AO

- Row D2, Compt C21 (Ventilation Supply Fan 1C 1-VA-1C-SF-CB) .... ☐
AO

2.1.3 Energizing E5 From E6 Actions (continued)

b. **IF** 1B battery room vent fans **NOT** OPERATING,
THEN: .... ☐
RO

(1) At MCC 1CB **ensure** ON:

- Row B1, Compt C43 (Vent Supply Fan 1B, 1B-SF-CB) .... ☐
AO

- Row B2, Compt C42 (Vent Exhaust Fan 1B, 1B-EF-CB) .... ☐
AO

(2) **Start** 1B-SF-CB and 1B-EF-CB (Battery Room 1B Vent Fans). .... ☐
RO

18. **Start** 1C-SF-CB and 1C-EF-CB (Battery Room 1A Vent Fans). .... ☐
RO

WARNING

Batteries generate hydrogen gas when charging. .... ☐

19. **IF** battery room vent fans **NOT** operating,
THEN: .... ☐
RO

a. **Tie open** door CTB-DR-EL023-118 between 1A Battery Room and Cable Spread. .... ☐
AO

b. **Notify** TSC batteries 1A-1 and 1A-2 charging without battery room ventilation. .... ☐
RO

20. **Confirm** for Battery Charger 1A-1:

- BAT-AM-6000 (D.C. Amperes) above 10 amps .... ☐
AO

- BAT-VM-6009 (D.C. Volts) 130 to 140 volts .... ☐
AO

2.1.3 Energizing E5 From E6 Actions (continued)

21. **Confirm** for Battery Charger 1A-2:

- BAT-AM-6001 (D.C. Amperes) above 10 amps .... ☐ AO

- BAT-VM-6010 (D.C. Volts) 130 to 140 volts.................................... ☐ AO

22. **<u>IF</u>** battery chargers **<u>NOT</u>** ENERGIZED,
<u>THEN</u>:

a. **Notify** Control Room .... ☐ AO

b. **Assess** upstream electrical alignment. .... ☐ RO

c. **Notify** CRS of assessment results. .... ☐ RO

ATTACHMENT 1
Page 1 of 1

<< Manually Charging 480v Breaker Charging Springs >>

1.0 MANPOWER REQUIRED

- 1 Auxiliary Operator

2.0 SPECIAL EQUIPMENT

- RO Desk Locked Drawer
  - ◊ 1 LEP toolbox key (LSV-1) .... ☐
- 1 flashlight.. ☐
- Diesel Building 23' LEP Toolbox
  - ◊ 1 Manual charging handle for 480V breakers.................................. ☐

3.0 OPERATOR ACTIONS

CAUTION
Manually overcharging closing springs may cause breaker to bind. ....☐ ☐

1. **Place** Charging Power toggle switch to OFF. ....☐ ☐ AO

2. **Open** breaker compartment door and **insert** manual charging handle. ....☐ ☐ AO

3. **Pump** manual charging handle until closing springs charged (clicks into position) and **confirm** charge satisfactory by SPRINGS CHARGED indicator. ....☐ ☐ AO

4. **Remove** manual charging handle and **close** breaker compartment door..☐ ☐ AO

5. **Place** Charging Power toggle switch to ON. ....☐ ☐ AO

6. **Return** to crosstie section..☐ ☐ AO

DUKE ENERGY

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

LESSON TITLE: PERFORM SJAE OFF-GAS RADIATION MONITORS CHANNEL CHECK CALCULATION

LESSON NUMBER: LOT-ADM-JP-201-D15

REVISION NO: 0

Daniel Hulgin 09/06/16
PREPARER / DATE

Bob Bolin 09/06/16
TECHNICAL REVIEWER / DATE

Kyle Cooper 09/06/16
Dwayne Wolf 09/06/16
Hunter Morris 09/06/16
VALIDATOR / DATE

Craig Oliver 09/27/2016
LINE SUPERVISOR / DATE

9-28-2016
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

299201B201
Perform Daily Surveillance Report Per OI-3.1 or OI-3.2

K/A REFERENCE AND IMPORTANCE RATING:

GEN 2.1.25 3.9/4.2
Ability to interpret reference materials, such as graphs, curves, tables, etc.

REFERENCES:

2OI-03.2, *Reactor Operator Daily Surveillance Report*
ODCM

TOOLS AND EQUIPMENT:

Student may use calculator

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Generic (Administrative)

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. Readings for D12-RM-K601A, *SJAE Off Gas Rad Monitor A*, and D12-RM-K601B, *SJAE Off Gas Rad Monitor B,* have been recorded on the Unit 2 Dayshift RODSR for Saturday 0630-1230.
2. Main Condenser Air Ejector is in operation.
3. HP has reported a local survey reading of 300 mR

INITIATING CUE:

RO, and SRO candidates:

You are directed by the Control Room Supervisor to complete item 108, *SJAE Off-Gas Radiation Monitors Channel Check*, of 2OI-03.2, *Reactor Operator Daily Surveillance Report,* and state the status of the channel check.

- Channel check is SAT
- Channel check is UNSAT

SRO ONLY:

Based on the above information, determine the required actions, if any.

____

____

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

TIME START: ___________

Step 1 – Record SJAE OFF-Gas Radiation Monitor readings from item 104 & 106 in Table 1.

*__**452**__ for D12-RM-K601A recorded in SAT block of table 1, and __**224**__ for D12-RM-K601B recorded in SAT block of table 1 .*

SAT/UNSAT

Step 2 – Determine D12-RM-K601A is the highest reading, and divide by 2.

*Value for D12-RM-K601A divided by 2 determined to be __**226**__.*

SAT/UNSAT

Step 3 – Compare lower reading monitor to value in step 3.

*Determines D12-RM-K601B value of __**224**__ is < value in step 3 (__**226)**__. Determines the channel check is not yet satisfactory.*

****CRITICAL STEP** SAT/UNSAT**

Step 4 – Contact E&RC health physics to obtain a local reading with an appropriate survey instrument.

Determines Information from the E&RC survey is needed.

SAT/UNSAT

Step 5 – Record local survey instrument reading

*Records __**300**__ in local survey instrument reading SAT block in attachment 1 .*

SAT/UNSAT

Step 6 – Multiply local survey instrument reading by 0.75

Determines local instrument times 0.75 *is __**225**__.*

SAT/UNSAT

Step 7 – Compare lower reading monitor to local survey results

Determines D12-RM-K601B value of ***224*** *is ≤0.75 of the local survey results (****225****), and therefore, the channel check is unsatisfactory.*

****CRITICAL STEP** SAT/UNSAT**

TERMINATING CUE:	When the results of the survey have been compared to D12-RM-K601B and the evaluation of the channel check has been made, this JPM is complete for **RO** candidates.

TIME COMPLETED: ____________

SRO Candidates ONLY:

Step 8 – Determines the deviation is non-conservative, and instrument is declared inoperable.

Determines that ODCM 7.3.2 Condition A is entered immediately, and ODCM Condition I requires the following actions:

1. *Gaseous Radwaste Treatment System is immediately verified not bypassed*
2. *Main stack effluent noble gas monitor is immediately verified operable*
3. *Grab sample taken once within 72 hours and every 4 hours thereafter and analyzed to verify that the noble gas gross gamma activity rate is ≤ 243,600 μCi/second*
4. *Channel restored to operable within 30 days*

****CRITICAL STEP** SAT/UNSAT**

TERMINATING CUE: When evaluation of the channel check has been made, and ODCM action statements are determined, this JPM is complete for SRO candidates.

TIME COMPLETED ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	Documentation of data in block
2	Not Critical	Math not documented
3	Critical	Error would prevent correct channel check
4	Not Critical	From task conditions
5	Not Critical	Documentation of data in block
6	Not Critical	Math not documented
7	Critical	Error would prevent correct channel check
8	Critical	Correct ODCM actions required

REVISION SUMMARY

0	New JPM

(Provide sufficient detail for reviewers and evaluators to understand the scope of any technical and/or administrative changes).

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ________________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ________________________________

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ________________________ Date: __________

TASK CONDITIONS:

1. Readings for D12-RM-K601A, *SJAE Off Gas Rad Monitor A*, and D12-RM-K601B, *SJAE Off Gas Rad Monitor B,* have been recorded on the Unit 2 Dayshift RODSR for Saturday 0630-1230.
2. Main Condenser Air Ejector is in operation.
3. HP has reported a local survey reading of 300 mR

INITIATING CUE:

RO, and SRO candidates:

You are directed by the Control Room Supervisor to complete item 108, *SJAE Off-Gas Radiation Monitors Channel Check*, of 2OI-03.2, *Reactor Operator Daily Surveillance Report,* and state the status of the channel check.

- Channel check is SAT
- Channel check is UNSAT

SRO ONLY:

Based on the above information, determine the required actions, if any.

____

____

____

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

LESSON TITLE: DETERMINE PRIMARY CONTAINMENT WATER LEVEL AND EVALUATE PCPL-A

LESSON NUMBER: LOT-ADM-JP-300-B00

REVISION NO: 4

Daniel Hulgin 09/6/16
PREPARER / DATE

Bob Bolin 09/6/16
TECHNICAL REVIEWER / DATE

Hunter Morris 09/06/16
Kyle Cooper 09/06/16
VALIDATOR / DATE

Craig Olivier 09/27/2016
LINE SUPERVISOR / DATE

9-28-16
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

200602B501
Determine Primary Containment water level per EOP-01-UG

K/A REFERENCE AND IMPORTANCE RATING:

GEN 2.1.7 4.4/4.7
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

REFERENCES:

0EOP-01-UG
0AOP-36.1

TOOLS AND EQUIPMENT:

Student may use calculator

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Administrative – Conduct Of Operations

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. Notify SM/CRS of JPM performance prior to commencing In-plant JPM.
2. Determine actual radiological conditions and potentially contaminated areas to achieve ALARA.
3. Ensure all electrical safety requirements are observed.
4. Review Work Practices section prior to conduct of the JPM.
5. DO NOT OPERATE any plant equipment during performance of this JPM.

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM will be performed Unit 2
4. This is an administrative JPM designed to be administered in any setting and may be administered to multiple candidates simultaneously in a classroom setting
5. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. An accident is in progress on Unit Two. The Unit CRS is directing actions of EOP-01-RVCP and EOP-02-PCCP.
2. 480 VAC Substation E7 is de-energized due to a fault. All other electrical buses are energized.
3. ERFIS is unavailable
4. See Attachment 1 for the Containment parameter readings that are available on the RTGB.

INITIATING CUE:

You are directed to determine Primary Containment water level per EOP-01-UG, Attachment 36. Determine the current region of operation (Safe/Unsafe) on Primary Containment Pressure Limit A (PCPL-A)

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

TIME START: ___________

Step 1 – Determine suppression pool water level instruments cannot be used to determine primary containment water level since CAC-LI-2601-1 is above +2 feet and CAC-LR-2602 is powered from E7 and is de-energized.

Determine suppression pool water level instruments cannot be used to determine primary containment water.

SAT/UNSAT

Step 2 – Determine suppression chamber pressure instruments cannot be used to determine primary containment water level since CAC-PI-1257-2B is not less than 75 psig and CAC-PI-1257-2A is powered from E7 and is de-energized

Determine suppression chamber pressure instruments cannot be used to determine primary containment water.

SAT/UNSAT

Step 3 – Determine primary containment water level should be calculated using CAC-PI-1230 and CAC-PI-4176 since both instruments have power and suppression chamber pressure is not less than 75 psig, determine CAC-PR-1257-1 is powered from E7 and should not be used.

Determine primary containment water level should be calculated using CAC-PI-1230 and CAC-PI-4176.

SAT/UNSAT

Step 4 – Calculate primary containment water level to be 2.3 ft/psi (72.5 – 67.5) + 28.5 ft.

Primary containment water level calculated to be 40 feet.

****CRITICAL STEP** SAT/UNSAT**

Step 5 – Determine operation to be in the safe region of PCPL-A using the PCPL-A graph, calculated primary containment water level and CAC-PI-4176 for drywell pressure, (or by using CAC-PI-1230 reading <70 psig).

Determine *PCPL-A is in the Safe region*

****CRITICAL STEP** SAT/UNSAT**

TERMINATING CUE: When primary containment water level is calculated, and PCPL-A is determined to be in the Safe region, this JPM is complete

TIME COMPLETED: ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	JPM can still be completed without performing this step.
2	Not Critical	JPM can still be completed without performing this step.
3	Not Critical	JPM can still be completed without performing this step.
4	Critical	Calculation required to complete this JPM.
5	Critical	Determination required to complete this JPM.

REVISION SUMMARY

4	New template incorporated. Modified torus press o read slightly >75 psig in att 1 Removed take a minute (step 1) reordered steps.
3	Changed Unit SCO to Unit CRS. No technical changes.
2	Revised to new JPM Template, Revision 3. No technical changes

(Provide sufficient detail for reviewers and evaluators to understand the scope of any technical and/or administrative changes).

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ________________________________

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ________________________________

☐ Comments reviewed with Performer

Evaluator Signature: ________________________ Date: ____________

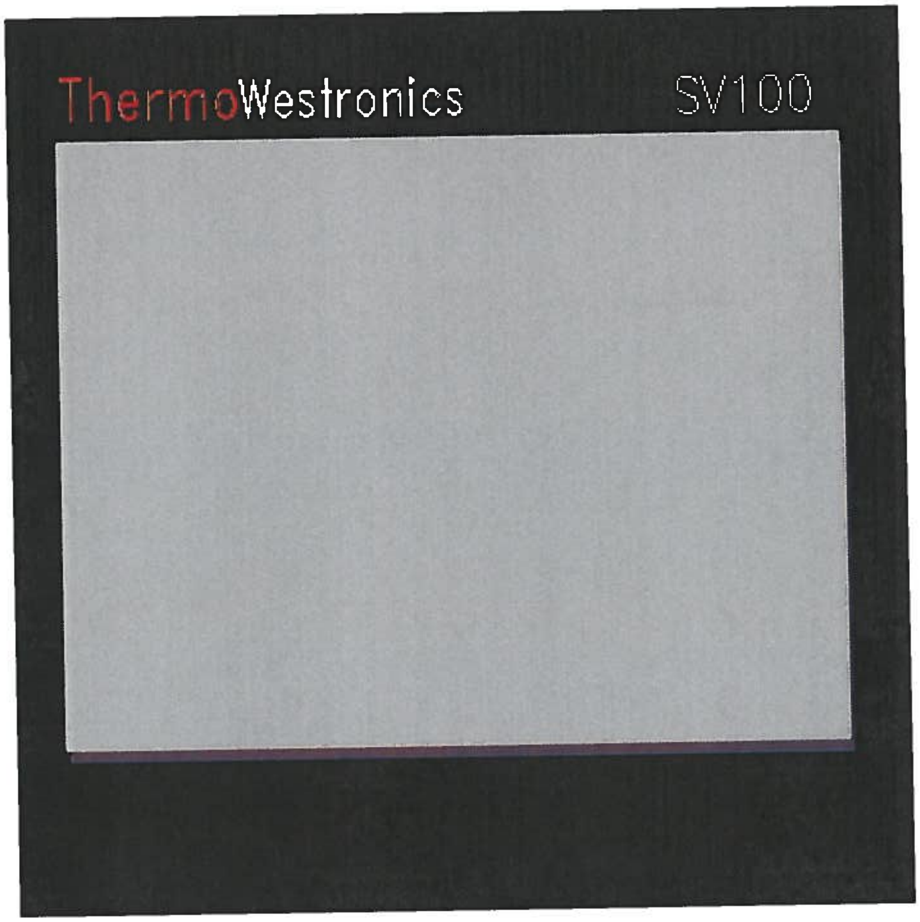
TASK CONDITIONS:

1. An accident is in progress on Unit Two. The Unit CRS is directing actions of EOP-01-RVCP and EOP-02-PCCP.
2. 480 VAC Substation E7 is de-energized due to a fault. All other electrical buses are energized.
3. ERFIS is unavailable.
4. See Attachment 1 for the Containment parameter readings that are available on the RTGB.

INITIATING CUE:

You are directed to determine Primary Containment water level per EOP-01-UG, Attachment 36. Determine the current region of operation (Safe/Unsafe) on Primary Containment Pressure Limit A (PCPL-A).

Attachment 1



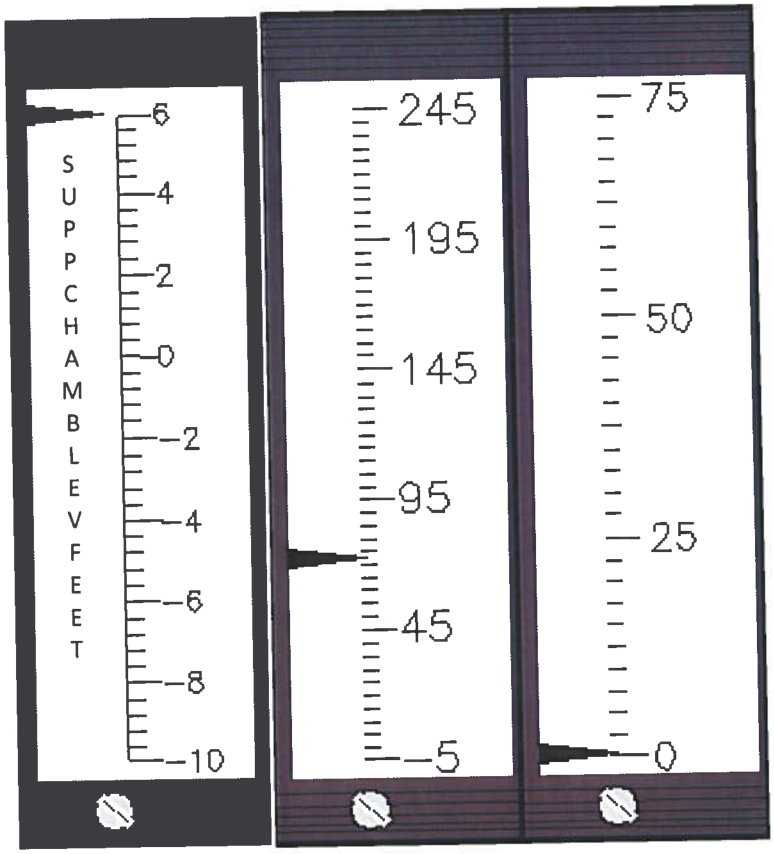
SUPRSSION
POOL
LEVEL
CAC-LR-2602
E3-E7

Attachment 1

DRYWELL PRESSURE

2-CAC-PR-1257-1

Attachment 1

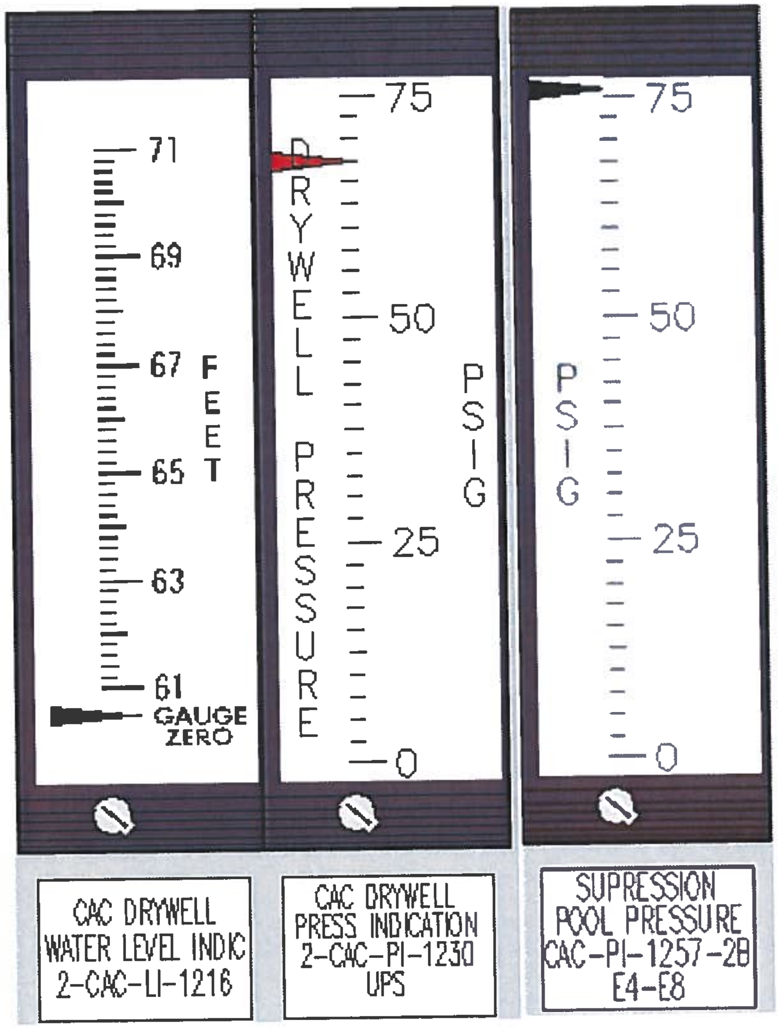


SUPP POOL LEVEL CAC-LI-2601-1 E4-E8

DRYWELL PRESSURE CAC-PI-4176 E4-E8

SUPP POOL PRESSURE CAC-PI-1257-2A E3-E7

Attachment 1



DUKE ENERGY

BRUNSWICK TRAINING SECTION

JOB PERFORMANCE MEASURE

LESSON TITLE: **CALCULATE DRYWELL LEAKAGE RATE**

LESSON NUMBER: **LOT-ADM-JP-201-D16**

REVISION NO: **0**

Daniel Hulgin 09/06/16
PREPARER / DATE

Bob Bolin 09/06/16
TECHNICAL REVIEWER / DATE

Hunter Morris 09/06/16
Kyle Cooper 09/06/16
Dwayne Wolf 09/06/16
VALIDATOR / DATE

09/27/2016
LINE SUPERVISOR / DATE

9-28-2016
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

K/A REFERENCE AND IMPORTANCE RATING:

GEN 2.2.12 3.7/4.1
Knowledge of surveillance procedures.

REFERENCES:

2OI-03.2
T.S 3.4.4

TOOLS AND EQUIPMENT:

Student may use calculator

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Generic (Administrative)

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. This task will be performed on Unit 2.
2. Unit 2 is in MODE 1.
3. The Equipment Drain Sump was manually pumped to an integrator reading of 457620, and the pump stopped at 2000 on Sunday Nightshift.
4. The Floor Drain Sump was manually pumped to an integrator reading of 13944891, and the pump stopped at 2000 on Sunday Nightshift.

INITIATING CUE:

RO, and SRO candidates:

You are directed by the Control Room Supervisor to determine the 24 hour leak rate for the equipment and floor drains, and the 24 hour total leak rate to the drywell IAW Attachment 1, *Drywell Leakage Calculation*, of 2OI-03.2, *Reactor Operator Daily Surveillance Report*, for Sunday Nightshift at time 2000.

SRO ONLY:

State if Tech Spec LCO 3.4.4 is or is NOT met, and If the LCO is NOT met, identify the **LATEST** time Unit 2 is required to be in MODE 3.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

TIME START: ___________

Step 1 – Calculate time interval from equipment drain manual pump at 2000 on Saturday.

Subtracts time equipment drain sump pump stops on 2000 Sunday (20) from time equipment drain sump pump stops on 2000 Saturday (20) for a value of 1440 minutes .

****CRITICAL STEP** SAT/UNSAT**

Step 2 – Calculate difference in equipment drain integrator reading from manual pump at 2000 on Saturday.

Subtracts 2000 Saturday equipment drain integrator reading (423780) from 2000 Sunday equipment drain integrator reading (457620) for a value of 33840 gal .

****CRITICAL STEP** SAT/UNSAT**

Step 3 – Calculate 24 hour equipment drain leak rate .

Divides leakage (value from step 3) by time interval (value from step 2) for a value of 23.5 gpm.

****CRITICAL STEP** SAT/UNSAT**

Step 4 – Calculate time interval from floor drain manual pump at 2000 on Saturday.

Subtracts time floor drain sump pump stops on 2000 Sunday (20) from time floor drain sump pump stops on 2000 Saturday (20) for a value of 1440 minutes .

****CRITICAL STEP** SAT/UNSAT**

Step 5 – Calculate difference in floor drain integrator reading from manual pump at 2000 on Saturday.

Subtracts 2000 Saturday floor drain integrator reading (13942587) from 2000 Sunday floor drain integrator reading (13944891) for a value of 2304 gal .

****CRITICAL STEP** SAT/UNSAT**

Step 6 – Calculate 24 hour equipment drain leak rate .

Divides leakage (value from step 6) by time interval (value from step 5) for a value of 1.6 gpm.

****CRITICAL STEP** SAT/UNSAT**

Step 7 – Calculate 24 hour total leak rate to drywell.

Adds value from step 4 and step 7 for a value of 25.1 gpm.

****CRITICAL STEP** SAT/UNSAT**

TERMINATING CUE:	When the results for the 24 hour floor drain, equipment drain, and 24 hour total leak rate to drywell have been recorded, this JPM is complete for RO candidates.

TIME COMPLETED: ____________

SRO Candidates ONLY:

Step 8 – State if Tech Spec LCO 3.4.4 is or is NOT met, and If the LCO is NOT met, identify the latest time Unit 2 is required to be in MODE 3.

Determines that the 24 hour total leak rate has exceeded the T.S. 3.4.4 limit, and entry into MODE 3 would be required no later than 1600 on Monday.

****CRITICAL STEP** SAT/UNSAT**

TERMINATING CUE: When evaluation of the T.S. has been made, and the earliest time to MODE 3 has been determined, this JPM is complete for SRO candidates.

TIME COMPLETED _____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Critical	Math Critical for JPM solution
2	Critical	Math Critical for JPM solution
3	Critical	Math Critical for JPM solution
4	Critical	Math Critical for JPM solution
5	Critical	Math Critical for JPM solution
6	Critical	Math Critical for JPM solution
7	Critical	Math Critical for JPM solution
8	Critical	TS determination is Critical

REVISION SUMMARY

0	New JPM

(Provide sufficient detail for reviewers and evaluators to understand the scope of any technical and/or administrative changes).

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ______________________________________

JPM: Pass _____ Fail _____

Remedial Training Required: Yes _____ No _____

Comments: ______________________________________

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: __________

EXAM KEY DO NOT GIVE TO STUDENTS

ATTACHMENT 1
Page 14 of 60
Reactor Operator Daily Surveillance Report (RODSR) – Unit 2
DRYWELL LEAKAGE CALCULATION

	20	00	04
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP)	20		
CALCULATE TIME INTERVAL (MINUTES) FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY	1440		
RECORD CURRENT INTEGRATOR READING	0 0 4 5 7 6 2 0		
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY (LEAKAGE)	33840		
CALCULATE 24 HOUR EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL)	23.5		
MANUALLY PUMP FLOOR DRAIN SUMP USING ONE PUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP)	20		
CALCULATE TIME INTERVAL (MINUTES) FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY	1440		
RECORD INTEGRATOR READING	1 3 9 4 4 8 9 1		
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY (LEAKAGE)	2304		
CALCULATE 24 HOUR FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL)	1.60		
24 HOUR EQUIPMENT DRAIN LEAK RATE	23.50		
24 HOUR FLOOR DRAIN LEAK RATE *	1.6		
24 HOUR TOTAL LEAK RATE TO DRYWELL **	25.1		
CHECK TECH SPEC 3.4.4 LEAKAGE LIMITS MET	unsat		

EXAM KEY DO NOT GIVE TO STUDENTS

TASK CONDITIONS:

1. This task will be performed on Unit 2.
2. Unit 2 is in MODE 1.
3. The Equipment Drain Sump was manually pumped to an integrator reading of 457620, and the pump stopped at 2000 on Sunday Nightshift.
4. The Floor Drain Sump was manually pumped to an integrator reading of 13944891, and the pump stopped at 2000 on Sunday Nightshift.

INITIATING CUE:

RO, and SRO candidates:

You are directed by the Control Room Supervisor to determine the 24 hour leak rate for the equipment and floor drains, and the 24 hour total leak rate to the drywell IAW Attachment 1, *Drywell Leakage Calculation*, of 2OI-03.2, *Reactor Operator Daily Surveillance Report*, for Sunday Nightshift at time 2000.

SRO ONLY:

State if Tech Spec LCO 3.4.4 is or is NOT met, and If the LCO is NOT met, identify the **LATEST** time Unit 2 is required to be in MODE 3.

Reactor Operator Daily Surveillance Report (RODSR) – Unit 2
DRYWELL LEAKAGE CALCULATION

	20	00	04
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME.PUMP STOPS (LOW LEVEL TRIP).	20	00	04
CALCULATE TIME INTERVAL (MINUTES) FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY.	1440	1440	1440
RECORD CURRENT INTEGRATOR READING.	0 0 4 2 3 7 8 0	0 0 4 2 9 1 8 1	0 0 4 3 4 6 6 6
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY (LEAKAGE).	30240	30384	30672
CALCULATE 24 HOUR EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).	21	21.1	21.3
MANUALLY PUMP FLOOR DRAIN SUMP USING ONE PUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP)	20	00	04
CALCULATE TIME INTERVAL (MINUTES) FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY.	1440	1440	1440
RECORD INTEGRATOR READING.	1 3 9 4 2 5 8 7	1 3 9 4 1 9 9 8	1 3 9 4 2 2 0 2
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY (LEAKAGE).	720	851	924
CALCULATE 24 HOUR FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).	0.5	0.59	0.64
24 HOUR EQUIPMENT DRAIN LEAK RATE.	21	21.1	21.3
24 HOUR FLOOR DRAIN LEAK RATE. *	0.5	0.59	0.64
24 HOUR TOTAL LEAK RATE TO DRYWELL. **	21.5	21.69	21.94
CHECK TECH SPEC 3.4.4 LEAKAGE LIMITS MET	V	V	V

Sump leak calculations required in Modes 1, 2, and 3 - reference - SR 3.4.4.1 and AOP-14.0.

* If floor drain leak rate exceeds 5 gpm averaged over the previous 24-hour period or if in Mode 1, increases by 2 gpm within the previous 24 hour period, enter Tech Spec 3.4.4 and 0AOP-14.0.

** If total leakage rate exceeds 25 gpm averaged over the previous 24-hour period, enter Tech Spec 3.4.4.

SHIFT Saturday Nightshift

Week Beginning December 10, 2016

Reactor Operator Daily Surveillance Report (RODSR) – Unit 2
DRYWELL LEAKAGE CALCULATION

	08	12	16
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME.PUMP STOPS (LOW LEVEL TRIP).	08	12	16
CALCULATE TIME INTERVAL (MINUTES) FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY.	1440	1440	1440
RECORD CURRENT INTEGRATOR READING.	0 0 4 4 0 2 5 9	0 0 4 4 5 8 5 7	0 0 4 5 1 4 9 3
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY (LEAKAGE).	31248	31968	32832
CALCULATE 24 HOUR EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).	21.7	22.2	22.8
MANUALLY PUMP FLOOR DRAIN SUMP USING ONE PUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP)	08	12	16
CALCULATE TIME INTERVAL (MINUTES) FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY.	1440	1440	1440
RECORD INTEGRATOR READING.	1 3 9 4 2 4 9 4	1 3 9 4 2 9 0 5	1 3 9 4 3 5 1 1
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY (LEAKAGE).	1012	1131	1326
CALCULATE 24 HOUR FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).	0.70	0.79	0.92
24 HOUR EQUIPMENT DRAIN LEAK RATE.	21.7	22.2	22.8
24 HOUR FLOOR DRAIN LEAK RATE. *	0.70	0.79	0.92
24 HOUR TOTAL LEAK RATE TO DRYWELL. **	22.4	22.99	23.72
CHECK TECH SPEC 3.4.4 LEAKAGE LIMITS MET	V	V	V

Sump leak calculations required in Modes 1, 2, and 3 - reference - SR 3.4.4.1 and AOP-14.0.

* If floor drain leak rate exceeds 5 gpm averaged over the previous 24-hour period or if in Mode 1, increases by 2 gpm within the previous 24 hour period, enter Tech Spec 3.4.4 and 0AOP-14.0.

** If total leakage rate exceeds 25 gpm averaged over the previous 24-hour period, enter Tech Spec 3.4.4.

SHIFT Sunday Dayshift

Week Beginning December 10, 2016

Reactor Operator Daily Surveillance Report (RODSR) – Unit 2
DRYWELL LEAKAGE CALCULATION

	20	00	04
MANUALLY PUMP EQUIPMENT DRAIN SUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP).			
CALCULATE TIME INTERVAL (MINUTES) FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY.			
RECORD CURRENT INTEGRATOR READING.			
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY (LEAKAGE).			
CALCULATE 24 HOUR EQUIPMENT DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).			
MANUALLY PUMP FLOOR DRAIN SUMP USING ONE PUMP - RECORD TIME PUMP STOPS (LOW LEVEL TRIP)			
CALCULATE TIME INTERVAL (MINUTES) FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY.			
RECORD INTEGRATOR READING.			
CALCULATE DIFFERENCE IN INTEGRATOR READING FROM MANUAL PUMP AT SAME TIME ON PREVIOUS DAY (LEAKAGE).			
CALCULATE 24 HOUR FLOOR DRAIN LEAK RATE (DIVIDE LEAKAGE BY TIME INTERVAL).			
24 HOUR EQUIPMENT DRAIN LEAK RATE.			
24 HOUR FLOOR DRAIN LEAK RATE. *			
24 HOUR TOTAL LEAK RATE TO DRYWELL. **			
CHECK TECH SPEC 3.4.4 LEAKAGE LIMITS MET			

Sump leak calculations required in Modes 1, 2, and 3 - reference - SR 3.4.4.1 and AOP-14.0.

* If floor drain leak rate exceeds 5 gpm averaged over the previous 24-hour period or if in Mode 1, increases by 2 gpm within the previous 24 hour period, enter Tech Spec 3.4.4 and 0AOP-14.0.

** If total leakage rate exceeds 25 gpm averaged over the previous 24-hour period, enter Tech Spec 3.4.4.

SHIFT Sunday Nightshift

Week Beginning December 10, 2016

DUKE ENERGY

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

LESSON TITLE: Determine Stay Time Limitations in High Radiation Areas

LESSON NUMBER: LOT-ADM-JP-102-A03

REVISION NO: 3

Daniel Hulgin 09/06/16
PREPARER / DATE

Bob Bolin 09/06/16
TECHNICAL REVIEWER / DATE

Hunter Morris 09/06/16
Kyle Cooper 09/06/16
VALIDATOR / DATE

09/27/2016
LINE SUPERVISOR / DATE

9-28-16
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

None

K/A REFERENCE AND IMPORTANCE RATING:

Generic 2.3.4 3.2/3.7
Knowledge of Radiation Exposure Limits under normal or emergency conditions
Generic 2.3.7 3.5/3/6
Ability to comply with radiation work permit requirements during normal and abnormal conditions

REFERENCES:

PD-RP-ALL-0001, Radiation Worker Responsibilities

TOOLS AND EQUIPMENT:

Calculator
Radiation Survey Map of 50' Reactor Building

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

A.3 Radiation Control

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL NOT** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. This JPM may be performed on Unit 1 or Unit 2 as selected by the evaluator. Survey map must reflect correct unit.
4. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

Two workers will be performing a lube check and coupling alignment on the Unit 2 RWCU Pump 2A.

Worker #1 has accumulated 800 mrem this year.
Worker #2 has accumulated 970 mrem this year.
The elevator is out of service
The following times for each worker have been estimated for performance of the job.

1. Traversing Southeast stairwell 20' – 50' Rx Bldg: 6 minutes
2. Staging time in access area directly outside the RWCU room: 45 minutes
3. Staging time in area directly inside room access door: 20 minutes
4. Work time at the "A" RWCU pump: 2.5 hours
5. Following completion of the job, an additional 60 mrem per worker will be received during de-staging activities and transit back to the maintenance shop.

INITIATING CUE:

Using the information above and the provided radiological survey using best ALARA practices:

1. Determine the total dose accumulated for both workers. (Assume the same task times for both workers).
2. Determine if any Brunswick administrative dose limitations will be exceeded.

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

TIME START ___________

Step 1 - Determines dose for each worker as follows:

a. *Traversing SE stairwell 20' – 50' Rx Bldg (SE is the lowest dose stairwell)*
(6 min) 0.1 Hr X 5 mr/hr = ***0.5 mrem***
Estimate 0.5 mrem dose accumulation

****CRITICAL STEP**　SAT/UNSAT**

b. *Staging time in access area directly outside the RWCU room*
(45 min) 0.75 Hr X 20 mr/hr = ***15 mrem***
Estimate 15 mrem dose accumulation

****CRITICAL STEP**　SAT/UNSAT**

c. *Staging time in area directly inside room access door*
(20 min) 0.33 Hr X 80 mr/hr = ***26.7 mrem***
Estimate 26.7 mrem dose accumulation.

****CRITICAL STEP**　SAT/UNSAT**

d. *Work time at the "A" RWCU pump*
2.5 Hrs X 200 mr/hr = ***500 mrem***
Estimate 500 millirem dose accumulation

****CRITICAL STEP**　SAT/UNSAT**

NOTE: *An additional 60 mr will be accumulated once the job is done for de-staging activities.*

e. *Total = 0.5 + 15 + 26.7 + 500 + 60 = 602.2* ***mrem***

****CRITICAL STEP**　SAT/UNSAT**

Step 2 - Determines that neither worker would exceed the Brunswick administrative limit of 2 REM per calendar year if the estimated dose were accumulated.

Worker #1: 800 mr + 602.2 mr = 1402.2 mr (< 2R limit)
Worker #2: 970 mr + 602.2 mr = 1572.2 mr (< 2R limit)

****CRITICAL STEP**   SAT/UNSAT**

TERMINATING CUE: When the total dose for each worker has been determined and the administrative limits addressed, the JPM is complete.

TIME COMPLETED: ________

NOTE: Comments required for any step evaluated as UNSAT.

Step	Critical / Not Critical	Reason
1a	Critical	Each calculation is critical to determine total dose for personnel safety.
1b	Critical	Each calculation is critical to determine total dose.
1c	Critical	Each calculation is critical to determine total dose.
1d	Critical	Each calculation is critical to determine total dose.
1e	Critical	Each calculation is critical to determine total dose.
2	Critical	Total calculation and knowledge of Admin Dose Limit is required to complete JPM.

REVISION SUMMARY

3	Removed take a minute-step 1. Reordered steps
2	Revised to new JPM Template Revised times so that calculations are different than previous versions.
1	Revised to new JPM Template, Revision 3. No technical changes.

Validation Time: 15 Minutes (approximate)

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	X	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	____	No	X	Time Limit	N/A
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ______________________________

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ______________________________

☐ Comments reviewed with Performer

Evaluator Signature: ______________________ Date: __________

TASK CONDITIONS:

Two workers will be performing a lube check and coupling alignment on the Unit 2 RWCU Pump 2A.

Worker #1 has accumulated 800 mrem this year.
Worker #2 has accumulated 970 mrem this year.
The elevator is out of service
The following times for each worker have been estimated for performance of the job.

1.	Traversing Southeast stairwell 20' – 50' Rx Bldg:	6 minutes
2.	Staging time in access area directly outside the RWCU room:	45 minutes
3.	Staging time in area directly inside room access door:	20 minutes
4.	Work time at the "A" RWCU pump:	2.5 hours
5.	Following completion of the job, an additional 60 mrem per worker will be received during de-staging activities and transit back to the maintenance shop.	

INITIATING CUE:

Using the information above and the provided radiological survey using best ALARA practices:

1. Determine the total dose accumulated for both workers. (Assume the same task times for both workers).
2. Determine if any Brunswick administrative dose limitations will be exceeded.

Results: ______________________________

DUKE ENERGY
BRUNSWICK TRAINING SECTION
JOB PERFORMANCE MEASURE

LESSON TITLE: CLASSIFY AN EMERGENCY PER PEP-02.1.

LESSON NUMBER: SOT-ADM-JP-301-A16

REVISION NO: 1

Daniel Hulgin 09/06/16
PREPARER / DATE

Bob Bolin 09/06/16
TECHNICAL REVIEWER / DATE

Dwayne Wolf 09/06/16
Kyle Cooper 09/06/16
VALIDATOR / DATE

Amy Oliver 09/27/2016
LINE SUPERVISOR / DATE

9-28-16
TRAINING SUPERVISION APPROVAL / DATE

RELATED TASKS:

344256B502
Direct initial emergency actions including emergency classification per 0PEP-02.1

K/A REFERENCE AND IMPORTANCE RATING:

GEN 2.4.29 3.1/4.4
Knowledge of the Emergency Plan

REFERENCES:

0PEP-02.1

TOOLS AND EQUIPMENT:

None

SAFETY FUNCTION (from NUREG 1123, Rev. 2, Supp. 1):

Admin – Emergency Procedures / Plan

SETUP INSTRUCTIONS

None

SAFETY CONSIDERATIONS:

1. None

EVALUATOR NOTES: (Do not read to performer)

1. The applicable procedure section **WILL** be provided to the trainee.
2. Prior to the first JPM of the JPM set, provide the JPM briefing contained in NUREG-1021, Appendix E, or similar briefing (for non-regulated exams) to the trainee(s).
3. Critical Step Basis
   a) Prevents Task Completion
   b) May Result in Equipment Damage
   c) Affects Public Health and Safety
   d) Could Result in Personal Injury

Read the following to the JPM performer.

TASK CONDITIONS:

1. Unit One is operating at 100% power.
2. Unit Two is operating at 100% power with DG4 under clearance when the following event occurs (consider all items that exceed EAL thresholds occur at the same time).

Unit Two Event Description
• A seismic event greater than the Operating Basis Earthquake results in a loss of the PBX Telephone System, Commercial Telephones, and NRC Emergency Telecommunications System • A Manual Scram is inserted, the Mode Switch is in Shutdown, ARI is initiated, and reactor power indicates 20%. • Driving rods IAW LEP-02, *Alternate Control Rod Insertion,* and SLC injection are in progress. • Current indications: reactor power is 1%, reactor water level maintained +60 to +90 inches, and reactor pressure is 945 psig on EHC. • NO radiological releases in progress, and NO indications of an onsite security event.

INITIATING CUE:

You are to evaluate the above event as the Control Room Site Emergency Coordinator (SEC) and determine the **HIGHEST** required classification and its EAL Identifier for Unit Two ONLY:

1. Write the required Classification and its associated EAL identifier in the table below.
2. Raise your hand when complete to have the evaluator stop the evaluation time and collect your cue sheet

This JPM is TIME CRITICAL.

CLASSIFICATION	EAL IDENTIFIER(s)

PERFORMANCE CHECKLIST

NOTE: Sequence is assumed unless otherwise indicated, comments required for any step evaluated as UNSAT.

PROMPT: Ensure a clock is visible for candidates. Announce and Write the Start Time on the board. Add 15 minutes to the Start Time and write that in the JPM Completion Time. If all candidates have not Declared a classification, this is the time to STOP all work, put pencils/pens down, and collect all remaining cue

NOTE: Declaration of event must be made in 15 minutes from the Start Time.

TIME START: ___________

NOTE: Loss of PBX Telephone System, Commercial Telephones, and NRC Emergency Telecommunications System does not reach EAL classification threshold

Step 1 – Determine required Classification threshold and associated EAL Number(s).

- *Unusual Event – HU2.1*
  -Seismic event > OBE per 0AOP-13.0

SAT/UNSAT

NOTE: Candidate may base the ALERT on SA8.1 if they determine the ATWS was a result of the earthquake. Critical Step is that the ALERT is based on EITHER SA8.1 or SA6.1 **EITHER Step 2 OR 3 is critical.**

Step 2 – Determine required Classification threshold and associated EAL Number(s).

- *Alert – SA8.1*
  -The occurrence of any Table S-4 hazardous event (Seismic event).
  AND EITHER
  -Event damage has caused indications of degraded performance in at least one train of a safety system needed for the current operating mode (ATWS).
  -The event has caused visible damage to a safety system component or structure needed for the current operating mode.

****CRITICAL STEP** SAT/UNSAT**

Step 3 – Determine required Classification threshold and associated EAL Number(s).

- *Alert – SA6.1*
  -An automatic or manual scram fails to reduce reactor power <2% (APRM downscale).
  AND
  -Manual scram actions taken at the reactor control console (Manual PBs, Mode Switch, ARI) are not successful in shutting down the reactor as indicated by reactor power ≥ 2% (note 8)
  Note 8: A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving control rods or boron injection strategies.

****CRITICAL STEP** SAT/UNSAT**

Step 4 – Classification made within required the required time (Declaration Time minus Start Time < 15 minutes).

- *Classification declared < 15 minutes of Start Time.*

****CRITICAL STEP** SAT/UNSAT**

TERMINATING CUE: When the event is classified with applicable EAL identifier(s) in the table, this JPM is complete.

TIME COMPLETED: ____________

COLLECT AND CONTROL ALL JPM EXAM MATERIALS FOR EXAM SECURITY.

Step	Critical / Not Critical	Reason
1	Not Critical	This EAL is not the highest classification
2 or 3	Critical	Highest EAL classification and EAL designator
4	Critical	Time to declare is critical

REVISION SUMMARY

1	Incorporated new format. Modified initial conditions as follows: • Changed earthquake magnitude to OBE per AOP13 • Changed Voicenet to PBX • Removed turbine trip • Changed reactor power to 1% following SLC and driving rods Changed initiating cue to include EAL identifier(s) instead of EAL identifier Changed step 1 to reflect new standard instead of obtaining a procedure Step 2 EAL criteria updated to new PEP2.1 criteria Step 3 EAL criteria updated to new PEP2.1 criteria. This is now a critical step. Its EAL designation is now part of the highest classification Step 4 3 EAL criteria updated to new PEP2.1 criteria. This is now an ALERT and no longer a SAE Removed take a minute (step 1) and reordered steps.
0	New JPM

(Provide sufficient detail for reviewers and evaluators to understand the scope of any technical and/or administrative changes).

Validation Time: 15 Minutes (approximate).

Time Taken: ____ Minutes

APPLICABLE METHOD OF TESTING

Performance:	Simulate	____	Actual	X	Unit:	2
Setting:	In-Plant	____	Simulator	____	Admin	X
Time Critical:	Yes	X	No	____	Time Limit	15 min
Alternate Path:	Yes	____	No	X		

EVALUATION

Performer: ____

JPM: Pass ______ Fail ______

Remedial Training Required: Yes ______ No ______

Comments: ____

____

____

____

____

____

____

☐ Comments reviewed with Performer

Evaluator Signature: ______________________________ Date: ____________

This JPM is TIME CRITICAL.

TASK CONDITIONS:

1. Unit One is operating at 100% power.
2. Unit Two is operating at 100% power with DG4 under clearance when the following event occurs (consider all items that exceed EAL thresholds occur at the same time).

Unit Two Event Description
• A seismic event greater than the Operating Basis Earthquake results in a loss of the PBX Telephone System, Commercial Telephones, and NRC Emergency Telecommunications System • A Manual Scram is inserted, the Mode Switch is in Shutdown, ARI is initiated, and reactor power indicates 20%. • Driving rods IAW LEP-02, *Alternate Control Rod Insertion,* and SLC injection are in progress. • Current indications: reactor power is 1%, reactor water level maintained +60 to +90 inches, and reactor pressure is 945 psig on EHC. • NO radiological releases in progress, and NO indications of an onsite security event.

INITIATING CUE:

You are to evaluate the above event as the Control Room Site Emergency Coordinator (SEC) and determine the **HIGHEST** required classification and its EAL Identifier for Unit Two ONLY:

1. Write the required Classification and its associated EAL identifier in the table below.
2. Raise your hand when complete to have the evaluator stop the evaluation time and collect your cue sheet

CLASSIFICATION	EAL IDENTIFIER

This JPM is TIME CRITICAL.