ML21028A339

DRAFT

Facility: HARRIS **Date of Exam:** NOVEMBER 2020

Tier	Group	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
		RO K/A Category Points												**SRO-Only Points**		
1. Emergency & Abnormal Plant Evolutions	1	✓3	✓3	✓3	N/A			✓3	✓3	N/A		✓3	✓18	✓3	✓3	✓6
	2	✓1	✓1	✓2				✓2	✓2			✓1	✓9	✓2	✓2	✓4
	Tier Totals	✓4	✓4	✓5				✓5	✓5			✓4	✓27	✓5	✓5	✓10
2. Plant Systems	1	✓3	✓2	✓3	✓2	✓2	✓3	✓3	✓2	✓3	✓3	✓2	✓28	2 ✓	3 ✓	✓5
	2	✓0	✓1	✓1	✓1	✓1	✓1	✓1	✓1	✓1	✓1	✓1	✓10	ME ~~2~~ 2	1 ✓	✓3
	Tier Totals	3	✓3	✓4	✓3	✓3	✓4	✓4	✓3	✓4	✓4	✓3	✓38	4 ✓	✓4	✓8

3. Generic Knowledge and Abilities Categories	1	2	3	4	10	1	2	3	4	7
	✓3	✓2	✓2	✓3		✓1	✓2	✓2	✓2	✓

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, 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 and justified; operationally important, site-specific systems 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 KAs.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics= importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note # 1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43..

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T1G1 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
007EK1.05	Reactor Trip - Stabilization - Recovery / 1	3.3	3.8	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Decay power as a function of time
008AA1.03	Pressurizer Vapor Space Accident / 3	2.8	2.6	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Turbine bypass in manual control to maintain header pressure
009EK3.02	Small Break LOCA / 3	2.8	3.2	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Opening excess letdown isolation valve
011EA2.13	Large Break LOCA / 3	3.7	3.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Difference between overcooling and LOCA indications
015AA2.11	RCP Malfunctions / 4	3.4	3.8	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	When to jog RCPs during ICC
022AK1.01	Loss of Rx Coolant Makeup / 2	2.8	3.2	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Consequences of thermal shock to RCP seals
025AA1.12	Loss of RHR System / 4	3.6	3.5	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	RCS temperature indicators
026AG2.1.23	Loss of Component Cooling Water / 8	4.3	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to perform specific system and integrated plant procedures during all modes of plant operation.
027AK2.03	Pressurizer Pressure Control System Malfunction / 3	2.6	2.8	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Controllers and positioners
029EK2.06	ATWS / 1	2.9	3.1	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Breakers, relays, and disconnects.
038EK1.03	Steam Gen. Tube Rupture / 3	3.9	4.2	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Natural circulation

T1G1 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
054AK3.04	Loss of Main Feedwater / 4	4.4	4.6	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Actions contained in EOPs for loss of MFW
056AA2.44	Loss of Off-site Power / 6	4.3	4.5	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Indications of loss of offsite power
057AG2.2.38	Loss of Vital AC Inst. Bus / 6	3.6	4.5	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of conditions and limitations in the facility license.
062AK3.03	Loss of Nuclear Svc Water / 4	4	4.2	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Guidance actions contained in EOP for Loss of nuclear service water
065AA1.04	Loss of Instrument Air / 8	3.5	3.4	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Emergency air compressor
WE05EK2.2	Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	3.9	4.2	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems and relations between the proper operation of these systems to the operation of the facility.
we12EG2.1.2	Steam Line Rupture - Excessive Heat Transfer / 4	4.6	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to execute procedure steps.
				3	3	3	0	0	0	3	3	0	0	3	

T1G2 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
003AK3.05	Dropped Control Rod / 1	3.4	4.1	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Tech-Spec limits for reduction of load to 50% power if flux cannot be brought back within specified target band
024AA2.06	Emergency Boration / 1	3.6	3.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	When boron dilution is taking place
059AK3.04	Accidental Liquid RadWaste Rel. / 9	3.8	4.3	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Actions contained in EOP for accidental liquid radioactive-waste release
061AA1.01	ARM System Alarms / 7	3.6	3.6	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Automatic actuation
069AA1.01	Loss of CTMT Integrity / 5	3.5	3.7	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Isolation valves, dampers and electropneumatic devices.
we02EG2.4.45	SI Termination / 3	4.1	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to prioritize and interpret the significance of each annunciator or alarm.
WE07EA2.1	Saturated Core Cooling Core Cooling / 4	3.2	4.0	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.
WE13EK2.1	Steam Generator Over-pressure / 4	3.0	3.1	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.
WE16EK1.2	High Containment Radiation / 9	2.7	3.2	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Normal, abnormal and emergency operating procedures associated with (High Containment Radiation).
				1	1	2	0	0	0	2	2	0	0	1	

T2G1 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
003K4.07	Reactor Coolant Pump	3.2	3.4	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Minimizing RCS leakage (mechanical seals)
004K5.44	Chemical and Volume Control	3.2	3.4	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Pressure response in PZR during in-and-out surge
005K2.03	Residual Heat Removal	2.7	2.8	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	RCS pressure boundary motor-operated valves
006A3.03	Emergency Core Cooling	4.1	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	ESFAS-operated valves
007K4.01	Pressurizer Relief/Quench Tank	2.6	2.9	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Quench tank cooling
008K1.05	Component Cooling Water	3.0	3.1	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Sources of makeup water
008K3.03	Component Cooling Water	4.1	4.2	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	RCP
010A3.01	Pressurizer Pressure Control	3.0	3.2	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	PRT temperature and pressure during PORV testing
012A1.01	Reactor Protection	2.9	3.4	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Trip setpoint adjustment
012K6.02	Reactor Protection	2.9	3.1	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Redundant channels
013K1.15	Engineered Safety Features Actuation	3.4	3.8	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	MFW System

T2G1 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
013K6.01	Engineered Safety Features Actuation	2.7	3.1	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Sensors and detectors
022K1.02	Containment Cooling	3.7	3.5	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	SEC/remote monitoring systems
026A1.03	Containment Spray	3.5	3.5	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Containment sump level
026K3.02	Containment Spray	4.2	4.3	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Recirculation spray system
039A3.02	Main and Reheat Steam	3.1	3.5	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Isolation of the MRSS
039G2.1.30	Main and Reheat Steam	4.4	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to locate and operate components, including local controls.
059A4.01	Main Feedwater	3.1	3.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	MFW turbine trip indication
061K2.03	Auxiliary/Emergency Feedwater	4.0	3.8	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	AFW diesel driven pump
061K5.05	Auxiliary/Emergency Feedwater	2.7	3.2	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Feed line voiding and water hammer
062A2.04	AC Electrical Distribution	3.4	3.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Effect on plant of de-energizing a bus
063A2.01	DC Electrical Distribution	2.5	3.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Grounds

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
063A4.03	DC Electrical Distribution	3.0	3.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	Battery discharge rate
064K6.07	Emergency Diesel Generator	2.7	2.9	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Air receivers
073A1.01	Process Radiation Monitoring	3.2	3.5	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Radiation levels
076G2.4.46	Service Water	4.2	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to verify that the alarms are consistent with the plant conditions.
078K3.01	Instrument Air	3.1	3.4	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Containment air system
103A4.03	Containment	2.7	2.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	ESF slave relays
				3	2	3	2	2	3	3	2	3	3	2	

T2G2 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
002K5.14	Reactor Coolant	3.8	4.2	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Consequences of forced circulation loss
015K6.01	Nuclear Instrumentation	2.9	3.2	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Sensors, detectors and indicators
016K3.12	Non-nuclear Instrumentation	3.4	3.6	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	S/G
027K2.01	Containment Iodine Removal	3.1	3.4	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Fans
028A1.02	Hydrogen Recombiner and Purge Control	3.4	3.7	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Containment pressure
034A2.03	Fuel Handling Equipment	3.3	4.0	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Mispositioned fuel element
071K4.06	Waste Gas Disposal	2.7	3.5	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Sampling and monitoring of waste gas release tanks
072G2.4.21	Area Radiation Monitoring	4.0	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the parameters and logic used to assess the status of safety functions
075A4.01	Circulating Water	3.2	3.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	Emergency/essential SWS pumps
086A3.01	Fire Protection	2.9	3.3	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Starting mechanisms of fire water pumps
				0	1	1	1	1	1	1	1	1	1	1	

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T3 PWR EXAMINATION OUTLINE

FORM ES-401-2

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
G2.1.1	Conduct of operations	3.8	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of conduct of operations requirements.
G2.1.27	Conduct of operations	3.9	4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of system purpose and or function.
G2.1.5	Conduct of operations	2.9	3.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to locate and use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.
G2.2.2	Equipment Control	4.6	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.
G2.2.42	Equipment Control	3.9	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to recognize system parameters that are entry-level conditions for Technical Specifications
G2.3.5	Radiation Control	2.9	2.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to use radiation monitoring systems
G2.3.7	Radiation Control	3.5	3.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to comply with radiation work permit requirements during normal or abnormal conditions
G2.4.1	Emergency Procedures/Plans	4.6	4.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of EOP entry conditions and immediate action steps.
G2.4.32	Emergency Procedures/Plans	3.6	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of operator response to loss of all annunciators.
G2.4.4	Emergency Procedures/Plans	4.5	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

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SRO T1G1 PWR EXAMINATION OUTLINE

FORM ES-401-2

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
055EA2.06	Station Blackout / 6	3.7	4.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Faults and lockouts that must be cleared prior to re-energizing buses
058AG2.1.19	Loss of DC Power / 6	3.9	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to use plant computer to evaluate system or component status.
077AA2.08	Generator Voltage and Electric Grid Disturbances / 6	4.3	4.4	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Criteria to trip the turbine or reactor
WE04EA2.2	LOCA Outside Containment / 3	3.6	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.
we11EG2.4.2	Loss of Emergency Coolant Recirc. / 4	3.8	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of operational implications of EOP warnings, cautions and notes.
we12EG2.4.1	Steam Line Rupture - Excessive Heat Transfer / 4	3.3	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the specific bases for EOPs.
											3	0	0	3	

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SRO T1G2 PWR EXAMINATION OUTLINE

FORM ES-401-2

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
005AG2.4.31	Inoperable/Stuck Control Rod / 1	4.2	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of annunciators alarms, indications or response procedures
032AA2.06	Loss of Source Range NI / 7	3.9	4.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Confirmation of reactor trip
051AG2.1.7	Loss of Condenser Vacuum / 4	4.4	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation.
WE03EA2.1	LOCA Cooldown - Depress. / 4	3.4	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.
											2	0	0	2	

ES-401, REV 11 **SRO T2G1 PWR EXAMINATION OUTLINE** **FORM ES-401-2**

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
004A2.03	Chemical and Volume Control	3.6	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Boundary isolation valve leak
006A2.13	Emergency Core Cooling	3.9	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Inadvertent SIS actuation
010G2.2.25	Pressurizer Pressure Control	3.2	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
076G2.2.40	Service Water	3.4	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to apply technical specifications for a system.
103G2.2.12	Containment	3.7	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of surveillance procedures.
											2	0	0	3	

ES-401, REV 11 **SRO T2G2 PWR EXAMINATION OUTLINE** **FORM ES-401-2**

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
001G2.1.32	Control Rod Drive	3.8	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to explain and apply all system limits and precautions.
011A2.09	Pressurizer Level Control	2.9	3.5	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	High ambient reflux boiling temperature effect or indicated PZR level
014A2.04	Rod Position Indication	3.4	3.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Misaligned rod
											2			1	

ES-401, REV 11

SRO T3 PWR EXAMINATION OUTLINE

FORM ES-401-2

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
G2.1.41	Conduct of operations	2.8	3.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the refueling processes
G2.2.15	Equipment Control	3.9	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to determine the expected plant configuration using design and configuration control documentaion
G2.2.18	Equipment Control	2.6	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process for managing maintenance activities during shutdown operations.
G2.3.14	Radiation Control	3.4	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities
G2.3.6	Radiation Control	2.0	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to aprove release permits
G2.4.26	Emergency Procedures/Plans	3.1	3.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage.
G2.4.8	Emergency Procedures/Plans	3.8	4.5	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Facility: Harris Nuclear Plant

Date of Examination: November 16, 2020

Examination Level: RO ☒ SRO ☐

Operating Test Number: 05000400/2020301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	N, R	Determine Axial Flux Difference with AFD Monitor Inoperable using OST-1021 and OP-163 (OST-1021) (JPM ADM-083-a) *K/A G 2.1.25* **2020 NRC RO A1-1**
Conduct of Operations	M, R	Determine Required boric acid flow using AOP-017 Attachment 4 and OP-107.01 (AOP-017) (JPM ADM-081-a) *K/A G2.1.23* **2020 NRC RO A1-2**
Equipment Control	M, R	Determine Clearance requirements for a CCW Pump (AD-OP-ALL-0200) (JPM ADM-003-b) *K/A G2.2.13* **2020 NRC RO A2**
Radiation Control	M, R	Given a set of conditions, determine and apply the facility dose limits (AD-RP-ALL-2000) (JPM ADM-028-c) *K/A G 2.3.7* **2020 NRC RO A3**
Emergency Plan	N/A	NOT SELECTED FOR RO

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	**(4)**
	(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)	**(0)**
	(N)ew or (M)odified from bank (≥ 1)	**(4)**
	(P)revious 2 exams (≤ 1; randomly selected)	**(0)**

2020 NRC RO Admin JPM Summary

2020 NRC RO A1-1 - Determine Axial Flux Difference with AFD Monitor Inop
(OST-1021)
(JPM ADM-083-a) **NEW**

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

The plant is at 90% power with a load reduction in progress when the load reduction is stopped to evaluate AFD following power oscillations. The candidate must perform Attachment 5 of OST-1021, Daily surveillance Requirements to determine the current AFD limit and if the AFD Monitor Alarm is operable or in operable.

2020 NRC RO A1-2 - Determine Required boric acid flow using AOP-017 Attachment 4 and OP-107.01 (AOP-017)
(JPM ADM-081-a) **MODIFIED**

K/A 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 4.3 SRO 4.4

The plant is in Mode 3. Instrument air header pressure is 45 psig and stable making automatic blender operating unavailable. VCT level is currently 19% and stable. The CRS directed the candidate perform a manual make to the VCT using the applicable procedure. The candidate will be provided with initial data and then be required to obtain AOP-017 which will direct the remaining values to be determined using OP-107.01, Attachment 7, calculate the maximum makeup flow rate to achieve the required boron concentration in the VCT along with the required boric acid flow rate and dilution flow rate.

NOTE: Modified by varying the initial data which will required the candidate to obtain different valves for the maximum makeup flow rate to achieve the required boron concentration in the VCT along with the required boric acid flow rate and dilution flow rate

2020 NRC RO Admin JPM Summary

RO Admin JPMs (continued)

2020 NRC RO A2 - Determine clearance requirements for a CCW Pump per AD-OP-ALL-0200 (AD-OP-ALL-0200)
(JPM ADM-021-f) **MODIFIED**

K/A G2.2.13 - Knowledge of tagging and clearance procedures.
(CFR: 41.10 / 45.13) RO 4.1 SRO 4.3

The plant is defueled. CCW Pump 1A-SA is required to be placed under a clearance for seal replacement. Cooling water and lube oil systems are NOT required to be placed under clearance. The candidate will be directed to determine the clearance requirements for CCW Pump 1A-SA using the SFDs, CWD and System Operating Procedures, as necessary. The candidate must provide electrical and mechanical protection and provide the necessary vent and drain paths.

NOTE: Modified by changing the component from the CSIP 1A-SA to the CCW Pump 1A-SA which will required the candidate to evaluate a different set of drawings to obtain the components required to be isolated to provide an adequate isolation boundary.

2020 NRC RO A3 - Given a set of conditions, determine and apply the facility dose limits (AD-RP-ALL-2000) (JPM ADM-028-c) **MODIFIED**

K/A G2.3.7 - Ability to comply with radiation work permit requirements during normal or abnormal conditions.
(CFR: 41.12 / 45.10) RO 3.5 SRO 3.6

The candidate will be supplied a survey map of a location in the RAB, a copy of AOP-36.08 and the required RWP for the radioactive area. The location also contains one or more hot spots. They must determine the individual stay time prior to exceeding the dose limits of the RWP. They will be provided Survey Maps, Simplified plant drawings to locate valves, Plant Maps of the area and a plant valve list to determine the location of the valves they will be required to operate in order to complete the task. The given information will supply the accumulated annual whole body dose for the AOs. They must perform their calculations based on RWP Stop Work Limits established for the RWP.

NOTE: Modified by varying the initial data which will required the candidate to obtain different valves for the required stay times based on updated limits for the RWP along with the dose rates of the survey map location. Additionally the candidate is required to determine when the RWP Stop Work Limit is reached vice the Facility dose limit

2020 NRC RO A4 – Not selected

Facility: Harris Nuclear Plant
Date of Examination: November 16, 2020
Examination Level: RO ☐ SRO ☒
Operating Test Number: 05000400/2020301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	N, R	Determine Axial Flux Difference with AFD Monitor Inoperable and Evaluate Technical Specifications using OST-1021 (OST-1021) (JPM ADM-083-a-SRO) *K/A G 2.1.25* **2020 NRC SRO A1-1**
Conduct of Operations	M, R	During a loss of shutdown cooling, determine the time that the RCS will reach core boiling and core boil-off conditions (AOP-020) (JPM ADM-005-c-SRO) *K/A G 2.1.25* **2020 NRC SRO A1-2**
Equipment Control	D, R	Review (for approval) the Completed OST-1017, Pressurizer PORV Block Valve Full Stroke Test (OST-1017) (JPM ADM-035-c-SRO) *K/A G2.2.12* **2020 NRC SRO A2**
Radiation Control	N, R	Review and complete Operations Actions of AP-545, Attachment 3, Section II. Pre-Entry Planning Actions (AP-545) (JPM ADM-075-a-SRO) *K/A G 2.3.13* **2020 NRC SRO A3**
Emergency Plan	N, R	Classify an Event (CSD-EP-HNP-0101-01) (JPM ADM-074-a-SRO) *K/A G2.4.41* **2020 NRC SRO A4**

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	**(5)**
	(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)	**(1)**
	(N)ew or (M)odified from bank (≥ 1)	**(4)**
	(P)revious 2 exams (≤ 1; randomly selected)	**(1)**

2020 NRC SRO Admin JPM Summary

2020 NRC SRO A1-1 - Determine Axial Flux Difference with AFD Monitor Inop and Evaluate Technical Specifications
(OST-1021)
(JPM ADM-083-a-SRO) **NEW**

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

The plant is at 90% power with a load reduction in progress when the load reduction is stopped to evaluate AFD following power oscillations. The candidate must perform Attachment 5 of OST-1021, Daily surveillance Requirements to determine the current AFD limit and if the AFD Monitor Alarm is operable or in operable.

2020 NRC SRO A1-2 - During a loss of shutdown cooling, determine the time that the RCS will reach core boiling and core boil-off conditions (AOP-020)
(JPM ADM-005-c-SRO) **MODIFIED**

K/A G2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.
(CFR: 41.10 / 43.5 / 45.12) RO 3.9 SRO 4.2

The candidate will be provided with initial plant conditions. A plant shutdown for refueling is in progress with the Reactor Vessel head off when a loss of RHR has occurred. The crew is implementing AOP-020, Loss of RCS Inventory or Residual Heat Removal While Shutdown. The SRO candidates must first determine which of the four plant curves to use (H-X-8 through H-X-11) and then calculate the time the RCS will reach core boiling and core boil-off based on the figures.

NOTE: Modified by changing the dates and times of plant shutdown and values of the core thermocouples. These changes have made the calculated answer substantially different than the bank JPM answer.

2020 NRC SRO A2 - Review (for approval) the Completed OST-1017, Pressurizer PORV Block Valve Full Stroke Test (OST-1017)
(JPM ADM-035-c) **DIRECT**

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

The candidate will be given a completed copy of OST-1017 to complete the Certification and Review by the CRS. The OST contains three (3) errors that the candidate must identify.

SRO Admin JPMs (continued)

2020 NRC SRO A3 – Review and complete Operations Actions of AP-545, Attachment 3, Section II. Pre-Entry Planning Actions (AP-545)
(JPM ADM-075-a-SRO) **NEW**

K/A G2.3.13 - Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

(CFR: 41.12 / 43.4 / 45.9 / 45.10) RO 3.4 SRO 3.8

The candidate will be supplied a partially completed copy of AP-545, Attachment 3, Containment Entry Permit, along with a JPM information sheet, an LCO Tracking Record and the most recently completed OST-1082 for the Containment Airlock. The candidate will be required to review the package and complete the Operations section for the Pre-Entry Planning section. Once the review is complete the candidate should determine that the PAL is considered Operable, however OST-1082 is required to be performed because it is beyond its periodicity.

2020 NRC SRO A4 - Classify an Event (CSD-EP-HNP-0101-01)
(JPM-ADM-082-a) **NEW**

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

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

Facility: Harris Nuclear Plant
Date of Examination: November 16, 2020
Exam Level: RO ☒ SRO-I ☒ **SRO-U (Bold)** ☒
Operating Test Number: 05000400/2020301

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

	System/JPM Title	Type Code*	Safety Function
a.	BTRS End of Life Dilution Operation (OP-108) (JPM-CR-280-a) K/A 004 A4.07	A, D, S	1
b.	Place Excess Letdown In Service (OP-107) (JPM-CR-211-b) *K/A 004 A4.06*	D, P, S	2
c.	**Take Corrective Action For Failure of CSIP Mini-Flow Valves to Re Position (EOP-E-0)** **(JPM-CR-225-e)** **K/A 006 A4.07**	**A, D, E, P, S**	**3**
d.	**Start an RCP (return to service following maintenance) w/ Spray Valve Failure (AOP-019)** **(JPM-CR-005-g)** ***K/A 002 A1.01***	**A, E, L, M, S**	**4P**
e.	Return the Containment Fan Coolers to normal following a Safety Injection actuation. (OP-169) (JPM CR-260-a) *K/A 022 A4.01*	D, EN, L, S	5
f.	**Shutdown EDG A-SA from MCB (for maintenance) Field Flash stays energized (OP-155)** **(JPM-CR-292-a)** ***K/A 064 A4.06***	**A, EN, M, S**	**6**
g.	Power Range NI Gain Adjustment (OP-105) (JPM CR-210-a) **RO Only** *K/A 015 A4.02*	D, S	7
h.	Align CCW to Support RHR System (OP-145) (JPM CR-085-b) *K/A 008 A4.10*	D, L, S	8

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for **SRO-U**			
i.	Restore Power To An Emergency Bus (OP-155) (JPM IP-239-a) *K/A 068 AA1.10*	A, M, EN, L	6
j.	**Place the ASI System in Standby Alignment (OP-185)** **(JPM-IP-291-a)** ***K/A 004 A4.11***	**D, L, R**	**2**
k.	**Isolate the SI Accumulators After a Control Room Evacuation (AOP-004)** **(JPM-IP-232-a)** ***K/A APE 068 AG2.1.30***	**D, E, EN, L**	**8**

* 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, and in-plant systems and functions may overlap those tested in the control room.

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

2020 NRC Control Room/In-Plant JPM Summary

Simulator JPMs

JPM a – BTRS End of Life Dilution Operation (OP-108)
(JPM-CR-280-a)

K/A 004 A4.07 – Ability to manually operate and/or monitor in the control room: Boration/dilution (CFR: 41/7 / 45.5 to 45.8) RO 3.9 / SRO 3.7

Evaluated position: Operator at the Controls (OATC) responsibilities.

Turnover: The candidate will assume the watch with the unit operating at 100% and the CRS has directed you to place BTRS in service for End of Life Dilution Operation per OP-108. The candidate will be informed that BTRS was initially placed in service earlier this week and the previously in service resin bed will be removed from service and realigned. The candidate will be directed to start at step 2 of section 8.9.2

Task: Place BTRS in service for EOL operations and respond to the failure of HC-387, BTRS Demin Bypass to operate.

Verifiable actions: The candidate will have to determine if flushing of the BTRS system to the RHT is required. Flushing of the BTRS system is required based on the realignment of the in service resin bed. The candidate will attempt to place BTRS in service and will not get the correct response for the White DIL light when repositioning HC-387.

Alternate Path – YES. When the White DIL Light is determined to be NOT illuminated and HC-387 is incorrectly operating the candidate will have to verify open the BTRS bypass and verify shut the BTRS inlet.

JPM completion: Once the candidate initiates a work request, evaluation on this JPM is complete.

JPM b –Place Excess Letdown in Service (OP-107)
(JPM-CR-211-b) – Direct - **Previous** from the 2016 Exam. (Randomly selected from the Simulator JPM bank)

K/A 004 A4.06 – Ability to manually operate and/or monitor in the control room: Letdown isolation and flow control valves
(CFR: 41/7 / 45.5 to 45.8) RO 3.6 / SRO 3.1

Evaluated position: Operator at the Controls (OATC) responsibilities.

Turnover: The plant is at 100%, steady state power middle of life (MOL). Normal Letdown needs to be secured for maintenance due to a problem with PCV-145. The CRS has directed the OATC to establish Excess Letdown to the VCT per OP-107, Section 8.2.

Task: Establish Excess Letdown to the VCT in accordance with OP-107, Section 8.2

Simulator JPMs (continued)

JPM b (continued)

Verifiable actions: The candidate will perform a valve lineup to establish a flow path from Excess Letdown to the Reactor Coolant Drain Tank. This flow path will be used to flush the lines to establish the same boron concentration as the RCS. They will then establish a valve lineup to the VCT and adjust a hand control valve to establish Excess Letdown flow at a rate that does not cause Excess Letdown temperature to exceed 174°F or pressure to exceed 150 psig. The MCB has indications and alarms for the parameters. Temperature and pressure limits prevent damage to the Excess Letdown Heat Exchanger and prevent lifting a relief in the Excess Letdown line.

Alternate Path – No - There are no failures with this JPM.

JPM completion: Excess letdown is in service and is flowing with temperature < 174°F and pressure < 150 psig in accordance with OP-107, Section 8.2.

JPM c – Take Corrective Action For Failure of CSIP Mini-Flow Valves to Re-Position (EOP-E-0)

(JPM-CR-225-e) SRO Upgrade - Direct - Previous from the 2018 Exam.
(Randomly selected from the Simulator JPM bank)

K/A 006 A4.07 Ability to manually operate and/or monitor in the control room: ECCS pumps and valves (CFR: 41.7 / 45.5 to 45.8) RO 4.4 SRO 4.4

Evaluated position: Operator at the Controls (OATC) responsibilities.

Turnover: The plant was operating at 100% when a technician's error resulted in an automatic Reactor Trip / Safety Injection signal. The crew is performing EOP-E-0, Reactor Trip or Safety Injection and is at step 37. The CRS has directed the OATC to begin at step 37 and continue performing EOP-E-0.

Task: Obtain adequate flow through a running CSIP.

Verifiable actions: The candidate will be required to change valve positions and stop one CSIP to secure the ECCS High Head injection flow path and establish a Normal Charging flow path from the lineup to RCS.

Alternate Path – YES. During the valve alignment 1CS-214, Common Normal Mini-flow Isolation Valve, will fail to open. This failure will require the operator to use RNO actions to ensure minimum Charging Flow is established for the running CSIP prior to terminating SI flow by shutting BIT outlet valves 1SI-3 and 1SI-4.

JPM completion: When Charging + Seal Injection flow is being maintained at >60 gpm the CRS will notify the OATC that the task is complete and another operator will continue implementing the procedure.

Simulator JPMs (continued)

JPM d – Start a RCP and respond to a subsequent Spray valve failure (OP-100, AOP-019) (JPM-CR-005-g) SRO Upgrade - Alternate Path - Modified

K/A 002 A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCS controls including: Primary and secondary pressure
(CFR: 41.5 / 45.7) RO 3.8 SRO 4.1

Evaluated position: Operator at the Controls (OATC) responsibilities

Turnover: A plant startup will be in progress with the 'B' and 'C' RCPs in operation. Maintenance has been completed on the 'A' RCP. The CRS has directed the OATC to start the 'A' RCP in accordance with OP-100, Reactor Coolant System.

Task: Start the 'A' RCP, identify the associated PRZ Spray valve (1RC-107) fails open and must be manually shut.

Verifiable actions: The candidate will be required to operate the RCP and its Oil Lift system to start the 'A' RCP in accordance with OP-100, while monitoring progress using MCB indicators and computer screens (ERFIS). The candidate will be required to operate the PRZ Spray valve (1RC-107).

Alternate Path – YES. After the RCP is started the 'A' RCP Spray valve will fail open resulting in lowering RCS pressure and various MCB annunciators. The candidate will be expected to enter AOP-019, Malfunction Of RCS Pressure Control and perform the immediate actions to take manual control of the spray valve and shut the valve. This will preclude an RCS pressure reduction to a Safety Injection actuation setpoint.

JPM completion: When the candidate has shut the RCP 'A' Spray valve, 1RC-107 and the SRO has been informed that the task is unsuccessful, evaluation on this JPM is complete.

Modification: Modified by changing the affected RCP from RCP 'A' to RCP 'B''. This will require the candidate to locate control switches and indications additionally the plant response will vary from the original JPM due to response of the PRZ Pressure system with RCP 'B' secured. RCP 'B' is the dominant Spray valve due to its location in reference to the Pressurizer the system is faster to respond to the changes in pressure.

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

K/A 022 A4.01 Ability to manually operate and/or monitor in the control room: CCS fans
(CFR: 41.7 / 45.5 to 45.8) RO 3.6 SRO 3.6

Evaluated position: Balance of Plant (BOP) Operator responsibilities.

Turnover: The plant is tripped due to an inadvertent SI initiation has occurred and the control room staff has entered EOP-E-0 and EOP-ES-1.1. Attachment 1 of EOP-ES-1.1 is being performed to realign plant systems. The CRS has directed the BOP to realign CMNT

Simulator JPMs (continued)

JPM e (continued)

Fan Coolers in accordance with OP-169 Section 8.4. The candidate will be directed to align the A Train of CNMT Fan Coolers for normal service.

Task: Place Containment Cooling Fans in Max Cooling Mode.

Verifiable actions: The candidate will secure both A Train CNMT Fan Coolers and verify proper damper alignment for the secured fans. The candidate will restart the A Train Fans per section 5.1 of OP-169. To minimize the starting current required for Hi-Speed operation the fans are initially started in Lo-Speed, then stopped and restarted in Hi-Speed

Alternate Path – NO.

JPM completion: Once the B Train of CNMT Fan Coolers are in standby and the determination is made that Maximum Cooling Mode is NOT required, evaluation on this JPM is complete.

JPM f – Shutdown EDG A-SA from MCB (for maintenance) Field Flash stays energized (OP-155)
(JPM-CR-292-c) SRO Upgrade - Alternate Path - Modified

K/A 064 A4.06 – Ability to manually operate and/or monitor in the control room: Manual start, loading, and stopping of the ED/G
(CFR: 41.7 / 45.5 to 45.8) RO 3.9 SRO 3.9

Evaluated position: Balance of Plant (BOP) Operator responsibilities.

Turnover: The plant is operating at 100% power steady state middle of life (MOL). The ‘A’ EDG is running in parallel with the grid to support testing of the governor. Testing of the governor is complete and the previous shift has reduced the EDG load from 6.0 MW to 2.3 MW and 1 MVAR over the last 30 minutes per OP-155, Diesel Generator Emergency Power System, Section 7.1. The CRS has directed the BOP to observe the NOTE prior to OP-155 section 7.1.2, Step 4, and continue shutting down the ‘A’ EDG.

Task: Shutdown EDG A-SA from the MCB.

Verifiable actions: The candidate will have to reduce load from 2.2 MW to 0.5 MW during this time and divorce the ‘EDG from the grid in accordance with OP-155, Diesel Generator Emergency Power System, Section 7.1.2. Once the ‘A’ EDG is separated from the grid after stack exhaust temperatures are checked the ‘A’ EDG is stopped.

Alternate Path – YES. The ‘A’ field breaker will remain shut and field voltage will remain on the ‘A’ EDG which will require the candidate to emergency stop the ‘A’ EDG in order to remove the field flashing voltage from the ‘A’ EDG.

JPM completion: When the candidate emergency stops the ‘A’ EDG and the SRO is informed, evaluation on this JPM is complete.

Simulator JPMs (continued)

JPM f (continued)

Modification: Modified by changing the affected EDG from EDG 'B' to EDG 'A'. This will require the candidate to locate control switches and indications from a different section of the MCB.

JPM g – Power Range NI Gain Adjustment (OP-105)
(JPM CR-210-a) **RO Only - Direct**

K/A 015 A4.02 Ability to manually operate and/or monitor in the control room: NIS indicators
(CFR: 41.7 / 45.5 to 45.8) RO 3.9 SRO 3.9

Evaluated position: Balance of Plant (BOP) Operator responsibilities.

Turnover: The plant is operating at 100% power steady state middle of life (MOL). Maintenance on PR Channel N-41, all required testing has been completed and the channel is ready to be returned to service. A calorimetric has just been performed per OST-1000, Power Range Heat Balance, ERFIS Online Calculation, Daily Interval, Mode 1 (Above 15% Power). The calculated power is 99.64%. The CRS has directed the BOP to perform the Power Range NI Gain Adjust for PR channel N-41 in accordance with OP-105, Excore Nuclear Instrumentation, Section 8.3 and Attachment 2.

Task: Power Range NI Gain Adjustment.

Verifiable actions: The candidate will be required to perform a calculation to determine the difference in the calculated power and the current indicated power of the Nuclear instrument and place the Rod Control system in manual to properly align the plant in accordance with OP-105, Excore Nuclear Instrumentation, Attachment 2, while monitoring progress using MCB.

Alternate Path – NO.

JPM completion: When the adjustments to return the NI's within 2% are complete and the switches are in the original configuration, evaluation on this JPM is complete.

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

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

Evaluated position: Operator at the Controls (OATC) responsibilities.

Turnover: The plant is in Mode 4, going to Mode 5. Preparations are underway to place both trains of RHR in service. Both ESW trains are in service. CCW Pump "A" is running. The CRS has directed the OATC to align CCW to support RHR operation in accordance with OP-145, Component Cooling Water.

Simulator JPMs (continued)

JPM h (continued)

Task: Align CCW to Support RHR System.

Verifiable actions: The candidate will be required to start a second CCW pump and realign the CCW system to supply the A and B train essential header to supply RHR, and isolate the A train essential header of the CCW from the non-essential header in accordance with OP-145, Component Cooling Water, Section 8.9 and 5.2 while monitoring CCW system operating parameters using MCB level and pressure indicators and computer screens (ERFIS).

Alternate Path – NO.

JPM completion: When the candidate contacts the AO to verify CCW flow locally then evaluation on this JPM is complete.

Modification: The most current revision of this OP has a new attachment which will modify this JPM by having the Operator document as found values for the RHR HX and RHR Pump Cooler Outlet flows along with the as left values of these flows. This attachment provides a new table for the operator to document the information along with new opportunities for the operator to direct local actions.

2020 NRC Control Room/In-Plant JPM Summary

In-Plant JPMs

JPM i – Restore Power to an Emergency Bus (OP-155)
(JPM IP-239-a) Alternate Path - Modified

K/A 068 AA1.10 Ability to operate and / or monitor the following as they apply to the Control Room Evacuation: Power distribution: ac and dc
(CFR 41.7 / 45.5 / 45.6) RO 3.7 / SRO 3.9

Evaluated position: EDG Building / Balance of the Plant Operator (BOP) responsibilities during AOP-004 implementation.

Turnover: AOP-004 has been entered due to a fire in the MCR. ‘B’ Safety bus is not energized due to a SUT fault. EDG 1B-SB was in standby operation but did not automatically start. AOP-004 has directed that the ‘B’ EDG be locally started and ‘B’ safety bus energized. Both safety and non-safety Plant DC Distribution Systems are in operation per OP-156.01 to support EDG operation. The manual transfer to LOCAL has been completed at the Main Transfer Panel 1B-SB.

Task: Locally start the ‘B’ EDG IAW OP-155 Section 8.14.2

Verifiable actions: Note- all actions will be simulated. Locate the EDG 1B-SB push to start pushbutton and start the 1B-SG EDG by depressing the pushbutton. Locate the K1 relay and position the switch in the reset position.

Alternate Path – YES. The EDG should automatically flash the field of the Generator once EDG speed is greater than approximately 200 RPM. This failure requires the candidate to locate the K1 relay behind the GCP left section door to manually reset the K2 relay.

JPM completion: Once the candidate has simulated starting the ‘B’ EDG and the K1 relay has been reset the JPM is complete.

Modification: This JPM has been modified by changing status of the K1 relay which requires the candidate to complete alternative field actions to reset the K1 relay to allow the EDG field to flash.

JPM j – Place the ASI System in Standby Alignment (OP-185)
(JPM-IP-291-a) SRO Upgrade - Direct

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

NOTE: This JPM is inside the RCA.

Evaluated position: Auxiliary Operator in the RAB (AO RAB)

Turnover: The plant is in Mode 4 and a heat up is in progress. The CRS directs the candidate to place the ASI system in automatic standby alignment in accordance with OP-185 section 5.1.

Task: Locally place the ASI system in automatic standby alignment.

In-plant JPMs (continued)

JPM j (continued)

Verifiable actions: The candidate will verify the ASI supply header isolation valves are open and the de-energized status of the ASI system control panel. The candidate will realign the ASI pump to automatic and return the Squib valve bypass control switches to normal alignment on the ASI control panel. The candidate will turn on the ASI system control panel feeder supply breaker and the ASI pump power supply breaker. The candidate will recheck the indications on the ASI system control panel for the proper standby alignment of the system.

Alternate Path – NO.

JPM completion: Once the candidate proceeds to section 5.1.3, Automatic Standby alignment configuration control closeout the evaluation on this JPM is complete.

JPM k – Isolate the SI Accumulators After a Control Room Evacuation (AOP-004) (JPM-IP-232-a) SRO Upgrade - Direct

K/A APE 068 AG2.1.30 Ability to locate and operate components, including local controls.
(CFR: 41.7 / 45.7) RO 4.4 / SRO 4.0

Evaluated position: Auxiliary Operator in the Turbine Building (AO TB)

Turnover: The unit Main Control Room has been evacuated due to a fire. The crew is performing a cooldown in accordance with AOP-004, Remote Shutdown. The CRS will direct the candidate to isolate SI Accumulators. The candidate will perform AOP-004 step 30.

Task: Locally isolate the SI accumulators after Control room evacuation.

Verifiable actions: The JPM cues include information of the proper status of the power supply light indications. The candidate will be required to locate each breaker cubicle and reposition both breakers, then obtain the key for the Auxiliary Transfer Panel in order to reposition the SI Accumulator isolation valves from this location. The candidate will be required to identify the individual indicating lights on the local control panel and operate the control panel pushbuttons.

Alternate Path – NO.

JPM completion: Once the CRS is notified that AOP-004, step 30 is complete and the SI Accumulators are isolated then evaluation on this JPM is complete.

1. 2020 NRC RO 001

The unit was operating at 100% power when a Reactor trip occurred.

Which ONE of the following completes the statements below?

Xenon-135 concentration will decay to zero (xenon-free) __(1)__ hours following the Reactor trip.

In accordance with EOP-ES-0.1, Reactor Trip Response, the operator will ensure that Source Range detectors energize when Intermediate Range flux FIRST lowers to __(2)__ AMPS.

A. (1) 30 - 40

(2) 5×10^{-11}

B. (1) 30 - 40

(2) 1×10^{-10}

C. (1) 70 - 80

(2) 5×10^{-11}

D. (1) 70 - 80

(2) 1×10^{-10}

Plausibility and Answer Analysis

Reason answer is correct: Xenon-135 concentration will peak and then decrease zero following a Reactor trip. Xenon free conditions will be reached in approximately 70 to 80 hours following the Reactor trip. EOP-ES-0.1 directs the operator to ensure Source Range detectors energize when Intermediate Range flux lowers to less than $5x10^{-11}$ AMPS.

A. Incorrect. *The first part is plausible since Xe-135 concentration will reach an equilibrium value 30 to 40 hours following a downpower. The second part is plausible since during a Reactor startup the Source Range detectors are de-energized at $1x10^{-10}$ AMPS (P-6 setpoint).*

B. Incorrect. *The first part is plausible since Xe-135 concentration will reach an equilibrium value 30 to 40 hours following a downpower. The second part is correct.*

C. Correct.

D. Incorrect. *The first part is correct. The second part is plausible since during a Reactor startup the Source Range detectors are de-energized at $1x10^{-10}$ AMPS (P-6 setpoint).*

000007 Reactor Trip - Stabilization - Recovery / 1

007 EK1.05; Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Decay power as a function of time

(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	3.3 3.8
Technical Reference:	EOP-ES-0.1, Step 18, Page 28, Rev. 4
References to be provided:	None
Learning Objective:	Fission Product Poisons - Instructor Guide, EO 1.5 GP-LP-3.07 Objective 1
Question Origin:	New
Comments:	Xe-135 is a fission product poison that has a correlation with power as it pertains to neutron population following a reactor trip. Xenon over time is a function of both production rate (fission and iodine decay) versus loss rate (burnout and decay).
Tier/Group:	T1/G1

2. 2020 NRC RO 002

Given the following plant conditions:

- A break in the Pressurizer steam space has resulted in a small break LOCA
- The crew is implementing EOP-ES-1.2, Post LOCA Cooldown and Depressurization
- RCS temperature is stable

Which ONE of the following completes the statements below?

The steam dumps will be operated in the ___(1)___ Mode of Control.

The operator must depress the OUTPUT ___(2)___ pushbutton to initiate an RCS cooldown to cold shutdown conditions.

A. (1) T-AVG

(2) RAISE

B. (1) T-AVG

(2) LOWER

C. (1) Steam Pressure

(2) RAISE

D. (1) Steam Pressure

(2) LOWER

Plausibility and Answer Analysis

Reason answer is correct: In accordance with EOP-ES-1.2, with the condenser available, the condenser steam dumps will be used to dump steam from the intact steam generators in the Steam Pressure Mode of Control. With the steam dumps in MANUAL, depressing the OUTPUT RAISE pushbutton will open the Group 1 steam dumps to initiate an RCS cooldown.

A. Incorrect. The first part is plausible since the steam dumps are normally operated in the T-AVG Mode of Control; however, this is incorrect as they are transferred to the Steam Pressure Mode of Control just prior to starting the RCS cooldown. The second part is correct.

B. Incorrect. The first part is plausible since the steam dumps are normally operated in the T-AVG Mode of Control; however, this is incorrect as they are transferred to the Steam Pressure Mode of Control just prior to starting the RCS cooldown. The second part is plausible since the SETPOINT LOWER pushbutton would be used if the Steam Dump Pressure Controller was in AUTO; however, the controller is in MANUAL and the OUTPUT RAISE pushbutton must be used to open the steam dumps.

C. Correct.

D. Incorrect. The first part is correct The second part is plausible since the SETPOINT LOWER pushbutton would be used if the Steam Dump Pressure Controller was in AUTO; however, the controller is in MANUAL and the OUTPUT RAISE pushbutton must be used to open the steam dumps.

000008 Pressurizer Vapor Space Accident / 3

008AA1.03; Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: Turbine bypass in manual control to maintain header pressure

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	2.8 2.6
Technical Reference:	OP-126, Section 5.3, Page 14, Rev. 41 EOP-ES-1.2, Step 10, Page 12, Rev. 4
References to be provided:	None
Learning Objective:	SDS-ILC Objective 6.f EOP-LP-3.05 Objective 5.a
Question Origin:	New
Comments:	K/A is matched since the applicant must demonstrate the ability to operate the Steam Dump controller in manual during a pressurizer steam space accident.
Tier/Group:	T1/G1

3. 2020 NRC RO 003

Given the following plant conditions:
- The unit was operating at 100% power when a LOCA occurred
- The Reactor was tripped and Safety Injection actuated
- 1CS-11, Letdown Isolation, is shut to isolate the break in accordance with EOP-ECA-1.2, LOCA Outside Containment

Subsequently:
- The crew is implementing EOP-ES-1.1, SI Termination, to terminate Safety Injection

Which ONE of the following completes the statements below?

Following the termination of Safety Injection, the **reason** EOP-ES-1.1 directs the establishment of excess letdown __(1)__ to prevent RCS overpressurization.

If excess letdown is established, based on plant system design, the __(2)__ signal is required to be reset to allow restoration of CCW to the excess letdown heat exchanger.

A. (1) is

(2) Safety Injection

B. (1) is

(2) Phase A

C. (1) is NOT

(2) Safety Injection

D. (1) is NOT

(2) Phase A

Reason answer is correct: With normal letdown isolated due to the break, excess letdown will need to be established to maintain RCS inventory control in order to offset RCP seal injection flow. The Phase A signal (generated from Safety Injection) isolated CCW to the excess letdown heat exchanger and is the signal that must be reset to allow the restoration of excess letdown flow.

A. Incorrect. The first part is plausible since this is the result of failing to control RCS inventory and allowing the RCS to go water solid; however, this is incorrect since the PRZ PORV's and Safeties will open to prevent RCS over pressurization. The second part is plausible since EOP-ES-1.1 will direct both the Safety Injection and Phase A signals be reset prior to establishing excess letdown flow; however, this is incorrect as the Phase A signal isolates CCW to the excess letdown heat exchanger, and is reset independently of Safety Injection, allowing the establishing of excess letdown regardless of the reset status of Safety Injection. Additional plausibility is found in the CCW cooled components that are isolated directly from a Safety Injection signal such as the Gross Failed Fuel Detector.

B. Incorrect. The first part is plausible since this is the result of failing to control RCS inventory and allowing the RCS to go water solid; however, this is incorrect since the PRZ PORV's and Safeties will open to prevent RCS over pressurization. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since EOP-ES-1.1 will direct both the Safety Injection and Phase A signals be reset prior to establishing excess letdown flow; however, this is incorrect as the Phase A signal isolates CCW to the excess letdown heat exchanger, and is reset independently of Safety Injection, allowing the establishing of excess letdown regardless of the reset status of Safety Injection. Additional plausibility is found in the CCW cooled components that are isolated directly from a Safety Injection signal such as the Gross Failed Fuel Detector.

D. Correct.

000009 Small Break LOCA / 3

009EK3.02; Knowledge of the reasons for the following responses as the apply to they small break LOCA: Opening excess letdown isolation valve

(CFR 41.5 / 41.10 / 45.6 / 45.13)

Importance Rating:	2.8 3.2
Technical Reference:	OMM-004, Attachment 4, Page 47, Rev. 42 EOP-ES-1.1, Step 17, Page 22, Rev. 3 SDD-ES-1.1, Step 17, Page 7, Rev. 1 ES-1.1 Background Document, Step 13, Page 23, Rev. 3
References to be provided:	None
Learning Objective:	CCW-ILC Objective 7.f CVCS-ILC Objective 5.i
Question Origin:	New
Comments:	Early Submittal Modified wording in part 2 of question to resolve conflict with RO Question #55. K/A is matched since applicant must demonstrate an understanding as to why excess letdown must be placed in service during a small break LOCA event.
Tier/Group:	T1/G1

4. 2020 NRC RO 004

Which ONE of the following completes the statements below regarding transferring the RHR system from the RWST to the Containment (CNMT) sumps (recirculation mode) during a large break LOCA?

The CNMT Sump to RHR Pump Suction valves will open automatically on ___(1)___ level.

The RWST to RHR Pump Suction valves will ___(2)___.

A. (1) lowering RWST

(2) remain open until shut by an operator

B. (1) lowering RWST

(2) begin to shut five seconds after the CNMT Sump to RHR Pump Suction Valves reach the full open position

C. (1) rising CNMT sump

(2) remain open until shut by an operator

D. (1) rising CNMT sump

(2) begin to shut five seconds after the CNMT Sump to RHR Pump Suction Valves reach the full open position

Plausibility and Answer Analysis

Reason answer is correct: The Containment Sump to RHR Pump Suction Valves will automatically open upon receipt of the RWST Low-Low Level alarm and once fully open, the operator will shut the RWST to RHR Pump Suction Valves to establish the RHR Pump Recirculation Alignment.

A. Correct.

B. Incorrect. The first part is correct. The second is plausible since this describes operation of the RWST to CSS Pump Suction Valves (1CT-26 and 1CT-71); however, this is incorrect since an operator must shut the RWST to RHR Pump Suction valves.

C. Incorrect. The first part is plausible since a NOTE in EOP-ES-1.3 states that 142 inches wide range level assures a long term recirculation suction source; however, this in incorrect since automatic opening of the CNMT suction valves is interlocked with RWST level, not CNMT sump level. The second part is correct.

D. Incorrect. The first part is plausible since a NOTE in EOP-ES-1.3 states that 142 inches wide range level assures a long term recirculation suction source; however, this in incorrect since automatic opening of the CNMT suction valves is interlocked with RWST level, not CNMT sump level. The second is plausible since this describes operation of the RWST to CSS Pump Suction valves (1CT-26 and 1CT-71); however, this is incorrect since an operator must shut the RWST to RHR Pump Suction Valves.

000011 Large Break LOCA / 3

011EA2.13; Ability to operate and monitor the following as they apply to a Large Break LOCA: Safety injection components

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	4.1 4.2
Technical Reference:	EOP-E-1, Step 16, Page 22, Rev. 5 EOP-ES-1.3, Steps 1 & 2, Pages 4 & 6, Rev. 4
References to be provided:	None
Learning Objective:	EOP-EP-3.01 Objective 3.b SIS-ILC, Objectives 6.g & 6.h
Question Origin:	Bank
Comments:	K/A is matched since the applicant must demonstrate the ability to monitor and operate Safety Injection valves during the transfer from the RWST to the CNMT sump during a large break LOCA.
Tier/Group:	T1/G1

5. 2020 NRC RO 005

Given the following plant conditions:
- EOP-FR-C.1, Response to Inadequate Core Cooling, is being implemented
- Containment pressure is 2.5 psig
- Maximum Core Exit Thermocouples (CET) temperatures are 1305°F
- All SGs have been depressurized to 130 psig
- Support conditions have been established to the 'B' and 'C' RCPs ONLY

Subsequent to the above conditions:
- RCP 'C' was started and CET temperatures are now 1220°F and stable
- The crew is evaluating if additional RCPs can be started to provide core cooling

Current SG narrow range levels are:
- SG 'A' level is 35%
- SG 'B' level is 15%
- SG 'C' level is 39%

Which ONE of the following identifies the operator action(s) required to be taken NEXT in accordance with EOP-FR-C.1?

A. Start RCP 'A'

B. Start RCP 'B'

C. Re-establish a heat sink in at least one SG

D. Open the PRZ PORVs and RCS vent valves

Plausibility and Answer Analysis

Reason answer is correct: EOP-FR-C.1 loops through starting all available RCPs, one at a time, in an idle RCS cooling loop with SG level > 25% [40%]. Normal conditions, although desired, are NOT required for starting the RCPs during ICC conditions.

A. Correct.

B. Incorrect. Plausible since normal conditions have been established for RCP 'B', but the 'B' RCS cooling loop is not available due to insufficient inventory in the 'B' SG.

C. Incorrect. Plausible since this choice would be correct if Containment conditions were adverse (≥ 3 psig). The adverse value for SG level is 40%. However, this is incorrect since Containment conditions are normal. Any SG level > 25% ensures that adequate secondary heat sink exists.

D. Incorrect. Plausible since once all available RCPs are started and CETs remain above 1200°F, the RCS should be vented to containment through the PORVs and RCS vent valves in an attempt to lower pressure to allow flow through the core. In this case, another RCP is available in an idle loop and must be started first.

000015 Reactor Coolant Pump Malfunctions / 4

015AA2.11; Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to jog RCPs during ICC

(CFR 43.5 / 45.13)

Importance Rating:	3.4 3.8
Technical Reference:	FR-C.1 Background, Section 2.3, Page 4, Rev. 3 EOP-FR-C.1, Step 20, Page 24, Rev. 6
References to be provided:	None
Learning Objective:	EOP-LP-3.10 Objectives 6.a & 9.e
Question Origin:	Bank
Comments:	Ask Chief Examiner if acceptable to ask restarting RCPs in accordance with our ICC procedure; does not address jogging the RCPs. Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by testing starting RCPs in EOP-FR-C.1. K/A is matched since applicant must evaluate plant conditions and determine that the conditions support starting an additional RCP during ICC conditions.
Tier/Group:	T1/G1

6. 2020 NRC RO 006

In accordance with AOP-018, Reactor Coolant Pump Abnormal Conditions, which ONE of the following completes the statements below?

If all RCP seal cooling is lost for greater than a MINIMUM of __(1)__ minutes, a controlled restoration of seal injection flow must be done.

The basis for this requirement is to __(2)__.

A. (1) 4

(2) preclude increased seal leakage

B. (1) 4

(2) protect against potential pump radial bearing damage

C. (1) 10

(2) preclude increased seal leakage

D. (1) 10

(2) protect against potential pump radial bearing damage

Reason answer is correct: Per the AOP-018-BD, DO NOT restore seal injection to an RCP that has lost all seal cooling for 4 minutes. If seals have had a total loss of seal cooling for 4 minutes, by not allowing restoration of seal cooling, the popping open of the seal should be precluded and limit the leakage to 21 gpm per pump.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since this is the basis for maintaining CCW flow to the RCP Bearing Oil Coolers.

C. Incorrect. The first part is plausible since 10 minutes is the RCP trip criteria if the RCPs were to lose CCW flow to either motor or cooler. The second part is correct.

D. Incorrect. The first part is plausible since 10 minutes is the RCP trip criteria if the RCPs were to lose CCW flow to either motor or cooler. The second part is correct. The second part is plausible since this is the basis for maintaining CCW flow to the RCP Bearing Oil Coolers.

000022 Loss of Reactor Coolant Makeup / 2

022AK1.01; Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Consequences of thermal shock to RCP seals

(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	2.8 3.2
Technical Reference:	AOP-18-BD, Section 3.3, Page 44, Rev. 25
References to be provided:	None
Learning Objective:	AOP-LP-3.18 Objective 5
Question Origin:	Previous (2018 NRC RO 28)
Comments:	K/A is matched since applicant must demonstrate an understanding of the consequences of loss of all seal cooling to RCPs for an extended period of time.
Tier/Group:	T1/G1

7. 2020 NRC RO 007

Given the following plant conditions:

- The unit is operating in Mode 5
- The RCS is in solid plant operation
- Both trains of RHR are aligned in the Shutdown Cooling Mode

Subsequently:

- A large RCS leak developed

Conditions are as follows:

- The crew has aligned flow through the BIT with 'A' CSIP in service as directed by AOP-020, Loss Of RCS Inventory Or Residual Heat Removal While Shutdown
- Core Exit Thermocouples continue to rise
- RCS water level continues to lower

Which ONE of the following completes the statement below regarding the action required by AOP-020 to lower Core Exit Thermocouple temperatures?

__(1)__ with flow through __(2)__.

A. (1) Start the 'B' CSIP

(2) 1SI-3 and 1SI-4, BIT Outlet Valves

B. (1) Start the 'B' CSIP

(2) 1SI-52, Alternate High Head SI to Cold Leg Valve

C. (1) Align 'A' RHR Pump for Low Head SI

(2) 1SI-340, Low Head SI Train A to Cold Leg Valve

D. (1) Align 'A' RHR Pump for Low Head SI

(2) 1SI-359, Low Head SI Trains A & B to Hot Leg valve

Plausibility and Answer Analysis

Reason answer is correct: Align one train of RHR for Low Head SI with flow through 1SI-340, Low Head SI Train A to Cold Leg Valve, is directed in AOP-020, Section 3.6.

A. Incorrect. Plausible since starting the second CSIP with flow through 1SI-3 and 1SI-4, BIT Outlet Valves, would provide additional flow; however this is incorrect as only one CSIP is operable in this mode.

B. Incorrect. Plausible since starting the second CSIP with flow through 1SI-52, Alternate High Head SI to Cold Leg Valve, would provide additional flow and this alignment is directed in EOP-ES-1.3, Transfer to Cold Leg Recirculation, with two CSIPs; however, this is incorrect as only one CSIP is operable in this mode

C. Correct.

D. Incorrect. Plausible since aligning one train of RHR for Low Head SI is directed in AOP-020; however, this is incorrect as flow is through 1SI-340, Low Head SI Train A to Cold Leg Valve, not 1SI-359. This alignment is used in EOP-ES-1.4, Transfer to Hog Leg Recirculation.

000025 Loss of Residual Heat Removal System / 4

025AA1.12; Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: RCS temperature indicators

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.6 3.5
Technical Reference:	AOP-020, Section 3.6, Steps 3 & 4, Pages 66 & 67. Rev. 39
References to be provided:	None
Learning Objective:	AOP-LP-3.20 Objective 3
Question Origin:	Bank (2014 NRC RO 75)
Comments:	Ask Chief Examiner if loss of RCS inventory while on RHR will satisfy the K/A. Phonecon 7/30: Chief Examiner stated K/A matched if question addresses loss or reduction in RHR flow. K/A is matched since RHR cooling is being impacted as evident by the rising Core Exit Thermocouple temperatures. The applicant must use this indication along with other plant indications to determine which injection flow path must be established to mitigate the loss of cooling. A large RCS leak will threaten the continued operations of the RHR system and HNP uses the same procedure for this event as it does for a complete loss of RHR flow.
Tier/Group:	T1/G1

8. 2020 NRC RO 008

Given the following plant conditions:

- The unit is operating at 100% power
- ALB-005-6-1, CCW Surge Tank High-Low Level, has just alarmed
- The OATC reports that CCW Surge Tank level is 39% and trending lower

Which ONE of the following automatic actions must be verified in accordance with APP-ALB-005?

A. 1DW-15, Makeup Valve, has opened

B. CCW Drain Tank Transfer Pump has tripped

C. CCW Holdup Tank Transfer Pump has tripped

D. CCW flow to the GFFD and RCS Sample Panel has isolated

Plausibility and Answer Analysis

Reason answer is correct: In accordance with APP-ALB-005, the operator must verify that CCW flow isolated to the GFFD and Primary Sample Panel on a low CCW Surge Tank level (40%).

A. Incorrect. Plausible since APP-ALB-005 directs the operator to go to AOP-014 for low CCW Surge Tank level which will direct opening 1DW-15 to add water; however, this is incorrect since this is a manual valve and does not open automatically.

B. Incorrect. Plausible since this pump has an auto trip feature; however, this is incorrect since the pump trips on high CCW Surge Tank level (75%), not low level (40%).

C. Incorrect. Plausible since this pump has an auto trip feature; however, this is incorrect since the pump trips on high CCW Surge Tank level (75%), not low level (40%).

D. Correct.

000026 Loss of Component Cooling Water / 8

026AG2.1.23; Ability to perform specific system and integrated plant procedures during all modes of plant operation.

(CFR: 41.10 / 43.5 / 45.2 / 45.6)

Importance Rating:	4.3 4.4
Technical Reference:	APP-ALB-005, Window 6-1, Page 39, Rev. 25
References to be provided:	None
Learning Objective:	CCWS-ILC Objective 7.d
Question Origin:	Bank
Comments:	K/A is matched since the applicant must demonstrate the ability verify that specific system automatic actions have occurred in accordance with an annunciator response procedure.
Tier/Group:	T1/G1

9. 2020 NRC RO 009

Given the following plant conditions:
- The unit is in Mode 3 at normal operating pressure
- Pressurizer (PRZ) Pressure Control is in AUTO

Subsequently:
- A PRZ pressure transmitter failure occurs
- PRZ pressure channel indications are:
  - PI-444 2050 psig
  - PI-445 2500 psig
  - PI-455 2050 psig
  - PI-456 1950 psig
  - PI-457 2050 psig

Which ONE of the following completes the statements below regarding the expected conditions of the PRZ PORVs and spray valves in accordance with AOP-019, Malfunction of RCS Pressure Control?

PRZ PORVs 1RC-116 (PCV-445B) and 1RC-118 (PCV-445A) will be __(1)__.

The PRZ spray valves (PCV-444C and PCV-444D) will be __(2)__.

(Assume NO operator actions)

A. (1) OPEN

(2) OPEN

B. (1) OPEN

(2) SHUT

C. (1) SHUT

(2) OPEN

D. (1) SHUT

(2) SHUT

Reason answer is correct: PT-445 controls PRZ PORVs 1RC-116 and 1RC-118. The PORVs will open and remain open until protection channels (455/456/457) drop below the P-11 setpoint of 2000 psig (2/3 logic). The spray valves are controlled by PT-444 and will be unaffected by the PT-445 failure.

A. Incorrect. The first part is correct. The second part is plausible since another channel (PT-444) failing high would cause the spray valves to open as well as opening PORV 1RC-114); however, this is incorrect as the spray valves are not controlled by PT-445.

B. Correct.

C. Incorrect. The first part is plausible since the PORVs would close once pressure drops below the P-11 setpoint of 2000 psig; however, this is incorrect since this is a 2/3 logic and only one protection channel is less than 2000 psig. The second part is plausible since another channel (PT-444) failing high would cause the spray valves to open as well as opening PORV 1RC-114); however, this is incorrect as the spray valves are not controlled by PT-445.

D. Incorrect. The first part is plausible since the PORVs would close once pressure drops below the P-11 setpoint of 2000 psig; however, this is incorrect since this is a 2/3 logic and only one protection channel is less than 2000 psig. The second part is correct.

000027 Pressurizer Pressure Control System Malfunction / 3

027AK2.03; Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners

(CFR 41.7 / 45.7)

Importance Rating:	2.6 2.8
Technical Reference:	AOP-019, Attachment 3, Page 21, Rev. 25
References to be provided:	None
Learning Objective:	AOP-LP-3.19 Objective 5
Question Origin:	Bank (2012 NRC RO 38)
Comments:	K/A is matched since the applicant must demonstrate an understanding of how a malfunctioning PRZ pressure transmitter affects the PRZ PORV and spray valve controllers.
Tier/Group:	T1/G1

10. 2020 NRC RO 010

Which ONE of the following completes the statements below regarding an ATWS?

Reactor Trip Breaker shunt trip coils are ___(1)___ to actuate devices.

In accordance with EOP-FR-S.1, Response to Nuclear Power Generation/ATWS, if the Reactor fails to trip following opening of the Reactor Trip and Bypass Breakers locally, the next PREFERRED action is to open the rod drive MG set ___(2)___ breakers.

A. (1) energize

(2) motor

B. (1) energize

(2) generator output

C. (1) de-energize

(2) motor

D. (1) de-energize

(2) generator output

Plausibility and Answer Analysis

Reason answer is correct: Each Reactor Trip Breaker (and Reactor Trip Bypass Breaker) has redundant trip coils: an undervoltage trip coil and a shunt trip coil. The undervoltage trip coil is maintained energized by the output of the SSPS logic bay when a trip signal is not active. When a Reactor trip is initiated by SSPS (or manually), power to the UV trip coil is removed, causing the breaker to open. The shunt trip coil, normally de-energized, is energized by 125 VDC power when its associated shunt trip relay de-energizes. Tripping the Reactor locally can be performed by performing ANY of the following (listed in order of preference): 1) trip both Reactor Trip Breakers, 2) trip both Reactor Trip Bypass Breakers (normally open and racked out), 3) trip both rod drive MG set generator output breakers, and 4) trip both rod drive MG set motor breakers.

A. Incorrect. The first part is correct. The second part is plausible since this is a method used in EOP-FR-S.1 to trip the Reactor locally, but it is the least preferred method.

B. Correct.

C. Incorrect. The first part is plausible since the Reactor Trip Breakers have undervoltage trip coils which de-energize to actuate; however, this is incorrect since the shut trip coils energize to actuate. The second part is plausible since this is a method used in EOP-FR-S.1 to trip the Reactor locally, but it is the least preferred method.

D. Incorrect. The first part is plausible since the Reactor Trip Breakers have undervoltage trip coils which de-energize to actuate; however, this is incorrect since the shut trip coils energize to actuate. The second part is correct.

000029 Anticipated Transient Without Scram / 1

029EK2.06; Knowledge of the interrelations between the and the following an ATWS: Breakers, relays, and disconnects

(CFR 41.7 / 45.7)

Importance Rating:	2.9 3.1
Technical Reference:	Drawings Emdrac 1364-0865 CWDs 6-B-401 Sheet 91 (A train) and Sheet 93 (B train) EOP-FR-S.1, Step 9, Page 9, Rev. 7
References to be provided:	None
Learning Objective:	ILC-RPS Objective 2.b EOP-LP-3.15 Objective 4.a
Question Origin:	New
Comments:	K/A is matched since the applicant must recall how the shut trip relays/coils function to mitigate an ATWS. Applicant must also determine which rod drive MG set breakers should be opened first to trip the reactor.
Tier/Group:	T1/G1

11. 2020 NRC RO 011

Given the following plant conditions:
- A tube rupture occurred in the 'A' SG
- Offsite power was lost
- The crew completed EOP-E-3, Steam Generator Tube Rupture, and transitioned to EOP-ES-3.1, Post-SGTR Cooldown Using Backfill

The following plant conditions presently exist:
- 6.9 KV Aux Buses 'A' and 'C' have been re-energized
- The crew is preparing to restart an RCP

Which ONE of the following completes the statement below?

In accordance with EOP-ES-3.1, the ___(1)___ RCP should be started FIRST to minimize any challenges to ___(2)___.

A. (1) 'A'

(2) vessel integrity

B. (1) 'A'

(2) core reactivity

C. (1) 'C'

(2) vessel integrity

D. (1) 'C'

(2) core reactivity

Reason answer is correct:

EOP-ES-3.1 NOTE: RCPs should be run in order of priority (B,A,C) to provide normal PRZ spray. (IF the preferred RCP is in the loop with the ruptured SG, THEN a different RCP should be started prior to starting the preferred one.)

EOP-ES-3.1 CAUTION: To prevent inadvertent criticality following natural circulation cooldown AND initiation of backfill, the RCP in the ruptured loop should NOT be the first RCP restarted.

A. Incorrect. The first part is plausible since the 'A' RCP is desired for spray flow (pressure control); however, this is incorrect since the first RCP restarted should not be in the ruptured loop. The second part is plausible since mixing the RCS will equalize loop temperatures (no stagnant cold water flowing to the downcomer) which would minimize integrity challenges; however, this is incorrect since the concern is inadvertent criticality.

B. Incorrect. The first part is plausible since the 'A' RCP is desired for spray flow (pressure control); however, this is incorrect since the first RCP restarted should not be in the ruptured loop. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since mixing the RCS will equalize loop temperatures (no stagnant cold water flowing to the downcomer) which would minimize integrity challenges; however, this is incorrect since the concern is inadvertent criticality.

D. Correct.

000038 Steam Generator Tube Rupture / 3

038EK1.03; Knowledge of the operational implications of the following concepts as they apply to the SGTR: Natural circulation

(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	3.9 4.2
Technical Reference:	EOP-ES-3.1, NOTE and CAUTION preceding Step 2, Page 6, Rev. 3
References to be provided:	None
Learning Objective:	EOP-LP-3.04 Objective 4
Question Origin:	Bank (2009B NRC SRO 24)
Comments:	K/A is matched since application must evaluate plant conditions (natural circulation) and determine which RCP must be re-started first to prevent an RCS dilution event. The question is testing knowledge of the operational implication of re-starting RCPs with a SGTR in progress.
Tier/Group:	T1/G1

12. 2020 NRC RO 012

Given the following plant conditions:

- A complete loss of all feedwater sources occurred
- RCS bleed and feed has been initiated

Subsequently:

- All SGs are completely dry and depressurized
- Auxiliary Feedwater (AFW) capability is restored
- RCS temperature is stable

Which ONE of the following completes the statements below?

In accordance with EOP-FR-H.1, Response to a Loss of Secondary Heat Sink, one intact SG will be fed using AFW at ___(1)___ KPPH.

The reason ONLY one SG is fed is to ensure ___(2)___.

A. (1) 50

(2) a failure due to excessive thermal stresses is limited to one SG

B. (1) 50

(2) RCS cooldown rates are maintained within Technical Specification limits

C. (1) 200

(2) a failure due to excessive thermal stresses is limited to one SG

D. (1) 200

(2) RCS cooldown rates are maintained within Technical Specification limits

Plausibility and Answer Analysis

Reason answer is correct: In accordance with EOP-FR-H.1, Attachment 1, with core exit TCs stable or dropping. AFW flow is limited to 50 KPPH. One SG is fed to minimize thermal shock and potential damage to the SG tubesheet when SGs are hot and dry. If a failure in an SG occurs due to excessive thermal stresses, the failure is isolated to one steam generator.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since EOP-FR-H.1 cautions the operator to control feedwater rates to prevent excessive cooldown for enhanced plant control; however, this is incorrect as complying with Tech Spec requirements is not the basis per the background document.

C. Incorrect. The first part is plausible since this is the AFW feed limit that would be used if restored before RCS bleed and feed was established. The second part is correct.

D. Incorrect. The first part is plausible since this is the AFW feed limit that would be used if restored before RCS bleed and feed was established. The second part is plausible since EOP-FR-H.1 cautions the operator to control feedwater rates to prevent excessive cooldown for enhanced plant control; however, this is incorrect as complying with Tech Spec requirements is not the basis per the background document.

000054 Loss of Main Feedwater /4

054AK3.04; Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Actions contained in EOPs for loss of MFW

(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	4.4 4.6
Technical Reference:	EOP-FR-H.1, Attachment 1, Page 70, Rev .5 SDD-FR-H.1, Attachment 1, Page 25, Rev. 1 FR-H.1 Background, Section 2.4, Page 29, Rev. 3
References to be provided:	None
Learning Objective:	EOP-LP-3.11 Objective 5.c
Question Origin:	Modified (2016 NRC RO 10)
Comments:	Ask Chief Examiner if acceptable to ask AOP actions addressing a loss of MFW. Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by asking a question about EOP-FR-H.1 where you cannot establish MFW. K/A is matched since the applicant must recall the reason for feeding only one SG when restoring secondary heat sink.
Tier/Group:	T1/G1

13. 2020 NRC RO 013

Given the following plant conditions:
- The unit is operating 100% power
- OST-1073, 1B-SB Emergency Diesel Generator Operability Test, in progress
- Emergency Diesel Generator 1B-SB is loaded to 6300 KW while operating in parallel with the grid

Subsequently:
- EDG 1B-SB output breaker (126) trips open then recloses

Which ONE of the following identifies an event that would cause breaker 126 to operate in this manner?

A. Safety Injection actuates

B. A loss of offsite power occurs

C. A Main Generator lockout trips

D. Breaker 124, Aux Bus 1E to Emergency Bus B-SB, opens

Plausibility and Answer Analysis

Reason answer is correct: If a loss of offsite power (LOOP) occurs while EDG 1B-SB is paralleled to the grid, breakers 125 and 126 should automatically trip which will leave the EDG running unloaded. Breaker 126 should then automatically reclose (due to the bus undervoltage condition) and the sequencer start to load Emergency Bus 1B-SB.

The LOOP signal used to open breakers 105 (125) and 106 (126) is generated by:

- Both breakers 101 (121) and 102 (122) open

OR

- Breaker 101 (102) open and either main generator lockout tripped

A. Incorrect. Plausible since a Safety Injection signal will open the EDG output breaker; however, this is incorrect since offsite power would still be supplying the emergency bus. The EDG output breaker would need to sense a UV (86) condition on the emergency bus to allow re-closure.

B. Correct.

C. Incorrect. Plausible since this is part of the LOOP circuit (as discussed above); however, this is incorrect since Breaker 102 would have shut on the fast bus transfer.

D. Incorrect. Plausible since this breaker 125 will open on interlock if breaker 124 opens; however, this is incorrect since the EDG output breaker would remain closed and continue to carry the load on the emergency bus.

000056 Loss of Offsite Power / 6

056AA2.44; Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Indications of loss of offsite power

(CFR: 43.5 / 45.13)

Importance Rating:	4.3 4.5
Technical Reference:	OP-155, Section 4.0, P&L #24, Page 9, Rev. 91
References to be provided:	None
Learning Objective:	EDG-ILC Objective 8.d
Question Origin:	New
Comments:	K/A is matched since applicant must evaluate plant conditions and determine that the one indication provided (EDG breaker cycling open then shut) would occur if a loss of offsite power occurred.
Tier/Group:	T1/G1

14. 2020 NRC RO 014

Given the following plant conditions:

- The unit is in Mode 6
- 'A-SA' Safety Train is in service
- Core Alterations are in progress
- Nuclear Flux Monitoring System (NFMS) N60 is being substituted for SR N31
- Source Range (SR) N32 is providing audible count rate in the MCR and CNMT

In accordance with Technical Specifications, which ONE of the following identifies a condition that would require suspension of Core Alterations?

A. RWST level lowers to 23%

B. 'B' EDG is declared inoperable

C. Instrument Bus IDP-1B-SII de-energizes

D. Equipment hatch is determined to be secured with 8 bolts

Reason answer is correct: GP-009 P&L #2 provides a list of times when Core Alterations must be suspended. Loss of Instrument Bus II results in a loss of NI-32 and audible count rate. When fewer than two Source Range Monitors are operable, or audible count rate indication is lost in the MCR or in Containment, core alterations shall be suspended. This same requirement is addressed by Tech Spec 3.9.2, ACTION a.

A. Incorrect. Plausible since this choice would be correct if level lowered to 12%. Tech Spec 3.1.2.5 (Modes 5 and 6) requires RWST level maintained > 12% (106,000 gallons) while fuel is in the Rx Vessel. Additionally, another plausibility of 23% is required BAT level.

B. Incorrect. Plausible since the applicant must consider the availability of EDGs during fuel movement. Tech Spec 3.8.1.2 requires that at least one EDG must be operable in order to continue fuel movement. There is no indication that the 'A' EDG is not available; therefore, suspension of core alterations would not be required.

C. Correct.

D. Incorrect. Plausible since Tech Spec 3.9.4 requires immediately suspending Core Alterations if the LCO for Containment Building Penetrations is not met with the equipment door being one of the penetrations addressed. However, the equipment door is only required to be held in place by a minimum of four bolts.

000057 Loss of Vital AC Instrument Bus / 6

057AG2.2.38; Knowledge of conditions and limitations in the facility license.

(CFR: 41.7 / 41.10 / 43.1 / 45.13)

Importance Rating: 3.6 4.5

Technical Reference: AOP-024, Attachment 2, Page 34, Rev. 60
GP-009, P&L #2, Pages 6 & 7, Rev. 68
Technical Specification 3.9.2, Page 3/4 9-3

References to be provided: None

Learning Objective: GP-LP-3.9 Objective 3

Question Origin: Bank (2009B NRC 13)

Comments: K/A is matched since applicant must determine the impact of a loss of an instrument bus during refueling operations and demonstrate an understanding of the impact this will have on core alterations (GP-009 and Tech Spec requirement for refueling).

Tier/Group: T1/G1

15. 2020 NRC RO 015

Given the following plant conditions:

- A LOCA has occurred
- 'A' ESW Booster Pump has tripped
- Containment pressure is 28 psig

Which ONE of the following completes the statement below?

In accordance with EOP-FR-Z.1, Response to High Containment Pressure, ESW to the 'A' Train Containment Fan Coolers is isolated to prevent _______.

A. an unmonitored release from Containment to the ESW system

B. infusion of hydrogen into the ESW system from the Containment atmosphere

C. damage to the Containment fan coolers from water hammer if the ESW Booster pump is restarted

D. damage to the containment fan coolers from water hammer due to ESW flashing to steam in piping inside Containment due to low fan cooler flow

Plausibility and Answer Analysis

Reason answer is correct: The ESW Booster Pump is provided to ensure that cooling water pressure inside the Containment fan cooler units is higher than Containment pressure during a LOCA. This prevents leakage of Containment radioactivity into the ESW system. An orifice downstream of the fan cooler units provides increased system resistance during booster pump operation. The booster pump is placed in service by an SI or LOSP sequencer actuation. Start of the booster pump causes the orifice to be placed into service by closing the orifice bypass valve. Flow bypasses the booster pump and orifice during normal plant operation. If the ESW Booster pump trips then the function of providing increased system resistance is not occurring; therefore, isolating the Containment Fan Cooler will prevent an unmonitored release from Containment occurring.

A. Correct.

B. Incorrect. Plausible since infusion of hydrogen from the Containment atmosphere into the ESW system is possible (as with the correct answer); however, this is incorrect as this is not the reason for isolating the fan coolers.

C. Incorrect. Plausible since damage to Containment fan cooler ESW piping has occurred when ESW was lost and restarted on a depressurized header. This could be construed to apply to restarting the ESW Booster pump; however, this is incorrect as the ESW header will remain pressurized with the ESW pump running.

D. Incorrect. Plausible since Containment temperature will be significant at 28 psig and it could be mistaken that lower ESW flow exists (lower ESW pressure does actually exist) with the ESW Booster pump tripped; however, this is incorrect as this is not the reason for isolating the fan coolers.

000062 Loss of Nuclear Service Water / 4

062AK3.03; Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Guidance actions contained in EOP for Loss of nuclear service water

(CFR 41.4, 41.8 / 45.7)

Importance Rating:	4.0 4.2
Technical Reference:	EOP-FR-Z.1, Step 8, Page 8, Rev. 2 SDD-FR-Z.1, Page 3, Rev. 0
References to be provided:	None
Learning Objective:	EOP-LP-3.13 Objective 5 SWS-ILC Objective 2.c
Question Origin:	Bank (2011 NRC RO 13)
Comments:	K/A is matched since applicant must demonstrate an understanding as to why ESW to the Containment Fan Coolers is isolated when a loss of an ESW Booster Pump occurs.
Tier/Group:	T1/G1

16. 2020 NRC RO 016

Given the following plant conditions:
- The unit is operating at 100% power
- Air Compressor 1C is the lead compressor
- Air Compressor 1B is under clearance for inspection
- Air Compressor 1A is in STANDBY and isolated from CAS Panel
- Instrument Air header pressure is 110 psig

Subsequently:
- Instrument Air header pressure begins to lower steadily

With regard to AOP-017, Loss of Instrument Air, which ONE of the following completes the statements below?

The HIGHEST value that Air Compressor 1A will start on lowering Instrument Air header pressure is ___(1)__ psig.

If Instrument Air header pressure continues to lower, the operators are FIRST required to manually trip the Reactor when pressure lowers to ___(2)___ psig.

A. (1) 96

(2) 60

B. (1) 96

(2) 35

C. (1) 90

(2) 60

D. (1) 90

(2) 35

Plausibility and Answer Analysis

Reason answer is correct: Air Compressor 1A will start at 96 psig when in STANDBY and isolated from CAS Panel. The FW regulating valves receive a shut signal when pressure falls to 60 psig on the Control Air header. A continuous action step in AOP-017 has the crew trip the Reactor when Main Feedwater flow to ALL Steam Generator cannot be maintained.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since at 35 psig all air-operated valves are no longer considered reliable; however, this is incorrect since the unit must be tripped at 60 psig due to the loss of all Main Feedwater flow.

C. Incorrect. The first part is plausible since 90 psig is a significant Instrument Air pressure setpoint (Service Air isolated from Instrument Air); however, this is incorrect since Air Compressor 1A will start at 95 psig. The second part is correct.

D. Incorrect. The first part is plausible since 90 psig is a significant Instrument Air pressure setpoint (Service Air isolated from Instrument Air); however, this is incorrect since Air Compressor 1A will start at 95 psig. The second part is plausible since at 35 psig all air-operated valves are no longer considered reliable; however, this is incorrect since the unit must be tripped at 60 psig due to the loss of all Main Feedwater flow.

000065 Loss of Instrument Air / 8

065AA1.04; Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Emergency air compressor

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.5 3.4
Technical Reference:	AOP-017, Section 3.0 Step 1 & Attachment 7, Pages 4 & 57, Rev. 40
References to be provided:	None
Learning Objective:	AOP-LP-3.17 Objective 2.a & 2.c
Question Origin:	Bank (Comanche Peak)
Comments:	Ask Chief Examiner what is considered an emergency air compressor. Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by testing operation of the 'A' and/or 'B' air compressors. K/A is matched since applicant must monitor plant conditions and determine when the backup (emergency) air compressor will automatically start on a loss of Instrument Air.
Tier/Group:	T1/G1

17. 2020 NRC RO 017

Given the following plant conditions:
- The unit is operating at 100% power
- MDAFW pump 'B' is under clearance

Subsequently the following occurs:
- A manual Reactor Trip was initiated due to a loss of the 'A' MFP
- The TDAFW pump tripped after starting
- MDAFW flow control valves are full open
- Total AFW flow is 212 KPPH and lowering
- SG NR levels are 41% and lowering
- Containment pressure is 2.8 psig and stable

Which ONE of the following would be the FIRST set of conditions that would require entry into EOP-FR-H.1, Response to Loss of Secondary Heat Sink?

All SG NR levels are __(1)__% AND total AFW flow is __(2)__ KPPH.

A. (1) 39

(2) 195

B. (1) 39

(2) 205

C. (1) 24

(2) 195

D. (1) 24

(2) 205

Reason answer is correct: Heat Sink CSFST indicates a loss of heat sink if AFW flow is less than 200 KPPH AND ALL SG NR levels are less than 25% with normal Containment conditions (40% adverse Containment conditions).

A. Incorrect. The first part is plausible since this level is less than the adverse Containment requirement (this would be the correct choice with adverse CNMT conditions). The second part is correct.

B. Incorrect. Plausible since the applicant may believe adverse Containment conditions exist and believe that only one of the requirements must be met (vice both) for entry into EOP-FR-H.1. In this case, SG NR levels would be less than the required value of 39%.

C. Correct

D. Incorrect. Plausible since the applicant may believe that only one of the requirements must be met (vice both) for entry into EOP-FR-H.1. In this case, SG NR levels are be less than the required value of 25%.

W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4

WE05EK2.2; Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

(CFR: 41.7 / 45.7)

Importance Rating:	3.9 4.2
Technical Reference:	EOP-CSFST, CSF-3 Heat Sink, Rev. 13
References to be provided:	None
Learning Objective:	EOP-LP-3.11 Objective 4
Question Origin:	Modified (2018 NRC RO 17)
Comments:	K/A is matched since the candidate must recall when a degradation in the AFW system (SG levels & flow) will lead to a loss of secondary heat sink condition.
Tier/Group:	T1/G1

18. 2020 NRC RO 018

Given the following plant conditions:

- The unit was operating at 100% power when a Reactor Trip and Safety Injection occurred due to a steam line break in Containment on the 'B' SG

Current plant conditions are as follows:

- Containment pressure is 28 psig

The crew has transitioned from EOP-E-0, Reactor Trip or Safety Injection and are at Step 1 of EOP-E-2, Faulted Steam Generator Isolation.

Which ONE of the following identifies the set of valves listed below that the operator must ensure are in the SHUT position for the conditions above?

1. All MSIV's
2. 1MS-70, Main Steam B to Aux FW Turbine
3. 'B' SG MDAFW AND TDAFW motor isolation valves
4. ONLY 'B' MSIV
5. All Blowdown isolation valves
6. 1SI-3, BIT Outlet

A. 1, 2, and 3

B. 4, 5, and 6

C. 1, 3, and 5

D. 2, 4, and 6

Reason answer is correct: During a Main Steam Line break inside Containment, a Safety Injection signal caused by Containment pressure exceeding 3 psig occurs. For the given conditions, a Main Steam Line Isolation signal and AFW Isolation signal would also have been generated

A MSLI signal, which would have been caused by either: Steamline low pressure (2/3 steamline pressures on any SG < 601 psig) or Containment Pressure > 3.0 psig. The MSLI signal will ensure that only one SG depressurizes following a steamline break upstream of the Main Steam Isolation Valves (MSIVs). The main steamline isolation signal automatically shuts the MSIVs (1MS-80, 1MS-82, 1MS-84), MSIV bypass valves (1MS-81, 1MS-83, 1MS-85), and MSIV before seat drain valves (1MS-231, 1MS-266, 1MS-301).

An AFW isolation is initiated if two of three differential pressures indicate any SG is 100 psi below the other two SGs and a main steam line isolation signal is present. The AFW isolation signal shuts the AFW flow control valves and the AFW isolation MOVs to the affected SG (from the MDAFW pumps and the TDAFW pump). The AFW isolation signal isolates AFW flow to a faulted SG on loss of secondary coolant to limit further RCS cooldown. For the given conditions, a large steam line break is present in the 'B' SG indicated by the Containment pressure of 28 psig. This large steam break would have caused the 'B' SG pressure to have decreased to > 100 psig below the 'A' and 'C' SG's by the time the crew had begun implementation of EOP-E-2.

A. Incorrect. Plausible since the MSIV's and AFW isolation valves shut, but 1MS-70 would be OPEN. 1MS-70 would be procedurally isolated during the implementation of E-2.

B. Incorrect. Plausible since 'B' MSIV and Blowdown would be isolated, but 1SI-3 would be OPEN. Plausibility of 1SI-3 is made because the valve is shut during the implementation of EOP-E-2. Since the given information is that the crew is at the beginning of implementation of E-2, the step to shut 1SI-3 would not have be implemented yet.

C. Correct.

D. Incorrect. Plausible since the 'B' MSIV would be isolated and the other 2 valves would be procedurally isolated during the implementation of E-2. Since the given information is that the crew is at the beginning of implementation of EOP-E-2 the step to shut 1MS-70 or 1SI-3 would not have been implemented yet.

W/E12 Steam Line Rupture—Excessive Heat Transfer / 4

WE12EG2.1.20; Ability to interpret and execute procedure steps.

(CFR: 41.10 / 43.5 / 45.12)

Importance Rating:	4.6 4.6
Technical Reference:	EOP-E-0, Attachment 3, Steps 7, 8, and 14, Pages 60 & 61, Rev. 15
References to be provided:	None
Learning Objective:	ESFAS-ILC Objective 8 EOP-LP-3.22 Objective 5 EOP-LP-3.09 Objective 5
Question Origin:	Bank (2014 NRC RO 10)
Comments:	Confirm with Chief Examiner that the intended K/A is WE12 EG2.1.20. Phonecon 4/14: Chief Examiner stated that the intended K/A is WE12 EG2.1.20. K/A is matched since applicant must demonstrate the ability to interpret and execute EOP steps (EOP-E-0 and EOP-E-2) during a Main Steam Line Break event. Both procedures have steps the operator must execute to ensure proper isolation of the faulted SG.
Tier/Group:	T1/G1

19. 2020 NRC RO 019

Which ONE of the following completes the statements below regarding recovery of a dropped rod in accordance with AOP-001, Malfunction of Rod Control and Indication System?

Prior to recovering the dropped rod, the lift coil disconnect switches for all rods except the dropped will be opened in the affected __(1)__.

During the recovery, withdrawal of the dropped rod will be stopped based on __(2)__.

A. (1) bank

(2) step counter position

B. (1) bank

(2) DRPI for the affected rod

C. (1) group ONLY

(2) step counter position

D. (1) group ONLY

(2) DRPI for the affected rod

Reason answer is correct: The basic sequence for recovery of a dropped rod is to open the lift coil disconnect switches for all rods in the affected bank except for the dropped rod, recording the step counter reading for the affected group and then setting it to zero, and then withdrawing the dropped rod until reaching the step counter reading just recorded.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since DRPI for the affected rod is checked AFTER rod withdrawal is stopped to determine if there may be a position indication problem or a problem with the rod being moved; however, this is incorrect since rod withdrawal is stopped based on step counter position.

C. Incorrect. The first part is plausible since the step counter for the affected GROUP is set to ZERO; however, this is incorrect since the lift coil disconnect switches must be lifted for all rods (except for the dropped rod) in the affected BANK. The second part is correct.

D. Incorrect. The first part is plausible since the step counter for the affected GROUP is set to ZERO; however, this is incorrect since the lift coil disconnect switches must be lifted for all rods (except for the dropped rod) in the affected BANK. The second part is plausible since DRPI for the affected rod is checked AFTER rod withdrawal is stopped to determine if there may be a position indication problem or a problem with the rod being moved; however, this is incorrect since rod withdrawal is stopped based on step counter position.

000003 Dropped Control Rod / 1

003AK3.05; Knowledge of the reasons for the following responses as they apply to the Dropped Control Rod: Reset of demand position counter to zero

(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	2.7 3.0
Technical Reference:	AOP-001-BD, Section 3.1, Page 3, Rev. 23
References to be provided:	None
Learning Objective:	AOP-LP-3.1 Objective 3.f
Question Origin:	New
Comments:	Ask Chief Examiner how to address reason for reducing power since this comes from our TS bases. Phonecon 4/14/2020: Chief Examiner stated that the reason power must be reduced can be something as simple as being outside the QPTR or AFD limits. Recommendation is to provide conditions where QPTR or AFD is outside allowable limits which the candidate will need to recognize and take appropriate action. The reasons can be at the RO level. Ask Chief Examiner for new K/A based on overlap concerns with operating exam. Phonecon 8/20/2020: HNP discussed being concerned with overlap between the Operating Test and this Written Exam K/A, so Chief Examiner selected a new K/A, keeping APE topic 003, Dropped Control Rod. New K/A 003AK3.06: Knowledge of the reasons for the following responses as they apply to the Dropped Control Rod: Reset of demand position counter to zero. K/A is matched since the applicant must demonstrate an understanding of how step counters are used during the recovery of a dropped rod.
Tier/Group:	T1/G2

20. 2020 NRC RO 020

Given the following plant conditions:

- The crew is implementing AOP-002, Emergency Boration
- Boration is occurring via 1CS-278, Emergency Boric Acid Addition
- Boric Acid flowrate (FI-110) indicates 25 gpm

With regard to AOP-002, which ONE of the following completes the statements below?

Boric acid flow to the CSIP suction __(1)__ adequate.

The NEXT required operator action is to __(2)__.

A. (1) is

(2) establish adequate charging flow to the RCS

B. (1) is

(2) control charging and letdown to maintain normal PRZ level

C. (1) is NOT

(2) establish emergency boration flow via the blender to the CSIP suction

D. (1) is NOT

(2) establish emergency boration flow via the blender to the top of the VCT

Reason answer is correct: Flowpaths for delivering sufficient boric acid to the RCS are listed in order of preference as follows:

a. Emergency boration flowpath
   * Via Emergency Boration Valve 1CS-278, to the CSIP suction
     - OR -
   * Via normal path through FCV-113A and FCV113B, to the CSIP suction
b. RWST path via LCV-115B and/or LCV-115D, to the CSIP suction
c. Alternate path via FCV-113A and FCV-114A, to the inlet of the VCT

Tech Spec LCOs 3.1.1.1 and 3.1.1.2 action statements require immediate initiation of flow at equal or greater than 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until required SDM is restored.

As such the minimum required flow rate from the BAT is 30 gpm and the minimum required flow rate from the RWST is 90 gpm.

A. Incorrect. The first part is plausible since this is a typical boric acid flow rate seen during auto makeup to the VCT; however, this is incorrect since the minimum required boric acid flow is 30 gpm from the BAT. The second is plausible since this is an action directed by AOP-002 if boric acid flow is adequate (≥ 30 gpm).

B. Incorrect. The first part is plausible since this is a typical boric acid flow rate seen during auto makeup to the VCT; however, this is incorrect since the minimum required boric acid flow is 30 gpm from the BAT. The second is plausible since this is an action directed by AOP-002 if boric acid flow is adequate (≥ 30 gpm).

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since this is a method directed by AOP-002 to deliver boric acid to the RCS; however, this is incorrect since this is the least preferred flow path (alternate). AOP-002 would first direct use the normal and RWST paths before utilizing this flow path.

000024 Emergency Boration / 1

024AA2.06; Ability to determine and interpret the following as they apply to the Emergency Boration: When boron dilution is taking place

(CFR: 43.5 / 45.13)

Importance Rating:	3.6 3.7
Technical Reference:	AOP-002-BD, Sections 1.0 and 3.0, Page 3, Rev. 6 AOP-002, Steps 2 & 3, Page 4, Rev. 24
References to be provided:	None
Learning Objective:	AOP-LP-3.02 Objective 2.a
Question Origin:	New
Comments:	K/A is matched since the applicant must interpret plant conditions and determine that emergency boration flow is inadequate and actions that must be taken to mitigate the event in progress.
Tier/Group:	T1/G2

21. 2020 NRC RO 021

Given the following plant conditions:

- A Refueling Water Storage Tank (RWST) leak has occurred
- Tank Area Drains are being pumped to the Storm Drain System in accordance with OP-120.09.01, Radioactive Floor Drain Collection
- REM-01MD-3530, Tank Area Drain Transfer Pumps Monitor, is in HIGH alarm
- Contaminated water is filling the retention dike area

Which ONE of the following completes the statements below?

As a result of this radiation alarm, __(1)__ automatically.

In accordance with AOP-008, Accidental Release of Liquid Waste, a leak from the RWST requires manual operation to __(2)__.

A. (1) the Tank Area Drain Transfer Pump stops

(2) shut 1FD-109, FD Tank Area Drain Pump 1X Discharge to Storm Drain Valve

B. (1) the Tank Area Floor Drain Sump Pump stops

(2) shut 1FD-109, FD Tank Area Drain Pump 1X Discharge to Storm Drain Valve

C. (1) 1FD-109, FD Tank Area Drain Pump 1X Discharge to Storm Drain Valve shuts

(2) secure the Tank Area Drain Transfer pump

D. (1) 1FD-109, FD Tank Area Drain Pump 1X Discharge to Storm Drain Valve shuts

(2) secure the Tank Area Floor Drain Sump pump

Reason answer is correct: In accordance with AOP-005 Attachment 9, Liquid Waste Effluent Monitors, a high radiation alarm on REM-012MD-3530 will stop the "Tank Area Drain Transfer Pump". With the leak in the RWST, which is located in the "Tank Building", the RNO action for this situation is to SHUT 1FD-109, FD Tank Area Drain Pump Discharge to Storm Drains Valve.

A. Correct.

B. Incorrect. The first part is plausible since this pump is in the Tank Building; however, this is incorrect since it does not receive a signal from the Radiation Monitoring System to automatically secure on radiation conditions. The second part is correct.

C. Incorrect. The first part is plausible since HNP has a number of radiation monitors that shut valves on high rad alarms (e.g. REM-3540 shuts 3LHS-296 and REM-3541 shuts 3FD-421) and 1FD-109 needs to be shut on a high radiation alarm; however, this is incorrect since this is done manually. The second part is plausible because the Tank Area Drain Transfer Pump needs to be secured on a high radiation signal; however, this is incorrect since this action is done automatically.

D. Incorrect. The first part is plausible since HNP has a number of radiation monitors that shut valves on high rad alarms (e.g. REM-3540 shuts 3LHS-296 and REM-3541 shuts 3FD-421) and 1FD-109 needs to be shut on a high radiation alarm; however, this is incorrect since this is done manually. The second part is plausible because the Tank Area Drain Sump Pump is located in the Tank building; however, this is incorrect since it does not take a suction on the RWST pit and as such will not need to be secured.

000059 Accidental Liquid Radwaste Release / 9

059AK3.04; Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release: Actions contained in EOP for accidental liquid radioactive-waste release

(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	3.8 4.3
Technical Reference:	AOP-005, Attachment 9, Page 27, Rev. 30 AOP-009, Section 3.0, Page 5, Rev. 14
References to be provided:	None
Learning Objective:	AOP-LP-3.05 Objectives 1.a & 4 AOP-LP-3.08 Objective 3
Question Origin:	Bank (2013 NRC RO 23)
Comments:	Ask Chief Examiner if acceptable to examine AOP actions related to accidental release. Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by testing the AOP that addresses accidental radioactive-waste releases. Ask Chief Examiner if asking automatic and manual actions for a high radiation alarm meets the K/A. Phonecon 7/30: Chief Examiner stated that the reason could be automatic actions occurring or manual actions required to be taken due to high radiation detected by a radiation monitor.
Tier/Group:	T1/G2

22. 2020 NRC RO 022

Given the following plant conditions:

- Core reload is in progress
- A spent fuel assembly in the Fuel Handling Building (FHB) is damaged

FHB area radiation levels are rising with monitor status as follows:

- RM-*1FR-3565A-SA - HIGH ALARM
- RM-*1FR-3565B-SB - ALERT
- No other Spent Fuel Pool Area monitors are in alarm

Which ONE of the following completes the statement below in accordance with AOP-005, Radiation Monitoring System?

__(1)__ train(s) of FHB Ventilation Emergency Exhaust has (have) automatically started and FHB Normal Operating Floor Ventilation __(2)__ shutdown.

A. (1) ONLY 'A'

(2) has

B. (1) ONLY 'A'

(2) has NOT

C. (1) BOTH 'A' and 'B'

(2) has

D. (1) BOTH 'A' and 'B'

(2) has NOT

Reason answer is correct: Per AOP-005, any FHB Spent Fuel Pool Area monitor in HIGH ALARM automatically starts the FHB Ventilation Emergency Exhaust system. Per OP-170 Section 8.1, this automatic start is train specific; thus, with only an 'A' train monitor in HIGH ALARM, only the 'A' train of emergency exhaust will start. But a HIGH ALARM on EITHER train will secure and isolate normal ventilation in the FHB.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since the logic could require a HIGH ALARM on both trains to secure normal ventilation; however, this is incorrect since a HIGH ALARM on either train is all that is required. Also plausible since another FHB ventilation system (Below Operating Floor) would remain running.

C. Incorrect. The first part is plausible since both rad monitors would alarm on the RMS Display; however, this is incorrect as only the one in High Alarm will start the associated Emergency Exhaust Fan. The second part is correct.

D. Incorrect. The first part is plausible since both rad monitors would alarm on the RMS Display; however, this is incorrect as only the one in High Alarm will start the associated Emergency Exhaust Fan. The second part is plausible since the logic could require a HIGH ALARM on both trains to secure normal ventilation; however, this is incorrect since a HIGH ALARM on either train is all that is required. Also plausible since another FHB ventilation system (Below Operating Floor) would remain running.

000061 Area Radiation Monitoring System Alarms / 7

061AA1.01; Ability to operate and / or monitor the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Automatic actuation

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.6 3.6
Technical Reference:	AOP-005, Attachment 2, Page 13, Rev. 30 OP-170, Section 8.1, NOTES, Page 23, Rev. 40
References to be provided:	None
Learning Objective:	FHVS-ILC Objective 5.a
Question Origin:	New
Comments:	K/A is matched since applicant must demonstrate an understanding of how the FHB ventilation systems respond automatically to area radiation alarms.
Tier/Group:	T1/G2

23. 2020 NRC RO 023

Given the following plant conditions:

- The unit is operating at 100% power when an event occurs causing a Phase A containment isolation

Subsequently:

- Five minutes after Phase A actuated, the RO noted that several of the Phase A isolation valves did NOT shut automatically

In accordance with EOP-E-0, Reactor Trip or Safety Injection, which ONE of the following pairs of valves can be used to identify which train of Phase A failed to actuate?

(Assume only one valve in each pair automatically shut)

A. 1CC-207/208, CCW TO RCPS

B. 1CS-470/472, RCP SEAL RTN

C. 1CS-235/238, RCS CHRG VALVE

D. 1CS-7/8, LETDOWN ORIFICE A/B

Plausibility and Answer Analysis

Reason answer is correct: 1CS-470/1CS-472 are motor operated valves that are operated by separate train relay actuations to isolate RCP seal return flow from Containment.

A. Incorrect. Plausible since 1CC-207/208 receive separate isolation signals; however, this is incorrect since these valves receive a Phase B (NOT Phase A) Containment Isolation Signal.

B. Correct.

C. Incorrect. Plausible since 1CS-235/238 receive separate isolation signals; however, this is incorrect since these valves receive a Safety Injection Signal.

D. Incorrect. Plausible since 1CS-7/8 receive a Phase A Containment Isolation Signal; however, this is incorrect since both valves receive a Containment Isolation Signal from one Train only.

000069 Loss of Containment Integrity / 5

069AA1.01; Ability to operate and / or monitor the following as they apply to the Loss of Containment Integrity: Isolation valves, dampers, and electropneumatic devices

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.5	3.7
Technical Reference:	EOP-E-0, Attachment 3, Page 59, Rev. 15 OMM-004, Attachment 4, Page 47, Rev. 42	
References to be provided:	None	
Learning Objective:	ESFAS-ILC Objective 8.c	
Question Origin:	Bank (Braidwood)	
Comments:	K/A is matched since applicant must evaluate isolation valve positions to determine which train of Phase A (Containment Isolation Signal) failed.	
Tier/Group:	T1/G2	

24. 2020 NRC RO 024

Given the following plant conditions:

- A faulted Steam Generator inside Containment occurred
- The faulted Steam Generator was isolated
- Containment pressure peaked at 12 psig
- The crew is implementing EOP-ES-1.1, SI Termination, to re-establish RCP seal return flow to the Volume Control Tank (VCT)

The following annunciators are currently in alarm:

- ALB-001-5-1, Containment Isolation Phase B
- ALB-005-1-5B, Seal Water HX CCW Low Flow

Which ONE of the following identifies (1) the annunciator that must be cleared to allow the re-establishment of RCP seal return flow to the VCT AND (2) the reason why?

A. (1) ALB-001-5-1

(2) Allows re-opening of Phase B valves that shut to isolate the seal return flowpath to the VCT.

B. (1) ALB-001-5-1

(2) Allows re-opening of Phase B valves that shut to isolate CCW flow to the Seal Water Return Heat Exchanger.

C. (1) ALB-005-1-5B

(2) Provides assurance that CCW flow to the Seal Water Return Heat Exchanger is available to provide adequate seal return cooling.

D. (1) ALB-005-1-5B

(2) Provides assurance that CCW pressure is sufficient to minimize any in-leakage from the Seal Water Return Heat Exchanger when flow is restored.

Plausibility and Answer Analysis

Reason answer is correct: The seal water HX CCW Low Flow alarm (ALB-5-1-5B) must be CLEAR prior to opening the seal return Phase A isolation valves which will establish RCP seal return flow to the VCT. This ensures adequate CCW seal return cooling.

A. Incorrect. Plausible since Phase A isolates the #1 seal return flow path to the VCT, not Phase B.

B. Incorrect. Plausible since Phase B isolates CCW flow to the RCPs, Thermal Barrier HX, and RCP Bearing Oil Coolers, but not to the Seal Water Return Heat Exchanger.

C. Correct.

D. Incorrect. Plausible since higher CCW flow would equate to higher pressure in the Seal Water Return Heat Exchanger and a higher pressure would reduce leakage into the CCW system if a leak existed, but the reason is to ensure adequate seal cooling.

W/E02 SI Termination / 3

WE02EG2.4.45; Ability to prioritize and interpret the significance of each annunciator or alarm.

(CFR: 41.10 / 43.5 / 45.3 / 45.12)

Importance Rating:	4.1 4.3
Technical Reference:	EOP-ES-1.1, Step 25, Page 34, Rev. 3 SDD-ES-1.1, Page 13, Rev. 1
References to be provided:	None
Learning Objective:	CVCS-ILC Objective 5.g
Question Origin:	New
Comments:	K/A is matched since the applicant must interpret annunciators and determine which one will impact restoration of RCP seal return to the Volume Control Tank (part of SI Termination procedure).
Tier/Group:	T1/G2

25. 2020 NRC RO 025

The crew has transitioned to EOP-E-1, Loss of Reactor or Secondary Coolant, and is presently evaluating if the RHR System is capable of Cold Leg Recirculation.

Current plant conditions:
- Offsite power has been lost
- EDG 1B-SB has tripped
- CNMT Pressure is 17 psig and rising
- CNMT Wide Range Sump Level is reading 211 inches
- RVLIS Full Range Level is reading 38%
- RCS Wide Range Pressure is reading 225 psig
- Core Exit Thermocouples are reading 740°F
- Containment Spray Pump 'A' has tripped

Which ONE of the following identifies the procedure the crew is required to implement at this time?

A. EOP-FR-Z.2, Response to Containment Flooding

B. EOP-FR-C.2, Response to Degraded Core Cooling

C. EOP-FR-C.1, Response to Inadequate Core Cooling

D. EOP-FR-Z.1, Response to High Containment Pressure

Plausibility and Answer Analysis

Reason the answer is correct: Core Exit Temperature is above the 730°F value (740°F) and RVLIS full range level is less than 39% (38%) so a RED path exists for Core Cooling. Implementation of EOP-FR-C.1 is required.

A. Incorrect. Plausible since the containment sump level (211 inches) is greater than the 196 inch value for an ORANGE path to exist; however, this is incorrect due having a higher priority for both high containment pressure and inadequate core cooling.

B. Incorrect. Plausible since the core exit temperature (740°F) is greater than the 730°F value and RVLIS full range level is less than 39%; however, this is incorrect since EOP-FR-C.2 would only be entered if one of these conditions existed (ORANGE path). With both present, a RED path exists for Core Cooling.

C. Correct.

D. Incorrect. Plausible since an ORANGE path exists due to containment pressure being greater than 10 psig with no CNMT spray pump running; however, this is incorrect due to having a higher priority CSFST for Core Cooling.

W/E07 Inadequate Core Cooling (Saturated Core Cooling) / 4

WE07EA2.1; Ability to determine and interpret the following as they apply to the (Saturated Core Cooling): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

(CFR: 43.5 / 45.13)

Importance Rating:	3.2 4.0
Technical Reference:	EOP-CSFST Page 2, Rev. 13
References to be provided:	None
Learning Objective:	EOP-LP-3.10 Objective 9.a
Question Origin:	Bank (2016 NRC RO 26)
Comments:	Ask Chief Examiner is acceptable to test inadequate or degraded core cooling conditions (red/orange path). Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by testing Inadequate and Degraded Core Cooling procedures (red/orange path). K/A is matched since the applicant must evaluate plant conditions and determine that entry into the inadequate core condition EOP is required.
Tier/Group:	T1/G2

26. 2020 NRC RO 026

Given the following plant conditions:

- The crew is implementing EOP-E-3, Steam Generator Tube Rupture
- The ruptured SG has been identified

Which ONE of the following completes the statement below?

The ruptured SG PORV controller setpoint is required to be adjusted to __(1)__ AND placed in __(2)__ to prevent lifting the SG code safety valves.

A. (1) 1135 psig (87%)

(2) AUTO

B. (1) 1135 psig (87%)

(2) MANUAL

C. (1) 1145 psig (88%)

(2) AUTO

D. (1) 1145 psig (88%)

(2) MANUAL

Plausibility and Answer Analysis

Reason answer is correct: In accordance with the Westinghouse Owners Group (WOG) Background document for step descriptions of the Steam Generator Tube Rupture procedure (E-3), the setpoint for the ruptured SG PORV controller should be adjusted so the setpoint is greater than no-load pressure in order to minimize atmospheric releases from the rupture steam generator and less than the minimum safety valve setpoint to prevent lifting of the code safety valves, which at HNP there are 5 safety valves with lift settings of 1170, 1185, 1200, 1215, and 1230 psig. The 25 psig margin is a typical value to allow for opening of the PORV prior to lifting of the safety valve.

A. Incorrect. The first part is plausible since this is the SG PORV controller setpoint used during plant startup operations (GP-005). This higher controller setting is to accommodate plant startup by placing an artificial load on the Reactor without causing the SG PORVs to open. The second part is correct.

B. Incorrect The first part is plausible since this is the SG PORV controller setpoint used during plant startup operations (GP-005). This higher controller setting is to accommodate plant startup by placing an artificial load on the Reactor without causing the SG PORVs to open. The second part is plausible since the operator may place the SG PORV controller in MANUAL to adjust the setpoint; however, this is incorrect since the controller must be placed back in AUTO following the setpoint adjustment.

C. Correct.

D. Incorrect The first part is correct. The second part is plausible since the operator may place the SG PORV controller in MANUAL to adjust the setpoint; however, this is incorrect since the controller must be placed back in AUTO following the setpoint adjustment.

W/E13 Steam Generator Overpressure / 4

WE13EK2.1; Knowledge of the interrelations between the (Steam Generator Overpressure) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

(CFR: 41.7 / 45.7)

Importance Rating:	3.0 3.1
Technical Reference:	E-3 Background Document, Page 59, Rev. 3 EOP-E-3, Step 5, Rev. 7
References to be provided:	None
Learning Objective:	EOP-LP-3.02 Objective 2.a
Question Origin:	New
Comments:	K/A is matched since the applicant must demonstrate an understanding of controller setting adjustment for a ruptured SG PORV to prevent challenging safety reliefs.
Tier/Group:	T1/G2

27. 2020 NRC RO 027

Given the following plant conditions:
- Core offload is in progress

Subsequently:
- An irradiated fuel assembly is damaged while being withdrawn from the core
- AOP-013, Fuel Handling Accident, has been entered
- Containment area radiation levels are rising with monitor status as follows:
  - RM-01CR-3561ASA - not in alarm
  - RM-01CR-3561BSB - ALERT
  - RM-01CR-3561CSA - not in alarm
  - RM-01CR-3561DSB - HIGH ALARM

Which ONE of the following completes the statements below?

Containment Ventilation Isolation __(1)__ automatically initiated.

In accordance with AOP-013, the damaged fuel assembly __(2)__ required to be placed in a safe storage location prior to evacuating Containment.

A. (1) has

(2) is

B. (1) has

(2) is NOT

C. (1) has NOT

(2) is

D. (1) has NOT

(2) is NOT

Reason answer is correct: High alarm on 2/4 Containment radiation monitors will automatically initiate Containment Ventilation Isolation. With only one monitor in high alarm, Containment Ventilation Isolation would not have occurred. With ANY Containment radiation monitor in alert or high alarm, the FIRST action directed by AOP-013 is performing an evacuation of Containment.

A. Incorrect. The first part is plausible since two Containment monitors are in alarm; however, only one is in high alarm so Containment Ventilation Isolation will not have automatically occurred. The second part is correct.

B. Incorrect. The first part is plausible since two Containment monitors are in alarm; however, only one is in high alarm so Containment Ventilation Isolation will not have automatically occurred. The second part is plausible since this action is directed by AOP-013 if no Containment radiation monitors were in alert or high alarm; however, this is incorrect as two radiation monitors are in alarm. All personnel must immediately evacuate Containment.

C. Incorrect. The first part is correct. The second part is plausible since this action is directed by AOP-013 if no Containment radiation monitors were in alert or high alarm; however, this is incorrect as two radiation monitors are in alarm. All personnel must immediately evacuate Containment.

D. Correct.

W/E16 High Containment Radiation / 9

WE16EK1.2; Knowledge of the operational implications of the following concepts as they apply to the (High Containment Radiation): Normal, abnormal and emergency operating procedures associated with (High Containment Radiation).

(CFR: 41.8 / 41.10, 45.3)

Importance Rating:	2.7 3.2
Technical Reference:	AOP-005-BD, Section 1.0, Page 2, Rev. 13 AOP-013, Section 3.2, Page 10, Rev. 16
References to be provided:	None
Learning Objective:	ESFAS-ILC Objective 4 AOP-LP-3.13 Objective 3
Question Origin:	New
Comments:	K/A is matched since the applicant must demonstrate an understanding of the operational implication of a high radiation condition occurring in Containment when moving fuel (immediate evacuation).
Tier/Group:	T1/G2

28. 2020 NRC RO 028

Given the following plant conditions:

- The unit is in Mode 3
- ALB-007-4-2, VCT HIGH-LOW PRESS, has alarmed
- Actual VCT pressure is 15 psig

Which ONE of the following completes the statement below?

If VCT pressure continues to lower, RCP #1 Seal Leakoff flow will ___(1)___ and RCP #2 Seal Leakoff flow will ___(2)___.

A. (1) rise

(2) rise

B. (1) rise

(2) lower

C. (1) lower

(2) rise

D. (1) lower

(2) lower

Reason answer is correct: The shaft seal section consists of three devices. They are the No. 1 controlled leakage, film riding face seal, and the No. 2 and 3 rubbing face seals. During normal system operation, the charging pump provides approximately 8 gpm injection flow to each RCP. The injection enters the pump between the thermal barrier and the pump bearing. The flow is then divided with approximately 5 gpm flowing down past the thermal barrier into the RCS and approximately 3 gpm flowing up past the pump bearing. The outlet from the No. 1 seal discharges to the Volume Control Tank (VCT). RCP No. 1 seal leak off flow is normally approximately 3 gpm to the VCT. The VCT maintains a back pressure of at least 15 psig to ensure a flow through the No. 2 seal. Because #1 and the #2 seal operate in close association with one another, the automatic adjustment of one will affect the other. When VCT pressure lowers, the differential pressure across the #1 Seal will rise, causing the seal leakoff flow to rise. When this occurs, the flow to the #2 Seal from the #1 Seal will lower causing its leakoff flow to be lower.

A. Incorrect. The first part is correct. The second part is plausible since the #2 Seal leakoff will be affected by the low VCT pressure; however, this is incorrect since leakoff from the #2 seal will lower, not rise. The #2 Seal leakoff is being directly impacted by the lower flow from the #1 Seal.

B. Correct.

C. Incorrect. Plausible since the #1 and #2 Seal leakoffs will be affected by the low VCT pressure with one seal leakoff rising and the other lowering; however, this is incorrect since leakoff from the #1 Seal will rise and leakoff from the #2 Seal will lower, not vice versa.

D. Incorrect. The first part is plausible since the #1 Seal leakoff will be directly affected by the low VCT pressure; however, this is incorrect since leakoff from the #1 Seal will rise, not lower. The second part is correct.

003 Reactor Coolant Pump / 4

003K4.07; Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Minimizing RCS leakage (mechanical seals)

(CFR: 41.7)

Importance Rating:	3.2 3.4
Technical Reference:	OP-100, Section 4.0, P&L #5, Page 5, Rev. 47 UFSAR, Section 5.4.1.3.10, Page 61, Amendment 63
References to be provided:	None
Learning Objective:	RCS-ILC Objectives 2, 5.b, & 9
Question Origin:	Bank (Robinson)
Comments:	K/A is matched since the applicant must demonstrate knowledge of an RCP seal package design feature (i.e. operate RCP #1 Seal with a minimum backpressure) which if operated properly will limit RCP seal leakage.
Tier/Group:	T1/G2

29. 2020 NRC RO 029

Given the following plant conditions:
- The unit is operating at 75% power
- Control Rods are in MANUAL

Subsequently:
- A DEH System malfunction causes a load rejection of approximately 50 MWe

Which ONE of the following completes the statement below regarding the INITIAL effect on Pressurizer pressure and Charging flow?

Pressurizer pressure will ___(1)___.

Charging flow will ___(2)___.

A. (1) rise

(2) rise

B. (1) rise

(2) lower

C. (1) lower

(2) rise

D. (1) lower

(2) lower

Reason answer is correct: Pressurizer pressure rises because RCS temperature rises during a load rejection. RCS water is forced into the Pressurizer (INSURGE) causing the bubble to be compressed. The water forced into the Pressurizer causes level to rise above program level. Charging flow is automatically reduced to bring level back to program level.

A. Incorrect. The first part is correct. The second part is plausible since program level is affected by RCS temperature. That is, when RCS temperature rises, program level rises requiring additional Charging flow. During the load rejection, RCS temperature will rise resulting in a rise in program level. This would support the applicant selecting Charging flow rising. However, this is incorrect since the magnitude of the program level change will be small when compared to the insurge itself and Charging flow will initially lower to bring Pressurizer level back to program level.

B. Correct.

C. Incorrect. The first part is plausible since the colder RCS water will be forced into the Pressurizer which will tend to depressurize the saturated Pressurizer; however, this is incorrect since the immediate effect on RCS pressure will be due to compression of the bubble. The second part is plausible since program level is affected by RCS temperature. That is, when RCS temperature rises, program level rises requiring additional Charging flow. During the load rejection, RCS temperature will rise resulting in a rise in program level. This would support the applicant selecting Charging flow rising. However, this is incorrect since the magnitude of the program level change will be small when compared to the insurge itself and Charging flow will initially lower to bring Pressurizer level back to program level.

D. Incorrect. The first part is plausible since the colder RCS water will be forced into the Pressurizer which will tend to depressurize the saturated Pressurizer; however, this is incorrect since the immediate effect on RCS pressure will be due to compression of the bubble. The second part is correct.

004 Chemical and Volume Control / 2

004K5.44; Knowledge of the operational implications of the following concepts as they apply to the CVCS: Pressure response in PZR during in-and-out surge

(CFR: 41.5/ 45.7)

Importance Rating:	3.2 3.4
Technical Reference:	AOP-15-BD, Section 3.0, Page 8, Rev. 18
References to be provided:	None
Learning Objective:	AOP-LP-3.15 Objective 5 CVCS-ILC, Objective 5.b
Question Origin:	Bank (Ginna)
Comments:	K/A is matched since the applicant must demonstrate an understanding how PRZ pressure and charging flow respond during a load rejection (PRZ INSURGE).
Tier/Group:	T2/G1

30. 2020 NRC RO 030

Which ONE of the following completes the statements below?

1RH-1, RCS Loop A to RHR Pump A-SA, is powered from 480V MCC ___(1)___.

In Mode 1, the supply breaker to 1RH-1 is ___(2)___.

A. (1) 1B21-SB

(2) ON

B. (1) 1B21-SB

(2) OFF

C. (1) 1B35-SB

(2) ON

D. (1) 1B35-SB

(2) OFF

Plausibility and Answer Analysis

Reason answer is correct: 1RH-1 is powered from 480V MCC-1B21-SB. In Mode 1, the breaker for 1RH-1 will be LOCKED OFF.

A. Incorrect. *The first part is correct. The second part is plausible since the valve positions are available on the MCB in Mode 1 powered from an independent power source.*

B. Correct.

C. Incorrect. *The first part is plausible since this is the power supply to another RHR valve (1RH-63, Header B to CSIP Suction). The second part is plausible since the valve positions are available on the MCB in Mode 1 powered from an independent power source.*

D. Incorrect. *The first part is plausible since this is the power supply to another RHR valve (1RH-63, Header B to CSIP Suction). The second part is correct.*

005 Residual Heat Removal / 4

005K2.03; Knowledge of bus power supplies to the following: RCS pressure boundary motor-operated valves

(CFR: 41.7)

Importance Rating:	2.7 2.8
Technical Reference:	OP-111, Step 5.1.2.17 & Attachment 1, Pages 17 & 150, Rev. 63
References to be provided:	None
Learning Objective:	RHR-ILC Objective 2.c
Question Origin:	Bank
Comments:	K/A is matched since the applicant must recall the power supply and breaker status for one of the RHR system isolation valves (1RH-1).
Tier/Group:	T2/G1

31. 2020 NRC RO 031

With the unit operating at 100% power, which ONE of the following identifies valves that will automatically re-position upon receipt of a Safety Injection signal?

A. RCP normal miniflow isolation valves (1CS-182/196/210/214)

B. TDAFW pump flow control valves (1AF-129/130/131)

C. SI accumulator discharge valves (1SI-246/247/248)

D. RWST to RHR pump suction valves (1SI-322/323)

Plausibility and Answer Analysis

Reason answer is correct: The RCP normal miniflow isolation valves automatically shut on a Safety Injection signal.

A. Correct.

B. Incorrect. Plausible since MDAFW Pump FCVs receive an OPEN signal on Safety Injection; however, this is incorrect since the TDAFW FCVs do not.

C. Incorrect. Plausible since these valves receive an OPEN signal on Safety Injection; however, this is incorrect since these valves are already open with power removed at power.

D. Incorrect. Plausible since these valves receive an OPEN signal on Safety Injection; however, this is incorrect since these valves are already open at power.

006 Emergency Core Cooling 2/3

006A3.03; Ability to monitor automatic operation of the ECCS, including:
ESFAS-operated valves

(CFR: 41.7 / 45.5)

Importance Rating:	4.1 4.1
Technical Reference:	OP-107, Attachment 2, Page 133, Rev. 117 OMM-004, Attachment 3, Page 38, Rev. 42
References to be provided:	None
Learning Objective:	ESFAS-ILC Objective 6.b
Question Origin:	New
Comments:	Ask Chief Examiner if question should evaluate RCS Inventory (SF2) or RX Pressure Control (SF3). Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by testing valves that respond to an ESFAS signal. The Safety Function suggestions are primarily provided to categorize JPMs. K/A is matched since the applicant must demonstrate an understanding of valves that automatically re-position upon receipt of a Safety Injection (ESFAS) signal.
Tier/Group:	T2/G1

32. 2020 NRC RO 032

Given the following plant conditions:

- ALB-009-8-1, Pressurizer Relief Tank High-Low Level Press Or Temp, alarms due to a high temperature condition

Which ONE of the following describes how the PRT is cooled in accordance with APP-ALB-009-8-1 and OP-100, Reactor Coolant System?

(Assume a rapid cooldown is NOT required)

A. Drain the PRT to the Reactor Coolant Drain Tank while making up to the PRT from the Demineralized Water Storage Tank.

B. Drain the PRT to the Reactor Coolant Drain Tank while making up to the PRT from the Reactor Makeup Water Storage Tank.

C. Recirculate the PRT through the Reactor Coolant Drain Tank heat exchanger using Service Water to cool the heat exchanger.

D. Recirculate the PRT through the Reactor Coolant Drain Tank heat exchanger using Component Cooling Water to cool the heat exchanger.

Plausibility and Answer Analysis

Reason answer is correct: Normal cooling of the PRT is accomplished by recirculating the PRT water through the RCDT heat exchanger which is cooling by CCW.

A. Incorrect. Plausible since this method would be used for a rapid cooldown of the PRT; however this is incorrect since a rapid cooldown is not required. Also incorrect since the makeup source for a rapid cooldown is RMUW, not the DWST.

B. Incorrect. Plausible since this method would be used for a rapid cooldown of the PRT; however this is incorrect since a rapid cooldown is not required.

C. Incorrect. Plausible since the normal cooling of the PRT is accomplished by recirculating the PRT water through the RCDT heat exchanger; however, this is incorrect since the RCDT heat exchanger is cooled by CCW, not SW.

D. Correct.

007 Pressurizer Relief/Quench Tank / 5

007K4.01; Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: Quench tank cooling

(CFR: 41.7)

Importance Rating:	2.6 2.9
Technical Reference:	APP-ALB-009, Page 29, Page 28, Rev. 18 OP-100, Section 8.5, Page 42, Rev. 47 OP-120.08, Section 8.1, Page 37, Rev. 27
References to be provided:	None
Learning Objective:	PRZ-ILC Objective 5.d
Question Origin	Bank (2004 NRC RO 23)
Comments:	K/A is matched since the applicant must demonstrate an understanding of a system design feature in that the PRT is cooled using another system's heat exchanger.
Tier/Group:	T2/G1

33. 2020 NRC RO 033

Which ONE of the following completes the statements below?

In accordance with OP-145, Component Cooling Water, the NORMAL source of makeup to the Component Cooling Water (CCW) System is __(1)__ Water.

Makeup from this source will be initiated __(2)__.

A. (1) Demineralized

(2) from the MCB

B. (1) Demineralized

(2) via local field actions

C. (1) Reactor Makeup

(2) from the MCB

D. (1) Reactor Makeup

(2) via local field actions

Plausibility and Answer Analysis

Reason answer is correct: 1DW-15, CCW Make Up, is remotely opened from the MCB to supply Demineralized Water (NORMAL source of makeup) to the CCW System. If CCW Surge Tank level continues to lower, local actions can be taken to supply the CCW System from the Reactor Makeup Water Storage Tank. This is considered the EMERGENCY source due to the tank containing potentially tritiated water which could result in CCW System contamination.

A. Correct.

B. Incorrect. Plausible since Demineralized Water is the normal makeup source to CCW and one of the makeup sources is supplied via local field actions; however, this is incorrect since Demineralized Water is supplied via switch operation on the MCB.

C. Incorrect. Plausible since one makeup source to CCW is supplied via local field actions and Reactor Makeup Water is a makeup source; however, this is incorrect since Reactor Makeup Water is supplied via local field actions.

D. Incorrect. Plausible since this choice would be correct if the question asked for the emergency source of makeup to CCW, not the normal source.

008 Component Cooling Water / 8

008K1.05; Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: Sources of makeup water

(CFR: 41.2 to 41.9 / 45.7 to 45.9)

Importance Rating:	3.0 3.1
Technical Reference:	AOP-014, Section 3.2, Pages 15 & 20, Rev. 39
References to be provided:	None
Learning Objective:	CCW-ILC Objective 8
Question Origin	Bank
Comments:	K/A is matched since the applicant must demonstrate an understanding of the makeup sources of water to the CCW system.
Tier/Group:	T2/G1

34. 2020 NRC RO 034

Given the following plant conditions:

- The unit is operating at 75% power
- ALB-005-1-2A, RCP Therm Bar Hdr High Flow, is in alarm

Which ONE of the following completes the statements below?

1CC-252, CCW Return Isolation from RCP Thermal Barriers Flow Control, will shut if CCW flow rises to a MINIMUM of __(1)__ gpm.

With 1CC-252 shut, RCP operational limits __(2)__ be exceeded.

A. (1) 198

(2) will

B. (1) 198

(2) will NOT

C. (1) 245

(2) will

D. (1) 245

(2) will NOT

Reason answer is correct: APP-ALB-005-1-2A has the operator verify that 1CC-252 shuts if flow increases to 198 gpm total flow (3 second time delay). CCW cooling to the thermal barrier heat exchangers would be lost upon closure of the isolation valve. However, a loss of CCW flow to the heat exchangers while maintaining seal injection results in a slight increase in pump lower bearing and seal temperatures, but temperatures are expected to remain below pump operational limitations.

A. Incorrect. The first part is correct. The second part is plausible since seal cooling from the thermal barrier heat exchangers has been lost; however, this is incorrect since normal seal cooling (CSIP) is still available.

B. Correct.

C. Incorrect. The first part is plausible since 245 gpm is the high flow alarm associated with the RCP oil coolers. The second part is plausible since seal cooling from the thermal barrier heat exchangers has been lost; however, this is incorrect since normal seal cooling (CSIP) is still available.

D. Incorrect. The first part is plausible since 245 gpm is the high flow alarm associated with the RCP oil coolers. The second part is correct.

008 Component Cooling Water / 8

008K3.03; Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: RCP

(CFR: 41.7 / 45.6)

Importance Rating:	4.1 4.2
Technical Reference:	APP-ALB-005, Window 1-2A, Page 5, Rev. 25
References to be provided:	None
Learning Objective:	CCWS-ILC Objective 7.b
Question Origin:	Bank (McGuire)
Comments:	K/A is matched since the applicant must demonstrate an understanding of how a loss of CCW to the thermal barrier heat exchangers affects RCP operation.
Tier/Group:	T2/G1

35. 2020 NRC RO 035

Given the following plant conditions:
- The unit is in Mode 3
- OST-1117, Pressurizer PORV Operability Quarterly Interval Modes 3 - 6, is in progress

One minute after opening 1RC-118, PRZ PORV PCV-445A SA:
- ALB-009-8-1, PRESSURIZER RELIEF TANK HIGH-LOW LEVEL PRESS OR TEMP, is received
- The RO determines the annunciator is caused by high temperature in the PRT
- Attempts to close 1RC-118 are unsuccessful

Which ONE of the following completes the statement below?

On lowering PRZ pressure, the Group 'C' heaters will FIRST receive a "full on" signal when pressure reduces to __(1)__ psig AND the PRT rupture discs will FIRST rupture when PRT pressure is ≥ __(2)__ psig.

(Assume no operator actions)

A. (1) 2220

(2) 50

B. (1) 2220

(2) 100

C. (1) 2210

(2) 50

D. (1) 2210

(2) 100

Reason answer is correct: The PRZ Group C heaters will be full on at 2220 psig and the PRT rupture discs will rupture at 100 psig.

A. Incorrect. The first part is correct. The second part is plausible because 50 psig is the approximate maximum PRT pressure expected during a design PZR safety valve discharge; however this is incorrect since the PRT rupture discs will rupture at 100 psig.

B. Correct.

C. Incorrect. The first part is plausible because the PRZ backup heaters will be full on at 2210 psig; however, this is incorrect since the PRZ Group C heaters will be full on at 2220 psig. Also plausible since 2210 psig will also cause a low pressure alarm. The second part is plausible because 50 psig is the approximate maximum PRT pressure expected during a design PZR safety valve discharge; however this is incorrect since the PRT rupture discs will rupture at 100 psig.

D. Incorrect. The first part is plausible because the PRZ backup heaters will be full on at 2210 psig; however, this is incorrect since the PRZ Group C heaters will be full on at 2220 psig. Also plausible since 2210 psig will also cause a low pressure alarm. The second part is correct.

010 Pressurizer Pressure Control / 3

010A3.01; Ability to monitor automatic operation of the PZR PCS, including: PRT temperature and pressure during PORV testing

(CFR: 41.7 / 45.5)

Importance Rating:	3.0 3.2
Technical Reference:	AOP-019, Attachment 1, Page 19, Rev. 25 APP-ALB-009-8-1, Page 29, Rev. 18 DBD-100, Step 2.1.5, Page 5, Rev. 19
References to be provided:	None
Learning Objective:	PRZ-ILC Objective 3.b AOP-LP-3.19 Objective 4.a
Question Origin:	Bank (2009B NRC RO 36)
Comments:	K/A is matched since the applicant must demonstrate an understanding of how PRZ pressure control is affected with a PORV stuck open and the impact this will have on PRT temperature and pressure (saturated system).
Tier/Group:	T2/G1

36. 2020 NRC RO 036

Given the following plant conditions:
- The unit is operating at 100% power

Subsequently:
- Pressurizer Spray Valve, 1RC-103, begins to slowly fail open

Which ONE of the following completes the statements below?

An __(1)__ turbine runback will occur when 2/3 channels are within __(2)__% of the Reactor trip setpoint.

OTΔT = OVERTEMPERATURE ΔT
OPΔT = OVERPOWER ΔT

A. (1) OTΔT

(2) 1.9

B. (1) OTΔT

(2) 3

C. (1) OPΔT

(2) 1.9

D. (1) OPΔT

(2) 3

Plausibility and Answer Analysis

Reason answer is correct:

The OTΔT Rx Trip Setpoint is automatically varied with coolant temperature, pressurizer pressure, and axial power distribution.

OTΔT Runback - 2/3 channels within 3% of OTΔT setpoint

The OPΔT Rx Trip Setpoint is automatically varied with coolant temperature, rate of change of temperature, and axial power distribution.

OPΔT Runback - 2/3 channels within 1.9% of OTΔT setpoint

A. Incorrect. The first part is correct. The second part is plausible since this is the setpoint for the OPΔT turbine runback.

B. Correct.

C. Incorrect. The first part is plausible since one of the variable Reactor trip setpoints is affected by pressurizer pressure, but OTΔT is affected, not OPΔT. The second part is plausible since this is the setpoint for the OPΔT turbine runback.

D. Incorrect. The first part is plausible since one of the variable Reactor trip setpoints is affected by pressurizer pressure, but OTΔT is affected, not OPΔT. The second part is correct.

012 Reactor Protection / 7

012A1.01; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including: Trip setpoint adjustment

(CFR: 41.5 / 45.5)

Importance Rating:	2.9 3.4
Technical Reference:	APP-ALB-020, Window 2-2, Page 12, Rev. 55 Technical Specifications, Pages 2-7 & 2-8 Technical Specifications Bases, Pages B 2-4 & B 2-5
References to be provided:	None
Learning Objective:	RPS-ILC Objective 8
Question Origin:	New
Comments:	Ask Chief Examiner if acceptable to ask how variable Rx trip setpoints are impacted by a plant transient. Operations does not adjust trip setpoints (I&C function). Phonecon 8/27: Chief Examiner stated that it was acceptable to ask this type of question to meet this K/A. K/A is matched since the applicant must predict how two RPS trip setpoints will be affected by a reduction in Pressurizer pressure.
Tier/Group:	T2/G1

37. 2020 NRC RO 037

Given the following plant conditions:
- Reactor power is 7%
- A plant startup is in progress in accordance with GP-005, Power Operation (Mode 2 to Mode 1)

Subsequently:
- Instrument Bus S-I de-energizes

Given the above plant conditions, which ONE of the following will result in a Reactor trip signal being generated?

A. LT-461, PRZ Level Channel III, fails high

B. LT-496, 'C' SG Level Channel III, fails low

C. PT-457, PRZ Pressure Channel III, fails low

D. A and C Aux buses are crosstied and breaker 107, Aux Bus A supply, fails open

Plausibility and Answer Analysis

Reason answer is correct: The SG low-low water level circuit trips the Reactor if two out of three level indicators of any one SG indicate below the low-low trip setpoint of 25% NR SG level. With the failure of Instrument Bus S-I, 2/3 'C' levels are < 25% (LS-494A and LS-496A) which would result in a Reactor trip.

A. Incorrect. Plausible since one channel for Pressurizer High Level (87%) would be actuated due to the failure of Instrument Bus S-I. The second failure would actuate a Reactor trip signal, but since Reactor power and Turbine power is <10%, the High PRZ level Reactor trip is blocked by P-7. Reactor trip logic for PRZ High Level - two of the three water level signals from LS-459A, LS-460A, or LS-461A above the trip setpoint (87%) will initiate a Reactor trip. The trip function is automatically blocked below P-7 (<10%).

B. Correct.

C. Incorrect. Plausible since one channel for Pressurizer Low Pressure (1960 psig) would be actuated due to the failure of Instrument Bus S-I. The second failure would actuate a Reactor trip signal, but since Reactor power and Turbine power is <10%, the Low PRZ Pressure Reactor trip is blocked by P-7. Reactor trip logic for PRZ Low Pressure - two of the pressure signals from PS-455C, PS-456C, or PS-457C below the trip setpoint (1960 psig) will initiate a Reactor trip. The trip function is automatically blocked below P-7 (<10%).

D. Incorrect. Plausible since this would cause 2 RCPs to lose power and a Reactor trip signal for loss of Reactor Coolant Flow (2 of 3 channels on one loop < 91.7%) would be activated, but because the Reactor power and Turbine power is <10% this trip is blocked by P-7.

012 Reactor Protection / 7

012K6.02; Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Redundant channels

(CFR: 41.7 / 45/7)

Importance Rating:	2.9 3.1
Technical Reference:	EOP-E-0, Attachment 10, Page 79, Rev. 15 AOP-024, Attachment 1, Page 32, Rev. 60 DBD-301, Section 4.1.2.4, Page 31, Rev. 9
References to be provided:	None
Learning Objective:	RPS-ILC Objective 8
Question Origin:	Bank (2014 NRC RO 39)
Comments:	K/A is matched since the applicant must demonstrate knowledge of an RPS trip setpoint, including channels used to satisfy the trip logic.
Tier/Group:	T2/G1

38. 2020 NRC RO 038

Given the following plant conditions:

- The unit is operating at 100% power

Which ONE of the following predicts the Main Feedwater Pump response to an inadvertent actuation of Train 'B' Safety Injection?

A. Both Main Feedwater pumps will immediately trip

B. 'B' Main Feedwater pump will immediately trip; 'A' Main Feedwater pump will remain running

C. 'B' Main Feedwater pump will immediately trip; 'A' Main Feedwater pump will be stripped by the sequencer

D. No Main Feedwater pump trip is initially generated; both Main Feedwater pumps will be stripped by the sequencers

Plausibility and Answer Analysis

Reason answer is correct: A Safety Injection signal will cause a Main Feedwater Isolation Signal (MFIS) to be generated. The main feedwater isolation signal (MFIS) is actuated from any SI signal or a two of four high-high SG levels (P-14 - 78%). The MFIS closes the main feedwater isolation valves (1FW-159, 1FW- 277, FW-217), the main feed regulating valves (FRVs, 1FW-133, 1FW-249, 1FW-191), the FRV bypass valves (1FW-140, 1FW-256, 1FW-198), and trips the Turbine and Main Feedwater pumps.

A. Correct.

B. Incorrect. Plausible that since only one train of SI has actuated that only one Main FW pump would be affected because multiple components in the ESFAS system are actuated by specific trains.

C. Incorrect. The first part is plausible since only one train of SI has actuated and that only one Main FW pump would be affected. The second part is plausible since a Reactor Trip with Tavg < 564°F will generate a signal to prevent excessive RCS cooldown. The signal affects the Feedwater system by shutting the main Feedwater Regulator valves, but does not send a trip signal to the MFW pumps.

D. Incorrect. The first part is plausible since the Main FW pumps continue to run during a Reactor Trip event, but will both trip due to a Main FWIS caused by the SI signal. The second part is plausible since a Reactor Trip with Tavg < 564°F will generate a signal to prevent excessive RCS cooldown. The signal affects the Feedwater system by shutting the main Feedwater Regulator valves, but does not send a trip signal to the MFW pumps.

013 Engineered Safety Features Actuation / 2

013K1.15; Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: MFW System

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating:	3.4 3.8
Technical Reference:	EOP-E-0, Attachment 3, Page 61, Rev. 15 EMDRAC 1364 000876, Functional Diagram Feedwater Control and Isolation, Rev. 8
References to be provided:	None
Learning Objective:	CFW-ILC Objective 9.a
Question Origin:	Bank (2013 NRC RO 10)
Comments:	K/A is matched since the applicant must demonstrate an understanding of how the MFW system is affected by a Safety Injection signal.
Tier/Group:	T2/G1

39. 2020 NRC RO 039

Given the following plant conditions:

- The unit is operating at 100% power
- Instrument Bus SIII is de-energized and actions are being taken in accordance with AOP-024, Loss of Uninterruptible Power Supply

Subsequently:

- PT-953, Containment Pressure Channel IV, fails high

Which ONE of the following identifies the effect on the Safety Injection (SI) and Containment Spray Actuation Signal (CSAS) systems?

	SI	CSAS
A.	NOT Actuated	NOT Actuated
B.	Actuated	NOT Actuated
C.	NOT Actuated	Actuated
D.	Actuated	Actuated

Plausibility and Answer Analysis

Reason answer is correct: *An SI actuation (de-energize to actuate) will occur, but a CSAS (energize to actuate) will not occur unless another energized channel senses a high pressure condition.*

A. Incorrect. Plausible since the applicant would select this choice if they believed both SI and CSAS were energize to actuate; however, this is incorrect since SI is de-energize to actuate.

B. Correct.

C. Incorrect. Plausible since one of the two signals is energize to actuate and the other is de-energize to actuate; however, this is incorrect since SI is de-energize to actuate and CSAS is energize to actuate. The applicant would select this choice if they could not recall which actuation was energize to actuate and which actuation was de-energize to actuate.

D. Incorrect. Plausible since the applicant would select this choice if they believed both SI and CSAS were de-energize to actuate; however, this is incorrect since CSAS is energize to actuate.

013 Engineered Safety Features Actuation / 2

013K6.01; Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	2.7 3.1
Technical Reference:	Plant Drawing 1364-000871 AOP-024, Attachment 3, Page 40, Rev. 60
References to be provided:	None
Learning Objective:	ESFAS-ILC Objectives 8.a & 8.b
Question Origin:	Previous (2016 NRC RO 40)
Comments:	K/A is matched since applicant must demonstrate an understanding how a failure of an ESFAS pressure transmitter affects multiple ESFAS systems.
Tier/Group:	T2G1

40. 2020 NRC RO 040

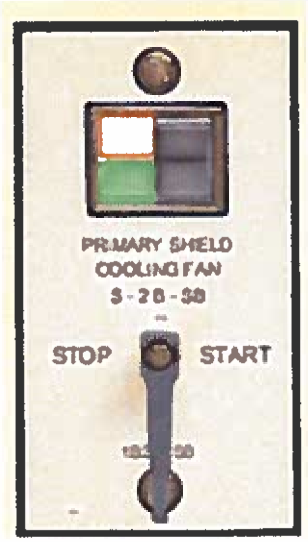
Given the following plant conditions:

- The unit is operating at 100% power
- S-2B-SB, Primary Shield Cooling Fan, is in operation

Subsequently:

- ALB-027-5-5, Reactor Primary Shield Clg Fans S2 Low-Flow-O/L, alarms

The S-2B-SB control switch indications are as follows:



Which ONE of the following completes the statement below?

In accordance with APP-ALB-027, S-2B-SB indicates the alarm was received due to actuation of the ___(1)___ AND S-2A-SA, Primary Shield Cooling Fan, ___(2)___.

A. (1) thermal overload device

(2) will start automatically

B. (1) thermal overload device

(2) must be manually started

C. (1) low flow switch

(2) will start automatically

D. (1) low flow switch

(2) must be manually started

Plausibility and Answer Analysis

Reason answer is correct: The flow sensor (FS-01RP-7970S) provides input into a flow switch to actuate the alarm ALB-027-5-5. While both the low flow switch and thermal overload device actuate ALB-027-5-5, only the thermal overload condition will energize the white light on the fan control switch. Because the Primary Shield Cooling fans do not automatically start, the APP response is for the operator to manually start the standby Primary Shield Cooling fan.

A. Incorrect. The first part is correct. The second part is plausible since the containment cooling system fans E80 and E81 for CRDM cooling automatically start the standby fan if a low flow condition occurs; however, this is incorrect because the S-2 and S-4 fans do not have an automatic start feature.

B. Correct.

C. Incorrect. The first part is plausible since the flow sensor will actuate ALB-027-5-5; however, this is incorrect since the white light indication on the control switch indicates the presence of a thermal overload condition. The second part is plausible since the containment cooling system fans E80 and E81 for CRDM cooling automatically start the standby fan if a low flow condition occurs; however, this is incorrect because the S-2 and S-4 fans do not have an automatic start feature.

D. Incorrect. The first part is plausible since the flow sensor will actuate ALB-027-5-5; however, this is incorrect since the white light indication on the control switch indicates the presence of a thermal overload condition. The second part is correct.

022 Containment Cooling / 5

022K1.02; Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SEC/remote monitoring systems

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating:	3.7 3.5
Technical Reference:	APP-ALB-027, Window 5-5, Page 18, Rev. 12
References to be provided:	None
Learning Objective:	CCS-ILC Objective 6
Question Origin:	Modified (2016 NRC RO 58)
Comments:	Ask Chief Examiner what SEC stands for. Phonecon 4/14: Chief Examiner stated to ignore SEC. K/A is matched since the applicant must demonstrate an understanding of remote indications (MCB) associated with a Containment Cooling System fan.
Tier/Group:	T2/G1

41. 2020 NRC RO 041

Given the following plant conditions:
- A LOCA occurred
- The crew has transitioned EOP-ES-1.3, Transfer to Cold Leg Recirculation
- Both trains of Safety Injection and Containment Spray are aligned for recirculation

Which ONE of the following completes the statements below?

A MINIMUM of ___(1)___ inches Containment (CNMT) wide range sump level assures a long term recirculation suction source.

The PREFERRED method for raising CNMT sump inventory is to re-align one of the running ___(2)___ suction back to the RWST.

A. (1) 142

(2) CSIPs

B. (1) 142

(2) CNMT Spray Pumps

C. (1) 196

(2) CSIPs

D. (1) 196

(2) CNMT Spray Pumps

Plausibility and Answer Analysis

Reason answer is correct: EOP-ES-1.3 states a minimum of 142 INCHES CNMT wide range level ensures recirculation sump strainers are completely submerged and assures a long term recirculation suction source. Attachment 2 contains actions to realign a CNMT spray pump (or CSIP, if absolutely necessary) for injection from the RWST.

A. Incorrect. The first part is correct. The second part plausible a CSIP can be used to raise CNMT sump inventory; however, this is incorrect since the preferred method is to use a CNMT Spray pump.

B. Correct.

C. Incorrect. The first part is plausible since 196 inches is the MAXIMUM containment sump water level addressed in EOP-ES-1.3. The second part is correct.

D. Incorrect. The first part is plausible since 196 inches is the MAXIMUM containment sump water level addressed in EOP-ES-1.3. The second part plausible a CSIP can be used to raise CNMT sump inventory; however, this is incorrect since the preferred method is to use a CNMT Spray pump.

026 Containment Spray / 5

026A1.03; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment sump level

(CFR: 41.5 / 45.5)

Importance Rating:	3.5 3.5
Technical Reference:	EOP-ES-1.3, NOTE preceding Step 1 & CAUTION preceding Attachment 2 Step 1, Pages 4 & 40, Rev. 4
References to be provided:	None
Learning Objective:	EOP-LP-3.03 Objectives 3.e, 5.c, & 5.j
Question Origin:	New
Comments:	K/A is matched since the applicant must be able to predict when the long term recirculation source is lost and demonstrate an understanding of how a Containment Spray pump can be used to recover this long term source.
Tier/Group:	T2/G1

42. 2020 NRC RO 042

Given the following plant conditions:

- The unit is operating at 100% power
- 'A' Containment Spray Pump is running on recirculation per OST-1118, Containment Spray Operability Train A Quarterly Interval Modes 1-4

Subsequently:

- A LOCA occurs
- Containment pressure rises to 7.5 psig

Which ONE of the following identifies the positions of 1CT-24, Containment Spray Eductor Test, and 1CT-50, Containment Spray Pump 1A-SA Discharge Valve?

	1CT-24	1CT-50
A.	OPEN	OPEN
B.	OPEN	SHUT
C.	SHUT	OPEN
D.	SHUT	SHUT

Plausibility and Answer Analysis

Reason answer is correct: In accordance with OST-1118, if a Phase A Containment Isolation Signal (CNMT pressure $\geq$ 3 psig) is received during the performance of this OST, the following components will realign as stated:

* Containment Spray Pump 1A-SA will trip
* 1CT-47, CNMT SPRAY PUMP A-SA RECIRC, will shut
* 1CT-24, CONTAINMENT SPRAY EDUCTOR TEST, will shut

A. Incorrect. The first part is plausible since 1CT-24 will be open for testing, but will shut on a Phase A signal. The second part is plausible since 1CT-50 will open on a CSAS signal (CNMT $\geq$ 10 psig).

B. Incorrect. The first part is plausible since both 1CT-24 and 1CT-25 will be open for testing, but will shut on a Phase A signal. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since 1CT-50 will open on a CSAS signal (CNMT $\geq$ 10 psig). This choice would be correct if CNMT pressure was $\geq$ 10 psig.

D. Correct.

026 Containment Spray / 5

026K3.02; Knowledge of the effect that a loss or malfunction of the CSS will have on the following: Recirculation spray system

(CFR: 41.7 / 45.6)

Importance Rating:	4.2 4.3
Technical Reference:	OST-1118, Section 4.0, P&L #1, Page 9, Rev. 55
References to be provided:	None
Learning Objective:	CSS-ILC Objective 6.b
Question Origin	Bank (2007 NRC Exam)
Comments:	Ask Chief Examiner if acceptable to ask question with Containment Spray in recirculation test lineup. Phonecon 8/27: Chief Examiner stated that it was acceptable to ask this type of question to meet this K/A. K/A is matched since the applicant must demonstrate an understanding of how a Containment Phase A Isolation signal will impact the Containment Spray system when recirculating back to the RWST during its quarterly surveillance testing.
Tier/Group:	T2/G1

43. 2020 NRC RO 043

Given the following plant conditions:

- The unit is operating at 100% power
- A Main Steam line rupture in the Turbine Building has occurred
- The crew has manually tripped the Reactor

Which ONE of the following completes the statement below?

The Turbine Ventilating valves (1GS-97, 1GS-98) are expected to __(1)__ AND the MSR Non-Return valves (1HD-2, 1HD-3, 1HD-302, 1HD-303) are expected to __(2)__.

Valve Noun Name:

Turbine Ventilating valves
1GS-97, HP Turbine Vent to Cond (FCV-01TA-0415B)
1GS-98, HP Turbine Vent to Cond (FCV-01TA-0415A)

MSR Non-Return valves
1HD-2, MSR 1A-NNS Outlet to MSDT 1A-NNS
1HD-3, MSRDT 1A-NNS Outlet to 5-1A-NNS
1HD-302, MSR 1B-NNS Outlet to MSDT 1B-NNS
1HD-303, MSRDT 1B-NNS Outlet to 5-1B-NNS

A. (1) SHUT

(2) SHUT

B. (1) SHUT

(2) OPEN

C. (1) OPEN

(2) SHUT

D. (1) OPEN

(2) OPEN

Reason answer is correct: Any Reactor Trip generates a Turbine Trip signal. Since a Turbine Trip signal is present all of the Turbine Throttle valves would be shut and the Auto Stop Trip header would be depressurized causing the Turbine Ventilating valves to OPEN and MSR Non-Return valves to SHUT. 1GS-97 and 1GS-98 automatically open while 1HD-2, 1HD-3, 1HD-302 and 1HD-303 shut automatically based on the status of the Turbine Throttle valves or the Auto Stop Trip header pressure which are used to determine if the Turbine is tripped or latched.

A. Incorrect. The first part is plausible since with the Turbine tripped, 1st stage pressure is reduced to the pressure of the Main Condenser which is less than the 5 psig. The Gland Sealing Steam Spillover Regulator to the condenser modulates open if header pressure is > 5 psig and therefore the valve would be shut on a turbine trip; however, the ventilating valves open to provide a flowpath to the condenser for the steam trapped in the HP turbine. The second part is correct.

B. Incorrect. The first part is plausible since with the Turbine tripped, 1st stage pressure is reduced to the pressure of the Main Condenser which is less than the 5 psig. The Gland Sealing Steam Spillover Regulator to the condenser modulates open if header pressure is > 5 psig and therefore the valve would be shut on a turbine trip; however, the ventilating valves open to provide a flowpath to the condenser for the steam trapped in the HP turbine. The second part is plausible since the turbine drain valves automatically open following a turbine trip to provide a drain path for the residual steam trapped in the turbine as this steam begins to condense.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since the turbine drain valves automatically open following a turbine trip to provide a drain path for the residual steam trapped in the turbine as this steam begins to condense.

039 Main and Reheat Steam / 4

039A3.02; Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS

(CFR: 41.5 / 45.5)

Importance Rating:	3.1 3.5
Technical Reference:	AOP-006, Section 3.1, Pages 5 and 6, Rev. 66
References to be provided:	None
Learning Objective:	MT-ILC Objective 9 MSR-ILC Objective 4
Question Origin:	Previous (2018 NRC RO 43)
Comments:	K/A is matched since the applicant must demonstrate an understanding of automatic operation of various Turbine and MSR valves following a turbine trip.
Tier/Group:	T2/G1

44. 2020 NRC RO 044

Which ONE of the following completes the statement below regarding operation of the SG PORVs?

Control power selector switches located in the ___(1)___ can be used to supply alternate control power from the instrument buses to ___(2)___ SG PORVs.

A. (1) Steam Tunnel

(2) ALL

B. (1) Steam Tunnel

(2) ONLY 'A' and 'B'

C. (1) RAB 286 Electrical Penetration Areas

(2) ALL

D. (1) RAB 286 Electrical Penetration Areas

(2) ONLY 'A' and 'B'

Plausibility and Answer Analysis

Reason answer is correct: The Control Power Selector Switches are installed on 286' of the RAB (electrical penetration areas) and are used to supply alternate control power to the 'A' and 'B' SG PORVs from the instrument buses during a station blackout.

A. Incorrect. The first part is plausible since this is the location of the SG PORVs; however, this is incorrect since the control power selector switches are located on 286' of the RAB. The second part is plausible since all three SG PORVs can be powered from the instrument buses; however, this is incorrect since instrument bus SIII is the normal control power supply for the 'C' SG PORV.

B. Incorrect. The first part is plausible since this is the location of the SG PORVs; however, this is incorrect since the control power selector switches are located on 286' of the RAB. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since all three SG PORVs can be powered from the instrument buses; however, this is incorrect since instrument bus SIII is the normal control power supply for the 'C' SG PORV.

D. Correct.

039 Main and Reheat Steam / 4

039G2.1.30; Ability to locate and operate components, including local controls.

(CFR: 41.7 / 45.7)

Importance Rating: 4.4 4.0

Technical Reference: EOP-ECA-0.0, Step 16.c, Page 26, Rev. 10

References to be provided: None

Learning Objective: MSS-ILC Objectives 2.b & 5.a

Question Origin: New

Comments: K/A is matched since the applicant must recall where local controls used to align the SG PORVs to their alternate power sources are located. Aligning the 'A' and 'B' SG PORVs to their alternate control power will allow them to be controlled from the MCB (vice locally).

Tier/Group: T2/G1

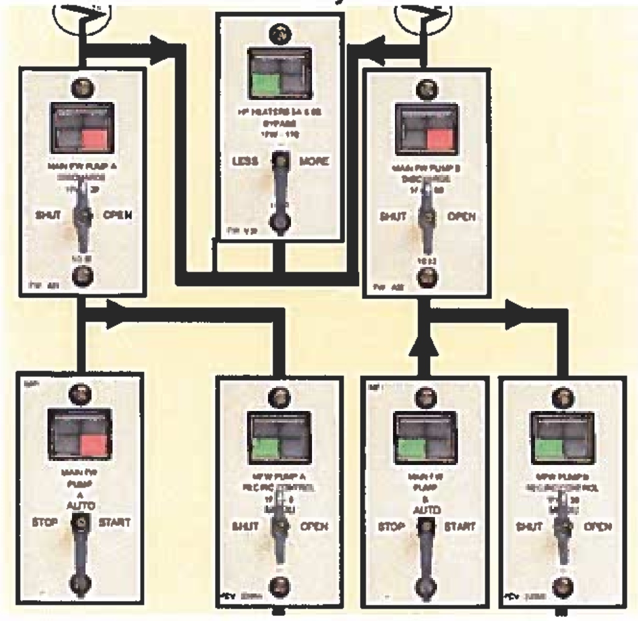
45. 2020 NRC RO 045

Given the following plant conditions:

- The unit is operating at 85% power

Subsequently:

- The following indications are observed by the Balance of Plant operator (BOP):



Which ONE of the following completes the statements below?

A loss of ___(1)___ has occurred.

In accordance with AOP-010, Feedwater Malfunctions, the operator is required to ___(2)___.

A. (1) 125 VDC DP 1A-1

(2) trip the Reactor

B. (1) 125 VDC DP 1A-1

(2) isolate Steam Generator Blowdown

C. (1) 6.9 KV Aux Bus 1B

(2) trip the Reactor

D. (1) 6.9 KV Aux Bus 1B

(2) isolate Steam Generator Blowdown

Reason answer is correct: The indications provided show the 'B' Main Feedwater Pump as tripped as a result of a loss of power to 6.9 kV Aux Bus 1B. The immediate actions of AOP-010 requires Steam Generator Blowdown to be isolated if Reactor power is $\geq$ 80% but less than 90%.

A. Incorrect. The first part is plausible since 125 VDC DP 1A-1 provides control power for the 6.9 kV breakers (non-safety related); however, this is incorrect since the Main Feedwater Pump indications would be extinguished with a loss of this DC power supply. The second part is plausible since this choice would be correct if Reactor power was $\geq$ 90%.

B. Incorrect. The first part is plausible since 125 VDC DP 1A-1 provides control power for the 6.9 kV breakers (non-safety related); however, this is incorrect since the Main Feedwater Pump indications would be extinguished with a loss of this DC power supply. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since this choice would be correct if Reactor power was $\geq$ 90%.

D. Correct.

059 Main Feedwater / 4

059A4.01; Ability to manually operate and monitor in the control room: MFW turbine trip indication

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	3.1 3.1
Technical Reference:	OP-134.01, Attachment 1, Page 53, Rev. 45 AOP-010, Section 3.0, Steps 2 & 3, Page 4, Rev. 40
References to be provided:	None
Learning Objective:	CFW-ILC Objective 2.d AOP-LP-3.10 Objective 4
Question Origin:	New
Comments:	Ask Chief Examiner if acceptable to test electric driven MFW pumps. Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by testing our electric driven MFW pumps. K/A is matched since the applicant must monitor control room indications and determine that a Main Feedwater Pump has tripped due to a loss of AC power. Then based on plant conditions, take the immediate action of the MFW AOP to isolate steam generator blowdown.
Tier/Group:	T2/G1

46. 2020 NRC RO 046

Which ONE of the following identifies the power supply for 1MS-72, Main Steam C to Aux FW Turbine?

A. PP-1B312-SB

B. DP-1B2-SB

C. 1B31-SB

D. IDP-SII

Plausibility and Answer Analysis

Reason answer is correct: 1MS-72 is powered from 125 VDC DP-1B2-SB.

A. Incorrect. Plausible since this is the alternate power supply for the TDAFW Pump FCVs.

B. Correct.

C. Incorrect. Plausible since this is the power supply for the MDAFW Pump MOVs.

D. Incorrect. Plausible since this is the power supply for the TDAFW Pump FCVs.

061 Auxiliary/Emergency Feedwater / 4

061K2.03; Knowledge of bus power supplies to the following: AFW diesel driven pump

(CFR: 41.7)

Importance Rating:	4.0 3.8
Technical Reference:	OP-137, Attachment 3, Page 74, Rev. 46
References to be provided:	None
Learning Objective:	AFW-ILC Objective 2.e
Question Origin:	New
Comments:	Ask Chief Examiner is acceptable to test electric driven or steam driven AFW pumps. Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by testing our electric driven or steam driven AFW pumps. K/A is matched since the applicant must recall the power supply to one of the steam admission valves for the TDAFW Pump.
Tier/Group:	T2/G1

47. 2020 NRC RO 047

Given the following plant conditions:

- The unit is operating at 100% power
- Annunciator ALB-014-7-4, SG A, B, C BACKLEAKAGE HIGH TEMP, has alarmed
- An AO has been dispatched to verify local temperatures

Which ONE of the following completes the statements below?

The reason this condition occurred is because a/an __(1)__ piping check valve is leaking.

In accordance with the AOP-010, Feedwater Malfunctions, under these conditions with the TDAFW piping local temperature > 212°F, the FIRST action required is to __(2)__.

A. (1) TDAFW pump steam supply

(2) start the TDAFW pump to flush the line through the exhaust

B. (1) TDAFW pump steam supply

(2) isolate the TDAFW pump discharge header

C. (1) Auxiliary Feedwater

(2) start the TDAFW pump to flush the line to the SGs

D. (1) Auxiliary Feedwater

(2) isolate the TDAFW pump discharge header

Plausibility and Answer Analysis

Reason answer is correct: Per the AOP-010 basis document, backleakage of steam through the AFW lines may occur if check valves leak. If steam should enter a horizontal portion of the line, restoring flow of cold water could create a bubble-collapse water hammer of damaging magnitude. The resulting damage could compromise the AFW system.

A. Incorrect. The first part is plausible since the TDAFW pump steam supply lines are equipped with check valves that have the potential to leak by and raise the temperature of the turbine side of the pump; however, the back leakage alarm is an indication of rising temperature on the pump discharge line. The second part is plausible since starting the TDAFW pump will flow steam to the supply piping and flush the exhaust header and remove any potential debris that is preventing the steam supply line check valve from seating; however, this is incorrect since the alarm addresses AFW line backleakage, not steam supply line backleakge.

B. Incorrect. The first part is plausible since the TDAFW pump steam supply lines are equipped with check valves that have the potential to leak by and raise the temperature of the turbine side of the pump; however, the back leakage alarm is an indication of rising temperature on the pump discharge line. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since this action is performed to cool the AFW piping; however, this is incorrect as this is not the FIRST action required. This action is only performed after the affected header is isolated and vented. Reactor power must also be reduced to less than 98% to prevent exceeding 100% power when cold AFW is introduced into the system.

D. Correct.

061 Auxiliary/Emergency Feedwater / 4

061K5.05; Knowledge of the operational implications of the following concepts as the apply to the AFW: Feed line voiding and water hammer

(CFR: 41.5 / 45.7)

Importance Rating:	2.7 3.2
Technical Reference:	AOP-010, Section 3.3 & Attachment 9, Pages 23 & 33, Rev. 40 AOP-010-BD, Section 3.0, Page 7, Rev. 21
References to be provided:	None
Learning Objective:	AOP-LP-3.10 Objective 5
Question Origin:	Bank (2013 NRC RO 47)
Comments:	K/A is matched since the applicant must demonstrate an understanding of potential voiding in the AFW supply lines due to backleakage and subsequent water hammer when AFW flow initiated.
Tier/Group:	T2/G1

48. 2020 NRC RO 048

Given the following plant conditions:

- The unit is operating at 100% power

Subsequently:

- A loss of offsite power occurs
- EDG 1B-SB fails to start
- The crew enters AOP-025, Loss of One Emergency AC Bus (6.9KV) or One Emergency DC Bus (125V)

One minute later, 125 VDC DP-1A-SA is lost.

Which ONE of the following completes the statements below?

EDG 1A-SA ___(1)___ supplying Emergency Bus 1A-SA.

Prior to re-energizing Emergency Bus 1B-SB from its EDG, the EDG output breaker and 6.9KV breakers will be opened ___(2)___ using Attachment 5, Emergency Bus 6.9KV and 480V Breakers.

A. (1) is

(2) locally

B. (1) is

(2) from the MCB

C. (1) is NOT

(2) locally

D. (1) is NOT

(2) from the MCB

Reason answer is correct: Because offsite power was lost first, EDG 1A-SA will start and sequence on loads using program 'A'. The EDG 1B-SB output breaker and 6.9KV breakers on Emergency Bus 1B-SB will be opened from the MCB using AOP-025 Attachment 5 as DC control power remains available to these breakers.

A. Incorrect. The first part is correct. The second part is plausible since if DC control power is not available, local manual operation of the EDG output breaker and 6.9KV breakers will be necessary; however, this is incorrect as DC power was never lost to the 1B-SB equipment.

B. Correct.

C. Incorrect. The first part is plausible since offsite power was lost and the applicant may determine that the EDG 1A-SA started with the loss of DP-1A-SA requiring local operation of the output breaker (breaker 106) to restore power to in accordance with EOP-ECA-0.0; however, this is incorrect since the EDG output breaker closes within 10 seconds of a loss of power (UV) condition to restore power to the Emergency Bus and the DP-1A-SA power is lost 60 seconds after the loss of offsite power; therefore, breaker 106 will be closed and the 6.9KV bus will be energized. The second part is plausible since if DC control power is not available, local manual operation of the EDG output breaker and 6.9KV breakers will be necessary; however, this is incorrect as DC power was never lost to the 1B-SB equipment.

D. Incorrect. The first part is plausible since offsite power was lost and the applicant may determine that the EDG 1A-SA started with the loss of DP-1A-SA requiring local operation of the output breaker (breaker 106) to restore power to in accordance with EOP-ECA-0.0; however, this is incorrect since the EDG output breaker closes within 10 seconds of a loss of power (UV) condition to restore power to the Emergency Bus and the DP-1A-SA power is lost 60 seconds after the loss of offsite power; therefore, breaker 106 will be closed and the 6.9KV bus will be energized. The second part is correct.

062 AC Electrical Distribution / 6

062A2.04; Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effect on plant of de-energizing a bus

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.1 3.4
Technical Reference:	AOP-025-BD, Section 1.0, Page 3, Rev. 21 AOP-025, Section 3.2, Step 45, Page 45, Rev. 45 OST-1085, Section 6.0, Step 1, Page 17, Rev. 36
References to be provided:	None
Learning Objective:	AOP-LP-3.25 Objectives 4, 5.d & 6
Question Origin:	New
Comments:	K/A is matched since the applicant must predict the impacts losses of both 6.9KV emergency buses and one 125VDC bus and then use procedural guidance to recover an emergency bus in a controlled manner.
Tier/Group:	T2/G1

49. 2020 NRC RO 049

Given the following plant conditions:

- 250 VDC Battery Charger 1A is in service
- Annunciator ALB-015-3-4, 250 VDC BUS TROUBLE, has alarmed
- Local observation confirms that a ground condition exists

Which ONE of the following completes the statement below concerning this ground?

The impact of this condition on the 250 VDC Bus is that ___(1)___ .

In accordance with APP-ALB-015-3-4, the crew should implement OP-156.06, Ground Isolation and Bus Drop, and ___(2)___ .

A. (1) the ground could result in the degradation of the DC system reliability

(2) energize the 1B 250 VDC Battery Charger then remove the 1A 250 VDC Battery Charger from service

B. (1) the ground could result in the degradation of the DC system reliability

(2) open the 1A charger DC output breaker allowing the batteries to power the 250 VDC bus and then place the 1B 250 VDC Battery Charger in service

C. (1) the battery charger(s) will automatically trip on a high ground condition if left in operation

(2) energize the 1B 250 VDC Battery Charger then remove the 1A 250 VDC Battery Charger from service

D. (1) the battery charger(s) will automatically trip on a high ground condition if left in operation

(2) open the 1A charger DC output breaker allowing the batteries to power the 250 VDC bus and then place the 1B 250 VDC Battery Charger in service

Reason answer is correct: In accordance with ALB-15-3-4, a NOTE in the response guidance to the annunciator states that a ground on the 250 VDC Bus could result in the degradation of the DC system reliability. Step 3.a response states that if a ground is suspected (reported by AO that a ground condition exists) then implement OP-156.06. Since one charger is already in service (1A 250 VDC Battery Charger), the procedure section that would be used to place the standby battery charger in service would be Section 8.1, Rotation of 250 VDC Battery Chargers. This section will place the standby charger in service then remove the initially running charger from service. At no time during this charger sway will the DC bus be powered solely on the batteries.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since opening the chargers DC output breaker would stop the ground associated with the DC charger from affected the DC Bus. Allowing the batteries to power the bus is a function of the batteries when the chargers do not have power and are not powering the bus. Therefore it is plausible to have the batteries power the bus for a short amount of time to power up the standby charger.

C. Incorrect. The first part is plausible since there could be a misconception on what trips the AC input or DC output breakers for the battery chargers. There is a high voltage trip associated with the AC Input Breaker, but there isn't a ground trip for the breakers on the bus. A NOTE in OP-156.01 states that to prevent an inadvertent high voltage trip, the output filters should be allowed to charge for a minimum of 30 seconds before closing the AC Input Breaker. The AC Input and Feeder Breaker may trip if the filter capacitors are not fully charged when the AC Input Breaker is closed. The second part is correct.

D. Incorrect. The first part is plausible since there could be a misconception on what trips the AC input or DC output breakers for the battery chargers. There is a high voltage trip associated with the AC Input Breaker, but there isn't a ground trip for the breakers on the bus. A NOTE in OP-156.01 states that to prevent an inadvertent high voltage trip, the output filters should be allowed to charge for a minimum of 30 seconds before closing the AC Input Breaker. The AC Input and Feeder Breaker may trip if the filter capacitors are not fully charged when the AC Input Breaker is closed. The second part is correct. The second part is plausible since opening the chargers DC output breaker would stop the ground associated with the DC charger from affected the DC Bus. Allowing the batteries to power the bus is a function of the batteries when the chargers do not have power and are not powering the bus. Therefore it is plausible to have the batteries power the bus for a short amount of time to power up the standby charger.

063 DC Electrical Distribution / 6

063A2.01; Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	2.5 3.2
Technical Reference:	APP-ALB-015-3-4, Page 19, Rev. 31 OP-156.06, Section 9.2, Page 16, Rev. 13 OP-156.01, Section 8.1, Page 35, Rev. 40
References to be provided:	None
Learning Objective:	DCP-ILC Objectives 4.c and 6.b
Question Origin:	Bank (2014 NRC RO 50)
Comments:	K/A is matched since the applicant must predict the impact of a ground in the DC electrical system and then use the system operating procedure to place the standby battery charger in service to mitigate the event.
Tier/Group:	T2/G1

50. 2020 NRC RO 050

Which ONE of the following completes the statements below?

The 125V DC Class 1E batteries are designed to provide power for __(1)__ hours during a station blackout event.

In addition to load shed, the Dedicated Shutdown Diesel Generator can be used to provide a non-safety-related feed through MCC 1D23 to __(2)__ safety-related battery charger(s) on EACH train to prolong the battery discharge time.

A. (1) two

(2) one

B. (1) two

(2) both

C. (1) four

(2) one

D. (1) four

(2) both

Plausibility and Answer Analysis

Reason answer is correct: The licensing basis of the plant requires the Class 1E batteries to provide DC power for four hours during a station blackout event. The DSDG is able to provide a non-safety-related feed through MCC 1D23 to one safety-related battery charger on each train (interlock prevents aligning to both).

A. Incorrect. The first part is plausible since the licensing basis of the plant requires the Class 1E batteries to provide DC power for two hours following a design basis event (LOCA/LOOP). The second part is correct.

B. Incorrect. The first part is plausible since the licensing basis of the plant requires the Class 1E batteries to provide DC power for two hours following a design basis event (LOCA/LOOP). The second part is plausible since there are two full capacity battery chargers provided for each redundant Class 1E DC bus; however, an interlock (Manual Transfer Switch) prevents aligning both battery chargers on each train to 1D23 at the same time.

C. Correct.

D. Incorrect. The first part is plausible since the licensing basis of the plant requires the Class 1E batteries to provide DC power for two hours following a design basis event (LOCA/LOOP). The second part is plausible since there are two full capacity battery chargers provided for each redundant Class 1E DC bus; however, an interlock (Manual Transfer Switch) prevents aligning both battery chargers on each train to 1D23 at the same time.

063 DC Electrical Distribution / 6

063A4.03; Ability to manually operate and/or monitor in the control room: Battery discharge rate

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	3.0 3.1
Technical Reference:	DBD-202, Section 2.1.6, Page 17, Rev. 41 EOP-ECA-0.0, Step 26, Page 44, Rev. 10
References to be provided:	None
Learning Objective:	DCP-ILC Objectives 1.a & 4
Question Origin:	New
Comments:	Ask Chief Examiner is acceptable to test battery capacity. Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by testing battery capacity or conditions where the candidate would need to know that the batteries were discharging to supply emergency power. K/A is matched since the applicant must demonstrate the manually operate the safety-related battery chargers during a station blackout event to extend battery life.
Tier/Group:	T2/G1

51. 2020 NRC RO 051

Which ONE of the following completes the statement below?

The Low Starting Air Pressure Interlock inhibits EDG __(1)__ and will FIRST occur when starting air pressure lowers to __(2)__ psig.

A. (1) auto starts ONLY

(2) 150

B. (1) auto AND manual starts

(2) 150

C. (1) auto starts ONLY

(2) 200

D. (1) auto AND manual starts

(2) 200

Reason answer is correct: At 150 psig, the Low Starting Air Pressure Interlock will inhibit the automatic EDG start; however, the EDG can still be started manually (MCR or local).

A. Correct.

B. Incorrect. The first part is plausible since the 86DG lockout relay inhibits all EDG starts; however, this is incorrect since the Low Starting Air Pressure Interlock only inhibits auto starts. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since this is when the Low Pressure Starting Air annunciator will be received; however, the Low Starting Air Pressure Interlock won't actuate until 150 psig.

D. Incorrect. The first part is plausible since the 86DG lockout relay inhibits all EDG starts; however, this is incorrect since the Low Starting Air Pressure Interlock only inhibits auto starts. The second part is plausible since this is when the Low Pressure Starting Air annunciator will be received; however, the Low Starting Air Pressure Interlock won't actuate until 150 psig.

064 Emergency Diesel Generator / 6

064K6.07; Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers

(CFR: 41.7 / 45.7)

Importance Rating: 2.7 2.9

Technical Reference: OP-155, Section 8.7, NOTE preceding Step 1, Page 70, Rev. 91

References to be provided: None

Learning Objective: EDG-ILC Objectives 9.b & 11.b

Question Origin: New

Comments: K/A is matched since the applicant must demonstrate an understanding of how low starting air pressure (air receivers) impacts EDG operations.

Tier/Group: T2/G1

52. 2020 NRC RO 052

Given the following plant conditions:

- A liquid release is in progress from the Treated Laundry and Hot Shower (TL&HS) Tank
- REM-*1WL-3540, Treated Laundry and Hot Shower Tank Pump Discharge Monitor, goes into HIGH ALARM during the release

Which ONE of the following will automatically terminate the release?

A. The running TREATED H&HS TANK PUMP PUMP trips

B. 3LHS-296, TREATED L&HS TKS DISCH ISOL VLV, shuts

C. 3LHS-293 (FCV HK-6193), TRTD L&HS TK TO ENVIRON, shuts

D. 3LHS-301, TREATED L&HS TKS DISCHARGE TO COOLING TOWER BLOWDOWN, shuts

Plausibility and Answer Analysis

Reason answer is correct: On a high radiation level as sensed by REM-*1WL-3540, the discharge isolation valve will automatically shut terminating any release in progress.

A. Incorrect. Plausible since the pump tripping would stop the release; however, there is no automatic trip on high radiation levels.

B. Correct.

C. Incorrect. Plausible since this valve shutting would stop the release; however, there is no automatic closure on high radiation levels.

D. Incorrect. Plausible since this valve shutting would stop the release; however, there is no automatic closure on high radiation levels.

073 Process Radiation Monitoring / 7

073A1.01; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels

(CFR: 41.5 / 45.7)

Importance Rating:	3.2 3.5
Technical Reference:	AOP-005, Attachment 9, Step 4, Page 27, Rev. 30
References to be provided:	None
Learning Objective:	AOP-LP-3.05 Objective 1.a
Question Origin:	Bank (2004 NRC RO 68)
Comments:	K/A is matched since the applicant must predict the PRM system response to a high radiation condition.
Tier/Group:	T2/G1

53. 2020 NRC RO 053

Given the following plant conditions:
- The unit is operating at 100% power
- NSW Pump 'B' is operating
- NSW Pump 'A' is in standby

Subsequently:
- ALB-002-7-1, SERV WATER SUPPLY HDR B LOW PRESS, alarms

After one (1) minute, which ONE of the following identifies the expected Service Water system alignment?

A. NSW Pump 'B' is running supplying NSW loads and both ESW headers.

No ESW pumps are running.

B. NSW Pump 'B' is running supplying NSW loads and the 'A' ESW header.

ESW Pump 'B' is running supplying the 'B' ESW header.

C. NSW Pump 'A' is running supplying NSW loads and both ESW headers.

No ESW pumps are running.

D. NSW Pump 'A' is running supplying NSW loads and the 'A' ESW header.

ESW Pump 'B' is running supplying the 'B' ESW header.

Plausibility and Answer Analysis

Reason answer is correct: Annunciator ALB-002-6-2, SERV WATER SUPPLY HDR B LOW PRESS will occur when 'B' ESW header pressure lowers to 53 psig. After a 20 second time delay, ESW Pump 'B' will auto start and supply the 'B' ESW header. NSW Pump 'B' will remain running supplying NSW loads and the 'A' ESW header.

A. Incorrect. Plausible since this is the initial system alignment and the alignment that would exist for the first 20 seconds following receipt of the alarm; however, this is incorrect since one minute has elapsed and the 'B' ESW Pump would have auto started.

B. Correct.

C. Incorrect. Plausible since the candidate may have a misconception that the standby NSW pump auto started on low pressure; however, this is incorrect since the standby NSW pump will only auto start on an overcurrent trip of the running pump's breaker.

D. Incorrect. The first part is plausible since the candidate may have a misconception that the standby NSW pump auto started on low pressure; however, this is incorrect since the standby NSW pump will only auto start on an overcurrent trip of the running pump's breaker. The second part is correct.

076 Service Water / 4

076G2.4.46; Ability to verify that the alarms are consistent with the plant conditions.

(CFR: 41.10 / 43.5 / 45.3 / 45.12)

Importance Rating: 4.2 4.2

Technical Reference: APP-ALB-002-7-1, Page 32, Rev. 53

References to be provided: None

Learning Objective: SWS-ILC Objective 4

Question Origin: Modified (2018 NRC RO 52)

Comments: K/A is matched since the applicant must verify plant alignment following receipt of a system low pressure alarm.

Tier/Group: T2/G1

54. 2020 NRC RO 054

Given the following plant conditions:

- The unit is operating at 100% power
- An Instrument Air leak is occurring
- Instrument Air pressure is currently 80 psig and stable

Which ONE of the following predicts the plant response for the current condition?

A. All FW flow control valves will SHUT

B. PRZ Spray valves drift to mid-position

C. RCS letdown flowpath valves drift to mid-position

D. Gland Steam Seal Spillover Regulator Valve will OPEN

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-017, Loss of Instrument Air, Attachment 7, when IA pressure lowers to 85 psig, the RCS letdown flowpath valves (located in Containment) begin to fail to mid-position.

A. Incorrect. *Plausible since the FW flow control valves auto shut with low IA pressure, but the pressure at which they close has not been reached yet. AOP-017 Attachment 7 states that the valves will auto shut when IA pressure is 60 psig.*

B. Incorrect. *Plausible since AOP-017, Attachment 7 states that all remaining air-operated valves will no longer be considered reliable, but this IA pressure has not been reached yet. Attachment 7 indicates that the IA pressure for this to occur is at 35 psig.*

C. Correct.

D. Incorrect. *Plausible since the majority of gland seal air operated valves fail open on the loss of air to the regulators. The spill over regulator line is an air operated valve that normally opens to relieve pressure from the gland sealing system back to the main condenser to prevent damage to the sealing surfaces when the turbine is self-sealing, but this valve fails shut on a loss of IA.*

078 Instrument Air / 8

078K3.01; Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Containment air system

(CFR: 41.7 / 45.6)

Importance Rating:	3.1 3.4
Technical Reference:	AOP-017, Attachment 7, Page 57, Rev. 40
References to be provided:	None
Learning Objective:	ISA-ILC Objective 9.c
Question Origin:	Bank (2013 NRC RO 54)
Comments:	Ask Chief Examiner if asking how a loss of Instrument Air affects components located inside Containment meets this K/A. Phonecon 8/27: Chief Examiner stated that it was acceptable to ask this type of question to meet this K/A. K/A is matched since the applicant must demonstrate an understanding of how low Instrument Air pressure will affect RCS letdown isolation valves which are located inside Containment.
Tier/Group:	T2/G1

55. 2020 NRC RO 055

Given the following plant conditions:

- A LOCA has occurred
- RCS pressure is 600 psig and stable
- Containment pressure is 11.5 psig and lowering
- SI has NOT been reset
- Phase 'A' and 'B' Containment Isolation reset switches have been placed to RESET

Which ONE of the following identifies the status of the Containment Isolation Phase A and Phase B signals?

	Phase A	Phase B
A.	NOT reset	reset
B.	NOT reset	NOT reset
C.	reset	reset
D.	reset	NOT reset

Plausibility and Answer Analysis

Reason answer is correct: Slave relays are reset using the MCB Phase A and Phase B RESET switches. Both signals are capable of being reset with standing signals and even though SI has not been reset.

A. Incorrect. Phase A not resetting is plausible since SI has not been reset and it is the initiating signal for the Phase A signal. Also plausible since RCS pressure is less that the actuation setpoint of 1850 psig. Phase B resetting is correct.

B. Incorrect. Phase A not resetting is plausible since SI has not been reset and it is the initiating signal for the Phase A signal. Also plausible since RCS pressure is less that the actuation setpoint of 1850 psig. Phase B not resetting is plausible since Containment pressure is still above the actuation setpoint of 10 psig.

C. Correct.

D. Incorrect. Phase A not resetting is correct. Phase B not resetting is plausible since Containment pressure is still above the actuation setpoint of 10 psig.

103 Containment / 5

103A4.03; Ability to manually operate and/or monitor in the control room: ESF slave relays

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	2.7 2.7
Technical Reference:	EOP-E-1, Step 19, Rev. 5
References to be provided:	None
Learning Objective:	ESFAS-ILC Objectives 8.c & 8.d
Question Origin:	Bank
Comments:	K/A is matched since the applicant must demonstrate the ability to reset ESF slave relays using the MCB switches.
Tier/Group:	T2/G1

56. 2020 NRC RO 056

Given the following plant conditions:
- A Steam Generator tube rupture occurred
- The Reactor was tripped and Safety Injection actuated
- All offsite power was lost following the Reactor trip

Subsequently:
- The crew is at the step in EOP-E-3, Steam Generator Tube Rupture, to depressurize the RCS to restore inventory

Which ONE of the following completes the statements below?

The RCS will be depressurized using ___(1)___.

Due to the loss of power during depressurization, the ___(2)___.

A. (1) one PRZ PORV

(2) Reactor Vessel upper head may void resulting in a rapidly rising PRZ level

B. (1) one PRZ PORV

(2) Steam Generator tubes may void causing a loss of natural circulation

C. (1) Auxiliary Spray

(2) Reactor Vessel upper head may void resulting in a rapidly rising PRZ level

D. (1) Auxiliary Spray

(2) Steam Generator tubes may void causing a loss of natural circulation

Plausibility and Answer Analysis

Reason answer is correct: With a loss of offsite power, no RCPs will be running post Reactor trip due to the loss of power to Aux Buses 1A, 1B, and 1C. With no RCPs running, normal PRZ spray will NOT be available and one PRZ PORV will be used to depressurize the RCS. A NOTE in EOP-E-3 states that the upper head region may void during RCS depressurization if RCPs are not running and that this will result in a rapidly rising PRZ level.

A. Correct.

B. Incorrect. The first part is correct. The second part would be a concern if the RCS was excessively depressurized (adequate RCS subcooling not maintained); however, this is incorrect since the upper head region will be hot with little flow through it and most susceptible to voiding during the RCS depressurization.

C. Incorrect. The first part is plausible since Auxiliary Spray can be used to RCS depressurization in EOP-E-3; however, this is incorrect since Safety Injection must first been terminated and letdown restored before this method of depressurization can be used. The second part is correct.

D. Incorrect. The first part is plausible since Auxiliary Spray can be used to RCS depressurization in EOP-E-3; however, this is incorrect since Safety Injection must first been terminated and letdown restored before this method of depressurization can be used. The second part would be a concern if the RCS was excessively depressurized (adequate RCS subcooling not maintained); however, this is incorrect since the upper head region will be hot with little flow through it and most susceptible to voiding during the RCS depressurization.

002 Reactor Coolant / 2/4

002K5.14; Knowledge of the operational implications of the following concepts as they apply to the RCS: Consequences of forced circulation loss

(CFR: 41.5 / 45.7)

Importance Rating:	3.7 4.2
Technical Reference:	EOP-E-3, Steps 53, 59, and NOTE preceding Step Rev. 7
References to be provided:	None
Learning Objective:	EOP-LP-3.02 Objective 4.f
Question Origin:	Bank (VC Summer)
Comments:	Ask Chief Examiner if question should evaluate RCS inventory (SF2) or Heat Removal from the Rx Core (SF4). Phonecon 4/14: Chief Examiner stated that it was acceptable to meet this K/A by testing the consequences of forced circulation loss. The Safety Function suggestions are primarily provided to categorize JPMs. K/A is matched since the applicant must demonstrate an understanding of the consequences of depressurizing the RCS without RCPs running during a SGTR event.
Tier/Group:	T2/G2

57. 2020 NRC RO 057

A failure of the compensating voltage for Intermediate Range channel NI-35 occurs resulting in NI-35 stabilizing at $2E^{-10}$ amps during a Reactor shutdown.

Which ONE of the following completes the statement below?

IF Intermediate Range channel NI-36 output lowers to less than P-6, THEN _______ will automatically energize.

A. BOTH SR NIs

B. NEITHER SR NI

C. ONLY SR channel NI-31

D. ONLY SR channel NI-32

Plausibility and Answer Analysis

Reason answer is correct: Both IR channels must be below the reset for P-6 for the SR NIs to automatically energize.

A. Incorrect. Plausible since one IR channel is below P-6 and SR NIs reset automatically when IR below P-6, but must have 2/2 IR channels <P-6.

B. Correct.

C. Incorrect. Plausible since one IR channel is below P-6 and SR NIs reset automatically when IR below P-6, but must have 2/2 IR channels <P-6 and SR resets are not train-related.

D. Incorrect. Plausible since one IR channel is below P-6 and SR NIs reset automatically when IR below P-6, but must have 2/2 IR channels <P-6 and SR resets are not train-related.

015 Nuclear Instrumentation / 7

015K6.01; Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Sensors, detectors, and indicators

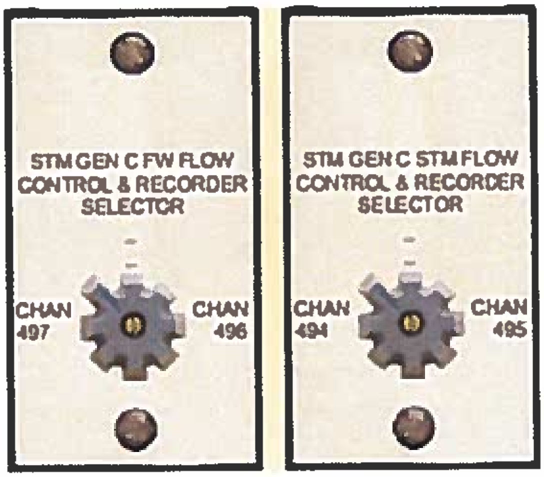
(CFR: 41.7 / 45.7)

Importance Rating:	2.9 3.2
Technical Reference:	GP-006, Section 6.3, CAUTION preceding Step 7, Page 40, Rev. 92
References to be provided:	None
Learning Objective:	NIS-ILC Objective 8.c
Question Origin:	Bank (2002 NRC RO Exam)
Comments:	K/A is matched since the applicant must demonstrate an understanding of how failure of IRNI compensating voltage impacts the operation of the SRNIs during a Reactor shutdown.
Tier/Group:	T2/G2

58. 2020 NRC RO 058

Given the following plant conditions:

- The unit is operating at 100% power
- The 'C' SG Control and Recorder Selector switches are as follows:



Subsequently:

- The controlling 'C' SG Feed Flow channel fails high
- Annunciator ALB-014-6-1A, SG C FW > STM Flow Mismatch, alarms

Which ONE of the following completes the statements below?

Immediately after the failure, the 'C' SG FRV will start to go __(1)__.

Once 'C' SG level is under operator control, OP-134, Feedwater System, will direct the operator to select __(2)__ to restore automatic water level control.

A. (1) OPEN

(2) STM GEN C FW Flow Chan 496 ONLY

B. (1) SHUT

(2) STM GEN C FW Flow Chan 496 ONLY

C. (1) OPEN

(2) STM GEN C FW Flow Chan 496 AND STM GEN C STM Flow Chan 495

D. (1) SHUT

(2) STM GEN C FW Flow Chan 496 AND STM GEN C STM Flow Chan 495

Plausibility and Answer Analysis

Reason answer is correct: If the controlling level channel (Channel III) fails high, SGWLC will see a level error and modulate the FRV shut for the affected SG to try to restore level to program. Actual SG level will lower. The lowering feed flow will result in a steam flow/feed flow mismatch, which will produce a flow error that opposes the level error (flow error will tend to modulate the FRV open). The operators will take manual control of the affected SG Feed Reg Value in accordance with AOP-010. In order to restore the affected SG Feed Reg Valve to automatic control, the operator must perform OP-134 Section 8.10. For the 'C' SG, the operators must position both the FW flow and STM Flow channels to the channels that were not in control. In this case, Channel 496 would be selected for FW Flow and Channel 495 would be selected for STM flow (Channel IV).

A. Incorrect. The first part is plausible since if this was the STM Flow channel that failed, the FRV would OPEN in response to the failure. The second part is plausible since only the FW Flow channel has failed; however, this is incorrect since the OP requires the selection of BOTH channels.

B. Incorrect. The first part is correct. The second part is plausible since only the FW Flow channel has failed; however, this is incorrect since the OP requires the selection of BOTH channels.

C. Incorrect. The first part is plausible since if this was the STM Flow channel that failed, the FRV would OPEN in response to the failure. The second part is correct.

D. Correct.

016 Non-Nuclear Instrumentation / 7

016K3.12; Knowledge of the effect that a loss or malfunction of the NNIS will have on the following: S/G

(CFR: 41.7 / 45.6)

Importance Rating: 3.4 3.6

Technical Reference: APP-ALB-014, Window 6-1A, Page 36, Rev. 30
AOP-010, Section 3.0, Step 7, Page 6, Rev. 40
OP-134, Section 8.10, Step 3, Page 46, Rev. 45

References to be provided: None

Learning Objective: SGWLC-ILC Objective 5.b

Question Origin: Bank (2014 NRC RO 59)

Comments: K/A is matched since the applicant must demonstrate an understanding how a failure of a feed flow transmitter will affect SG water level control.

Tier/Group: T2/G2

59. 2020 NRC RO 059

Which ONE of the following identifies the 480V power supply for S-1B, Containment Airborne Radioactivity Removal (ARR) Fan?

A. MCC 1B21-SB

B. MCC 1E11

C. Bus 1B1

D. Bus 1D2

Plausibility and Answer Analysis

Reason answer is correct: In accordance with the electrical lineup checklist for the containment ventilation and relief system:

1E11-1E Airborne Radioactive Removal Fan S-1 (1B-NNS) (both breakers)

A. Incorrect. Plausible since other Containment fans (Containment Fan Coolers 1 & 4) are powered from MCC 1B21-SB.

B. Correct.

C. Incorrect. Plausible since other Containment fans (CRDM Fans E-80B and E-81B) are powered from MCC 1B24 via Bus 1B1.

D. Incorrect. Plausible since other Containment Fans (Containment Purge Fans AH-81A and AH-82A) are powered from MCC 1D21 via Bus 1D2.

027 Containment Iodine Removal / 5

027K2.01; Knowledge of bus power supplies to the following: Fans

(CFR: 41.7)

Importance Rating:	3.1 3.4
Technical Reference:	OP-168, Attachment 1, Page 29, Rev. 41
References to be provided:	None
Learning Objective:	CVS-ILC Objective 2a
Question Origin:	Modified (2016 NRC RO 59)
Comments:	K/A is matched since applicant must recall the power supply for a fan used to remove airborne radioactivity from the Containment atmosphere.
Tier/Group:	T2/G2

60. 2020 NRC RO 060

Given the following plant conditions:

- The unit was operating at 100% power when a LOCA develops inside Containment

Subsequently:

- Containment pressure rises to a peak value of 12 psig
- Containment hydrogen concentration is 0.5%

Current plant conditions:

- Safety Injection System is aligned for Cold Leg Recirculation
- Containment hydrogen concentration is 5%
- Containment pressure is 3.5 psig

Which ONE of the following completes the statements below in accordance with OP-125, Hydrogen Monitoring System (HMS)?

The Containment Isolation Phase ___(1)___ signal must be reset to allow aligning the Hydrogen Monitoring System from Standby to Continuous Sample Mode.

The Hydrogen Purge System ___(2)___ designed to be placed in service based on the current plant conditions.

A. (1) A

(2) is

B. (1) A

(2) is NOT

C. (1) B

(2) is

D. (1) B

(2) is NOT

Reason answer is correct: During the performance of the EOP network, once the Low Head and High Head Safety Injection systems are in Cold Leg Recirculation Mode, the Hydrogen Monitoring System is placed in Continuous Sample Mode. In order to do so, Phase A Containment Isolation must first be reset to allow re-opening the sampling containment isolation valves. The hydrogen concentration is monitored until the concentration rises to 4% or more, at which time the plant staff evaluates additional recovery actions including the use of hydrogen purge in order to reduce hydrogen concentration. With containment hydrogen concentration above the normal levels (normally below the minimum detectable), the Hydrogen Purge System may be placed in service to reduce hydrogen concentration; however, the system is NOT designed for operation with the containment pressurized and is not placed in service until the containment building is at atmospheric conditions.

Reason answer is correct:

A. Incorrect. The first part is correct. The second part is plausible since other containment ventilation systems such as containment cooling are allowed to be in service when containment is above atmospheric pressure and the applicant may improperly determine operation of the hydrogen purge system is allowed. Additionally, the applicant may determine that because the hydrogen concentration is 5%, reducing hydrogen concentration is required in order to restore containment to normal conditions; however, this is incorrect because the hydrogen purge system is designed to dilute the hydrogen concentration and reduce it below 4% when containment is at atmospheric conditions, not while pressurized.

B. Correct.

C. Incorrect. The first part is plausible since a Containment Phase B Isolation occurred when containment pressure exceeded 10 psig; however, this is incorrect since the hydrogen sampling valves SHUT on a Phase A signal. The second part is plausible since other containment ventilation systems such as containment cooling are allowed to be in service when containment is above atmospheric pressure and the applicant may improperly determine operation of the hydrogen purge system is allowed. Additionally, the applicant may determine that because the hydrogen concentration is 5%, reducing hydrogen concentration is required in order to restore containment to normal conditions; however, this is incorrect because the hydrogen purge system is designed to dilute the hydrogen concentration and reduce it below 4% when containment is at atmospheric conditions, not while pressurized.

D. Incorrect. The first part is plausible since a Containment Phase B Isolation occurred when containment pressure exceeded 10 psig; however, this is incorrect since the hydrogen sampling valves SHUT on a Phase A signal. The second part is correct.

028 Hydrogen Recombiner and Purge Control / 5

028A1.02; Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Containment pressure

(CFR: 41.5 / 45.5)

Importance Rating: 3.4 3.7

Technical Reference: EOP-E-1, Step 25, Page 30, Rev. 5
OP-125, P&L #4 & Section 8.4, Pages 5 & 23, Rev. 27

References to be provided: None

Learning Objective: HMS-ILC Objectives 1.b & 7.a

Question Origin: Modified (2016 NRC RO 60)

Comments: K/A is matched since the applicant must predict the impact high Containment pressure has on the Hydrogen Monitoring System (both monitoring and purge functions)

Tier/Group: T2/G2

61. 2020 NRC RO 061

The following plant conditions exist:

- Core reload is in progress in accordance with GP-009, Refueling Cavity, Refueling and Drain of the Refueling Cavity, Modes 5-6-5

Subsequently:

- A fuel assembly is being lowered into a wrong core location
- An unanticipated rise on both Source Range channels is observed

Which ONE of the following completes the statements below?

Manipulator Crane Overload and Underload Interlocks stop hoist travel when the hoist load reaches a MINIMUM of ___(1)___ lbs above or below the weight of the mast and fuel assembly to prevent fuel damage.

In accordance with GP-009, core reload must be suspended if BOTH Source Range channels count rates rise by a MINIMUM factor of ___(1)___.

A. (1) 150

(2) two

B. (1) 150

(2) five

C. (1) 1200

(2) two

D. (1) 1200

(2) five

Plausibility and Answer Analysis

Reason answer is correct: Per FHP-400, Underload and Overload interlocks prevent fuel damage by stopping hoist travel when hoist load is 150 lbs above or below the weight of the mast and fuel assembly. In accordance with GP-009, core Alterations shall be suspended if an unanticipated increase in count rate by a factor of two occurs simultaneously on both Source Range channels.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since an unanticipated increase in count rate by a factor of FIVE on EITHER Source Range channel is one of the criteria for suspending Core Alterations; however, this is incorrect since the MINIMUM rise on BOTH Source Channels simultaneously is a factor of TWO.

C. Incorrect. The first part is plausible since a 1200 lbs Safety Interlock protects fuel by ensuring that a fuel assembly is not lifted by a disengaged gripper; however, the Overload and Underload Interlock setpoint is 150 psig. The second part is correct.

D. Incorrect. The first part is plausible since a 1200 lbs Safety Interlock protects fuel by ensuring that a fuel assembly is not lifted by a disengaged gripper; however, the Overload and Underload Interlock setpoint is 150 psig. The second part is plausible since an unanticipated increase in count rate by a factor of FIVE on EITHER Source Range channel is one of the criteria for suspending Core Alterations; however, this is incorrect since the MINIMUM rise on BOTH Source Channels simultaneously is a factor of TWO.

034 Fuel-Handling Equipment / 8

034A2.03; Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Mispositioned fuel element

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.3 4.0
Technical Reference:	FHP-400, Attachment 3, Pages 68 & 69, Rev. 4 GP-009 P&L #2, Pages 6, Rev. 68
References to be provided:	None
Learning Objective:	FHS-ILC Objective 6 GP-LP-3.09 Objective 3
Question Origin:	New
Comments:	K/A is matched since the applicant must predict the potential impact of lowering a fuel assembly into a wrong core location (manipulator crane interlocks) and use the refueling procedure to determine when fuel movement must be stopped to mitigate the consequences.
Tier/Group:	T2/G2

62. 2020 NRC RO 062

Given the following plant conditions:
- A waste gas release is in progress
- WPB Stack 5 PIG Monitor, REM-*1WV-3546, exceeds the HIGH ALARM setpoint

Which ONE of the following identifies how the release will be automatically terminated?

A. 3WG-230, Gas Decay Tanks to Plant Vent Isolation Valve, SHUTS

B. 3WG-229, WG Decay Tanks E&F to Plant Vent Valve, SHUTS

C. Filtered Exhaust Fans, E-46, E-47, E-48, and E-49 TRIP

D. Running Waste Gas Compressor TRIPS

Plausibility and Answer Analysis

Reason answer is correct: A high radiation level on the waste gas discharge line closes the trip valve (3WG-229) automatically.

A. Incorrect. Plausible since closure of this valve would terminate the release; however, this is incorrect since 3WG-230 is a manual valve that must be locally operated.

B. Correct.

C. Incorrect. Plausible since these fans supply the release path and would terminate the forced air release; however, this is incorrect since these fans do not receive trip signals on high radiation. Examples of other fans that trip on high radiation levels include the CNMT normal and pre-entry purge fans.

D. Incorrect. Plausible since the running compressor tripping would terminate the release; however, this is incorrect since the compressors do not receive trip signals on high radiation. Examples of other pumps (motors) that trip on high radiation levels include auxiliary condensate, sump, and transfer pumps.

071 Waste Gas Disposal / 9

071K4.06; Knowledge of design feature(s) and/or interlock(s) which provide for the following: Sampling and monitoring of waste gas release tanks

(CFR: 41.7)

Importance Rating:	2.7 3.5
Technical Reference:	AOP-005, Attachment 3, Page 15, Rev. 30
References to be provided:	None
Learning Objective:	RMS-ILC Objective 6.a
Question Origin:	Bank (2002 NRC RO Exam)
Comments:	K/A is matched since the applicant must demonstrate an understanding of the interlock which terminates a waste gas release on high radiation condition.
Tier/Group:	T2/G2

63. 2020 NRC RO 063

Which ONE of the following is an input to the Containment Critical Safety Function Status Tree (CSF-5)?

A. RM-01CR-3589SA, High Range Containment Post Accident

B. REM-01LT-3502ASA, Containment RCS Leak Detection

C. RM-01CR-3561BSB, Containment Ventilation Isolation

D. REM-01LT-3502B, Containment Pre-Entry Purge

Plausibility and Answer Analysis

Reason answer is correct: RM-01CR-3589SA and RM-01CR-3589SB are both inputs for the Containment Critical Safety Function Status Tree (CSF-5).

A. Correct.

B. Incorrect. Plausible since a high alarm on REM-01LT-3502ASA isolates Normal Containment Purge; however, this is incorrect as it is not an input to the Containment Critical Safety Function Status Tree.

C. Incorrect. Plausible since a high alarm on RM-01CR-3561BSB is an input to Containment Ventilation Isolation; however, this is incorrect as it is not an input to the Containment Critical Safety Function Status Tree.

D. Incorrect. Plausible since a high alarm on REM-01LT-3502B isolates Containment Pre-Entry Purge; however, this is incorrect as it is not an input to the Containment Critical Safety Function Status Tree.

072 Area Radiation Monitoring / 7

072G2.4.21; 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)

Importance Rating:	4.0 4.6
Technical Reference:	OP-118, Attachment 3, Page 99, Rev. 39 EOP-CSFST, Containment CSFST, Page 3, Rev. 13
References to be provided:	None
Learning Objective:	EOP-LP-3.13 Objective 6.a
Question Origin:	New
Comments:	Ask Chief Examiner if acceptable to ask a question which does not test the logic portion of K/A statement. Phonecon 8/27: Chief Examiner stated that it was acceptable to read this K/A as "and/or", that is "parameters and/or logic". K/A is matched since the applicant must recall which Containment area radiation monitor is used to monitor Containment conditions via the Containment CSFST.
Tier/Group:	T2/G2

64. 2020 NRC RO 064

Given the following plant conditions:
- The unit is operating at 100% power
- 'A' Normal Service Water (NSW) Pump is running

Subsequently:
- The 'A' Emergency Service Water (ESW) Pump control switch is taken to START

Which ONE of the following completes the statements below regarding the Service Water valve alignment two (2) minutes following the pump start?

1SW-39, NSW Supply to 'A' ESW Header, will be __(1)__.

1SW-276, ESW to NSW Common Return, will be __(2)__.

A. (1) SHUT

(2) SHUT

B. (1) SHUT

(2) OPEN

C. (1) OPEN

(2) SHUT

D. (1) OPEN

(2) OPEN

Plausibility and Answer Analysis

Reason answer is correct: Normally open 1SW-39 is interlocked to shut on the start of its associated ESW pump. Normally open 1SW-276 does not receive a shut signal upon the start of any ESW pump (1SW-276 will only auto shut due to SI, LOOP, or 'A' ESW Pump start when control is transferred to the ACP).

A. Incorrect. The first part is correct. The second part is plausible since the individual return valve (1SW-275) will shut; however, this is incorrect since a manual ESW pump start will not shut the common return (1SW-276).

B. Correct.

C. Incorrect. The first part is plausible since one NSW supply valve (1SW-40) will remain open; however, 1SW-39 will receive a shut signal upon start of the 'A' ESW Pump. The second part is plausible since the individual return valve (1SW-274) will shut; however, this is incorrect since a manual ESW pump start will not shut the common return (1SW-276).

D. Incorrect. The first part is plausible since one NSW supply valve (1SW-40) will remain open; however, 1SW-39 will receive a shut signal upon start of the 'A' ESW Pump. The second part is correct.

075 Circulating Water / 8

075A4.01; Ability to manually operate and/or monitor in the control room: Emergency/essential SWS pumps

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	3.2 3.2
Technical Reference:	OP-139, Section 5.2.2, Page 23, Rev. 138
References to be provided:	None
Learning Objective:	SWS-ILC Objective 4
Question Origin	Bank
Comments:	K/A is matched since the applicant must demonstrate an understanding of Service Water System alignment following the manual start of one Service Water pump.
Tier/Group:	T2/G2

65. 2020 NRC RO 065

Given the following plant conditions:

- Fire header pressure is 123 psig

Subsequently:

- A fire occurs on site
- Fire header pressure lowers to 70 psig for 30 seconds then recovers

Fire header pressure is currently 102 psig with the Fire Jockey Pump running.

Which ONE of the following completes the statements below?

The Motor Driven Fire Pump will be __(1)__.

The Diesel Driven Fire Pump will be __(2)__.

(Assume NO operator actions have been taken)

A. (1) OFF

(2) OFF

B. (1) OFF

(2) RUNNING

C. (1) RUNNING

(2) OFF

D. (1) RUNNING

(2) RUNNING

Plausibility and Answer Analysis

Reason answer is correct: FPT 3001 acceptance criteria # 2 states that Motor Driven Fire Pump starts at greater than or equal to 90 psig as indicated on PI-*1FP-8622B. FPT 3010 acceptance criteria # 2 states that Diesel Driven Fire Pump pressure switch actuates at greater than or equal to 73 psig as indicated on PI-*1FP-8622A. There is an 8 second time delay associated with the Diesel Driven Fire Pump auto start.

A. Incorrect. Plausible if the applicant believes the Motor Driven and Diesel Driven Fire Pumps will automatically stop once fire header pressure rises above the auto start setpoints.

B. Incorrect. The first part is plausible since the applicant may believe the Motor Driven Fire Pump will automatically stop once fire header pressure rises above its auto start setpoint. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since the applicant may believe the Diesel Driven Fire Pump will automatically stop once fire header pressure rises above its auto start setpoint.

D. Correct.

086 Fire Protection / 8

086A3.01; Ability to monitor automatic operation of the Fire Protection System including: Starting mechanisms of fire water pumps

(CFR: 41.7 / 45.5)

Importance Rating:	3.3 3.3
Technical Reference:	FPT-3001, Section 6.0, Page 4, Rev. 14 FPT-3010, Section 6.0, Page 5, Rev. 19
References to be provided:	None
Learning Objective:	FP-ILC Objective 9
Question Origin:	Modified (2016 NRC RO 65)
Comments:	K/A is matched since the applicant must demonstrate the ability to determine which fire pumps auto started with a reduction in fire header pressure.
Tier/Group:	T2/G2

66. 2020 NRC RO 066

Which ONE of the following completes the statement below in accordance with AD-OP-ALL-1000, Conduct of Operations?

If needed to protect the plant, the ___(1)___ can authorize resetting a protective device without knowing the cause provided a(an) ___(2)___ condition is NOT evident.

A. (1) Control Room Supervisor

(2) thermal overload

B. (1) Control Room Supervisor

(2) overcurrent

C. (1) Shift Manager

(2) thermal overload

D. (1) Shift Manager

(2) overcurrent

Reason answer is correct: In accordance with AD-OP-ALL-1000, Section 5.5.16, the SM is authorized to reset any tripped protective device without knowing the cause, provided an overcurrent condition is NOT evident.

A. Incorrect. The first part is plausible since the CRS is the authorizing watchstander for many actions required to protect the plant; however, this is incorrect since AD-OP-ALL-1000 requires SM authorization to reset any tripped protective device. The second part is plausible since overload relays are used to protect safety-related equipment and in turn protect the plant; however, this is incorrect since the SM is authorized to reset this type of protective device.

B. Incorrect. The first part is plausible since the CRS is the authorizing watchstander for many actions required to protect the plant; however, this is incorrect since AD-OP-ALL-1000 requires SM authorization to reset any tripped protective device.

C. Incorrect. The first part is correct. The second part is plausible since overload relays are used to protect safety-related equipment and in turn protect the plant; however, this is incorrect since the SM is authorized to reset this type of protective device. The second part is correct.

D. Correct.

2.1 Conduct of Operations

G2.1.1; Knowledge of conduct of operations requirements.

(CFR: 41.10 / 45.13)

Importance Rating:	3.8 4.2
Technical Reference:	AD-OP-ALL-1000, Section 5.5.16, Page 43, Rev. 17
References to be provided:	None
Learning Objective:	PP-LP-3.00 Objective 10.g
Question Origin:	New
Comments:	K/A is matched since applicant must demonstrate an understanding of the requirements found in the fleet conduct of operations procedure regarding resetting protective devices.
Tier/Group:	T3

67. 2020 NRC RO 067

Given the following plant conditions:

- A post-maintenance lineup is being performed
- Circuit Breaker 1A-SA-5, Charging/SI Pump 1A-SA Breaker, is being racked in

Which ONE of the following completes the statements below in accordance with AD-HU-ALL-0005, Human Performance Tools?

For this evolution, verification of "racked in" status for the breaker MUST ___(1)___.

Circuit Breaker 1A-SA-5 ___(2)___ require Independent Verification.

A. (1) be performed LOCALLY

(2) does

B. (1) be performed LOCALLY

(2) does NOT

C. (1) use the MCB indicating light

(2) does

D. (1) use the MCB indicating light

(2) does NOT

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AD-HU-ALL-0005, Attachment 11, verification of racking in status for rackable breakers (e.g., 6900 VAV, 4160 VAC, and 480 VAC load distribution breakers) must be performed LOCALLY since remote indicating lights are functional prior to the breaker reaching the fully racked in position. Independent Verification is required for system alignments of safe-related or important equipment following an outage when the system was NOT maintained in the normal alignment or when returning a system to service following maintenance.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since not all 6.9KV breakers (i.e. non-safety-related equipment) require Independent Verification per AD-HU-ALL-0005.

C. Incorrect. The first is plausible since the remote indicating light can be used, but only after local observation has been performed to verify that the remote indication reflects the component's position. The second part is correct.

D. *Incorrect. The first is plausible since remote indicating light can be used, but only after local observation has been performed to verify that the remote indication reflects the component's position. The second part is plausible since not all 6.9KV breakers (i.e. non-safety-related equipment) require Independent Verification per AD-HU-ALL-0005.*

2.1 Conduct of Operations

G2.1.29; Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

(CFR: 41.10 / 45.1 / 45/12)

Importance Rating:	4.1 4.0
Technical Reference:	AD-HU-ALL-0005, Attachment 11, Pages 29 & 32, Rev. 4
References to be provided:	None
Learning Objective:	PP-LP-3.11 Objective 8
Question Origin:	Bank (Robinson)
Comments:	Phonecon 7/2/2020: HNP discussed being unable to create a generic T3 question based on system purpose and/or function for the generic K/A topic knowledge of system purpose and /or function, so selected a new K/A, keeping Generic topic 2.1, Conduct of Operations and determined a different randomly selected K/A: New K/A G2.1.29: Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc. K/A is matched since the applicant must demonstrate an understanding of fleet procedure requirements regarding restoration of a safety-related breaker post-maintenance.
Tier/Group:	T3

68. 2020 NRC RO 068

Which ONE of the following completes the statement below in accordance with OMM-002, Shift Turnover Package?

With the unit in Mode 4, the MINIMUM shift crew composition must include __(1)__ Reactor Operator(s) and __(2)__ Auxiliary Operator(s).

A. (1) one

(2) one

B. (1) one

(2) two

C. (1) two

(2) one

D. (1) two

(2) two

Plausibility and Answer Analysis

Reason answer is correct: Per OMM-002 Section 5.1, minimum shift crew composition shall comply with Technical Specification 6.2.2. In Modes 1 - 4, this requires a minimum of two ROs and two AOs.

A. Incorrect. Plausible since this choice would be correct if the unit was in Mode 5 or 6.

B. Incorrect. The first part is plausible since this is the minimum number of ROs required in Mode 5 or 6. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since this is the minimum number of AOs required in Mode 5 or 6.

D. Correct.

2.1 Conduct of Operations

G2.1.5; Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

(CFR: 41.10 / 43.5 / 45.12)

Importance Rating:	2.9 3.9
Technical Reference:	OMM-002, Section 5.1, page 8, Rev. 69 Technical Specifications, Table 6.2-1
References to be provided:	None
Learning Objective:	PP-LP-3.0 Objective 8
Question Origin:	Bank
Comments:	K/A is matched since the applicant must demonstrate the ability to apply OMM-002 minimum shift staffing requirements to Mode 4.
Tier/Group:	T3

69. 2020 NRC RO 069

A Reactor startup is in progress in accordance with GP-004, Reactor Startup (Mode 3 to Mode 2).

Which ONE of the following completes the statements below?

In accordance with AD-OP-ALL-0203, Reactivity Management, the dedicated Reactor Operator for this evolution __(1)__ be one of the Reactor Operators on the crew.

The Reactor Operator conducting the Reactor startup should expect criticality to be achieved (based on the ECP) when Control Bank __(2)__ reaches 90 steps.

A. (1) can

(2) C

B. (1) can

(2) D

C. (1) can NOT

(2) C

D. (1) can NOT

(2) D

Reason answer is correct: In accordance with AD-OP-ALL-0203, Section 5.2.1, the dedicated RO can be one of the unit reactor operators. In accordance with GP-004, most startups use 90 steps on Control Bank D as the target for criticality, but special conditions may warrant a target anywhere between 90 and 130 steps, as determined by Reactor Engineering.

A. Incorrect. The first part is correct. The second part is plausible since Control Bank C rods are still being withdrawn when criticality is achieved on Control Bank D (rod bank overlap); however, this is incorrect since most startups will use 90 steps on Control Bank D as the target for criticality. Criticality should not be achieved solely due to withdrawing Control Bank C rods.

B. Correct.

C. Incorrect. The first part is plausible since the dedicated Reactor Operator can be in addition to the crew, but does not have to be. The second part is plausible since Control Bank C rods are still being withdrawn when criticality is achieved on Control Bank D (rod bank overlap); however, this is incorrect since most startups will use 90 steps on Control Bank D as the target for criticality. Criticality should not be achieved solely due to withdrawing Control Bank C rods.

D. Incorrect. The first part is plausible since the dedicated Reactor Operator can be in addition to the crew, but does not have to be. The second part is correct.

2.2 Equipment Control

G2.2.2; 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)

Importance Rating:	4.6 4.1
Technical Reference:	AD-OP-ALL-0203, Section 5.2.1 & Attachment 2, Pages 23 & 63, Rev. 13 GP-004, Section 6.0, NOTE preceding Step 1, Page 12, Rev. 66
References to be provided:	None
Learning Objective:	GP-3.04 Objective 6 SU-SIM-3.23 Objective 2
Question Origin:	New
Comments:	Ask Chief Examiner if question needs to address plant startup or shutdown conditions. Phonecon 7/30: Chief Examiner stated that K/A does not need to address startup/shutdown conditions. K/A is matched since the applicant must demonstrate the ability to operate control rods during a Reactor startup.
Tier/Group:	T3

70. 2020 NRC RO 070

Given the following plant conditions:

- The Reactor is shutdown for a scheduled refueling outage
- An RCS cooldown is in progress IAW GP-007, Normal Plant Cooldown

The following information is a plot of the cooldown:

TIME	RCS Tcold
0830	516°F
0845	505°F
0900	487°F
0915	477°F
0930	465°F
0945	441°F
1000	405°F
1015	378°F
1030	363°F

Of the times listed below, when was the Technical Specification RCS cooldown rate limit FIRST exceeded?

A. 0900

B. 0930

C. 1000

D. 1030

Plausibility and Answer Analysis

Reason answer is correct: At 1030, the RCS has cooled down 102°F in the last hour which is in excess of the 100°F/hr Tech Spec limit for Mode 3 (Tcold ≥ 350°F).

A. Incorrect. Plausible since 15 minute change in temperature is 18°F which corresponds to rate (72°F/hr) that would exceed the 1 hour cooldown limit of 50°F/ hr when RCS Tcold < 350°F.

B. Incorrect. Plausible since 51°F in 1 hour is greater than 50°F required by Tech Specs if Tcold < 350°F, but the RCS Tcold ≥ 350°F at this time.

C. Incorrect. Plausible since 15 minute change in temperature is 36°F which corresponds to rate (144°F/hr) that would exceed the 1 hour cooldown limit of 100°F/ hr when RCS Tcold ≥ 350°F.

D. Correct.

2.2 Equipment Control

G2.2.42; Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

(CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

Importance Rating:	3.9 4.6
Technical Reference:	Tech Spec 3.4.9.1, Page 3/4 4-33
References to be provided:	None
Learning Objective:	RCS-ILC Objective 12.o
Question Origin:	Bank (2011 NRC RO 69)
Comments:	Early Submittal K/A is matched since applicant must evaluate plant conditions and determine when Technical Specifications are required to be entered for excessive RCS cooldown.
Tier/Group:	T3

71. 2020 NRC RO 071

Which ONE of the following completes the statements below regarding operation of the DISCP (RMS) Human Machine Interface?

Operators may navigate between screens and choose options using the ___(1)___.

Only the ___(2)___ can be used to control the functions of the safety-related monitors.

(DISCP = Distributed Instrumentation and Control System Platform)

A. (1) keyboard ONLY

(2) RM-23

B. (1) keyboard ONLY

(2) DICSP

C. (1) mouse AND keyboard

(2) RM-23

D. (1) mouse AND keyboard

(2) DICSP

Reason answer is correct: In accordance with OP-118, Attachment 10, the operators may NAVIGATE between screens and CHOOSE options using any of multiple methods including the mouse and keyboard. The RM-23 can control the functions of safety-related monitors. The DICSP (RMS) can only monitor the functions of the safety-related monitors.

A. Incorrect. The first part is plausible since the RM-11 platform has just recently been replaced by DICSP. The RM-11 required use of the keyboard to navigate screens and choose options. However, DISCP allows use of the keyboard or a mouse to navigate and choose. The second part is correct.

B. Incorrect. The first part is plausible since the RM-11 platform has just recently been replaced by DICSP. The RM-11 required use of the keyboard to navigate screens and choose options. However, DISCP allows use of the keyboard or a mouse to navigate and choose. The second part is plausible since the DICSP (RMS) can monitor the functions of the safety-related monitors, but not control the functions.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since the DICSP (RMS) can monitor the functions of the safety-related monitors, but not control the functions.

2.3 Radiation Control

G2.3.5; Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

(CFR: 41.11 / 41.12 / 43.4 / 45.9)

Importance Rating:	2.9 2.9
Technical Reference:	OP-118, Attachment 10, Page 113, Rev. 39
References to be provided:	None
Learning Objective:	RMS-ILC Objective 2.h & 7.a
Question Origin:	New
Comments:	K/A is matched since the applicant must demonstrate the ability to use the DICSP HMI as it relates to operation of the Radiation Monitoring System.
Tier/Group:	T3

72. 2020 NRC RO 072

Given the following:

- A valve lineup will be performed in the RCA
- Highest general radiation levels are 20 mrem/hr
- Highest general area contamination levels are 1,000 dpm/100 cm^2
- The valve lineup requires accessing one valve 10 feet in the overhead

Which ONE of the following completes the statement below in accordance with PD-RP-ALL-0001, Radiation Worker Responsibilities?

The RWP Self-Briefing process is _______.

A. allowed for the given conditions

B. NOT allowed due to the overhead work

C. NOT allowed due to the area radiation levels

D. NOT allowed due to the area contamination levels

Plausibility and Answer Analysis

Reason answer is correct: PD-RP-ALL-0001 does not allow the Self-Briefing process to be used if there will be entry or work in:

* High Radiation Areas
* Radiation Areas greater than 25 mrem/hr
* Contaminated Areas greater than 10,000 dpm/100 cm^2
* Posted Alpha or Airborne Radioactivity Areas
* Overhead work above 7 feet

A. Incorrect. Plausible since this choice would be correct if the overhead work was less than or equal to 7 feet. All conditions would be met for Self-Briefing.

B. Correct.

C. Incorrect. Plausible since this choice would be correct if radiation levels were greater than 25 mrem/hr and the overhead work was less than or equal to 7 feet.

D. Incorrect. Plausible since this choice would be correct if contamination levels were greater than 10,000 dpm/100 cm^2 and the overhead work was less than or equal to 7 feet. A value of 1,000 dpm/100 cm^2 was used for plausibility since this is the value found on the Operations RWP for expected radiological conditions in the RCA.

2.3 Radiation Control

G2.3.7; Ability to comply with radiation work permit requirements during normal or abnormal conditions.

(CFR: 41.12 / 45.10)

Importance Rating:	3.5 3.6
Technical Reference:	PD-RP-ALL-0001, Section 5.4.3, Step 9, Page 30, Rev. 13
References to be provided:	None
Learning Objective:	PP-LP-3.07 Objective 5
Question Origin:	New
Comments:	K/A is matched since the applicant must demonstrate the ability to comply with RWP self-briefing requirements per PD-RP-ALL-0001.
Tier/Group:	T3

73. 2020 NRC RO 073

Which ONE of the following completes the statements below in accordance with AD-OP-ALL-1001, Conduct of Abnormal Operations?

Event Procedure immediate actions ___(1)___ require CRS concurrence to perform.

If entry conditions are met for multiple Event Procedures, the CRS will ___(2)___ the Event Procedures.

A. (1) do

(2) concurrently enter

B. (1) do

(2) prioritize entry into

C. (1) do NOT

(2) concurrently enter

D. (1) do NOT

(2) prioritize entry into

Plausibility and Answer Analysis

Reason answer is correct: AD-OP-ALL-1001 states that immediate actions will NOT be delayed while waiting on formal entry into the Event Procedure by the CRS. It also directs prioritization of implementation of the Event Procedures (EOPs and AOPs) if multiple Event Procedure entry conditions are met.

A. Incorrect. The first part is a plausible misconception the applicant may have believing formal entry into the Event Procedure is first required before immediate actions can be performed The second part is plausible since concurrent Event Procedure execution may be required, but not required to be entered.

B. Incorrect. The first part is a plausible misconception the applicant may have believing formal entry into the Event Procedure is first required before immediate actions can be performed The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since concurrent Event Procedure execution may be required, but not required to be entered.

D. Correct.

2.4 Emergency Procedures / Plan

G2.4.1; Knowledge of EOP entry conditions and immediate action steps.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	4.6 4.8
Technical Reference:	AD-OP-ALL-1001, Section 5.7, Page 23, Rev. 3
References to be provided:	None
Learning Objective:	ONO-LP-3.0 Objective 4
Question Origin:	New
Comments:	K/A is matched since the applicant must demonstrate an understanding of Event Procedure entry conditions and performance of Event Procedure immediate actions per AD-OP-ALL-1001.
Tier/Group:	T3

74. 2020 NRC RO 074

Given the following plant conditions:
- The unit is operating at 100% power

Subsequently:
- At 0800, a loss of MCB annunciators occurred and the crew entered AOP-037, Loss of Main Control Room Annunciators
- The CRS determined that the following Alarm Light Boxes (ALBs) were lost:
  - ALB-001, Containment Spray & Accumulator System
  - ALB-002, Emergency Service Normal Service Water System
  - ALB-003, Miscellaneous Systems
  - ALB-004, RHR/RWST System

Which ONE of the following completes the statement below regarding the required action per Technical Specifications for loss of these ALBs?

___(1)___ must be first logged no LATER than ___(2)___.

A. (1) Containment sump level

(2) 0810

B. (1) Containment sump level

(2) 0900

C. (1) Temperature and level for both reservoirs

(2) 0810

D. (1) Temperature and level for both reservoirs

(2) 0900

Plausibility and Answer Analysis

Reason answer is correct: A loss of the System 1 24 VDC power supply (1A#1) will result in the loss of ALBs 1, 2, 3, and 4. ALB-001-6-1 meets the requirement for T.S. 3.4.6.b. With this alarm lost, AOP-16 Attachment 16 provides for manual determination of sump in-leakage and the first sump level must be recorded within 10 minutes of alarm inoperability.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since unidentified leakage must be calculated in less than one hour per Attachment 16; however, the first determination of sump level must be obtained within 10 minutes of alarm inoperability.

C. Incorrect. Plausible since the Main and Aux Reservoir alarms for both temperature and level have been lost on ALB-002-7-5; however, this surveillance is met by the normal logging in OST-1021 daily. No additional actions are required. The second part is plausible since actions must be taken within a specified period of time following alarm inoperability and the applicant may have a misconception that this requirement applies to alarms associated with other Technical Specification related parameters (i.e. LCO 3.7.5 - Ultimate Heat Sink).

D. Incorrect. Plausible since the Main and Aux Reservoir alarms for both temperature and level have been lost on ALB-002-7-5; however, this surveillance is met by the normal logging in OST-1021 daily. No additional actions are required. The second part is plausible since actions must be taken within a specified period of time following alarm inoperability and the applicant may have a misconception that this requirement applies to alarms associated with other Technical Specification related parameters (i.e. LCO 3.7.5 - Ultimate Heat Sink).

2.4 Emergency Procedures / Plan

G2.4.32; Knowledge of operator response to loss of all annunciators.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.6 4.0
Technical Reference:	AOP-037, Section 3.0, Step 19, Page 8, Rev. 27 Technical Specifications, Page 3/4 4-21 ALB-001-6-1, Page 22, Rev. 24 AOP-016, Attachment 16, Page 104, Rev. 58 OP-163, Section 6.3.1, Page 16, Rev. 42
References to be provided:	None
Learning Objective:	AOP-LP-3.37 Objective 3
Question Origin:	Bank (2009 NRC RO 74)
Comments:	Ask Chief Examiner if question meets Tier 3 criteria. Email 7/2: Chief Examiner reviewed question and stated it met Tier 3 criteria. K/A is matched since the applicant must demonstrate an understanding of how the loss of an ALB impacts the ability to monitor Containment sump in-leakage and the response required to mitigate.
Tier/Group:	T3

75. 2020 NRC RO 075

Given the following plant conditions:

- The Reactor has tripped and Safety Injection has actuated due to a large break Loss of Coolant Accident (LOCA)
- The crew is implementing EOP-E-1, Loss of Reactor or Secondary Coolant
- The OATC reports the following for Critical Safety Function Status Trees:
  - Containment - Orange
  - Subcriticality - Orange
  - Heat Sink - Red
  - Integrity - Red

Which ONE of the following identifies the procedure required to be entered?

A. EOP-FR-S.1, Response to Nuclear Generation/ATWS

B. EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock

C. EOP-FR-H.1, Response to Loss of Secondary Heat Sink

D. EOP-FR-Z.1, Response to High Containment Pressure

Plausibility and Answer Analysis

Reason answer is correct: In accordance with the EOP-USERS-GUIDE, Section 5.2.2 determines priority of the CSFSTs as: Subcriticality, Core Cooling, Heat Sink, Integrity, Containment, and Inventory. Section 5.2.3, General Usage determines RED as the highest priority followed by ORANGE, YELLOW and GREEN. It also describes how RED paths on lower priority trees must be addressed before ORANGE paths on higher priority trees due to the severe challenge to the safety function. Therefore, FR-H.1 is correct because it is the highest priority Red path CSFST.

A. Incorrect. Plausible due to the Subcriticality tree being the highest priority tree, but an Orange Path is not a higher priority than a Red path.

B. Incorrect. Plausible if the applicant believes that a Red path on the Integrity tree would require the transition to FR-P.1, however this is incorrect because FR-H.1 is a higher priority.

C. Correct.

D. Incorrect. Plausible if the applicant believes that an Orange path on the Containment Tree is a higher priority and would require transition to FR-Z.1.

2.4 Emergency Procedures / Plan

G2.4.4; Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

(CFR: 41.10 / 43.2 / 45.6)

Importance Rating: 4.5 4.7

Technical Reference: EOP Users Guide, Sections 5.2.2 & 5.2.3, Pages 23 & 24, Rev. 51

References to be provided: None

Learning Objective: EOP-LP-3.19 Objective 2.b

Question Origin: Bank (2013 NRC RO 73)

Comments: Early Submittal

K/A is matched since the applicant must evaluate plant conditions and determine entry-level conditions for a functional recovery procedure (EOP).

Tier/Group: T3

76. 2020 NRC SRO 001

Given the following plant conditions:

- The unit is operating at 100% power when a Station Blackout occurs
- 125 VDC power to Auxiliary Buses 1D and 1E has been lost
- Start Up XFMR 1A Lockout SU 1A Relay is tripped
- Aux Bus 1E 86 Lockout Relay is tripped

Subsequently:

- Offsite power is restored
- The appropriate lockout has been reset

Which ONE of the following completes the statement below?

The crew will restore power to 6.9 KV Emergency Bus __(1)__ using EOP-ECA-0.0, Loss of All AC Power, __(2)__.

A. (1) 1A-SA

(2) Attachment 1, Restoration of Offsite Power to Emergency Buses

B. (1) 1A-SA

(2) Attachment 2, Local Restoration of Offsite Power to Emergency Buses

C. (1) 1B-SB

(2) Attachment 1, Restoration of Offsite Power to Emergency Buses

D. (1) 1B-SB

(2) Attachment 2, Local Restoration of Offsite Power to Emergency Buses

Reason answer is correct: Cautions in EOP-ECA-0.0 Attachments 1 and 2 allow resetting any tripped Start Up XFMR Lockout Relays with Load Dispatcher's permission. Resetting this relay allows the re-energization of 6.9 KV Auxiliary Bus D and 6.9 KV Emergency Bus 1A-SA. The Auxiliary Bus 1E 86 Lockout Relay tripped is an indication of an electrical fault on the bus itself and Auxiliary Bus 1E should not be re-energized (Auxiliary Bus 1E must be re-energized to energize Emergency Bus 1B-SB). With DC control power (125 VDC NNS) unavailable to Auxiliary Buses 1D and 1E, Attachment 2 will be used to locally strip all potential bus loads, re-energize the auxiliary buses from the respective startup transformer, and close the supply breaker to the respective 6.9 KV emergency bus.

A. Incorrect. The first part is correct. The second part is plausible since Attachment 1 would normally be used to align offsite power from the MCR; however, this is incorrect since breaker control power is not available with the loss of 125 VDC power to Auxiliary Buses 1D and 1E.

B. Correct.

C. Incorrect. The first part is plausible since some relays can be reset with Load Dispatcher's permission (i.e. Start Up XFMR lockout relays) and resetting the 86 Lockout Relay would allow restoration of power to Aux Bus 1E and ultimately to Emergency Bus 1B-SB; however, this is incorrect since any 87 Bus Differential or 86 Lockout relay tripped is an indication of an electrical fault on the bus itself and the auxiliary bus should not be re-energized. The second part is plausible since Attachment 1 would normally be used to align offsite power from the MCB; however, this is incorrect since breaker control power is not available with the loss of 125 VDC power to Auxiliary Buses 1D and 1E.

D. Incorrect. The first part is plausible since some relays can be reset with Load Dispatcher's permission (i.e. Start Up XFMR lockout relays) and resetting the 86 Lockout Relay would allow restoration of power to Aux Bus 1E and ultimately to Emergency Bus 1B-SB; however, this is incorrect since any 87 Bus Differential or 86 Lockout relay tripped is an indication of an electrical fault on the bus itself and the auxiliary bus should not be re-energized. The second part is correct.

000055 Station Blackout / 6

055EA2.06; Ability to determine or interpret the following as they apply to a Station Blackout: Faults and lockouts that must be cleared prior to re- energizing buses

(CFR 43.5 / 45.13)

Importance Rating:	3.7 4.1
Technical Reference:	EOP-ECA-0.0 Attachment 2, Pages 105 & 106, Rev. 10
References to be provided:	None
Learning Objective:	EOP-LP-3.07 Objective 6
Question Origin:	New
Comments:	K/A is matched as the applicant must interpret multiple lockouts and use this information to make a determination as to which safety bus can be safely re-energized.
Tier/Group:	T1/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

77. 2020 NRC SRO 002

Given the following plant conditions:
- The unit is operating at 100% power
- Group Display AOP-016 is being used to monitor a small SG tube leak
- ERFIS point FCS0150, LETDN HX OUTLET FLOW, indicates 105.8 gpm

Subsequently:
- A loss of 125 VDC DP-1A-SA occurs

Which ONE of the following completes the statements below regarding the status of FCS0150 and the Technical Specification bases for the Electrical Power Systems in Modes 1 through 4?

Following the loss of 125 VDC DP-1A-SA, ERFIS point FCS0150 will ___(1)___.

The OPERABILITY of the power sources are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed ___(2)___.

A. (1) lower to zero gpm

(2) loss of offsite power ONLY

B. (1) lower to zero gpm

(2) loss of offsite power AND single failure of the other onsite A.C. source

C. (1) remain unchanged

(2) loss of offsite power ONLY

D. (1) remain unchanged

(2) loss of offsite power AND single failure of the other onsite A.C. source

Plausibility and Answer Analysis

Reason answer is correct: Per the AOP-025 Basis Document, the letdown orifice isolation valves will fail shut is DP-1A-SA is lost. Per the Technical Specification Bases, the OPERABILITY of the power sources are based upon maintaining at least one redundant set of A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

A. Incorrect. The first part is correct. The second part is plausible since the safety analyses initial conditions assume a loss of offsite power; however, this is incorrect since the analyses also assume a single failure of an onsite A.C source.

B. Correct.

C. Incorrect. The first part is plausible since have a misconception that the letdown orifice isolation valves will be unaffected by the loss of DP-1A-SA since they are AOVs; however, this is incorrect since the letdown orifice valves will fail shut if DP-1A-SA is lost. The second part is correct.

D. Incorrect. The first part is plausible since have a misconception that the letdown orifice isolation valves will be unaffected by the loss of DP-1A-SA since they are AOVs; however, this is incorrect since the letdown orifice valves will fail shut if DP-1A-SA is lost. The second part is plausible since the safety analyses initial conditions assume a loss of offsite power; however, this is incorrect since the analyses also assume a single failure of an onsite A.C source.

000058 Loss of DC Power / 6

058AG2.1.19; Ability to use plant computers to evaluate system or component status.

(CFR: 41.10 / 45.12)

Importance Rating:	3.9 3.8
Technical Reference:	AOP-025-BD, Section 1.0, Page 5, Rev. 21 Technical Specifications Bases, Page B 3/4 8-1
References to be provided:	None
Learning Objective:	AOP-LP-3.25 Objectives 4 & 6 DCP-ILC Objective 11
Question origin:	New
Comments:	K/A is matched since the applicant must demonstrate the ability to monitor a computer (ERFIS) point to evaluate the status of the Letdown system following a loss of DC power.
Tier/Group:	T1/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

78. 2020 NRC SRO 003

Given the following plant conditions:

- The unit is operating at 100% power

Subsequently:

- The Load Dispatcher reports a large disturbance occurring on the grid
- The crew enters AOP-028, Grid Instability

The following conditions are observed:

Time	Grid Frequency (Hz)
0107	59.6
0110	59.2
0113	58.9
0116	58.7
0118	58.5
0121	58.3

Which ONE of the following completes the statements below?

In accordance with AOP-028, the EARLIEST time that the Reactor must be tripped is __(1)__.

When the 6.9 KV Emergency AC Buses are energized from the Emergency Diesel Generators, declaration of an emergency event __(2)__ required.

A. (1) 0118

(2) is

B. (1) 0118

(2) is NOT

C. (1) 0121

(2) is

D. (1) 0121

(2) is NOT

Reason answer is correct: At time 0118, a Reactor trip is required since frequency has been less than 59 Hz for 5 minutes. A CAUTION is AOP-028 states that upon intentionally separating the Emergency AC Buses from the grid, for the purpose of EAL classification ALL Offsite Power should be considered LOST. As such, an Unusual Event would be declared.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since offsite power may still be considered available (based on reports from the Load Dispatcher); however, this is incorrect since Attachment 2 dictates that Offsite Power be considered LOST with respect to EAL classifications if divorced from the grid.

C. Incorrect. The first part is plausible since at time 0121 Reactor trip criteria is met (frequency less than 58.4 Hz); however, this is incorrect since this is not the EARLIEST time trip conditions are met. The second part is correct.

D. Incorrect. The first part is plausible since at time 0121 Reactor trip criteria is met (frequency less than 58.4 Hz); however, this is incorrect since this is not the EARLIEST time trip conditions are met. The second part is plausible since offsite power may still be considered available (based on reports from the Load Dispatcher); however, this is incorrect since Attachment 2 dictates that Offsite Power be considered LOST with respect to EAL classifications if divorced from the grid.

000077 Generator Voltage and Electric Grid Disturbances / 6

077AA2.08; Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Criteria to trip the turbine or reactor

(CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

Importance Rating:	4.3 4.4
Technical Reference:	AOP-028, Section 3.0, Step 2 & Attachment 2 (Caution), Pages 4 & 13, Rev. 38
References to be provided:	None
Learning Objective:	AOP-LP-3.28 Objectives 2.a & 4.b
Question Origin:	New
Comments:	Ask Chief Examiner if asking if event declaration has to be made will meet SRO-only part of question. Phonecon 7/30: Chief Examiner stated that asking if an event declaration is required to be made is at the SRO level of knowledge. K/A is matched since the applicant must evaluate plant conditions and determine when Reactor trip criteria is first met with respect to degrading grid frequency.
Tier/Group:	T1/G1
SRO Justification:	Declaration of an emergency event is an SRO-only task. Task #345001H602 - Determine EAL classifications per AD-EP-ALL-0111, EP-EAL, and EAL Matrix.

79. 2020 NRC SRO 004

Given the following plant conditions:

- A small break LOCA outside Containment occurred
- The crew implemented EOP-E-0, Reactor Trip or Safety Injection, and transitioned to EOP-ECA-1.2, LOCA Outside Containment
- RP will NOT allow personnel entry while RAB radiological conditions are being evaluated

Which ONE of the following completes the statements below?

Rising ___(1)___ is the indication used in EOP-ECA-1.2 to determine that the break is isolated.

After the break is isolated, a transition to ___(2)___ will be made.

A. (1) PRZ level

(2) EOP-E-1, Loss of Reactor or Secondary Coolant

B. (1) PRZ level

(2) EOP-ECA-1.1, Loss Of Emergency Coolant Recirculation

C. (1) RCS pressure

(2) EOP-E-1, Loss of Reactor or Secondary Coolant

D. (1) RCS pressure

(2) EOP-ECA-1.1, Loss Of Emergency Coolant Recirculation

Plausibility and Answer Analysis

Reason answer is correct: RCS pressure is the parameter used in EOP-ECA-1.2 as an indication of the LOCA being isolated. EOP-ECA-1.2 directs a transition to EOP-E-1 after the break is isolated.

A. Incorrect *The first part is plausible since PRZ level is a parameter used to determine the status of RCS inventory; however, this is incorrect as it is not a sufficient indication of break isolation. EOP-ECA-1.2 uses RCS pressure since PRZ level could be affected by head voiding during LOCA conditions. The second part is correct.*

B. Incorrect *The first part is plausible since PRZ level is a parameter used to determine the status of RCS inventory; however, this is incorrect as it is not a sufficient indication of break isolation. EOP-ECA-1.2 uses RCS pressure since PRZ level could be affected by head voiding during LOCA conditions. The second part is plausible since EOP-ECA-1.2 does direct a transition to EOP-ECA-1.1 if the leak outside of containment is NOT isolated.*

C. Correct

D. Incorrect *The first part is correct. The second part is plausible since EOP-ECA-1.2 does direct a transition to EOP-ECA-1.1 if the leak outside of containment is NOT isolated.*

W/E04 LOCA Outside Containment / 3

WE04EA2.2; Ability to determine and interpret the following as they apply to the (LOCA Outside Containment): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

(CFR: 43.5 / 45.13)

Importance Rating:	3.6 4.2
Technical Reference:	EOP-ECA-1.2, Step 6, Page 5, Rev. 1
References to be provided:	None
Learning Objective:	EOP-LP-3.03 Objective 2.d
Question Origin:	Bank
Comments:	K/A is matched as the applicant must interpret plant conditions and determine which procedure will be used to mitigate the event in progress (LOCA outside CNMT) once the break is isolated.
Tier/Group:	T1/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

80. 2020 NRC SRO 005

Given the following plant conditions:

- A large break LOCA occurred
- The RHR system was determined to be not capable of cold leg recirculation and the crew transitioned from EOP-E-1, Loss of Reactor or Secondary Coolant, to EOP-ECA-1.1, Loss of Emergency Coolant Recirculation Capability
- The SGs are currently being depressurized to inject the SI Accumulators

Which ONE of the following completes the statements below?

In accordance with EOP-ECA-1.1, SG depressurization should be controlled to maximize the amount of time the accumulators are available as a makeup source while maintaining ___(1)___.

If emergency coolant recirculation capability is restored during SG depressurization, EOP-ECA-1.1 requires the crew to transition from EOP-ECA-1.1 to ___(2)___.

A. (1) core exit TCs stable or dropping

(2) EOP-E-1, Loss of Reactor or Secondary Coolant, step in effect

B. (1) core exit TCs stable or dropping

(2) EOP-ES-1.3, Transfer to Cold Leg Recirculation, step 1

C. (1) RVLIS at or above its required value

(2) EOP-E-1, Loss of Reactor or Secondary Coolant, step in effect

D. (1) RVLIS at or above its required value

(2) EOP-ES-1.3, Transfer to Cold Leg Recirculation, step 1

Plausibility and Answer Analysis

Reason answer is correct: Per the NOTE in EOP-ECA-1.1 prior to Step 38: "SG depressurization to inject SI accumulators to maximize the amount of time the accumulators are available as a makeup source. Accumulator injection should maintain RVLIS at or just above the required value". The Restoration of Emergency Coolant Recirculation foldout directs a return to procedure and step in effect when emergency coolant recirculation capability is restored.

A. Incorrect. The first part is plausible since core exit TCs are monitored to ensure sufficient SI (or charging) flows; however, this is incorrect since the NOTE specifically addresses monitoring RVLIS indication when injecting from the SI accumulators. The second part is correct.

B. Incorrect. The first part is plausible since core exit TCs are monitored to ensure sufficient SI (or charging) flows; however, this is incorrect since the NOTE specifically addresses monitoring RVLIS indication when injecting from the SI accumulators. The second part is plausible since emergency coolant recirculation will be established using EOP-ES-1.3; however, this is incorrect since the EOP-ECA-1.1 foldout directs a return to EOP-E-1 which will then direct a transition to EOP-ES-1.3.

C. Incorrect. The first part is correct. The second part is plausible since emergency coolant recirculation will be established using EOP-ES-1.3; however, this is incorrect since the EOP-ECA-1.1 foldout directs a return to EOP-E-1 which will then direct a transition to EOP-ES-1.3.

D. Correct.

W/E11 Loss of Emergency Coolant Recirculation / 4

WE11EG2.4.20; Knowledge of the operational implications of EOP warnings, cautions, and notes.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.8 4.3
Technical Reference:	EOP-ECA-1.1 Foldout & NOTE preceding Step 38, Pages 53 & 54, Rev. 4
References to be provided:	None
Learning Objective:	EOP-LP-3.03 Objective 5.e
Question Origin:	New
Comments:	Ask Chief Examiner if intended K/A was WE11 G2.2.20. Phonecon 4/14: Chief Examiner stated that the intended K/A was WE11 G.2.20. Early Submittal K/A is matched since the applicant must demonstrate an understanding of a NOTE in EOP-ECA-1.1 addressing injection of SI accumulators during SG depressurization.
Tier/Group:	T1/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

81. 2020 NRC SRO 006

Given the following plant conditions:

- The unit was operating at 100% power when a Main Steam Line Break occurred
- The Reactor was tripped and Safety Injection actuated
- Main Steam Isolation failed and all MSIVs failed to shut from the MCB
- The crew established a minimum feed flow of 12.5 KPPH to all SGs in accordance with EOP-ECA-2.1, Uncontrolled Depressurization of All Steam Generators

Which ONE of the following completes the statements below?

The basis for maintaining minimum feed flow to each SG is to minimize thermal stresses on the __(1)__.

If any SG pressure begins to rise, a transition from EOP-ECA-2.1 to ___(2)___, Step 1, will be required.

A. (1) SGs when feed flow is eventually raised

(2) EOP-E-2, Faulted Steam Generator Isolation

B. (1) SGs when feed flow is eventually raised

(2) EOP-E-3, Steam Generator Tube Rupture

C. (1) Reactor Vessel due to continued RCS cooldown

(2) EOP-E-2, Faulted Steam Generator Isolation

D. (1) Reactor Vessel due to continued RCS cooldown

(2) EOP-E-3, Steam Generator Tube Rupture

Plausibility and Answer Analysis

Reason answer is correct: If feed flow to a SG is isolated and the SG is allowed to dry out, subsequent re-initiation of feed flow to the SG could create significant thermal stress conditions on SG components. Maintaining a minimum verifiable feed flow to the SG allows the components to remain in a "wet" condition, thereby minimizing any thermal shock effects if feed flow is increased. If any SG pressure begins to rise, then a transition to EOP-E-2 is required per the EOP-ECA-2.1 foldout criteria.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since a rise in SG pressure would be an indication of a SG tube rupture requiring a transition to EOP-E-3; however, this is incorrect as a pressure rise is an indication of an intact SG requiring a transition to EOP-E-2. Instead, a level rise in the SG is used as the foldout criteria for a SG tube rupture.

C. Incorrect. The first part is plausible since this is the basis for lowering feed flow to 12.5 KPPH to each SG per the ECA-2.1 Background Document; however, this is incorrect since the basis for maintaining a minimum feed flow is to minimize thermal stresses on a SG when feed flow is raised. The second part is correct.

D. Incorrect. The first part is plausible since this is the basis for lowering feed flow to 12.5 KPPH to each SG per the ECA-2.1 Background Document; however, this is incorrect since the basis for maintaining a minimum feed flow is to minimize thermal stresses on a SG when feed flow is raised. The second part is plausible since a rise in SG pressure would be an indication of a SG tube rupture requiring a transition to EOP-E-3; however, this is incorrect as a pressure rise is an indication of an intact SG requiring a transition to EOP-E-2. Instead, a level rise in the SG is used as the foldout criteria for a SG tube rupture.

W/E12 Steam Line Rupture - Excessive Heat Transfer / 4

WE12EG2.4.18; Knowledge of the specific bases for EOPs.

(CFR: 41.10 / 43.1 / 45.13)

Importance Rating:	3.3 4.0
Technical Reference:	EOP-ECA-2.1 Foldout, Page 3, Rev. 2 HECA21BG (ECA-2.1 Background), Page 23, Rev. 3
References to be provided:	None
Learning Objective:	EOP-LP-3.09 Objectives 4 & 5
Question Origin:	New
Comments:	Ask Chief Examiner if intended K/A was WE12 EG2.4.18. Phonecon 4/14: Chief Examiner stated that the intended K/A was WE12 EG2.4.18. K/A is matched since the applicant must recall the basis for maintaining minimum feedwater flow to each SG during an uncontrolled SG depressurization event.
Tier/Group:	T1/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

82. 2020 NRC SRO 007

Given the following plant conditions:

- The unit was operating at 100% power when a load rejection occurred

Following the load rejection:

- The OATC reports that Control Rod D-4 is misaligned from the Group D step counter demand position by approximately 20 steps
- ALB-013-7-1, ROD CONTROL URGENT ALARM, is in alarm
- All rods are verified to be above the Rod Insertion Limits

Which ONE of the following completes the statements below?

In accordance with AOP-001, Malfunction of Rod Control and Indication System, Control Rod D-4 ___(1)___ considered trippable.

In accordance with Technical Specification 3.1.3.1, Movable Control Assemblies - Group Height, the bases for reducing Thermal Power is to ___(2)___.

A. (1) is

(2) minimize the effects of a control rod ejection accident

B. (1) is

(2) provide assurance of fuel rod integrity during continued operation

C. (1) is NOT

(2) minimize the effects of a control rod ejection accident

D. (1) is NOT

(2) provide assurance of fuel rod integrity during continued operation

Reason answer is correct: In accordance with AOP-001, to be considered trippable, a control rod must meet ANY of the following 4 criteria:

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

In accordance with the bases for Technical Specification 3.1.3.1, the ACTION statement restrictions on THERMAL POWER provide assurance of fuel rod integrity during continued operation.

A. Incorrect. The first part is correct. The second plausible since one of the reasons for maintaining Control Rod alignment within limits is to ensure the effects of a rod ejection accident are within analyzed limits. For this condition, if the rod ejection occurred on the dropped rod, the reactivity effects would be more pronounced at 100% power than at 75% power; however, this is not the reason the bases requires reducing power to < 75%. Specifically, a rod ejection is the basis for the Power Range Positive Rate trip as it provides protection against rapid flux increases which are characteristic of a rupture of the control rod drive housing.

B. Correct.

C. Incorrect. The first part is plausible since there are only four conditions per AOP-001 where a control rod can be considered trippable; however, this is incorrect since one of the four conditions is present. The second plausible since one of the reasons for maintaining Control Rod alignment within limits is to ensure the effects of a rod ejection accident are within analyzed limits. For this condition, if the rod ejection occurred on the dropped rod, the reactivity effects would be more pronounced at 100% power than at 75% power; however, this is not the reason the bases requires reducing power to < 75%. Specifically, a rod ejection is the basis for the Power Range Positive Rate trip as it provides protection against rapid flux increases which are characteristic of a rupture of the control rod drive housing.

D. Incorrect. The first part is plausible since there are only four conditions per AOP-001 where a control rod can be considered trippable; however, this is incorrect since one of the four conditions is present. The second is correct.

000005 Inoperable/Stuck Control Rod / 1

005AG2.4.31; Knowledge of annunciator alarms, indications, or response procedures.

(CFR: 41.10 / 45.3)

Importance Rating:	4.2 4.1
Technical Reference:	AOP-001, Attachment 5, Page 53, Rev. 54 Technical Specification 3.1.3.1 Bases, Page 3/4 1-4
References to be provided:	None
Learning Objective:	AOP-LP-3.01 Objective 5.b RODCS-ILC Objective 15
Question Origin:	New
Comments:	K/A is matched since applicant must demonstrate an understanding of an annunciator alarm associated with the Rod Control System and how it relates to trippability for a stuck rod.
Tier/Group:	T1/G2
SRO Justification:	10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

83. 2020 NRC SRO 008

Given the following timeline:

- 0000 The unit is in Mode 5
  All shutdown rods are fully withdrawn for testing
- 0001 Source Range Nuclear Instrument N-31 fails LOW
- 0010 Source Range Nuclear Instrument NI-32 fails LOW
  The OATC manually trips the Reactor using MCB Switch #1

Which ONE of the following completes the statements below?

In accordance with the EOP Users Guide, EOP-E-0, Reactor Trip or Safety Injection, __(1)__ required to be entered to confirm the Reactor trip.

A maximum allowable extension of 25% __(2)__ be used when verifying compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.2.

(Reference provided)

A. (1) is

(2) can

B. (1) is

(2) can NOT

C. (1) is NOT

(2) can

D. (1) is NOT

(2) can NOT

Plausibility and Answer Analysis

Reason answer is correct: EOP-E-0 is not required to be entered for a Reactor Trip in Mode 5. The EOP User's Guide identifies EOP-E-0 as only applicable when > 350°F. SR 4.0.2 states that each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval; however, this 25% allowance does not apply when performing a surveillance item to comply with a Technical Specification action statement.

A. Incorrect. The first part is plausible since the stem of the question presents the applicant with a Reactor trip and no other procedure other than EOP-E-0 is written to address a "reactor trip response"; however, this is incorrect since per the EOP Users Guide, EOP-E-0 in only applicable in Modes 1, 2, or 3. The second part is correct.

B. Incorrect. The first part is plausible since the stem of the question presents the applicant with a Reactor trip and no other procedure other than EOP-E-0 is written to address a "reactor trip response"; however, this is incorrect since per the EOP Users Guide, EOP-E-0 in only applicable in Modes 1, 2, or 3. The second part is plausible since SR 4.0.2 allows a 25% extension for Surveillance Requirements; however, this is incorrect since this extension does not apply when performing a surveillance item to comply with a Technical Specification action statement.

C. Incorrect. The first part is correct. The second part is plausible since SR 4.0.2 allows a 25% extension for Surveillance Requirements; however, this is incorrect since this extension does not apply when performing a surveillance item to comply with a Technical Specification action statement.

D. Correct.

000032 Loss of Source Range Nuclear Instrumentation / 7

032AA2.06; Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Confirmation of reactor trip

(CFR: 43.5 / 45.13)

Importance Rating:	3.9 4.1
Technical Reference:	EOP Users Guide, Attachment 1, Page 66, Rev. 51 Technical Specification 3.3.1, Pages 3/4 3-1, 3-2, & 3-7
References to be provided:	Technical Specification 3.3.1
Learning Objective:	EOP-LP-3.19 Objective 1.c NIS-ILC Objective 12
Question Origin:	Bank (Watts Bar)
Comments:	Ask Chief Examiner if second part of question meets SRO-only criteria. Phonecon 7/30: Chief Examiner reviewed question and felt it met the criteria. K/A is matched since the applicant must interpret plant conditions and make a determination as to whether the reactor trip procedure must be used to confirm the trip.
Tier/Group:	T1/G2
SRO Justification:	10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

84. 2020 NRC SRO 009

Given the following plant conditions:
- The unit is operating at 100% power
- The 'A' and 'B' Circulating Water Pumps are operating
- 'C' Circulating Water Pump is under clearance

Subsequently:
- Degrading condenser vacuum is observed
- AOP-012, Partial Loss of Condenser Vacuum, is entered
- CTMP-7-1, COOLING TOWER 1 LEVEL HI/LO, alarms due to low level

Which ONE of the following completes the statements below?

A Reactor trip would be required in accordance with AOP-012 if ___(1)___.

When the Reactor is tripped, the crew will GO TO EOP-E-0, Reactor Trip or Safety Injection, and ___(2)___.

A. (1) ALB-021-5 alarms due to Condenser Pit High Level

(2) exit AOP-012

B. (1) ALB-021-5 alarms due to Condenser Pit High Level

(2) continue to perform the actions of AOP-012 as time allows

C. (1) ONE of the running Circulating Water Pumps trips

(2) exit AOP-012

D. (1) ONE of the running Circulating Water Pumps trips

(2) continue to perform the actions of AOP-012 as time allows

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-012, a Reactor Trip is required if Reactor Power is greater than 10% and BOTH ALB-021-8-5 in in alarm due to Condenser Pit high level and CTMP-7-1, Cooling Tower 1 Level Hi/Lo, is in alarm due to low level. These are indications of a complete Circulating Water System expansion joint failure. The crew is required to immediately go to EOP-E-0 and perform AOP-012 substeps 15.c -.f as time allows.

A. Incorrect. The first part is correct. The second part is plausible as other AOPs (e.g. AOP-001) are exited upon entry into EOP-E-0; however, this is incorrect since AOP-012 has additional actions that must be completed as time permits.

B. Correct.

C. Incorrect. The first part is plausible since CWP status is addressed in AOP-012 and an expansion joint failure would challenge pump operation; however, this is incorrect since AOP-012 only directs a Reactor trip if NO CWPs are running. The second part is correct.

D. Incorrect. The first part is plausible since CWP status is addressed in AOP-012 and an expansion joint failure would challenge pump operation; however, this is incorrect since AOP-012 only directs a Reactor trip if NO CWPs are running. The second part is plausible as other AOPs (e.g. AOP-001) are exited upon entry into EOP-E-0; however, this is incorrect since AOP-012 has additional actions that must be completed as time permit.

000051 Loss of Condenser Vacuum / 4

051AG2.1.7; Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

(CFR: 41.5 / 43.5 / 45.12 / 45.13)

Importance Rating:	4.4 4.7
Technical Reference:	AOP-012, Section 3.0, Steps 14 & 15, Pages 8 & 9, Rev. 32
References to be provided:	None
Learning Objective:	AOP-LP-3.12 Objectives 2.a & 3
Question Origin:	Bank
Comments:	K/A is matched since the applicant must evaluate plant parameters to make an operational decision related to tripping the reactor with degrading vacuum conditions.
Tier/Group:	T1/G2
SRO Justification:	10 CFR Part 55 Content - 43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

85. 2020 NRC SRO 010

Given the following plant conditions:

- The crew is implementing EOP-ES-1.2, Post LOCA Cooldown and Depressurization
- AFW flow to the SGs has been secured to maintain current levels
- Charging flow is 150 gpm
- Pressurizer level is 8% and lowering
- RCS subcooling is 23°F and lowering
- 'C' SG level is rising steadily

Which ONE of the following completes the statement below?

The required operator action will be to ___(1)___ and transition to ___(2)___.

A. (1) actuate Safety Injection

(2) EOP-E-3, Steam Generator Tube Rupture

B. (1) actuate Safety Injection

(2) EOP-ECA-3.1, SGTR with Loss of Reactor Coolant: Subcooled Recovery

C. (1) manually align flow through the BIT

(2) EOP-E-3, Steam Generator Tube Rupture

D. (1) manually align flow through the BIT

(2) EOP-ECA-3.1, SGTR with Loss of Reactor Coolant: Subcooled Recovery

Plausibility and Answer Analysis

Reason answer is correct: With PRZ level less than 10%, SI Reinitiation Criteria is met. EOP-ES-1.2 Foldout directs manually aligning flow through the BIT by shutting the charging line isolation valves and opening the BIT outlet valves. With steadily rising level in the 'C' SG, E-3 Transition Criteria is met.

A. Incorrect. The first part is plausible since other EOPs have SI Actuation Criteria (e.g. EOP-ES-0.1); however, this is incorrect since EOP-ES-1.2 has SI Reinitiation Criteria which requires actions to manually re-align flow through the BIT. The second part is correct.

B. Incorrect. The first part is plausible since other EOPs have SI Actuation Criteria (e.g. EOP-ES-0.1); however, this is incorrect since EOP-ES-1.2 has SI Reinitiation Criteria which requires actions to manually re-align flow through the BIT. The second part is plausible since EOP-ECA-3.1 addresses a SG tube rupture with a loss of RCS inventory (conditions which do exist in the question stem); however, this is incorrect as EOP-ES-1.2 has E-3 Transition Criteria, not EOP-ECA-3.1 Transition Criteria. EOP-ECA-3.1 is only entered from EOP-E-3, EOP-ES-3.1, EOP-ES-3.2, or EOP-ES-3.3.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since EOP-ECA-3.1 addresses a SG tube rupture with a loss of RCS inventory (conditions which do exist in the question stem); however, this is incorrect as EOP-ES-1.2 has E-3 Transition Criteria, not EOP-ECA-3.1 Transition Criteria. EOP-ECA-3.1 is only entered from EOP-E-3, EOP-ES-3.1, EOP-ES-3.2, or EOP-ES-3.3.

W/E03 LOCA Cooldown—Depressurization / 4

WE03EA2.1; Ability to determine and interpret the following as they apply to the (LOCA Cooldown and Depressurization): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

(CFR: 43.5 / 45.13)

Importance Rating:	3.4 4.2
Technical Reference:	EOP-ES-1.2 Foldout, Page 3, Rev. 4
References to be provided:	None
Learning Objective:	EOP-LP-3.05 Objective 4
Question Origin:	Bank
Comments:	K/A is matched since the applicant must interpret plant conditions and determine the required transition to the appropriate procedure to address RCS leakage.
Tier/Group:	T1/G2
SRO Justification:	10 CFR Part 55 Content - 43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

86. 2020 NRC SRO 011/NEW/C/A//TS 3.4.6.2, TS 3.0.4/TS 3.4.6.2, SFD-1300//011 A2.09/

Given the following plant conditions:

- The unit is in Mode 4
- A plant heatup is in progess in accordance with GP-002, Normal Plant Heatup from Solid to Hot Subcritical, Mode 5 to Mode 3

Subsequently:

- RCS leakage develops
- Inspection reveals that 1RC-38, Normal Letdown Isol. Vlv., has developed through-wall leakage (upstream side of valve body)

Which ONE of the following completes the statements below in accordance with Technical Specifications?

The RCS leakage will be classified as __(1)__ LEAKAGE.

Entry into Mode 3 __(2)__ allowed.

(Reference provided)

A. (1) UNIDENTIFIED

(2) is

B. (1) UNIDENTIFIED

(2) is NOT

C. (1) PRESSURE BOUNDARY

(2) is

D. (1) PRESSURE BOUNDARY

(2) is NOT

Plausibility and Answer Analysis

Reason answer is correct: Through-wall leakage would be considered PRESSURE BOUNDARY LEAKAGE. No PRESSURE BOUNDARY LEAKAGE is allowed per Technical Specification 3.4.6.2. ACTION b allows 4 hours to reduce the leakage rate within limits or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Since the ACTION statement does not permit continued operation of the facility for an unlimited period of time, LCO 3.0.4 will NOT allow entry into Mode 3.

A. Incorrect. The first part is plausible since this type of leakage would most likely be classified initially as UNIDENTIFIED LEAKAGE until an inspection was completed (quote in TS bases); however, this is incorrect since the inspection has already identified through-wall leakage. The second part is plausible since Technical Specification 3.4.6.2 allows 4 hours to reduce the leakage rate and the applicant may interpret this as an allowance for mode ascension; however, this is incorrect since Mode 3 entry is prohibited by LCO 3.0.4.A. The second part is plausible since Technical Specification 3.4.6.2 allows 4 hours to reduce the leakage rate and the applicant may interpret this as an allowance for mode ascension; however, this is incorrect since Mode 3 entry is prohibited by LCO 3.0.4.

B. Incorrect. The first part is plausible since this type of leakage would most likely be classified initially as UNIDENTIFIED LEAKAGE until an inspection was completed (quote in TS bases); however, this is incorrect since the inspection has already identified through-wall leakage. The second part is plausible since Technical Specification 3.4.6.2 allows 4 hours to reduce the leakage rate and the applicant may interpret this as an allowance for mode ascension; however, this is incorrect since Mode 3 entry is prohibited by LCO 3.0.4.A. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since Technical Specification 3.4.6.2 allows 4 hours to reduce the leakage rate and the applicant may interpret this as an allowance for mode ascension; however, this is incorrect since Mode 3 entry is prohibited by LCO 3.0.4.

D. Correct.

004 Chemical and Volume Control / 1/2

004A2.03; Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Boundary isolation valve leak

(CFR: 41.5/ 43/5 / 45/3 / 45/5)

Importance Rating: 3.6 4.2

Technical Reference:
Technical Specification, Definitions, Page 1-4
Technical Specification 3.4.6.2, Page 3/4 4-23
Technical Specifications, LCO 3.0.4, Page 3/4 0-1
5-S-1300, Simplified Flow Diagram, Reactor Control System, Rev. 23

References to be provided:
Technical Specification 3.4.6.2
5-S-1300, Simplified Flow Diagram, Reactor Control System

Learning Objective:
CVCS-ILC Objective 5.a
TS-LP-3.0 Objectives 1.f & 3.a

Question Origin: New

Comments: K/A is matched since the applicant must use plant conditions to classify the RCS leakage and use Technical Specifications to determine the impact on continued plant operations.

Tier/Group: T2/G2

SRO Justification: 10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

87. 2020 NRC SRO 012

Given the following plant conditions:
- The unit is operating at 100% power

Subsequently:
- An inadvertent Safety Injection occurs
- RCS pressure is 2100 psig and rising

Which ONE of the following completes the statements below?

For the conditions above, the CSIP alternate mini-flow valves will be __(1)__.

Safety Injection will be terminated using __(2)__.

A. (1) SHUT

(2) EOP-ES-1.1, SI Termination

B. (1) SHUT

(2) EOP-E-0, Reactor Trip or Safety Injection

C. (1) OPEN

(2) EOP-ES-1.1, SI Termination

D. (1) OPEN

(2) EOP-E-0 Reactor Trip or Safety Injection

Plausibility and Answer Analysis

Reason answer is correct: Above 2000 psig with a Safety Injection signal present, the CSIP alternate mini-flow valves will open to provide mini-flow protection for the CSIPs. This mini-flow will be directed to the RWST. With an inadvertent Safety Injection, SI Termination Criteria will be met and Safety Injection flow will be terminated in EOP-E-0.

A. Incorrect. The first part is plausible since this choice would be correct if RCS pressure was less than 2000 psig; however, this is incorrect since the RCS pressure is 2100 psig. Also plausible since this setpoint used to be 2200 psig prior to implementation of an EC. The second part is plausible since the purpose of EOP-ES-1.1 is to terminate Safety Injection; however, this is incorrect since EOP-E-0 provides direction for terminating Safety Injection for an inadvertent actuation. Also plausible since this would be the required procedural transition for termination of SI.

B. Incorrect. The first part is plausible since this choice would be correct if RCS pressure was less than 2000 psig; however, this is incorrect since the RCS pressure is 2100 psig. Also plausible since this setpoint used to be 2200 psig prior to implementation of an EC. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since the purpose of EOP-ES-1.1 is to terminate Safety Injection; however, this is incorrect since EOP-E-0 provides direction for terminating Safety Injection for an inadvertent actuation. Also plausible since this would be the required procedural transition for termination of SI.

D. Correct.

006 Emergency Core Cooling / 2/3

006A2.13; Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent SIS actuation

(CFR: 41.5 / 45.5)

Importance Rating:	3.9 4.2
Technical Reference:	EOP-E-0, Foldout & Step 42, Pages 37 & 38, Rev. 15
References to be provided:	None
Learning Objective:	CVCS-ILC Objective 5.e EOP-LP-3.22 Objectives 1 & 4
Question Origin:	New
Comments:	Early Submittal K/A is matched since applicant must predict operation of the CSIP alternate miniflow valves with a Safety Injection signal present and evaluate plant conditions to determine which procedure will be used to mitigate the consequences (terminate Safety Injection).
Tier/Group:	T2/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

88. 2020 NRC SRO 013

Which ONE of the following completes the statement below regarding the Pressurizer Power-Operated Relief Valves (PORVs)?

In accordance with Technical Specification bases, a SAFETY-RELATED function of the PORVs in Modes 1, 2, and 3 is to _______.

A. provide automatic pressure control to minimize challenges to the safety valves

B. prevent the RCS from being pressurized above its Safety Limit of 2735 psig

C. provide manual RCS pressure control for mitigation of a SGTR accident

D. prevent RCS overpressurization from occuring during a Turbine Trip

Reason answer is correct. Tech 3.4.4 requires all PORVs and block valves to be OPERABLE in Modes 1, 2, and 3. Providing an RCS pressure boundary and manual pressure control for mitigation of accidents are the safety-related functions of the PORVs. The automatic RCS pressure control function of the PORVs is not a safety-related function in Modes 1, 2, and 3.

A. Incorrect. Plausible since providing automatic RCS pressure control is a function of the PORVs, but not a safety-related function.

B. Incorrect. Plausible since this is the bases for the pressurizer Safety Valves, not the PORVs.

C. Correct.

D. Incorrect. Plausible since this is the bases for the maximum pressurizer water level limit, not the PORVs.

010 Pressurizer Pressure Control / 3

010G2.2.25; Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

(CFR: 41.5 / 41.7 / 43.2)

Importance Rating:	3.2 4.2
Technical Reference:	Technical Specification Bases (3/4.4.4), Page B 3/4 4-2
References to be provided:	None
Learning Objective:	PRZPC-ILC Objective 12
Question Origin:	New
Comments:	K/A is match since the applicant must recall the bases in Technical Specifications regarding safety-related function of the PRZ PORVs in Modes 1, 2, and 3.
Tier/Group:	T2/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

89. 2020 NRC SRO 014

Given the following plant conditions:

- The unit is operating at 100% power
- At 1000 Emergency Service Water Pumps 1A-SA and 1B-SB are determined to be NOT Operable but Available due to a common cause

Subsequently:

- At 1030 a downpower to shutdown the unit is initiated

Which ONE of the following completes the statements below?

In accordance with Technical Specification 3.7.4, Plant Systems - Emergency Service Water System, the bases for the Limiting Condition of Operation is to ensure that sufficient cooling capacity is available for continued operation of safety related equipment during __(1)__conditions.

The LATEST time the unit is required to be in Hot Standby is __(2)__.

(Reference provided)

A. (1) normal AND accident

(2) 1630

B. (1) normal AND accident

(2) 1700

C. (1) ONLY accident

(2) 1630

D. (1) ONLY accident

(2) 1700

Plausibility and Answer Analysis

Reason answer is correct: The bases for Technical Specification 3.7.4 states the OPERABILITY of the Emergency Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system. assuming a single failure is consistent with the assumptions used in the safety analyses. Because both Emergency Service Water Pumps are inoperable concurrently and Technical Specification 3.7.4 does not have an action for this condition LCO 3.0.3 must be applied. LCO 3.0.3 allows 1 hour to initiate action to place the unit in a mode in which the specification does not apply and 6 hours to be in Hot Standby. Since LCO 3.7.4 was determined to not be met at 1000 the unit must be in Hot Standby not later than 1700 (1 hour to initiate action plus 6 hours to be in Hot Standby).

A. Incorrect. The first part is correct. The second part is plausible since the unit shutdown commenced at 1030 the candidate may have the misconception that the unit must be placed in Hot Standby 6 hours from the time the shutdown was initiated; however this is incorrect because LCO 3.0.3 allows 1 hour to initiate a shutdown for a total of 7 hours from the time of discovery.

B. Correct.

C. Incorrect. The first part is plausible because the ESW system does provide cooling capacity for accident conditions to safety related equipment, however the normal system line up of the ESW header which is supplied by the NSW system during normal operation and components on the ESW header such as the CSIPs and CCW pumps are required to be cooled during these normal operating conditions. The second part is plausible since the unit shutdown commenced at 1030 the candidate may have the misconception that the unit must be placed in Hot Standby 6 hours from the time the shutdown was initiated; however this is incorrect because LCO 3.0.3 allows 1 hour to initiate a shutdown for a total of 7 hours from the time of discovery.

D. Incorrect. The first part is plausible because the ESW system does provide cooling capacity for accident conditions to safety related equipment, however the normal system line up of the ESW header which is supplied by the NSW system during normal operation and components on the ESW header such as the CSIPs and CCW pumps are required to be cooled during these normal operating conditions. The second part is correct.

076 Service Water / 4

076G2.2.40; Ability to apply Technical Specifications for a system.

(CFR: 41.10 / 43.2 / 43.5 / 45.3)

Importance Rating: 3.4 4.7

Technical Reference: Technical Specification 3.7.4, Bases, Page B 3/4 7-3
Technical Specification 3.7.4, Page 3/4 7-12
Technical Specification 3.0.3, Page 3/4 0-1

References to be provided: Technical Specification 3.7.4

Learning Objective: SWS-ILC Objectives 2.c & 6.a

Question Origin: Early Submittal

K/A is match since applicant must apply LCO 3.0.3 to determine the latest time the unit is required to be in Hot Standby (Mode 3).

Comments: None

Tier/Group: T2/G1

SRO Justification: 10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

90. 2020 NRC SRO 015

Given the following plant conditions:

- OST-1029, Containment Penetration Outside Isolation Valve Verification, Monthly Interval, Modes 1-6, was performed satisfactorily on September 30 at midnight
- On November 2 at 0800, it was discovered that OST-1029 was not performed on October 30 as scheduled

Which ONE of the following identifies the LATEST date this surveillance item must be completed satisfactory to be within its specified surveillance interval?

(Monthly frequency is 31 days)

A. November 3

B. November 7

C. November 10

D. December 2

Plausibility and Answer Analysis

Reason answer is correct: TS 4.0.2 states that each Surveillance Requirement (SR) shall be performed within its specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified interval. In the case of OST-1029, the 25% grace period would be 7.75 days (31 days x 0.25). Since OST-1029 was last performed satisfactory on at midnight on September 30, the LATEST time in which the SR could be completed satisfactory and the specified frequency met would be 1800 on November 7.

A. Incorrect. Plausible since if it's discovered that a surveillance was not performed within its specified surveillance interval, TS 4.0.3 allows, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, to allow performance of the surveillance without declaring the LCO not met. This choice is 24 hours from the time of discovery; however, this is incorrect since the SR is still within its maximum allowable extension of 25% of the surveillance interval and TS 4.0.3 is not applicable at this time.

B. Correct.

C. Incorrect. Plausible since choice is 7.75 days (grace period) from the time of discovery; however, this is incorrect since the grace period is added to the surveillance interval and applied to the time the last time the surveillance was performed satisfactory.

D. Incorrect. Plausible since if it's discovered that a surveillance was not performed within its specified surveillance interval, TS 4.0.3 allows, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, to allow performance of the surveillance without declaring the LCO not met. This choice is 31 days from the time of discovery; however, this is incorrect since the SR is still within its maximum allowable extension of 25% of the surveillance interval and TS 4.0.3 is not applicable at this time.

103 Containment / 5

103G2.2.12; Knowledge of surveillance procedures.

(CFR: 41.10 / 45.13)

Importance Rating: 3.7 4.1

Technical Reference: Technical Specifications, Surveillance Requirements (4.0.2), Page 3/4 0-2

References to be provided: None

Learning Objective: TS-LP-3.0 Objective 3.b

Question Origin: New

Comments: K/A is matched since the applicant must apply SR 4.0.2 to a scenario where a Containment surveillance was not performed as scheduled.

Tier/Group: T2/G1

SRO Justification: 10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

91. 2020 NRC SRO 016

Given the following plant conditions:

- A load reduction was initiated in accordance with GP-006, Normal Plant Shutdown from Power Operation to Hot Standby (Mode 1 to Mode 3)

The following indications are observed as load is reduced:

Time	Power	Control Bank C	Control Bank D
0600	75%	228 steps	158 steps
0630	70%	228 steps	128 steps
0700	65%	228 steps	113 steps
0730	60%	224 steps	98 steps
0800	55%	216 steps	88 steps

Which ONE of the following completes the statement below?

The EARLIEST time that the action statement is required to be entered for Technical Specification 3.1.3.6, Control Rod Insertion Limits, is ___(1)___ AND the action required to satisfy the LCO is to restore ___(2)___.

(Reference provided)

A. (1) 0630

(2) Thermal Power to less than 67% by no later than 1030

B. (1) 0630

(2) control banks to within the insertion limit specified by 0830

C. (1) 0730

(2) Thermal Power to less than 51% by no later than 1130

D. (1) 0730

(2) control banks to within the insertion limit specified by 0930

Reason answer is correct: OP-104 P&L #2 states that whenever the Reactor is critical and except for approved physics testing, control rods must be above the low-low insertion limit. The Rod Insertion Technical Specification 3.1.3.6 limit is a linear curve that increases the limit 1.86 steps for each percent power. With the Reactor at 70% power, the rod insertion limits for control bank C and D are 225 and 130 steps respectively. The control rods indicate they are at 225 on control bank C and 113 steps on control bank D; therefore, control bank D is clearly below the Technical Specification 3.1.3.6 limits at time 0630 as indicated. At that time, the applicant must apply action statement a or b within 2 hours to either restore rods to above the insertion limits for action a or reduce thermal power below the required fraction of rated thermal power for the rod height at that time for action b.

A. Incorrect. The first part is correct. The second part is plausible since the normal progression of LCO action statements is to perform the first action, i.e. action a, then if not completed perform the second action, i.e. action b, within the following time frame after the elapse of the first action; however, this is incorrect since the LCO allows the candidate to perform either action statement to restore compliance with the LCO within the 2 hour timeframe, therefore the cumulative time of 4 hours is improperly applying the Technical Specification 3.1.3.6 LCO.

B. Correct.

C. Incorrect. The first part is plausible since it is correct at that current time as CB C is below the curve limit at this time; however, this is incorrect because CB D was out of spec at an earlier time, therefore it is not the earliest time. The second part is plausible since the normal progression of LCO action statements is to perform the first action, i.e. action a, then if not completed perform the second action, i.e. action b, within the following time frame after the elapse of the first action; however, this is incorrect since the LCO allows the candidate to perform either action statement to restore compliance with the LCO within the 2 hour timeframe, therefore the cumulative time of 4 hours is improperly applying the Technical Specification 3.1.3.6 LCO.

D. Incorrect. The first part is plausible since it is correct at that current time as CB C is below the curve limit at this time; however, this is incorrect because CB D was out of spec at an earlier time, therefore it is not the earliest time. The second part is plausible since it is the correct action based on the current time; however, this is incorrect since it is not the earliest time.

001 Control Rod Drive / 1

001G2.1.32; Ability to explain and apply system limits and precautions.

(CFR: 41.10 / 43.2 / 45.12)

Importance Rating: 3.8 4.0

Technical Reference: OP-145, Section 4.0, P&L #2, Page 5, Rev. 45
Rod Control Manual, Section 2.2, Control Rod Insertion Limits, Rev. 0
Technical Specification 3.1.3.6, Page 3/4 1-21

References to be provided: Rod Control Manual, Section 2.2, Control Rod Insertion Limits
Technical Specification 3.1.3.6

Learning Objective: RODCS-ILC Objectives 12.d & 14

Question Origin: Previous (2016 NRC SRO 16)

Comments: K/A is matched since the applicant must apply a Rod Control System precaution and limitation regarding insertion limits when operating at power.

Tier/Group: T2/G2

SRO Justification: 10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

92. 2020 NRC SRO 017

Given the following plant conditions:
- The unit is operating at 100% power

At 0930, Pressurizer (PRZ) level indications are as follows:
- LI-459 is 88% and rising
- LI-460 is 56% and lowering
- LI-461 is 55% and lowering

Which ONE of the following completes the statements below?

At 0930, ___(1)___ will be in alarm due to the level transmitter failure.

At 1020, the inoperable channel is placed into bypass for testing of other channels.

In accordance with Technical Specification 3.3.1, Instrumentation - Reactor Trip System Instrumentation, the inoperable channel may be bypassed for surveillance testing of the other channels until no LATER than ___(2)___.

A. (1) ALB-009-4-1, PRESSURIZER HIGH LEVEL

(2) 1420

B. (1) ALB-009-4-1, PRESSURIZER HIGH LEVEL

(2) 1620

C. (1) ALB-009-2-1, PRZ CONT HIGH LEVEL DEVIATION AND HEATERS ON

(2) 1420

D. (1) ALB-009-2-1, PRZ CONT HIGH LEVEL DEVIATION AND HEATERS ON

(2) 1620

Plausibility and Answer Analysis

Reason answer is correct: Based on the indications given, the controlling channel LT-459 is failing high. Once LI-459 is 5% above program level (60%), ALB-009-2-1, PRZ CONT HIGH LEVEL DEVIATION AND HEATERS ON, will alarm and the PRZ backup heaters will energize. Technical Specification 3.3.1 ACTION 6 allows the inoperable PRZ level channel to be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

A. Incorrect. The first part is plausible since this alarm would be received if the non-controlling channel LT-460 was failing high; however, this is incorrect since the controlling channel LT-459 is failing high which causes another ALB-009 alarm to be received. The second part is correct.

B. Incorrect. The first part is plausible since this alarm would be received if the non-controlling channel LT-460 was failing high; however, this is incorrect since the controlling channel LT-459 is failing high which causes another ALB-009 alarm to be received. The second part is plausible since ACTION 6 also requires the inoperable channel to be a tripped condition within 6 hours; however, this is incorrect since only 4 hours is allowed to bypass the inoperable channel for surveillance testing.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since ACTION 6 also requires the inoperable channel to be a tripped condition within 6 hours; however, this is incorrect since only 4 hours is allowed to bypass the inoperable channel for surveillance testing.

011 Pressurizer Level Control / 2

011A2.09; Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High ambient reflux boiling temperature effect or indicated PZR level

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	2.9 3.5
Technical Reference:	APP-ALB-009, Window 2-1, Page 8, Rev. 18 Technical Specification 3.3.1, Pages 3/4 3-1, 3-2, & 3-7
References to be provided:	None
Learning Objective:	PRZLC-ILC Objectives 6, 9.c & 11
Question Origin:	Modified (2018 NRC SRO 92)
Comments:	K/A is matched since the applicant must predict the impact of a failed PRZ level transmitter (alarm) and then use Technical Specifications to determine the requirements for bypassing the failed channel for testing of another channel.
Tier/Group:	T2/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

93. 2020 NRC SRO 018

At time 0030, the following plant conditions exist:

- The unit is operating at 75% power
- Both Control Bank 'D' step counters indicate 182 steps
- Control Rod H-14 indicates 168 steps on DRPI
- All other Control Bank 'D' rods indicate 180 steps on DRPI
- I&C estimates 8 hours to repair to faulty indicator

Which ONE of the following completes the statements below?

ALB-013-8-5, COMPUTER ALARM ROD DEV/SEQ NIS PWR RANGE TILTS, __(1)__ be in alarm due to its Rod to Bank Deviation input.

In accordance with Technical Specification 3.1.3.1, Movable Control Assemblies - Group height, the LATEST time the High Neutron Flux Setpoint is required to be reduced to less than or 85% of Rated Thermal Power is __(2)__.

(Reference provided)

A. (1) will

(2) 0430

B. (1) will

(2) 0630

C. (1) will NOT

(1) 0430

D. (1) will NOT

(2) 0630

Reason answer is correct:

In accordance with APP-ALB-013, Rod to Bank Deviation:

* Individual rod position does not match average rod position within group by greater than 12 steps

* DRPI rod bank position input disagrees with demand pulse inputs from rod control

A 16 step difference exists between the stuck rod and the group counters so the alarm will be in due to its deviation input.

In accordance with Technical Specification 3.1.3.1, Movable Control Assemblies - Group Height, POWER OPERATION may continue provided that within 1 hour the rod is declared inoperable (0130), THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour (0230), and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER (0630).

A. Incorrect. The first part is correct. The second part is plausible since Technical Specification 3.1.3.1 ACTION d allows 4 hours to reduce the High Neutron Flux Trip Setpoint to less than or equal to 85% of RATED THERMAL POWER and 0430 is 4 hours from time 0030; however, this is incorrect since 1 hour is allowed first to declared the rod inoperable and another hour is allowed to reduce power to less or equal to 75%. Also, plausible since the initial power level was 75% and the applicant may believe the first hour in Action d.3.d) does not apply.

B. Correct.

C. Incorrect. The first part is plausible since only a 12 step difference exists between DRPI for the affected rod and DRPI for all other rods in the group; however, this is incorrect since a 14 step difference exists between the affected rod [DRPI] and demand [step counters]. The second part is plausible since Technical Specification 3.1.3.1 ACTION d allows 4 hours to reduce the High Neutron Flux Trip Setpoint to less than or equal to 85% of RATED THERMAL POWER and 0430 is 4 hours from time 0030; however, this is incorrect since 1 hour is allowed first to declared the rod inoperable and another hour is allowed to reduce power to less or equal to 75%. Also, plausible since the initial power level was 75% and the applicant may believe the first hour in Action d.3.d) does not apply.

D. Incorrect. The first part is plausible since only a 12 step difference exists between DRPI for the affected rod and DRPI for all other rods in the group; however, this is incorrect since a 14 step difference exists between the affected rod [DRPI] and demand [step counters]. The second part is correct.

014 Rod Position Indication / 1

014A2.04; Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Misaligned rod

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.4 3.9
Technical Reference:	APP-ALB-013, Window 8-5, Pages 36 & 37, Rev. 38 Technical Specification 3.1.3.1, Pages 3/4 1-14 & 1-15
References to be provided:	Technical Specification 3.1.3.1
Learning Objective:	RPI-ILC Objective 4.b RODCS-ILC Objective 12.a
Question Origin:	New
Comments:	K/A is matched since the applicant predict receipt of an alarm associated with a misaligned rod then use Technical Specifications to mitigate the event.
Tier/Group:	T2/G2
SRO Justification:	10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases.

94. 2020 NRC SRO 019

Which ONE of the following completes the statement below regarding the refueling process in accordance with AD-NS-ALL-1001, Conduct of Refueling?

With fuel movement in progress, bypassing of fuel handling equipment interlocks which are **NOT** specified in approved procedures shall require permission of the Refueling SRO and concurrence of the _______ .

A. Shift Manager

B. Reactor Engineer

C. Reactor Services Supervisor

D. Refueling Equipment Engineer

Reason answer is correct: Per AD-NS-ALL-1001, Conduct of Refueling, Section 5.2.4, bypassing of fuel handling equipment interlocks which are **NOT** specified in approved procedures shall require permission of the Refueling SRO and concurrence of the Shift Manager.

A. Correct.

B. Incorrect. Plausible since the Reactor Engineer provides instructions and approved move sheets to fuel handlers for alternate moves after obtaining approval from the Refueling SRO; however, this is incorrect as the Shift Manager must concur with bypassing fuel handling equipment interlocks.

*C. Incorrect. Plausible since Reactor Services Supervision approves bypass of refueling equipment interlocks when **NO** movement of fuel assemblies, fuel components, or irradiated components and bypass is not being performed per approved procedures; however, this is incorrect since fuel movement is in progress. Refueling SRO permission is required when fuel movement is in progress.*

D. Incorrect. Plausible since the Refueling Equipment Engineer assists in solving fuel handling related problems; however, this is incorrect as the Shift Manager must concur with bypassing fuel handling equipment interlocks.

2.1 Conduct of Operations

G2.1.41; Knowledge of the refueling process.

(CFR: 41.2 / 41.10 / 43.6 / 45.13)

Importance Rating:	2.8 3.7
Technical Reference:	AD-NS-ALL-1001, Section 5.2.4, Page 21, Rev. 7
References to be provided:	None
Learning Objective:	PP-LP-2.08 Objectives 2.d & 2.e FHS-ILC Objective 6
Question Origin:	New
Comments:	Early Submittal K/A is matched since the applicant must recall the permissions required to bypass fuel handling equipment interlocks.
Tier/Group:	T3
SRO Justification:	10 CFR Part 55 Content - 43(b)(7): Fuel-Handling Facilities and Procedures

95. 2020 NRC SRO 020

Given the following:
- A clearance is ready for approval
- The clearance uses single valve isolation

System conditions are as follows:
- Pressure is 450 psig
- Temperature is 175°F

Which ONE of the following completes the statement below regarding the approval process for this clearance?

.

The clearance can be approved _______.

A. as written since this is a low energy system

B. provided the clearance is designated 'Exceptional'

C. ONLY after double valve isolation has been obtained

D. ONLY after system conditions have been established to allow single valve isolation

Reason answer is correct: In accordance with AD-OP-ALL-0200, double valve isolation shall be provided when available when system has greater than 500 psid across boundary valves or fluids above 200°F. With both parameters less than these values, single valve isolation can be used.

A. Correct.

B. Incorrect. Plausible since clearances which lack double valve isolation when required can be designated 'Exceptional' with SM approval; however, this is incorrect since system conditions do not require double valve isolation.

C. Incorrect. Plausibility since Duke Energy defines Hazardous Energy as 60 psig pressure and 120°F temperature for the purposes of clearance application and system conditions provided are above these values. As such, the applicant would conclude double valve isolation required. Also plausible the applicant may have a misconception that all clearances must use double valve isolation regardless of system conditions.

D. Incorrect. Plausibility since Duke Energy defines Hazardous Energy as 60 psig pressure and 120°F temperature for the purposes of clearance application and system conditions provided are above these values. As such, the applicant may conclude that system conditions will need to be established (lower pressure and temperature) to allow use of single valve isolation.

2.2 Equipment Control

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)

Importance Rating:	3.9 4.3
Technical Reference:	AD-OP-ALL-0200, Section 5.8.2 & Attachment 5, Pages 42 & 123, Rev. 20
References to be provided:	None
Learning Objective:	PP-LP-3.20 Objective 4.c
Question Origin:	New
Comments:	K/A is matched since the applicant demonstrate the ability to use configuration control documentation (clearance) to determine if isolations are adequate to support system maintenance.
Tier/Group:	T3
SRO Justification:	Clearance approval is an SRO-only task. Task # 342003H302 - Approve Removal of Plant Equipment From Operation

96. 2020 NRC SRO 021/NEW/FUNDAMENTAL//AD-WC-ALL-0420/NONE//G2.2.18/
Which ONE of the following completes the statement below regarding maintenance activities during a refueling outage in accordance with AD-WC-ALL-0420, Shutdown Risk Management?

REDUCED INVENTORY is a plant condition in which fuel is in the reactor vessel and reactor vessel inventory level is lowered to less than ___(1)___ inches below the reactor vessel flange.

The ___(2)___ is reponsible for confirming organizational readiness for scheduled activities prior to commencing a drain of the reactor coolant system to a reduced inventory condition.

A. (1) 12

(2) Shift Manager

B. (1) 12

(2) Shift Outage Manager

C. (1) 36

(2) Shift Manager

D. (1) 36

(2) Shift Outage Manager

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AD-WC-ALL-0420, Shutdown Risk Management, reduced inventory is a plant condition in which fuel is in the reactor vessel and reactor vessel inventory level is lower than 36 inches below the reactor vessel flange. The Shift Manager confirms organizational readiness prior to entering Lowered or Reduced Inventory conditions.

A. Incorrect. The first part is plausible since water level must be at least 12 inches below the Reactor Vessel Flange before the Reactor Vessel Head can be detensioned and removed (GP-008); however, this is not considered a reduced inventory condition. The second part is correct.

B. Incorrect. The first part is plausible since water level must be at least 12 inches below the Reactor Vessel Flange before the Reactor Vessel Head can be detensioned and removed (GP-008); however, this is not considered a reduced inventory condition. The second part is plausible since the Shift Outage Manager has responsibilities per AD-WC-ALL-0420 including communicating the status of the Key Safety Functions as well as communicating information for emergent activities to the Shift Manager and maintenance crew supervisors.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since the Shift Outage Manager has responsibilities per AD-WC-ALL-0420 including communicating the status of the Key Safety Functions as well as communicating information for emergent activities to the Shift Manager and maintenance crew supervisors.

2.2 Equipment Control

G2.2.18; Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	2.6 3.9
Technical Reference:	AD-WC-ALL-0420, Sections 3.0 & 4.0, Pages 7 & 12, Rev. 6
References to be provided:	None
Learning Objective:	PP-LP-2.04 Objective 5.b
Question Origin:	New
Comments:	K/A is matched since applicant must recall who is responsible for coordinating maintenance activities when shutdown.
Tier/Group:	T3
SRO Justification:	Managing work activities during shutdown conditions is an SRO-only task. Task #342027H202 - Coordinate Maintenance Activities

97. 2020 NRC SRO 022

Given the following:

- An employee was injured and contaminated during a fuel handling accident
- The employee was transported offsite for treatment before he was de-contaminated
- Duke Energy is planning a news release for this event

Which ONE of the following completes the statements below?

In accordance with AOP-013, Fuel Handling Accident, __(1)__ is the primary radiological concern for fuel off-loaded more than 6 months ago because it will NOT be detected by personal dosimetry or area radiation monitors.

In accordance with AD-LS-ALL-0006, Notification/Reportability Evaluation, the EARLIEST required NRC notification of this event is within __(2)__ hours.

(Reference provided)

A. (1) Krypton-85

(2) 4

B. (1) Iodine-131

(2) 4

C. (1) Krypton-85

(2) 8

D. (1) Iodine-131

(2) 8

Reason answer is correct: The fuel in the Spent Fuel Pool has been there for more than 6 months. The personal non-detectable radiation hazard would be Krypton-85 which is a beta emitter. AOP-013 has a note stating: Kr-85 is the primary radiological concern for fuel off-loaded more than 6 months ago. Kr-85 is a beta hazard and will NOT be detected by personal dosimetry or area radiation monitors. There is also a caution stating: Airborne radiation may be present and gas bubbles may be visible if a fuel assembly is ruptured. Personnel should remain clear until Health Physics has established access controls

The basis document states the activity of most concern is that which is contained in the volatile fission product gases contained in the fuel pellet to cladding gap. When a fuel pin is damaged, this fission product inventory can be released to the SFP water. Technical Specifications 3.9.10 and 3.9.11 require a minimum water level of 23 feet in the SFP and Refueling Cavity specifically to reduce the potential dose resulting from a fuel handling accident. This amount of water will capture 99% of the assumed 10% iodine activity present in the pellet to clad gap before it breaks the surface of the water. However, although the water is expected to retain a large fraction of this activity, a portion of it will reach the surface and bubble out into the FHB or CNMT atmosphere. (Since halogens are soluble, a large fraction of these halogens will be retained by the water, whereas noble gases, being insoluble, will not be retained.) Once in the atmosphere, much of this fission product activity will cause an observed increase in area radiation levels. (Gases such as Kr-85 which are primarily beta hazards will not be detectable using installed monitors.)

The transportation of a potentially contaminated individual must be reported to the NRC within 8 hours per AD-LS-ALL-0006; however, since a press release regarding the event is planned by Duke Energy, this incident must be reported to the NRC within 4 hours.

A. Correct.

B. Incorrect. The first part is plausible since volatile fission product gases have escaped from the damaged fuel assembly and would be observable as bubbles coming to the surface of the SFP. Iodine-131 would be part of the volatile gases. I-131 is gamma emitter which would be detectable with personal dosimetry and therefore would NOT be a non-detectable radiation concern. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since under other circumstances, an 8-hour NRC report would be required for transport of a potentially contaminated individual. A 4-hour report applies only because a related offsite notification to another agency will be made.

D. Incorrect. The first part is plausible since volatile fission product gases have escaped from the damaged fuel assembly and would be observable as bubbles coming to the surface of the SFP. Iodine-131 would be part of the volatile gases. I-131 is gamma emitter which would be detectable with personal dosimetry and therefore would NOT be a non-detectable *radiation concern. The second part is plausible since under other circumstances, an 8-hour NRC report would be required for transport of a potentially contaminated individual. A 4-hour report applies only because a related offsite notification to another agency will be made.*

2.3 Radiation Control

G2.3.14; Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

(CFR: 41.12 / 43.4 / 45.10)

Importance Rating:	3.4 3.8
Technical Reference:	AD-LS-ALL-0006, Attachment 2, Page 38, Rev. 3 AOP-013-BD, Section 1.0, Page 3, Rev. 4 AOP-013, Section 3.1, NOTE preceding Step 7, Page 7, Rev. 16
References to be provided:	AD-LS-ALL-0006, Attachments 2 & 3
Learning Objective:	PP-LP-2.17 Objective 4 AOP-LP-3.13 Objective 2.c
Question Origin:	Bank (2013 NRC SRO 22)
Comments:	K/A is matched since applicant must demonstrate an understanding of radiological concerns associated with off-loading fuel.
SRO Justification:	Notifications to the NRC are an SRO-only task. Task #344039H504 - Perform Notifications for Events per AP-617 (now AD-LS-ALL-0006)

98. 2020 NRC SRO 023

Which ONE of the following personnel are RESPONSIBLE for preparing and approving a Batch Gaseous Effluent Permit in accordance with OP-120.07, Waste Gas Processing?

	Prepares Permit	Approves Permit
A.	Chemistry	Control Room Supervisor
B.	Chemistry	Shift Manager
C.	Radiation Protection	Control Room Supervisor
D.	Radiation Protection	Shift Manager

Reason answer is correct: In accordance with OP-120.07, Waste Gas Processing, Section 8.12, prior to discharging a gaseous batch release a copy of the Discharge Log is given to Chemistry for issuance of the Batch Gaseous Effluent Permit per Step 31. Chemistry prepares the permit in accordance with CRC 853. Prior to starting the release, the Shift Manager reviews and signs the Batch Gaseous Effluent Permit per Step 44.

A. Incorrect. The first part is correct. The second part is plausible since the CRS is a licensed on-shift crew member in charge of control room activities (e.g. approves DSRs, OSTs, etc.); however this is incorrect since he/she is not responsible for approving the release of gaseous permits for the station.

B. Correct.

C. Incorrect. The first part is plausible since the release is a radioactive release and Radiation Protection is responsible for monitoring the dose of station activities and related radioactive conditions; however this is incorrect since they are not responsible for preparing Batch Gaseous Effluent permits. The second part is plausible since the CRS is a licensed on-shift crew member in charge of control room activities (e.g. approves DSRs, OSTs, etc.); however this is incorrect since he/she is not responsible for approving the release of gaseous permits for the station.

D. Incorrect. The first part is plausible since the release is a radioactive release and Radiation Protection is responsible for monitoring the dose of station activities and related radioactive conditions; however this is incorrect since they are not responsible for preparing Batch Gaseous Effluent permits. The second part is correct.

2.3 Radiation Control

G2.3.6; Ability to approve release permits.

(CFR: 41.13 / 43.4 / 45.10)

Importance Rating: 2.0 3.8

Technical Reference: OP-120.07, Section 8.12.2, Pages 72 & 77, Rev. 83

References to be provided: None

Learning Objective: GWPS-ILC Objective 5.a

Question Origin: Bank (2014 NRC SRO 24)

Comments: K/A is matched since the applicant must determine who approves a Batch Gaseous Effluent Permit.

Tier/Group: T3

SRO Justification: 10 CFR Part 55 Content - 43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

One example cited in ES-401 Attachment 2 for this topic is the process for gaseous/liquid release approvals (i.e., release permits).

99. 2020 NRC SRO 024

Which ONE of the following completes the statements below in accordance with AD-OP-ALL-0207, Fire Brigade Administrative Controls?

The ___(1)___ is responsible for providing qualified staffing for the Fire Brigade.

In addition to the Incident Commander, a MINIMUM of ___(2)___ Fire Brigade members must be knowledgeable of plant safety-related equipment and the affects of fire suppressants on safe shutdown capabilities.

A. (1) Shift Manager

(2) one

B. (1) Shift Manager

(2) two

C. (1) Control Room Supervisor

(2) one

D. (1) Control Room Supervisor

(2) two

Reason answer is correct: In accordance with AD-OP-ALL-0207, the Shift Manager is responsible for providing staffing for the Fire Brigade. This responsibility includes ensuring that in addition to the Incident Commander, at least two Fire Brigade members are knowledgeable of plant safety-related equipment and the affects of fire suppressants on safe shutdown capabilities.

A. Incorrect. The first part is correct. The second part is one individual is designated as the safe shutdown operator on the watchbill who must be knowledgeable of plant safety-related equipment for safe shutdown of the unit.

B. Correct.

C. Incorrect. The first part is plausible since the CRS has responsibilities during a fire which include directing the activities of Control Room personnel; however, this is incorrect as the Shift Manager is responsible for Fire Brigade staffing. The second part is one individual is designated as the safe shutdown operator on the watchbill who must be knowledgeable of plant safety-related equipment for safe shutdown of the unit.

D. Incorrect. The first part is plausible since the CRS has responsibilities during a fire which include directing the activities of Control Room personnel; however, this is incorrect as the Shift Manager is responsible for Fire Brigade staffing. The second part is correct.

2.4 Emergency Procedures / Plan

G2.4.26; Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.

(CFR: 41.10 / 43.5 / 45.12)

Importance Rating:	3.1 3.6
Technical Reference:	AD-OP-ALL-0207, Sections 4.0 % 5.3.2, Pages 6 & 10, Rev. 3
References to be provided:	None
Learning Objective:	PP-LP-3.0 Objective 8
Question Origin:	New
Comments:	K/A is matched since the applicant must demonstrate an understanding of specific fire brigade staffing requirements.
Tier/Group:	T3
SRO Justification:	Fire brigade staffing (part of shift turnover process) is an SRO-only task. Task #341001H102 - Perform Shift Relief and Turnover Process per OMM-002 & AD-OP-ALL-0108

100. 2020 NRC SRO 025

Given the following plant conditions:

- A seismic event has occurred resulting in a small break LOCA inside Containment
- Damage to the intake structure has resulted in a loss of Emergency Service Water
- Containment pressure is 12 psig and rising
- No Containment Spray pumps are running
- Offsite power remains available

The crew is currently implementing EOP-E-1, Loss of Reactor or Secondary Coolant.

Which ONE of the following describes restoration of containment cooling?

PROCEDURE TITLES:
AOP-022, Loss of Service Water
EOP-FR-Z.1, Response to High Containment Pressure
ISG-CC, Containment Cooling

A. Continue in EOP-E-1 and perform AOP-022 in parallel to establish Service Water to the Containment Fan Cooler Units.

B. Transition to EOP-FR-Z.1 and perform AOP-022 in parallel to establish Service Water to the Containment Fan Cooler Units.

C. Continue in EOP-E-1 and perform ISG-CC in parallel to supply cooling water using fire water to the Containment Fan Cooler Units.

D. Transition to EOP-FR-Z.1 and perform EOP-E-1 in parallel to re-establish containment cooling using the Containment Spray System.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with the EOP Users Guide, while implementing EOPs, it may be necessary to implement actions identified in the AOPs. This is acceptable assuming that referencing an AOP does not delay accident mitigation as outlined in the EOPs. Particular attention should be given to actions that will protect major plant equipment and/or enhance plant control.

With offsite power still available, AOP-022, Attachment 4, can be used to supply the ESW headers with NSW. Once cooling water is restored, the Containment Fan Cooler Units can then be restarted using EOP-FR-Z.1 to provide cooling to the containment.

A. Incorrect. Plausible since one of the entry conditions for EOP-FR-Z.1 is not met (CNMT pressure ≥ 45 psig); however, this is incorrect EOP-FR-Z.1 must also be entered when CNMT pressure is ≥ 10 psig with no Containment Spray Pumps running. Also plausible since AOP-022 will be performed in parallel to restore Service Water to the Containment Fan Cooler Units.

B. Correct.

C. Incorrect. Plausible since one of the entry conditions for EOP-FR-Z.1 is not met (CNMT pressure ≥ 45 psig); however, this is incorrect EOP-FR-Z.1 must also be entered when CNMT pressure is ≥ 10 psig with no Containment Spray pumps running. Also plausible since ISC-CC can be used to provide alternate sources/methods for containment cooling in the event of damage to areas of the plant; however, the ISG uses fire water to flood the containment, not supply cooling water to the Containment Fan Cooler Units. Plausibility further supported in that fire water can be used to supply cooling to the Spent Fuel Pool Heat Exchangers.

D. Incorrect. Plausible since entry conditions for EOP-FR-Z.1 are met (Containment pressure ≥ 10 psig with no Containment Spray pumps running) and both EOP-FR-Z.1 and EOP-E-1 address restoration of Containment Spray; however, this is incorrect since functional recovery procedures are not performed in parallel with EOPs per the Users Guide.

You have completed the test!

2.4 Emergency Procedures / Plan

G2.4.8; Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.8 4.5
Technical Reference:	EOP Users Guide, Section 5.1.2, Page 13, Rev. 51 AOP-022 Attachment 4, Page 74, Rev. 40 EOP-FR-Z.1, Entry Conditions, Page 2, Rev. 2 EOP-CSFST, Containment CSF-5, Page 3, Rev. 13
References to be provided:	None
Learning Objective:	EOP-LP-3.19 Objective 1.b
Question Origin:	Bank (Indian Point)
Comments:	K/A is matched since the applicant must demonstrate an understanding use of an AOP in parallel with an EOP to mitigate the accident. AOP-022 will address restoration of Service Water to the Containment Fan Coolers and EOP-FR-Z.1 will address starting them.
Tier/Group:	T3
SRO Justification:	10 CFR Part 55 Content - 43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

You have completed the test!

HARRIS 2020 NRC SCENARIO 1

Facility:	Harris Nuclear Plant	Scenario No.:	1	Op Test No.:	05000400/2020301
Examiners:			Operators:	SRO:	
				RO:	
				BOP:	

Initial Conditions: IC-26 MOL, 88% power

- ‘B’ MDAFW Pump is under clearance for pump packing repairs
- 1SI-3 Out Of Service
- ‘B’ DEH Pump Out of Service

Turnover:	The plant is at 88% power, middle of core life. GP-006 step 10 to Adjust MS Flow to HP Turbine per OP-131.04 Section 8.6 as applicable
Critical Task:	• Manually align at least one high head ECCS pump flow path to prevent RVLIS Dynamic Range Level from lowering below 60% • Depressurize the RCS to minimize primary to secondary leakage to prevent SG ‘C’ exceeding 95% level

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R – RO/SRO N – BOP/SRO	Power reduction from 88% power
2	crf14b	C – RO/SRO	Control rods fail to move in Auto - continue down power with rods in manual (AOP-001)
3	lt:112	I – RO/SRO	Failure of VCT LT-112 to 100%, which will full divert letdown to RHT (AOP-003)
4	hva011 z3274tib	C – BOP/SRO TS – SRO	Trip of running AH-85C fan, standby fails to Auto Start
5	pt:2307	I – BOP/SRO	MFW Pump Suction Pressure to CBP controller failure
6	sgn05c	C – RO/SRO TS – SRO	'C' Steam Generator Tube Leak (AOP-016)
7	sgn05c	M – ALL	'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-0 and EOP-E-3)
8	zrpk603a	C – BOP/SRO	Relay failure on resultant SI signal K603A
9	zdsq2:6b jpb9101b	C – RO/SRO	‘B’ ESW Pump fails to auto start on SI

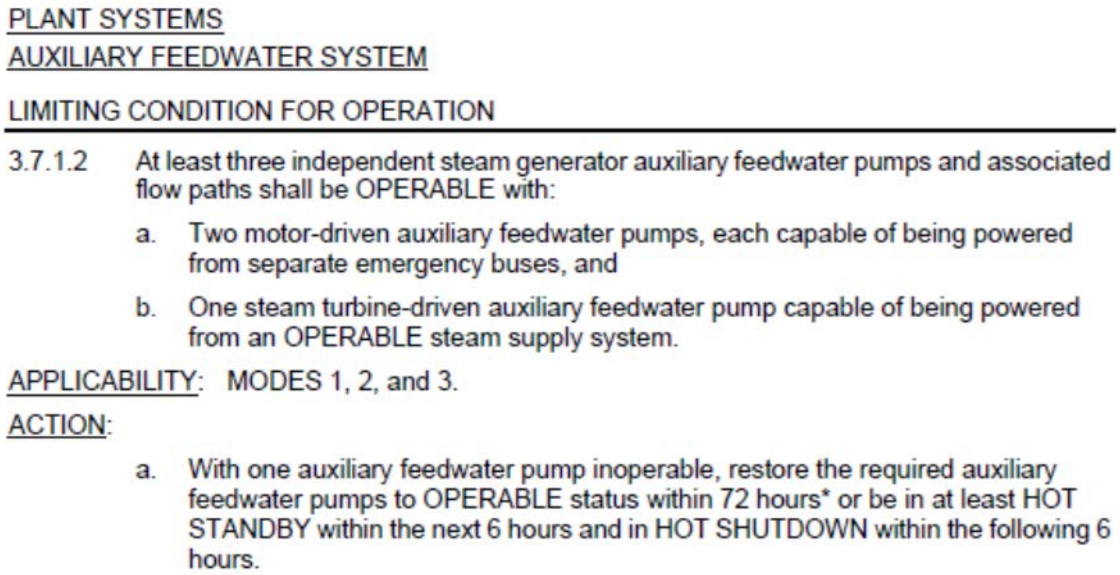
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 1

The plant is at 88% power, middle of core life. Due to the ‘B’ MDAFW pump LCO expiring, a normal shutdown in accordance with GP-006, Normal Plant Shutdown From Power Operation To Hot Standby (Mode 1 To Mode 3) is in progress as directed by plant management. It is to continue after shift turnover at 4 MW / minute.

The following equipment is under clearance:

- MDAFW Pump B-SB is under clearance for pump packing repairs. The pump has been inoperable for 66 hours and cannot be restored to operable status. Tech Spec 3.7.1.2 Action **a** and Tech Spec 3.3.3.5.b Action **c** applies.



PLANT SYSTEMS
AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

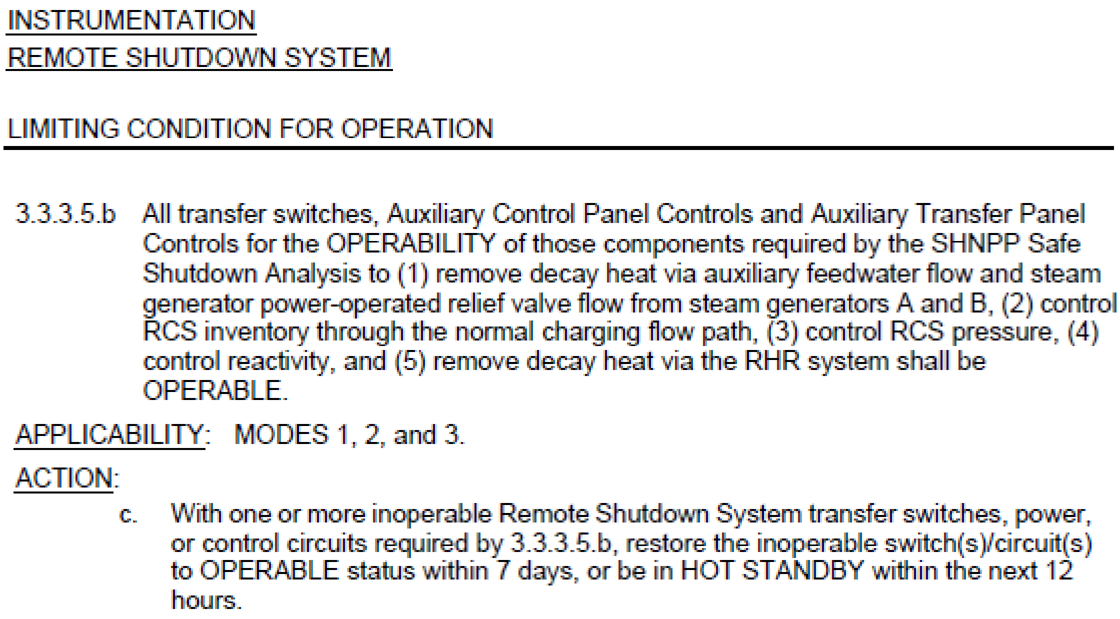
a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency buses, and

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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



INSTRUMENTATION
REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

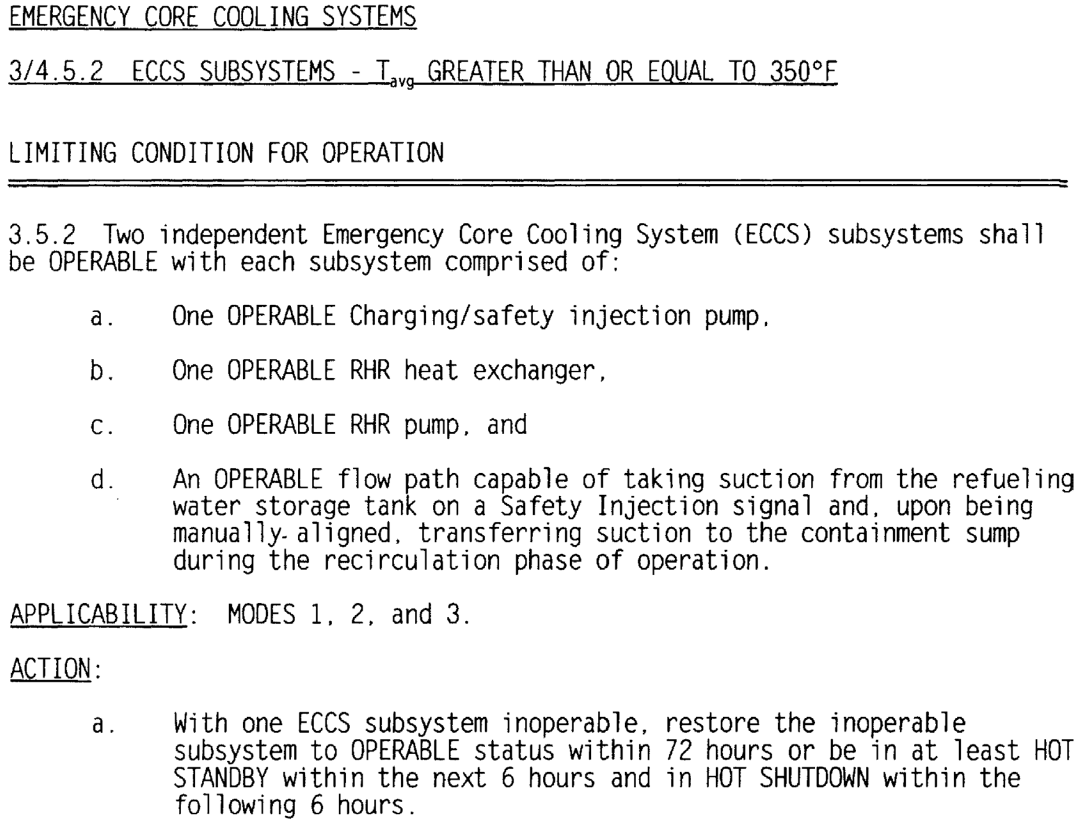
c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 1 continued

The following equipment is under clearance (continued):

- 'B' DEH Pump is under clearance for motor repairs. The pump has been unavailable for 8 hours. Repairs are expected to be completed within 24 hours.

- 1SI-3, Boron Injection Tank Outlet valve has been under clearance the last 12 hours for breaker repairs. The repairs are close to completion and the valve is expected to be returned to service within the next hour. The valve is currently shut with power removed. OWP-SI-01 has been completed. Tech Specs 3.5.2 Action **a** and Tech Specs 3.6.3 applies.



EMERGENCY CORE COOLING SYSTEMS

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

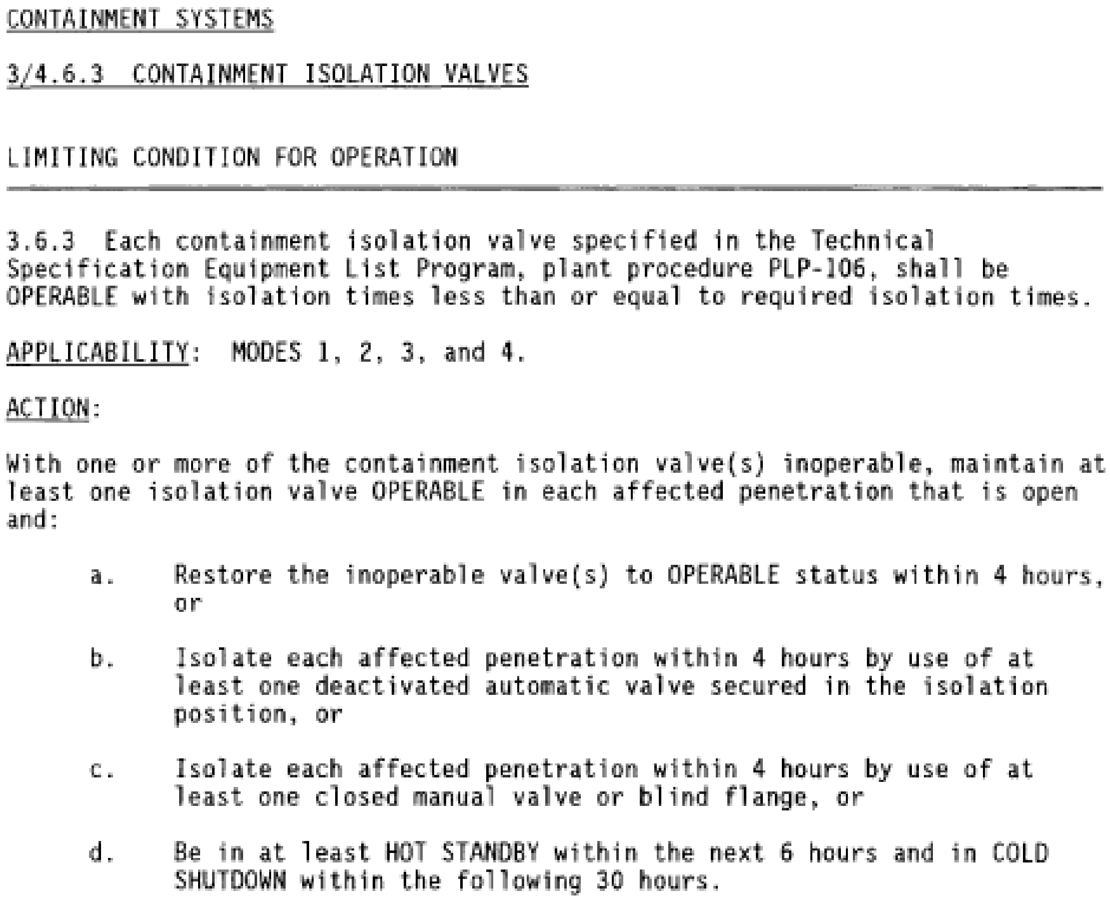
d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 1 (Continued)



CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

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

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

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 1 (Continued)

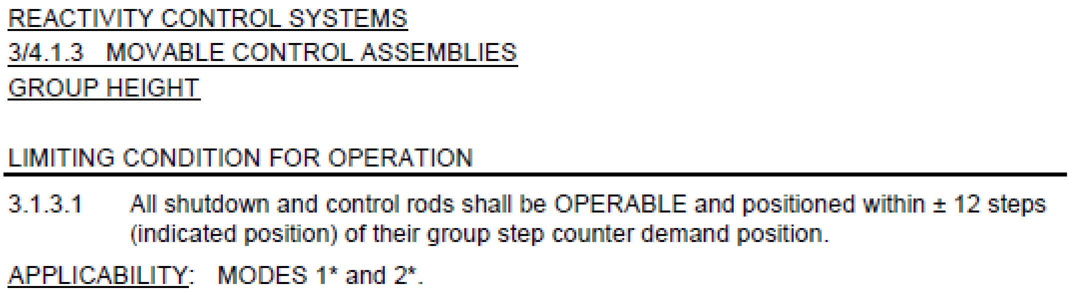
Event 1: Plant Shutdown (GP-006). Turnover takes place with the unit at 88% Reactor power. The crew will be given credit for a reactivity manipulation during the down power.

Verifiable Action: It is expected that the SRO will conduct a reactivity brief, the RO will borate and monitor auto rod insertion per the reactivity plan. The BOP will operate the TCS Load control to adjust the Turbine ramp rate from 1 GVPC units to 4 GVPC units per minute then ensure the controls are set correctly to lower power. After power is reduced 3% - 5% and the crew has demonstrated that they have control of the plant during a shutdown Event 2 is pre- inserted and will be identified once the T_{avg}/T_{ref} mismatch is greater than 2°F.

Event 2: Control rods fail to move in Auto. T_{avg}/T_{ref} recorder TR-408 along with ERFIS quick plot Tave will provide indication of the T_{avg}/T_{ref} mismatch. If the crew allows the mismatch to reach +/- 5°F ALB 010-6-4B, RCS Tref/Tavg High-Low, will alarm.

Verifiable Action: The crew will enter AOP-001 and carry out the immediate actions. The RO will perform the immediate actions of AOP-001 by verifying that <2 rods are dropped (no rods have dropped), place Rod Control in MANUAL and then verify no rod motion. Once the immediate actions are complete the BOP should place the Turbine in Hold it stabilize the plant. With concurrence from the SRO the RO will restore T_{avg} to match T_{ref} by inserting the rods in manual.

The SRO should evaluate Tech Spec 3.1.3.1, Reactivity Control Systems- Movable Control Assemblies - Group Height and 3.1.3.5, Reactivity Control Systems- Shutdown Rod Insertion Limit both conditions are satisfied based on AOP-001 Attachment 5.



REACTIVITY CONTROL SYSTEMS

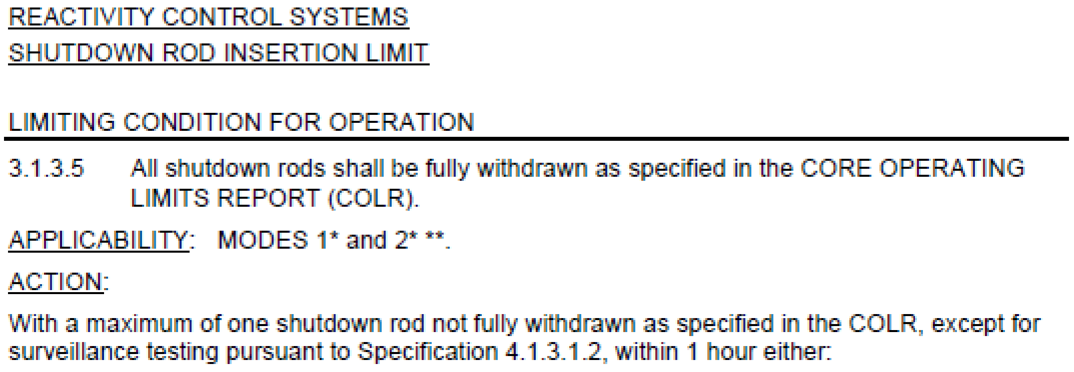
3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*.



REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2* **.

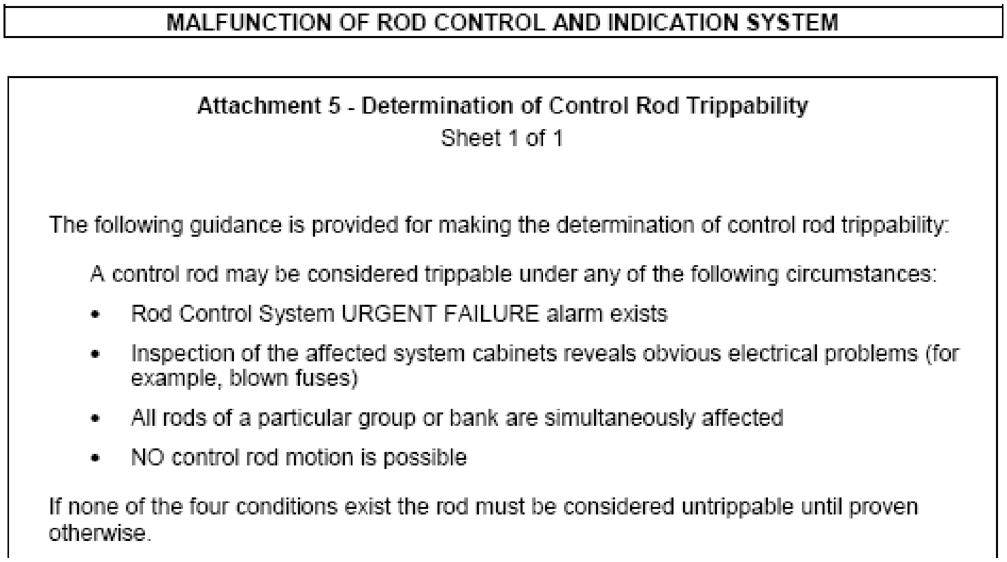
ACTION:

With a maximum of one shutdown rod not fully withdrawn as specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 1 (Continued)

Event 2: Tech Spec evaluation continued

AOP-001 Attachment 5



MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 5 - Determination of Control Rod Trippability

Sheet 1 of 1

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

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

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

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

The SRO should provide a temperature band of +/- 5°F to the RO in accordance with OMM-001, Attachment 11, Control Bands And Administrative Limits. The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 3: Failure of VCT LT-112 to 100% which will full divert letdown to RHT. ALB 007-5-5, Computer Alarm Chem & Vol Systems, will alarm due to LT-112 being greater than 75%. If the crew allows actual level on LT-115 to reach 20% an auto makeup from the Reactor Makeup System will occur.

Verifiable Action: The crew will respond by entering AOP-003 which has NO immediate actions. A failure of LT-112 high will cause 1CS-120, Letdown VCT/Hold Up Tank valve to shift to the Hold Up Tank. The RO will have to return the MCB switch to the VCT position. Since VCT level has failed HIGH auto CSIP suction switch over on 5% VCT level to the RWST will not occur until Maintenance has lifted the leads associated with LT-112. The operator will have to monitor VCT level and communicate with Maintenance to resolve this failure.

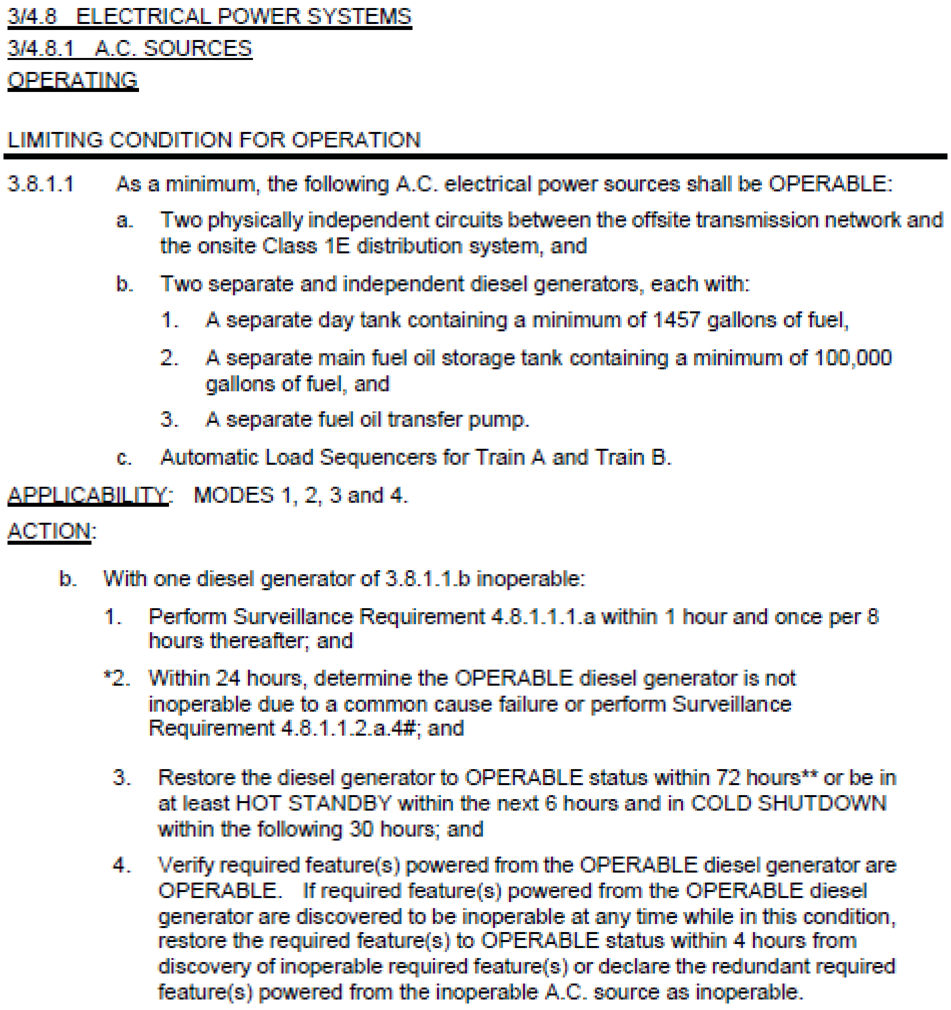
The SRO should provide a level band of 20 to 70% to the RO in accordance with AOP-003, Section 3.1, Step 4 RNO. The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 1 (Continued)

Event 4: Trip of running AH-85C fan, standby fails to Auto Start. This will cause the running Diesel Generator electrical equipment room supply fan AH-85 1C-SB. ALB 027-1-4, Diesel Gen Elec Equip Rm Sup Fans AH-85 Low Flow – O/L, will alarm and the standby fan AH-85 1D-SB fails to automatically start.

Verifiable Action: The BOP should identify that the auto start feature of the standby AH-85 1D-SB has failed. The crew will use the APP-ALB 027 to start the standby fan AH-85 1D-SB.

The SRO should evaluate Tech Spec 3.8.1.1, Electrical Power Systems - AC Sources – Operating and 3.3.3.5.b, Instrumentation - Remote Shutdown System Action: **b** and **c** respectively.



3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. A separate day tank containing a minimum of 1457 gallons of fuel,
2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel, and
3. A separate fuel oil transfer pump.

c. Automatic Load Sequencers for Train A and Train B.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

b. With one diesel generator of 3.8.1.1.b inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

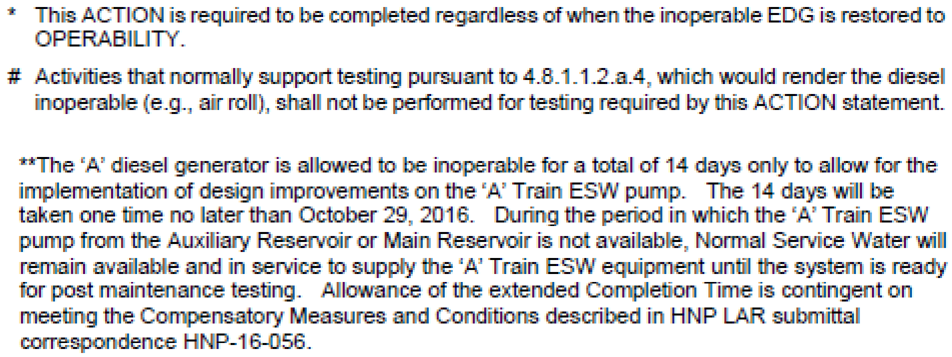
*2. Within 24 hours, determine the OPERABLE diesel generator is not inoperable due to a common cause failure or perform Surveillance Requirement 4.8.1.1.2.a.4#; and

3. Restore the diesel generator to OPERABLE status within 72 hours** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. Verify required feature(s) powered from the OPERABLE diesel generator are OPERABLE. If required feature(s) powered from the OPERABLE diesel generator are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 4 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 1 (Continued)

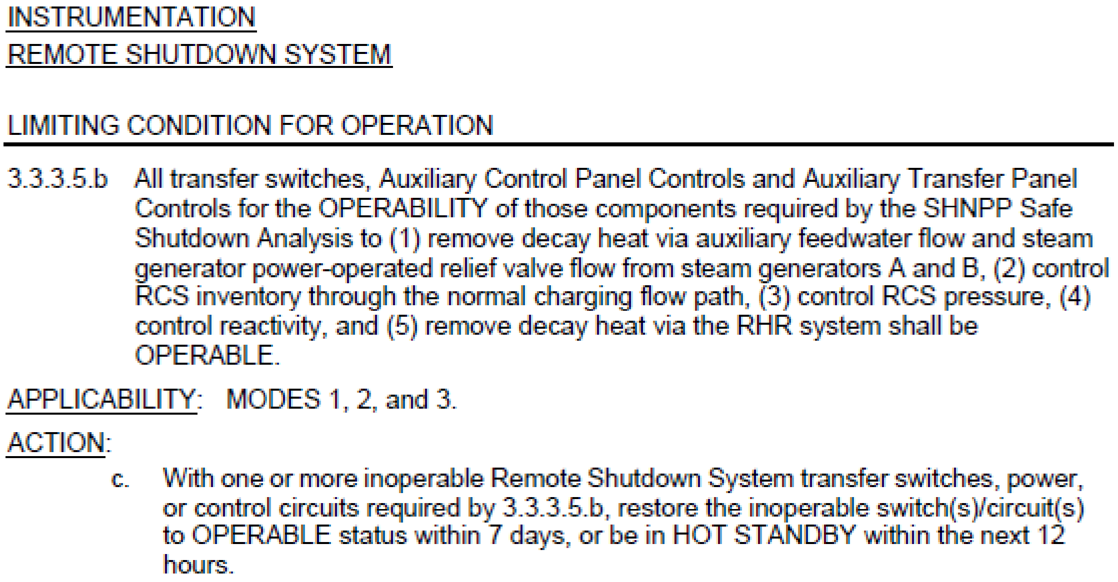
Event 4: Tech Spec evaluation continued



* This ACTION is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

\# Activities that normally support testing pursuant to 4.8.1.1.2.a.4, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

**The 'A' diesel generator is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence HNP-16-056.



INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

OWP-HVAC Attachment 1, HVAC Support System Requirements, lists AH-85 1C-SB TS 3.3.3.5.b since ONLY AH-85 1C-SB can be credited for supported system operability, since AH-85 1D-SB does not start automatically during an accident.

The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 5: MFW Pump Suction Pressure to CBP controller failure. A transmitter failure will cause the Condensate Booster Pump controller to reject to manual. ALB 019-4-1A and 4-1B, Cndbstr Pmps 10% Deviation and Cndbstr Pmps 20% Dev/Man Rej, respectively will alarm and both Condensate Booster pump M/A stations to go manual. There will also be alarms on Feedwater heater levels and if the crew does not respond quickly then SG level deviation alarms will alarm. The failure will cause SG levels increase due to the higher suction pressure being supplied to the MFW pumps.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 1 (Continued)

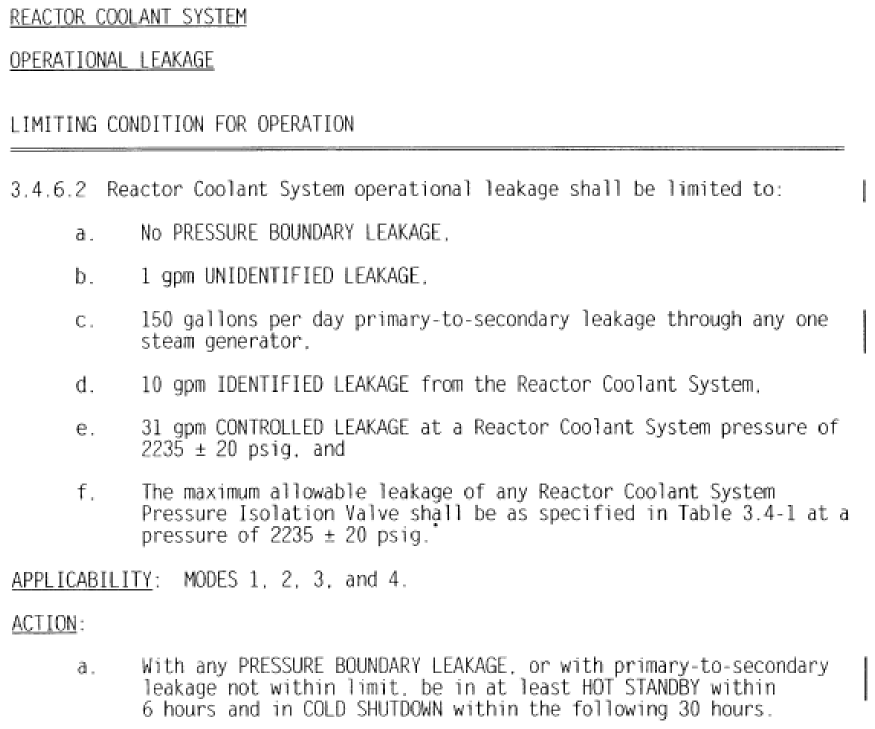
Verifiable Action: The BOP will respond to the failure by taking actions contained in the APP-ALB 019 4-1B by manually controlling PI-2200, FW pumps suction header pressure at 430 psig using both Condensate Booster pump M/A stations (PK-2307 and PK-2308). Both controls will quickly reach 100% and must be individually lowered to regain normal supply pressure.

The SRO should provide a pressure band of 430 psig +/- 5 psig to the BOP in accordance with AD-OP-ALL-1000, Conduct of Operations, (5.5.6) for operation Control Bands and APP-ALB 019-4-1B. The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 6: 'C' Steam Generator Tube Leak (AOP-016). The RCS Loop ‘C’ will leak into its associated SG requiring the crew to implement the actions for AOP-016. Minor changes in Pressurizer Level and Charging flow will occur. Radiation monitors will alarm on the RMS computer for CVPETS and MSL ‘C’. Additionally ALB 010-4-5, Rad Monitor System Trouble, will alarm due to the MSL ‘C’ RM-23 alarming.

Verifiable Action: The crew will respond by entering AOP-016 which has NO immediate actions. The RO will perform a leak rate calculation and determine the leakage is ~30 gpm. The BOP will make plant announcements and contact various support organizations (HP, Chemistry, etc.) as directed by the AOP. The SRO should determine that leak rate is in excess of Action Level 3 and the unit must be less than 50% within the hour and removed from service within the next 2 hours. The crew will implement AOP-038, Rapid Down power to complete this action.

The SRO should evaluate Tech Spec 3.4.6.2, Reactor Coolant System – Operational Leakage Action: a. which will be completed by performing the more restrictive PSAL 3 requirements.



REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 31 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 ± 20 psig.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 1 (Continued)

The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 7: 'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-0 and EOP-E-3). The major event is a Steam Generator Tube Rupture. The RCS Loop ‘C’ leak will degrade to a rupture into its associated SG requiring the crew to implement the continuous actions for AOP-016 with leak rate in excess of VCT makeup ability and trip the Reactor and actuate Safety Injection. Major changes in Pressurizer Level and Charging flow will occur.

Verifiable Action: The RO will manually trip the Reactor in accordance with AOP-016, then following verification of the Turbine trip actuate Safety Injection and the crew will continue with EOP-E-0. The crew will then transition from EOP-E-0 and go to EOP-E-3, Steam Generator Tube Rupture.

Event 8: Relay failure on resultant SI signal K603A. The failure of K603A will result in the failure of 3 ‘A’ train SI signals 1SI-4 fails to open, 1CS-238 fails to shut and CRI fails to occur.

Verifiable Action: The RO will manually open 1SI-4 (**Critical Task #1**) and shut 1CS-238 in accordance with EOP-E-0, Attachment 1, SI Emergency Alignment. The BOP should identify the ‘A’ train Control Room Area Ventilation are not properly aligned and will manually align the components in accordance with EOP-E-0, Attachment 3, Safeguards Actuation Verification or AD-OP-ALL-1000, Conduct of Operations, (5.6.3.8) for Equipment Manipulation and Status Control.

Event 9: ‘B’ ESW Pump fails to auto start on SI.

Verifiable Action: The crew should identify this failure and manually start the Emergency Service Water Pump once the ‘B’ Sequencer reaches Load Block 9, Automatic Manual Loading Permissive, in accordance with AD-OP-ALL-1000, Conduct of Operations, (5.6.3.8) for Equipment Manipulation and Status Control or EOP-E-0, Attachment 3, Safeguards Actuation Verification. The crew may elect to perform the immediate actions of AOP-022, Service Water Malfunctions, and secure both the ‘B’ EDG and the ‘B’ CSIP with service water loss to the respective running component.

The scenario termination is met in EOP-E-3 after the RCS has been depressurized to minimize primary to secondary leakage prior to SG ‘B’ exceeding 95% level (**Critical Task #2**) and all but one CSIP is secured.

CRITICAL TASK JUSTIFICATION:

1. Manually align at least one high head ECCS pump flow path to prevent RVLIS Dynamic Range Level from lowering below 60%.

In this scenario the 1SI-3 is out of service and the 1SI-4 does not automatically open from sequencer actuation. The operator must manually open 1SI-4 which was currently in the shut position. Plant parameter grading criteria for the task is opening 1SI-4 to prevent RVLIS Dynamic Range Level from lowering below 60% which constitutes a significant core uncover with 3 Reactor Coolant Pumps in operation.

2. Depressurize the RCS to minimize primary to secondary leakage to prevent SG ‘C’ exceeding 95% level

Failure to depressurize the RCS needlessly complicates mitigation of a SGTR event by allowing the reactor coolant leak to continue. It constitutes a significant reduction of safety margin beyond that introduced by the SGTR event analysis.
If primary to secondary leakage is not stopped the SG pressure will increase until either the SG PORV or Safety valve(s) open releasing radioactivity to the environment. If leakage is allowed to continue the increased inventory will result in water release through the PORV once SG overfill conditions are reached.
At Harris the plant 95% level on the narrow range indicators is the value at which overfill conditions will start to exist and the adverse effects of the condition may start to manifest themselves.

Note: Causing an unnecessary plant trip or ESF actuation may constitute a CT failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

Simulator Setup

Reset to IC-141 password “NRC3sros”

Go to RUN

Silence and Acknowledge annunciators

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner.

Set ERFIS screens for normal full power conditions, ensure VCT Level Channel LCS0112 is indicated on QP VCT

(The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

SPECIAL INSTRUCTIONS

Provide a Reactivity Plan to candidates for shutting down the plant

Provide a copy of the following procedures:

- GP-006, NORMAL PLANT SHUTDOWN FROM POWER OPERATION TO HOT STANDBY (MODE 1 TO MODE 3) **marked up** through section 6.2 step 10

Press START on Counter Scaler
Post conditions for status board from IC-26 Reactor Power 88%
Control Bank D at 201 steps
RCS boron 980 ppm

Turnover: The plant is at 88% power, middle of core life. Due to the ‘B’ MDAFW pump LCO expiring, a normal shutdown in accordance with GP-006, Normal Plant Shutdown From Power Operation To Hot Standby (Mode 1 To Mode 3) is in progress as directed by plant management. It is to continue after shift turnover at 4 MW / minute.

Equipment Under Clearance:

- ‘B-SB’ MDAFW Pump is under clearance for motor high vibrations. The pump has been inoperable for 66 hours and cannot be restored to operable status. Tech Spec 3.7.1.2 LCO Action **a** and Tech Spec 3.3.3.5.b Action **c** applies.

- ‘B’ DEH Pump is under clearance for motor repairs. The pump has been unavailable for 8 hours. Repairs are expected to be completed within 24 hours.

- 1SI-3, Boron Injection Tank Outlet valve has been under clearance the last 12 hours for breaker repairs. The repairs are close to completion and the valve is expected to be returned to service within the next hour. The valve is currently shut with power removed. OWP-SI-01 has been completed. Tech Specs 3.5.2 Action **a** and Tech Specs 3.6.3 applies.

Simulator Setup (continued)

Align equipment for repairs:

Place CIT on 'B-SB' MDAFW pump MCB Switch
Place protected train placards in accordance with OMM-001 Attachment 5
    Protected Train placards on 'A-SA' MDAFW pump, 'B-SB' RHR Pump, 'B-SB' CCW Pump, 'B-SB' ESW Pump, 1MS-70 and 1MS-72

Place the ‘B’ DEH Pump in PTL and then hang a CIT on MCB switch
Place protected train placards in accordance with AD-OP-ALL-0210, Single Point Vulnerabilities
    Protected Train placards on ‘A’ DEH Pump

Place a CIT on the switch for 1SI-3.

Place protected train placards in accordance with Response to Industry Best Practices, Expectations
    Protected train placards on ‘A-SA’ ESW Pump, ‘A-SA’ CCW Pump, and ‘A-SA’ SFP Hx

Place filled out copies of OWP’s into the OWP book – ensure they are removed at end of day

- OWP-SI-01 and place in MCR OWP book for 1SI-3 clearance

Hang restricted access signs on MCR entry swing gates

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	14	of	83
Event Description:			**Power Reduction**						

Time	Position	Applicant's Actions or Behavior

Lead Evaluator:	**The crew has been directed to re-commence a power reduction from 88% to the unit is off line. The power reduction is on hold for turnover. The SRO is expected to conduct a reactivity brief prior to commencing the power reduction. This brief may be conducted outside the simulator prior to starting the scenario.**
	When the crew has completed their board walk down and are ready to take the shift inform the Simulator Operator to place the Simulator in Run. When the Simulator is in run announce: **CREW UPDATE – (SRO's Name) Your crew has the shift. END OF UPDATE**

Simulator Operator:	**When directed by the Lead Evaluator, ensure that the annunciator horns are on and place the Simulator in RUN.**

Evaluator Note:		**The crew may elect to begin Boration prior to lowering turbine load.**
	RO	OP-107.01, Section 5.2
	RO	DETERMINE the volume of boric acid to be added. (Current OPT-1536 data or approved reactivity plan from Engineering may be used.)
	SRO	Directs Boration
Procedure Note:		**FIS-113, BORIC ACID BATCH COUNTER, has a tenths position.**

Op Test No.: NRC	Scenario # 1	Event # 1	Page 15 of 83
Event Description:	**Power Reduction**		

Time	Position	Applicant's Actions or Behavior
Procedure Caution:		**If the translucent covers associated with the Boric Acid and Total Makeup Batch counters FIS-113 and FIS-114, located on the MCB, are not closed, the system will not automatically stop at the preset value.**
	RO	• SET FIS-113, BORIC ACID BATCH COUNTER, to obtain the desired quantity. • ENSURE the RMW CONTROL switch has been placed in the STOP position. • ENSURE the RMW CONTROL switch green light is lit.
Procedure Note:		• **Boric Acid flow controller must be set between 0.2 and 6 (1 and 30 gpm.).** • **Performing small borations at high flow rates may result in an overboration based on equipment response times. Boration flow should be set such that the time required to reach the desired setpoint will happen after release of the control switch.**
	RO	IF the current potentiometer setpoint of controller 1CS-283, FK-113 BORIC ACID FLOW, needs to be changed to obtain makeup flow, THEN: (N/A)
		• RECORD the current potentiometer setpoint of controller 1CS-283, FK-113 BORIC ACID FLOW, in Section 5.2.3. • SET controller 1CS-283, FK-113 BORIC ACID FLOW, for the desired flow rate.
	RO	PLACE control switch RMW MODE SELECTOR to the BOR position.

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	16	of 83
Event Description:			**Power Reduction**					

Time	Position	Applicant's Actions or Behavior

Procedure Note:	• **Boration may be manually stopped at any time by turning control switch RMW CONTROL to STOP.** • **During makeup operations following an alternate dilution, approximately 10 gallons of dilution should be expected due to dilution water remaining in the primary makeup lines.**

	RO	• START the makeup system as follows: o TURN control switch RMW CONTROL to START momentarily. o ENSURE the RED indicator light is LIT. o IF expected system response is not obtained, THEN TURN control switch RMW CONTROL to STOP. • ENSURE boration automatically terminates when the desired quantity of boron has been added.
	RO	IF controller 1CS-283, FK-113 BORIC ACID FLOW, was changed in Step 5.2.2.5, THEN: (N/A)
		• REPOSITION controller 1CS-283, FK-113 BORIC ACID FLOW, to the position recorded in Step 5.2.2.5.a. • INDEPENDENTLY VERIFY controller 1CS-283, FK-113 BORIC ACID FLOW, position.
	RO	• Monitor Tavg and rod control for proper operation. • Establish VCT pressure between 20-30 psig. • Turn control switch RMW MODE SELECTOR to AUTO. • START the makeup system as follows: o TURN control switch RMW CONTROL to START momentarily. o ENSURE the RED indicator light is LIT. o IF expected system response is not obtained, THEN TURN control switch RMW CONTROL to STOP. (Ref. 4.0.31)

Time	Position	Applicant's Actions or Behavior
Evaluator Note:		**The following steps have been completed to achieve the current power level. The crew should validate status of the turbine load reduction in accordance with GP-006 section 6.2 step 5 before re-initiating the turbine load reduction.**
GP-006		GP-006, Section 6.2
Procedure Note:		**Routine load changes must be coordinated with the Load Dispatcher to meet system load demands** **GVPC is the preferred method of Load Control. Megawatt Control is normally used only during GV and TV testing** **Controls and indications in following steps are on the TCS Load Control screen** **If Oper Entry is selected with the Turbine in GO, the value currently in the Ramp Rate Entry Window will become the load rate in effect. It may be desirable to place the turbine in HOLD to avoid undesirable ramp rates**
Evaluator Note:		**There is no procedural guidance directing when the boration to lower power is required. The crew may elect to perform the boration prior to placing the Turbine in GO.**
	SRO	DIRECTS BOP to start power reduction at 4 MW/Min. May direct initiation of a boration before the power reduction begins.
	BOP	Requests PEER check prior to manipulations of TCS Load Control screen

Time	Position	Applicant's Actions or Behavior
	BOP	a. IF GVPC indicator is TRUE, THEN go to Step 5.c c. Select Ramp Rate Selection, Select button d. Select the desired ramp rate OR Oper Entry on Load Ramp Rate Selection menu • ENTER the desired rate, **NOT** to exceed 5 MW/MIN, in the DEMAND display. (4 DEH Units/minute) e. IF Oper Entry is selected, THEN enter the desired loading rate in the Ramp Rate Entry window and depress Enter. • ENTER the desired rate, NOT to exceed 5 MW/MIN, in the DEMAND display. (4 DEH Units/minute) • DEPRESS the ENTER push-button.
Procedure Note:		**The unloading of the unit can be stopped at any time by selecting the Hold button. The load reduction can be resumed by selecting the Go button**
	BOP	Reduce turbine load as follows: a. Enter desired Target Load (120 MW if shutting down) in Target Entry window and depress Enter b. Select the Go button c. Check that Demand window indication counts down towards desired Target Load d. Check that load ramps towards desired Target Load
Procedure Note:		**Once a raise/lower command button is activated, it will remain in the visually depressed state as an indication the button cannot be activated again for approximately two seconds. After two seconds, command buttons automatically return to their default visual state indicating the button may be activated again**
	BOP	IF AT ANY TIME, a small incremental change of Target Load value (1 or 5 megawatts) is desired, THEN select any of the following buttons: • ▲1 MW • ▲▲5 MW • ▼ 1 MW • ▼▼ 5 MW

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	19	of 83
Event Description:				**Power Reduction**				

Time	Position	Applicant's Actions or Behavior
	BOP	Ensure Generator load is lowering
Evaluator Note:		**As the crew demonstrates a satisfactory load reduction Event 2, Control rods fail to move in AUTO (AOP-001) will become apparent as Tavg/Tref mismatch continues to grow with no rod motion. NO Trigger is required for this event.**

Op Test No.:	NRC	Scenario #	1	Event #	2		Page	20	of	83
Event Description:			Control rods fail to move in AUTO (AOP-001)							

Time	Position	Applicant's Actions or Behavior
Evaluator Note:		**Event 2 (Rods do not move in AUTO) will become apparent when the crew identifies that rod control system signals from power and temperature mismatches have no effect on the rod control system.**
Simulator Operator:		**No triggers are required for this malfunction. The malfunction is pre-loaded**
Evaluator Note:		**The crew may take action to enter AOP-001 prior to receiving any alarms based on monitoring TAVG-TREF deviation indicated by ERFIS points TRC0408Z (median TAVG) and TRC0408b (TREF). The first section of the guide is written to the response of the APP and then AOP-001.**
Indications Available		• **ALB 010-6-4B, RCS TREF/TAVG HIGH-LOW** o NOTE: This alarm is only expected if the Tavg/Tref mismatch reaches the alarm setpoint of +5°F/-5°F • **Tavg/Tref recorder indications**
	RO	Responds to ALB-010-6-4B, RCS TREF/TAVG HIGH-LOW **OR** identifies that the Tavg/Tref indications should have provided a step signal to rod control and has not
	CREW	• CONFIRM alarm using: o Tavg/Tref recorder TR-408 (MCB) o Turbine first stage pressure indicators (PI-446 and PI 447)
Simulator Communicator		**If I&C is contacted to investigate the rod control failure, wait approximately 3 minutes and report back that an I&C technician is at the rod control system and looking for indications of a failure.**

Op Test No.:	NRC	Scenario #	1	Event #	2	Page	21	of	83
Event Description:			**Control rods fail to move in AUTO (AOP-001)**						

Time	Position	Applicant's Actions or Behavior	
	RO	• VERIFY Automatic Functions: ○ None • **IF** there is an indication of a control rod malfunction (MCB and AEP-1), **THEN GO TO** AOP-001, Malfunction of Rod Control and Indication System.	
AOP-001		Malfunction of Rod Control and Indication System	
	SRO	ENTERS and directs actions of AOP-001 Conducts a Crew Update Makes PA announcement for AOP entry	
	RO	PERFORMS immediate actions.	
Immediate Action	RO	CHECK that LESS THAN TWO control rods are dropped.	(YES)
Immediate Action	RO	POSITION Rod Bank Selector Switch to MAN.	
Immediate Action	RO	CHECK Control Bank motion STOPPED.	(YES)
	SRO	READS immediate actions and proceeds to Section 3.3, Failure of a Control Bank To Move. Directs BOP to place Turbine to HOLD if in GO.	
	BOP	Places Turbine to HOLD if in GO.	
	RO	CHECK that AT LEAST ONE of the following conditions is present: • ALB 13-7-1, ROD CONTROL URGENT ALARM, is ALARMED • Control Bank will NOT move • Shutdown Bank will NOT move	(NO) (YES) (NO)

Op Test No.:	NRC	Scenario #	1	Event #	2	Page	22	of	83

Event Description: **Control rods fail to move in AUTO (AOP-001)**

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>RO</td><td>PERFORM the following:
ADJUST Turbine load OR Boron concentration to equalize Tavg with Tref</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>SRO</td><td>DIRECTS RO to equalize Tavg with Tref (Boron or Turbine adjustments) then proceeds to step 6
Directs RO to maintain TAVG within 2°F of Tref per OMM-001 attachment 11.
(NOTE: during a transient such as continuation of the power reduction the control band will change to TAVG within 5°F of Tref)
<table>
<tr><th rowspan="2">Controller</th><th rowspan="2">Control Band</th><th colspan="2">Administrative Limit</th></tr>
<tr><th>Low</th><th>High</th></tr>
<tr><td>Rod Control
Stable Plant</td><td>T Avg within 2° of T Ref</td><td>T Avg Within 10° of T Ref</td><td>T Avg Within 10° of T Ref</td></tr>
<tr><td>Rod Control
Transient Plant</td><td>T Avg within 5° of T Ref</td><td>T Avg Within 10° of T Ref</td><td>T Avg Within 10° of T Ref</td></tr>
</table></td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>RO/ BOP</td><td>Adjusts RCS Boron or turbine load to equalize Tavg with Tref. (may borate based on SRO direction)</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Procedure Note:</td><td>• Surveillance Requirement 4.1.1.1.1.a requires performing a shutdown margin calculation upon detecting an inoperable rod. [C.1]
• It is acceptable to use incore detectors or ERFIS Point from DRPI (or other methods if developed) to meet 4.1.3.1.1 and the Rod Insertion Limit SRs 4.1.3.5 and 4.1.3.6.</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>SRO</td><td>Reviews note</td></tr>
<tr><td></td><td></td><td></td></tr>
</table>

Op Test No.:	NRC	Scenario #	1	Event #	2	Page	23	of	83
Event Description:		**Control rods fail to move in AUTO (AOP-001)**							

Time	Position	Applicant's Actions or Behavior	
	SRO	Refer To the following AND CHECK that ALL control rods are operable: • Tech Spec 3.1.3.1 (does not apply) • Tech Spec 3.1.3.5 (does not apply) **Does not apply in this situation since rod control can be demonstrated operable by rods moving in MANUAL** • Attachment 5, Determination of Control Rod Trippability (can determine rods are trippable)	
	SRO	**DETERMINE** if the Westinghouse Rod Control System Troubleshooting Guidelines should be initiated. (Priority 1 Work Request is required) May contact Reactor Engineering or asks for help when contacting Work Control	
	RO	Determines Tref based on 1st Stage pressure using Curve G-4. He/she may instead use Tref just before the failure to determine the current value of Tref or use OSI-PI plot values.	
Procedure Caution:		If ALB-13-7-1, ROD CONTROL URGENT ALARM, is alarming due to a logic error, resetting the alarm before correcting the cause could result in dropping rods supplied from the affected power cabinet.	
	SRO	Reviews Caution	
	SRO	**CHECK** that ALB-13-7-1, ROD CONTROL URGENT ALARM, is CLEARED.	(YES)
	SRO	**CHECK** automatic AND manual Rod Control FUNCTIONING PROPERLY.	(NO)
Evaluator Note:		**Step 6 of AOP-001 will not be met until rod control has been repaired. Plant shutdown will need to continue with rod control in MANUAL.**	

Op Test No.:	NRC	Scenario #	1	Event #	2	Page	24	of	83
Event Description:		**Control rods fail to move in AUTO (AOP-001)**							

Time	Position	Applicant's Actions or Behavior
	SRO	Completes an Emergent Issue Checklists and contacts WCC for assistance. (WR, LCOTR and Maintenance support)
	CREW	Dispatch operators to rod control cabinets to determine if urgent failure alarms are on locally.
Simulator Communicator		**2-3 minutes after WCC/Engineering or Maintenance has been contacted, report that System Engineer has identified the problem exists in the AUTO circuit only inside PIC-8.**
Evaluator Note:		**If necessary – prompt the crew to continue the plant shutdown by having the Manager of Ops call and direct that the plant shutdown continue with rod control in manual. The SM and AOM-Shift concur that JITT is not required for Maneuvering Plant with a Controller in Manual.** **Crew resumes load reduction.** **SRO asks RO for reactivity addition recommendation.** **BOP places the Turbine in GO to lower load** **With Turbine load lowering cue Simulator Operator to insert Trigger 3** **Event 3: Failure of VCT LT-112 to 100%, which will full divert letdown to RHT (AOP-003)**

Op Test No.:	NRC	Scenario #	1	Event #	3	Page	25	of 83
Event Description:	**Failure of VCT LT-112 to 100%, which will full divert letdown to RHT (AOP-003)**							

Time	Position	Applicant's Actions or Behavior	
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 3 "Failure of VCT LT-112 to 100%, which will full divert letdown to RHT (AOP-003)"**	
Indications Available:		**• ALB 007-5-5, COMPUTER ALARM CHEM & VOL SYSTEMS • 1CS-120 (LCV-115A), Letdown VCT / Holdup Tank, aligns to HUT**	
	RO	Refers to ALB-007-5-5, COMPUTER ALARM CHEM & VOL SYSTEMS	
Evaluator Note:		**Crew may place 1CS-120 (LCV-115A) to the VCT position per AD-OP-ALL-1000.**	
	RO	• **CHECK** instrumentation on MCB associated with alarm point. • **DISPATCH** an operator to check local indications associated with alarming points.	
Simulator Communicator:		**Acknowledge the request to check for local indications of alarming points.**	
	CREW	Identifies entry conditions to AOP-003, Malfunction of Reactor Makeup Control are met	
AOP-003		Malfunction of Reactor Makeup Control	
	SRO	ENTERS and directs actions of AOP-003, Conducts a Crew Update Makes PA announcement for AOP entry	
	RO	Check IA available	(YES)

Op Test No.:	NRC	Scenario #	1	Event #	3	Page	26	of 83
Event Description:	**Failure of VCT LT-112 to 100%, which will full divert letdown to RHT (AOP-003)**							

Time	Position	Applicant's Actions or Behavior
	SRO	CHECK BOTH LT-112 and LT-115 functioning properly. • Determines LK-112 output has failed and goes to Section 3.1, LT-112 or LT-115 Malfunction
	RO	Assesses effects of LT-112 failure (Attachment 1)
Simulator Communicator:		**When directed to report local indication for LT:112, Wait 1 minute then report that local indication is 100%.**
Procedure Note:		An instrument malfunction may manifest itself as a slow drift rather than a "full high" or "full low" failure. Until the instrument has failed fully high or fully low, all steps should be reviewed for applicability periodically, even if not continuously applicable.
	SRO	• **CHECK** that LT-115 is FAILING- (NO)
	SRO	• Determines that LT-112 is failed high and DIRECTS RO to place 1CS-120 (LCV-115A), Letdown VCT / Holdup Tank, to VCT position

	RO	• Determines failure is NOT due to LT-115 and go to Step 8
		• Determines failure caused by LT-112
		  • Monitor VCT level using either:
		    • ERFIS point LCS0115
		    • LT-115
		• Check LT-112 is failing LOW - NO
		• RNO action: Place 1CS-120 (LCV-115A), Letdown VCT / Holdup Tank, to VCT position – (places control to VCT)

Procedure Note:		Normally, VCT level is maintained between 20 and 40% by auto makeup.
	SRO	• Reviews note • DIRECTS RO to **CONTROL** VCT level in AUTO

Op Test No.:	NRC	Scenario # 1	Event # 3	Page 27 of 83
Event Description:	**Failure of VCT LT-112 to 100%, which will full divert letdown to RHT (AOP-003)**			

Time	Position	Applicant's Actions or Behavior
	RO	Maintains VCT level > 5%
Procedure Note:		Lifting leads in the following Step will simulate a low-low level signal from the failed instrument. This allows a valid low-low level signal from the good instrument to initiate emergency makeup.
	SRO	• Reviews note: • Determines LT-112 has failed high and directs Maintenance to lift leads in SSPS for auto switchover to RWST (Step 19)
		• **DIRECT** Maintenance to investigate and repair the instrument malfunction. • Completes an Emergent Issue Checklist and contacts WCC for assistance. (WR, and Maintenance support)
Simulator Communicator:		**Acknowledge requests for assistance.**
Evaluator Note:		**After VCT level has been stabilized, cue Simulator Operator to insert Trigger 4** **Event 4: Trip of running AH-85C fan, standby fails to Auto Start.**

Op Test No.: NRC Scenario # 1 Event # 4		Page 28 of 83
Event Description:		**Trip of running AH-85C fan, standby fails to Auto Start**
Time	Position	Applicant's Actions or Behavior

Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 4** **"Trip of running AH-85C fan, standby fails to Auto Start"**
Indications Available:		• **ALB 027-1-4, DIESEL GEN ELEC EQUIP RM SUP FANS AH-85 LOW FLOW - O/L**
ALB-027	BOP	RESPONDS to alarm on APP-ALB-027-1-4
	BOP	IDENTIFIES the tripped fan, AH-85 1C-SB
	BOP	REPORTS failure of the AH-85 1D-SB standby fan to start
	BOP	STARTS standby AH-85 1D-SB Contacts AO's to investigate breaker failure
Simulator Communicator:		**Breaker failure was overcurrent – IF requested to take breaker to OFF acknowledge the request.** **Simulator Operator – do not take breaker off – not required to continue with scenario**
Evaluator Note:		**(Any Tech Spec evaluation can be conducted with a follow up question after the scenario).**
	SRO	REFER to Tech Specs • T.S 3.8.1.1.b, Action b, items 1-4 One EDG Inoperable Restore EDG to operable within 72 hours Requests BOP to contact AO's to perform OST-1023 Verify required features powered from the Operable EDG are operable • T.S. 3.3.3.5.b, Remote Shutdown System (7 days) OWP-HVAC – Attachment 1, Only AH-85 1C-SB can be credited for supported system operability, since AH-85 1D-SB does not start automatically during an accident.

Op Test No.:	NRC	Scenario #	1	Event #	4	Page	29	of	83

Event Description: **Trip of running AH-85C fan, standby fails to Auto Start**

Time	Position	Applicant's Actions or Behavior
Simulator Communicator:		**Acknowledge the request wait approximately 30 minutes and report back that OST-1023 is complete.**
	SRO	Completes an Emergent Issue Checklist and contacts WCC for assistance. (WR, LCOTR, and Maintenance support)
Simulator Communicator:		**Acknowledge requests for assistance.**
Lead Evaluator:		**Crew will probably place the Turbine on HOLD.** **Once the crew completes starts the standby Air Handler and Tech Specs have been evaluated, cue Simulator Operator to insert Trigger 5** **Event 5: MFW Pump Suction Pressure to CBP controller failure**

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	30	of	83
Event Description:	**MFW Pump Suction Pressure to CBP controller failure**								

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator insert Trigger 5** **MFW Pump Suction Pressure to CBP controller failure**
Available Indications		• **ALB 019 4-1A CNDBSTR PMPS 10% DEVIATION** • **ALB 019 4-1B CNDBSTR PMPS 20% DEVIATION** • **ALB 019 5-5 COMPUTER ALARM CONDENSATE SYSTEM** • **Both Condensate Booster Pump discharge pressures rises to >600 psig** • **FW Pump suction pressure PI-220 lowering** • **Both Condensate Booster Pump controllers PK-2307 and PK-2308 shift from Auto to Manual control** • **SG levels rising**
	Crew	Responds to multiple Condensate Booster Pump alarms and diagnoses that a failure has occurred in the Condensate Booster pump controller that caused both pump M/A stations to go to manual.
Evaluator Note:		**The crew may enter AOP-010 based on the changes to Feedwater flow (may be considered a flow transient but it really is a pressure transient). Page 32 lists the AOP-010 response.** **NOTE: Responding with ONLY AOP-010 guidance and NOT lowering the output of PK-2307 and PK-2308 in accordance with the APP directions will cause all SG levels to continue to rise.**
ALB 019 4-1B	SRO	Directs BOP to manually control PI-2200, FW Pumps Suction Hdr Press, at 430 psig using PK-2307 and PK-2308, Condensate Booster Pump 'A' and 'B' speed controllers in accordance with ALB-019 4-1B.
	BOP	Takes PK-2307 and PK-2308 controllers and lowers the output to reduce PI-2200, FW Pumps Suction Hdr Press to 430 psig Verifies that SG levels are recovering and FRVs are responding correctly

Op Test No.: NRC	Scenario # 1	Event # 5	Page 31 of 83
Event Description:	MFW Pump Suction Pressure to CBP controller failure		

Time	Position	Applicant's Actions or Behavior
	SRO	• VERIFY 1CE-227 (1CE-268), Condensate Booster Pump A (B) Discharge OPEN. • VERIFY the position of 1CE-220 (1CE-261), Condensate Booster Pump A (B) Recirc. • DISPATCH an Operator to perform the following: (1) CHECK system line up using OP-134, Condensate System. (2) CHECK pump operation normal. (3) CHECK for leakage. (4) CHECK normal ∆P at Condensate Polishing Demins, AND BYPASS as necessary. • IF necessary, THEN GO TO AOP-010, Feedwater Malfunctions.
	BOP	• Verifies1CE-227 (1CE-268), Condensate Booster Pump A (B) Discharge is OPEN. • Verifies the position of 1CE-220 (1CE-261), Condensate Booster Pump A (B) Recirc (as seen)
	BOP	Dispatches AO to check for system leakage and other abnormal system indications
Simulator Communicator:		**Acknowledge communications** **After 2-3 minutes report back that nothing is abnormal with the system and no leaks were found**
Evaluator Note:		**If the SRO enters AOP-010 then the crew will perform the immediate actions of the AOP and enter the AOP. The AOP will address SG level issues but will not provide directions for the CBP speed control problems.** **The BOP will have to maintain FW pump suction pressure with both CBP speed controllers in manual for the remainder of this scenario.** **AOP-010 actions are on the next page.**

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	32	of 83
Event Description:		MFW Pump Suction Pressure to CBP controller failure						

Time	Position	Applicant's Actions or Behavior	
AOP-010		**Feedwater Malfunctions**	
	SRO	ENTERS and directs actions of AOP-010 Conducts a Crew Update Makes PA announcement for AOP entry	
Procedure Note:		**Steps 1 through 4 are immediate actions.**	
Immediate Action	BOP	CHECK Feedwater Regulator valves operating properly.	(YES)
Immediate Action	BOP	CHECK ANY Main Feedwater Pump TRIPPED **RNO** GO TO STEP 6	(NO)
	BOP	MAINTAIN ALL of the following: • At least ONE Main Feedwater Pump RUNNING • Main Feedwater flow to ALL Steam Generators • ALL Steam Generator levels greater than 30% Maintains all of the above	
	BOP	CHECK Feedwater Regulator Valves operating properly in AUTO: • Response to SG levels • Valve position indication • Response to feed flow/steam flow mismatch	(YES)
Procedure Note:		Inability to monitor one or more Safety System Parameters concurrent with a turbine runback of greater than 25%, requires a change of event classification per the HNP Emergency Plan. [C.2, C.3]	
	BOP	CHECK turbine runs back less than 25% turbine load	YES

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	33	of	83
Event Description:		**MFW Pump Suction Pressure to CBP controller failure**							

Time	Position	Applicant's Actions or Behavior	
Procedure Note:		A feedwater train consists of a Condensate Pump, Condensate Booster Pump and Main Feedwater Pump.	
	SRO	GO TO the applicable section: EVENT: All Condensate/Feedwater flow malfunctions (other than pump trips) Section 3.1 Page 10	
	BOP	CHECK the following Recirc and Dump Valves operating properly in MODU:	
		• Main Feedwater Pumps	(YES)
		• Condensate Booster Pumps	(YES)
		• Condensate Pumps	(YES)
		• 1CE-293, Condensate Recirc	(YES)
		• 1CE-142, Condensate Dump To CST Isolation Valve (SLB-4/7-1)	(YES)
	BOP	CHECK the Condensate and Feedwater System INTACT.	
Procedure Note:		Pumps should be stopped in the order of higher to lower pressure. (To stop a Condensate Pump, stop a Main Feedwater Pump followed by a Condensate Booster Pump and then the Condensate Pump.)	
	BOP	CHECK pumps for NORMAL OPERATION.	(YES)
	SRO	NOTIFY Load Dispatcher of ANY load limitations. **(No load limitations so Dispatcher will not be called)**	
	SRO	CHECK Reactor thermal power changed by less than 15% in any one hour period.	(NO)
	SRO	EXIT this procedure.	

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	34	of	83
Event Description:		**MFW Pump Suction Pressure to CBP controller failure**							

Time	Position	Applicant's Actions or Behavior
	SRO	Completes an Emergent Issue Checklist and contacts WCC for assistance. (WR, LCOTR, and Maintenance support)
Lead Evaluator:		**Once the plant has stabilized, cue Simulator Operator to insert Trigger 6** **Event 6: 'C' Steam Generator Tube Leak (AOP-016)**

Op Test No.: NRC	Scenario # 1	Event # 6	Page	35	of	83	
Event Description:	**'C' Steam Generator Tube Leak (AOP-016)**						

Time	Position	Applicant's Actions or Behavior	
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 6** **"'C' Steam Generator Tube Leak (AOP-016)"**	
Indications Available:		• **Charging Flow rising** • **VCT Level lowering** • **Pressurizer Level and Pressure lowering** • **'C' MSL Rad monitor**	
	CREW	Identifies entry conditions to AOP-016, Excessive Primary Plant Leakage are met	
AOP-016		Excessive Primary Plant Leakage	
	SRO	ENTERS and directs actions of AOP-016, Conducts a Crew Update Makes PA announcement for AOP entry	
Procedure Note:		**This procedure contains no immediate actions.**	
	RO	CHECK RHR in operation	(NO)
	SRO	REFER TO PEP-110, Emergency Classification And Protective Action Recommendations, AND ENTER the EAL Matrix.	
	RO	CHECK RCS leakage within VCT makeup capability May report that the leak is exceeding Tech Spec SG leakage.	(YES)
Procedure Note:		If CSIP suction is re-aligned to the RWST, negative reactivity addition should be anticipated.	
	RO	MAINTAIN VCT level GREATER THAN 5% GO TO STEP 10	(YES)

Op Test No.:	NRC	Scenario #	1	Event #	6	Page	36 of 83
Event Description:	'C' Steam Generator Tube Leak (AOP-016)						

Time	Position	Applicant's Actions or Behavior	
	SRO	• CHECK valid CNMT Ventilation Isolation monitors • (REM-3561A, B, C and D) ALARM CLEAR • CHECK RM 3502A, RCS Leak Detection Radiation Monitor, ALARM CLEAR • CHECK ALL valid Area Radiation Monitors ALARM CLEAR • CHECK valid Stack Monitors ALARM CLEAR	(YES) (YES) (YES) (YES)
	SRO	DETERMINE if unnecessary personnel should be evacuated from affected areas, as follows: CHECK that a valid RMS Secondary Monitor HIGH ALARM indicates a SG tube leak may exist.	
	BOP	SOUND local evacuation alarm. ANNOUNCE on the PA: "Attention all personnel. High radiation levels may exist in portions of the power block due to SG tube leakage. Unnecessary personnel evacuate the RAB and Turbine Building, including the Steam Tunnel. Further announcements will be made as surveys are performed."	
	BOP	NOTIFY Chemistry to stop any primary sampling activities.	
Simulator Communicator:		**Acknowledge request to stop primary sampling activities.**	
Procedure Note:		• The following qualitative flow balance is to quickly determine if RCS leakage exceeds Tech Spec limits, EAL classification thresholds, or RCS makeup capability. • RCS influent and effluent flow rates are compared and PRZ level rate of change is used to determine the RCS flow balance.	

Op Test No.: NRC	Scenario # 1 Event # 6	Page 37 of 83
Event Description:		**'C' Steam Generator Tube Leak (AOP-016)**
Time	Position	Applicant's Actions or Behavior

	RO	PERFORM a qualitative RCS flow balance, as follows: a. ESTIMATE leak rate considering the following parameters: • PRZ level rate of change (~55 gal/% at 653°F) • Charging flow • Total seal injection flow • Letdown flow • Total seal return flow Reports estimate to SRO of ~ 30 gpm
		b. OPERATE the following letdown orifice valves as necessary to maintain charging flow on scale: • 1CS-7, 45 gpm Letdown Orifice A • 1CS-8, 60 gpm Letdown Orifice B • 1CS-9, 60 gpm Letdown Orifice C (No changes required)
Procedure Note:		Performance of surveillance tests to determine if leakage exceeds Tech Spec limits, or to more accurately quantify leakage is up to CRS discretion.
	SRO	Determines that more accurate quantification is not needed due to excessive leakage indications present.
Evaluator Note:		**Any Tech Spec evaluation can be conducted as a follow up question after the scenario.**
	SRO	EVALUATE RCS leakage (refer to Tech Spec 3.4.6.2).
		Reviews Reactor Coolant System TS 3.4.6.2 Reactor Coolant System operational leakage shall be limited to: c. 150 gallons per day primary-to-secondary leakage through any one steam generator. ACTION **a.** - With any PRESSURE BOUNDARY LEAKAGE or with primary-to-secondary leakage not within limits, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Op Test No.:	NRC	Scenario #	1	Event #	6	Page	38 of 83
Event Description:		'C' Steam Generator Tube Leak (AOP-016)					

Time	Position	Applicant's Actions or Behavior
	SRO	DETERMINE leak location from one or more of the following: MCB indications and Valid Radiation Monitors
	BOP	NOTIFY Health Physics of the following: a. Leak location: • Source inside or outside CNMT • To closed system, SG or to atmosphere b. Applicable radiation levels.
Simulator Communicator:		**Acknowledge communications**
	SRO	WHEN leakage location has been determined, THEN PERFORM the applicable Attachment: Primary-to-Secondary Attachment 1 page 13
	BOP	NOTIFY Chemistry to implement CRC-804, Primary-To-Secondary Leak Rate Monitoring, to accomplish the following: a. NOTIFY the MCR as soon as the leaking SG has been determined. b. NOTIFY the MCR when the following results are obtained: • quantify leak rate • quantify leak rate trend
Simulator Communicator:		**Acknowledge communications**
Procedure Note:		For a known leak rate greater than 100 gpd (PSAL 3 threshold), the CRS may direct performance of Attachments 9, 10 and 11 while the remaining steps of Attachment 1 are being completed.

Op Test No.: NRC	Scenario # 1	Event # 6	Page 39 of 83
Event Description:	'C' Steam Generator Tube Leak (AOP-016)		

Time	Position	Applicant's Actions or Behavior
	SRO	CHECK known leak rate is LESS THAN 100 gpd (0.0694 gpm).- NO leak is > 100 gpm – GO TO STEP 4
	SRO	DETERMINE leaking SG(s) using the following information: • Individual SGBD samples • Main Steam Line radiation monitor levels • Local surveys of SGBD lines Determines leak is from 'C' SG from various indication sources
	SRO	CHECK the following valid radiation monitors ALARM CLEAR: • RM-01MS-3593 SB, Main Steam Line C (DICSP Grids 5, 6) • REM-01TV-3534, Condenser Vacuum Pump Effluent (DICSP Grid 2) • REM-01BD-3527, Steam Generator Blowdown (DICSP Grid 2) • RM-01TV-3536-1, Turbine Building Vent Stack Effluent (DICSP Grids 2, 5, 6) NO not clear
	BOP	PERFORM the following: a. DIRECT Health Physics to survey the following outside the RCA: • SG Blowdown piping • Vicinity of Main Steam piping b. IF ANY valid monitor is in HIGH ALARM, THEN: (1) NOTIFY HP to evaluate the alarm (refer to HPP-780, Radiation Monitoring Systems Operator's Manual). (2) SOUND the local evacuation alarm. (3) ANNOUNCE evacuation of the following areas: • Steam Tunnel • SG PORVs/SG Safety valves area • Turbine Building 314' elevation (4) REPEAT sounding the local evacuation alarm AND the announcement.

Op Test No.:	NRC	Scenario #	1	Event #	6	Page	40	of 83
Event Description:		**'C' Steam Generator Tube Leak (AOP-016)**						

Time	Position	Applicant's Actions or Behavior
	SRO	(5) IF ANY valid Main Steam Line Monitor is in HIGH ALARM, THEN PERFORM an Offsite Dose Calculation (Refer to PEP-340, Dose Assessment). - Refers to the STA for this assessment.
	SRO	CHECK BOTH of the following: • Turbine Building Vent Stack radiation monitor HIGH ALARM CLEAR • SG tube leakage is less than Tech Spec limits. NO – RNO actions: START CVPETS (refer to OP-133, Main Condenser Air Removal System).
	BOP	Contacts TB AO to Start CVPETS in accordance with OP-133
Simulator Communicator:		**Acknowledge communications to start CVPETS**
Simulator Operator:		**Perform the following actions from Sim Diagram CVP01 to operate start the CVPETS 'A' fan:** **Start CVPETS 'A' fan modify rf cnd035 ON, then have Communicator report back when completed**
Procedure Note:		B train Aux Condensate Equipment is in long term shutdown per EC 264640.

Op Test No.: NRC	Scenario # 1	Event # 6	Page 41 of 83
Event Description:	'C' Steam Generator Tube Leak (AOP-016)		

Time	Position	Applicant's Actions or Behavior
	SRO	CHECK valid Aux Steam Condensate radiation monitors ALARM CLEAR: • REM-21AC-3525, RAB Auxiliary Steam Condensate (DICSP Grid 1) • REM-21AC-3543A, AUX Steam Condensate Tank Pump Discharge A (DICSP Grid 4) YES - clear
	BOP	NOTIFY Chemistry to sample the Auxiliary Steam System for activity.
Simulator Communicator:		**Acknowledge communications**
	SRO	CHECK Chemistry reports Auxiliary Steam System activity is satisfactory. (No reports yet – continues with procedure)
Procedure Note:		• For initial leakage reports, where no previous leakage existed, leakage should be assumed to have changed from zero to the current value in the last hour. • The monitoring requirements of Step 3 become optional if Step 10 directs performance of Attachment 9, 10, or 11.
	SRO	PERFORM the required actions based on the following: Action Level 3 Greater than or equal to 150 + Greater than or equal to 30 = • Verify sustained rate of change above 30 gpd/hr (not followed by a reduction - spike) • Perform Attachment 11 • Isolate SG Blowdown from leaking SG (Shut 1BD-56) • Reduce power to 50% within 1 hour • Be in Mode 3 within the next 2 hours (3 hours total time) • Be in Mode 5 within the next 30 hours (33 hours total time)

Op Test No.:	NRC	Scenario #	1	Event #	6	Page	42 of	83
Event Description:		**'C' Steam Generator Tube Leak (AOP-016)**						

Time	Position	Applicant's Actions or Behavior
	SRO	Determines that SG leakage will require the unit power level to be reduced to <50% within 1 hour and Mode 3 in next 2 • Requires AOP-038 entry to accomplish these time limits per Attachment 11 step 7 RNO
	SRO	WHEN required actions are complete OR leaking SG(s) are cooled down and depressurized to Mode 5, THEN: a. CONSULT plant operations staff concerning plant conditions needed to support recovery efforts. b. EXIT this procedure.
	SRO	Informs crew that they are transitioning to AOP-038
AOP-038		Rapid Downpower
	SRO	Enters AOP-038, RAPID DOWNPOWER Makes PA announcement Conducts a crew brief
Simulator Communicator:		**The crew may make calls to notify plant management in accordance with AD-OP-ALL-1000, Section 5.5.13 before or during the power reduction. Acknowledge and request a report from the MCR when more information becomes available.**

Op Test No.:	NRC	Scenario #	1	Event #	6	Page	43 of	83
Event Description:	**'C' Steam Generator Tube Leak (AOP-016)**							

Time	Position	Applicant's Actions or Behavior
Procedure Note:		• This procedure contains no immediate actions. • Steps may be performed simultaneously or out of sequence at the discretion of the Shift Manager. • If the ASI System is supplying RCP seal injection and no CSIP is available, boration is accomplished by the operation of the ASI pump and is not under control of the operator. Steps that perform boration or dilution cannot be performed and should be marked NA. Turbine load should be reduced at a rate between 5 MW/MIN (EOL) and 10 MW/MIN (BOL). Target rod heights as a function of power in Attachment 2 remain valid.
	SRO	ENTER the EAL Matrix (Refer to the following): [C.1] • AD-EP-ALL-0101, Emergency Classification • AD-EP-ALL-0109, Offsite Protective Action Recommendations
	BOP	NOTIFY Load Dispatcher that the Unit is reducing load.
Procedure Note:		Boration of the RCS commences at Step 9.
	RO	DETERMINE required boric acid addition as follows: CHECK BOTH of the following conditions exist: • Reactor power is 100% • Target power level is provided in OPT-1536, Routine Reactivity Data Calculation. [C.3]
Evaluator Note:		**AOP-038 Attachment 2 is located in this guide on page 77.**
		NO – RNO actions: OBTAIN values from Attachment 2, Gallons of Boric Acid/Target Rod Height Required for Power Reduction. [C.3] • Desired Boration _______ gal Target Rod height (D Bank) _______

Op Test No.:	NRC	Scenario #	1	Event #	6	Page	44	of	83
Event Description:			**'C' Steam Generator Tube Leak (AOP-016)**						

Time	Position	Applicant's Actions or Behavior
Procedure Note:		• If load reduction rates in excess of 45 MW/min are required, the Unit should be tripped. • GVPC is the preferred method of Load Control. Megawatt Control is normally used only during GV and TV testing. • If Oper Entry is selected with the Turbine in GO, the value currently in the Ramp Rate Entry Window will become the load rate in effect. It may be desirable to place the turbine in HOLD to avoid undesirable ramp rates.
	BOP	PERFORM the following on TCS Load Control screen, Load Control section: a. CHECK GVPC indicator is TRUE – YES b. SELECT Ramp Rate Selection, Select button c. CHECK desired ramp rate is listed on Load Ramp Rate Selection – YES NO – RNO actions: d. SELECT desired ramp rate (NOT to exceed 45 MW/min). **(Should select 25 MW/min)** e. ENTER desired load (120 MW if shutting down) in Target Entry window. **(Should be previously select for 120 MW)** f. DEPRESS Enter
	RO	CHECK Rod Control in AUTO (NO – auto is failed) • MANUALLY POSITION Control Rods to maintain Tavg within 5°F of Tref.
	RO	ENERGIZE ALL available PRZ Backup heaters. (ALL ON)
	SRO	DISCUSS Attachment 3, Reactivity Brief, with the MCR staff.
Procedure Note:		The MW output indication is displayed on the TCS Turbine Load Control screen. An accurate indication of Main Generator output can also be obtained from ERFIS point JEE1568B (Gross MWe).

Op Test No.: NRC	Scenario # 1	Event # 6	Page	45	of	83	
Event Description:	**'C' Steam Generator Tube Leak (AOP-016)**						

Time	Position	Applicant's Actions or Behavior
	BOP	COMMENCE turbine load reduction at the TCS Load Control screen: • SELECT GO pushbutton. • CHECK that Demand value counts-down to Target Load value.
Procedure Note:		• To prevent over-boration, only the amount of boron required to reduce power to the desired power level should be added. If the situation merits that the downpower may have to be halted for any substantial time (>2 hours) at lower powers prior to taking the plant off-line, the effects of Xenon and changes in feed flow should be considered. Reactivity control may become challenging if boron manipulations are not appropriately implemented. • Adjustments should be made to boric acid flow based on actual core/rod response.
	RO	COMMENCE RCS boration as required to maintain Control Rods above the Rod Insertion Limit (ROD Manual Sect 2.2).
Evaluator Note:		**The following boration steps of OP-107.01 are provided for evaluator use. They are not in AOP-038.** **Section 8.7 is provided below.**
OP-107.01		CVCS Boration, Dilution, And Chemistry Control Section 8.7, Rapid Addition of Boric Acid to the RCS

Op Test No.: NRC	Scenario # 1	Event # 6	Page 46 of 83
Event Description:	**'C' Steam Generator Tube Leak (AOP-016)**		

Time	Position	Applicant's Actions or Behavior
Procedure Note:		• If performing a rapid shutdown of the plant per AOP-038, the following calculation does not have to be completed before boration begins, but should be completed before half of the estimated (or before 500 gallons whichever is less) boron addition has been dispensed. • Reactivity Evolution category to be determined by the CRS. • If an RCS leak or SGTL is occurring and an Auto Makeup is in progress, it may be necessary to take RMUW control to stop in order to avoid flow deviations on RMUW system while performing the boration.
	RO	IF it is desired to stop Auto Makeup due to RCS leakage, THEN PERFORM the following: a. PLACE RMW CONTROL switch to stop. b. CHECK the green light is lit on the RMW Control switch.
		DETERMINE the volume of boric acid necessary to achieve the required RCS boron concentration. • Required gallons of Boric Acid _____________Gal. ENSURE the backup Boric Acid Transfer Pump control switch is in STOP.
Procedure Note:		• Required boration flow rate of greater than 45 gpm, is best achieved by using Step 8.7.2.4, 1CS 278 SB, EMERGENCY BORIC ACID ADDITION. • Required boration flow rate of less than 45 gpm, is best achieved by using Step 8.7.2.5, 1CS 283, BORIC ACID TO BORIC ACID BLENDER FCV-113A, and 1CS 156, MAKE UP TO CSIP SUCTION FCV 113B.

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Event Description: **'C' Steam Generator Tube Leak (AOP-016)**

Time	Position	Applicant's Actions or Behavior
	RO	IF using 1CS-278 SB, EMERGENCY BORIC ACID ADDITION for Boric Acid addition, THEN PERFORM the following: a. RECORD the initial BAT level for backup calculation of Boric Acid addition. - Initial BAT level: _____________ % b. START the Boric Acid Transfer Pump aligned for Auto Makeup (switch in AUTO) by placing the control switch to START. c. SIMULTANEOUSLY PERFORM the following: - OPEN 1CS-278 SB, EMERGENCY BORIC ACID ADDITION. - MARK the START time. Time 1CS-278 opened. Time _____________ d. RECORD the Boric Acid flowrate from FI-110. - Boric Acid flowrate (FI-110) _____________ Gpm e. CALCULATE the amount of time in minutes it will take to deliver the required amount of Boric Acid. - Required gallons BA / BA flowrate = Time ________ ÷ ________ = ________ min 8.7.2.2 8.7.2.4.d
		f. CONTROL charging and letdown to maintain normal PRZ and VCT levels. g. CALCULATE the final BAT level for the required amount of Boric Acid being added. - Initial BAT Lvl % – [(Required gallons BA) / (330 gal/%)] = Final BAT Lvl % ________ - (________ / 330) = ________ % 8.7.2.4.a 8.7.2.2
Procedure Note:		- Boration flow may be interrupted as needed by cycling 1CS-278, while maintaining the total boration time calculated in Step 8.7.2.4.e

Op Test No.: NRC	Scenario # 1	Event # 6	Page	48 of 83
Event Description:		**'C' Steam Generator Tube Leak (AOP-016)**		

Time	Position	Applicant's Actions or Behavior	
	RO	h. WHEN the calculated amount of time has elapsed, THEN SIMULTANEOUSLY: • SHUT 1CS-278 SB. • MARK the STOP time. Time 1CS-278 shut. Time ____________ i. ENSURE, using calculated final BAT level, that the required amount of Boric Acid has been dispensed	
Procedure Note:		Boration flow may be interrupted as needed by cycling 1CS-278, while maintaining the total boration time calculated. During makeup operations following an alternate dilution, approximately 10 gallons of dilution should be expected due to dilution water remaining in the primary makeup lines.	
	RO	REQUEST Chemistry to sample the RCS boron concentration. PLACE Reactor Makeup in Auto per Section 5.1.	
AOP-038		Rapid Downpower Actions - Continued (step 11)	
	BOP	ENSURE Generator load AND Reactor power LOWERING.	
	BOP	MAINTAIN Generator reactive load (VARs) within guidelines.	
Procedure Note:		Opening 3A and 3B Feedwater Heater vents helps minimize water hammer in 3A and 3B Feedwater Heaters.	
	RO	CHECK Tavg within 5°F of Tref	(YES)
	CREW	NOTIFY Chemistry of the following: Reactor power change will exceed 15% in a one hour period.	

Op Test No.: NRC	Scenario # 1	Event # 6	Page	49 of 83		
Event Description:	'C' Steam Generator Tube Leak (AOP-016)					

Time	Position	Applicant's Actions or Behavior	
	SRO	DIRECT Chemistry to initiate surveillances specified in the applicable sections of the following: • RST-204, Reactor Coolant System Chemistry and Radiochemistry Surveillance • RST-211, Gaseous Effluent Radiochemistry Surveillance	
	SRO	• CHECK that a planned load reduction will take the Unit to Turbine shutdown	(YES)
	SRO	• DISPATCH an operator to start the Auxiliary Boiler using OP-130.02, • Auxiliary Boiler and Fuel Oil. • NOTIFY Radwaste Control Room to be prepared for the increased water processing requirements due to boration.	
	CREW	CHECK Power level at the target value	(NO)
Examiner Note:		**With AOP-038 in progress, cue Simulator Operator to insert Trigger 7** **Event 7: 'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-0 and EOP-E-3).**	

Op Test No.:	NRC	Scenario #	1	Event #	7	Page	50	of	83

Event Description: **'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-0)**

Time	Position	Applicant's Actions or Behavior

Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 7 "'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-0 and EOP-E-3)"**
Indications Available:		• **ALB-009-2-2, Pressurizer Control Low Level Deviation** • **ALB-010-4-5 Rad Monitor System Trouble** • **Pressurizer Level and Pressure lowering** • **Charging Flow rising** • **VCT Level lowering** • **'C' MSL Rad monitor** • **'C' SG level rising**
	CREW	Identifies re-entry conditions to AOP-016, Excessive Primary Plant Leakage are met
AOP-016		Excessive Primary Plant Leakage
	SRO	RE-ENTERS and directs continuous action step 4 of AOP-016, Conducts a Crew Update
	RO	CHECK RCS leakage within VCT makeup capability — NO
		NO – RNO actions PERFORM the following: a. TRIP the Reactor, AND GO TO EOP-E-0. (Perform RNO substeps 4.b. and 4.c as time permits) **(Actuates Manually Rx Trip using MCB switch)**
EOP-E-0		Reactor Trip Or Safety Injection
	SRO	Enters EOP-E-0 Holds crew update
	RO/BOP	Performs E-0 Immediate Actions.

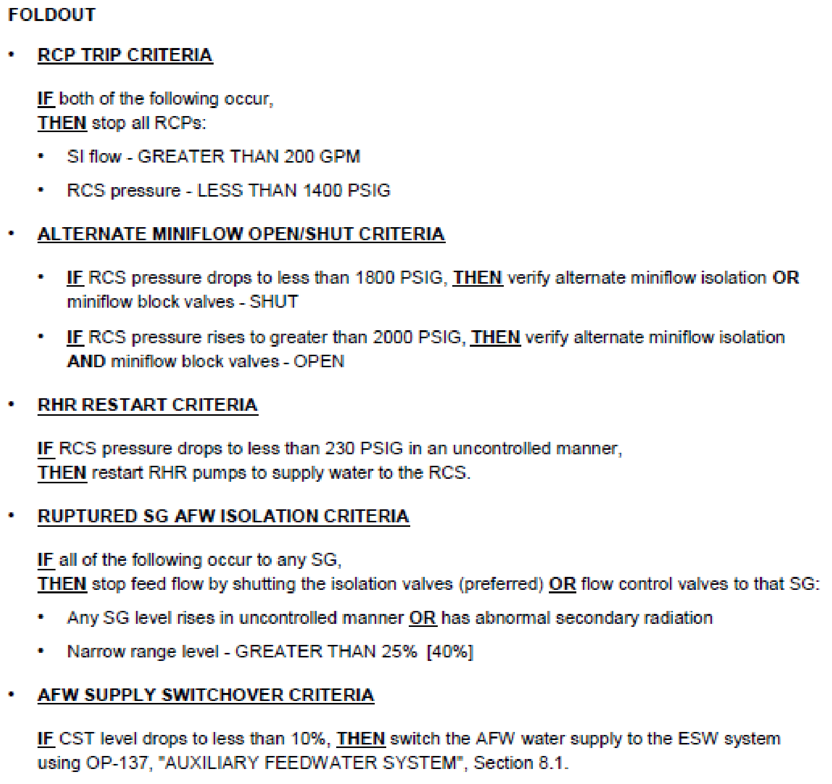
Op Test No.:	NRC	Scenario #	1	Event #	7	Page	51	of 83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-0)**							

Time	Position	Applicant's Actions or Behavior	
Immediate Actions	RO	VERIFY Reactor Trip: REACTOR TRIP CONFIRMATION Reactor Trip AND Bypass BKRs – OPEN Rod Bottom Lights (Zero Steps) – LIT Neutron Flux – DROPPING	YES YES YES
Immediate Actions	BOP	Check Turbine Trip – ALL THROTTLE VALVES SHUT TURB STOP VLV 1 – TSLB-2-11-1 TURB STOP VLV 2 – TSLB-2-11-2 TURB STOP VLV 3 – TSLB-2-11-3 TURB STOP VLV 4 – TSLB-2-11-4	YES YES YES YES
AOP-016		Excessive Primary Plant Leakage	
Procedure Note:		If SI Actuation is required, the Reactor and Turbine should be verified tripped in EOP-E-0 before manually actuating SI.	
	RO	b. MANUALLY INITIATE Safety Injection. [C.1] **(Actuates Manually Safety Injection using MCB switch)**	
	SRO	c. EXIT this procedure.	
EOP-E-0	SRO	E-0, Reactor Trip Or Safety Injection	

Op Test No.:	NRC	Scenario #	1	Event #	7	Page	52	of	83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-0)**								

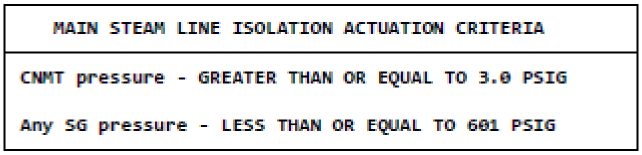
Time	Position	Applicant's Actions or Behavior	
Immediate Actions	BOP	Perform The Following: • AC Emergency Buses – AT LEAST ONE ENERGIZED • AC Emergency Buses – BOTH ENERGIZED	YES YES
Immediate Actions	RO	Safety Injection – ACTUCATED (BOTH TRAINS) `BPLP 4-1,"SI ACTUATED" - LIT (CONTINUOUSLY)`	YES
Procedure Note:		Steps 1 through 4 are immediate action steps Foldout applies. (Immediate actions should be completed prior implementing Foldout Page items.)	
	SRO	Reviews Foldout page	

Op Test No.:	NRC	Scenario #	1	Event #	7	Page	53	of 83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-0)**							

Time	Position	Applicant's Actions or Behavior	
Evaluator Note:		**FOLDOUT** • **RCP TRIP CRITERIA** **IF** both of the following occur, **THEN** stop all RCPs: • SI flow - GREATER THAN 200 GPM • RCS pressure - LESS THAN 1400 PSIG • **ALTERNATE MINIFLOW OPEN/SHUT CRITERIA** • **IF** RCS pressure drops to less than 1800 PSIG, **THEN** verify alternate miniflow isolation **OR** miniflow block valves - SHUT • **IF** RCS pressure rises to greater than 2000 PSIG, **THEN** verify alternate miniflow isolation **AND** miniflow block valves - OPEN • **RHR RESTART CRITERIA** **IF** RCS pressure drops to less than 230 PSIG in an uncontrolled manner, **THEN** restart RHR pumps to supply water to the RCS. • **RUPTURED SG AFW ISOLATION CRITERIA** **IF** all of the following occur to any SG, **THEN** stop feed flow by shutting the isolation valves (preferred) **OR** flow control valves to that SG: • Any SG level rises in uncontrolled manner **OR** has abnormal secondary radiation • Narrow range level - GREATER THAN 25% [40%] • **AFW SUPPLY SWITCHOVER CRITERIA** **IF** CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1. **If ‘C’ SG previously identified as the Ruptured Generator due to rising SG level, then Ruptured SG AFW Isolation foldout will apply**	
	SRO	Assigns Foldout items: RCP Trip Criteria, Alternate Miniflow Open/Shut Criteria, RHR Restart Criteria, Ruptured SG AFW Isolation criteria, AFW Supply Switchover Criteria Directs Shift Manager to Evaluate EAL Matrix	
	SRO	Evaluate EAL Matrix (Refer to PEP-110)	
	RO	Verify CSIPs – ALL RUNNING	(YES)
	RO	Verify RHR pumps – ALL RUNNING	(YES)

Op Test No.:	NRC	Scenario #	1	Event #	8	Page	54 of 83

Event Description: **Relay failure on resultant SI signal K603A**

Time	Position	Applicant's Actions or Behavior	
	RO	Safety Injection flow – GREATER THAN 200 GPM	NO
(Event 8) Critical Task #1	RO	NO – RNO actions Perform the following: a) Ensure high head safety injection alignment: (1) CSIP suction from RWST valves - OPEN (2) VCT outlet valves – SHUT (3) Charging line isolation valves - SHUT **(Shut 1CS-238 manually)** (4) BIT outlet valves - OPEN **(Open 1SI-4 manually)** Critical to manually align at least one high head ECCS pump flow path before RCS temperature reaches 730°F and RVLIS Full Range Level lowers below 39%.	YES YES NO NO
	RO	RCS pressure – LESS THAN 230 PSIG	(NO)
	SRO	RNO: GO TO Step 12.	
	BOP	MAIN Steam isolation – ACTUATED.	(NO)
	SRO	RNO: Perform the following:	
	BOP	• Check MAIN Steam isolation – REQUIRED  MAIN STEAM LINE ISOLATION ACTUATION CRITERIA CNMT pressure - GREATER THAN OR EQUAL TO 3.0 PSIG Any SG pressure - LESS THAN OR EQUAL TO 601 PSIG • IF Main Steam Isolation is NOT required , THEN GO TO Step 16.	(NO)

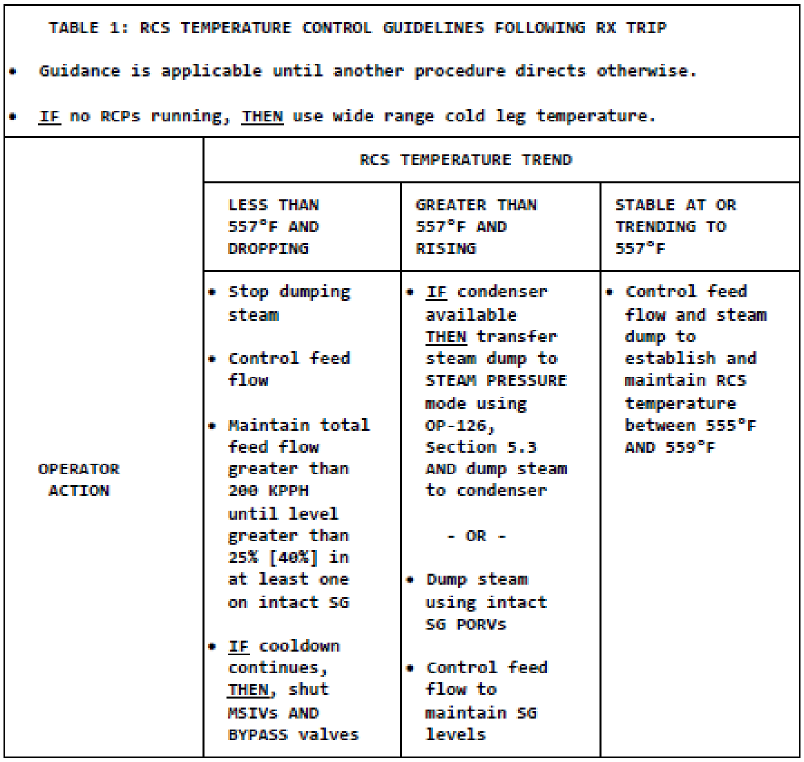
Op Test No.:	NRC	Scenario #	1	Event #	7	Page	55	of 83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-0)**							

Time	Position	Applicant's Actions or Behavior	
	RO	CHECK CNMT Pressure – HAS REMAINED LESS THAN 10 PSIG	(YES)
	BOP	Verify AFW flow – AT LEAST 200 KPPH ESTABLISHED	(YES)
	BOP	Sequencer Load Block 9 (Manual Loading Permissive) - ACTUATED (BOTH TRAINS)	(YES)
	BOP	Energize AC buses 1A1 AND 1B1	
Evaluator Note:		**The RO will perform all board actions until the BOP completes Attachment 3. The BOP is permitted to properly align plant equipment in accordance with Attachment 3 without SRO approval. The Scenario Guide still identifies tasks by board position because the time frame for completion of Attachment 3 is not predictable.** **To follow BOP actions E-0 Attachment 3 is located in the back of this guide.**	
	BOP	VERIFY Alignment of Components From Actuation of ESFAS Signals Using Attachment 3, "Safeguards Actuation Verification", While Continuing with this Procedure.	
	BOP	Directs TB AO – Place air compressor 1A and 1B in the Local Control mode. Directs RAB AO – Locally unlock and turn on the breakers for the CSIP Suction and Discharge Cross-Connect valves	

Op Test No.: NRC	Scenario # 1	Event # 8/9	Page	56	of	83	
Event Description:	**Failure of Control Room Isolation to actuate 'B' ESW Pump fails to auto start on SI**						

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**When contacted to place A/B air compressors in Local Control mode, run CAEP :\air\ACs_to_local.txt.**
Simulator Communicator:		**When CAEP is complete, report that the air compressors are running in local control mode.**
Simulator Operator:		**When contacted to Unlock and Turn ON the breakers for the CSIP suction and discharge cross-connect valves, run CAEP *:*\cvc\E-0 Att 2 CSIP suct & disc valve power.txt.**
Simulator Communicator:		**When the CAEP is complete, report task to the MCR.**
Event 9	RO	Ensure All ESW AND ESW Booster Pumps – RUNNING Identifies that the 'B' ESW Pump is NOT running and manually starts pump.
Event 8	BOP	Ensure Control Room Area Ventilation - Main Control Room Aligned For Emergency Operation (Refer to OMM-004, "Post Trip/Safeguards Actuation Review", Attachment 5, Sheets 1 and 2, Sections for Main Control Board, SLB-5 and SLB-6.) Identifies that the Control Room Area Ventilation is NOT aligned for Emergency Operation and aligns the ventilation system correctly. **OMM-004 Attachment 5 is located in the back of this guide on page 79**.
	BOP	The following items should be completed due to Control Room Ventilation not being aligned: Opens CZ-D66 Starts R2 A-SA fan (Emergency Filtration) Stops E9A fan (Normal Exhaust) Opens Battery Room A Return Dampers AC-D4

<table>
<tr><td colspan="3">Op Test No.: <u>NRC</u> Scenario # 1 Event # 7 Page <u>57</u> of <u>83</u>
Event Description: **'C' Steam Generator Tube Rupture of 250 gpm (E-0) Continued**</td></tr>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>
Stabilize AND maintain temperature between 555°F AND 559°F using Table 1.



<table>
<tr><td colspan="4">TABLE 1: RCS TEMPERATURE CONTROL GUIDELINES FOLLOWING RX TRIP
• Guidance is applicable until another procedure directs otherwise.
• <u>IF</u> no RCPs running, <u>THEN</u> use wide range cold leg temperature.</td></tr>
<tr><td></td><td colspan="3">RCS TEMPERATURE TREND</td></tr>
<tr><td></td><td>LESS THAN 557°F AND DROPPING</td><td>GREATER THAN 557°F AND RISING</td><td>STABLE AT OR TRENDING TO 557°F</td></tr>
<tr><td>OPERATOR ACTION</td><td>• Stop dumping steam
• Control feed flow
• Maintain total feed flow greater than 200 KPPH until level greater than 25% [40%] in at least one on intact SG
• <u>IF</u> cooldown continues, <u>THEN</u>, shut MSIVs AND BYPASS valves</td><td>• <u>IF</u> condenser available <u>THEN</u> transfer steam dump to STEAM PRESSURE mode using OP-126, Section 5.3 AND dump steam to condenser
- OR -
• Dump steam using intact SG PORVs
• Control feed flow to maintain SG levels</td><td>• Control feed flow and steam dump to establish and maintain RCS temperature between 555°F AND 559°F</td></tr>
</table>

- Control feed flow and steam dump to stabilize temperature between 555˚F AND 559˚F
</td></tr>
</table>

Op Test No.:	NRC	Scenario #	1	Event #	7	Page	58 of 83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (E-0) Continued**						

Time	Position	Applicant's Actions or Behavior	
	RO	PRZ PORVs – SHUT	(YES)
	RO	PRZ spray valves – SHUT	(YES)
	RO	PRZ PORV block valves – AT LEAST ONE OPEN (All OPEN)	(YES)
	BOP/SRO	ANY SG pressures – DROPPING IN AN UNCONTROLLED MANNER <u>OR</u> COMPLETELY DEPRESSURIZED GO To Step 27	(NO) (NO)
	BOP/SRO	ANY SG ABNORMAL RADIATION <u>OR</u> UNCONTROLLED LEVEL RISE Crew identifies 'C' SG	(YES) (YES)
	SRO	Check Feed Flow to Ruptured SG(s) – ISOLATED (The crew should have isolated the 'C' SG Feed Flow earlier utilizing the Ruptured SG AFW Isolation Criteria Foldout)	(YES)
	SRO	GO TO E-3, "STEAM GENERATOR TUBE RUPTURE", Step 1.	

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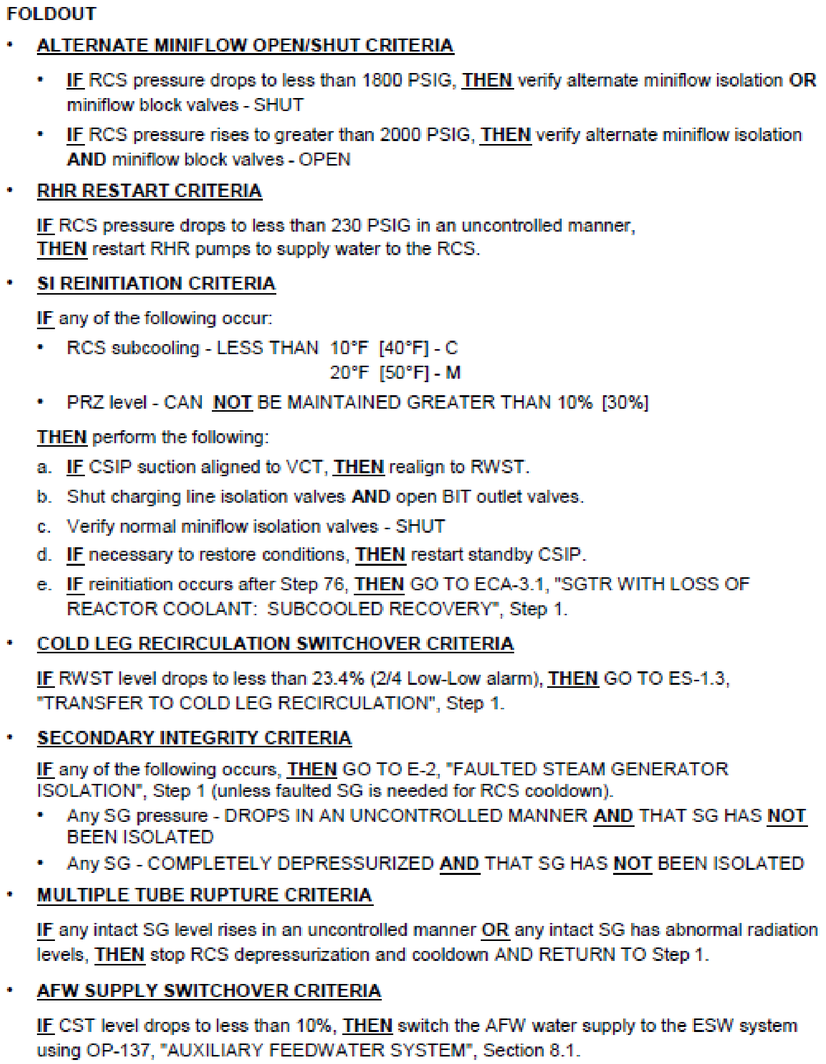
Event Description: **'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**

Time	Position	Applicant's Actions or Behavior
EOP-E-3	SRO	Enters E-3, Steam Generator Tube Rupture Holds crew update
Procedure Note:		**Foldout applies.**

Evaluator Note:



FOLDOUT

- **ALTERNATE MINIFLOW OPEN/SHUT CRITERIA**
  - **IF** RCS pressure drops to less than 1800 PSIG, **THEN** verify alternate miniflow isolation **OR** miniflow block valves - SHUT
  - **IF** RCS pressure rises to greater than 2000 PSIG, **THEN** verify alternate miniflow isolation **AND** miniflow block valves - OPEN
- **RHR RESTART CRITERIA**

  IF RCS pressure drops to less than 230 PSIG in an uncontrolled manner, **THEN** restart RHR pumps to supply water to the RCS.
- **SI REINITIATION CRITERIA**

  IF any of the following occur:
  - RCS subcooling - LESS THAN 10°F [40°F] - C
    20°F [50°F] - M
  - PRZ level - CAN **NOT** BE MAINTAINED GREATER THAN 10% [30%]

  THEN perform the following:

  a. **IF** CSIP suction aligned to VCT, **THEN** realign to RWST.
  b. Shut charging line isolation valves **AND** open BIT outlet valves.
  c. Verify normal miniflow isolation valves - SHUT
  d. **IF** necessary to restore conditions, **THEN** restart standby CSIP.
  e. **IF** reinitiation occurs after Step 76, **THEN** GO TO ECA-3.1, "SGTR WITH LOSS OF REACTOR COOLANT: SUBCOOLED RECOVERY", Step 1.
- **COLD LEG RECIRCULATION SWITCHOVER CRITERIA**

  IF RWST level drops to less than 23.4% (2/4 Low-Low alarm), **THEN** GO TO ES-1.3, "TRANSFER TO COLD LEG RECIRCULATION", Step 1.
- **SECONDARY INTEGRITY CRITERIA**

  IF any of the following occurs, **THEN** GO TO E-2, "FAULTED STEAM GENERATOR ISOLATION", Step 1 (unless faulted SG is needed for RCS cooldown).
  - Any SG pressure - DROPS IN AN UNCONTROLLED MANNER **AND** THAT SG HAS **NOT** BEEN ISOLATED
  - Any SG - COMPLETELY DEPRESSURIZED **AND** THAT SG HAS **NOT** BEEN ISOLATED
- **MULTIPLE TUBE RUPTURE CRITERIA**

  IF any intact SG level rises in an uncontrolled manner **OR** any intact SG has abnormal radiation levels, **THEN** stop RCS depressurization and cooldown AND RETURN TO Step 1.
- **AFW SUPPLY SWITCHOVER CRITERIA**

  IF CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1.

No actions should result from FOLDOUT page during the remainder of the scenario.

Op Test No.: NRC	Scenario # 1	Event # 7	Page 60 of 83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**		

Time	Position	Applicant's Actions or Behavior	
	SRO	Assigns Foldout items: Alternate Miniflow Open/Shut Criteria, RHR Restart Criteria, SI Reinitiation Criteria, Cold Leg Recirculation Switchover Criteria Secondary Integrity Criteria, Multiple Tube Rupture Criteria, AFW Supply Switchover Criteria Initiates Monitoring Of Critical Safety Function Status Trees.	
	RO	Any RCP – RUNNING	(YES)
Procedure Note:		The RCP Trip Criteria is in effect until an RCS cooldown is initiated.	
	RO	CHECK RCP Trip Criteria: • Check all of the following: ○ SI flow - GREATER THAN 200 GPM ○ Check RCS pressure - LESS THAN 1400 PSIG	(YES) (NO)
	SRO	RNO: GO TO Step 4.	
	BOP	CHECK RCP Ruptured SG(s) - IDENTIFIED Ruptured SG Identification (Any of the following) SG level - RISING IN AN UNCONTROLLED MANNER SG Sample - HIGH RADIATION Main Steamlines - HIGH RADIATION • RM-01MS-3591 SB, Main Steam Line A • RM-01MS-3592 SB, Main Steam Line B • RM-01MS-3593 SB, Main Steam Line C	(YES)

Op Test No.: NRC	Scenario # 1	Event # 7	Page 61 of 83		
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**				
Time	Position	Applicant's Actions or Behavior			

<table>
<tr><td></td><td>BOP</td><td colspan="2">ADJUST ruptured SG PORV controller setpoint to 88% (1145 PSIG) AND place in AUTO.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td>CHECK ruptured SG PORV – SHUT.</td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td>Check Feed Flow To Intact SG(s) - AVAILABLE FROM MDAFW PUMP</td><td>(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Procedure Caution:</td><td colspan="2">The steam supply valve from the ruptured SG to the TDAFW pump should be shut OR isolated before initiating RCS cooldown (unless this prevents feeding SGs to be used for cooldown).</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td>SHUT ruptured SG steam supply valve to TDAFW pump:
SG B: 1MS-70
SG C: 1MS-72
(May have been closed previously in E-0)</td><td></td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td>VERIFY blowdown isolation valves from ruptured SG – SHUT
<table>
<tr><th colspan="3">SG Blowdown Isolation Valves</th></tr>
<tr><th>Process Line</th><th>Outside CNMT (MLB-1A-SA)</th><th>Inside CNMT (MLB-1B-SB)</th></tr>
<tr><td>SG A Blowdown</td><td>1BD-11</td><td>1BD-1</td></tr>
<tr><td>SG B Blowdown</td><td>1BD-30</td><td>1BD-20</td></tr>
<tr><td>SG C Blowdown</td><td>1BD-49</td><td>1BD-39</td></tr>
</table></td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td>SHUT ruptured SG main steam drain isolation before MSIV:
SG A: 1MS-231
SG B: 1MS-266
SG C: 1MS-301</td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
</table>

Op Test No.:	NRC	Scenario # 1	Event # 7	Page	62	of 83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**					
Time	Position	Applicant's Actions or Behavior				

	BOP	SHUT ruptured SG MSIV AND BYPASS valve.	(YES)
Procedure Caution:		IF ruptured SG is faulted AND is NOT needed for RCS cooldown, THEN feed flow to that SG should remain isolated.	
	BOP	Ruptured SG Level – GREATER THAN 25% [40%]	(YES)
	BOP	Ensure Feed Flow To Ruptured SG(s) - ISOLATED	(YES)
	BOP	CHECK Ruptured SG(s) Pressure – GREATER THAN 260 PSIG [350 PSIG]	(YES)
	RO	Check PRZ Pressure - LESS THAN 2000 PSIG	(NO)
	SRO	RNO: WHEN pressure lowers to less than 2000 PSIG during RCS cooldown, THEN perform Steps 16 AND 17. Continue with Step 18.	
Evaluator Note:		**During validation the pressure was greater than 2000 psig The "Check PRZ Pressure" could be answered YES or NO, depending on the pace at which the SRO progresses through the EOP network. The following two steps are the actions to be taken once PRZ Pressure is less than 200 psig.**	

Op Test No.: NRC Scenario # 1 Event # 7 Page 63 of 83

Event Description: **'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>RO</td><td>Check Steamline High Pressure Rate Bistables - CLEAR (NOT LIT)
<table>
<tr><th colspan="3">TSLB-1</th></tr>
<tr><td>STMLN A
HP RATE
PB 474C
(4-2)</td><td>STMLN B
HP RATE
PB 484C
(5-2)</td><td>STMLN C
HP RATE
PB 494C
(6-2)</td></tr>
<tr><td>STMLN A
HP RATE
PB 475C
(4-3)</td><td>STMLN B
HP RATE
PB 485C
(5-3)</td><td>STMLN C
HP RATE
PB 495C
(6-3)</td></tr>
<tr><td>STMLN A
HP RATE
PB 476C
(4-4)</td><td>STMLN B
HP RATE
PB 486C
(5-4)</td><td>STMLN C
HP RATE
PB 4956
(6-4)</td></tr>
</table></td><td>(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Procedure Note:</td><td colspan="2">After the low steam pressure SI signal is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td colspan="2">Block Low Steam Pressure SI.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td>At least one intact SG - AVAILABLE FOR RCS COOLDOWN</td><td>(YES)</td></tr>
<tr><td></td><td>SRO</td><td colspan="2">GO TO Step 23.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
</table>

Op Test No.:	NRC	Scenario #	1	Event #	7	Page	64	of	83

Event Description: **'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>SRO</td><td colspan="2">Determine required core exit temperature based on lowest ruptured SG pressure:
<table>
<tr><th>LOWEST RUPTURED SG PRESSURE (PSIG)</th><th>ERFIS AVAILABLE: CORE EXIT TEMPERATURE (°F)</th><th>ERFIS <u>NOT</u> AVAILABLE: HIGHEST CORE EXIT TC (PREFERRED) OR ACTIVE LOOP WIDE RANGE T-HOT (°F)</th></tr>
<tr><td>ABOVE 1100</td><td>530 [495]</td><td>520 [490]</td></tr>
<tr><td>1000 TO 1100</td><td>515 [485]</td><td>505 [475]</td></tr>
<tr><td>900 TO 1000</td><td>505 [470]</td><td>495 [465]</td></tr>
<tr><td>800 TO 900</td><td>490 [460]</td><td>480 [450]</td></tr>
<tr><td>700 TO 800</td><td>475 [445]</td><td>465 [435]</td></tr>
<tr><td>600 TO 700</td><td>460 [425]</td><td>450 [420]</td></tr>
<tr><td>500 TO 600</td><td>440 [410]</td><td>430 [400]</td></tr>
<tr><td>400 TO 500</td><td>420 [385]</td><td>410 [380]</td></tr>
<tr><td>300 TO 400</td><td>390 [360]</td><td>380 [350]</td></tr>
<tr><td>200 TO 300</td><td>360 [NA]</td><td>350 [NA]</td></tr>
</table></td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td>Condenser Available For Steam Dump:
<table>
<tr><th>Condenser Available Requirements</th></tr>
<tr><td>Any Intact SG MSIV - OPEN
Condenser Available (C-9)- LIT (BPLB 3-3)
Steam Dump Contol - AVAILALBE</td></tr>
</table></td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td colspan="2">Place steam dump pressure controller in manual AND decrease output to 0%.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td colspan="2">Place steam dump mode select switch in STEAM PRESS.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
</table>

Op Test No.:	NRC	Scenario #	1	Event #	7	Page	65	of 83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**							

Time	Position	Applicant's Actions or Behavior	
	BOP	Check RCS temperature - LESS THAN OR EQUAL 553˚F (P-12) BPLB-4-4 LOW-LOW TAVG STEAM DUMP BLOCKED (P-12)	(YES)
	BOP	Momentarily place both steam dump interlock bypass switches to INTLK BYP.	
	BOP	Check LOW-LOW STEAM DUMP (P-12) BYPASSED Status Light - ILLUMINATED BPLB-5-4 LOW-LOW TAVG STEAM DUMP BLOCKED (P-12) BYPASSED	(YES)
	BOP	Dump steam from intact SGs to Condenser at Maximum Rate	
	SRO	Core Exit TCs - LESS THAN REQUIRED TEMPERATURE	(NO)
	SRO	RNO: WHEN core exit TCs less than required temperature, THEN perform Steps 32 AND 33. • Observe CAUTION Prior To Step 34 AND Continue with Step 34.	
Evaluator Note:		**During cooldown at Max Rate, Main Steam Line Isolation may occur, requiring use of SG 'A' and 'B' PORVs to continue cooling down.** **The crew will continue with the procedure while the cooldown is in progress. When the CET temperature is less than the target then the crew should terminate the cooldown and continue with the procedure.**	

Op Test No.:	NRC	Scenario #	1	Event #	7	Page	66	of 83

Event Description: **'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**

Time	Position	Applicant's Actions or Behavior	
Procedure Caution:		If no RCPs running, the following actions may cause a false indication for the INTEGRITY CSFST. Disregard ruptured SG wide range cold leg temperature until Step 94 complete.	
	RO	Maintain RCP Seal Injection Flow Between 8 GPM And 13 GPM.	
Procedure Caution:		If an AFW isolation to an intact SG occurs, the signal may be reset to allow restoration of AFW. (An AFW isolation will occur if a main steam line isolation signal is present AND one SG pressure decreases 100 PSIG below the other two SGs.) If the steam supply valve from the ruptured SG to TDAFW pump reopens due to decreasing SG level, it must be restored to the shut position. (Two out of three SG levels decreasing below 25% will open both steam supply vales to the TDAFW pump.)	
	BOP	Any Intact SG Level - GREATER THAN 25% [40%]	(YES)
	BOP	AFW flow - AT LEAST 200 KPPH AVAILABLE	(YES)
	BOP	Control Feed Flow To Maintain Intact SG Levels Between 25% And 50% [40% and 50%]	
	RO	Verify Power To PORV Block Valves - AVAILABLE	(YES)
	RO	PRZ PORVs - SHUT	(YES)
	RO	Check block valves - AT LEAST ONE OPEN	(YES)
	RO	Reset SI.	

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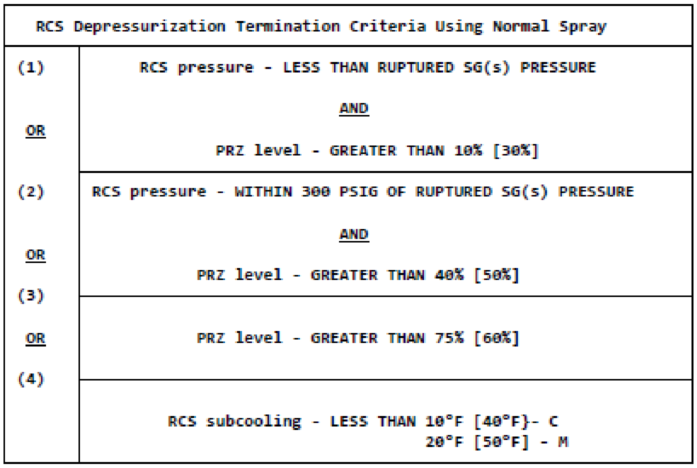
Event Description: **'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**

Time	Position	Applicant's Actions or Behavior	
	SRO	Manually Realign Safeguards Equipment Following A Loss Of Offsite Power. (Directs RO)	
	RO	Reset Phase A AND Phase B Isolation Signals. (Phase A only is actuated)	
	RO	Open Instrument Air **AND** Nitrogen Valves To CNMT: 1IA-819 (ISOL VALVE CONT. BLDG 236' PENETRATION (M-80)) 1SI-287 (ACCUMULATOR & PRZ PORV N2 SUPPLY ISO VLV)	
	RO	Check RHR pump suction - ALIGNED TO RWST RWST SUCTION (OPEN) RHR A: 1SI-322 RHR B: ISI-323	(YES)
	RO	RCS pressure - GREATER THAN 230 PSIG	(YES)
	RO	Stop RHR pumps.	
	RO	Core exit TCs - LESS THAN REQUIRED TEMPERATURE	(YES/NO)
	BOP	Stop RCS cooldown	
	BOP	Maintain core exit TCs less than required temperature.	

Op Test No.:	NRC	Scenario #	1	Event #	7	Page	68 of 83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**						

Time	Position	Applicant's Actions or Behavior	
	BOP	Check ruptured SG pressure - STABLE OR RISING	(YES)
	RO	Check RCS Subcooling - GREATER THAN 30 ˚F – C	(YES)
	RO	Normal PRZ spray - AVAILABLE (INCLUDING INSTRUMENT AIR TO CNMT) Normal PRZ Spray Valves 1RC-107 (PCV-444C) 1RC-103 (PCV-444D)	(YES)
	RO	Check PRZ level - LESS THAN OR EQUAL TO 75% [60%]	(YES)
Critical Task #3	RO	Manually Open All Available Normal PRZ Spray Valves AND Spray At Maximum Rate (Until ANY Of The RCS Depressurization Termination Criteria in Step 56 Satisfied). *Critical to minimize primary to secondary leakage prior to SG 'C' exceeding 95% level*	
Evaluator Note:		**Crew will maintain the spray valves open until one of the RCS Depressurization Termination Criteria on the following page is SATISFIED**	

Op Test No.:	NRC	Scenario #	1	Event #	7	Page	69	of 83
Event Description:	**'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)**							

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>RO</td><td>Check RCS Depressurization Termination Criteria – SATISFIED

<table>
<tr><th colspan="2">RCS Depressurization Termination Criteria Using Normal Spray</th></tr>
<tr><td>(1)
<u>OR</u></td><td>RCS pressure - LESS THAN RUPTURED SG(s) PRESSURE
<u>AND</u>
PRZ level - GREATER THAN 10% [30%]</td></tr>
<tr><td>(2)
<u>OR</u></td><td>RCS pressure - WITHIN 300 PSIG OF RUPTURED SG(s) PRESSURE
<u>AND</u>
PRZ level - GREATER THAN 40% [50%]</td></tr>
<tr><td>(3)
<u>OR</u></td><td>PRZ level - GREATER THAN 75% [60%]</td></tr>
<tr><td>(4)</td><td>RCS subcooling - LESS THAN 10°F [40°F]- C
20°F [50°F] - M</td></tr>
</table></td><td>(NO)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">RNO: Continue to monitor termination criteria.
• WHEN criteria satisfied, THEN GO TO Step 57.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td colspan="2">Shut spray valve used for depressurization:</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">GO TO Step 65.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td>RCS subcooling – GREATER THAN 10°F - C</td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td>Level In At Least One Intact SG - GREATER THAN 25% [40%]</td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">GO TO Step 69.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td>RCS pressure - STABLE OR RISING</td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>RO</td><td>PRZ level - GREATER THAN 10% [30%]</td><td>(YES)</td></tr>
</table>

<table>
<tr><td colspan="3">Op Test No.: NRC Scenario # 1 Event # 7 Page 70 of 83</td></tr>
<tr><td colspan="3">Event Description: 'C' Steam Generator Tube Rupture of 250 gpm (EOP-E-3)</td></tr>
<tr><td>Time</td><td>Position</td><td>Applicant's Actions or Behavior</td></tr>
</table>

<table>
<tr><td></td><td>RO</td><td colspan="2">Stop All But One CSIP.</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>RO</td><td>Check CSIP Suction - ALIGNED TO RWST
<table>
<tr><td>VCT OUTLET
(SHUT)</td><td>RWST SUCTION
(OPEN)</td></tr>
<tr><td>1CS-165 (LCV-115C)
1CS-166 (LCV-115E)</td><td>1CS-291 (LCV-115B)
1CS-292 (LCV-115D)</td></tr>
</table></td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>RO</td><td colspan="2">Open Normal Miniflow Isolation Valves:
<table>
<tr><td>CSIP A:</td><td>1CS-182</td></tr>
<tr><td>CSIP B:</td><td>1CS-196</td></tr>
<tr><td>CSIP C:</td><td>1CS-210</td></tr>
<tr><td>COMMON:</td><td>1CS-214</td></tr>
</table></td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td colspan="2">Shut BIT outlet valves:
<table>
<tr><td>1SI-3
1SI-4</td></tr>
</table></td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Lead Evaluator:</td><td colspan="2">Terminate the scenario after BIT outlet valves 1SI-3 and 1SI-4 are SHUT.
Announce 'Crew Update' - End of Evaluation - I have the shift.
Have crew remain in the Simulator without discussing the exam. Examiners will formulate any follow-up questions.</td></tr>
</table>

Simulator Operator:	**When directed by Lead Evaluator go to FREEZE**

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 1 of 7
SAFEGUARDS ACTUATION VERIFICATION

NOTE

- General guidance for verification of safeguards equipment is contained in Attachment 4 of this procedure.
- ERFIS displays of safeguards equipment status are not reliable while any associated safety-related electrical buses are de-energized.

☐ 1. **Ensure** Two CSIPs - RUNNING

☐ 2. **Ensure** Two RHR Pumps - RUNNING

☐ 3. **Ensure** Two CCW Pumps - RUNNING

☐ 4. **Ensure** All ESW **AND** ESW Booster Pumps - RUNNING

☐ 5. **Ensure** SI Valves - PROPERLY ALIGNED

(**Refer** to Attachment 1.)

☐ 6. **Ensure** CNMT Phase A Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 4.)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 2 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 7. **Ensure** SG Blowdown **AND** SG Sample Isolation Valves In Table 1 - SHUT

Table 1: SG Blowdown And Sample Isolation Valves

Process Line	Outside CNMT (MLB-1A-SA)	Inside CNMT (MLB-1B-SB)
SG A Sample	1SP-217	1SP-214/216
SG B Sample	1SP-222	1SP-219/221
SG C Sample	1SP-227	1SP-224/226
SG A Blowdown	1BD-11	1BD-1
SG B Blowdown	1BD-30	1BD-20
SG C Blowdown	1BD-49	1BD-39

8. IF Main Steam Line Isolation Actuated OR Is Required By Any Of The Following, THEN **Ensure** MSIVs **AND** MSIV Bypass Valves - SHUT

☐ • Steam line pressure - LESS THAN 601 PSIG

☐ • CNMT pressure - GREATER THAN 3.0 PSIG

9. IF CNMT Spray Actuation Signal Actuated OR Is Required, THEN **Ensure** The Following:

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 9.)

☐ • CNMT spray pumps - RUNNING

☐ • CNMT spray valves - PROPERLY ALIGNED

☐ • Phase B isolation valves - SHUT

☐ • All RCPs - STOPPED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 3 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 10. **Ensure** Both Main FW Pumps - TRIPPED

☐ 11. **Ensure** FW Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 6.)

☐ 12. **Ensure** Both MDAFW pumps - RUNNING

13. **<u>IF</u>** Any Of The Following Conditions Exist, **<u>THEN</u> Ensure** The TDAFW Pump - RUNNING

- ☐ Undervoltage on either 6.9 KV emergency bus
- ☐ Level in two SGs - LESS THAN 25%
- ☐ Manual actuation to control SG level

14. **Ensure** AFW Valves - PROPERLY ALIGNED

- ☐ **<u>IF</u>** no AFW Isolation Signal, **<u>THEN</u> ensure** isolation **<u>AND</u>** flow control valves - OPEN

> **<u>NOTE</u>**
>
> An AFW Isolation signal signal requires a Main Steam Line Isolation coincident with one SG pressure 100 PSIG below the other two SGs.

- ☐ **<u>IF</u>** AFW Isolation Signal present, **<u>THEN</u> ensure** MDAFW **<u>AND</u>** TDAFW isolation **<u>AND</u>** flow control valves to affected SG - SHUT

☐ 15. **Ensure** Both EDGs - RUNNING

☐ 16. **Ensure** CNMT Fan Coolers - ONE FAN PER UNIT RUNNING IN SLOW SPEED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 4 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 17. **Ensure** CNMT Ventilation Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 7.)

☐ 18. **Ensure** Control Room Area Ventilation - MAIN CONTROL ROOM ALIGNED FOR EMERGENCY OPERATION

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 5, Sheets 1 and 2, Sections for MAIN CONTROL BOARD, SLB-5 and SLB-6.)

19. **Ensure** Essential Service Chilled Water System Operation:

☐ • **Ensure** both WC-2 chillers - RUNNING

☐ • **Ensure** both P-4 pumps - RUNNING

☐ (**Refer** to AOP-026, "LOSS OF ESSENTIAL SERVICE CHILLED WATER SYSTEM" for loss of any WC-2 chiller.)

20. **Ensure** CSIP Fan Coolers - RUNNING

☐ AH-9 A SA
☐ AH-9 B SB
☐ AH-10 A SA
☐ AH-10 B SB

NOTE

Security systems are powered by bus 1A1 (normal supply) or bus 1B1 (alternate supply). Backup power will be available for approximately 30 MINUTES after the supplying bus is de-energized. (Refer to OP-115, "CENTRAL ALARM STATION ELECTRICAL SYSTEMS", Section 8.9 and 8.10.)

☐ 21. **Ensure** AC buses 1A1 **AND** 1B1 - ENERGIZED

☐ 22. **Place** Air Compressor 1A **AND** 1B In The LOCAL CONTROL Mode.

(**Refer to** Attachment 7.)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 5 of 7
SAFEGUARDS ACTUATION VERIFICATION

CAUTION

The maximum calculated dose rate in the vicinity of MCC 1A35-SA and MCC 1B35-SB is between 10 MREM/HR and 150 MREM/HR.

☐ 23. Dispatch An Operator To **Unlock** And **Close** The Breakers For The CSIP Suction **AND** Discharge Cross-Connect Valves:

(**Refer** to Attachment 2.)

MCC 1A35-SA		MCC 1B35-SB	
VALVE	CUBICLE	VALVE	CUBICLE
1CS-170	4A	1CS-171	4D
1CS-169	4B	1CS-168	7D
1CS-218	14D	1CS-220	9D
1CS-219	14E	1CS-217	12C

24. **Check** If C CSIP Should Be Placed In Service:

☐ • **IF** two charging pumps can **NOT** be verified to be running, **AND** C CSIP is available, **THEN place** C CSIP in service in place of the non-running CSIP using OP-107, "CHEMICAL AND VOLUME CONTROL SYSTEM, Section 8.5 or 8.7.

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 6 of 7
SAFEGUARDS ACTUATION VERIFICATION

25. **Start** The Spent Fuel Pump Room Ventilation System:

    a. At AEP-1, **ensure** the following ESCWS isolation valves - OPEN

    1) SLB-11 (Train A)

    ☐ • AH-17 SUP CH 100 (Window 9-1)

    ☐ • AH-17 RTN CH 105 (Window 10-1)

    2) SLB-9 (Train B)

    ☐ • AH-17 SUP CH 171 (Window 9-1)

    ☐ • AH-17 RTN CH 182 (Window 10-1)

    b. At AEP-1, **start** one SFP PUMP ROOM FAN COOLER:

    ☐ • AH-17 1-4A SA

    ☐ • AH-17 1-4B SB

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 7 of 7
SAFEGUARDS ACTUATION VERIFICATION

NOTE

- Fuel pool levels and temperatures should be monitored approximately every 1 to 2 HOURS.
- Following the initial check of fuel pool levels and temperature, monitoring responsibilities may be assumed by the plant operations staff (including the TSC or STA).
- Only fuel pools containing fuel are required to be monitored.

26. **Check** Status Of Fuel Pools:

☐ a. **Operate** spent fuel cooling pumps to maintain fuel pool temperatures between 85°F to 105°F.

b. **Monitor** fuel pool levels **AND** temperatures:

☐ • **Refer to** AOP-041, "SPENT FUEL POOL EVENT" Attachments 7, 8, 9, 10 and 11 for SFP parameter monitoring methods.

☐ • **Refer to** Curves H-X-24, H-X-25 and H-X-26 for SFP time to 200°F.

☐ • Levels - GREATER THAN LO ALARM (284 FT, 0 IN)

☐ • Temperatures - LESS THAN HI TEMP ALARM (105°F)

NOTE

If control room ventilation was previously aligned to an emergency outside air intake for post-accident operations, then follow-up actions will be required to restore the alignment.

27. **Consult** Plant Operations Staff Regarding Alignment Of The Control Room Ventilation System:

☐ • Site Emergency Coordinator - Control Room

☐ • Site Emergency Coordinator - Technical Support Center

(Refer to PEP-230, "CONTROL ROOM OPERATIONS".)

- END -

RAPID DOWNPOWER

Attachment 2 - Boric Acid/Target Rod Height for Power Reduction

Sheet 1 of 2

NOTE

- This Attachment serves as a reactivity plan. **[C.3]**
- These tables are developed from HNEI-0400 Series Harris Cycle-Specific Startup Operations Report (SOR). These tables are cycle-specific, but will only need to be updated for the new cycle if the table values fall outside of the acceptable range established in the SOR.
- Target rod heights correspond to the lower (target) power level in each row and are applicable regardless of the rate of power reduction or source of boration.
- Gallons of boric acid in Table 1 are for 10% power reduction increments. These are additive for power reductions of greater than 10%.
  Example: A power reduction from 90% to 60% at BOL would require [180 gal + 163 gal +146 gal = 489 gal]
- For purposes of this procedure, 5% increments can be obtained by dividing by two, or by referring to Table 2 - 5% Power Reduction Increments.
- As used in this table, the following times in core life are assumed:
  BOL (0 ≤ EFPD ≤150) (3000 MWD/MTU)
  MOL (150 < EFPD ≤ 350) (10,000 MWD/MTU)
  EOL (350 < EFPD) (17,000 MWD/MTU)

Table 1 - 10% Power Reduction Increments

Power Level (%)	Target Rod Height (D Bank)	Gallons of Boric Acid Required for Power Reduction		
		BOL 0 ≤ EFPD ≤ 150	MOL 150 < EFPD ≤ 350	EOL 350 < EFPD
100 to 90	206	223	273	285
90 to 80	194	180	215	234
80 to 70	183	163	200	212
70 to 60	171	146	167	198
60 to 50	159	138	159	192
50 to 40	147	139	151	194
40 to 30	135	122	144	204
30 to 20	124	141	154	230
20 to 10	112	123	137	266

RAPID DOWNPOWER

Attachment 2 - Boric Acid/Target Rod Height for Power Reduction

Sheet 2 of 2

Table 2 - 5% Power Reduction Increments

Power Level (%)	Target Rod Height (D Bank)	Gallons of Boric Acid Required for Power Reduction		
		BOL 0 ≤ EFPD ≤ 150	MOL 150 < EFPD ≤ 350	EOL 350 < EFPD
100 to 95	212	112	137	143
95 to 90	206	111	136	142
90 to 85	200	90	108	117
85 to 80	194	90	107	117
80 to 75	188	82	100	106
75 to 70	183	81	100	106
70 to 65	177	73	84	99
65 to 60	171	73	83	99
60 to 55	165	69	80	96
55 to 50	159	69	79	96
50 to 45	153	70	76	97
45 to 40	147	69	75	97
40 to 35	141	61	72	102
35 to 30	135	61	72	102
30 to 25	129	71	77	115
25 to 20	124	70	77	115
20 to 15	118	62	69	133
15 to 10	112	61	68	133

--END OF ATTACHMENT 2--

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ATTACHMENT 5
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<< Reference Use - Control Room Ventilation Isolation Verification >>

TRAIN - A Components	REQ POS	POS CK	TRAIN - B Components	REQ POS	POS CK
MAIN CONTROL BOARD					
CZ-19 SA EMERGENCY FILTRATION DISCHARGE	OPEN [Note 1]		CZ-20 SB EMERGENCY FILTRATION DISCHARGE	OPEN [Note 1]	
R2 A-SA EMERGENCY FILTRATION FAN	START		R2 B-SB EMERGENCY FILTRATION FAN	START	
CZ-9 SA EMERG FILT SOUTH OUTSIDE AIR INLET	SHUT		CZ-10 SB EMERG FILT SOUTH OUTSIDE AIR INLET	SHUT	
CZ-11 SA EMERG FILT NORTH OUTSIDE AIR INLET	SHUT		CZ-12 SB EMERG FILT NORTH OUTSIDE AIR INLET	SHUT	
CZ-D66 SA EMERGENCY FILTRATION RECIRC	OPEN		CZ-D61 SB EMERGENCY FILTRATION RECIRC	OPEN	
ES-1A PURGE EXHAUST FAN	STOP		ES-1B PURGE EXHAUST FAN	STOP	
CZ-13 SA PURGE EXHAUST	SHUT		CZ-14 SB PURGE EXHAUST	SHUT	
CZ-17 SA PURGE MAKE UP	SHUT		CZ-18 SB PURGE MAKE UP	SHUT	
CZ-D69 SA CONT RM NORMAL REC DAMPER	OPEN [Note 1]		CZ-D70 SB CONT RM NORMAL REC DAMPER	OPEN [Note 1]	
CZ-1 SA NORMAL INTAKE	SHUT		CZ-2 SB NORMAL INTAKE	SHUT	
CZ-3 SA NORMAL EXHAUST	SHUT		CZ-4 SB NORMAL EXHAUST	SHUT	
E-9A NORMAL EXHAUST FAN	STOP		E-9B NORMAL EXHAUST FAN	STOP	
ACTUATED BY EITHER TRAIN A OR B			E-5A CNMT PRE-ENTRY PURGE EXHAUST FAN	STOP	
			E-5B CNMT PRE-ENTRY PURGE EXHAUST FAN	STOP	

Note:

1. This component does not receive direct actuation signal but is slaved to other equipment.

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ATTACHMENT 5
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<< Reference Use - Control Room Ventilation Isolation Verification >>

TRAIN - A Components		REQ POS	POS CK	TRAIN - B Components		REQ POS	POS CK
SLB - 5 TRAIN A				**SLB - 6 TRAIN B**			
8-1	AH-15 IN CZ-D1	OPEN		8-1	AH-15 IN CZ-D2	OPEN	
8-2	AH-15 DISCH CZ-25	OPEN		8-2	AH-15 DISCH CZ-26	OPEN	
8-3	R2 IN CZ-23	OPEN [Note 2]		8-3	R2 IN CZ-24	OPEN [Note 2]	
8-4	R2 OUT CZ-21	OPEN [Note 2]		8-4	R2 OUT CZ-22	OPEN [Note 2]	
AEP - 1							
E-28 A SA BATTERY ROOM A EXHAUST FAN		STOP		E-29 A SB BATTERY ROOM B EXHAUST FAN		STOP	
E-28 B SA BATTERY ROOM A EXHAUST FAN		STOP		E-29 B SB BATTERY ROOM B EXHAUST FAN		STOP	
E-10 A SA NORMAL EXHAUST FAN		STOP		E-10 B SB NORMAL EXHAUST FAN		STOP	
AC-D4 SA BATTERY ROOM A RETURN DAMPER		OPEN		AC-D6 SB BATTERY ROOM B RETURN DAMPER		OPEN	
1CZ-7 SA EXHAUST FAN DISCHARGE ISOL		SHUT		1CZ-8 SB EXHAUST FAN DISCHARGE ISOL		SHUT	
1CZ-5 SA RAB ELEC EQUIP ROOM OAI PURGE ISOL		SHUT		1CZ-6 SB RAB ELEC EQUIP ROOM OAI PURGE ISOL		SHUT	
E-6 A SA EMERGENCY EXHAUST FAN		START [Note 3]		E-6 B SB EMERGENCY EXHAUST FAN		START [Note 3]	
ACTUATED BY EITHER TRAIN A OR B				E-17 X NNS NORMAL EXHAUST FAN		STOP	
				E-18 X NNS NORMAL EXHAUST FAN		STOP	
				E-19 X NNS NORMAL EXHAUST FAN		STOP	
				E-20 X NNS NORMAL EXHAUST FAN		STOP	
				S-3 A NNS RAB NORMAL SUPPLY FAN		STOP	
				S-3 B NNS RAB NORMAL SUPPLY FAN		STOP	

Notes:

2. This component does not receive direct actuation signal but is slaved to other equipment.
3. This component starts from the SI signal not the CRIS.

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<< Reference Use - Control Room Ventilation Isolation Verification >>

TRAIN - A Components		REQ POS	POS CK	TRAIN - B Components		REQ POS	POS CK
SLB - 10				**SLB - 8**			
1-2	ISOL AV D 23	SHUT		1-2	ISOL AV D 24	SHUT	
1-3	ISOL AV D 21	SHUT		1-3	ISOL AV D 22	SHUT	
1-4	ISOL AV D 19	SHUT		1-4	ISOL AV D 20	SHUT	
2-2	ISOL AV D 62	SHUT		2-2	ISOL AV D 63	SHUT	
2-3	ISOL AV D 70	SHUT		2-3	ISOL AV D 71	SHUT	
2-4	ISOL AV D 66	SHUT		2-4	ISOL AV D 67	SHUT	
3-2	ISOL AV D 15	SHUT		3-2	ISOL AV D 16	SHUT	
3-3	ISOL AV D 27	SHUT		3-3	ISOL AV D 28	SHUT	
3-4	ISOL AV D 25	SHUT		3-4	ISOL AV D 26	SHUT	
4-2	ISOL AV D 82	SHUT		4-2	ISOL AV D 83	SHUT	
4-3	ISOL AV D 74	SHUT		4-3	ISOL AV D 75	SHUT	
4-4	ISOL AV D 58	SHUT		4-4	ISOL AV D 59	SHUT	
5-2	ISOL AV D 17	SHUT		5-2	ISOL AV D 18	SHUT	
5-3	ISOL AV D 13	SHUT		5-3	ISOL AV D 14	SHUT	
5-4	ISOL AV D 11	SHUT		5-4	ISOL AV D 12	SHUT	
6-2	ISOL AV D 78	SHUT		6-2	ISOL AV D 79	SHUT	
6-3	ISOL AV D 52	SHUT		6-3	ISOL AV D 53	SHUT	
6-4	ISOL AV D 33	SHUT		6-4	ISOL AV D 34	SHUT	
7-2	ISOL AV D 35	SHUT		7-2	ISOL AV D 36	SHUT	
7-3	ISOL AV D 3	SHUT		7-3	ISOL AV D 4	SHUT	
7-4	ISOL AV D 9	SHUT		7-4	ISOL AV D 10	SHUT	
8-2	ISOL AV D 37	SHUT		8-2	ISOL AV D 38	SHUT	
8-3	ISOL AV D 7	SHUT		8-3	ISOL AV D 8	SHUT	
8-4	ISOL AV D 5	SHUT		8-4	ISOL AV D 6	SHUT	
9-2	ISOL AV D 86	SHUT		9-2	ISOL AV D 87	SHUT	
9-3	ISOL AV D 31	SHUT		9-3	ISOL AV D 32	SHUT	
9-4	ISOL AV D 29	SHUT		9-4	ISOL AV D 30	SHUT	

Attachment 3 OMM-004 Attachment 5

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<< Reference Use - Control Room Ventilation Isolation Verification >>

Components		REQ POS	POS CK
AEP - 3			
ACTUATED BY EITHER TRAIN A OR B	R-13 (1 & 2A NNS) EMER FAN	START	
	R-13 (1 & 2B NNS) EMER FAN	START	
	ES-7 (1 & 2 NNS) SMOKE PURGE EXHAUST FAN	STOP	
	CK-B6 1 & 2 SMOKE PURGE INTAKE VLV	SHUT	
	CK-B7 1 & 2 NORMAL INTAKE VLV	SHUT	
	CK-B8 1 & 2 NORMAL INTAKE VLV	SHUT	

Comment No.	Description

Signature: ______________________ Time ________ ... Date ________

HARRIS 2020 NRC SCENARIO 2

Facility:	Harris Nuclear Plant	Scenario No.:	2	Op Test No.:	05000400/2020301
Examiners:				Operators:	SRO:
					RO:
					BOP:

Initial Conditions: IC-5 BOL, 53% power

- ‘B-SB’ Boric Acid Transfer Pump is under clearance for breaker repairs
- 1CS-9, Letdown Isolation Valve is under clearance for solenoid replacement
- ‘A’ Gland Steam Condenser Exhauster Fan is under clearance due to high vibrations on the motor bearing

Turnover:	The plant is at 53% power, beginning of core life. GP-005 step 134.e, comparison of diverse indications of power after exceeding 50% power is complete.
Critical Task:	• Manually start the standby DEH Pump prior to DEH pressure lowering below 1150 psig to prevent an automatic Turbine Trip/Reactor trip • Manually maintain control of SG ‘B’ level below 78% to prevent an automatic Reactor trip after steam generator level transmitter LT-486 fails low • Manually trip all RCPs within 10 minutes of a Phase B isolation signal • Shut BIT Outlet valve 1SI-4 prior to establishing flow through the charging header

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R – RO/SRO N – BOP/SRO	Power ascension from 53% power
2	nis08b	I – RO/SRO TS – SRO	PRNIS Channel NI-42 fails HIGH (AOP-001)
3	lt:460	I – RO/SRO TS – SRO	Pressurizer Level Transmitter for LT-460 fails low
4	xd1i142 xd1o142w xn27e05	C – BOP/SRO	Reactor Primary Shield Fan Failure
5	tur24a jmsehpas	C – BOP/SRO	DEH pump shaft shear and failure of the standby pump to start
6	lt:486	C – BOP/SRO TS – SRO	'B' SG Controlling Level Transmitter fails Low (AOP-010)
7	mss01b	M – ALL	Steam line Break on 'B' SG inside Containment (EOP-E-0 and EOP-E-2)
8	zrpk643b zrpk644b zrpk645b	C – RO/SRO	‘B’ Containment Spray pump fails to auto start
9	sis017 sis018	C – BOP/SRO	1SI-4 failure to close from MCB switch

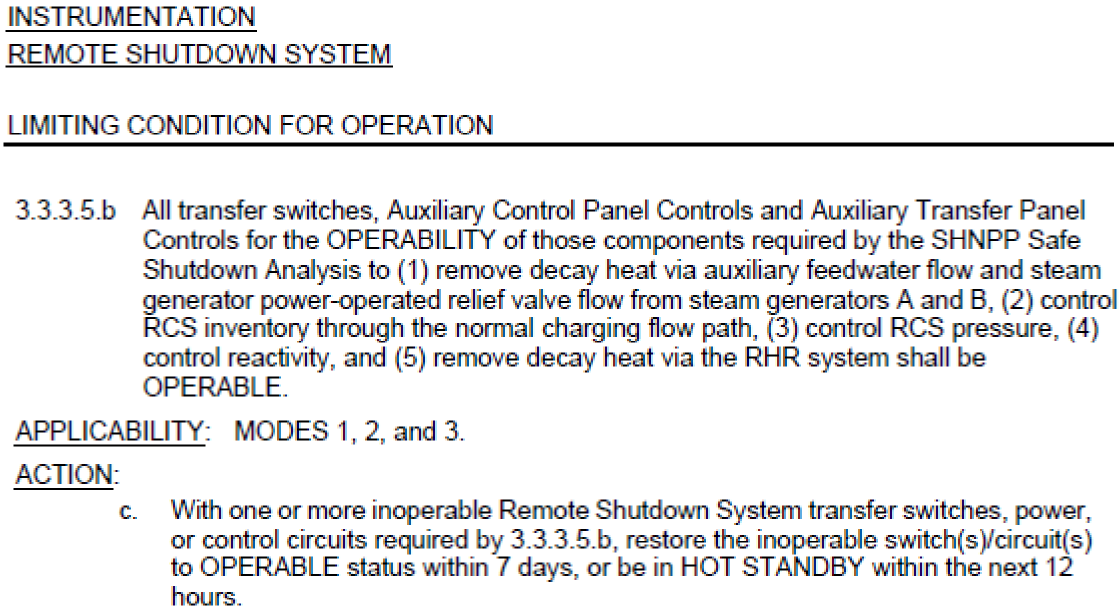
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 2

The plant is at 53% power, beginning of core life. GP-005, Power Operation (Mode 2 To Mode 1) is in progress as directed by plant management. GP-005 step 134.e, comparison of diverse indications of power after exceeding 50% power is complete and the turbine is in hold for turnover. Once turnover is complete, raise TCS Load Control to 4 GVPC units/ min and continue the power ascension @ 4 MW/min.

The following equipment is under clearance:

- Boric Acid Transfer Pump B-SB is under clearance due to breaker blown control power fuses. Has been under clearance for 12 hours. The problem with the breaker has been repaired and the clearance will be removed later this shift. Tech Spec 3.3.3.5.b Action **c** and 3.1.2.2 applies (3.1.2.2 is for tracking only). OWP-CS-05 has been completed.



INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

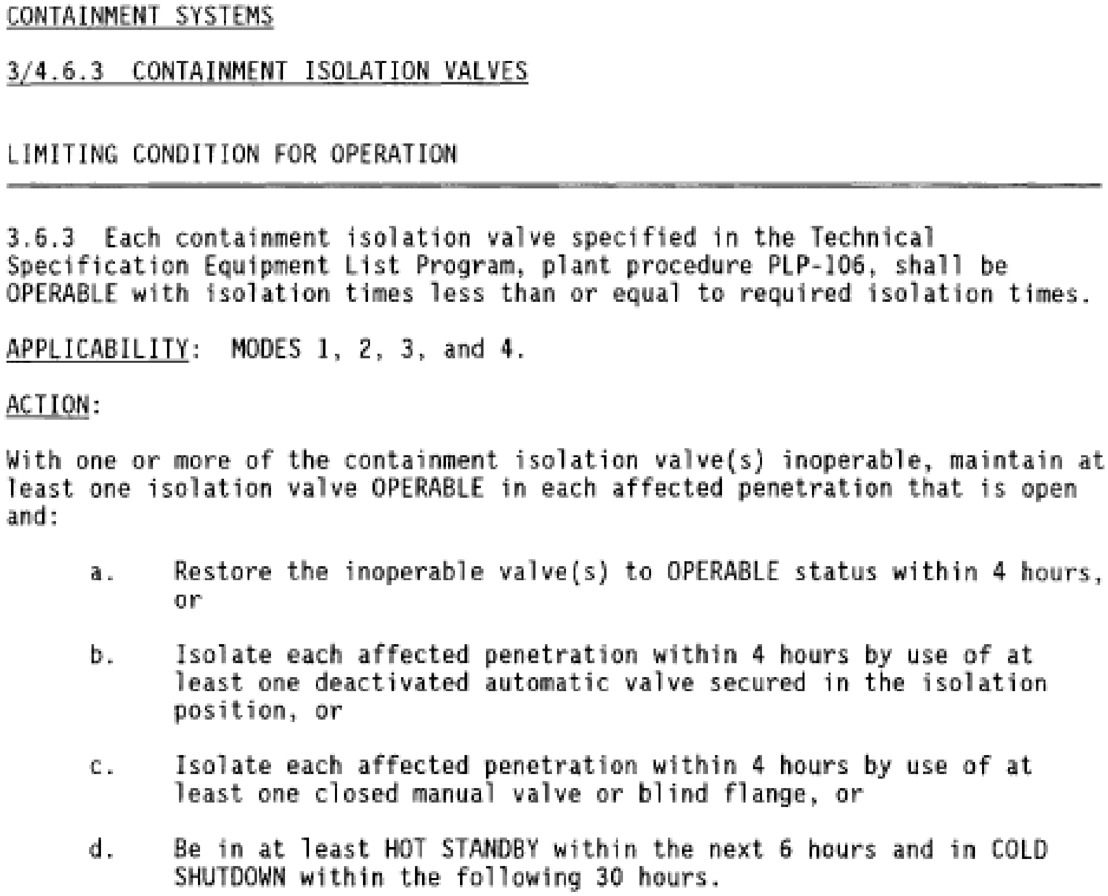
c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

- 1CS-9, Letdown Orifice Isolation valve, has been under clearance the last 12 hours for solenoid replacement. The repairs are close to completion and the valve is expected to be returned to service within the next hour. The valve is currently shut with power removed. OWP-CS-09 has been completed. Tech Specs 3.6.3 Action **b** applies.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 2 (continued)

The following equipment is under clearance (continued):

- 1CS-9, Letdown Orifice Isolation valve Tech Spec (continued)



CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

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

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

- ‘A’ Gland Seal Exhauster Fan is under clearance for high vibrations on the motor bearing. The fan has been under clearance for 8 hours. Repairs are expected to be completed within 24 hours.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 2 (Continued)

Event 1: Power ascension from 53% power (GP-005). Turnover takes place with the unit at 53% Reactor power. The crew will be given credit for a reactivity manipulation during the power ascension.

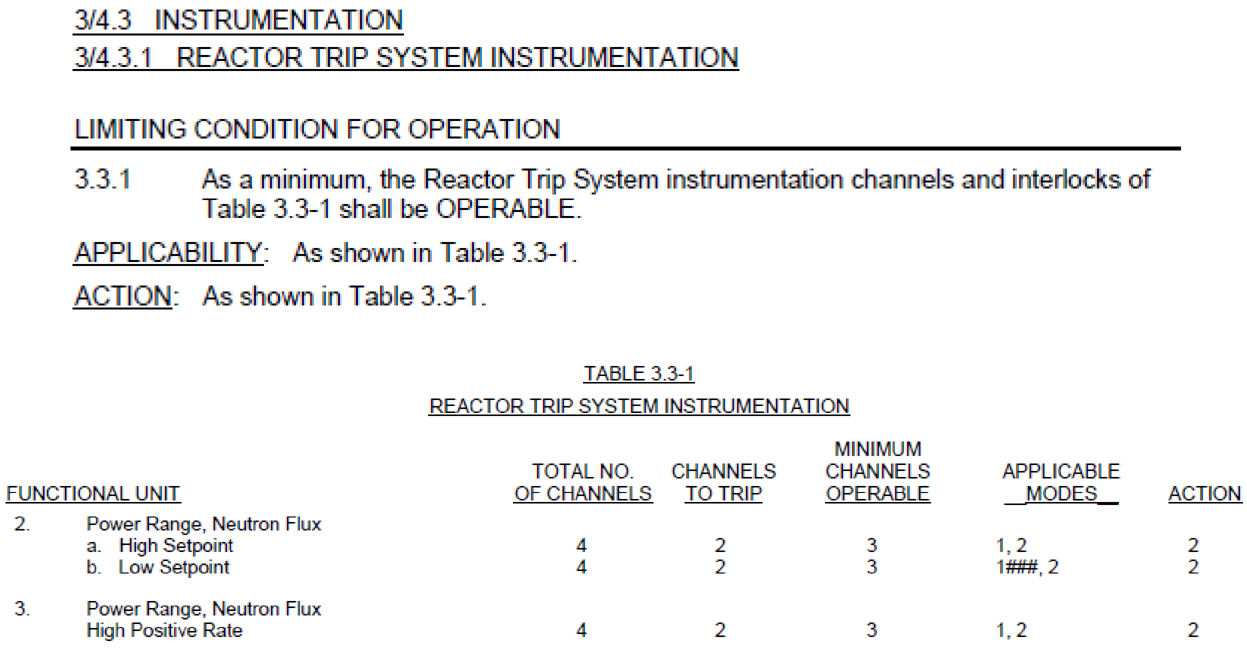
Verifiable Action: It is expected that the SRO will conduct a reactivity brief, the OATC will borate and monitor auto rod insertion per the reactivity plan. The BOP will operate the DEH Turbine controls as necessary to raise power. After power is raise 3% - 5% and the crew has demonstrated that they have control of the plant Event 2 may be inserted.

Event 2: PRNIS Channel NI-42 fails HIGH (AOP-001). NI-42 along with the Rod Control system MCB response will provide indications of the malfunction. Multiple ALB 013 annunciator window associated with the Power Range Nuclear Instruments will alarm.

Verifiable Action: The crew will enter AOP-001 and carry out the immediate actions. The OATC will perform the immediate actions of AOP-001 by verifying that <2 rods are dropped (no rods have dropped), place Rod Control in MANUAL and then verify no rod motion. Once the immediate actions are complete the BOP should place the Turbine in Hold it stabilize the plant. The SRO should continue with the implementation of AOP-001 to bypass NI-42 and restore T_{avg} to match T_{ref} in order to return Rod Control to Auto.

The SRO should provide a temperature band of +/- 5°F to the OATC in accordance with OMM-001, Attachment 11, Control Bands And Administrative Limits. The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

The SRO should evaluate Tech Spec 3.3.1, Instrumentation – Reactor Trip System Instrumentation Action: 2.



3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 2 (Continued)

Event 2: Tech Spec evaluation continued

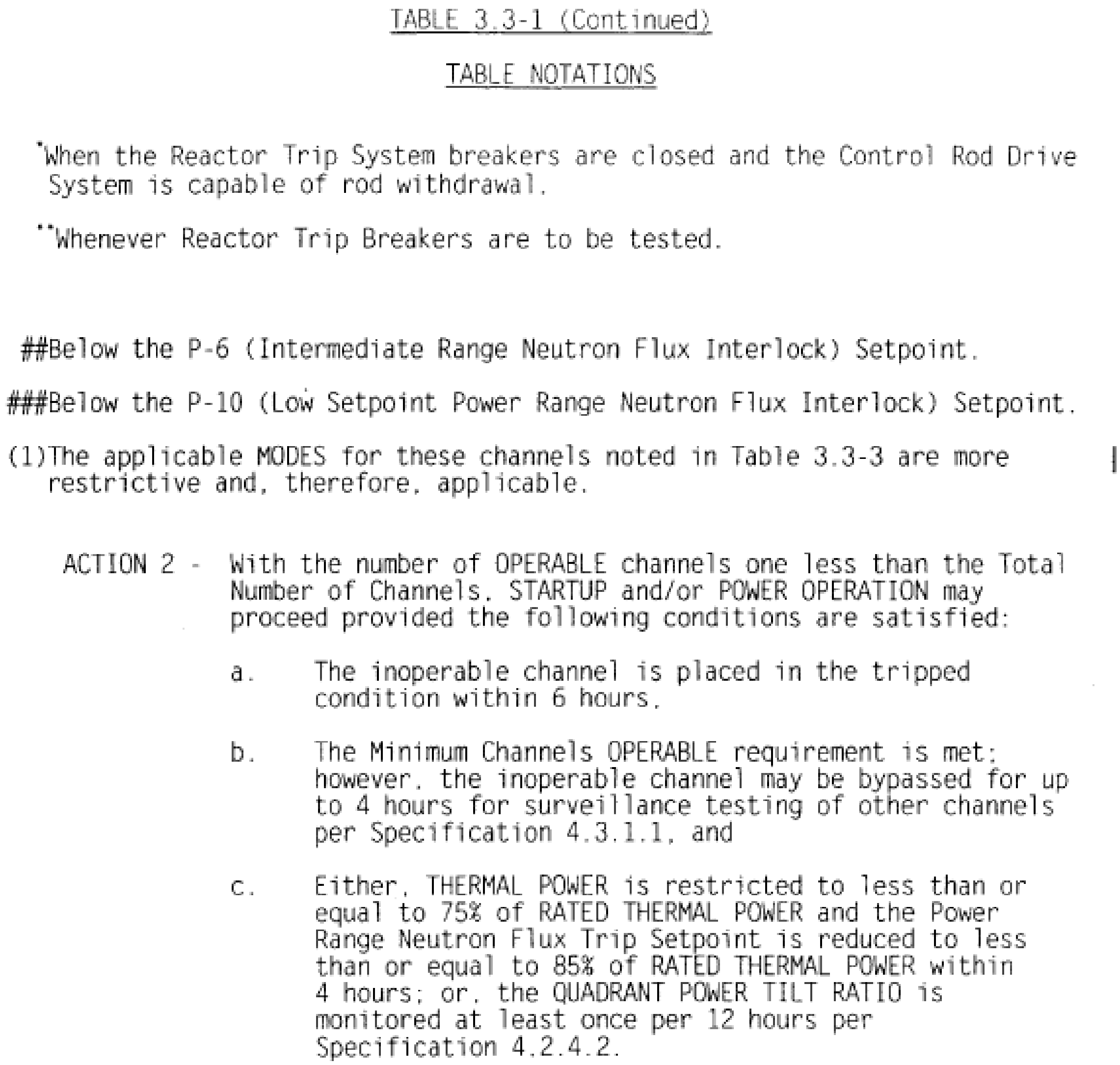


TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Whenever Reactor Trip Breakers are to be tested.

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

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

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

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

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

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

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

Event 3: Pressurizer Level Transmitter for LT-460 fails low. ALB 009-4-3, Pressurizer Low Level Ltdn Secured And Htrs Off, will alarm due to LT-460 being less than 17%.

Verifiable Action: The crew will respond by in accordance with APP-ALB-009 and verify all Pressurizer Heaters off and Letdown has automatically isolated. The OATC will select the 459/461 position on the MCB to restore two operable channels and reset each pressurizer heater as required. The BOP will ensure the failed channel is not the selected recorder channel.

The SRO should provide a level band of +/- 5% to the OATC in accordance with OMM-001, Attachment 11, Control Bands And Administrative Limits. The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists, for the failure and request assistance from the WCC.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 2 (Continued)

The SRO should evaluate Tech Spec 3.3.1, Instrumentation – Reactor Trip System Instrumentation Action: 6.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Whenever Reactor Trip Breakers are to be tested.

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

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

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

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

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

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

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 2 (Continued)

Event 4: Reactor Primary Shield Fan Failure. This will cause a loss of both fans on S-2. ALB 027-5-5, Reactor Primary Shield Clg Fans S2 Low Flow-O/L, will alarm along with ALB 001-6-5, Engineering Safeguard Features System Train A Bypassed Or Inoperable.

Verifiable Action: The crew will use the APP-ALB 027 to shift Primary Shield Cooling Fans to 'B' Train (S-2 1B-SB) in accordance with OP-169, Containment Cooling And Ventilation.

The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 5: DEH pump shaft shear and failure of the standby pump to start. The running pump will continue to indicate running until DEH pressure lowers to < 1600 psig when annunciator ALB-020-4-2B, DEH Fluid Low Press, will alarm. The crew should dispatch an Aux Operator to investigate the cause and confirm the standby pump is not affect by the failure.

Verifiable Action: The BOP will respond to the failure by taking actions contained in the APP-ALB 020-4-2B and ensuring the start the standby DEH pump **(Critical Task #1)**. If DEH pressure lowers to < 1500 psig the standby pump should auto start but a relay failure will prevent the pump from auto starting which will require the pump to be started manually. AD-OP-ALL-1000, Conduct of Operations, (5.6.3.8) for Equipment Manipulation and Status Control maybe used to start the standby DEH pump prior to 1500 psig.

The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 6: 'B' SG Controlling Level Transmitter fails Low (AOP-010). ALB 014-2-1B, 5-1A, 5-4B, SG B NR LVL/SP Hi/Lo Dev, SG B FW > Stm Flow Mismatch, and Steam Gen B Low-Low Level, respectively will alarm. The crew will respond by entering AOP-010, Feedwater Malfunction and taking manual control of 'B' Main Feedwater Regulating Valve to reduce Feedwater flow and stabilize level.

Verifiable Action: Taking manual control of 'B' Main Feedwater Regulating Valve to reduce Feedwater flow and stabilize level **(Critical Task #2)**. With the controller in manual and the plant stabilized the crew will implement OWP-RP-06 to remove the failed channel from service.

The SRO should provide a level band of 52% to 62% to the BOP in accordance with AOP-010 and OMM-001, Attachment 11, Control Bands And Administrative Limits. The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists, for the failure and request assistance from the WCC.

The SRO should evaluate Tech Spec 3.3.1, Instrumentation – Reactor Trip System Instrumentation and Tech Spec 3.3.2, Instrumentation – Engineered Safety Features Actuation System Instrumentation Action: 6 and 19 apply respectively.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 2 (Continued)

Event 6: Tech Spec evaluation continued

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Whenever Reactor Trip Breakers are to be tested.

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

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

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

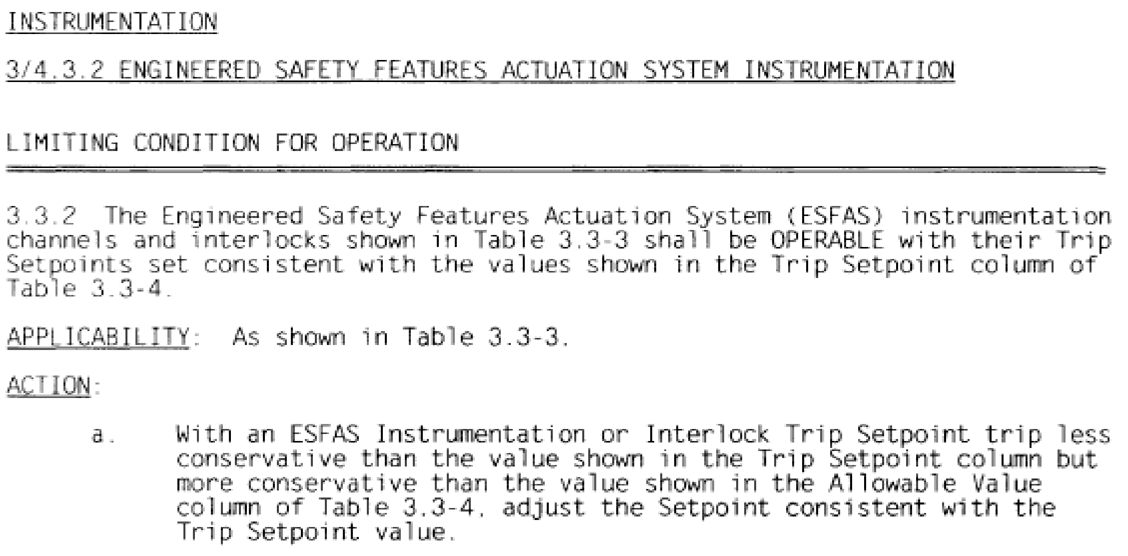
ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

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

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 2 (Continued)

Event 6: Tech Spec evaluation continued



INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the the Trip Setpoint value.

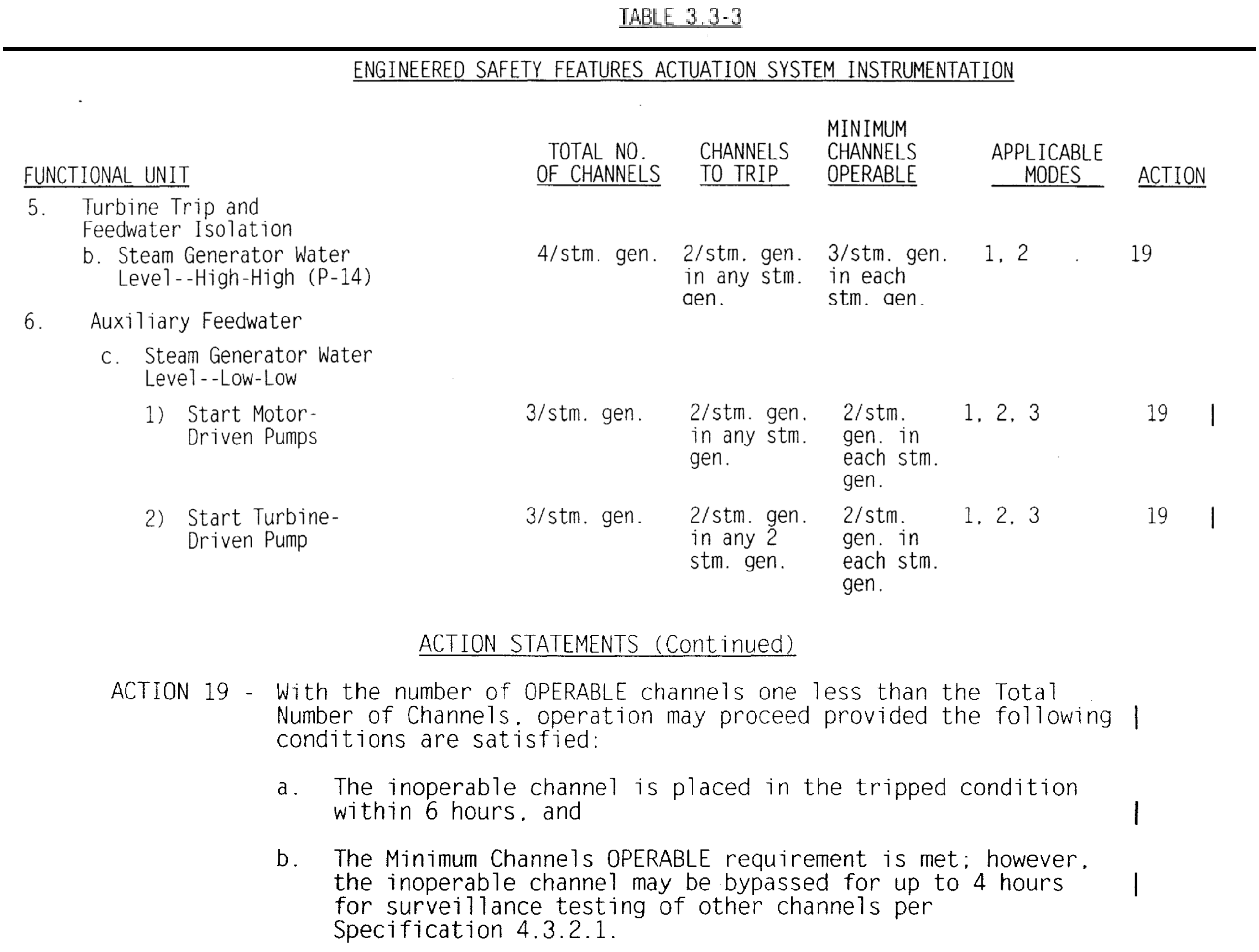


TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

ACTION STATEMENTS (Continued)

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

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

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

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 2 (Continued)

Event 7: Steam line Break on 'B' SG inside Containment (EOP-E-0 and EOP-E-2). The major event is a Steam line break. The RCS Loop ‘B’ will degrade into a break inside containment requiring the crew to implement the continuous actions for AOP-016 with leak rate in excess of VCT makeup ability and trip the Reactor and actuate Safety Injection. Major changes in Pressurizer Level and Charging flow will occur.

Verifiable Action: The OATC will manually trip the Reactor in accordance with AOP-016, then following verification of the Turbine trip actuate Safety Injection and the crew will continue with EOP-E-0. The crew will then transition from EOP-E-0 and go to EOP-E-2, Faulted Steam Generator Isolation. While the crew is performing actions of EOP-E-2 the Containment pressure will continue to rise beyond 10 psig which will actuate a Phase B isolation signal. This will require ALL RCPs to be secured.

All RCPs will need to be manually tripped within 10 minutes of a Phase B isolation signal. **(Critical Task #3)**

Event 8: ‘B’ Containment Spray pump fails to auto start. ‘B’ CT pump should auto start when Containment pressure is > 10 psig but does not.

Verifiable Action: The operator will first attempt to actuate Containment Spray using the MCB actuation switches but the actuation still does not occur requiring manual starting of the ‘B’ CT pump and alignment of the ‘B’ Train CT valves AD-OP-ALL-1000, Conduct of Operations, (5.6.3.8) for Equipment Manipulation and Status Control.

Event 9: 1SI-4 failure to close from MCB switch. While implementing EOP-E-2 the crew will be directed to reset SI and shut BIT outlet valves then establish a normal Charging lineup. When the crew attempts to shut 1SI-4 from the MCB the valve will not close.

Verifiable Action: The crew should identify this failure and direct an Aux Operator to locate and shut the 1SI-4 locally in accordance with the EOP-E-2 RNO step. Not shutting 1SI-4 prior to establishing a normal Charging lineup will cause simultaneous flow through the Charging and SI lines and cause a CSIP run out condition indicated by oscillating discharge pressure. **(Critical Task #4).**

The scenario termination is met in EOP-ES-1.1 when Safety Injection has been terminated and the crew restores letdown to service. With PZR level lowering and RCS Hot Leg Temperatures stable or lowering the RCS pressure challenge will be removed.

CRITICAL TASK JUSTIFICATION:

1. Manually start the standby DEH Pump prior to DEH pressure lowering below 1150 psig to prevent an automatic Turbine Trip/Reactor trip

An unnecessary automatic Reactor Trip for this event will create critical task. See note below.

2. Manually maintain control of SG 'B' level below 78% to prevent an automatic Reactor trip after steam generator level transmitter LT-486 fails low

An unnecessary automatic Reactor Trip for this event will create critical task. See note below.

3. Manually trip all RCPs within 10 minutes of a Phase B isolation signal

Securing RCPs during a large steam break inside Containment is procedurally required when Containment pressure has exceeded the High 3 setpoint of 10 psig. Exceeding this pressure causes a Phase B actuation which will isolate CCW flow to the RCP motor coolers. Operation of RCPs for greater than 10 minutes without CCW cooling to the motor oil coolers may result in RCP bearing damage.

4. Shut BIT Outlet valve 1SI-4 prior to establishing flow through the charging header.

Isolation of Safety Injection is required to allow the operator to stabilize RCS plant conditions. Eventually the Pressurizer will fill with water rendering pressurizer control ineffective. Consequently, in order to decrease RCS pressure to conserve makeup water, Safety Injection flow must be decreased. Because Safety Injection flow cannot be throttled, once the criteria to reduce Safety Injection flow is met Safety Injection is terminated by isolating Safety Injection flow, reducing to one CSIP in operation and realigning the CSIP discharge to the normal charging header. Shutting the BIT outlet valves is the first step in realigning normal charging to the RCS. Not shutting 1SI-4 prior to establishing a normal Charging lineup will cause simultaneous flow through the Charging and SI lines and cause a CSIP run out condition indicated by oscillating discharge pressure.

Note: Causing an unnecessary plant trip or ESF actuation may constitute a CT failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

Simulator Setup

Reset to IC-142 password "NRC3sros"

Go to RUN

Silence and Acknowledge annunciators

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner.

Set ERFIS screens for normal full power conditions

(The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

SPECIAL INSTRUCTIONS

Provide a Reactivity Plan to candidates for shutting down the plant

Provide a copy of the following procedures:

- GP-005, Power Operations, WN FROM POWER OPERATION TO HOT STANDBY (MODE 1 TO MODE 3) **marked up** through section 6.2 step 134

Press START on Counter Scaler
Post conditions for status board from IC-5 Reactor Power 538%
Control Bank D at 156 steps
RCS boron 1725 ppm

Turnover: The plant is at 53% power, beginning of core life. GP-005 step 134.e on hold for turnover. Once turnover is complete, raise TCS Load Control to 4 GVPC units/ min and continue the power ascension @ 4 MW/min.

Equipment Under Clearance:

- Boric Acid Transfer Pump B-SB is under clearance due to breaker blown control power fuses. Has been under clearance for 12 hours. The problem with the breaker has been repaired and the clearance will be removed later this shift. Tech Spec 3.3.3.5.b Action **c** and 3.1.2.2 applies (3.1.2.2 is for tracking only). OWP-CS-05 has been completed.
- 1CS-9, Letdown Orifice Isolation valve, has been under clearance the last 12 hours for solenoid replacement. The repairs are close to completion and the valve is expected to be returned to service within the next hour. The valve is currently shut with power removed. OWP-CS-09 has been completed. Tech Specs 3.6.3 Action **b** applies.
- 'A' Gland Seal Exhauster Fan is under clearance for high vibrations on the motor bearing. The fan has been under clearance for 8 hours. Repairs are expected to be completed within 24 hours.

Simulator Setup (continued)

Align equipment for repairs:

Place CIT on ‘A’ Boric Acid Transfer pump MCB Switch
Place protected train placards IAW OMM-001 Attachment 5
    Protected Train placards on "B" BA Transfer pump

Place CIT on ‘A’ Gland Steam Condenser Exhaust Fan MCB switch

Place CIT on 1CS-9 MCB switch

Place filled out copies of OWP’s into the OWP book – ensure they are removed at end of day

- OWP-CS-05 and place in MCR OWP book for "A" BA Transfer pump
- OWP-CS-09 and place in MCR OWP book for 1CS-9 clearance

Hang restricted access signs on MCR entry swing gates

Op Test No.:	NRC	Scenario #	2	Event #	1	Page	14	of	83
Event Description:					Power Ascension				

Time	Position	Applicant's Actions or Behavior

Lead Evaluator:	**The crew has been directed to re-commence a power ascension from 48% to the unit is at 100%. The power ascension is on hold for turnover. The SRO is expected to conduct a reactivity brief prior to commencing the power ascension. This brief may be conducted outside the simulator prior to starting the scenario.**
	When the crew has completed their board walk down and are ready to take the shift inform the Simulator Operator to place the Simulator in Run. When the Simulator is in run announce: **CREW UPDATE – (SRO's Name) Your crew has the shift. END OF UPDATE**

Simulator Operator:	**When directed by the Lead Evaluator, ensure that the annunciator horns are on and place the Simulator in RUN.**

Evaluator Note:		**The crew may elect to begin dilution prior to raising turbine load.**
	OATC	OP-107.01, Section 5.4
	OATC	• DETERMINE the volume of makeup water to be added. (Current OPT-1536 data may be used.)
	SRO	Directs dilution
Procedure Note:		**FIS-114 may be set for one gallon less than desired. A pressure transient caused by 1CS-151 shutting results in FIS-114 normally indicating one gallon more than actual flow but two gallons more would be unexpected.**

Op Test No.:	NRC	Scenario #	2	Event #	1	Page	15	of	83
Event Description:				**Power Ascension**					

Time	Position	Applicant's Actions or Behavior
Procedure Caution:		**If the translucent covers associated with the Boric Acid and Total Makeup Batch counters FIS-113 and FIS-114, located on the MCB, are not closed, the system will not automatically stop at the preset value.**
	OATC	SET FIS-114, TOTAL MAKEUP WTR BATCH COUNTER, to obtain the desired quantity.
	OATC	• ENSURE the RMW CONTROL switch has been placed in the STOP position. • ENSURE the RMW CONTROL switch green light is lit.
	OATC	IF the current potentiometer setpoint of controller 1CS-151, FK-114 RWMU FLOW, needs to be changed to obtain makeup flow, THEN PERFORM the following: (N/A)
		• RECORD the current potentiometer setpoint of controller 1CS-151, FK-114 RWMU FLOW, in Section 5.4.3. • SET controller 1CS-151, FK-114 RWMU FLOW, for the desired flow rate.
	OATC	PLACE control switch RMW MODE SELECTOR to the ALT DIL position.
Procedure Note:		• **Alternate Dilution may be manually stopped at any time by turning the control switch RMW CONTROL to STOP.**

Op Test No.:	NRC	Scenario #	2	Event #	1	Page	16	of	83
Event Description:				**Power Ascension**					

Time	Position	Applicant's Actions or Behavior
	OATC	• START the makeup system as follows: ○ TURN control switch RMW CONTROL to START momentarily. ○ ENSURE the RED indicator light is LIT. ○ IF expected system response is not obtained, THEN TURN control switch RMW CONTROL to STOP. • ENSURE dilution automatically terminates when the desired quantity has been added.
	OATC	IF controller 1CS-151, FK-114 RWMU FLOW, potentiometer was changed in Step 5.4.2.5, THEN PERFORM the following: (N/A)
		• REPOSITION controller FK-114 to the position recorded in Section 5.4.3. • INDEPENDENTLY VERIFY FK-114 potentiometer position of Step 5.4.2.9.a is correct.
	OATC	• Monitor Tavg and rod control for proper operation. • Establish VCT pressure between 20-30 psig. • Turn control switch RMW MODE SELECTOR to AUTO. • START the makeup system as follows: ○ TURN control switch RMW CONTROL to START momentarily. ○ ENSURE the RED indicator light is LIT. ○ IF expected system response is not obtained, THEN TURN control switch RMW CONTROL to STOP. (Ref. 4.0.31)
Evaluator Note:		**There is no procedural guidance directing when the dilution to raise power is required. The crew may elect to perform the raise prior to placing the Turbine in GO.**
	SRO	DIRECTS BOP to start power ascension at 4 MW/Min. May direct initiation of a dilution before the power ascension begins.

Op Test No.:	NRC	Scenario #	2	Event #	1	Page	17	of	83	
Event Description:				**Power Ascension**						

Time	Position	Applicant's Actions or Behavior
Evaluator Note:		**The following steps have been completed to achieve the current power level. The crew should validate status of the turbine load ascension in accordance with GP-005 section 6.2 step 108 and 110 before re-initiating the turbine load ascension.**
	BOP	Requests PEER check prior to manipulations of TCS Load Control screen
	BOP	On the TCS Load Control screen, Load Control section, perform the following: a. Select Ramp Rate Selection, Select button b. Select the desired ramp rate determined in Step 16.a OR Oper Entry • ENTER the desired rate, **(4 GVPC Units/minute)** c. IF Oper Entry is selected, THEN enter the desired ramp rate determined in Step 16.a in the Ramp Rate Entry window and depress Enter. • ENTER the desired rate, **(4 GVPC Units/minute)** • DEPRESS the ENTER push-button.
Procedure Note:		**If Oper Entry is selected with the Turbine in GO, the value currently in the Ramp Rate Entry Window will become the rate in effect. It may be desirable to place the turbine in HOLD to avoid undesirable rates.**
	BOP	WHEN ready to continue raising turbine load, THEN perform the following on TCS Load Control screen, Load Control section:
		a. IF 960 GVPC Units was NOT entered in the Target Entry window in Step 109.b, THEN enter 960 GVPC Units in the Target Entry window and depress Enter. **(960 GVPC Units)**. b. Select the Go button
	BOP	Ensure Generator load is rising

Op Test No.: NRC Scenario # 2 Event # 1		Page 18 of 83
Event Description:		**Power Ascension**
Time	Position	Applicant's Actions or Behavior

Evaluator Note:	**Once the crew demonstrates a satisfactory load ascension cue Simulator Operator to insert Trigger 2** **Event 2: PRNIS Channel NI-42 fails HIGH (AOP-001)**

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	19	of 83
Event Description:				PRNIS Channel NI-42 fails HIGH (AOP-001)				

Time	Position	Applicant's Actions or Behavior	
Simulator Operator:		**On cue from Lead Evaluator actuate Trigger 2** **"PRNIS Channel NI-42 fails HIGH (AOP-001)"**	
Indications Available		**• Uncontrolled inward rod motion** **• ALB-013-4-1, POWER RANGE HIGH NEUTRON FLUX HIGH SP ALERT** **• ALB-013-4-2, POWER RANGE HIGH NEUTRON FLUX HIGH ALERT** **• ALB-013-4-5, POWER RANGE CHANNEL DEVIATION** **• ALB-013-5-1, OVERPOWER ROD STOP** **• ALB-013-8-5, COMPUTER ALARM ROD DEV/SEQ NIS PWR RANGE TILTS**	
	OATC	RESPONDS to uncontrolled rod motion.	
AOP-001		Malfunction of Rod Control and Indication System	
	SRO	ENTERS and directs actions of AOP-001 Conducts a Crew Update Makes PA announcement for AOP entry	
	OATC	PERFORMS immediate actions.	
Immediate Action	OATC	CHECK that LESS THAN TWO control rods are dropped.	(YES)
Immediate Action	OATC	POSITION Rod Bank Selector Switch to MAN.	
Immediate Action	OATC	CHECK Control Bank motion STOPPED.	(YES)
	SRO	READS immediate actions and proceeds to Section 3.2. Directs BOP to place Turbine to HOLD if in GO.	

Op Test No.: NRC Scenario # 2 Event # 2 Page 20 of 83

Event Description: **PRNIS Channel NI-42 fails HIGH (AOP-001)**

Time	Position	Applicant's Actions or Behavior	
	BOP	Places Turbine to HOLD if in GO.	
	OATC	CHECK that instrument channel failure has NOT OCCURRED by observing the following: • RCS Tavg • RCS Tref • Power Range NI channels • Turbine first stage pressure	(NO) (NO) (YES) (NO)
	OATC	PERFORM the following: • IF a power supply is lost, THEN GO TO AOP-024, Loss of Uninterruptible Power Supply. • IF an individual instrument failed, THEN MAINTAIN manual rod control until corrective action is complete. • IF a Power Range NI Channel failed, THEN BYPASS the failed channel using OWP-RP.	(NO) (YES) (YES)
	BOP	Proceeds to the Detector Current Comparator Drawer and places NI-42 Rod Stop Bypass switch to BYPASS • Reports completion of task to the SRO.	
Procedure Note:		**Failure of RCS Median TAVE will cause an improper response of the PRESSURIZER AUTOMATIC LEVEL CONTROL and AUTOMATIC STEAM DUMP CONTROL systems.**	
	OATC	• IF RCS MEDIAN Tavg is failed THEN PERFORM the following: o ENSURE Charging FK-122.1 charging flow 1CS-231 is in manual and CONTROL charging to maintain pressurizer level. o ENSURE steam dumps are in Steam Pressure Mode using OP-126, section 5.3.	(NO) (N/A) (N/A)

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	21	of	83
Event Description:	**PRNIS Channel NI-42 fails HIGH (AOP-001)**								

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>OATC</td><td colspan="2">MANUALLY OPERATE affected control bank to restore the following:
• EQUILIBRIUM power and temperature conditions
• RODS above the insertion limits of Tech Spec 3.1.3.6 and PLP-106, Technical Specification Equipment List Program and Core Operating Limits Report.
• Withdraws Control Bank ‘D’ to restore Tave with Tref.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">• Directs RO to maintain TAVG within 5°F of Tref per OMM-001 attachment 11.
<table>
<tr><th rowspan="2">Controller</th><th rowspan="2">Control Band</th><th colspan="2">Administrative Limit</th></tr>
<tr><th>Low</th><th>High</th></tr>
<tr><td>Rod Control Stable Plant</td><td>T Avg within 2° of T Ref</td><td>T Avg Within 10° of T Ref</td><td>T Avg Within 10° of T Ref</td></tr>
<tr><td>Rod Control Transient Plant</td><td>T Avg within 5° of T Ref</td><td>T Avg Within 10° of T Ref</td><td>T Avg Within 10° of T Ref</td></tr>
</table></td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Evaluator Note:</td><td colspan="2">The following will be done when Tave is restored.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>OATC</td><td>VERIFY proper operation of the following:
• CVCS demineralizers
• BTRS
• REACTOR Makeup Control System</td><td>
(YES)
(N/A)
(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td>CHECK that this section was entered due to control banks MOVING OUT.</td><td>(NO)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td>CHECK that NEITHER of the following OCCURRED:
• Unexplained RCS Boration
• Unplanned RCS dilution</td><td>
(NO)
(NO)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
</table>

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	22	of	83
Event Description:		**PRNIS Channel NI-42 fails HIGH (AOP-001)**							

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td colspan="2">Procedure Note:</td><td colspan="2">Failure of RCS Median TAVE will cause an improper response of the PRESSURIZER AUTOMATIC LEVEL CONTROL and AUTOMATIC STEAM DUMP CONTROL systems.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td>CHECK that spurious rod motion is due to malfunction of the Automatic Rod Control System.
NO – RNO
GO TO Step 9.</td><td>(NO)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">EXIT this procedure.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">Exits AOP-001</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td>OWP-RP-24</td><td>SRO</td><td colspan="2">Refer to OWP-RP-24 to remove channel from service.</td></tr>
<tr><td></td><td>SRO</td><td colspan="2">• Direct operator and I&C to perform OWP-RP-24
• Completes an Emergent Issue Checklists for the failure of NI-42.
• Contacts WCC for assistance (WR, LCOTR and Maintenance support)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Simulator Communicator:</td><td colspan="2">Acknowledge request and reports from SRO.
IF asked to report to MCR to perform OWP-RP-24 state that you will report as soon as possible.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Simulator Operator:</td><td colspan="2">It is not required to implement the OWP prior to continuing with the scenario.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Evaluator Note:</td><td colspan="2">Any Tech Spec evaluation may be completed with a follow-up question after the scenario.</td></tr>
</table>

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	23	of	83
Event Description:	**PRNIS Channel NI-42 fails HIGH (AOP-001)**								

Time	Position	Applicant's Actions or Behavior
	SRO	Enters Instrumentation TS 3.3.1 Functional Unit 2, and 3 ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels. STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: a. The inoperable channel is placed in the tripped condition within 6 hours. b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1. and c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or,. the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2 **Reference the below T.S. but it will not apply for this conditions because 3 instruments is the Minimum Number required** 3.3.1 Functional Unit 19 b, c, and d. ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.
Evaluator's Note:		**When Tavg is restored and AOP-001 exited, cue Simulator Operator to insert Trigger 3** **Event 3: Pressurizer Level Transmitter for LT-460 fails low**

Op Test No.:	NRC	Scenario #	2	Event #	3	Page	24	of 83
Event Description:	**Pressurizer Level Transmitter for LT-460 fails low**							

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 3 "Pressurizer Level Transmitter for LT-460 fails low"**
Indications Available:		**• ALB-009-4-3, PRESSURIZER LOW LEVEL LTDN SECURED AND HTRS OFF** **• LI-460, Pressurizer Level Indication** **• FI-150.1, Letdown Flow Indication**
	RO	Responds to ALB-009-4-3 or indication of a failed Pressurizer Level Channel on LI-460.
APP-ALB-009	SRO	Enters APP-ALB-009-4-3
Evaluator Note:		Operator may use AD-OP-ALL-1000 guidance to take manual control of charging to avoid a trip or transient prior to the SRO direction.
	RO	CONFIRM alarm using: • Pressurizer level LI-459A1, LI-460, LI-461.1 (LI-460 low) • Letdown flow FI-150.1
	RO	VERIFY Automatic Functions: • All pressurizer heaters off • Letdown isolated
	SRO	• Directs RO to maintain controlling band +/- 5% of reference level per OMM-001 attachment 11.

Controller	Control Band	Administrative Limit	
		Low	High
Pressurizer Level	Within 5% of Reference Level	10%	75%

Op Test No.: NRC	Scenario # 2	Event # 3		Page	25	of	83
Event Description:	**Pressurizer Level Transmitter for LT-460 fails low**						

Time	Position	Applicant's Actions or Behavior	
	RO	PERFORM Corrective Actions: • IF PRZ level is low, THEN VERIFY letdown is isolated AND heaters are off. • IF RCS leakage is indicated, THEN GO TO AOP-016, Excessive Primary Plant Leakage. • IF alarm is due to malfunction of level control system, THEN MANUALLY RESTORE normal level. **(LT-459 is controlling channel for PZR level)** • IF the alarm is due to a failed level instrument o USING the Pressurizer Level Controller Selector switch, THEN SELECT a position which places the two operable channels into service. **(Select channels 459/461)** o VERIFY the failed channel is not selected, at the MCB recorder panel. o RESET the control heaters by placing the control switch to OFF and then back to ON. • IF maintenance is to be performed, THEN REFER TO OWP-RP, Reactor Protection.	(YES) (NO) (NO) (YES)
	RO	SELECT 459/461 on Pressurizer Level Controller Selector	
Evaluator Note:		**Any Tech Spec evaluation may be completed with a follow-up question after the scenario.**	
	SRO	Enters Instrumentation TS <u>3.3.1 Functional Unit 11</u> ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels. STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: a. The inoperable channel is placed in the tripped condition within 6 hours. b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.	

Op Test No.:	NRC	Scenario #	2	Event #	3	Page	26	of 83
Event Description:	**Pressurizer Level Transmitter for LT-460 fails low**							

Time	Position	Applicant's Actions or Behavior
	SRO	• Completes an Emergent Issue Checklists for the failure of LI-460. • Contacts WCC for assistance (WR, LCOTR and Maintenance support)
Simulator Communicator		**Acknowledge request.**
Evaluator's Note:		**Once the crew has taken manual control of Charging FCV-122 and selects an alternate controlling Pressurizer channel normal letdown flow may be restored** **IF desired to observe the restoration of normal letdown the actions have been listed on pages 27-31.** **IF desired to have normal letdown remain isolated continue to page 32 and cue Simulator Operator to insert Trigger 4** **Event 4: Reactor Primary Shield Fan Failure**

Op Test No.:	NRC	Scenario # 2	Event # 3	Page	27	of	83
Event Description:	**Restore letdown IAW OP-107, Chemical and Volume Control System**						

Time	Position	Applicant's Actions or Behavior
OP-107		OP-107, Section 5.4
	RO	Verifies Initial Conditions: • Charging flow has been established per Section 5.3 (YES) • Pressurizer level is greater than 17% (YES) • The following valves are shut: (YES) ○ 1CS-7, 45 GPM Letdown Orifice A ○ 1CS-8, 60 GPM Letdown Orifice B ○ 1CS-9, 60 GPM Letdown Orifice C
Procedure Caution:		If Charging flow was stopped or greatly reduced prior to letdown being secured, there is a possibility that the Letdown line contains voids due to insufficient cooling. This is a precursor to water hammer, and should be evaluated prior to initiating letdown flow.
	RO	VERIFY 1CC-337, TK-144 LTDN TEMPERATURE, controller is: • In AUTO AND • set for 110 to 120 ˚F (4.0 to 4.7 on potentiometer) normal operation OR • set for 90 to 120 ˚F (2.67 to 4.7 on potentiometer) if operating per Section 8.11
Procedure Note:		PK-145.1 LTDN PRESSURE, 1CS-38, may have to be adjusted to control at lower pressures.
	RO	VERIFY 1CS-38 Controller, PK-145.1 LTDN PRESSURE: • in MAN • output set at 50%
		VERIFY open the following Letdown Isolation Valves: • 1CS-2, LETDOWN ISOLATION LCV-459 • 1CS-1, LETDOWN ISOLATION LCV-460

Op Test No.: NRC	Scenario # 2	Event # 3	Page 28 of 83
Event Description:	**Restore letdown IAW OP-107, Chemical and Volume Control System**		

Time	Position	Applicant's Actions or Behavior
	RO	VERIFY open 1CS-11, LETDOWN ISOLATION.
Procedure Note:		The following table gives the minimum charging flow required to keep the regenerative heat exchanger temperature below the high temperature alarm when letdown is established:

Letdown flow (to be established)	Minimum Charging Flow necessary when letdown is established
45 gpm	20 gpm
60 gpm	26 gpm
105 gpm	46 gpm
120 gpm	53 gpm

If Pressurizer level is above the programmed level setpoint, charging flow should be adjusted to a point above the minimum required to prevent regenerative heat exchanger high temperature alarm but low enough to reduce pressurizer level.

Time	Position	Applicant's Actions or Behavior
	RO	ADJUST controller 1CS-231, FK-122.1 CHARGING FLOW, as required to: • Maintain normal pressurizer level program • Keep regenerative heat exchanger temperature below the high temperature alarm when the desired letdown orifice is placed in service.
Procedure Note:		If CVCS Demins have cooled from normal operating temperature, an increased reactivity affect may be observed. Consideration may be given to increasing CVCS Demins to operating temperature by flushing them to the RHT prior to restoring letdown. TIS-250, Recycle evaporator Feed Demineralizer Temperature Switch, can be used to determine temperature during flushing to the RHT.

Op Test No.: NRC	Scenario # 2	Event # 3	Page 29 of 83
Event Description:	**Restore letdown IAW OP-107, Chemical and Volume Control System**		

Time	Position	Applicant's Actions or Behavior	
	RO	IF flushing CVCS Demins to the RHT is desired for increasing temperature, THEN PERFORM the following: • NOTIFY Radwaste Control Room that letdown flow will be diverted to the RHT. • PLACE 1CS-120, LETDOWN TO VCT/HOLDUP TANK LCV-115A to the RHT position.	(N/A)
Procedure Note:		Changes in Letdown flowrate will affect the displayed value for RM-3502A (Channel 2303) due to the detector's proximity to the LTDN line.	
	RO	OPEN an Orifice Isolation Valve (1CS-7, 1CS-8, 1CS-9) for the orifice to be placed in service. ADJUST 1CS-38 position by adjusting PK-145.1 output as necessary to control LP LTDN Pressure (PI-145.1) at 340 to 360 psig, to prevent lifting the LP Letdown Relief.	
	RO	WHEN Letdown pressure has stabilized at 340 to 360 psig on PI-145.1, LP LTDN PRESS, THEN PERFORM the following: • ADJUST PK-145.1 LTDN PRESSURE setpoint to 58% • PLACE the controller in AUTO. VERIFY PK-145.1 LTDN PRESSURE Controller maintains Letdown pressure stable at 340 to 360 psig.	
	RO	IF Step 5.4.2.6 was performed AND CVCS Demin temperature is at normal operating temperature, THEN PERFORM the following: • PLACE 1CS-120, LETDOWN TO VCT/HOLDUP TANK LCV-115A to the AUTO position. • NOTIFY Radwaste Control Room that diversion to the RHT has been terminated.	(N/A)

<table>
<tr><td colspan="2">Op Test No.: NRC</td><td>Scenario # 2 Event # 3 Page 30 of 83</td></tr>
<tr><td colspan="2">Event Description:</td><td>Restore letdown IAW OP-107, Chemical and Volume Control System</td></tr>
<tr><td>Time</td><td>Position</td><td>Applicant’s Actions or Behavior</td></tr>
</table>

<table>
<tr><td colspan="2">Procedure Note:</td><td>Changes in Letdown flowrate will affect the displayed value for RM-3502A (Channel 2303) due to the detector's proximity to the LTDN line.</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>RO</td><td>OPEN additional orifice isolation valves (1CS-7, 1CS-8, 1CS-9) as required.
ADJUST charging flow as necessary to:
• prevent high temperature alarm (per table above)
• maintain pressurizer programmed level.</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Evaluator Note:</td><td>Placing LK-459F in AUTO may take several minutes due to matching PRZ level to reference level.</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>RO</td><td>PLACE PRZ level controller, LK-459F, in AUTO, as follows:
• PLACE PRZ level controller, LK-459F, in MAN to cancel any integrated signal.
• Record FI-112A.1, Charging flow
• Determine PRZ level controller , LK-459F setpoint by using the table below:
<table>
<tr><td>LTDN Flow</td><td>Charging Flow</td><td>LK-459F Setpoint (approx. value)</td></tr>
<tr><td>45 gpm</td><td>27 gpm</td><td>*3%</td></tr>
<tr><td>60 gpm</td><td>42 gpm</td><td>*8%</td></tr>
<tr><td>105 gpm</td><td>87 gpm</td><td>*34%</td></tr>
<tr><td>120 gpm</td><td>102 gpm</td><td>*46%</td></tr>
<tr><td colspan="3">* Approximate values based on NOT/NOP</td></tr>
</table>
• ADJUST PRZ level controller, LK-459F, to the calculated setpoint.
• Place PRZ level controller, LK-459F, in AUTO</td></tr>
</table>

Op Test No.:	NRC	Scenario #	2	Event #	3	Page	31	of 83
Event Description:	**Restore letdown IAW OP-107, Chemical and Volume Control System**							

Time	Position	Applicant's Actions or Behavior
	RO	WHEN the following occurs: • Program pressurizer level is matching the current pressurizer level AND • Letdown and seal return are balanced with seal injection flow and charging flow. THEN place controller 1CS-231, FK-122.1 CHARGING FLOW, in AUTO. COMPLETE Section 5.4.3. (Position Verification)
Lead Evaluator:		**After the actions to restore Normal Letdown are complete, cue Simulator Operator to insert Trigger 4** **Event 4: Reactor Primary Shield Fan Failure**

Op Test No.:	NRC	Scenario #	2	Event #	4	Page	32	of 83
Event Description:	**Reactor Primary Shield Fan Failure**							

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 4 "Reactor Primary Shield Fan Failure"**
Indications Available:		• **ALB-027-5-5, REACTOR PRIMARY SHIELD CLG FANS S2 LOW FLOW-O/L** • **S-2 Fan control switch indicating lights:** ○ **White light on indicates thermal overload**
ALB-027	BOP	ENTERS APP-ALB-027-5-5
	BOP	CONFIRM alarm using
		Control switch indicating lights: • White light ON indicates thermal overload • All indication lost indicates power supply de-energized
	SRO	VERIFY Automatic Functions: None
	BOP	PERFORM Corrective Actions: • START the standby Primary Shield Cooling fan per OP-169, Containment Cooling and Ventilation. (YES)
	BOP	DISPATCH an operator to check the status of the following breakers: • 1A21-SA-4C, S-2 (1A-SA) Primary Shield Cooling Fan (both breakers)
Simulator Communicator:		**If dispatched to investigate the breaker, report back (in 1- 2 mins) that both breakers for 1A21-SA-4C, S-2 1A-SA Primary Shield Cooling Fan are closed with the Voltage Vision is de-energized. No other abnormalities evident.**
	SRO/BOP	IF the breaker has tripped, OR has a thermal overload, THEN ENSURE that the cause of the trip has been investigated and corrected prior to resetting breaker.

Op Test No.:	NRC	Scenario #	2	Event #	4	Page	33	of 83

Event Description: **Reactor Primary Shield Fan Failure**

Time	Position	Applicant's Actions or Behavior
Evaluator Note:		**Due to the slow nature of the EHC system depressurizing (approximately 5-6 minutes) and only one MCB indication for pressure the crew may not notice a pressure reduction until the annunciator for EH fluid low pressure alarms.** ***For efficiency the Lead Evaluator should cue Trigger 5 once the local actions to evaluate the status of the breakers on 1A21 have been dispatched by the BOP.***
	SRO	• Completes an Emergent Issue Checklists for the failure of S-2A. • Contacts WCC for assistance (WR, and Maintenance support)
Lead Evaluator:		**After the crew has restored Reactor Primary Shield Cooling, cue Simulator Operator to insert Trigger 5** **Event 5: DEH pump shaft shear and failure of the standby pump to start**

Op Test No.: NRC	Scenario # 2	Event # 5	Page 34 of 83
Event Description:	**DEH pump shaft shear and failure of the standby pump to start**		

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 5** **"DEH pump shaft shear and failure of the standby pump to start"**
Evaluator Note:		**Due to the slow nature of the EHC system depressurizing (approximately 5-6 minutes) and only one MCB indication for pressure the crew may not notice a pressure reduction until the annunciator for EH fluid low pressure alarms**
Available Indications:		• **ALB-020-4-2B, EH FLUID LOW PRESS** • **PI-4221 lowering trend**
	BOP	Responds to ALB-20-4-2B or indication of degrading EHC pressure on PI-4221
ALB-020	BOP	Enters APP-ALB-020-4-2B
	BOP	CONFIRM alarm using
		• PI-4221, DEH Fluid Pressure indication • PI-4220A and PI-4220B, Local DEH Pump discharge pressure indicators
	BOP	VERIFY Automatic Functions:
		• Standby DEH Pump starts at 1500 psig, as sensed by PS-01TA-4223V (NO)
Evaluator Note:		**The BOP may immediately start the standby pump or wait until after reading the APP and the report from the AO. IF pressure is allowed to continue to lower when pressure reaches 1150 psig the Main Turbine will trip.**

Op Test No.:	NRC	Scenario # 2	Event # 5	Page	35	of	83
Event Description:		**DEH pump shaft shear and failure of the standby pump to start**					

Time	Position	Applicant's Actions or Behavior	
Critical Task #1	BOP	Starts EHC Pump 'B' and observes pressure returning to normal on PI-4221. Critical to start the standby DEH Pump prior to DEH pressure lowering below 1150 psig to prevent an automatic Turbine Trip/Reactor trip. (ALB-018 window 3-4, Turbine Trip Auto Stop Oil Trip)	
	BOP	PERFORM Corrective Actions:	
		• IF the Reactor is tripped, THEN GO TO EOP-E-0. • START the standby DEH Pump. ○ **Manually starts standby DEH Pump** • IF EH fluid pressure drops to 1500 psig, THEN INITIATE a rapid plant shutdown using AOP-038, Rapid Downpower, while continuing with this procedure.	(NO) (NO) (NO)
	BOP	• DISPATCH an operator to perform the following: ○ MONITOR DEH Pump and PCV operation. ○ VERIFY OPEN the following: ▪ 1EH-1, A EH Pump Suction Vlv ▪ 1EH-8, B EH Pump Suction Vlv ▪ 1EH-31, Main Hdr Press Switch Isol Vlv ○ INVESTIGATE system for leaks. ○ IF a leak is found, THEN ISOLATE the leak AND IMMEDIATELY NOTIFY Control Room.	(NO)
		Dispatches AO to investigate failure of EHC Pump 'A'.	
Simulator Communicator:		**When dispatched to investigate, report the 'A' EHC Pump shaft is sheared and not producing any discharge pressure.**	
	SRO	• Completes an Emergent Issue Checklists for the failure of DEH Pump A. • Contacts WCC for assistance (WR, and Maintenance support)	

Op Test No.: NRC Scenario # 2 Event # 5		Page 36 of 83
Event Description:		**DEH pump shaft shear and failure of the standby pump to start**
Time	Position	Applicant's Actions or Behavior

Lead Evaluator:	**After the crew has restored DEH header pressure, cue Simulator Operator to insert Trigger 6** **Event 6: 'B' SG Controlling Level Transmitter fails Low (AOP-010)**

Op Test No.: NRC	Scenario # 2	Event # 6	Page	37	of	83
Event Description:	**SG 'B' Controlling Level Transmitter fails Low (AOP-010)**					

Time	Position	Applicant's Actions or Behavior	
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 6** **"SG 'B' Controlling Level Transmitter fails Low (AOP-010)"**	
Indications Available:		**ALB-014-2-1B, SG B NR LVL/SP HI/LO DEV** **ALB-014-5-1A, SG B FW > STM FLOW MISMATCH** **ALB-014-5-4B, STEAM GEN B LOW-LOW LVL** **SG 'B' levels rising**	
	BOP	RESPONDS to alarms and ENTERS AOP-010	
AOP-010		Feedwater Malfunctions	
	SRO	ENTERS and directs actions of AOP-010 Conducts a Crew Update Makes PA announcement for AOP entry	
Procedure Note:		**Steps 1 through 4 are immediate actions.**	
Critical Task # 2 **Immediate Action**	BOP	CHECK Feedwater Regulator valves operating properly. **RNO** PERFORM the following: • PLACE affected Feedwater Regulator valve(s) in MANUAL. Places SG 'B' Feedwater Reg valve in MANUAL • MAINTAIN Steam Generator level(s) between 52 and 62%. Checks SG level and operates manual controller to maintain level between 52%-62% ***Critical Task: Maintain control of SG 'B' level below 78% to prevent an automatic Reactor trip after the controlling level transmitter LT-486 fails low.*** IF Steam Generator level(s) cannot be controlled, THEN TRIP the Reactor AND GO TO EOP-E-0. **(Should be controlled)**	(NO)

Op Test No.:	NRC	Scenario #	2	Event #	6	Page	38	of 83
Event Description:	SG 'B' Controlling Level Transmitter fails Low (AOP-010)							

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td>Immediate Action</td><td>BOP</td><td>CHECK ANY Main Feedwater Pump TRIPPED

RNO
GO TO STEP 6</td><td>(NO)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td colspan="2">MAINTAIN ALL of the following:
• At least ONE Main Feedwater Pump RUNNING
• Main Feedwater flow to ALL Steam Generators
• ALL Steam Generator levels greater than 30%
Maintains all of the above</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td>CHECK Feedwater Regulator Valves operating properly in AUTO: (NO not 'B')
• Response to SG levels
• Valve position indication
• Response to feed flow/steam flow mismatch

RNO
PERFORM the following:
• IF automatic SG water level control can be restored by selecting out a failed instrument, THEN USE OP-134.01, Feedwater System, Section 8.10 to swap Steam Flow/Feed Flow Control and Recorder Channels and restore level control to automatic.
• REFER to Tech Spec 3.3.1 AND IMPLEMENT OWP-RP or OWP-ESF where appropriate.
• IF needed, THEN CONTROL feed flow to SGs using Main Feed Reg Valve Bypass FCVs</td><td>(NO)

(NO)

(NO)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">• Directs BOP to maintain controlling band of 52% to 62% per OMM-001 attachment 11.
<table>
<tr><th rowspan="2">Controller</th><th rowspan="2">Control Band</th><th colspan="2">Administrative Limit</th></tr>
<tr><th>Low</th><th>High</th></tr>
<tr><td>Steam Generator Level</td><td>52% to 62%</td><td>30%</td><td>73%</td></tr>
</table></td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Procedure Note:</td><td colspan="2">Inability to monitor one or more Safety System Parameters concurrent with a turbine runback of greater than 25%, requires a change of event classification per the HNP Emergency Plan. [C.2, C.3].</td></tr>
</table>

Op Test No.:	NRC	Scenario #	2	Event #	6	Page	39 of	83
Event Description:	**SG 'B' Controlling Level Transmitter fails Low (AOP-010)**							

Time	Position	Applicant's Actions or Behavior	
	BOP	CHECK turbine runs back less than 25% turbine load	(YES)
Procedure Note:		A feedwater train consists of a Condensate Pump, Condensate Booster Pump and Main Feedwater Pump.	
	SRO	GO TO the applicable section: EVENT: All Condensate/Feedwater flow malfunctions (other than pump trips) Section 3.1 Page 10	
	BOP	CHECK the following Recirc and Dump Valves operating properly in MODU: • Main Feedwater Pumps • Condensate Booster Pumps • Condensate Pumps • 1CE-293, Condensate Recirc • 1CE-142, Condensate Dump To CST Isolation Valve (SLB-4/7-1)	 (YES) (YES) (YES) (YES) (YES)
	BOP	CHECK the Condensate and Feedwater System INTACT.	
Procedure Note:		Pumps should be stopped in the order of higher to lower pressure. (To stop a Condensate Pump, stop a Main Feedwater Pump followed by a Condensate Booster Pump and then the Condensate Pump.)	
	BOP	CHECK pumps for NORMAL OPERATION	(YES)
	SRO	NOTIFY Load Dispatcher of ANY load limitations. (No load limitations so Dispatcher will not be called)	
	SRO	CHECK Reactor thermal power changed by less than 15% in any one hour period.	(NO)
	SRO	EXIT this procedure.	

Op Test No.:	NRC	Scenario #	2	Event #	6	Page	40 of 83
Event Description:	**SG 'B' Controlling Level Transmitter fails Low (AOP-010)**						

Time	Position	Applicant's Actions or Behavior
OWP-RP-06	SRO	Refer to OWP-RP-06 to remove channel from service.
	SRO	Contacts WCC for support, requests WR and LCOTR. Contacts I&C to have channel removed from service.
Simulator Communicator:		**Respond to crew requests.**
Evaluator Note:		**Any Tech Spec evaluation may be completed with a follow-up question after the scenario.**
	SRO	Enters Instrumentation TS <u>3.3.1 Functional Unit 13 and 14</u> ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels. STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: a. The inoperable channel is placed in the tripped condition within 6 hours. b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1. <u>3.3.2 Functional Unit 5 and 6</u> ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied : a. The inoperable channel is placed in the tripped condition within 6 hours, and b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

Op Test No.: NRC	Scenario # 2	Event # 6	Page	41 of 83	
Event Description:	**SG 'B' Controlling Level Transmitter fails Low (AOP-010)**				

Time	Position	Applicant's Actions or Behavior

Evaluator Note:	**Channel does NOT have to be removed from service using the OWP to continue the scenario. Once after SG level is under control and the TS has been identified, cue Simulator Operator to insert Trigger 7** **Event 7: Steam line Break on 'B' SG inside Containment (EOP-E-0 and EOP-E-2)**

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	42	of 83
Event Description:		**Steam line Break on 'B' SG inside Containment (EOP-E-0 and EOP-E-2)**						
Time	Position	Applicant's Actions or Behavior						

Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 7 "Steam line Break on 'B' SG inside Containment (EOP-E-0 and EOP-E-2)"**
Evaluator Note:		**The crew should identify the leak. The crew will enter E-0 and perform the immediate actions. The SRO may also direct a manual Steam Line Isolation. The crew should diagnose that a LOCA is NOT in progress and transition from E-0 to E-2, Faulted Steam Generator Isolation.** • **When SG 'B' pressure is < 100 psi of 'A' and 'C' SG (with MSLI) an AFW isolation signal will close the 'B' MD and TD AFW valves.** • **When Containment pressure > 3 psig the crew should identify 'Adverse Containment' conditions are required to be implemented.** • **When 1SI-4 is closed from the MCB it will fail to close requiring the RAB Aux Operator to locally close the valve** • **When Containment pressure exceeds 10 psig 'B' CT Pump should start but will NOT autostart. It must be manually started and aligned for spray.**
Indications Available		• ALB-028-5-1 CONTAINMENT AIR HIGH VACUUM will clear (if in due to earlier ESW Pump start) • ALB-028-8-5 COMPUTER ALARM VENTILATION SYSTEM • Rising pressure in Containment • Rising temperature in Containment • Rising SG steam flow • Tavg lowers • PRZ level and pressure lower • Power rises
Evaluator Note:		**The crew may go to AOP-042. They will not have time to make progress before requiring a trip.** **Depending on how timing the crew may or may not actuate a Manual Reactor trip based on conditions that will exceed an ESF actuation setpoint**

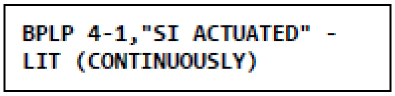
Op Test No.: NRC Scenario # 2 Event # 7 Page 43 of 83

Event Description: **Steam line Break on 'B' SG inside Containment (EOP-E-0)**

Time	Position	Applicant's Actions or Behavior	
	RO	(Time permitting – an auto Reactor Trip may occur prior to announcement) Informs SRO then actuates a Manual Reactor Trip	
	SRO	Directs manual Reactor Trip and Ensure Safety Injection activation	
EOP-E-0		Reactor Trip Or Safety Injection	
	SRO	Enters EOP-E-0 Makes plant PA announcement Conducts a Crew Update	
Immediate Action	RO	Verify Reactor Trip **Reactor Trip Confirmation** • Reactor Trip AND Bypass Bkrs - OPEN • Rod Bottom Lights (Zero Steps) - LIT • Neutron Flux - DROPPING	(YES)
Immediate Action	BOP	Check Turbine is Tripped – All throttle valves shut  TURB STOP VLV 1 | TSLB-2-11-1 TURB STOP VLV 2 | TSLB-2-11-2 TURB STOP VLV 3 | TSLB-2-11-3 TURB STOP VLV 4 | TSLB-2-11-4	(YES)

Op Test No.: NRC Scenario # 2 Event # 7 Page 44 of 83

Event Description: **Steam line Break on 'B' SG inside Containment (EOP-E-0)**

Time	Position	Applicant's Actions or Behavior	
Immediate Action	BOP	Perform The Following: • AC emergency buses ‐ AT LEAST ONE ENERGIZED • AC emergency buses – BOTH energized	(YES) (YES)
Immediate Action	RO	Safety Injection ‐ ACTUATED (BOTH TRAINS)  BPLP 4-1,"SI ACTUATED" - LIT (CONTINUOUSLY)	(YES)
	SRO	Perform The Following: • Review Foldout page and assign foldout. ▪ RCP Trip criteria ▪ Alternate Miniflow Open/Shut criteria ▪ RHR restart criteria ▪ Ruptured SG AFW Isolation criteria ▪ AFW supply switchover criteria	

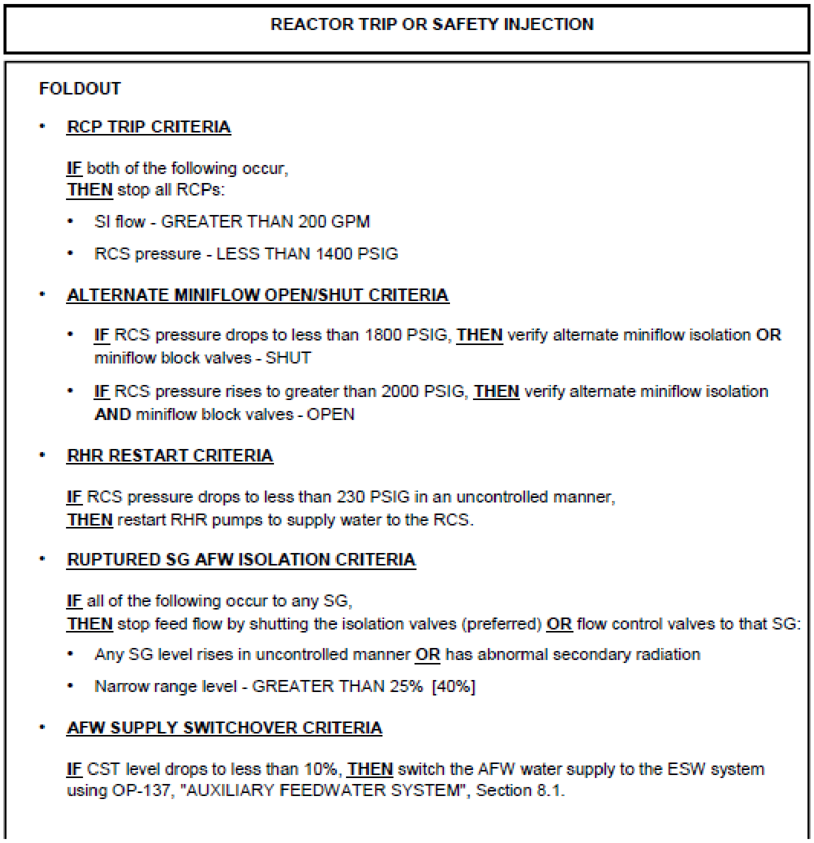
Op Test No.: NRC Scenario # 2 Event # 7 Page 45 of 83

Event Description: **Steam line Break on ‘B’ SG inside Containment (EOP-E-0)**

Time	Position	Applicant’s Actions or Behavior	

Evaluator Aide:

E-0 Foldout



REACTOR TRIP OR SAFETY INJECTION

FOLDOUT

- **RCP TRIP CRITERIA**

  IF both of the following occur,
  THEN stop all RCPs:

  - SI flow - GREATER THAN 200 GPM
  - RCS pressure - LESS THAN 1400 PSIG

- **ALTERNATE MINIFLOW OPEN/SHUT CRITERIA**

  - **IF** RCS pressure drops to less than 1800 PSIG, **THEN** verify alternate miniflow isolation **OR** miniflow block valves - SHUT
  - **IF** RCS pressure rises to greater than 2000 PSIG, **THEN** verify alternate miniflow isolation **AND** miniflow block valves - OPEN

- **RHR RESTART CRITERIA**

  IF RCS pressure drops to less than 230 PSIG in an uncontrolled manner,
  THEN restart RHR pumps to supply water to the RCS.

- **RUPTURED SG AFW ISOLATION CRITERIA**

  IF all of the following occur to any SG,
  THEN stop feed flow by shutting the isolation valves (preferred) **OR** flow control valves to that SG:

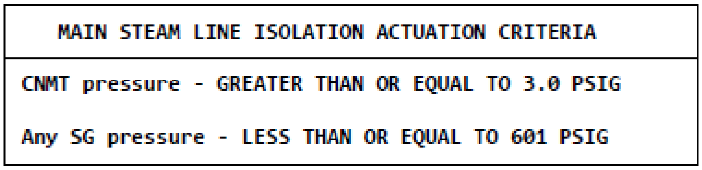
  - Any SG level rises in uncontrolled manner **OR** has abnormal secondary radiation
  - Narrow range level - GREATER THAN 25% [40%]

- **AFW SUPPLY SWITCHOVER CRITERIA**

  IF CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1.

	SRO	• Evaluate EAL Matrix.	
	CREW	Identifies Containment Adverse Conditions Containment Pressure > 3 psig	
	RO	Ensure CSIPs – ALL RUNNING ‘A’ and ‘B’ running	(YES)
	RO	Ensure RHR Pumps – ALL RUNNING ‘A’ and ‘B’ running	(YES)
	RO	Safety Injection flow > 200 gpm	(YES)

Op Test No.: NRC	Scenario # 2	Event # 7	Page 46 of 83		
Event Description:	**Steam line Break on ‘B’ SG inside Containment (EOP-E-0)**				

Time	Position	Applicant’s Actions or Behavior	
	SRO	RCS pressure LESS than 230 PSIG RNO GO to Step 12	(NO)
	BOP	Main Steam Line Isolation – ACTUATED  MAIN STEAM LINE ISOLATION ACTUATION CRITERIA CNMT pressure - GREATER THAN OR EQUAL TO 3.0 PSIG Any SG pressure - LESS THAN OR EQUAL TO 601 PSIG	(YES)
	BOP	Ensure All MSIVs AND Bypass Valves – SHUT	(YES)
	BOP	Any SG pressure - 100 PSIG LOWER THAN PRESSURE IN TWO OTHER SGs	(YES)
	BOP	Ensure MDAFW AND TDAFW Isolation Valves AND Flow Control Valves To Affected SG – SHUT Both MDAFW and TDAFW isolation valve and FCV to the ‘B’ SG • 1AF-93 • 1AF-143 • FCV-2071 B (1AF-130) • FCV-2051B (1AF-51)	(YES) (SHUT) (SHUT) (SHUT) (SHUT)

Op Test No.: NRC	Scenario # 2	Event # 8	Page 47 of 83
Event Description:	**Failure of 'B' Train Containment Spray Pump to actuate (EOP-E-0)**		

Time	Position	Applicant's Actions or Behavior	
	BOP	Check CNMT Pressure – HAS REMAINED LESS THAN 10 PSIG (YES/NO time dependent when YES) Perform the following: • Ensure Containment Spray – ACTUATED Identifies that the 'B' Containment Spray pump has not started and attempts to actuate Containment Spray using the MCB Containment Spray switches (2 per train) – Pump does not start Manually starts 'B' Containment Spray pump and aligns spray valves Opens 1CT-11 and 1CT-88	(NO)

Op Test No.: NRC	Scenario # 2	Event # 7	Page	48 of 83		
Event Description:	**Steam line Break on 'B' SG inside Containment (EOP-E-0)**					

Time	Position	Applicant's Actions or Behavior	
Critical Task #3	RO	Check CNMT Pressure – HAS REMAINED LESS THAN 10 PSIG (YES/NO time dependent when YES) Start time: **Time ALB-001-5-1, Containment Isolation Phase B, received at ______________** Perform the following: • Stop ALL RCP's ○ Locates MCB switches and STOPS ALL 3 RCP's **Time ALL RCP's secured: ______________** **Total time: _____________** ***Critical to have all RCP's secured in < 10 Minutes***	
	BOP	Ensure AFW flow - AT LEAST 200 KPPH ESTABLISHED	(YES)
	BOP	Sequencer Load Block 9 (Manual Loading Permissive) - ACTUATED (BOTH TRAINS)	(YES)
	BOP	Energize AC buses 1A1 AND 1B1	
Evaluator Note:		**E-0, Attachment 3 is located on page 67.**	
Evaluator Note:		**The RO will perform all board actions until the BOP completes Attachment 3. The BOP is permitted to properly align plant equipment IAW E-0 Attachment 3 without SRO approval.** **The Scenario Guide still identifies tasks by board position because the time frame for completion of Attachment 3 is not predictable.**	

Op Test No.: NRC	Scenario # 2	Event # 7	Page 49 of 83
Event Description:	**Steam line Break on 'B' SG inside Containment (EOP-E-0)**		

Time	Position	Applicant's Actions or Behavior
	BOP	Ensure Alignment Of Components From Actuation Of ESFAS Signals Using Attachment 3, "Safeguards Actuation Verification", While Continuing With This Procedure.
	BOP	Directs AO to place 1A and 1B Air Compressor in the local control mode per E-0 Attachment 3 step 22
Simulator Communicator		**Acknowledge the request to place 1A and 1B Air Compressor in the local control mode per E-0 Attachment 3 step 22**
Simulator Operator		**When directed to place the 1A and 1B Air Compressor in the local control mode: Run APP\air\acs_to_local**
Simulator Communicator		**When the APP for 1A and 1B Air Compressor has completed running call the MCR and inform them that the air compressors are running in local control.**
	BOP	Directs AO to Unlock AND Turn ON The Breakers For The CSIP Suction AND Discharge Cross-Connect Valves per E-0 Attachment 3 step 23 (or from step 11 - refer to Attachment 2)
Simulator Communicator		**Acknowledge the request to Unlock AND Turn ON The Breakers For The CSIP Suction AND Discharge Cross-Connect Valves**
Simulator Operator		**When directed to Unlock AND Turn ON The Breakers for the CSIP Suction AND Discharge Cross-Connect Valves: Run APP\cvc\E-0 Att 2 CSIP suct & disch valve power.txt.**
Simulator Communicator		**When the APP for CSIP Suction AND Discharge Cross-Connect Valves has completed running call the MCR and inform them that CSIP Suction AND Discharge Cross-Connect Valves are energized.**
Examiners Note:		**RCP's are secured therefore WR CL temperatures should be used when checking RCS temperature. RCS temp trend will be < 557°F and dropping – control FF, maintain total FF > 200 KPPH until SG level > 40% (all MSIV's are shut)**

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	50 of 83

Event Description: **Steam line Break on ‘B’ SG inside Containment (EOP-E-0)**

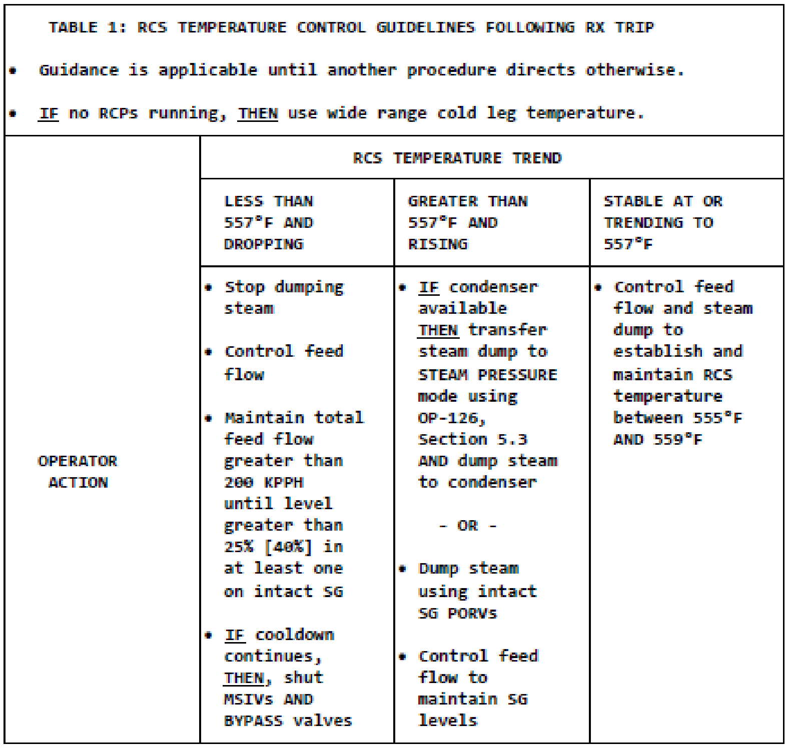
Time	Position	Applicant’s Actions or Behavior
	RO	Stabilize AND Maintain Temperature Between 555°F AND 559°F Using Table 1.
	RO	

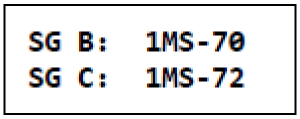
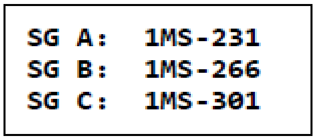
TABLE 1: RCS TEMPERATURE CONTROL GUIDELINES FOLLOWING RX TRIP

- Guidance is applicable until another procedure directs otherwise.
- IF no RCPs running, THEN use wide range cold leg temperature.

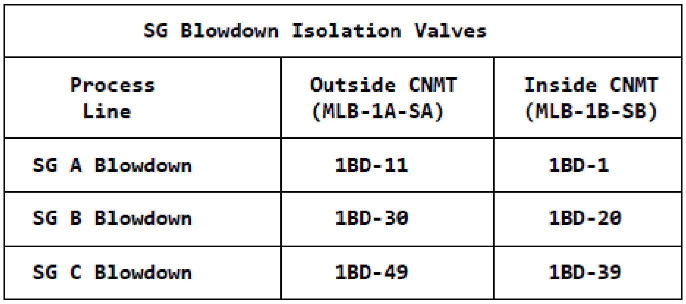
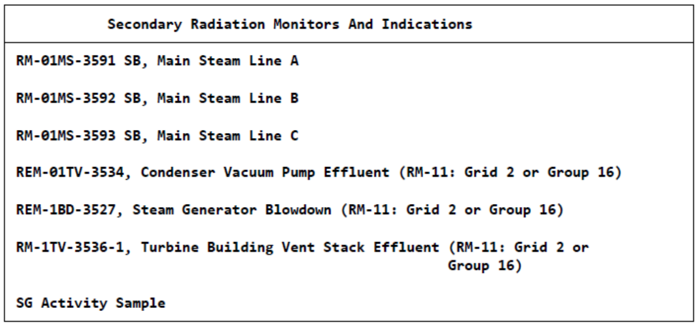
	RCS TEMPERATURE TREND		
	LESS THAN 557°F AND DROPPING	GREATER THAN 557°F AND RISING	STABLE AT OR TRENDING TO 557°F
OPERATOR ACTION	• Stop dumping steam • Control feed flow • Maintain total feed flow greater than 200 KPPH until level greater than 25% [40%] in at least one on intact SG • IF cooldown continues, THEN, shut MSIVs AND BYPASS valves	• IF condenser available THEN transfer steam dump to STEAM PRESSURE mode using OP-126, Section 5.3 AND dump steam to condenser - OR - • Dump steam using intact SG PORVs • Control feed flow to maintain SG levels	• Control feed flow and steam dump to establish and maintain RCS temperature between 555°F AND 559°F

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	51	of 83
Event Description:	**Steam line Break on 'B' SG inside Containment (EOP-E-2)**							

Time	Position	Applicant's Actions or Behavior	
	RO	PRZ PORVs – SHUT PRZ Spray Valves – SHUT (RCPs are secured) PRZ PORV Block Valves - AT LEAST ONE OPEN	(YES) (YES) (YES)
	SRO	Any SG pressure – DROPPING IN AN UNCONTROLLED MANNER OR COMPLETELY DEPRESSURIZED ('B' SG) GO TO E-2, "FAULTED STEAM GENERATOR ISOLATION", Step 1.	(YES)
EOP-E-2		Faulted Steam Generator Isolation	
		Enters EOP-E-2 Conducts a Crew Update	
		•	
Procedure Caution:		• At least one SG must be maintained available for RCS cooldown. • Any faulted SG OR secondary break should remain isolated during subsequent recovery actions unless needed for RCS cooldown.	
	SRO	Initiate Monitoring Of Critical Safety Function Status Trees.	
	BOP	Verify All MSIVs – SHUT Verify All MSIV bypass valves – SHUT	(YES) (YES)
	BOP	Check Any SG pressure - STABLE OR RISING (NOT FAULTED) **('A' and 'C' SG)**	(YES)
	BOP	Any SG pressure – DROPPING IN AN UNCONTROLLED MANNER OR COMPLETELY DEPRESSURIZED **('B' SG)**	(YES)

Procedure Caution:		IF the TDAFW pump is the only available source of feed flow, THEN maintain steam supply to the TDAFW pump from one SG.	
	BOP	Isolate Faulted SG(s) (Identified In Step 5): • Verify faulted SG(s) PORV – SHUT • Verify main FW isolation valves – SHUT (Automatically)	(YES) (YES)
	BOP	Ensure MDAFW AND TDAFW pump isolation valves to faulted SG(s) – SHUT • 1AF-93 • 1AF-143 (YES / NO time dependent – may have identified and isolated these valves in E-0)	(SHUT) (SHUT)
	BOP	Shut faulted SG(s) steam supply valve to TDAFW pump – SHUT  SG B: 1MS-70 SG C: 1MS-72 Shuts 1MS-70	(SHUT)
	BOP	Ensure main steam drain isolation(s) before MSIVs - SHUT:  SG A: 1MS-231 SG B: 1MS-266 SG C: 1MS-301	(YES)

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	53	of 83
Event Description:	**Steam line Break on ‘B’ SG inside Containment (EOP-E-2)**							

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant’s Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>Ensure SG blowdown isolation valves – SHUT

<table>
<tr><th colspan="3">SG Blowdown Isolation Valves</th></tr>
<tr><th>Process Line</th><th>Outside CNMT (MLB-1A-SA)</th><th>Inside CNMT (MLB-1B-SB)</th></tr>
<tr><td>SG A Blowdown</td><td>1BD-11</td><td>1BD-1</td></tr>
<tr><td>SG B Blowdown</td><td>1BD-30</td><td>1BD-20</td></tr>
<tr><td>SG C Blowdown</td><td>1BD-49</td><td>1BD-39</td></tr>
</table></td><td>(YES)
(YES)
(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td>Ensure main steam analyzer isolation valves – SHUT
Check CST Level - GREATER THAN 10%</td><td>(YES)
(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Procedure Note:</td><td colspan="2">A SG may be suspected to be ruptured if it fails to dry out following isolation of feed flow. Local checks for radiation can be used to confirm primary-to-secondary leakage.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>CREW</td><td>Any SG - ABNORMAL RADIATION OR UNCONTROLLED LEVEL RISE

<table>
<tr><th>Secondary Radiation Monitors And Indications</th></tr>
<tr><td>RM-01MS-3591 SB, Main Steam Line A</td></tr>
<tr><td>RM-01MS-3592 SB, Main Steam Line B</td></tr>
<tr><td>RM-01MS-3593 SB, Main Steam Line C</td></tr>
<tr><td>REM-01TV-3534, Condenser Vacuum Pump Effluent (RM-11: Grid 2 or Group 16)</td></tr>
<tr><td>REM-1BD-3527, Steam Generator Blowdown (RM-11: Grid 2 or Group 16)</td></tr>
<tr><td>RM-1TV-3536-1, Turbine Building Vent Stack Effluent (RM-11: Grid 2 or Group 16)</td></tr>
<tr><td>SG Activity Sample</td></tr>
</table>
RNO
Go to Step 10</td><td>(NO)</td></tr>
</table>

Op Test No.: NRC	Scenario # 2	Event # 7	Page	54 of 83
Event Description:		**Steam line Break on 'B' SG inside Containment (EOP-E-2)**		

Time	Position	Applicant's Actions or Behavior	
	RO	Check If SI Has Been Terminated: • Check for all of the following: ○ Check BIT outlet valves – SHUT OR ISOLATED ▪ 1SI-3 (OPEN) ▪ 1SI-4 (OPEN) **RNO** Go to step 13	(NO) (NO)
	BOP	Check SI Termination Criteria: • Check Subcooling – > 40°F • Level in at least one SG > 40%	(YES) (YES)
	RO	• RCS pressure – STABLE OR RISING • PRZ level - > 30% (YES / NO – time dependent action)	(YES)
Evaluator Note:		**PRZ level > 30%** **IF YES** then crew will continue with E-2 below **IF NO** then crew will transition to E-1 – the actions for E-1 follow E-2 (included later in guide)	
E-2 Continues	RO	Reset SI	
	Crew	Manually Realign Safeguards Equipment Following A Loss Of Offsite Power. (There is no loss of power – N/A)	
	RO	Resets Phase A AND Phase B Isolation Signals. (both were actuated)	

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	55 of 83

Event Description: **Steam line Break on 'B' SG inside Containment (EOP-E-2)**

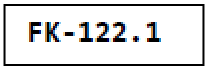
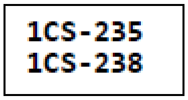
Time	Position	Applicant's Actions or Behavior	
	RO	Open Instrument Air AND Nitrogen Valves to Containment: 1IA-819 (ISOL VALVE CONT. BLDG 236' PENETRATION (M-80)) 1SI-287 (ACCUMULATOR & PRZ PORV N2 SUPPLY ISO VLV) Locates and OPENS both valves	
	RO	Stop all but ONE CSIP (STOPS A / B CSIP) RCS pressure – STABLE OR RISING	(YES)
	RO	Check CSIP suction - ALIGNED TO RWST VCT OUTLET (SHUT): 1CS-165 (LCV-115C), 1CS-166 (LCV-115E) RWST SUCTION (OPEN): 1CS-291 (LCV-115B), 1CS-292 (LCV-115D)	(YES)
	RO	Open Normal Miniflow Isolation Valves: CSIP A: 1CS-182 CSIP B: 1CS-196 CSIP C: 1CS-210 COMMON: 1CS-214 Locates controls and OPENS each valve	

Op Test No.: NRC Scenario # 2 Event # 9 Page 56 of 83

Event Description: **Failure of BIT outlet isolation valve 1SI-4 to close (EOP-E-2)**

Time	Position	Applicant's Actions or Behavior	
		Event 9 - Failure of 1SI-4 to close	
Critical Task #4	RO	Shut BIT Outlet Valves: 1SI-3 1SI-4 Shuts 1SI-3 from MCB switch Attempts to shut 1SI-4 will not SHUT from MCB switch Dispatches RAB Aux Operator to locally shut 1SI-4 (may also request that the breaker for the valve OPEN) Critical Task to shut BIT Outlet valve 1SI-4 prior to establishing flow through the charging header or CSIP run out conditions will occur as indicated by oscillating discharge pressure.	
Simulator Communicator:		**IF this valve has not been previously shut then:** **Acknowledge request to locally shut 1SI-4 (A-230-FX32-W3-S2) AND if requested acknowledge request to OPEN breaker prior to locally valve operation.** **Report back approximately 1 minute after Simulator Operator completes actions below that 1SI-4 is SHUT.**	
Simulator Operator -		**Perform the following actions from Sim Diagram SIS02 to operate 1SI-4:** **(IF requested) OPEN control power rf sis016** **Engage handwheel rf sis017** **Shut valve modify rf sis018**	
	RO	Verify Cold Leg AND Hot Leg Injection Valves – SHUT 1SI-52 1SI-86 1SI-107	(YES)

Op Test No.: NRC	Scenario # 2 Event # 7	Page 57 of 83
Event Description:	**Steam line Break on ‘B’ SG inside Containment (EOP-E-2)**	

Time	Position	Applicant’s Actions or Behavior	
Procedure Note:		High head SI flow should be isolated before continuing.	
	RO	Establish Charging Lineup: • Shut charging flow control valve:  FK-122.1 • Open charging line isolation valves:  1CS-235 1CS-238	(SHUT) (OPEN) (OPEN)
	RO	Monitor RCS Hot Leg Temperature: Check RCS hot leg temperature – STABLE (YES / NO – time dependent - probably rising) YES / NO – BOP action next step	
	BOP	**IF YES** – Manually dump steam AND control feed flow to maintain RCS temperature stable.	
	BOP	**IF NO** - If temperature rising, THEN manually dump steam from intact SG PORVs at maximum rate to stabilize temperature.	
Procedure Note:		RCS temperature must be stabilized to allow evaluation of PRZ level trend.	
	BOP	**IF NO** - WHEN temperature stabilizes, THEN manually dump steam AND control feed flow to maintain RCS temperature stable.	
Procedure Caution:		Charging flow should NOT exceed 150 GPM to prevent damage to the regenerative heat exchanger.	

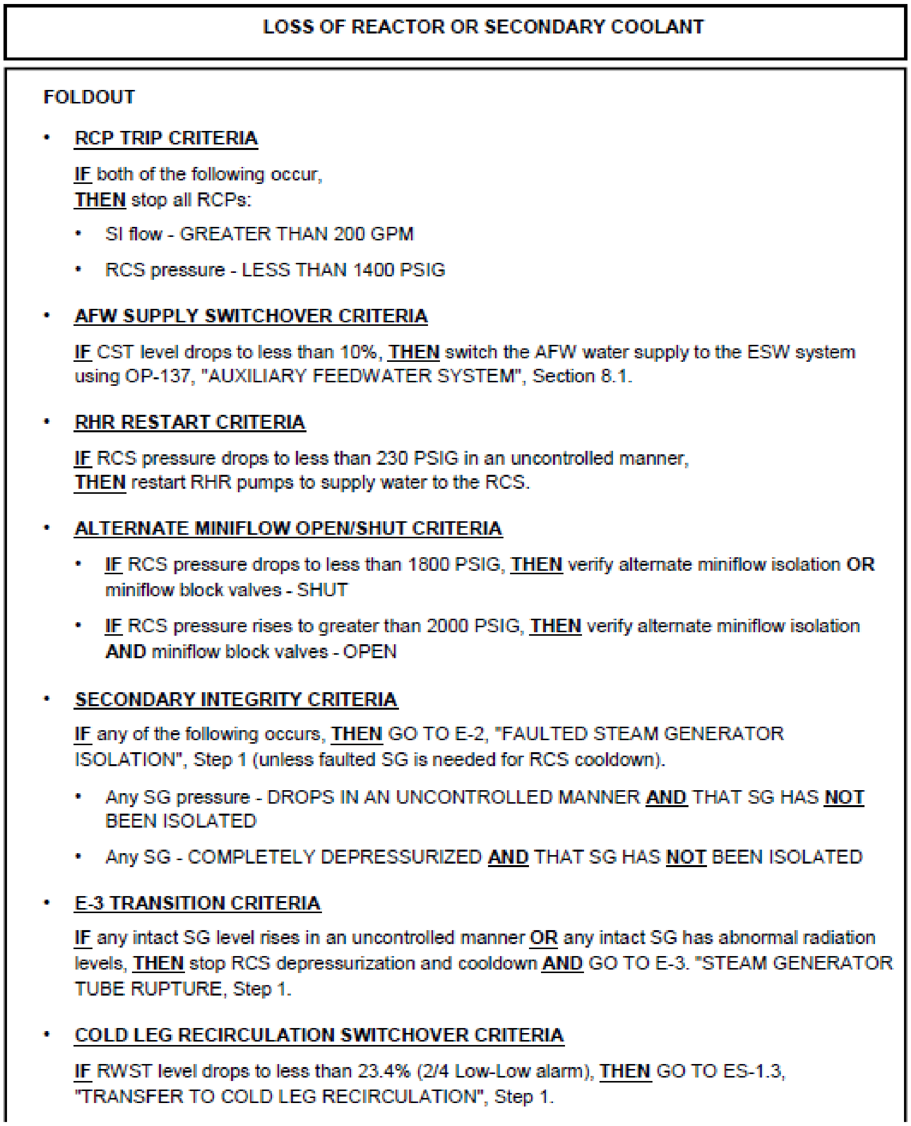
Op Test No.: NRC	Scenario # 2	Event # 7	Page	58 of 83	
Event Description:	**Steam line Break on ‘B’ SG inside Containment (EOP-E-1)**				

Time	Position	Applicant’s Actions or Behavior	
	RO	Control Charging Flow To Maintain PRZ Level: • Control charging using charging flow control valve: FK-122.1 • Maintain charging flow less than 150 GPM	
	RO	• PRZ level – CAN BE MAINTAINED STABLE OR RISING	(YES)
	SRO	GO TO ES-1.1, "SI TERMINATION", step 1	
Evaluator Note:		**IF the crew transitioned to EOP-E-1 based on PRZ level < 30% then continue on next page.** **If PRZ level is > 30% then continue with EOP- ES-1.1, SI Termination step 1 (see page 62 in this guide)**	
EOP-E-1		Loss of Reactor or Secondary Coolant	
Procedure Note:		Foldout applies	
	SRO	Assigns Foldout items to RO and or BOP RO: RCP Trip criteria, RHR Restart criteria, Alternate Miniflow Open/Shut criteria, Cold Leg Recirculation switchover criteria BOP: AFW supply switchover criteria, Secondary integrity criteria, E-3 transition criteria	

Evaluator Aide:	**E-1** Foldout

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	59 of 83

Event Description: **Steam line Break on 'B' SG inside Containment (EOP-E-1)**



LOSS OF REACTOR OR SECONDARY COOLANT

FOLDOUT

- **RCP TRIP CRITERIA**

  IF both of the following occur,
  THEN stop all RCPs:

  - SI flow - GREATER THAN 200 GPM
  - RCS pressure - LESS THAN 1400 PSIG

- **AFW SUPPLY SWITCHOVER CRITERIA**

  IF CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1.

- **RHR RESTART CRITERIA**

  IF RCS pressure drops to less than 230 PSIG in an uncontrolled manner,
  THEN restart RHR pumps to supply water to the RCS.

- **ALTERNATE MINIFLOW OPEN/SHUT CRITERIA**

  - **IF** RCS pressure drops to less than 1800 PSIG, **THEN** verify alternate miniflow isolation **OR** miniflow block valves - SHUT
  - **IF** RCS pressure rises to greater than 2000 PSIG, **THEN** verify alternate miniflow isolation **AND** miniflow block valves - OPEN

- **SECONDARY INTEGRITY CRITERIA**

  IF any of the following occurs, **THEN** GO TO E-2, "FAULTED STEAM GENERATOR ISOLATION", Step 1 (unless faulted SG is needed for RCS cooldown).

  - Any SG pressure - DROPS IN AN UNCONTROLLED MANNER **AND** THAT SG HAS **NOT** BEEN ISOLATED
  - Any SG - COMPLETELY DEPRESSURIZED **AND** THAT SG HAS **NOT** BEEN ISOLATED

- **E-3 TRANSITION CRITERIA**

  IF any intact SG level rises in an uncontrolled manner **OR** any intact SG has abnormal radiation levels, **THEN** stop RCS depressurization and cooldown **AND** GO TO E-3. "STEAM GENERATOR TUBE RUPTURE, Step 1.

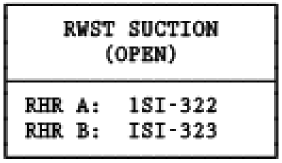
- **COLD LEG RECIRCULATION SWITCHOVER CRITERIA**

  IF RWST level drops to less than 23.4% (2/4 Low-Low alarm), **THEN** GO TO ES-1.3, "TRANSFER TO COLD LEG RECIRCULATION", Step 1.

Time	Position	Applicant's Actions or Behavior
	CREW	Initiate Monitoring Of Critical Safety Function Status Trees.
	RO	Maintain RCP Seal Injection Flow Between 8 GPM and 13 GPM.

	BOP	Check Intact SG Levels: • Any level - GREATER THAN 40% • Control Feed Flow to maintain all intact levels between 40% - 50%	(YES)
	BOP	• Any level – RISING IN AN UNCONTROLLED MANNER	(NO)
	RO	Check PRZ PORV AND Block Valves:	
	RO	• Verify AC buses 1A1 AND 1B1 – ENERGIZED • Check PRZ PORVs – SHUT • Check block valves - AT LEAST ONE OPEN	(YES) (YES) (YES)
	RO	Check SI Termination Criteria: • RCS subcooling - >40°F	(YES)
	BOP	• Level in at least one intact SG > 40% • Total feed flow to intact SGs > 200 KPPH	(YES) (YES)
	RO	PRZ level > 30% (YES / NO time dependent) **YES** – GO TO **ES-1.1, SI Termination, Step 1 (later in guide)** **NO** – Continue with E-1 actions below	

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	61	of 83
Event Description:	**Steam line Break on 'B' SG inside Containment (EOP-E-1)**							

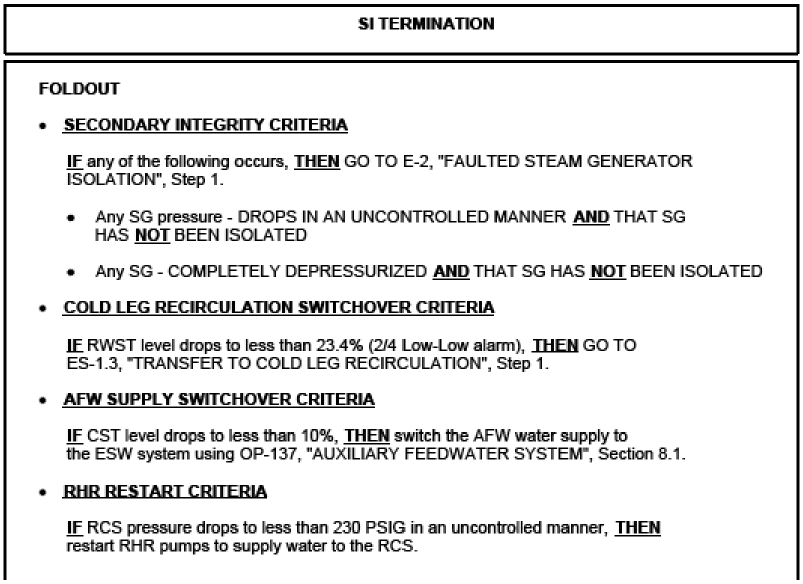
Time	Position	Applicant's Actions or Behavior	
E-1 Continues	RO	Check CNMT Spray Status: • Check any CNMT spray pump – RUNNING • Consult plant operations staff to determine if CNMT spray should be placed in standby. CNMT spray - TO BE PLACED IN STANDBY (When directed by plant operations staff)	(YES)
	RO	Check Source Range Detector Status: • Intermediate range flux – LESS THAN 5x10-11 AMPS • Verify source range detectors – ENERGIZED • Transfer nuclear recorder to source range scale.	(YES) (YES)
	RO	Check RHR Pump Status: • Check RHR pump suction – ALIGNED TO RWST  RWST SUCTION (OPEN) RHR A: 1SI-322 RHR B: ISI-323	(YES) (YES)
	RO	• RCS Pressure - GREATER THAN 230 PSIG • RCS pressure - STABLE OR RISING • Stop RHR pumps (STOPS both RHR pumps)	(YES) (YES)

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	62	of 83
Event Description:		**Steam line Break on 'B' SG inside Containment (EOP-ES-1.1)**						

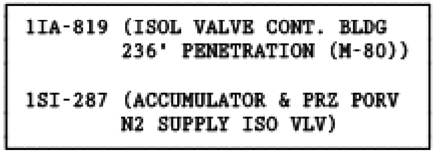
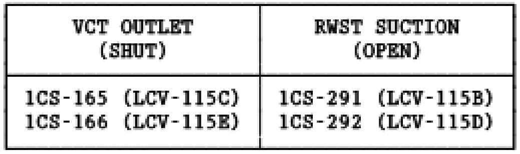
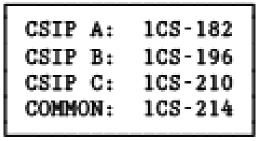
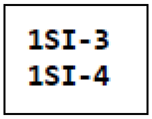
Time	Position	Applicant's Actions or Behavior
	RO	Check RCS And SG Pressures: **(time dependent)** Check for both of the following: All SG Pressures - STABLE OR RISING (YES / NO) RCS pressure - STABLE OR DROPPING (YES / NO) **IF NO** - the crew will return to step 1 and loop back to through the procedure. When they reach step 5 to check PRZ level they will have adequate level and transition to ES-1.1, SI Termination.
Evaluator Note:		**SI Termination is entered from either E-2 step 29 or E-1 Step 5.e**
EOP-ES-1.1		SI Termination
Procedure Note:		Foldout Applies
	SRO	Assigns foldout action items to RO and or BOP • Cold leg recirculation switchover criteria • RHR restart criteria • Secondary integrity criteria • AFW switchover criteria

Op Test No.: NRC Scenario # 2 Event # 7 Page 63 of 83

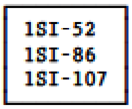
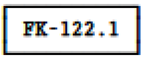
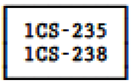
Event Description: **Steam line Break on 'B' SG inside Containment (EOP-ES-1.1)**

Time	Position	Applicant's Actions or Behavior	
Evaluator Aide:		**ES-1.1** Foldout	
		**SI TERMINATION** **FOLDOUT** • **SECONDARY INTEGRITY CRITERIA** **IF** any of the following occurs, **THEN** GO TO E-2, "FAULTED STEAM GENERATOR ISOLATION", Step 1. • Any SG pressure - DROPS IN AN UNCONTROLLED MANNER **AND** THAT SG HAS **NOT** BEEN ISOLATED • Any SG - COMPLETELY DEPRESSURIZED **AND** THAT SG HAS **NOT** BEEN ISOLATED • **COLD LEG RECIRCULATION SWITCHOVER CRITERIA** **IF** RWST level drops to less than 23.4% (2/4 Low-Low alarm), **THEN** GO TO ES-1.3, "TRANSFER TO COLD LEG RECIRCULATION", Step 1. • **AFW SUPPLY SWITCHOVER CRITERIA** **IF** CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1. • **RHR RESTART CRITERIA** **IF** RCS pressure drops to less than 230 PSIG in an uncontrolled manner, **THEN** restart RHR pumps to supply water to the RCS.	
	SRO	Initiate Monitoring Of Critical Safety Function Status Trees.	
	RO	Check If SI Has Been Terminated: Check for all of the following: Check BIT outlet valves – SHUT OR ISOLATED	
	RO	• 1SI-3 (YES / NO – shut in E-2 step 22 OR will be shut in ES-1.1 step 9.c – coming up) • 1SI-4 (YES / NO – shut in E-2 step 22 OR will be shut in ES-1.1 step 9.c – coming up) IF answer is **NO** then perform actions on following pages for "NO" response to reset SI If **YES** then do the following step and the actions then follow steps on page 66 of this guide after "NO" response ends.	
	RO	Check cold leg AND hot leg injection valves – SHUT • 1SI-52 • 1SI-86 • 1SI-107	(YES) (YES) (YES)

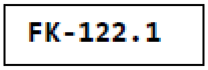
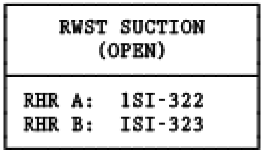
Op Test No.:	NRC	Scenario #	2	Event #	7	Page 64 of 83
Event Description:		**Steam line Break on 'B' SG inside Containment (EOP-ES-1.1)**				

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant’s Actions or Behavior</th></tr>
<tr><td>“NO” response</td><td>RO</td><td>Reset SI
Manually realign Safeguards Equipment Following A Loss of Offsite Power (NO action required)
Reset Phase A and Phase B Isolation Signals
Open IA and Nitrogen Valves to CNMT:

1IA-819 (ISOL VALVE CONT. BLDG 236' PENETRATION (M-80))
1SI-287 (ACCUMULATOR & PRZ PORV N2 SUPPLY ISO VLV)
Stop all but ONE CSIP
Check RCS Pressure – STABLE OR RISING
Isolate High Head SI Flow:
• Check CSIP suction – aligned to RWST

<table><tr><th>VCT OUTLET (SHUT)</th><th>RWST SUCTION (OPEN)</th></tr><tr><td>1CS-165 (LCV-115C)
1CS-166 (LCV-115E)</td><td>1CS-291 (LCV-115B)
1CS-292 (LCV-115D)</td></tr></table>
• Open normal miniflow isolation valves:

CSIP A: 1CS-182
CSIP B: 1CS-196
CSIP C: 1CS-210
COMMON: 1CS-214</td><td>(DONE)
(DONE)
(DONE)
(DONE)
(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td>Critical Task #4
“NO” response</td><td>RO</td><td colspan="2">Shut BIT Outlet Valves:

1SI-3
1SI-4
Shuts 1SI-3 from MCB switch
Attempts to shut 1SI-4 will not SHUT from MCB switch
Dispatches RAB Aux Operator to locally shut 1SI-4
(may also request that the breaker for the valve OPEN)
Critical Task to shut BIT Outlet valve 1SI-4 prior to establishing flow through the charging header or CSIP run out conditions will occur as indicated by oscillating discharge pressure.</td></tr>
</table>

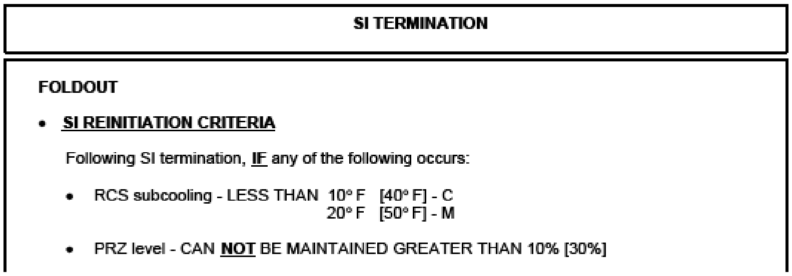
Op Test No.:	NRC	Scenario #	2	Event #	7	Page	65	of	83
Event Description:	**Steam line Break on 'B' SG inside Containment (EOP-ES-1.1)**								

Time	Position	Applicant's Actions or Behavior	
Simulator Communicator:		**IF this valve has not been previously shut then:** **Acknowledge request to locally shut 1SI-4 (A-230-FX32-W3-S2) AND if requested acknowledge request to OPEN breaker prior to locally valve operation.** **Report back approximately 1 minute after Simulator Operator completes actions below that 1SI-4 is SHUT.**	
Simulator Operator -		**Perform the following actions from Sim Diagram SIS02 to operate 1SI-4:** **(IF requested) OPEN control power rf sis016** **Engage handwheel rf sis017** **Shut valve rf sis018**	
"NO" response	RO	- Verify cold leg AND hot leg injection valves – SHUT  1SI-52 1SI-86 1SI-107	(YES)
Procedure Caution:		High head SI flow should be isolated before continuing	
"NO" response * ends after this step	RO	Establish Charging Lineup: - Shut charging flow control valve:  FK-122.1 - Open charging line isolation valves:  1CS-235 1CS-238	(SHUTS) (OPEN) (OPEN)

Op Test No.: NRC	Scenario # 2	Event # 7	Page 66 of 83
Event Description:		**Steam line Break on 'B' SG inside Containment (EOP-ES-1.1)**	

Time	Position	Applicant's Actions or Behavior	
Procedure Caution:		Charging flow should NOT exceed 150 GPM to prevent damage to the regenerative heat exchanger.	
	RO	Control Charging Flow To Maintain PRZ Level: • Control charging using charging flow control valve: FK-122.1 • Maintain charging flow < 150 gpm • PRZ level – CAN BE MAINTAINED STABLE OR RISING	(YES)
	RO	Check If RHR Pumps Should Be Stopped: • Check RHR pumps – ANY RUNNING WITH SUCTION ALIGNED TO RWST RWST SUCTION (OPEN) RHR A: 1SI-322 RHR B: ISI-323 • Stop RHR pumps (locates MCB stop switches and STOPs both RHR pumps)	(YES)
Procedure Caution:		• Simultaneous flow through the charging and SI lines may cause CSIP runout (as indicated by oscillating discharge pressure). • Charging flow should NOT exceed 150 GPM to prevent damage to the regenerative heat exchanger.	
	RO	Check SI Reinitiation Criteria: • RCS subcooling - GREATER THAN 40°F • PRZ level - GREATER THAN 30% • PRZ level - Can Be Maintained GREATER THAN 30%	(YES) (YES) (YES)

Op Test No.: NRC	Scenario # 2	Event # 7	Page 67 of 83
Event Description:	**Steam line Break on 'B' SG inside Containment (EOP-ES-1.1)**		

Time	Position	Applicant's Actions or Behavior	
Procedure Note:		Additional foldout item, "SI REINITIATION CRITERIA" applies.  **SI TERMINATION** **FOLDOUT** • **SI REINITIATION CRITERIA** Following SI termination, **IF** any of the following occurs: • RCS subcooling - LESS THAN 10° F [40° F] - C 20° F [50° F] - M • PRZ level - CAN **NOT** BE MAINTAINED GREATER THAN 10% [30%]	
	SRO	Assigns foldout for SI Reinitiation criteria	
	BOP	Establish Steam Generator Pressure Control Mode: • Check if steam dump to condenser AVAILABLE:  Condenser Available Requirements Any Intact SG MSIV - OPEN Condenser Available (C-9)- LIT (BPLB 3-3) Steam Dump Contol - AVAILALBE • Use intact SG PORV for steam dumping in subsequent steps.	(NO)
Procedure Note:		RCS temperature must be stabilized to allow evaluation of PRZ level trend.	
	RO	Monitor RCS Hot Leg Temperature: • Check RCS hot leg temperature - STABLE	(YES)
Procedure Caution:		Excessive RCS activity can cause adverse radiological conditions when letdown is placed in service.	

Op Test No.: NRC	Scenario # 2	Event # 7	Page 68 of 83
Event Description:	**Steam line Break on 'B' SG inside Containment (EOP-ES-1.1)**		
Time	Position	Applicant's Actions or Behavior	

Procedure Note:		Pressure controller PK-145.1 is normally set to maintain 350 PSIG (58%). If RCS pressure is low, the setpoint may have to be reduced to obtain proper letdown flow.	
	RO	Check If Letdown Can Be Placed In Service: • Check PRZ Level – GREATER THAN 40% • Establish Letdown	(YES)
Examiners Note:		After letdown is established Pressurizer level can be lowered and Pressurizer pressure should no longer be a problem. END OF SCENARIO	

Lead Evaluator:	**Terminate the scenario when RCS hot leg temperature stable or stabilizing under the crews control and letdown established.** **Announce 'Crew Update' - End of Evaluation - I have the shift.** **Have crew remain in the Simulator without discussing the exam. Examiners will formulate any follow-up questions.**

Simulator Operator:	**When directed by Lead Evaluator go to FREEZE**

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 1 of 7
SAFEGUARDS ACTUATION VERIFICATION

NOTE

- General guidance for verification of safeguards equipment is contained in Attachment 4 of this procedure.
- ERFIS displays of safeguards equipment status are not reliable while any associated safety-related electrical buses are de-energized.

☐ 1. **Ensure** Two CSIPs - RUNNING

☐ 2. **Ensure** Two RHR Pumps - RUNNING

☐ 3. **Ensure** Two CCW Pumps - RUNNING

☐ 4. **Ensure** All ESW **AND** ESW Booster Pumps - RUNNING

☐ 5. **Ensure** SI Valves - PROPERLY ALIGNED

(**Refer** to Attachment 1.)

☐ 6. **Ensure** CNMT Phase A Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 4.)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 2 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 7. **Ensure** SG Blowdown **AND** SG Sample Isolation Valves In Table 1 - SHUT

Table 1: SG Blowdown And Sample Isolation Valves

Process Line	Outside CNMT (MLB-1A-SA)	Inside CNMT (MLB-1B-SB)
SG A Sample	1SP-217	1SP-214/216
SG B Sample	1SP-222	1SP-219/221
SG C Sample	1SP-227	1SP-224/226
SG A Blowdown	1BD-11	1BD-1
SG B Blowdown	1BD-30	1BD-20
SG C Blowdown	1BD-49	1BD-39

8. IF Main Steam Line Isolation Actuated OR Is Required By Any Of The Following, THEN **Ensure** MSIVs AND MSIV Bypass Valves - SHUT

☐ • Steam line pressure - LESS THAN 601 PSIG

☐ • CNMT pressure - GREATER THAN 3.0 PSIG

9. IF CNMT Spray Actuation Signal Actuated OR Is Required, THEN **Ensure** The Following:

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 9.)

☐ • CNMT spray pumps - RUNNING

☐ • CNMT spray valves - PROPERLY ALIGNED

☐ • Phase B isolation valves - SHUT

☐ • All RCPs - STOPPED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 3 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 10. **Ensure** Both Main FW Pumps - TRIPPED

☐ 11. **Ensure** FW Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 6.)

☐ 12. **Ensure** Both MDAFW pumps - RUNNING

13. **IF** Any Of The Following Conditions Exist, **THEN** **Ensure** The TDAFW Pump - RUNNING

- ☐ Undervoltage on either 6.9 KV emergency bus
- ☐ Level in two SGs - LESS THAN 25%
- ☐ Manual actuation to control SG level

14. **Ensure** AFW Valves - PROPERLY ALIGNED

- ☐ **IF** no AFW Isolation Signal, **THEN** **ensure** isolation **AND** flow control valves - OPEN

NOTE

An AFW Isolation signal signal requires a Main Steam Line Isolation coincident with one SG pressure 100 PSIG below the other two SGs.

- ☐ **IF** AFW Isolation Signal present, **THEN** **ensure** MDAFW **AND** TDAFW isolation **AND** flow control valves to affected SG - SHUT

☐ 15. **Ensure** Both EDGs - RUNNING

☐ 16. **Ensure** CNMT Fan Coolers - ONE FAN PER UNIT RUNNING IN SLOW SPEED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 4 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 17. **Ensure** CNMT Ventilation Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 7.)

☐ 18. **Ensure** Control Room Area Ventilation - MAIN CONTROL ROOM ALIGNED FOR EMERGENCY OPERATION

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 5, Sheets 1 and 2, Sections for MAIN CONTROL BOARD, SLB-5 and SLB-6.)

19. **Ensure** Essential Service Chilled Water System Operation:

- ☐ • **Ensure** both WC-2 chillers - RUNNING
- ☐ • **Ensure** both P-4 pumps - RUNNING
- ☐ (**Refer** to AOP-026, "LOSS OF ESSENTIAL SERVICE CHILLED WATER SYSTEM" for loss of any WC-2 chiller.)

20. **Ensure** CSIP Fan Coolers - RUNNING

☐ AH-9 A SA
☐ AH-9 B SB
☐ AH-10 A SA
☐ AH-10 B SB

<u>NOTE</u>

Security systems are powered by bus 1A1 (normal supply) or bus 1B1 (alternate supply). Backup power will be available for approximately 30 MINUTES after the supplying bus is de-energized. (Refer to OP-115, "CENTRAL ALARM STATION ELECTRICAL SYSTEMS", Section 8.9 and 8.10.)

☐ 21. **Ensure** AC buses 1A1 **AND** 1B1 - ENERGIZED

☐ 22. **Place** Air Compressor 1A **AND** 1B In The LOCAL CONTROL Mode.

(**Refer to** Attachment 7.)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 5 of 7
SAFEGUARDS ACTUATION VERIFICATION

CAUTION

The maximum calculated dose rate in the vicinity of MCC 1A35-SA and MCC 1B35-SB is between 10 MREM/HR and 150 MREM/HR.

☐ 23. Dispatch An Operator To **Unlock** And **Close** The Breakers For The CSIP Suction **AND** Discharge Cross-Connect Valves:

(**Refer** to Attachment 2.)

MCC 1A35-SA		MCC 1B35-SB	
VALVE	CUBICLE	VALVE	CUBICLE
1CS-170	4A	1CS-171	4D
1CS-169	4B	1CS-168	7D
1CS-218	14D	1CS-220	9D
1CS-219	14E	1CS-217	12C

24. **Check** If C CSIP Should Be Placed In Service:

☐ • **IF** two charging pumps can **NOT** be verified to be running, **AND** C CSIP is available, **THEN place** C CSIP in service in place of the non-running CSIP using OP-107, "CHEMICAL AND VOLUME CONTROL SYSTEM, Section 8.5 or 8.7.

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 6 of 7
SAFEGUARDS ACTUATION VERIFICATION

25. **Start** The Spent Fuel Pump Room Ventilation System:

    a. At AEP-1, **ensure** the following ESCWS isolation valves - OPEN

    1) SLB-11 (Train A)

    ☐ • AH-17 SUP CH 100 (Window 9-1)

    ☐ • AH-17 RTN CH 105 (Window 10-1)

    2) SLB-9 (Train B)

    ☐ • AH-17 SUP CH 171 (Window 9-1)

    ☐ • AH-17 RTN CH 182 (Window 10-1)

    b. At AEP-1, **start** one SFP PUMP ROOM FAN COOLER:

    ☐ • AH-17 1-4A SA

    ☐ • AH-17 1-4B SB

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 7 of 7
SAFEGUARDS ACTUATION VERIFICATION

NOTE

- Fuel pool levels and temperatures should be monitored approximately every 1 to 2 HOURS.
- Following the initial check of fuel pool levels and temperature, monitoring responsibilities may be assumed by the plant operations staff (including the TSC or STA).
- Only fuel pools containing fuel are required to be monitored.

26. **Check** Status Of Fuel Pools:

☐ a. **Operate** spent fuel cooling pumps to maintain fuel pool temperatures between 85°F to 105°F.

b. **Monitor** fuel pool levels **AND** temperatures:

☐ • **Refer to** AOP-041, "SPENT FUEL POOL EVENT" Attachments 7, 8, 9, 10 and 11 for SFP parameter monitoring methods.

☐ • **Refer to** Curves H-X-24, H-X-25 and H-X-26 for SFP time to 200°F.

☐ • Levels - GREATER THAN LO ALARM (284 FT, 0 IN)

☐ • Temperatures - LESS THAN HI TEMP ALARM (105°F)

NOTE

If control room ventilation was previously aligned to an emergency outside air intake for post-accident operations, then follow-up actions will be required to restore the alignment.

27. **Consult** Plant Operations Staff Regarding Alignment Of The Control Room Ventilation System:

☐ • Site Emergency Coordinator - Control Room

☐ • Site Emergency Coordinator - Technical Support Center

(Refer to PEP-230, "CONTROL ROOM OPERATIONS".)

- END -

SECONDARY STEAM LEAK/ EFFICIENCY LOSS

INSTRUCTIONS	RESPONSE NOT OBTAINED
3.0 OPERATOR ACTIONS	
NOTE This procedure contains no immediate actions.	
✸ 1. **CHECK** that the plant can be operated safely:	1. **PERFORM** the following:
☐ • **CHECK ALL** Reactor Protection parameters will remain **WITHIN TRIP LIMITS**.	☐ a. **TRIP** the Reactor **AND GO TO** EOP-E-0. (Continue with RNO Step 1.b.)
☐ • **CHECK** Turbine Building envelope safe for personnel entry.	**NOTE** If Main Steam Line Isolation is required, the Reactor and Turbine should be verified tripped in EOP-E-0 before manually initiating MSLI.
☐ • **CHECK** RAB Steam Tunnel safe for personnel entry.	
	☐ b. **IF** the Reactor was tripped due to a steam leak, **THEN MANUALLY INITIATE** a Main Steam Line Isolation signal.
	☐ c. **EXIT** this procedure.
☐2. **CHECK** a steam leak exists.	☐2. **GO TO** Step 4.

SECONDARY STEAM LEAK/ EFFICIENCY LOSS

INSTRUCTIONS	RESPONSE NOT OBTAINED
3.0 OPERATOR ACTIONS	
☐3. **NOTIFY** personnel of evacuation requirements.	
☐ a. **SOUND** the local evacuation alarm.	
☐ b. **ANNOUNCE** on the PA: "Attention all personnel. There is a steam leak (give location). All personnel stand clear of (give location)."	
☐ c. **ESTABLISH** a boundary to prevent unauthorized personnel entry.	
☐4. **REFER TO** PEP-110, Emergency Classification and Protective Action Recommendations, **AND ENTER** the EAL Matrix.	

NOTE

Initial target reduction may be up to 100 MW less than current REFERENCE value and may be changed as necessary to reduce power to less than 100%.

☐5. **DETERMINE** the required megawatt change needed for the power reduction.	☐5. **IF** no power reduction is required, **THEN GO TO** Step 17 to determine leak location.
☐6. **NOTIFY** Load Dispatcher that the Unit is reducing load.	

SECONDARY STEAM LEAK/ EFFICIENCY LOSS

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

NOTE

- If load reduction rates in excess of 45 MW/min are required, the Unit should be tripped.
- If OSI-PI is available, VIDAR is functioning properly if the DEH_MEGAWATTS point is updating. (Attachment 1, Checking VIDAR Functioning, provides alternative methods of checking VIDAR functioning.)

CAUTION

Failure of the DEH computer VIDAR Unit while in OPER AUTO has resulted in a plant trip.

INSTRUCTIONS	RESPONSE NOT OBTAINED
7. **CHECK BOTH** of the following: ☐ • DEH System in AUTO ☐ • VIDAR functioning properly	☐ 7. **PREPARE** to reduce Turbine load manually using OP-131.01, Main Turbine, **AND GO TO** Step 9.

SECONDARY STEAM LEAK/ EFFICIENCY LOSS

INSTRUCTIONS	RESPONSE NOT OBTAINED
3.0 OPERATOR ACTIONS	
8. **PERFORM** the following at the DEH panel:	
☐ a. **DEPRESS** the LOAD RATE MW/MIN pushbutton.	
☐ b. **ENTER** desired rate (NOT to exceed 45 MW/MIN) in DEMAND display.	
☐ c. **DEPRESS** ENTER pushbutton.	
☐ d. **DEPRESS** REF pushbutton.	
☐ e. **ENTER** desired load in DEMAND display.	
☐ f. **DEPRESS** ENTER pushbutton.	
☐ g. **CHECK** HOLD pushbutton LIT.	
☐9. **CHECK** Rod Control in AUTO.	9. **PERFORM** ONE of the following: ☐ a. **PLACE** Rod Control selector switch in AUTO. ☐ b. **MANUALLY POSITION** Control Rods to maintain T_{avg} within 5°F of T_{ref}.

SECONDARY STEAM LEAK/ EFFICIENCY LOSS

INSTRUCTIONS	RESPONSE NOT OBTAINED
3.0 OPERATOR ACTIONS	

> **NOTE**
>
> During the load reduction, it is permissible to periodically move between GO and HOLD and to vary the load rate.

INSTRUCTIONS	RESPONSE NOT OBTAINED
10. **COMMENCE** turbine load reduction at the DEH panel:	
a. **CHECK** OPER AUTO Mode AVAILABLE.	☐ a. **MANUALLY REDUCE** Turbine load using OP-131.01, Main Turbine.
☐ (1) **DEPRESS** GO pushbutton.	☐ b. GO TO Step 11.
☐ (2) **VERIFY** the value in the REFERENCE display LOWERS.	
☐ 11. **VERIFY** Generator load AND Reactor power LOWERING.	
✱ ☐ 12. **MAINTAIN** Generator reactive load (VARs) within guidelines.	
✱ ☐ 13. **CHECK** T_{avg} within 5°F of T_{ref}.	13. **RESTORE** T_{avg} to within 5°F of T_{ref} by ANY of the following methods: ☐ • **ADJUST** Turbine load ☐ • **ADJUST** boron concentration ☐ • **MANUALLY CONTROL** rod insertion or withdrawal.

SECONDARY STEAM LEAK/ EFFICIENCY LOSS

INSTRUCTIONS	RESPONSE NOT OBTAINED
3.0 OPERATOR ACTIONS	
☐14. **WHEN** Reactor power is less than 100%, **THEN DEPRESS** the HOLD pushbutton.	
☐15. **CHECK** the HOLD pushbutton is LIT.	
☐16. **CHECK** a steam leak exists.	☐16. GO TO Step 19.
☐17. **DISPATCH** personnel to identify the leak location using all necessary safety practices.	
✱☐18. **CHECK** that the steam leak can be isolated.	☐18. GO TO ONE of the following, as applicable:
☐ a. **ISOLATE** the leak.	☐ • GP-006, Normal Plant Shutdown From Power Operation to Hot Standby (Mode 1 To Mode 3), for normal plant shutdown
	☐ • AOP-038, Rapid Downpower
☐19. **NOTIFY** the Load Dispatcher that power reduction is complete.	
☐20. **CHECK** REFERENCE and DEMAND windows equalized.	20. **PERFORM** the following:
	☐ a. **DEPRESS** the REF pushbutton.
	☐ b. **ENTER** the REFERENCE value in the DEMAND window.
	☐ c. **DEPRESS** the ENTER pushbutton.

SECONDARY STEAM LEAK/ EFFICIENCY LOSS

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

21. GO TO ONE of the following, as applicable:

☐ • GP-005, Power Operation (Mode 2 to Mode 1), for continued plant operation

☐ • GP-006, Normal Plant Shutdown From Power Operation to Hot Standby (Mode 1 To Mode 3), for normal plant shutdown

☐ • AOP-038, Rapid Downpower

☐**22. EXIT** this procedure.

-- END OF SECTION 3.0 --

SECONDARY STEAM LEAK/ EFFICIENCY LOSS

INSTRUCTIONS	RESPONSE NOT OBTAINED

Attachment 1 – Checking VIDAR Functioning

Sheet 1 of 1

NOTE

If OSI-PI is NOT available, then accessing the ANALOG INPUTS screen on the Graphics Display Computer (located in the Termination Cabinet Room near the ATWS Panel) will show several points, most of which should be updating if the VIDAR Unit is functioning properly.

1. IF the DEH graphics computer is out of service,
THEN VIDAR can be checked as updating on the operator panel as follows:

☐ a. **DEPRESS** TURBINE PROGRAM DISPLAY button.

☐ b. **CHECK** TURBINE PROGRAM DISPLAY button is illuminated.

☐ c. **CHECK** REFERENCE and DEMAND displays indicate 0000.

☐ d. **DEPRESS** 1577.

☐ e. **DEPRESS** "ENTER".

f. **CHECK** the DEMAND display:

☐ • IF the DEMAND display indicates 0000, VIDAR is updating.

☐ • IF the DEMAND display indicates 0001, VIDAR is NOT updating.

HARRIS 2020 NRC SCENARIO 3

Facility:	Harris Nuclear Plant	Scenario No.: 3	Op Test No.: 05000400/2020301
Examiners:		Operators:	SRO:
			RO:
			BOP:

Initial Conditions: IC-27, MOL, 3% power

- 'B' NSW Pump is under clearance for breaker repairs

Turnover:	The plant is at 3% power, middle of core life. Startup on HOLD for briefing GP-005 Rev 107 Step 87
Critical Task:	• Manually start AFW flow to maintain control of SG level above 25% to prevent an automatic Reactor trip after trip of the last running Main Feed Pump • Manually start at least one high head ECCS pump to prevent RVLIS Dynamic Range Level from lowering below 33% • During a Small Break LOCA secure all RCPs with SI flow > 200 gpm and RCS pressure < 1400 psig to prevent RVLIS Dynamic Range Level from lowering below 33%

Event No.	Malf. No.	Event Type*	Event Description
1	xd2i085 xd2o085w xn30d06	C – BOP/SRO TS – SRO	Control Room Air Handler AH-15 trip requiring standby Air Handler startup – (APP-030)
2	tt:144 jtb143b	I – RO/SRO	Letdown Temperature Controller fails LOW/Diversion Valve fails to bypass demineralizers
3	cnd04a	C – BOP/SRO	Main Condenser Evacuation Pump trips – (AOP-012)
4	rcs10	C – RO/SRO TS – SRO	Reactor Vessel Flange Leak – (AOP-016)
5	cfw16a xb1i155 zr211158 zr211113	C – BOP/SRO TS – SRO	Running MFW Pump trips Standby MFW Pump fails to start Both MDAFW Pump AUTO start failure
6	rcs09b	C – RO/SRO	RCP "B" rising vibration (AOP-018). Vibrations require a manual Reactor trip (E-0) , then secure 'B' RCP and PRZ spray valve.
7	rps01b	M – ALL	Failure of the Reactor Trip breakers to open auto or manual – (EOP-FR-S.1)
8	rcs01a	M – ALL	Small Break LOCA – (EOP-E-1)
9	dsg04a	C – RO/SRO	Failure of 'B' Sequencer Load Block 1 to actuate during the Safety Injection which fails to start 'B' CSIP
10	zrpk601a	C – BOP/SRO	Failure of Safety Injection Isolation valves on 'A' Train CSIP normal mini flow 1CS-214 fails to close automatically

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

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 3

The crew will assume the watch while the JITT Trained Startup crew is attending a briefing by Reactor Engineering. The plant was in Mode 1 with Turbine valve testing complete when secondary chemistry parameters degraded and Reactor power was lowered to < 5%. The plant startup is on hold in MODE 2. The candidates are to maintain current plant conditions with Reactor Power ~ 3%.

The following equipment is under clearance:

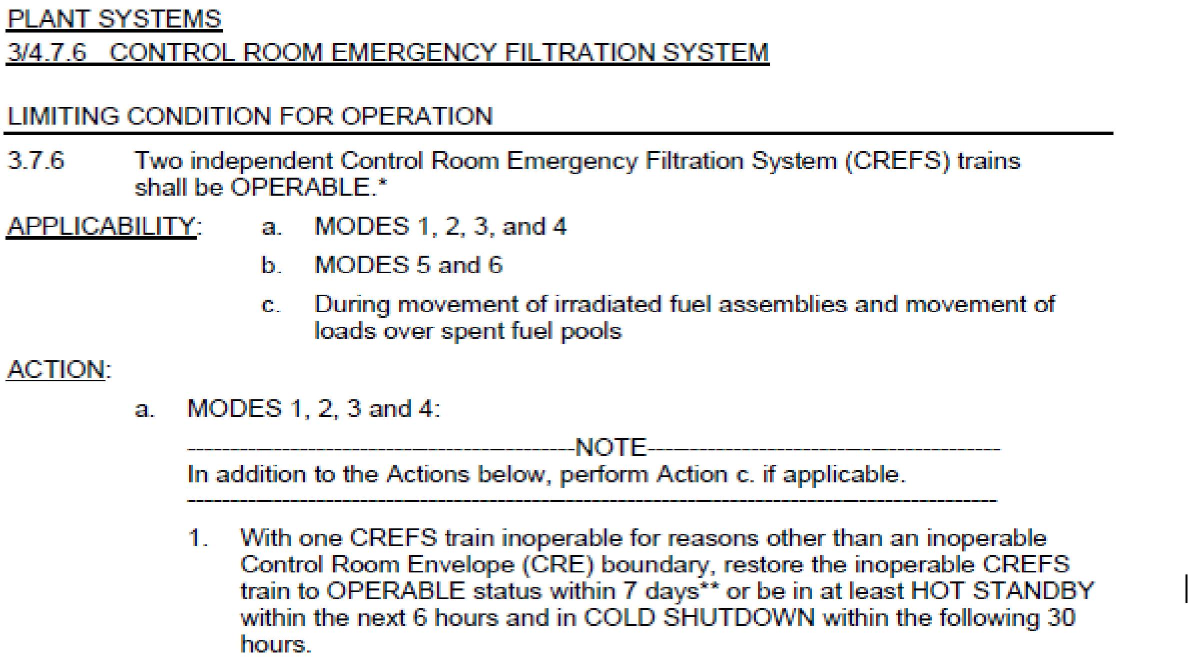
- 'B' NSW Pump is under clearance for motor repairs. The pump has been unavailable for 8 hours. Repairs are expected to be completed within 24 hours.

Event 1: Control Room Air Handler AH-15A-SA trips. Annunciator ALB-030-6-4, Control Room HVAC Normal Supply Fans AH-15A-SA Low Flow – O/L will alarm. The HVAC dampers will automatically reposition and all Control Room Ventilation will secure.

Verifiable Action: The BOP will respond in accordance with the alarm procedure for ALB-030-6-4. The BOP should identify that the standby fan has failed to automatically start and report the failure to the SRO. The SRO should direct the BOP to manually start the standby fan using OP-173, Control Room Area HVAC Systems.

The SRO should evaluate Tech Spec 3.7.6, Control Room Emergency Filtration system and determine action a.1 applies.

- With one Control Room Emergency Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.



PLANT SYSTEMS
3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Room Emergency Filtration System (CREFS) trains shall be OPERABLE.*

APPLICABILITY:
a. MODES 1, 2, 3, and 4
b. MODES 5 and 6
c. During movement of irradiated fuel assemblies and movement of loads over spent fuel pools

ACTION:

a. MODES 1, 2, 3 and 4:

--NOTE--
In addition to the Actions below, perform Action c. if applicable.
----

1. With one CREFS train inoperable for reasons other than an inoperable Control Room Envelope (CRE) boundary, restore the inoperable CREFS train to OPERABLE status within 7 days** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The SRO should prepare AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 3 (Continued)

Event 2: Letdown Temperature Controller fails - LD/Diversion Valve fails to bypass demineralizers. This failure will cause temperature controller TK-144 output to decrease to zero. Without cooling to the letdown heat exchanger, temperatures observed on TI-143 will rise. At 135°F annunciator ALB-007-3-2, Demin Flow Diversion High Temp will alarm.

Verifiable Action: The OATC will respond in accordance with the alarm procedure for ALB 007-3-2. The OATC should identify that the divert valve to the VCT has failed to respond and report the failure to the SRO. The OATC should manually bypass the CVCS Demineralizers with 1CS-50 (TCV-143), and then take manual control of TK-144 to restore letdown temperature to normal.

The SRO should provide a temperature band between 110°F to 120°F to the OATC in accordance with AD-OP-ALL-1000, Conduct of Operations, (5.5.6) for operation Control Bands. (Temperature band guidance can be found in OP-107, Chemical Volume And Control). The CVCS Demineralizers should remain bypassed pending an evaluation for continued resin use. The SRO should prepare AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 3: Main Condenser Evacuation Pump ‘A’ trips. – ALB -021-4-1, Condenser Vacuum Pump A Trip, will alarm and the breaker for the MCES Pump ‘A’ will indicate open on the MCB. Main condenser Vacuum will degrade slowly.

Verifiable Action: The BOP will respond in accordance with the alarm procedure for ALB 021-4-1 and identify that the ‘A’ MCES Pump has tripped based on MCB indication. The BOP will report the failure to the SRO and manually start the ‘B’ MCES Pump. The SRO should review AOP-012, Partial Loss of Condenser Vacuum, and work through the procedure to determine if any additional actions are required.

The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 4: Reactor Vessel Flange leak of ~ 15 gpm. The crew should notice Pressurizer level slowly lowering and a rise in charging flow. Annunciator ALB-010-5-5, Reactor vessel flange leakoff high temp will alarm when MCB temperature indicator TI-401 reaches 140°F.

Verifiable Action: The OATC will respond in accordance with the alarm procedure for ALB 010-3-2. The OATC should identify that the rising temperature on TI-401 and report the failure to the SRO. The OATC should shut 1RC-46 in accordance with the alarm response. manually actions for Reactor Vessel leakage directs shutting 1RC-46, Head Flange Seal Leakoff Line Isolation.
The closure of this valve will stop leakage from the inner Reactor head seal. AOP-016, Excessive Primary Plant Leakage may also be entered by the crew to address the flange leakage but the leakage will be stopped when addressed with the APP actions.

The SRO should evaluate Tech Spec 3.4.6.2, Reactor Coolant System – Operational Leakage and determine action b applies for condition d (briefly until 1RC-46 is shut):

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 3 (Continued)

Event 4: Tech Spec evaluation continued

T.S. 3.4.6.2: Reactor Coolant System operational leakage shall be limited to:

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System (Modes 1, 2, 3, and 4)

Action:

b. With any Reactor Coolant System operational leakage greater than anyone of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC for entry into Containment to complete the APP-ALB-010 actions.

Event 5: ‘A’ MFP trips with ‘B’ MFP failure to start and initiate AFW in accordance with AOP-010, Feedwater Malfunctions – ‘A’ MFP trip, with the ‘B’ MFW pump failing to auto start may be inserted once Pressurizer level and RCS Leakoff temperature have stabilized. Both MDAFW pumps fail to auto start but can be started in the MCR.

Verifiable Action: The BOP will respond in accordance with the alarm procedure for ALB 016-1-4. The BOP should identify that the ‘A’ MFP has tripped and the ‘B’ MFP failed to start based on Feedwater discharge pressure and plant response and report the failure to the SRO and verbalize the immediate actions of AOP-010. The SRO should enter AOP-010, Feedwater Malfunctions, and work through the procedure to initiate AFW flow to maintain Steam Generator Level between 52 and 62%(**Critical Task #1**). The BOP may place the ‘A’ MFP and ‘B’ MFP control switches in the stop position for the tripped MFP in accordance with APP-ALB-016-1-4.

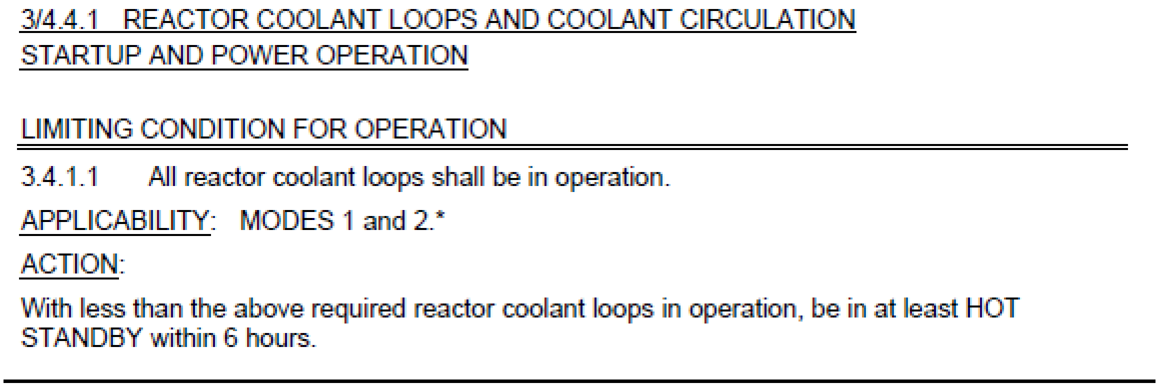
Event 6: ’B’ RCP high vibration. During this event the ‘B’ RCP vibrations will begin to rise over 3 minutes and peak at 28 mils shaft. Note: the shaft vibration instrumentation reads up to 30 mils. The crew will respond to the ‘B’ RCP malfunction by either identify rising vibrations or when ALB-010-2-5, RCP-B Trouble alarms.

Verifiable Action: The OATC will respond in accordance with the alarm procedure for ALB 010-2-5 and report this to the alarm to the SRO. The BOP should see the ‘A’ RCP vibration probe readings are rising and report the failure to the SRO. The SRO should enter AOP-018, Reactor Coolant Pump Abnormal Conditions and the OATC should perform the immediate actions of checking any CSIP running. Vibrations will continue to rise and exceed AOP-018 Attachment 1 RCP trip criteria of 20 mils shaft. The OATC will perform a manual Reactor trip and at which time the Reactor will fail to trip (ATWS) and the will have to implement EOP-FR-S.1.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 3 (Continued)

Event 6: Tech Spec evaluation continued

The SRO should evaluate Tech Spec 3.4.1.1, Reactor Coolant Loops and Coolant Circulation Startup and Power Operation, and determine this action is applicable prior to opening the Reactor Trip Breakers. This may be discussed after the scenario based on the sequence of this event.



3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION
STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

Once the immediate actions of EOP-FR-S.1 are completed the crew will transition to EOP-E-0 and secure the ‘A’ RCP and associated PRZ spray valve after EOP-E-0 immediate actions are completed.

Event 7: Major - ATWS Reactor Trip breakers fail to open auto or manual. The crew should recognize that the Reactor has failed to trip and enter FR-S.1, Response to Nuclear Power Generation/ATWS. The Reactor Trip breakers will be opened locally one minute after a field operator has been dispatched to perform those actions. Once the crew has inserted negative reactivity via rod insertion (Auto or manual) or initiated the emergency Boration and have verified that the Reactor is tripped in FR-S.1, they should exit FR-S.1 and return to EOP-E-0.

Verifiable Action: The OATC will respond in accordance with EOP-FR-S.1 immediate actions and attempt to trip the Reactor via the second MCB Rector Trip switch. The BOP will respond in accordance with EOP-FR-S.1 immediate actions trip the Turbine from the MCB via the Turbine Trip switch. Once the immediate actions of EOP-FR-S.1 are complete the SRO should make a plant announcement for an available operator to come to the MCR for directions to locally trip the Reactor.

The next event Small Break LOCA will ramp in over the 4 minutes from the time the Reactor Trip breakers open allowing the crew to then transition from EOP E-0 to ES-0.1, Reactor Trip Response.

Event 8: Major - Small Break LOCA caused by a Loop 1 Cold Leg break resulting in either a Manual OR Automatic SI initiation. The crew should recognize a changing plant conditions with Pressurizer level and RCS pressure lowing. If the crew responds quickly to the event they may manually actuate a Safety Injection based on ES-0.1 foldout criteria of not being able to maintain Pressurizer level > 5% or RCS subcooling < 10°F. If they do not respond quickly an Automatic

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 3 (Continued)

Event 8: Continued

Safety Injection will occur. The crew will then transition from ES-0.1 back to E-0, Reactor Trip or Safety Injection. After returning to EOP-E-0 and with SI actuated the crew will identify the ‘A’ CSIP has tripped and the ‘B’ CSIP has failed to start from the Sequencer and pressure in the Containment will continue to rise due to the LOCA. The degrading conditions in Containment will cause the crew to transition from EOP-E-0 to EOP-E-1, Loss of Reactor or Secondary Coolant.

Verifiable Action: Once the crew starts the ‘B’ CSIP (**Critical Task #2**). The Foldout Criteria for securing RCPs will be met and secure the RCPs (**Critical Task #3**). The OATC will report this condition to the SRO. The SRO should continue in EOP-E-0, and direct the OATC to implement the procedure foldout to ensure all RCPs are stopped.

Event 9: During the Safety Injection activation the ‘B’ Load Sequencer will skip the ‘B’ CSIP load block.

Verifiable Action: The OATC will report this condition to the SRO. The SRO should continue in EOP-E-0, and work through the procedure to ensure the OATC starts the ‘B’ CSIP in accordance with step 6. Provided the sequencer has reached Load Block 9 (Manual Loading Permissive) the OATC may start ‘B’ CSIP when the automatic function failure is observed in accordance with AD-OP-ALL-1000, Conduct of Operations, (5.6.3.8) for Equipment Manipulation and Status Control after notifying the SRO.

Event 10: Failure of Safety Injection Isolation valves on 'A' Train CSIP normal mini flow 1CS-214 fails to close automatically.

Verifiable Action: The BOP or the OATC will report this condition to the SRO. The SRO should continue in EOP-E-0, and work through the procedure to ensure the BOP or the OATC shuts 1CS-214 in accordance with EOP-E-0, Attachment 1. The BOP or the OATC may shut 1CS-214 when the automatic function failure is observed in accordance with AD-OP-ALL-1000, Conduct of Operations, (5.6.3.8) for Equipment Manipulation and Status Control after notifying the SRO.

The crew will continue in EOP-0 into EOP-E-1, Loss of Reactor Or Secondary Coolant until the transition to EOP-ES-1.2, Post-LOCA Cooldown and Depressurization, is made. During the implementation of EOP-ES-1.2 a transition to EOP-FR-P.1, Response To Imminent Pressurized Thermal Shock may be required based on Cold Leg temperature of the broke RCS loop dropping below 240°F. The crew will return back to EOP-ES-1.2 where the scenario termination is met after the first SG pressure reduction has been completed.

CRITICAL TASK JUSTIFICATION:

1. Manually start AFW flow to maintain control of SG level above 25% to prevent an automatic Reactor trip after trip of the last running Main Feed Pump

An unnecessary automatic Reactor Trip for this event will create critical task. See note below.

2. Manually start at least one high head ECCS pump to prevent RVLIS Dynamic Range Level from lowering below 33%

In this scenario the ‘A’ CSIP has tripped and the ‘B’ CSIP has did not automatically start from sequencer actuation. The operator must manually start the ‘B’ CSIP which was currently in standby. Plant parameter grading criteria for the task is starting the ‘B’ CSIP to prevent RVLIS Dynamic Range Level from lowering below 33% which constitutes a significant core uncover with 2 Reactor Coolant Pumps in operation.

3. During a Small Break LOCA secure all RCPs with SI flow > 200 gpm and RCS pressure < 1400 psig to prevent RVLIS Dynamic Range Level from lowering below 33%

In this scenario EOP-E-0 foldout will apply following the completion of the immediate actions. The RCP trip criteria is BOTH of the following: SI flow > 200 gpm and RCS pressure < 1400 psig. These plant parameters are to be monitored continuously and when those conditions are met the operator must secure the operating RCPs. Plant parameter grading criteria for the task is tripping RCPs if SI flow > 200 gpm to prevent RVLIS Dynamic Range Level from lowering below 33% which constitutes a significant core uncover with 2 Reactor Coolant Pumps in operation.

Note: Causing an unnecessary plant trip or ESF actuation may constitute a CT failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

Simulator Setup

Reset to IC-143 password “NRC3sros”

Go to RUN

Silence and Acknowledge annunciators

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner.

Set ERFIS screens for normal full power conditions

(The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

SPECIAL INSTRUCTIONS

Provide a Reactivity Plan to candidates for shutting down the plant

Provide a copy of the following procedures:

- GP-005, POWER OPERATION (MODE 2 TO MODE 1) **marked up** through section 6.0 step 87

Press START on Counter Scaler
Post conditions for status board from IC-27
Reactor Power 3%
Control Bank D at 102 steps
RCS boron 1472 ppm

Turnover: The plant is at 3% power, middle of core life. The crew will assume the watch while the JITT Trained Startup crew is attending a briefing by Reactor Engineering. The plant was in Mode 1 with Turbine valve testing complete when secondary chemistry parameters degraded and Reactor power was lowered to < 5%. The plant startup is on hold in MODE 2. The candidates are to maintain current plant conditions with Reactor Power ~ 3%.

Equipment Under Clearance:

- ‘B’ NSW Pump is under clearance for breaker repairs.

Align equipment for repairs:

Place protected train placards IAW OMM-001 Attachment 5
Protected Train placards on 'A' NSW pump

Place a CIT on the switch for ‘B’ NSW Pump.

Hang restricted access signs on MCR entry swing gates
Set CRT screen 3 to "QP STARTUP"

Op Test No.:	NRC	Scenario #	3	Event #	1	Page	9	of	66
Event Description:	**Control Room Air Handler AH-15 trips, standby fails to Auto Start**								

Time	Position	Applicant’s Actions or Behavior
Lead Evaluator:		**The crew has been directed to hold power at 3% while the oncoming crew conducts a turnover briefing.** **When the crew has completed their board walk down and are ready to take the shift inform the Simulator Operator to place the Simulator in Run. When the Simulator is in run announce:** **CREW UPDATE – (SRO’s Name) Your crew has the shift. END OF UPDATE**
Simulator Operator:		**When directed by the Lead Evaluator, ensure that the annunciator horns are on and place the Simulator in RUN.**
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 1** **“Control Room Air Handler AH-15 trips, standby fails to Auto Start”**
Indications Available:		• **ALB-030-6-4, Control Room HVAC Normal Supply Fans AH-15 Low Flow – O/L** • **Control Room ventilation damper re-alignment** • **White Overload light lit on AH-15 MCB switch**
ALB-030	BOP	RESPONDS to alarm on APP-ALB-030-6-4
	BOP	CONFIRM alarm using: • Fan status indication at MCB for AH-15 (1A-SA and 1B-SB) • Damper position indication on MCB for CZ-D2SB, CZ-25, and CZ-26 • ALB-030-6-3, Cont Room Normal Supply AH-15 Filter High ΔP
	BOP	VERIFY Automatic Functions: • Fans trip on overload

Op Test No.:	NRC	Scenario # 3	Event #	1	Page	10 of	66
Event Description:	**Control Room Air Handler AH-15 trips, standby fails to Auto Start**						

Time	Position	Applicant's Actions or Behavior	
	BOP	PERFORM Corrective Actions:	
		• CHECK AH-15 fans status indication on MCB. • IF fan is tripped, THEN PERFORM the following: ○ START the standby fan using OP-173, Control Room Area HVAC System.	(YES)
	SRO	Directs BOP to start Control Room ventilation alignment in accordance with OP-173	
	BOP	○ IF white fan trouble light is LIT, THEN DISPATCH an operator to check overload relays on 1A36-SA-5A or 1B36-SB-3A. ○ DISPATCH an operator to check for tripped breaker on 1A36-SA-5A or 1B36-SB-3A. • CHECK damper alignment on MCB for CZ-D1SA-1, CZ-D2SB-1, CZ-25 and CZ-26. • IF ALB-030-6-3 is ALARMING, THEN REFER TO ALB-030-6-3.	(NO)
Simulator Communicator:		**When contacted to investigate fan failure report back in 2 minutes that breaker 1A-36-SA Cubical 5A is tripped on overload and no problems are noted locally at the fan unit.**	
Evaluator Note:		**(Any Tech Spec evaluation can be conducted with a follow up question after the scenario).**	
	SRO	Enters Instrumentation TS 3.7.6 ACTION **a.1** - With one Control Room Emergency Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.	
	SRO	Completes an Emergent Issue Checklist and contacts WCC for assistance. (WR, LCOTR, and Maintenance support)	

Op Test No.:	NRC	Scenario #	3	Event #	1	Page	11	of 66

Event Description: **Control Room Air Handler AH-15 trips, standby fails to Auto Start**

Time	Position	Applicant's Actions or Behavior
Simulator Communicator:		**Acknowledge requests for assistance.**
Lead Evaluator:		**Once the crew completes start of the standby Air Handler and Tech Specs have been evaluated, cue Simulator Operator to insert Trigger 2** **Event 2: Letdown Temperature Control Failure**

Op Test No.:	NRC	Scenario #	3	Event #	2	Page	12	of	66
Event Description:			**Letdown Temperature Control Failure**						

Time	Position	Applicant's Actions or Behavior	
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 2 "Letdown Temperature Control Failure"**	
Indications Available		• **ALB-007-3-2, DEMIN FLOW DIVERSION HIGH TEMP** • **TK-144 output lowers to 0** • **TI-143 temperature rising**	
ALB-007	RO	RESPONDS to alarm on APP-ALB-030-6-4	
	RO	CONFIRM alarm using TI-143, LP Letdown Temperature.	
	RO	VERIFY Automatic Functions:	
		• 1CS-50, Letdown to VCT/Demin, diverts flow to the VCT, bypassing the BTRS and Purification Demineralizers **(Manually positions 1CS-50, Letdown to VCT/Demin, to divert flow to the VCT)**	(NO)
	RO	PERFORM Corrective Actions:	
		• VERIFY that 1CS-50 diverts flow to the VCT, bypassing the BTRS and Purification • PERFORM the following as needed to lower letdown temperature: o VERIFY proper charging flow is established. o LOWER letdown flow. o IF CCW flow to the Letdown Heat Exchanger appears low, THEN: ➢ TAKE manual control of TK-144. ➢ OPEN 1CC-337, to raise CCW flow.	(YES) (YES) (N/A) (YES)
	SRO	Directs RO to maintain a TK-144 outlet temperature controlling band of 110°F to 120°F per OP-107.	

Op Test No.:	NRC	Scenario # 3	Event #	2	Page	13	of	66
Event Description:		**Letdown Temperature Control Failure**						

Time	Position	Applicant's Actions or Behavior	
		• IF letdown temperature can NOT be lowered, THEN REFER TO OP-107, Chemical and Volume Control System, AND PERFORM the following: ○ REMOVE letdown from service. ○ IF desired, THEN PLACE Excess Letdown in service.	(NO) (NO) (N/A)
	SRO	NOTIFY RP that due to high temperatures closure of 1CS-50 has bypassed the demineralizers. Surveillance is necessary to identify areas in the plant that could have experienced changes to radiological conditions.	
	SRO	Completes an Emergent Issue Checklist and contacts WCC for assistance. (WR, and Maintenance support)	
Simulator Communicator:		**If contacted as WCC, System Engineer or Chemistry: Direct the control room to maintain flow bypassing the demineralizers until a resin damage assessment is completed.**	
Lead Evaluator:		**When letdown temperature is under control, cue Simulator Operator to insert Trigger 3** **Event 3: Main Condenser Evacuation Pump 'A' trip**	

Op Test No.:	NRC	Scenario #	3	Event #	3	Page	14	of	66
Event Description:	**Main Condenser Evacuation Pump 'A' trip**								

Time	Position	Applicant's Actions or Behavior	
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 3 "Main Condenser Evacuation Pump 'A' trip"**	
Indications Available		**• ALB-021-4-1, CONDENSER VACUUM PUMP A TRIP • MCES 'A' Pump MCB switch green light lit**	
ALB-021	BOP	RESPONDS to alarm on APP-ALB-021-4-1	
	BOP	CONFIRM alarm using:	
		• Condenser Vacuum Pump status • Condenser vacuum indication	
	BOP	VERIFY Automatic Functions:	
		• Standby Vacuum Pump auto-starts on rising condenser pressure only if running Vacuum Pump has not tripped. **(Manually starts MCES 'B' Pump from MCB)**	(NO)
	BOP	PERFORM Corrective Actions:	
		• IF Condenser vacuum is degrading, THEN GO TO AOP-012, Partial Loss of Condenser Vacuum. • CHECK Vacuum Pump breaker indication (MCB). • IF necessary, THEN START the standby Vacuum Pump. • DISPATCH an Operator to check operation of seal water make-up to Vacuum Pump.	(N/A) (YES)
Simulator Communicator:		**When contacted to investigate pump trip report back in 2 minutes that breaker 1D3 Cubical 3D is tripped on overload and no problems are noted locally at the Vacuum Pump.**	
	SRO	Completes an Emergent Issue Checklist and contacts WCC for assistance. (WR, and Maintenance support)	

Op Test No.:	NRC	Scenario #	3	Event #	3	Page	15	of	66

Event Description: **Main Condenser Evacuation Pump ‘A’ trip**

Time	Position	Applicant’s Actions or Behavior

Simulator Communicator:		**Acknowledge requests for assistance.**
Lead Evaluator:		**Once the crew completes start of the standby MCES ‘B’ Pump, cue Simulator Operator to insert Trigger 4** **Event 4: Reactor Vessel Flange leak of ~ 15 gpm**

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	16	of 66
Event Description:			**Reactor Vessel Flange leak of ~ 15 gpm**					

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 4 "Reactor Vessel Flange leak of ~ 15 gpm"**
Indications Available:		• **ALB-10-5-5, REACTOR VESSEL FLANGE LEAKOFF HIGH TEMP** • **TI-401, Reactor Vessel Flange Leakoff Temp rising**
Evaluator Note:		**Responding to the annunciator will direct the operator to shut 1RC-46, Head Flange Seal leakoff Line Isolation to stop leakage from the inner Reactor head seal. With the condition clear the crew may not enter AOP-016.**
ALB-010	RO	Responds to alarm and evaluates APP-ALB-010-5-5
		• CONFIRM alarm using: o TI-401 o Reports TI-401 reading or trending high. • VERIFY Automatic Functions: None • PERFORM Corrective Actions: o CHECK containment temperature trend for high containment temperature resulting from a nearby steam/RCS leak (NONE) o Shut 1RC-46, Head Flange Seal Leakoff Line Isolation to stop leakage from inner Reactor head seal o Monitors TI-401 indications and identifies temperature is lowering
	RO	Informs SRO Reactor Vessel Flange leakage is isolated
	SRO	• Completes an Emergent Issue Checklists for the failure of the Rx Vessel Flange. • Contacts WCC to coordinate Containment entry per AP-545 (WR, LCOTR and Maintenance support).

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	17	of	66	
Event Description:		**Reactor Vessel Flange leak of ~ 15 gpm**								
Time	Position	Applicant's Actions or Behavior								

Evaluator Note:		**Any Tech Spec evaluation can be conducted with a follow up question after the scenario. Leakrate may not be easily determinable due to changing RCS Temperature and may require Engineering assistance**	
	SRO	Evaluates Reactor Coolant System TS 3.4.6.2 Reactor Coolant System operational leakage shall be limited to: d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System. ACTION **b.** - With any Reactor Coolant System operational leakage greater than anyone of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.	
Evaluator Note:		**The following write up is if AOP-016 is used for the response to the Reactor Vessel Flange Leak.**	
	CREW	Identifies entry conditions to AOP-016, Excessive Primary Plant Leakage are met	
AOP-016		Excessive Primary Plant Leakage	
	SRO	ENTERS and directs actions of AOP-016, Conducts a Crew Update Makes PA announcement for AOP entry	
Procedure Note:		**This procedure contains no immediate actions.**	
	OATC	CHECK RHR in operation	(NO)
	SRO	REFER TO PEP-110, Emergency Classification And Protective Action Recommendations, AND ENTER the EAL Matrix.	

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	18	of	66
Event Description:				**Reactor Vessel Flange leak of ~ 15 gpm**					

Time	Position	Applicant's Actions or Behavior	
	OATC	CHECK RCS leakage within VCT makeup capability May report that the leak is exceeding Tech Spec SG leakage.	(YES)
Procedure Note:		If CSIP suction is re-aligned to the RWST, negative reactivity addition should be anticipated.	
	OATC	MAINTAIN VCT level GREATER THAN 5%	(YES)
	SRO	RNO: GO TO Step 10.	
	OATC	CHECK valid CNMT Ventilation Isolation monitors (REM-3561A, B, C and D) ALARM CLEAR CHECK RM 3502A, RCS Leak Detection Radiation Monitor, ALARM CLEAR CHECK ALL valid Area Radiation Monitors ALARM CLEAR CHECK valid Stack Monitors ALARM CLEAR	(YES) (YES) (YES) (YES)
	SRO	DETERMINE if unnecessary personnel should be evacuated from affected areas, as follows:	
		CHECK that a valid RMS Secondary Monitor HIGH ALARM CHECK that an RCS leak outside Containment, other than SG tube leakage, has caused a valid RMS alarm.	(NO) (NO)
	SRO	RNO: GO TO Step 15.	
	BOP	NOTIFY Chemistry to stop any primary sampling activities.	
Simulator Communicator:		**Acknowledge request to stop primary sampling activities.**	

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	19	of	66
Event Description:		**Reactor Vessel Flange leak of ~ 15 gpm**							

Time	Position	Applicant's Actions or Behavior
Procedure Note:		• The following qualitative flow balance is to quickly determine if RCS leakage exceeds Tech Spec limits, EAL classification thresholds, or RCS makeup capability. • RCS influent and effluent flow rates are compared and PRZ level rate of change is used to determine the RCS flow balance.
	OATC	PERFORM a qualitative RCS flow balance, as follows: a. ESTIMATE leak rate considering the following parameters: • PRZ level rate of change (~55 gal/% at 653°F) • Charging flow • Total seal injection flow • Letdown flow • Total seal return flow Reports estimate to SRO of ~ 15 gpm
		b. OPERATE the following letdown orifice valves as necessary to maintain charging flow on scale: • 1CS-7, 45 gpm Letdown Orifice A • 1CS-8, 60 gpm Letdown Orifice B • 1CS-9, 60 gpm Letdown Orifice C (No changes required)
Procedure Note:		Performance of surveillance tests to determine if leakage exceeds Tech Spec limits, or to more accurately quantify leakage is up to CRS discretion.
	SRO	Determines that more accurate quantification is not needed due to excessive leakage indications present.
Evaluator Note:		**Any Tech Spec evaluation can be conducted as a follow up question after the scenario.**
	SRO	EVALUATE RCS leakage (refer to Tech Spec 3.4.6.2).

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	20	of	66
Event Description:		**Reactor Vessel Flange leak of ~ 15 gpm**							

Time	Position	Applicant's Actions or Behavior
		Reviews Reactor Coolant System TS 3.4.6.2 Reactor Coolant System operational leakage shall be limited to: d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System. ACTION **b.** - With any Reactor Coolant System operational leakage greater than anyone of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
	SRO	DETERMINE leak location from one or more of the following: MCB indications and Valid Radiation Monitors • From RV Flange
	BOP	NOTIFY Health Physics of the following: a. Leak location: • Source inside or outside CNMT • To closed system, SG or to atmosphere b. Applicable radiation levels. NOTIFY HP of Reactor Vessel Flange leakage
Simulator Communicator:		**Acknowledge RCs leakage is coming from Reactor Vessel Flange.**
	SRO	WHEN leakage location has been determined, THEN PERFORM the applicable Attachment: Leakage From RV Flange Attachment 6 page 28
	SRO	Transitions to Attachment 6: • Consult with Operation Management to determine leak isolation and recovery actions

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	21	of	66
Event Description:	**Reactor Vessel Flange leak of ~ 15 gpm**								

Time	Position	Applicant's Actions or Behavior
Procedure Note:		Radiation Control personnel must identify radiological conditions or provide coverage and issue a special RWP prior to CNMT entry.
	SRO	IF CNMT entry is made to isolate the leak, THEN VERIFY valves manipulated for leak isolation are documented per the following: • OMM-001, Operations Administrative Requirements • OPS-NGGC-1303, Verification Practices
	SRO	Completes an Emergent Issue Checklists for the failure of the Rx Vessel Flange. Contacts WCC to coordinate Containment entry per AP-545 (WR, LCOTR and Maintenance support).
	SRO	Exit AOP-016
Evaluator Note:		**After Rx Vessel leakage has stabilized, cue Simulator Operator to insert Trigger 5** **Event 5: 'A' MFP trips with 'B' MFP failure to start AOP-010).**

Op Test No.:	NRC	Scenario #	3	Event #	5	Page	22	of 66
Event Description:	**MFP 'A' trips with MFP 'B' failure to start (AOP-010)**							

Time	Position	Applicant's Actions or Behavior	
Simulator Operator:		**On cue from the Lead Evaluator insert Trigger 5** **"MFP 'A' trips with 'B' MFP failure to start (AOP-010)"**	
Indications Available:		**ALB-016-1-4, FW PUMP A/B O/C TRIP –GND OR BKR FAIL TO CLOSE** **Multiple FW flow alarms**	
	BOP	RESPONDS to alarms and ENTERS AOP-010	
AOP-010		Feedwater Malfunctions	
	SRO	ENTERS and directs actions of AOP-010 Conducts a Crew Update Makes PA announcement for AOP entry	
Procedure Note:		**Steps 1 through 4 are immediate actions.**	
Immediate Action	BOP	CHECK Feedwater Regulator valves operating properly.	(YES)
Immediate Action	BOP	CHECK ANY Main Feedwater Pump TRIPPED	(YES)
Immediate Action	BOP	CHECK initial Reactor power less than 90%.	(YES)
Immediate Action	BOP	CHECK initial Reactor power less than 80%.	(YES)
Procedure Note:		Turbine runback will automatically terminate at approximately 50% power. Turbine runbacks are identified as follows: • ALB-20/2-2, TURBINE RUNBACK OPERATIVE in alarm • TCS Runback in Urgent Priority alarm	
	BOP	CHECK initial Reactor power less than 60%.	(YES)

Op Test No.:	NRC	Scenario #	3	Event #	5	Page	23	of	66

Event Description: **MFP ‘A’ trips with MFP ‘B’ failure to start (AOP-010)**

Time	Position	Applicant’s Actions or Behavior	
	BOP	MAINTAIN ALL of the following: • At least ONE Main Feedwater Pump RUNNING • Main Feedwater flow to ALL Steam Generators • ALL Steam Generator levels greater than 30%	(NO) (YES) (YES)
	SRO	RNO: PERFORM the following:	
Critical Task # 1	BOP	• IF ANY SG level drops to 30% THEN TRIP the Reactor AND GO TO EOP-E-0.Places SG ‘B’ Feedwater Reg valve in MANUAL • IF Above POAH AND Reactor power is LESS THAN 10%, THEN: ○ INITIATE AFW flow to maintain Steam Generator levels between 52 and 62%. ***Critical Task: Manually start AFW flow to maintain control of SG level above 25% to prevent an automatic Reactor trip after trip of the last running Main Feed Pump***	(NO) (YES)
Procedure Note:		Mode change occurs at 5% Reactor power.	
	SRO	○ REDUCE power as necessary to maintain SG level.	
		• IF below POAH, THEN:	(NO)
	BOP	CHECK Feedwater Regulator Valves operating properly in AUTO: • Response to SG levels • Valve position indication • Response to feed flow/steam flow mismatch	(YES)
Procedure Note:		Inability to monitor one or more Safety System Parameters concurrent with a turbine runback of greater than 25%, requires a change of event classification per the HNP Emergency Plan. [C.2, C.3]	

Op Test No.:	NRC	Scenario #	3	Event #	5	Page	24	of 66
Event Description:	**MFP 'A' trips with MFP 'B' failure to start (AOP-010)**							
Time	Position	Applicant's Actions or Behavior						

	BOP	CHECK turbine runs back less than 25% turbine load	YES
Procedure Note:		A feedwater train consists of a Condensate Pump, Condensate Booster Pump and Main Feedwater Pump.	
	SRO	GO TO the applicable section: EVENT: Loss of Running Pumps Section 3.2 Page 14	
Procedure Note:		• A feedwater train consists of a Condensate Pump, Condensate Booster Pump and Main Feedwater Pump. • Target load for loss of a running pump is as follows: o One Heater Drain Pump with both FW Trains operating: Less than 100% Power o No Heater Drain Pumps with both FW Trains operating: 90% turbine load. o Single Main Feedwater Pump running with both Condensate Pumps and both Condensate Booster Pumps operating: 7.0 mpph Total FW Flow. o Single Feedwater Train with both Heater Drain Pumps operating: 7.0 mpph Total FW Flow. o Single Feedwater Train operating: 5.5 mpph Total FW Flow.	
	BOP	MAINTAIN ALL of the following: • At least ONE Main Feedwater Pump RUNNING • Main Feedwater flow to ALL Steam Generators • ALL Steam Generator levels greater than 30%	 (NO) (YES) (YES)
	SRO	RNO: PERFORM the following:	

Op Test No.: NRC	Scenario # 3	Event # 5	Page 25 of 66
Event Description:	**MFP 'A' trips with MFP 'B' failure to start (AOP-010)**		

Time	Position	Applicant's Actions or Behavior	
Critical Task # 1	BOP	• IF ANY SG level drops to 30% THEN TRIP the Reactor AND GO TO EOP-E-0.Places SG 'B' Feedwater Reg valve in MANUAL • IF Above POAH AND Reactor power is LESS THAN 10%, THEN: ○ INITIATE AFW flow to maintain Steam Generator levels between 52 and 62%. ***Critical Task: Manually start AFW flow to maintain control of SG level above 25% to prevent an automatic Reactor trip after trip of the last running Main Feed Pump***	(NO) (YES)
Procedure Note:		Mode change occurs at 5% Reactor power.	
	SRO	○ REDUCE power as necessary to maintain SG level.	
		• IF below POAH, THEN:	(NO)
	RO	CHECK control rods INSERTING to reduce Tavg - Tref mismatch. **(Rod Control is in Manual at this time)**	
	BOP	CHECK Main Steam pressure less than PORV controller setpoint. (nominally 1106 psig).	(YES)
Procedure Caution:		Improper operation of the Steam Dumps while in manual control can lead to excessive SG swell or overcooling of the RCS.	
	BOP	CHECK proper Steam Dump Valve operation.	(YES)
	BOP	CHECK SG levels TRENDING to between 52% and 62%.	(YES)

Op Test No.:	NRC	Scenario #	3	Event #	5	Page	26	of	66

Event Description: **MFP 'A' trips with MFP 'B' failure to start (AOP-010)**

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>RO</td><td>CHECK PZR PORVs SHUT.</td><td>(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td>CHECK PZR pressure TRENDING to 2235 psig.</td><td>(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td>CHECK PZR Level TRENDING to reference level.</td><td>(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td colspan="2">ALIGN Main Feedwater Pump control switches, as applicable:
<u>Pumps</u>
• Tripped Pump - STOP (spring-return to AUTO)
• Auto-started Pump - START (spring-return to AUTO)

<u>Pump Recirc Valves</u>
• Tripped Pump - SHUT
• Auto-started Pump - MODU</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td>CHECK BOTH Heater Drain Pumps TRIPPED.</td><td>(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td>CHECK the following high-high level alarms CLEAR:</td><td>(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Procedure Note:</td><td colspan="2">A feedwater train consists of a Condensate Pump, Condensate Booster Pump and Main Feedwater Pump.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td>CHECK load less than or equal to target based on final condition.
<table>
<tr><th colspan="2">TARGET</th></tr>
<tr><th>Condition</th><th>Load</th></tr>
<tr><td>One HDP Running</td><td>Less than 100% Power</td></tr>
<tr><td>No HDPs Running</td><td>90% Turbine Load</td></tr>
<tr><td>One MFP Running</td><td>7.0 mpph Total FW Flow</td></tr>
<tr><td>Single FW Train - both HDPs Running</td><td>7.0 mpph Total FW Flow</td></tr>
<tr><td>Single FW Train</td><td>5.5 mpph Total FW Flow</td></tr>
</table>
</td><td>(YES)</td></tr>
</table>

Op Test No.:	NRC	Scenario #	3	Event #	5	Page	27	of 66
Event Description:	**MFP 'A' trips with MFP 'B' failure to start (AOP-010)**							

Time	Position	Applicant's Actions or Behavior	
	BOP	DISPATCH an operator to check the following seated, observing tailpipes: • MSR Relief Valves • SG Safety Valves	
Simulator Communicator:		**Acknowledge communications** **After 2-3 minutes report back that nothing is abnormal with the tailpipes and no leaks were found** **IF contacted by MCR to investigate the causes of the "A" and later the "B" MFW pump trip report that both breakers have tripped on overcurrent. There are no signs of damage at the pumps.**	
	BOP	CHECK Hotwell level trending to between 71% and 76%.	(YES)
	BOP	RESET Loss of Load interlocks C7A and C7B, as follows: **(Steam Dumps are in Steam Pressure Mode at this time)**	
	SRO	NOTIFY Load Dispatcher of ANY load limitations. **(Generator is not connected to the Grid at this time)**	
	SRO	CHECK Reactor thermal power changed by less than 15% in any one hour period. [C.1] Within 1.5 hours of load rejection, CHECK control rods above insertion limits.	(YES) (YES)
	SRO	EXIT this procedure.	
	SRO	Completes an Emergent Issue Checklist and contacts WCC for assistance. (WR, and Maintenance support)	

Op Test No.: NRC Scenario # 3 Event # 5 Page 28 of 66		
Event Description: **MFP ‘A’ trips with MFP ‘B’ failure to start (AOP-010)**		
Time	Position	Applicant’s Actions or Behavior

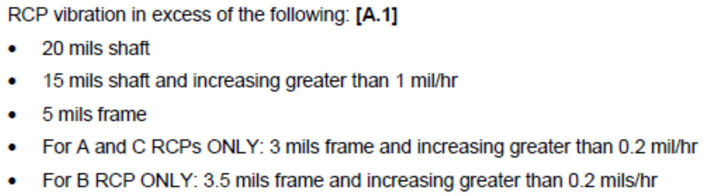
Simulator Communicator:		**IF WCC is contacted then report that Electrical Maintenance is investigating the problems with the breakers any repairs will be made as quickly as possible.**
Lead Evaluator:		**Once the plant has stabilized, cue Simulator Operator to insert Trigger 6** **Event 6: RCP ‘B’ rising vibration (AOP-018)**

Op Test No.:	NRC	Scenario #	3	Event #	6	Page	29	of 66
Event Description:		**RCP 'B' rising vibration (AOP-018)**						

Time	Position	Applicant's Actions or Behavior

Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 6 "RCP 'B' rising vibration (AOP-018)"**	
Available Indications		• **ALB-010-2-5, RCP-B TROUBLE** • **RCP 'B' vibration monitors rising and red high vibration lights lit**	
	RO	RESPONDS to alarms and ENTERS AOP-018	
AOP-018		Reactor Coolant Pump Abnormal Conditions	
Immediate Action	RO	CHECK any CSIP running.	(YES)
	SRO	ENTERS and directs actions of AOP-018, Conducts a Crew Update Makes PA announcement for AOP entry	
	SRO	REFER TO PEP-110, Emergency Classification And Protective Action Recommendations, AND ENTER the EAL Matrix.	
Procedure Note:		Minimum allowable flow for a CSIP is 60 gpm which is provided by normal miniflow during normal operation and alternate miniflow during safety injection. Maintaining CSIP flow greater than or equal to 60 gpm also satisfies this requirement.	
		EVALUATE plant conditions AND GO TO the appropriate section: MALFUNCTION: High Reactor Coolant Pump Vibration, Section 3.2 Page 8	
Evaluator Note:		**The following question may be YES at this time but the limit will be exceeded quickly. This is a continuous action step and implemented when the limit is exceeded. The guide is therefore written as if the limit is exceeded.**	
	RO	CHECK ALL RCPs operating within the limits of	(NO)

Op Test No.: NRC	Scenario # 3	Event # 6	Page	30	of	66
Event Description:	**RCP 'B' rising vibration (AOP-018)**					

Time	Position	Applicant's Actions or Behavior	
		Attachment 1 (Page 22).  RCP vibration in excess of the following: **[A.1]** • 20 mils shaft • 15 mils shaft and increasing greater than 1 mil/hr • 5 mils frame • For A and C RCPs ONLY: 3 mils frame and increasing greater than 0.2 mil/hr • For B RCP ONLY: 3.5 mils frame and increasing greater than 0.2 mils/hr	
	SRO	RNO: GO TO STEP 3	
	RO	CHECK the Reactor is TRIPPED.	(NO)
Evaluator Note:		**The SRO should conduct a Crew Update and review AOP-018 Section 3.2 steps 4 through 7 and direct these actions to be performed after the E-0 immediate actions are ensured complete.**	
	SRO	RNO: TRIP the Reactor AND GO TO EOP-E-0. (Perform Steps 4 through 7 as time permits.)	
Evaluator Note:		**Upon entering EOP-E-0, Rx WILL NOT trip from RPS or MCB switches**	
EOP-E-0		REACTOR TRIP OR SAFETY INJECTION	
	SRO	Directs manual Reactor Trip	
	RO	Initiates a MANUAL Reactor trip. from center section of the Main Control Board (switch is failed).	
	RO	Attempts to initiate a MANUAL Reactor Trip from left section of the Main Control Board (switch is failed).	
	SRO	IF the reactor will not trip after using both MCB manual trip switches, THEN go to FR-S.1, "RESPONSE TO NUCLEAR POWER GENERATION/ATWS", Step 1.	

Op Test No.:	NRC	Scenario #	3	Event #	7	Page	31	of 66
Event Description:	**Failure of the Reactor Trip breakers to open auto or manual (EOP-FR-S.1)**							

Time	Position	Applicant's Actions or Behavior

Simulator Communicator:		**During the ATWS - the crew makes a PA announcement for an operator to contact or report to the MCR for instructions to locally trip the Reactor. CALL the MCR as the TB AO and get the instructions.**	
Simulator Operator:		**After the TB AO has received instructions to locally trip the Reactor, wait 1 minute then run TRG-15. Trigger 15 will delete the ATWS malfunction (RPS01B) and trip the Reactor locally. After running TRG-15 call MCR and report that the Rx trip breakers were locally opened.**	
Evaluator Note:		**EOP-FR-S.1 is the first transition step from EOP-E-0 and contains immediate action step required to be performed from memory. Because of this the SRO may proceed directly to EOP-FR-S.1.**	
EOP- FR-S.1		Response to Nuclear Power Generation / ATWS	
Procedure Caution:		To maximize core cooling, RCPs should NOT be tripped with reactor power GREATER THAN 5%. (Normal support conditions for running RCPs are NOT required for these circumstances. The RCP TRIP CRITERIA for small break LOCA conditions is NOT applicable to this procedure.)	
Procedure Note:		Steps 1 through 2 are immediate action steps.	
Immediate Action	RO	Ensure Reactor Trip: • Reactor Trip AND Bypass BKRs – OPEN • Rod bottom lights (Zero Steps) – LIT • Neutron flux – DROPPING	(NO) (NO) (NO)

Op Test No.:	NRC	Scenario #	3	Event #	7	Page	32 of 66
Event Description:	**Failure of the Reactor Trip breakers to open auto or manual (EOP-FR-S.1)**						

Time	Position	Applicant's Actions or Behavior	
	RO	RNO actions: IF the reactor will NOT trip (automatically AND after using both manual trip switches), THEN verify negative reactivity inserted by any of the following while continuing with this procedure: • Manually insert control rods • Ensure control rods inserting in automatic	
Immediate Action	BOP	Check Turbine Trip: • All turbine throttle valves – SHUT	(NO)
	BOP	RNO actions: Manually Trip Main Turbine from MCB	
Evaluator Note:		**When the Main Turbine is tripped RCS pressure will rapidly raise and one or more Pressurizer PORV's will lift. With RCS break flow occurring, the RCS pressure will steadily drop. SG pressure will also rapidly rise and cause all SG PORV's to OPEN and most of the SG safety valves to lift.**	
	BOP	Ensure All AFW Pumps – RUNNING (Starts ALL available AFW pumps)	
	SRO	Direct an operator to report to the main control room to receive instructions for local actions (Local Actions to trip the reactor or turbine are directed in step 9).	
	SRO	Inform SM to Evaluate EAL Matrix (Refer to PEP-110).	

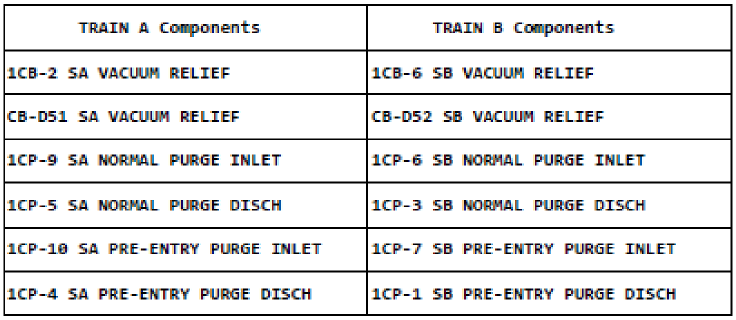
Op Test No.:	NRC	Scenario #	3	Event #	7	Page	33	of	66
Event Description:	**Failure of the Reactor Trip breakers to open auto or manual (EOP-FR-S.1)**								

Time	Position	Applicant's Actions or Behavior	
Procedure Note:		Actuation of the sequencer inhibits operation of the boric acid pumps. (If the sequencer runs on Program A, the pumps can be started manually after LB-9. Otherwise, the sequencer must be reset to restore operation of the pumps) SI flow accomplishes emergency boration.	
Evaluator Note:		**After the reactor is tripped, RCS pressure will rapidly lower to the Auto SI setpoint (1850 psig). The crew may/may not have time to manually actuate SI; as such, there is no problem with the crew NOT performing a manual SI.** **After the reactor is tripped, the 'A' CSIP will trip on an electrical fault, and the Safeguards Sequencer will fail to start the 'B' CSIP.**	
	RO	Initiate Emergency Boration of RCS: • Check SI flow – GREATER THAN 200 GPM. • Emergency borate from the BAT: o Start a boric acid pump. • Perform any of the following (listed in order of preference): o Open Emergency Boric Acid Addition valve: ▪ 1CS-278 o Open normal boration valves: ▪ FCV-113A ▪ FCV-113B • Ensure boric acid flow to CSIP suction – AT LEAST 30 GPM • Ensure CSIP flow to RCS – AT LEAST 30 GPM	(NO)
	RO	Check PRZ Pressure – LESS THAN 2335 PSIG.	(YES)
	SRO	Go to Step 8.	

Op Test No.: NRC Scenario # 3 Event # 7 Page 34 of 66

Event Description: **Failure of the Reactor Trip breakers to open auto or manual (EOP-FR-S.1)**

Time	Position	Applicant's Actions or Behavior	
	BOP	Isolate CNMT Ventilation: • Stop the following fans: (If running) ○ AH-82A NORMAL PURGE SUPPLY FAN ○ AH-82B NORMAL PURGE SUPPLY FAN ○ E-5A CNMT PRE-ENTRY PURGE EXHAUST FAN ○ E-5B CNMT PRE-ENTRY PURGE EXHAUST FAN	
	BOP	Ensure the valves and dampers listed in the table – SHUT.	(YES)



TRAIN A Components	TRAIN B Components
1CB-2 SA VACUUM RELIEF	1CB-6 SB VACUUM RELIEF
CB-D51 SA VACUUM RELIEF	CB-D52 SB VACUUM RELIEF
1CP-9 SA NORMAL PURGE INLET	1CP-6 SB NORMAL PURGE INLET
1CP-5 SA NORMAL PURGE DISCH	1CP-3 SB NORMAL PURGE DISCH
1CP-10 SA PRE-ENTRY PURGE INLET	1CP-7 SB PRE-ENTRY PURGE INLET
1CP-4 SA PRE-ENTRY PURGE DISCH	1CP-1 SB PRE-ENTRY PURGE DISCH

Time	Position	Applicant's Actions or Behavior	
Evaluator Note:		**The following actions will be complete IF local AO actions have been completed; IF NOT the RNO steps will be directed by the Crew.**	
	RO	Check Trip Status: • Check Reactor – TRIPPED	(NO)
	SRO	RNO actions: Direct and AO to perform the following local actions: Locally trip the reactor using any of the following (listed in order of preference): • Locally trip both reactor trip breakers: • Locally trip both reactor trip bypass breakers: • Locally trip both rod drive MG set generator output breakers: • Locally trip both rod drive MG set motor breakers:	

Op Test No.: NRC	Scenario # 3 Event # 7	Page 35 of 66
Event Description:	**Failure of the Reactor Trip breakers to open auto or manual (EOP-FR-S.1)**	
Time	Position	Applicant's Actions or Behavior

	BOP	Check turbine – TRIPPED	(YES)
Evaluator Note:		**The following actions will be performed once ATWS condition is clear (1 minute after the crew directs the Local AO actions).** **After the reactor is tripped, the 'A' CSIP will trip on an electrical fault,** **IF ATWS condition is not clear the REACTOR SUBCRITICALITY CRITERIA FOLDOUT will apply and the Crew will continue with EOP-FR-S.1 until the foldout criteria is satisfied.**	
	RO	Check Reactor Subcritical: • Check for both of the following: o Power range channels – LESS THAN 5% o Intermediate range startup rate channels – NEGATIVE	(YES) (YES)
	SRO	Observe CAUTION prior to Step 25 and go to Step 25.	
Procedure Caution:		Boration should continue to obtain adequate shutdown margin during subsequent recovery actions.	
	SRO	Initiate Monitoring of Critical Safety Function Status Trees.	
	SRO	Return to Procedure And Step In Effect.	
	SRO	Returns to procedure in effect (EOP-E-0, Step 1)	
Evaluator Note:		**The SRO should return to EOP-E-0, Step 1**	

Op Test No.: NRC Scenario # 3 Event # 8 Page 36 of 66

Event Description: **Small Break LOCA (EOP-E-0)**

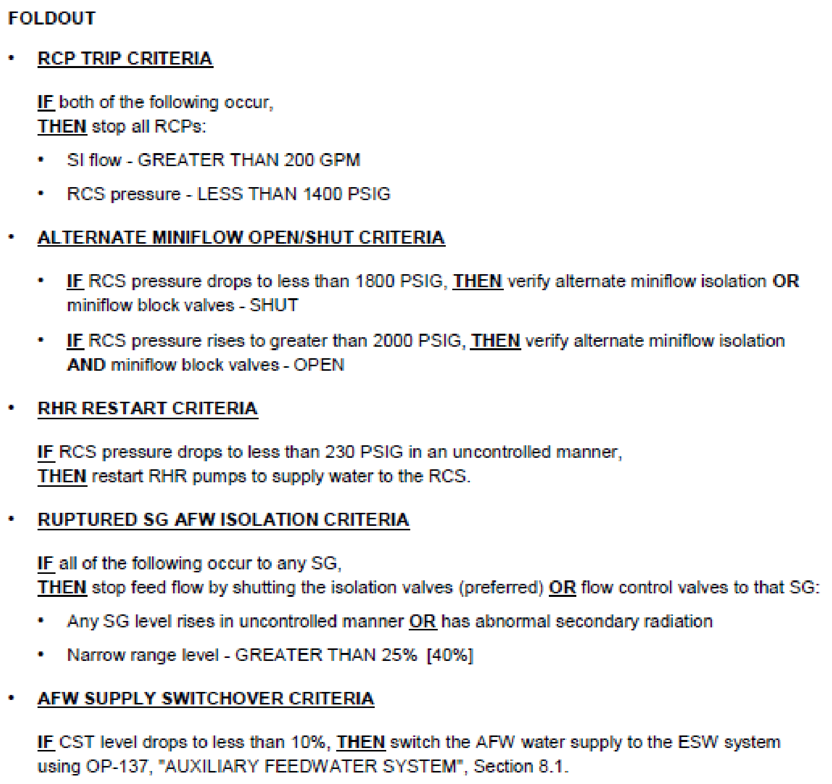
Time	Position	Applicant's Actions or Behavior	
	SRO	Transitions to EOP-E-0, Step 1	
EOP-E-0		Reactor Trip Or Safety Injection	
	SRO	Enters EOP-E-0 Holds crew update	
	RO/BOP	Re-performs E-0 Immediate Actions.	
Immediate Actions	RO	VERIFY Reactor Trip: REACTOR TRIP CONFIRMATION Reactor Trip AND Bypass BKRs – OPEN Rod Bottom Lights (Zero Steps) – LIT Neutron Flux – DROPPING	(YES) (YES) (YES)
Immediate Actions	BOP	Check Turbine Trip – ALL THROTTLE VALVES SHUT TURB STOP VLV 1 – TSLB-2-11-1 TURB STOP VLV 2 – TSLB-2-11-2 TURB STOP VLV 3 – TSLB-2-11-3 TURB STOP VLV 4 – TSLB-2-11-4	(YES) (YES) (YES) (YES)
Immediate Actions	BOP	Perform The Following: • AC Emergency Buses – AT LEAST ONE ENERGIZED • AC Emergency Buses – BOTH ENERGIZED	(YES) (YES)

Op Test No.: NRC Scenario # 3 Event # 8 Page 37 of 66

Event Description: **Small Break LOCA (EOP-E-0)**

Time	Position	Applicant's Actions or Behavior	
Evaluator Note:		**After the reactor is tripped, RCS pressure will rapidly lower, to the Auto SI setpoint (1850 psig). The crew may/may not have time to manually actuate SI; as such, there is no problem with the crew NOT performing a manual SI.** **After the reactor is tripped, the 'A' CSIP will trip on an electrical fault, and the Safeguards Sequencer will fail to start the 'B' CSIP.**	
Immediate Actions	RO	Safety Injection – ACTUCATED (BOTH TRAINS) BPLP 4-1,"SI ACTUATED" - LIT (CONTINUOUSLY)	(YES)
Procedure Note:		Steps 1 through 4 are immediate action steps Foldout applies. (Immediate actions should be completed prior implementing Foldout Page items.)	
Evaluator Note:		**Following completion of the EOP-E-0 immediate actions the RO should complete AOP-018, section 3.2 Step 4-7 as directed by the SRO prior to the ATWS Event.**	
AOP-018		Reactor Coolant Pump Abnormal Conditions	
	SRO	Directs RO/BOP to secure the RCP 'B' and continue with AOP-018 steps 4-7	
	RO/BOP	STOPS RCP 'B' and places PK-444D.1 to manual then shuts valve with demand at 0%	

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	38	of	66
Event Description:	**Failure of 'B' Sequencer Load Block 1 to actuate during the Safety Injection which fails to start 'B' CSIP (EOP-E-0)**								

Time	Position	Applicant's Actions or Behavior	
	SRO	Reviews Foldout page	
Evaluator Note:		 **FOLDOUT** • **RCP TRIP CRITERIA** **IF** both of the following occur, **THEN** stop all RCPs: • SI flow - GREATER THAN 200 GPM • RCS pressure - LESS THAN 1400 PSIG • **ALTERNATE MINIFLOW OPEN/SHUT CRITERIA** • **IF** RCS pressure drops to less than 1800 PSIG, **THEN** verify alternate miniflow isolation **OR** miniflow block valves - SHUT • **IF** RCS pressure rises to greater than 2000 PSIG, **THEN** verify alternate miniflow isolation **AND** miniflow block valves - OPEN • **RHR RESTART CRITERIA** **IF** RCS pressure drops to less than 230 PSIG in an uncontrolled manner, **THEN** restart RHR pumps to supply water to the RCS. • **RUPTURED SG AFW ISOLATION CRITERIA** **IF** all of the following occur to any SG, **THEN** stop feed flow by shutting the isolation valves (preferred) **OR** flow control valves to that SG: • Any SG level rises in uncontrolled manner **OR** has abnormal secondary radiation • Narrow range level - GREATER THAN 25% [40%] • **AFW SUPPLY SWITCHOVER CRITERIA** **IF** CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1.	
	SRO	Assigns Foldout items: Alternate Miniflow Open/Shut Criteria, RHR Restart Criteria, AFW Supply Switchover Criteria Directs Shift Manager to Evaluate EAL Matrix (Refer to PEP-110)	
	RO	Ensure CSIPs – ALL RUNNING	(NO)
Critical Task # 2	RO	Checks Safeguards Sequencer has reached Load Block 9 (Manual Load Permissive) Starts 'B' CSIP ***(Critical to manually start 'B' CSIP to prevent RCS temperature from reaching 730°F and RVLIS Full Range Level from lowering below 39%.***	

Op Test No.:	NRC	Scenario #	3	Event #	8	Page	39	of	66

Event Description: **Small Break LOCA (EOP-E-0) Continued**

Time	Position	Applicant's Actions or Behavior	
Evaluator Note:		**The following actions should be taken in accordance with EOP-E-0 Foldout criteria during the scenario:** • **When RCP trip criteria is met per Foldout the crew should have the 'B' CSIP running, identify the condition and then trip all running RCP's** • **Ensure Alternate Miniflow Isolation Valves CLOSE or CLOSE the Miniflow Block Valves when RCS Pressure lowers to less than 1800 PSIG.**	
	RO	Ensure RHR pumps – ALL RUNNING	(YES)
	RO	Safety Injection flow – GREATER THAN 200 GPM	(YES)
Critical Task # 3	RO	Identifies Foldout RCP Trip Criteria is MET SI flow > 200 GPM RCS pressure < 1400 psig Informs SRO that RCP trip criteria is met Secures ALL RCPs ***(Critical to secure all RCPs with SI flow > 200 gpm and before RVLIS Full Range Level lowers below 39%.)***	
	RO	RCS pressure – LESS THAN 230 PSIG	(NO)
	SRO	RNO: GO TO Step 12.	
	BOP	MAIN Steam isolation – ACTUATED.	(NO)
	SRO	RNO: Check Main Steam Line Isolation - REQUIRED Perform the following: • IF Main Steam Isolation is NOT required, THEN go to Step 16.	(NO)

Op Test No.:	NRC	Scenario #	3	Event #	8	Page	40 of 66
Event Description:	**Small Break LOCA (EOP-E-0) Continued**						

Time	Position	Applicant's Actions or Behavior	
	RO	CHECK CNMT Pressure – HAS REMAINED LESS THAN 10 PSIG	(YES)
	RO/BOP	Ensure AFW flow – AT LEAST 200 KPPH ESTABLISHED	(YES)
	BOP	Sequencer Load Block 9 (Manual Loading Permissive) – ACTUATED (BOTH TRAINS)	(YES)
	BOP	Energize AC buses 1A1 AND 1B1	
Evaluator Note:		**The RO will perform all board actions until the BOP completes Attachment 3. The BOP is permitted to properly align plant equipment in accordance with EOP-E-0, Attachment 3 without SRO approval.** **The Scenario Guide still identifies tasks by board position because the time frame for completion of Attachment 3 is not predictable.** **EOP-E-0, Attachment 3, "Safeguards Actuation Verification" has been included as Attachment 3 (Pg 60 of 67) at the end of this scenario.**	
	BOP	Ensure Alignment of Components From Actuation of ESFAS Signals Using Attachment 3, "Safeguards Actuation Verification", While Continuing with this Procedure.	

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	41	of 66

Event Description: **Failure of Safety Injection Isolation valves on 'A' Train CSIP normal mini flow 1CS-214 fails to close automatically**

Time	Position	Applicant's Actions or Behavior
Event 8	BOP	Ensure SI Valves - PROPERLY ALIGNED (Refer to Attachment 1.) Identifies that 1CS-214, CSIP normal miniflow valve is not SHUT and manually shuts valve to align the system correctly. **(Attachment 1 is located in the back of this guide)**
	BOP	Directs AO to place 1A and 1B Air Compressor in the local control mode per E-0 Attachment 3 step 22
Simulator Communicator		**Acknowledge the request to place 1A and 1B Air Compressor in the local control mode per E-0 Attachment 3 step 22**
Simulator Operator		**When directed to place the 1A and 1B Air Compressor in the local control mode: Run APP\air\acs_to_local**
Simulator Communicator		**When the APP for 1A and 1B Air Compressor has completed running call the MCR and inform them that the air compressors are running in local control.**
	BOP	Directs AO to Unlock AND Turn ON The Breakers For The CSIP Suction AND Discharge Cross-Connect Valves per E-0 Attachment 3 step 23 (or from step 11 - refer to Attachment 2)
Simulator Communicator		**Acknowledge the request to Unlock AND Turn ON The Breakers For The CSIP Suction AND Discharge Cross-Connect Valves**
Simulator Operator		**When directed to Unlock AND Turn ON The Breakers for the CSIP Suction AND Discharge Cross-Connect Valves: Run APP\cvc\E-0 Att 2 CSIP suct & disch valve power.txt.**
Simulator Communicator		**When the APP for CSIP Suction AND Discharge Cross-Connect Valves has completed running call the MCR and inform them that CSIP Suction AND Discharge Cross-Connect Valves are energized.**

Op Test No.: NRC Scenario # 3 Event # 9 Page 42 of 66

Event Description: **Small Break LOCA (EOP-E-0) Continued**

Time	Position	Applicant's Actions or Behavior	
	BOP/RO	Stabilize And Maintain Temperature Between 555°F To 559°F Using Table 1.	

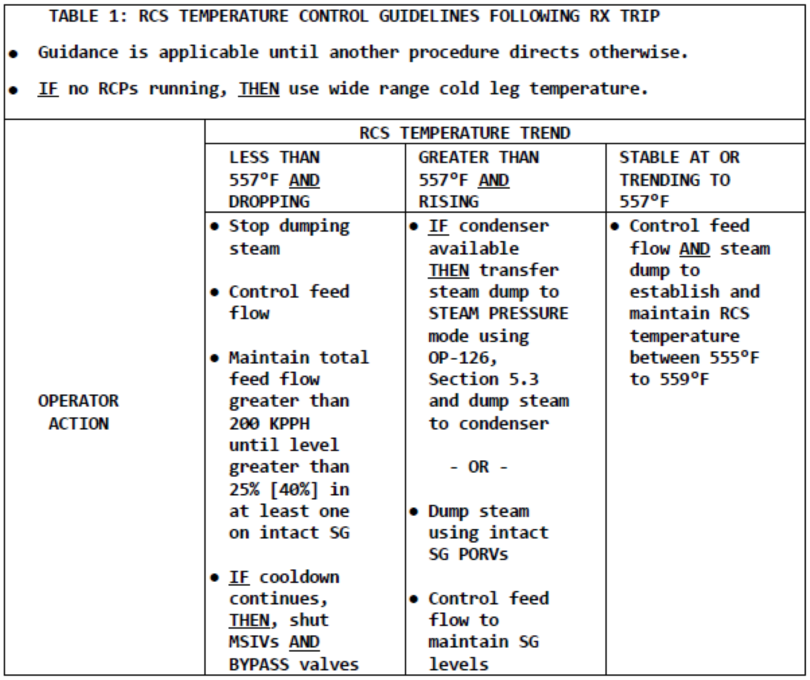


TABLE 1: RCS TEMPERATURE CONTROL GUIDELINES FOLLOWING RX TRIP

- Guidance is applicable until another procedure directs otherwise.
- IF no RCPs running, THEN use wide range cold leg temperature.

	RCS TEMPERATURE TREND		
	LESS THAN 557°F AND DROPPING	GREATER THAN 557°F AND RISING	STABLE AT OR TRENDING TO 557°F
OPERATOR ACTION	• Stop dumping steam • Control feed flow • Maintain total feed flow greater than 200 KPPH until level greater than 25% [40%] in at least one on intact SG • IF cooldown continues, THEN, shut MSIVs AND BYPASS valves	• IF condenser available THEN transfer steam dump to STEAM PRESSURE mode using OP-126, Section 5.3 and dump steam to condenser - OR - • Dump steam using intact SG PORVs • Control feed flow to maintain SG levels	• Control feed flow AND steam dump to establish and maintain RCS temperature between 555°F to 559°F

	CREW	Identifies RCS cooldown continues and shuts MSIV's NOTE: MSIVs may have been shut from MSLI	
	RO	PRZ PORVs – SHUT PRZ Spray Valves – SHUT (RCPs are secured) PRZ PORV Block Valves - AT LEAST ONE OPEN	(YES) (YES) (YES)
	SRO	Any SG pressure – DROPPING IN AN UNCONTROLLED MANNER OR COMPLETELY DEPRESSURIZED	(NO)
	SRO	RNO: GO TO Step 27.	
	SRO	Any SG - ABNORMAL RADIATION OR UNCONTROLLED LEVEL RISE	(NO)

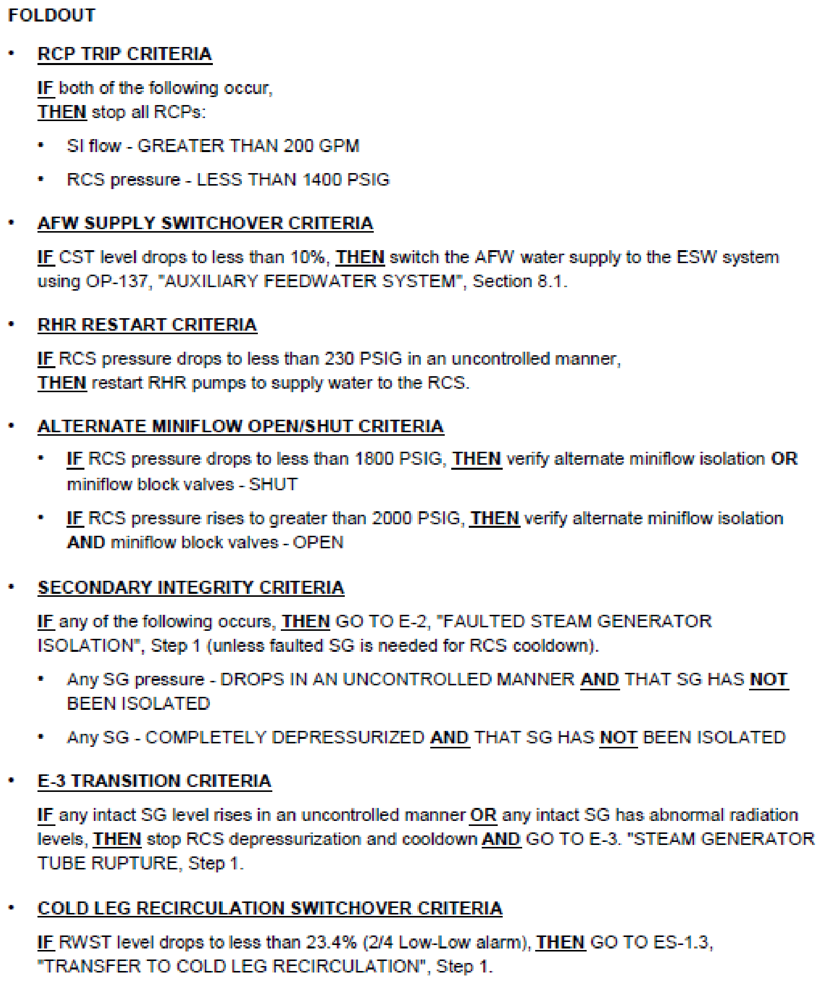
Op Test No.:	NRC	Scenario #	3	Event #	9	Page	43	of	66
Event Description:	**Small Break LOCA** **(EOP-E-1)**								

Time	Position	Applicant's Actions or Behavior	
	SRO	RNO: GO TO Step 30.	
	SRO	CNMT pressure – NORMAL GO TO E-1, "LOSS OF REACTOR OR SECONDARY COOLANT", Step 1	(NO)
	SRO	RNO: GO TO E-1, "LOSS OF REACTOR OR SECONDARY COOLANT", Step 1.	
	SRO	Transitions to EOP-E-0, Step 1	
EOP-E-1		Loss Of Reactor Or Secondary Coolant	
	SRO	Enters EOP-E-1 Holds crew update	
Procedure Note		Foldout applies.	
	SRO	Review Foldout page	

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	44	of 66
Event Description:	**Small Break LOCA (EOP-E-1)**							

Time	Position	Applicant's Actions or Behavior

Evaluator Note:



FOLDOUT

- **RCP TRIP CRITERIA**

  IF both of the following occur,
  THEN stop all RCPs:

  - SI flow - GREATER THAN 200 GPM
  - RCS pressure - LESS THAN 1400 PSIG

- **AFW SUPPLY SWITCHOVER CRITERIA**

  IF CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1.

- **RHR RESTART CRITERIA**

  IF RCS pressure drops to less than 230 PSIG in an uncontrolled manner,
  THEN restart RHR pumps to supply water to the RCS.

- **ALTERNATE MINIFLOW OPEN/SHUT CRITERIA**

  - **IF** RCS pressure drops to less than 1800 PSIG, **THEN** verify alternate miniflow isolation **OR** miniflow block valves - SHUT
  - **IF** RCS pressure rises to greater than 2000 PSIG, **THEN** verify alternate miniflow isolation **AND** miniflow block valves - OPEN

- **SECONDARY INTEGRITY CRITERIA**

  IF any of the following occurs, **THEN** GO TO E-2, "FAULTED STEAM GENERATOR ISOLATION", Step 1 (unless faulted SG is needed for RCS cooldown).

  - Any SG pressure - DROPS IN AN UNCONTROLLED MANNER **AND** THAT SG HAS **NOT** BEEN ISOLATED
  - Any SG - COMPLETELY DEPRESSURIZED **AND** THAT SG HAS **NOT** BEEN ISOLATED

- **E-3 TRANSITION CRITERIA**

  IF any intact SG level rises in an uncontrolled manner **OR** any intact SG has abnormal radiation levels, **THEN** stop RCS depressurization and cooldown **AND** GO TO E-3. "STEAM GENERATOR TUBE RUPTURE, Step 1.

- **COLD LEG RECIRCULATION SWITCHOVER CRITERIA**

  IF RWST level drops to less than 23.4% (2/4 Low-Low alarm), **THEN** GO TO ES-1.3, "TRANSFER TO COLD LEG RECIRCULATION", Step 1.

	SRO	Assigns Foldout items:
	SRO	Initiate Monitoring Of Critical Safety Function Status Trees.
Evaluator Note:		**The crew should review foldout criteria.** **The crew identify and use Adverse Values identified in brackets in the EOP procedures [ ] when Containment Pressure exceeds 3 PSIG**
	RO	Maintain RCP Seal Injection flow between 8 GPM AND 13 GPM.

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	45 of 66
Event Description:		**Small Break LOCA (EOP-E-1)**					

Time	Position	Applicant's Actions or Behavior	
	BOP	Check Intact SG Levels: • Any level – GREATER THAN 25% [40%]. **(Dependent on timing – same results)** • Control feed flow to maintain all intact levels between 25% AND 50% [40% AND 50%]. • Any level - RISING IN AN UNCONTROLLED MANNER.	(YES) (NO)
	SRO	RNO: GO TO Step 4.	
	RO	Check PZR PORV block valves: • Ensure AC buses 1A1 AND 1B1 – Energized • Check PORV's Shut • Check block valves - AT LEAST ONE OPEN.	(YES) (YES) (YES)
	RO	Check SI Termination Criteria: • RCS subcooling – GREATER THAN ○ 10°F [40°F] – C ○ 20°F [50°F] – M **(Dependent on timing – same results)**	(YES/ NO)
Evaluator Note:		**IF Subcooling > 10°F then the check is performed, otherwise the following is N/A**	
	BOP	• Check secondary heat sink by observing any of the following: ○ Level in at least one intact SG – >25% [40%] ○ Total feed flow to intact SGs – > 200 KPPH	(YES) (YES)
	RO	• RCS pressure – Stable Or Rising • PRZ level – GREATER THAN 10% [30%] **(Dependent on timing – same results)**	(YES/ NO)

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	46 of 66
Event Description:	**Small Break LOCA (EOP-E-1)**						

Time	Position	Applicant's Actions or Behavior	
	SRO	WHEN the SI termination criteria are met, THEN GO TO EOP-ES-1.1, "SI TERMINATION", Step 1. **(not met initially)**	
	RO	Check CNMT Spray Status: • Check any CNMT Spray Pump – RUNNING.	(NO)
	RO	Check Source Range Detector Status: • Intermediate range flux – LESS THAN 5x10^{-11} AMPS • Ensure source range detectors – ENERGIZED • Transfer nuclear recorder to source range scale	(YES)
	RO	Check RHR Pump status: • Check RHR pump suction – ALIGNED TO RWST • RCS Pressure greater than 230 PSIG o YES – Stop RHR pumps • RCS Pressure greater than 230 PSIG o NO – leave RHR pumps on. **(Dependent on timing)**	(YES) (YES) (NO)
Evaluator's Note:		**The evaluation/trend of RCS pressure in the next several steps is dependent on how long it took the crew to reach these steps (Decay Heat/Break Flow/ECCS flow). Pressure should be stable or lowering at this point.**	
		Check RCS And SG Pressures: • Check for both of the following: o All SG Pressures – STABLE OR RISING. o RCS pressure – STABLE OR DROPPING.	(YES) (YES)
Evaluator's Note:		**If the evaluation/trend of RCS pressure in the previous step was rising the SRO will return to EOP-E-1, Step 1 (Pg 43 of 67) and wait for the plant to stabilize.**	
		Establish CCW Flow To The RHR Heat Exchangers:	

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	47	of	66
Event Description:	**Small Break LOCA** **(EOP-E-1)**								

Time	Position	Applicant's Actions or Behavior	
	RO	• Ensure both CCW Pumps running • Open the following valves: (CCW Return From RHR HX) ○ 1CC-147 Train A ○ 1CC-167 Train B **(locates MCB switch and opens valves listed)**	(YES)
	RO	• Ensure CCW flow to the RHR Heat Exchangers	(YES)
	RO	• Perform one of the following to establish two independent CCW systems: ○ Shut train A CCW non-essential supply AND return valves: • 1CC-99 • 1CC-128 **OR** ○ Shut train B CCW non-essential supply AND return valves: • 1CC-113 • 1CC-127 **(locates MCB switch and shuts one Train of valves listed)**	
	BOP/RO	Check EDG status: • Check AC emergency buses 1A-SA AND 1B-SB – ENERGIZED BY OFFSITE POWER ○ Check Bus voltages (Normal) ○ Check breakers 105 and 125 closed	(YES)
	SRO	• GO TO Step 13.e.	
	BOP/RO	• Check any EDG – running unloaded	(YES)
	RO	• Reset SI **(takes both SI reset switches to RESET and observes status light change from SI active to SI reset)**	

Op Test No.: NRC	Scenario # 3	Event # 9	Page 48 of 66
Event Description:	**Small Break LOCA (EOP-E-1)**		

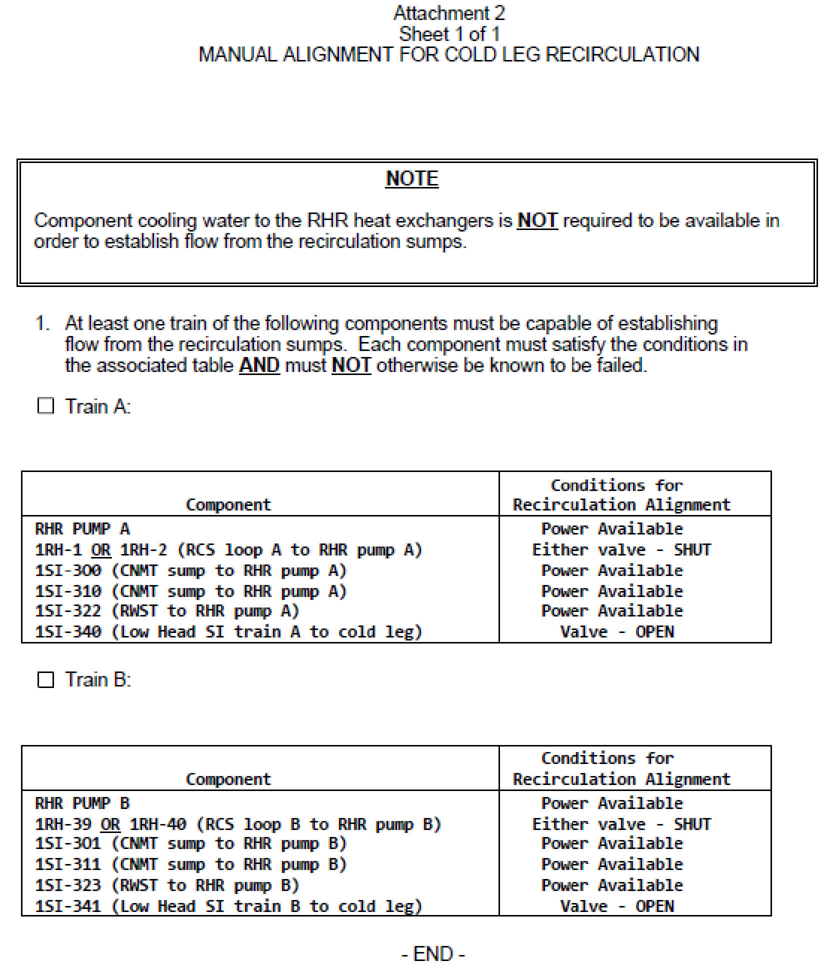
Time	Position	Applicant's Actions or Behavior	
	SRO	• Manually realign safeguards equipment following a loss of offsite power. (Refer to E-0, "REACTOR TRIP OR SAFETY INJECTION", Attachment 6.)	
	BOP	• Shutdown any unloaded EDGs using OP-155, "Diesel Generator Emergency Power System", Section 7.0	
	SRO	Initiate Evaluation of Plant Status: • RHR system – CAPABLE OF COLD LEG RECIRCULATION (Refer to Attachment 2) • Check Auxiliary AND Radwaste Processing Building radiation – NORMAL	 (YES) (YES)

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	49	of	66

Event Description: **Small Break LOCA (EOP-E-1)**

Time	Position	Applicant's Actions or Behavior

Evaluator Note:



Attachment 2
Sheet 1 of 1
MANUAL ALIGNMENT FOR COLD LEG RECIRCULATION

NOTE

Component cooling water to the RHR heat exchangers is **NOT** required to be available in order to establish flow from the recirculation sumps.

1. At least one train of the following components must be capable of establishing flow from the recirculation sumps. Each component must satisfy the conditions in the associated table **AND** must **NOT** otherwise be known to be failed.

☐ Train A:

Component	Conditions for Recirculation Alignment
RHR PUMP A	Power Available
1RH-1 OR 1RH-2 (RCS loop A to RHR pump A)	Either valve - SHUT
1SI-300 (CNMT sump to RHR pump A)	Power Available
1SI-310 (CNMT sump to RHR pump A)	Power Available
1SI-322 (RWST to RHR pump A)	Power Available
1SI-340 (Low Head SI train A to cold leg)	Valve - OPEN

☐ Train B:

Component	Conditions for Recirculation Alignment
RHR PUMP B	Power Available
1RH-39 OR 1RH-40 (RCS loop B to RHR pump B)	Either valve - SHUT
1SI-301 (CNMT sump to RHR pump B)	Power Available
1SI-311 (CNMT sump to RHR pump B)	Power Available
1SI-323 (RWST to RHR pump B)	Power Available
1SI-341 (Low Head SI train B to cold leg)	Valve - OPEN

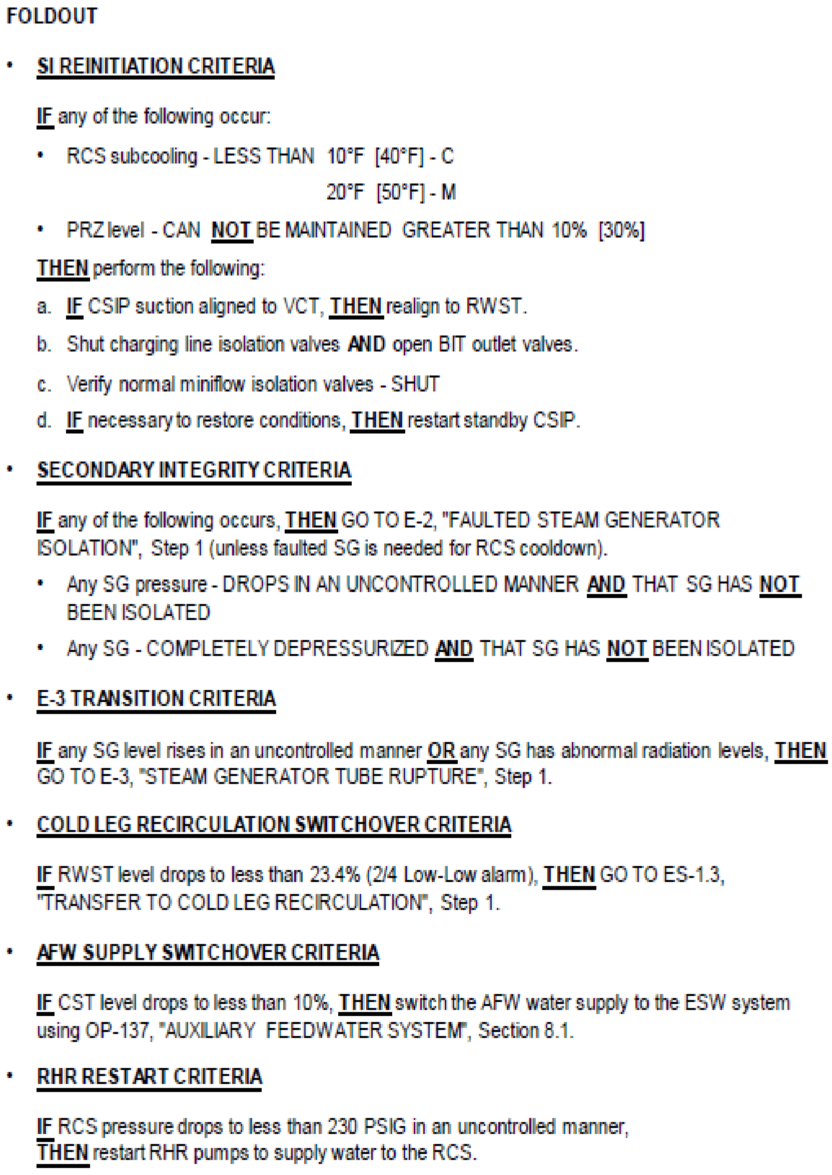
- END -

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	50	of	66

Event Description: **Small Break LOCA (EOP-ES-1.2)**

Time	Position	Applicant's Actions or Behavior	
	SRO	- GO TO Step 15.	
	RO	Check RCS Status - Check for both of the following: o RCS pressure – LESS THAN 230 PSIG o Any RHR HX header flow - GREATER THAN 1000 GPM	 (NO) (NO)
	SRO	RNO:GO TO **Go to** ES-1.2, "POST LOCA COOLDOWN AND DEPRESSURIZATION", Step 1	
	SRO	Transitions to EOP-ES-1.2, Step 1	
Evaluator Note:		**At some point during the implementation of EOP-ES-1.2 the break will clear and the Safety Injection flow filling the RCS with cold RWST water will cause a reduction in RCS pressure and temperature.** **The critical safety function status tree for RCS integrity will begin to toggle from Green to Yellow to Orange to Red. Eventually RCS Integrity will remain RED and the crew will transition to EOP-FR-P.1.** **The scenario guide is written for the transition that occurred based on the plant response during validation. When this transition occurs will vary based on the pace of implantation by the crew.**	
EOP-ES-1.2		Post LOCA Cooldown and Depressurization	
	SRO	Enters EOP-ES-1.2 Holds crew update	
Procedure Note		Foldout applies.	
	SRO	Review Foldout page	

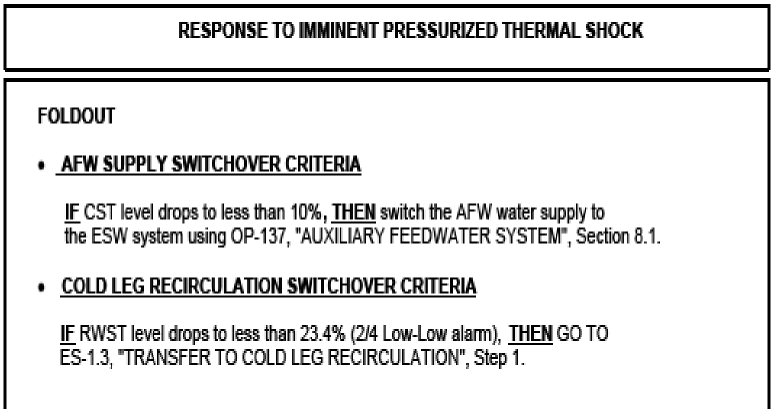
Op Test No.:	NRC	Scenario #	3	Event #	9	Page	51	of	66
Event Description:	**Small Break LOCA (EOP-ES-1.2)**								

Time	Position	Applicant's Actions or Behavior
Evaluator Note:		 **FOLDOUT** • **SI REINITIATION CRITERIA** **IF** any of the following occur: • RCS subcooling - LESS THAN 10°F [40°F] - C 20°F [50°F] - M • PRZ level - CAN **NOT** BE MAINTAINED GREATER THAN 10% [30%] **THEN** perform the following: a. **IF** CSIP suction aligned to VCT, **THEN** realign to RWST. b. Shut charging line isolation valves **AND** open BIT outlet valves. c. Verify normal miniflow isolation valves - SHUT d. **IF** necessary to restore conditions, **THEN** restart standby CSIP. • **SECONDARY INTEGRITY CRITERIA** **IF** any of the following occurs, **THEN** GO TO E-2, "FAULTED STEAM GENERATOR ISOLATION", Step 1 (unless faulted SG is needed for RCS cooldown). • Any SG pressure - DROPS IN AN UNCONTROLLED MANNER **AND** THAT SG HAS **NOT** BEEN ISOLATED • Any SG - COMPLETELY DEPRESSURIZED **AND** THAT SG HAS **NOT** BEEN ISOLATED • **E-3 TRANSITION CRITERIA** **IF** any SG level rises in an uncontrolled manner **OR** any SG has abnormal radiation levels, **THEN** GO TO E-3, "STEAM GENERATOR TUBE RUPTURE", Step 1. • **COLD LEG RECIRCULATION SWITCHOVER CRITERIA** **IF** RWST level drops to less than 23.4% (2/4 Low-Low alarm), **THEN** GO TO ES-1.3, "TRANSFER TO COLD LEG RECIRCULATION", Step 1. • **AFW SUPPLY SWITCHOVER CRITERIA** **IF** CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1. • **RHR RESTART CRITERIA** **IF** RCS pressure drops to less than 230 PSIG in an uncontrolled manner, **THEN** restart RHR pumps to supply water to the RCS.
	SRO	Assigns Foldout items:
	RO	Reset SI (already performed)
	SRO	Manually realign safeguards equipment following a loss of offsite power.

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	52	of 66
Event Description:	**Small Break LOCA (EOP-ES-1.2)**							

Time	Position	Applicant's Actions or Behavior	
		(Refer to E-0, "REACTOR TRIP OR SAFETY INJECTION", Attachment 6.)	
	RO	Reset Phase A and Phase B Isolation signals • Reset Phase A (if actuated) **(locates MCB Phase A switch and resets Phase A)** • Reset Phase B (if actuated) **(Phase B not actuated)**	
	RO	Open Instrument Air and Nitrogen valves to CNMT • 1IA-819 • 1SI-287 **(locates MCB switches and opens valve)**	
	BOP/RO	Check EDG status: • Check AC emergency buses 1A-SA AND 1B-SB – ENERGIZED BY OFFSITE POWER ○ Check Bus voltages (Normal) ○ Check breakers 105 and 125 closed	(YES)
	SRO	• GO TO Step 5.e.	
Evaluator Note:		**The scenario guide is written for the transition to EOP-FR-P.1 at this point in the scenario based on the plant response during validation. When this transition occurs will vary based on the pace of implantation by the crew.**	
EOP-FR-P.1		Response to Imminent Pressurized Thermal Shock	
	SRO	Enters EOP-FR-P.1 Holds crew update	
Procedure Note		Foldout applies.	

Op Test No.: NRC	Scenario # 3	Event # 9	Page 53 of 66		
Event Description:	**Small Break LOCA** **(EOP-FR-P.1)**				
Time	Position	Applicant's Actions or Behavior			

	SRO	Review Foldout page	
Evaluator Note:		 **RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK** **FOLDOUT** • **AFW SUPPLY SWITCHOVER CRITERIA** **IF** CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1. • **COLD LEG RECIRCULATION SWITCHOVER CRITERIA** **IF** RWST level drops to less than 23.4% (2/4 Low-Low alarm), **THEN** GO TO ES-1.3, "TRANSFER TO COLD LEG RECIRCULATION", Step 1.	
	RO	Check RCS Pressure: Check for both of the following: • RCS pressure - LESS THAN 230 PSIG • Any RHR HX header flow - > 1000 GPM RO restarts RHR pumps when RCS pressure < RHR shutoff head – EOP-ES-1.2 foldout action item	(NO) (NO)
	SRO	• GO TO Step 2.	
	RO	Check RCS Cold Leg Temperature Trend: • Check RCS Cold Leg Temperatures - STABLE OR RISING	(NO)
	SRO	• GO TO Step 3.	
Procedure Note:		A faulted SG is any SG that is depressurizing in an uncontrolled manner or is completely depressurized.	

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	54	of 66

Event Description: **Small Break LOCA (EOP-FR-P.1)**

Time	Position	Applicant's Actions or Behavior	
	BOP	Stop RCS Cooldown: • Ensure SG PORVs – SHUT • Ensure condenser steam dump valves – SHUT • Check RHR system – IN SHUTDOWN COOLING MODE • Any non-faulted SG level - > 25% [40%] • Control feed flow to non-faulted SG(s) to stop RCS cooldown.	(YES) (YES) (NO) (YES)
Procedure Caution:		IF the TDAFW pump is the only available source of feed flow, THEN maintain steam supply to the TDAFW pump from one SG.	
	BOP	Minimize RCS Cooldown From Faulted SG(s): • Check any SG – FAULTED	(NO)
	SRO	• GO TO Step 5.	
	RO	Check PRZ PORV Block Valves: • Ensure power to block valves – AVAILABLE • Check block valves - AT LEAST ONE OPEN	(YES) (YES)
Procedure Note:		IF PRZ PORV opens on high pressure, Step 6 should be repeated after pressure drops to less than PORV setpoint.	
	RO	Check PRZ PORVs: • Check all of the following: ○ Check LTOPS control switches - IN NORMAL (NOT BLOCKED)	(NO)
	SRO	• GO TO Step 6.d.	
	RO	• Check PRZ pressure - < 2335 psig (YES) • Verify PRZ PORVs – SHUT (YES)	(YES) (YES)
	RO	Check SI Flow - > 200 gpm	(YES)

Op Test No.: NRC	Scenario # 3	Event # 9	Page 55 of 66	
Event Description:	**Small Break LOCA (EOP-FR-P.1)**			
Time	Position	Applicant's Actions or Behavior		

	SRO	Check SI Termination Criteria: • Check for both of the following: ○ RCS subcooling - > 60°F [90°F] – C	(NO)
		• GO TO Step 9.	
Procedure Caution:		Following a complete loss of normal seal cooling, the affected RCP(s) should NOT be started prior to a status evaluation performed by the Plant Operations Staff or responsible engineer.	
	SRO	Check If An RCP Should Be Started: • RCS subcooling - GREATER THAN 10°F [40°F] – C	(NO)
		• GO TO Step 33.	
Procedure Caution:		Following an excessive cooldown, reactor vessel stress must be relieved to enhance and maintain vessel integrity. Do NOT perform any actions that raise pressure OR cause an RCS cooldown until the soak is complete.	
Procedure Note:		Even if a soak period is required, steam may be released from intact SGs with pressure higher than the saturation pressure for lowest cold leg temperature.	
	SRO	Determine RCS Soak Requirements: • RCS cooldown rate - > 100°F in any 60 min period • Perform one hour RCS soak: ○ Maintain RCS temperature stable. ○ Maintain RCS pressure stable. ○ Perform actions of other procedures that do NOT cause an RCS cooldown OR raise pressure.	(YES)

Op Test No.: NRC Scenario # 3 Event # 9 Page 56 of 66

Event Description: **Small Break LOCA (EOP-ES-1.2)**

Time	Position	Applicant's Actions or Behavior	
	SRO	Establish Subsequent Cooldown: • RCS subcooling monitor - AVAILABLE • Maintain RCS subcooling between 10°F and 190°F [40°F and 160°F]. • Maintain RCS cooldown rate less than 50°F in any sixty minute period.	(YES)
	SRO	Return to Procedure And Step In Effect.	
EOP-ES-1.2			
	SRO	Returns to EOP-ES-1.2 Step 5.e Holds crew update	
	SRO	• GO TO Step 5.e.	
	BOP/RO	• Check all non-emergency AC buses – ENERGIZED	(YES)
Procedure Caution:		PRZ heaters should NOT be energized until PRZ water level indicates greater than minimum recommended by plant operations staff to ensure heaters are covered.	
	RO	Secure PRZ Heaters: • Place backup heaters in the OFF position • Ensure control heaters – OFF	
	SRO	• Consult plant operations staff for a recommended minimum indicated PRZ water level that will ensure heaters are covered. (Refer to ERG Executive Volume, Generic Issue: Evaluations by the Plant Engineering Staff.)	

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	57	of	66

Event Description: **Small Break LOCA (EOP-ES-1.2)**

Time	Position	Applicant's Actions or Behavior	
	RO	Check if RHR Pumps should be stopped: • Check RHR pump suction – ANY RUNNING WITH SUCTION ALIGNED TO RWST • RCS Pressure greater than 230 PSIG • RCS Pressure STABLE OR RISING ○ YES – Stop RHR pumps • RCS Pressure STABLE OR RISING ○ NO – leave RHR pumps on. **(Dependent on timing)**	(YES) (YES) (YES) (NO)
	BOP	Check Intact SG Levels: • Any level - GREATER THAN 25% [40%] • Control feed flow to maintain all intact levels between 25% and 50% [40% and 50%].	(YES)
Procedure Note:		After the low steam pressure SI signal is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.	
	RO	Check PRZ Pressure: • Pressure – less than 2000 PSIG • Block low steam pressure SI **(Locates Low Steam Line Pressure SI MCB block switch and places switch to block – verifies block on status lights)**	(YES)
Procedure Note:		Even if the lowest RCS cold leg temperature has dropped by 100°F in the last 60 minutes, steam may be released from intact SGs with pressure higher than the saturation pressure for lowest cold leg temperature.	
	SRO	Initiate RCS Cooldown To Cold Shutdown: • Maintain Cooldown rate in RCS cold legs - <100˚F/HR **(SRO should maintain requirements of EOP-FR-P.1 <50˚F/HR until the 1 hour soak is complete.)**	

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	58	of 66
Event Description:	**Small Break LOCA (EOP-ES-1.2)**							
Time	Position	Applicant's Actions or Behavior						

	RO	• Check RHR system - OPERATING IN SHUTDOWN COOLING MODE	(NO)
	SRO	GO TO Step 10.f	
	BOP	• Check all of the following to determine if steam can be dumped to condenser: ○ Check any intact SG MSIV – OPEN ○ Condenser Available (C-9)- LIT (BPLB 3-3) ○ Steam Dump Control – AVAILABLE	(NO) (NO) (YES)
Evaluator Note:		**The may recouple RCS with SG's which will require SG PORV's to be opened and SG pressure reduced. The SRO should maintain requirements of EOP-FR-P.1 until the 1 hour soak is complete.** **o Maintain RCS temperature stable.** **o Maintain RCS pressure stable.** **o Perform actions of other procedures that do NOT cause an RCS cooldown OR raise pressure.**	
	BOP	RNO: Dump steam from intact SGs using any of the following (listed in order of preference): ○ SG PORVs ○ Locally operate SG PORVs ○ TDAFW pump	

Lead Evaluator:	**Terminate the scenario when the crew discusses their plan for cooldown of the RCS.** **Announce 'Crew Update' - End of Evaluation - I have the shift.** **Have crew remain in the Simulator without discussing the exam. Examiners will formulate any follow-up questions.**

Simulator Operator:	**When directed by Lead Evaluator go to FREEZE**

REACTOR TRIP OR SAFETY INJECTION

Attachment 1
Sheet 1 of 1
SI EMERGENCY ALIGNMENT

- Charging line isolation valves - SHUT:

  1CS-235
  1CS-238

- CSIP suction from RWST valves - OPEN:

  1CS-291 (LCV-115B)
  1CS-292 (LCV-115D)

- VCT outlet valves - SHUT:

  1CS-165 (LCV-115C)
  1CS-166 (LCV-115E)

- BIT outlet valves - OPEN:

  1SI-3
  1SI-4

- CSIP alternate miniflow isolation valves - SHUT (IF RCS PRESSURE LESS THAN 1800 PSIG) OR OPEN (IF RCS PRESSURE GREATER THAN 2000 PSIG):

  1CS-746
  1CS-752

- CSIP alternate miniflow block valves - OPEN (UNLESS SHUT TO ISOLATE AN ALTERNATE MINIFLOW ISOLATION VALVE)

  1CS-745
  1CS-753

- CSIP normal miniflow valves - SHUT:

  1CS-214
  1CS-182
  1CS-196
  1CS-210

- Low head SI to cold leg valves - OPEN:

  1SI-340
  1SI-341

- Low head SI to hot leg crossover valves - OPEN:

  1SI-326
  1SI-327

- Low head SI to hot leg valve - SHUT:

  1SI-359

- RWST to RHR pump suction valves - OPEN:

  1SI-322
  1SI-323

- END -

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 1 of 7
SAFEGUARDS ACTUATION VERIFICATION

NOTE

- General guidance for verification of safeguards equipment is contained in Attachment 4 of this procedure.
- ERFIS displays of safeguards equipment status are not reliable while any associated safety-related electrical buses are de-energized.

☐ 1. **Ensure** Two CSIPs - RUNNING

☐ 2. **Ensure** Two RHR Pumps - RUNNING

☐ 3. **Ensure** Two CCW Pumps - RUNNING

☐ 4. **Ensure** All ESW **AND** ESW Booster Pumps - RUNNING

☐ 5. **Ensure** SI Valves - PROPERLY ALIGNED

(**Refer** to Attachment 1.)

☐ 6. **Ensure** CNMT Phase A Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 4.)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 2 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 7. **Ensure** SG Blowdown **AND** SG Sample Isolation Valves In Table 1 - SHUT

Table 1: SG Blowdown And Sample Isolation Valves

Process Line	Outside CNMT (MLB-1A-SA)	Inside CNMT (MLB-1B-SB)
SG A Sample	1SP-217	1SP-214/216
SG B Sample	1SP-222	1SP-219/221
SG C Sample	1SP-227	1SP-224/226
SG A Blowdown	1BD-11	1BD-1
SG B Blowdown	1BD-30	1BD-20
SG C Blowdown	1BD-49	1BD-39

8. IF Main Steam Line Isolation Actuated OR Is Required By Any Of The Following, THEN **Ensure** MSIVs **AND** MSIV Bypass Valves - SHUT

☐ • Steam line pressure - LESS THAN 601 PSIG

☐ • CNMT pressure - GREATER THAN 3.0 PSIG

9. IF CNMT Spray Actuation Signal Actuated OR Is Required, THEN **Ensure** The Following:

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 9.)

☐ • CNMT spray pumps - RUNNING

☐ • CNMT spray valves - PROPERLY ALIGNED

☐ • Phase B isolation valves - SHUT

☐ • All RCPs - STOPPED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 3 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 10. **Ensure** Both Main FW Pumps - TRIPPED

☐ 11. **Ensure** FW Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 6.)

☐ 12. **Ensure** Both MDAFW pumps - RUNNING

13. **IF** Any Of The Following Conditions Exist, **THEN** **Ensure** The TDAFW Pump - RUNNING

- ☐ Undervoltage on either 6.9 KV emergency bus
- ☐ Level in two SGs - LESS THAN 25%
- ☐ Manual actuation to control SG level

14. **Ensure** AFW Valves - PROPERLY ALIGNED

- ☐ **IF** no AFW Isolation Signal, **THEN** **ensure** isolation **AND** flow control valves - OPEN

NOTE

An AFW Isolation signal signal requires a Main Steam Line Isolation coincident with one SG pressure 100 PSIG below the other two SGs.

- ☐ **IF** AFW Isolation Signal present, **THEN** **ensure** MDAFW **AND** TDAFW isolation **AND** flow control valves to affected SG - SHUT

☐ 15. **Ensure** Both EDGs - RUNNING

☐ 16. **Ensure** CNMT Fan Coolers - ONE FAN PER UNIT RUNNING IN SLOW SPEED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 4 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 17. **Ensure** CNMT Ventilation Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 7.)

☐ 18. **Ensure** Control Room Area Ventilation - MAIN CONTROL ROOM ALIGNED FOR EMERGENCY OPERATION

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 5, Sheets 1 and 2, Sections for MAIN CONTROL BOARD, SLB-5 and SLB-6.)

19. **Ensure** Essential Service Chilled Water System Operation:

- ☐ • **Ensure** both WC-2 chillers - RUNNING
- ☐ • **Ensure** both P-4 pumps - RUNNING
- ☐ (**Refer** to AOP-026, "LOSS OF ESSENTIAL SERVICE CHILLED WATER SYSTEM" for loss of any WC-2 chiller.)

20. **Ensure** CSIP Fan Coolers - RUNNING

☐ AH-9 A SA
☐ AH-9 B SB
☐ AH-10 A SA
☐ AH-10 B SB

NOTE

Security systems are powered by bus 1A1 (normal supply) or bus 1B1 (alternate supply). Backup power will be available for approximately 30 MINUTES after the supplying bus is de-energized. (Refer to OP-115, "CENTRAL ALARM STATION ELECTRICAL SYSTEMS", Section 8.9 and 8.10.)

☐ 21. **Ensure** AC buses 1A1 **AND** 1B1 - ENERGIZED

☐ 22. **Place** Air Compressor 1A **AND** 1B In The LOCAL CONTROL Mode.

(**Refer to** Attachment 7.)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 5 of 7
SAFEGUARDS ACTUATION VERIFICATION

CAUTION

The maximum calculated dose rate in the vicinity of MCC 1A35-SA and MCC 1B35-SB is between 10 MREM/HR and 150 MREM/HR.

☐ 23. Dispatch An Operator To **Unlock** And **Close** The Breakers For The CSIP Suction **AND** Discharge Cross-Connect Valves:

(**Refer** to Attachment 2.)

MCC 1A35-SA		MCC 1B35-SB	
VALVE	CUBICLE	VALVE	CUBICLE
1CS-170	4A	1CS-171	4D
1CS-169	4B	1CS-168	7D
1CS-218	14D	1CS-220	9D
1CS-219	14E	1CS-217	12C

24. **Check** If C CSIP Should Be Placed In Service:

☐ • **IF** two charging pumps can **NOT** be verified to be running, **AND** C CSIP is available, **THEN place** C CSIP in service in place of the non-running CSIP using OP-107, "CHEMICAL AND VOLUME CONTROL SYSTEM, Section 8.5 or 8.7.

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 6 of 7
SAFEGUARDS ACTUATION VERIFICATION

25. **Start** The Spent Fuel Pump Room Ventilation System:

    a. At AEP-1, **ensure** the following ESCWS isolation valves - OPEN

    1) SLB-11 (Train A)

    ☐ • AH-17 SUP CH 100 (Window 9-1)

    ☐ • AH-17 RTN CH 105 (Window 10-1)

    2) SLB-9 (Train B)

    ☐ • AH-17 SUP CH 171 (Window 9-1)

    ☐ • AH-17 RTN CH 182 (Window 10-1)

    b. At AEP-1, **start** one SFP PUMP ROOM FAN COOLER:

    ☐ • AH-17 1-4A SA

    ☐ • AH-17 1-4B SB

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 7 of 7
SAFEGUARDS ACTUATION VERIFICATION

NOTE

- Fuel pool levels and temperatures should be monitored approximately every 1 to 2 HOURS.
- Following the initial check of fuel pool levels and temperature, monitoring responsibilities may be assumed by the plant operations staff (including the TSC or STA).
- Only fuel pools containing fuel are required to be monitored.

26. **Check** Status Of Fuel Pools:

☐ a. **Operate** spent fuel cooling pumps to maintain fuel pool temperatures between 85°F to 105°F.

b. **Monitor** fuel pool levels **AND** temperatures:

☐ • **Refer to** AOP-041, "SPENT FUEL POOL EVENT" Attachments 7, 8, 9, 10 and 11 for SFP parameter monitoring methods.

☐ • **Refer to** Curves H-X-24, H-X-25 and H-X-26 for SFP time to 200°F.

☐ • Levels - GREATER THAN LO ALARM (284 FT, 0 IN)

☐ • Temperatures - LESS THAN HI TEMP ALARM (105°F)

NOTE

If control room ventilation was previously aligned to an emergency outside air intake for post-accident operations, then follow-up actions will be required to restore the alignment.

27. **Consult** Plant Operations Staff Regarding Alignment Of The Control Room Ventilation System:

☐ • Site Emergency Coordinator - Control Room

☐ • Site Emergency Coordinator - Technical Support Center

(Refer to PEP-230, "CONTROL ROOM OPERATIONS".)

- END -

HARRIS 2020 NRC SCENARIO 4

Facility: Harris Nuclear Plant Scenario No.: 4 Op Test No.: 05000400/2020301

Examiners: ______________________ Operators: SRO: ______________________

______________________ RO: ______________________

______________________ BOP: ______________________

Initial Conditions: IC-19 MOL, 100% power

- 1CS-9, Letdown Isolation Valve is under clearance for solenoid replacement
- 'B' MDAFW Pump is under clearance for pump packing repairs
- 'B' DEH Pump Out of Service

Turnover:	The plant is at 100% power, middle of core life. GP-006 step 4
Critical Task:	• Manually maintain control of SG 'A' level above 25% to prevent an automatic Reactor trip after steam pressure transmitter PT-475 fails low • Manually maintain control of PRZ Pressure above 1960 psig to prevent an automatic Reactor trip after the pressure transmitter PT-444 fails high • Initiate RCS Bleed and Feed for Successful High-Head SI Pump Injection prevent RVLIS Full Range Level from lowering below 39%.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R – RO/SRO N – BOP/SRO	Power reduction from 100% power
2	sws07a	C – RO/SRO TS – SRO	Normal Service Water Pump 'A' sheared shaft (AOP-022)
3	prs06a	I – RO/SRO TS – SRO	Pressurizer PORV 445A Leakage (AOP-016)
4	gen15	C – BOP/SRO	Generator Voltage Regulator Failure
5	pt:475	I – BOP/SRO TS – SRO	Feed pressure transmitter failure low on 'A' SG PT-475 (AOP-010)
6	pt:444	C – RO/SRO TS – SRO	PT-444 Fails HIGH (AOP-019)
7	eps01 cfw01c cfw20a	M – ALL	Loss of Offsite Power with a Feed line break inside CNMT (EOP-E-0)
8	mss05a mss05b mss05c dsg04b	C – BOP/SRO	Main Steamline Isolation fails, 'B' CCW pump fails to Auto start
9	cfw01a	M – ALL	'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1)
10	prs03e	C – RO/SRO	Pressurizer PORV 445B fails to open

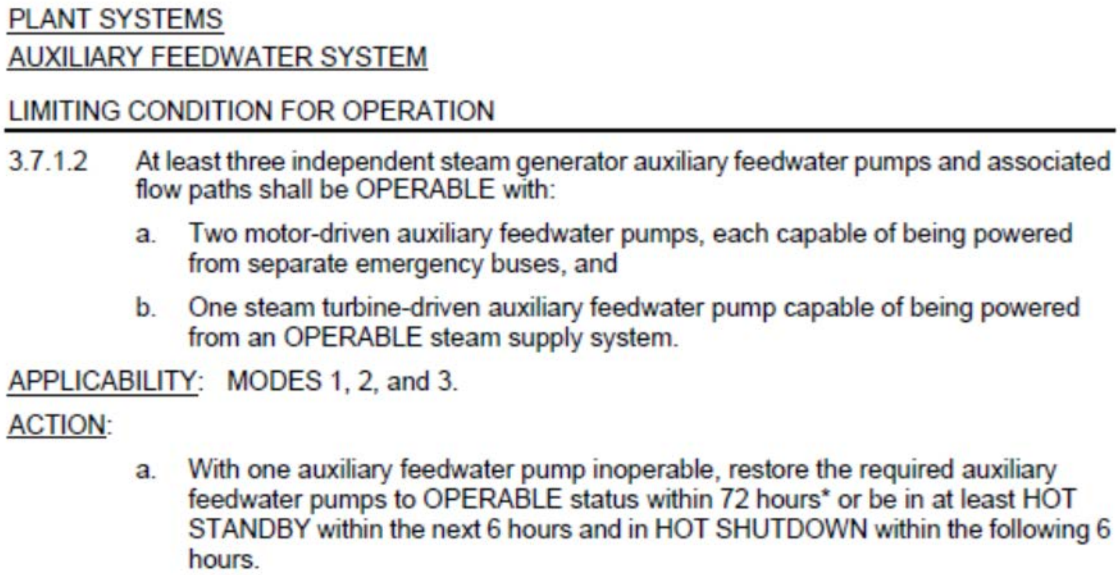
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 4

The plant is at 100% power, middle of core life. Due to the ‘B’ MDAFW pump LCO expiring, a normal shutdown in accordance with GP-006, Normal Plant Shutdown From Power Operation To Hot Standby (Mode 1 To Mode 3) is in progress as directed by plant management. Continue the shutdown @ 4 MW/min with TCS Load Control at 4 GVPC units/ min

The following equipment is under clearance:

- MDAFW Pump B-SB is under clearance for pump packing repairs. The pump has been inoperable for 66 hours and cannot be restored to operable status. Tech Spec 3.7.1.2 Action **a** and Tech Spec 3.3.3.5.b Action **c** applies.



PLANT SYSTEMS
AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

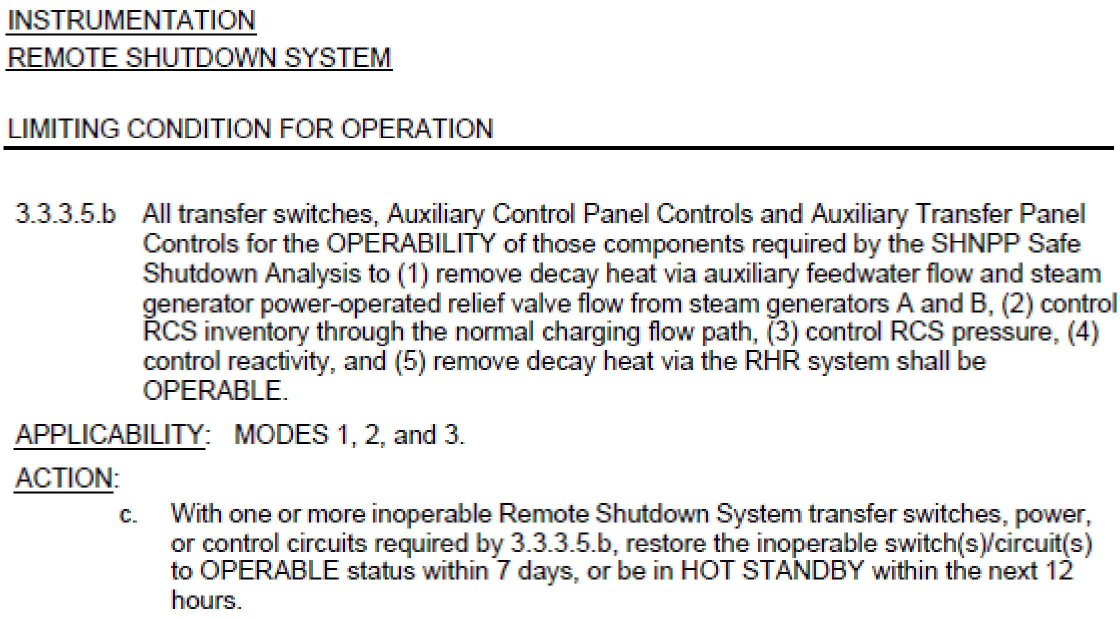
a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency buses, and

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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



INSTRUMENTATION
REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

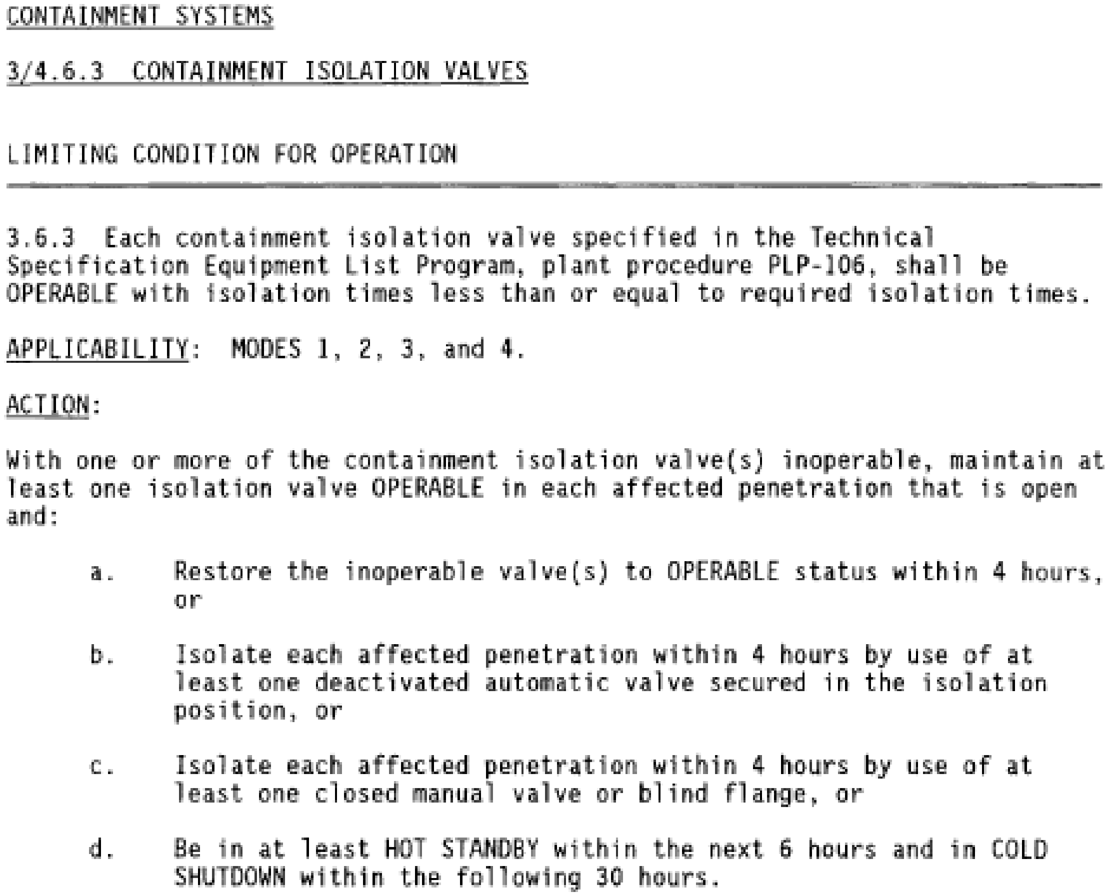
c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 4 continued

The following equipment is under clearance (continued):

- ‘B’ DEH Pump is under clearance for motor repairs. The pump has been unavailable for 8 hours. Repairs are expected to be completed within 24 hours.

- 1CS-9, Letdown Orifice Isolation valve, has been under clearance the last 12 hours for solenoid replacement. The repairs are close to completion and the valve is expected to be returned to service within the next hour. The valve is currently shut with power removed. OWP-CS-09 has been completed. Tech Specs 3.6.3 Action **b** applies.



CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

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

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

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 4 (Continued)

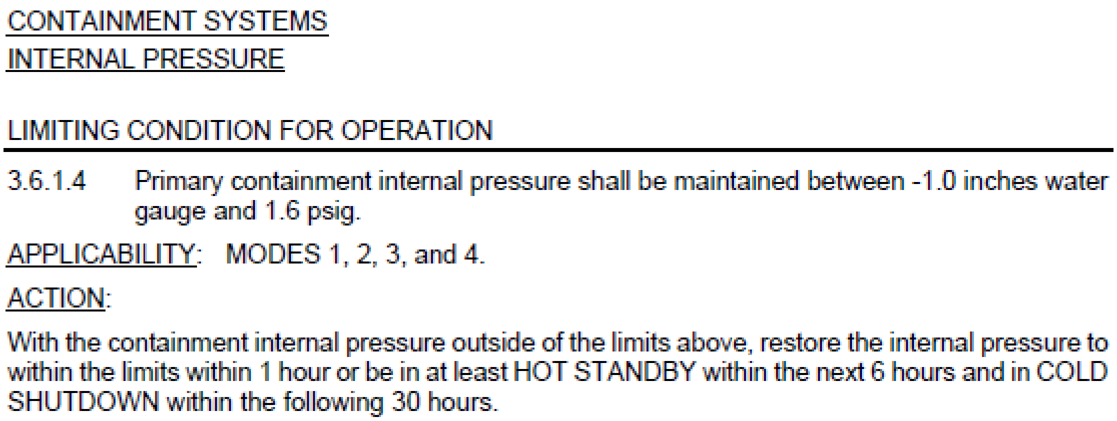
Event 1: Plant Shutdown (GP-006). Turnover takes place with the unit at 100% Reactor power. The crew will be given credit for a reactivity manipulation during the down power.

Verifiable Action: It is expected that the SRO will conduct a reactivity brief, the RO will borate and monitor auto rod insertion per the reactivity plan. The BOP will operate the DEH Turbine controls as necessary to lower power. After power is reduced 3% - 5% and the crew has demonstrated that they have control of the plant during a shutdown Event 2 will be inserted.

Event 2: Normal Service Water Pump ‘A’ sheared shaft (AOP-022). This failure will result in multiple NSW alarms on ALB 002 and the crew should enter AOP-022. While NSW system pressure is low the ESW system will automatically start and isolate into the ‘A’ and ‘B’ train headers. With lower temperature ESW water providing cooling into Containment the potential exists for a low pressure condition to occur. This will be indicated by ALB 028-5-1, Containment Air High Vacuum.

Verifiable Action: The crew will enter AOP-022 and carry out the immediate actions. The RO will perform the immediate actions of AOP-022 by verifying that the ESW pump automatically starts and the running CSIP does not operate greater than 1 minute without cooling water. The BOP will verbalize that no EDG is running to complete the immediate actions. Once the immediate actions are complete the BOP should place the Turbine in Hold it stabilize the plant and the crew should use the AOP to start up the standby NSW pump and verify proper system operation.

The SRO should evaluate Tech Spec 3.6.1.4, Containment Systems – Internal Pressure Action.



CONTAINMENT SYSTEMS
INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -1.0 inches water gauge and 1.6 psig.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

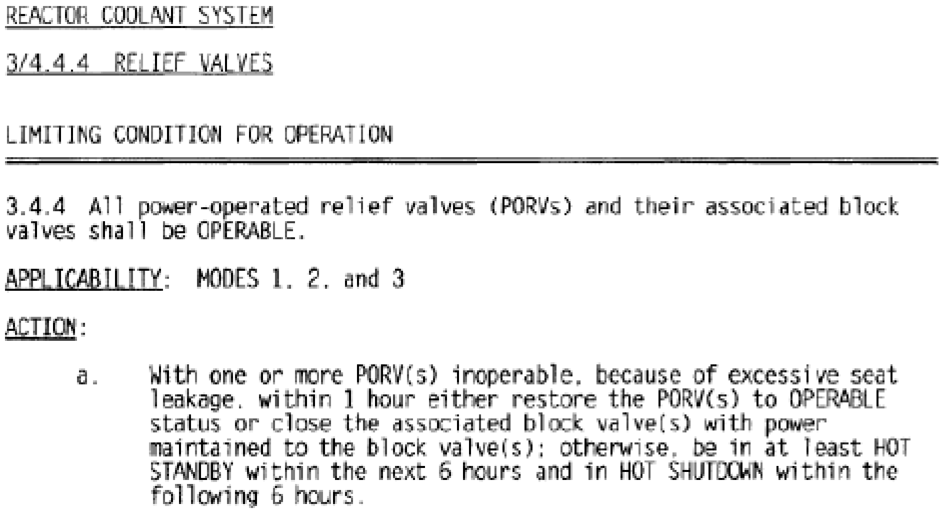
The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 3: Pressurizer PORV 445A Leakage (AOP-016). This failure will cause PRZ PORV 445A to leak, resulting in rising PRT pressure and level. PORV Line Temp indicator TI-463 will rise as observed on the MCB and the crew will respond in accordance with ALB 009-8-2, Pressurizer Relief Discharge High Temp. The crew may utilize AOP-016, Excessive Primary Plant Leakage, Attachment 5 to determine which PORV is leaking.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 4 (Continued)

Verifiable Action: The crew may respond in accordance with the alarm response procedure APP-ALB-009 or by entering AOP-016, which has NO immediate actions. The RO will place the block valve (1RC-117) for the affected PRZ PORV (1RC-118) in the shut position and monitor the PRT parameters to confirm isolation of the PORV.

The SRO should evaluate Tech Spec 3.4.4, Reactor Coolant System – Relief Valves Action **a**.



REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1. 2. and 3

ACTION:

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

The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 4: Generator Voltage Regulator Failure. This will cause the automatic function of the voltage regulator to oscillate which will be indicated on the ERFIS computer Quick Plot “VAC” and the MCB indication EI-567, Megavars. As the amplitude of the oscillations grows if the crew continues to operate the AVR in Auto the system will reject to manual after 10 minutes.

Verifiable Action: Event 4: Generator Voltage Regulator Failure. This will cause the automatic function of the voltage regulator to oscillate which will be indicated on the ERFIS computer Quick Plot “VAC” and MCB indication EI-565 and EI-567, Generator Megawatts and Megavars respectively. As the amplitude of the oscillations grows ALB 022-9-4, Computer Alarm Gen/Exciter Systems and 4-3, Gen Volt/Freq Ratio Limiter Active Or Under-Freq, alarm requiring the BOP to take manual control of the AVR in order to restore control of Generator Megavars.

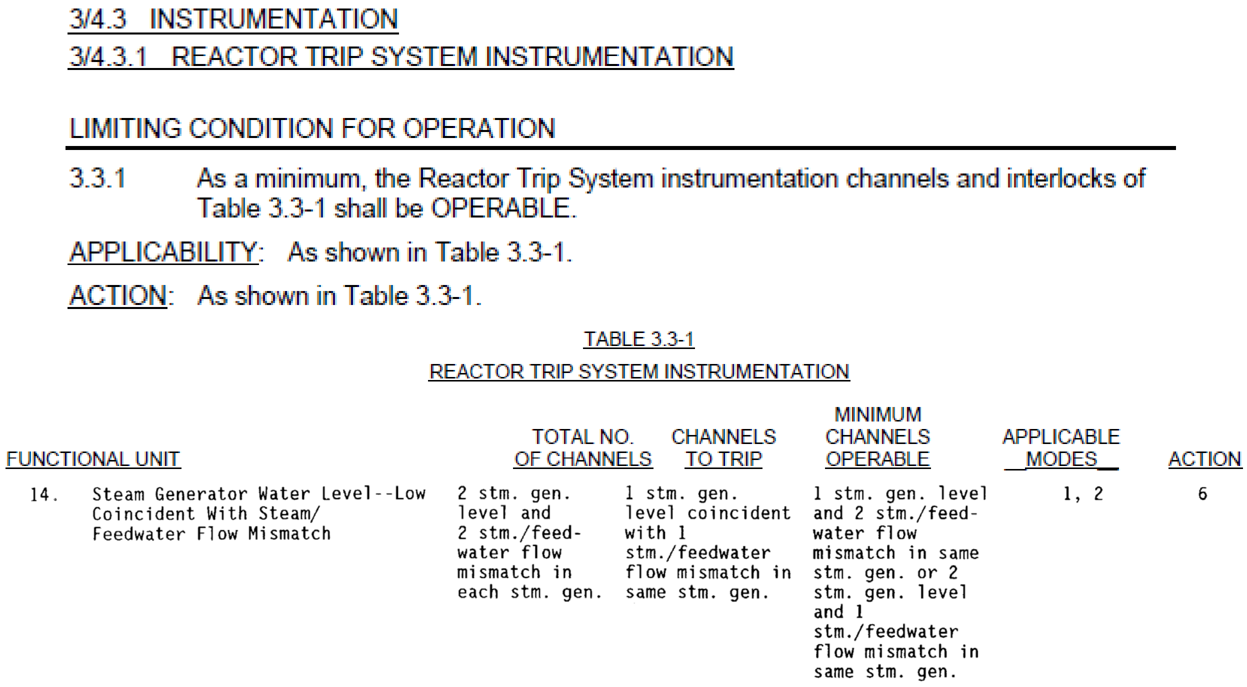
The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 5: Feed pressure transmitter failure low on ‘A’ SG FT-475 (AOP-010). This failure will cause ALB 014-1-2, 1-4, 4-1B, 4-2A, Loop A Hi Steam Line ΔP Low-P1, Loop A Hi Steam Line Press Rate Alert, SG A Stm > FW Flow Mismatch, and Loop A Low Stm Line Press Alert respectively to alarm. The crew will respond by entering AOP-010, Feedwater Malfunction and taking manual control of ‘A’ Main Feedwater Regulating Valve to raise Feedwater flow and stabilize level.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 4 (Continued)

Verifiable Action: The BOP will respond to the failure by taking the immediate actions of AOP-010 by manually controlling the ‘A’ FRV and restoring level 57%. With the controller in manual and the plant stabilized the crew will implement OWP-ESF-02 to remove the failed channel from service (**Critical Task #1**).

The SRO should evaluate Tech Spec 3.3.1, Instrumentation – Reactor Trip System Instrumentation, Tech Spec 3.3.2, Instrumentation – Engineered Safety Features Actuation System Instrumentation and Tech Spec 3.3.3.6. Action: 6 and 19 apply respectively.



3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

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

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 4 (Continued)

Event 5: Tech Spec evaluation continued

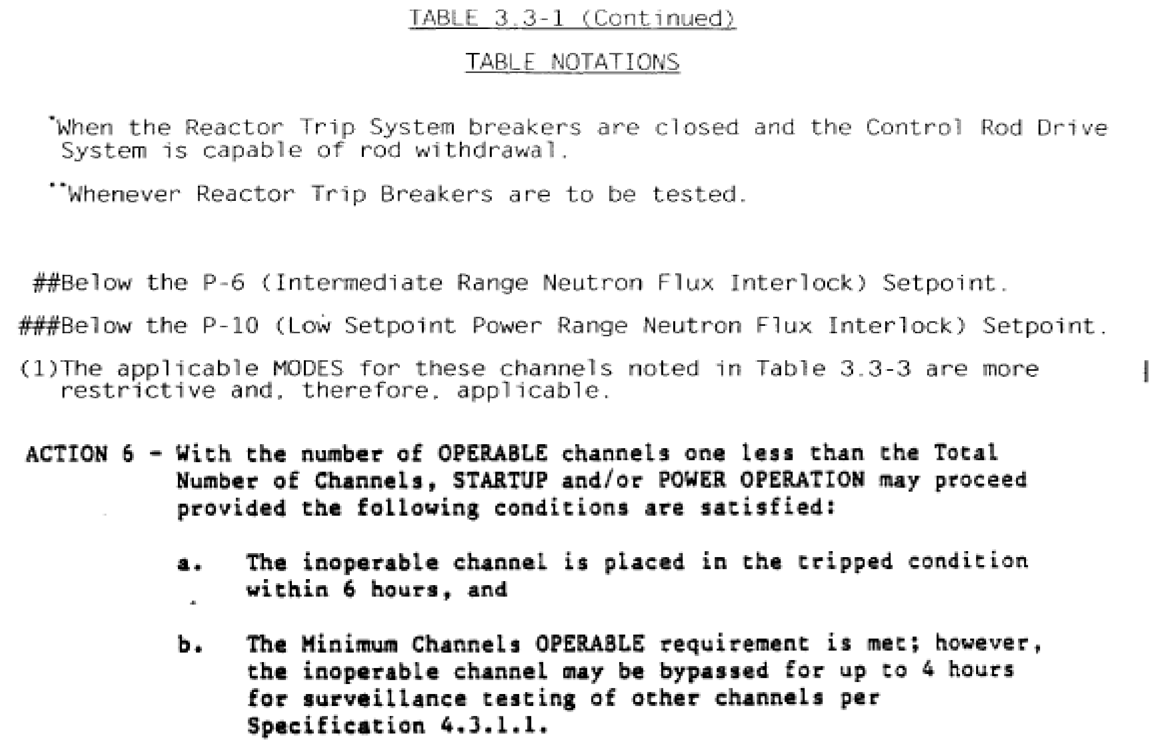


TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Whenever Reactor Trip Breakers are to be tested.

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

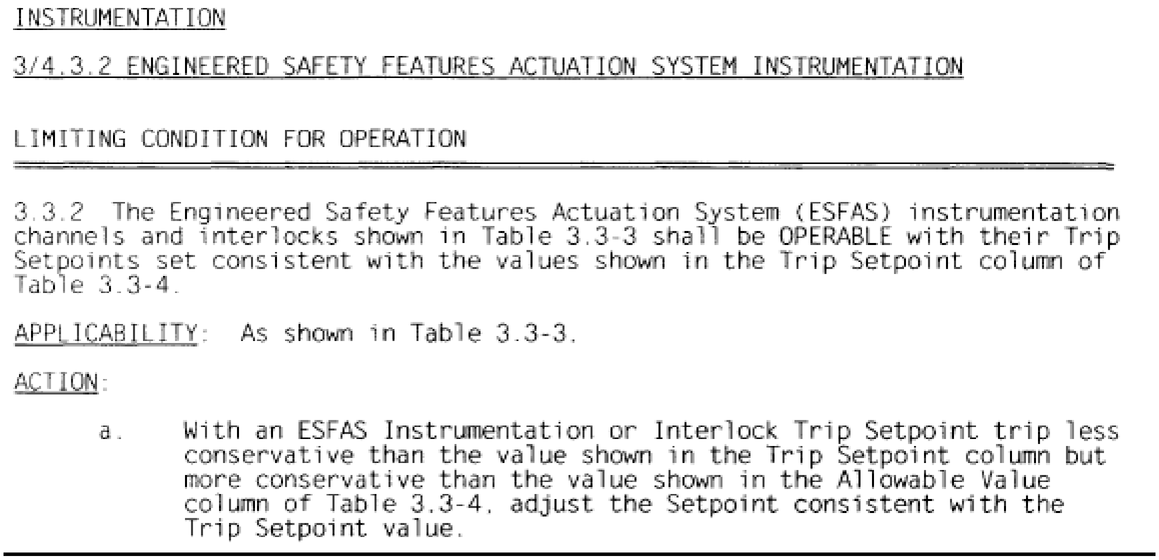
###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

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

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

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

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



INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 4 (Continued)

Event 5: Tech Spec evaluation continued

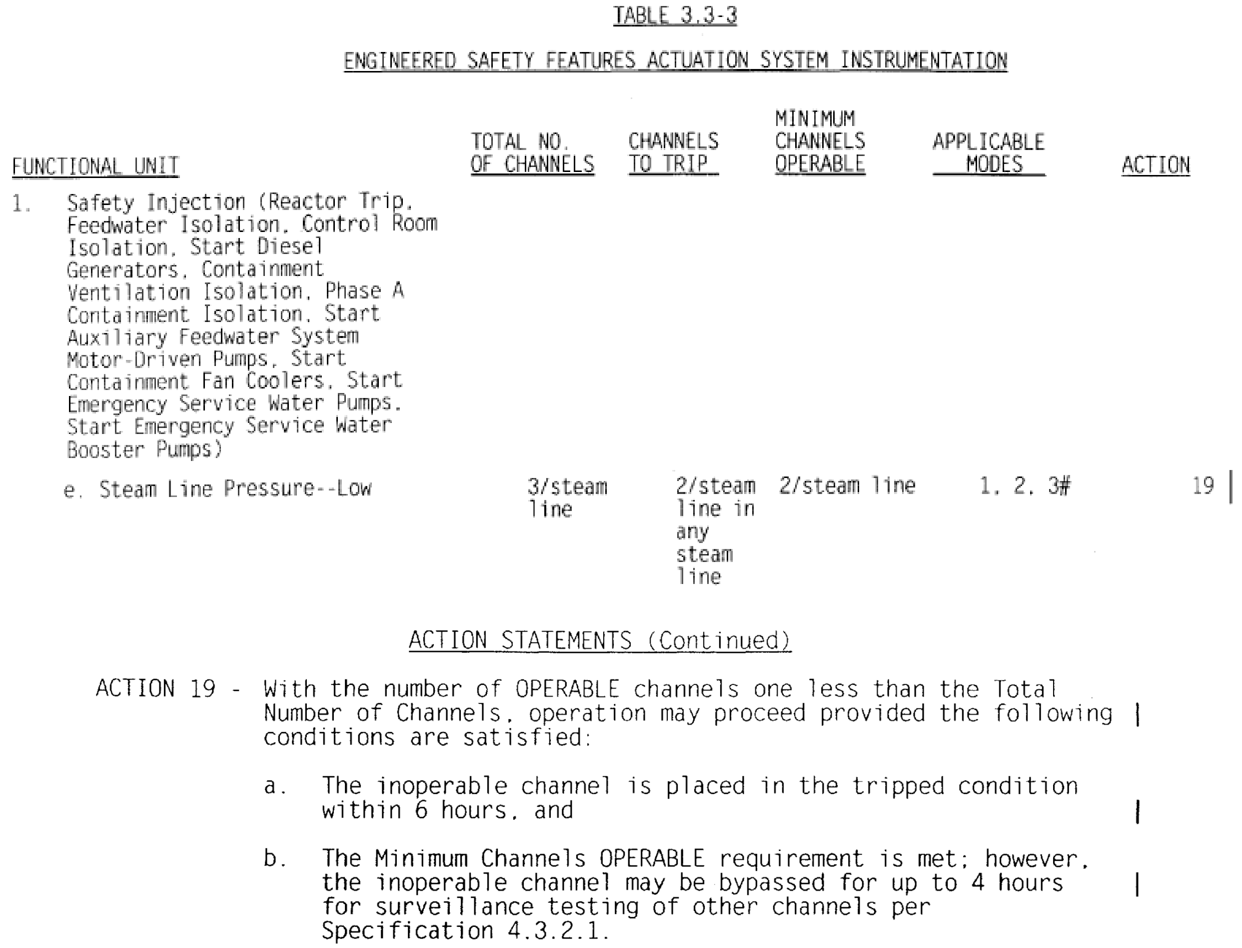


TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

ACTION STATEMENTS (Continued)

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

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

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

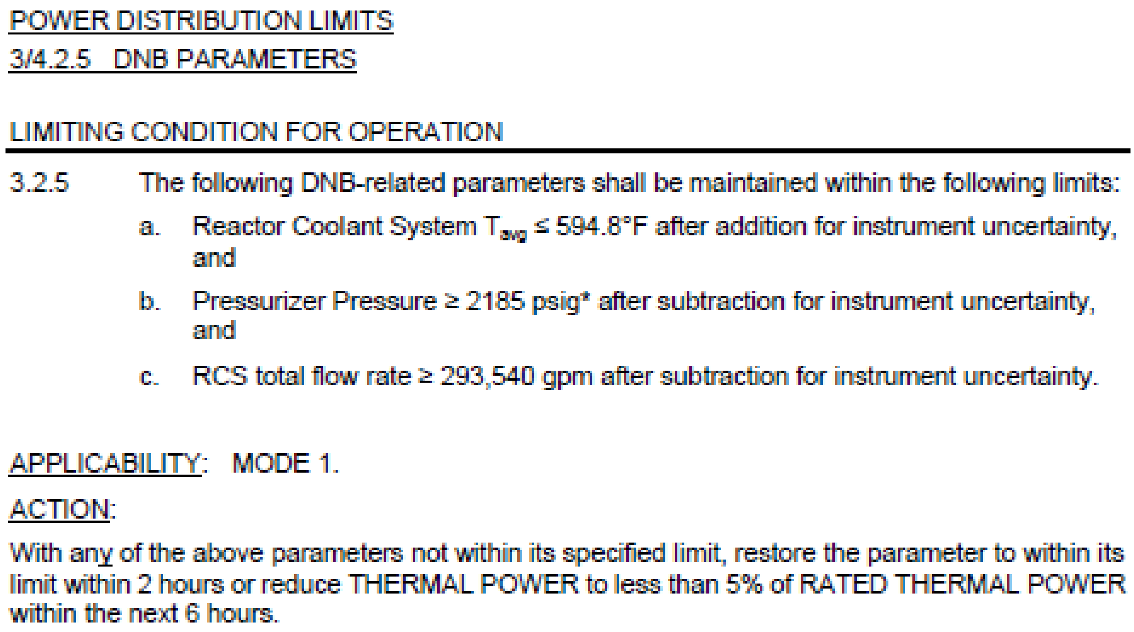
The SRO should provide a level band of 52% to 62% to the BOP in accordance with AOP-010 and OMM-001, Attachment 11, Control Bands And Administrative Limits. The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 6: PT-444 Fails HIGH (AOP-019). This failure will cause multiple ALB 009 annunciators to alarm along with lowering RCS pressure and changes in Pressurizer Level and Charging flow. This will require the crew to implement the immediate actions for AOP-019. Additionally ALB 010-8-5A, Cmptr Alarm Rx Coolant, will alarm if RCS pressure is allowed to lower below 2215 psig.

Verifiable Action: The crew will respond by entering AOP-019 and performing the immediate actions. The RO will place the 1RC-114, PRZ PORV 444B SB in the shut position which will not be successful requiring the block valve 1RC-113 to be shut (**Critical Task #2**).

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 4 (Continued)

The SRO should evaluate Tech Spec 3.2.5, Power Distribution Limits – DNB Parameters Action.



POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

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

b. Pressurizer Pressure ≥ 2185 psig* after subtraction for instrument uncertainty, and

c. RCS total flow rate ≥ 293,540 gpm after subtraction for instrument uncertainty.

APPLICABILITY: MODE 1.

ACTION:

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

The SRO should refer to AD-OP-ALL-1000 Attachment 4, Emergent Issue Checklists for the failure and request assistance from the WCC.

Event 7: Loss of Offsite Power with a Feed line break inside CNMT (EOP-E-0). The major event is a Feed line Break inside containment. The SG ‘A’ will degrade to a fault inside containment coincident with a loss of offsite power requiring the crew to implement the immediate actions of EOP-E-0 and stabilize the unit.

Verifiable Action: The crew will perform the EOP-E-0 immediate actions to ensure the Reactor is tripped, Turbine is tripped, and both AC emergency buses are energized. The crew should determine Safety Injection actuation is required based on rising containment pressure and sump level. They should monitor Safety Injection to ensure it automatically actuates at 3.0 psig in containment and continue with EOP-E-0. The BOP will stabilize RCS temperature using EOP-E-0, Table 1 and energize AC buses 1A1 and 1B1.

Event 8: Main Steam line Isolation fails, 'B' CCW pump fails to Auto start. The MSIVs will fail to close at 3.0 psig in containment and the sequencer will fail to start the ‘B’ CCW pump.

Verifiable Action: The BOP will attempt to manually actuate MSLI from the MCB in accordance with EOP-E-0, which will not be successful and then manually place each switch in the shut position in accordance with AD-OP-ALL-1000, Conduct of Operations, (5.6.3.8) for Equipment Manipulation and Status Control, but this will not be successful as well. The RO will manually start the ‘B’ CCW Pump once the ‘B’ Sequencer reaches Load Block 9, Automatic Manual

SCENARIO SUMMARY: 2020 NRC EXAM SCENARIO 4 (Continued)

Loading Permissive, in accordance with AD-OP-ALL-1000, Conduct of Operations, (5.6.3.8) for Equipment Manipulation and Status Control or EOP-E-0, Attachment 3, Safeguards Actuation Verification.

Event 9: 'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1). The crew should identify this failure and attempt to restore a source of Feedwater. Transition to EOP-FR-H.1 will be required at this time. The crew will continue with EOP-FR-H.1 until heat sink is restored or the requirement to initiate Bleed and Feed are met.

Verifiable Action: The RO will be required to secure any running RHR pumps in in accordance with EOP-FR-H.1
Event 10: Pressurizer PORV 445B fails to open. During the performance of EOP-FR-H.1 actions to establish Bleed and Feed PORV 445B the non-safety PRZ PORV will fail to open.

Verifiable Action: The crew should identify this failure and open the Reactor Vent valve to ensure an adequate RCS Bleed Path is established, in accordance with EOP-FR-H.1 step 30.

The scenario termination is met in EOP-FR-H.1 after RCS Bleed and Feed has been established prior to PRZ PORVs automatically opening (**Critical Task #3**).

CRITICAL TASK JUSTIFICATION:

1. Manually maintain control of SG ‘A’ level above 25% to prevent an automatic Reactor trip after steam pressure transmitter PT-475 fails low

An unnecessary automatic Reactor Trip for this event will create critical task. See note below.

2. Manually maintain control of PRZ Pressure above 1960 psig to prevent an automatic Reactor trip after the pressure transmitter PT-444 fails high

An unnecessary automatic Reactor Trip for this event will create critical task. See note below.

3. Initiate RCS Bleed and Feed for Successful High-Head SI Pump Injection to prevent RVLIS Full Range Level from lowering below 39%

Failure to initiate RCS bleed and feed before the RCS saturates at a pressure above the shutoff head of the high-head ECCS pumps results in significant and sustained core uncovery. If RCS bleed is initiated so that the RCS is depressurized below the shutoff head of the high-head ECCS pumps, then core uncovery is prevented or minimized. At Harris the plant with no Reactor Coolant Pump operating RVLIS Full Range Level lowering below 39% will provide indication of significant core uncovery.

Note: Causing an unnecessary plant trip or ESF actuation may constitute a CT failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

Simulator Setup

Reset to IC-144 password “NRC3sros”

Go to RUN

Silence and Acknowledge annunciators

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner.

Set ERFIS screens for normal full power conditions

(The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

SPECIAL INSTRUCTIONS

Provide a Reactivity Plan to candidates for shutting down the plant

Provide a copy of the following procedures:

- GP-006, NORMAL PLANT SHUTDOWN FROM POWER OPERATION TO HOT STANDBY (MODE 1 TO MODE 3) **marked up** through section 6.2 step 4

Press START on Counter Scaler
Post conditions for status board from IC-19 Reactor Power 100%
Control Bank D at 218 steps
RCS boron 954 ppm

Turnover: The plant is at 100% power, middle of core life. Due to the ‘B’ MDAFW pump LCO expiring, a normal shutdown in accordance with GP-006, Normal Plant Shutdown From Power Operation To Hot Standby (Mode 1 To Mode 3) is in progress as directed by plant management. Continue the shutdown @ 4 MW/min with TCS Load Control at 4 GVPC units/ min

Equipment Under Clearance:

- MDAFW Pump B-SB is under clearance for motor high vibrations. The pump has been inoperable for 66 hours and cannot be restored to operable status. Tech Spec 3.7.1.2 LCO Action **a** and Tech Spec 3.3.3.5.b Action **c** applies.

- ‘B’ DEH Pump is under clearance for motor repairs. The pump has been unavailable for 8 hours. Repairs are expected to be completed within 24 hours.

- 1CS-9, Letdown Orifice Isolation valve, has been under clearance the last 12 hours for solenoid replacement. The repairs are close to completion and the valve is expected to be returned this shift. The valve is currently shut with power removed. OWP-CS-09 has been completed. Tech Specs 3.6.3 Action **b** applies.

Simulator Setup (continued)

Align equipment for repairs:

Place CIT on 'B-SB' MDAFW pump MCB Switch
Place protected train placards in accordance with OMM-001 Attachment 5
    Protected Train placards on 'A-SA' MDAFW pump, 'B-SB' RHR Pump, 'B-SB' CCW Pump, 'B-SB' ESW Pump, 1MS-70 and 1MS-72

Place the “B” DEH Pump in PTL and then hang a CIT on MCB switch
Place protected train placards in accordance with AD-OP-ALL-0210, Single Point Vulnerabilities
    Protected Train placards on “A” DEH Pump

Place CIT on 1CS-9 MCB switch

Place protected train placards in accordance with Response to Industry Best Practices, Expectations
    Protected train placards on ‘A-SA’ ESW Pump, ‘A-SA’ CCW Pump, and ‘A-SA’ SFP Hx

Place filled out copies of OWP’s into the OWP book – ensure they are removed at end of day

- OWP-CS-09 and place in MCR OWP book for 1CS-9 clearance

Op Test No.:	NRC	Scenario #	4	Event #	1	Page	14	of	70	
Event Description:				**Power Reduction**						

Time	Position	Applicant's Actions or Behavior

Lead Evaluator:	**The crew has been directed to commence a power reduction from 100% to the unit is off line. The power reduction is on hold for turnover. The SRO is expected to conduct a reactivity brief prior to commencing the power reduction. This brief may be conducted outside the simulator prior to starting the scenario.**
	When the crew has completed their board walk down and are ready to take the shift inform the Simulator Operator to place the Simulator in Run. When the Simulator is in run announce: **CREW UPDATE – (SRO's Name) Your crew has the shift. END OF UPDATE**

Simulator Operator:	**When directed by the Lead Evaluator, ensure that the annunciator horns are on and place the Simulator in RUN.**

GP-006		GP-006, Section 6.2
Procedure Note:		**When PRZ backup heaters are energized in manual, then PK-444A1 (PRZ Master Pressure Controller) (a PI controller) will integrate up to a greater than normal output, opening PRZ Spray Valves to return and maintain RCS pressure at setpoint. The result is as follows:** • **PORV PCV-444B will open at a lower than expected pressure.** • **ALB-009-3-2 (Pressurizer High Press Deviation Control), will activate at a lower than expected pressure.** • **Higher probability for exceeding Tech Spec DNB limit for RCS pressure.**
	RO	Energize all available Pressurizer Backup Heaters per OP-100 Section 8.15.
Evaluator Note:		**The crew may elect to begin boration prior to lowering turbine load.**

Op Test No.:	NRC	Scenario #	4	Event #	1		Page	15	of	70
Event Description:					**Power Reduction**					

Time	Position	Applicant's Actions or Behavior
	RO	OP-107.01, Section 5.2
	RO	DETERMINE the volume of boric acid to be added. (Current OPT-1536 data or approved reactivity plan from Engineering may be used.)
	SRO	Directs boration
Procedure Note:		**FIS-113, BORIC ACID BATCH COUNTER, has a tenths position.**
Procedure Caution:		**If the translucent covers associated with the Boric Acid and Total Makeup Batch counters FIS-113 and FIS-114, located on the MCB, are not closed, the system will not automatically stop at the preset value.**
	RO	• SET FIS-113, BORIC ACID BATCH COUNTER, to obtain the desired quantity. • ENSURE the RMW CONTROL switch has been placed in the STOP position. • ENSURE the RMW CONTROL switch green light is lit.
Procedure Note:		• **Boric Acid flow controller must be set between 0.2 and 6 (1 and 30 gpm.).** • **Performing small borations at high flow rates may result in an overboration based on equipment response times. Boration flow should be set such that the time required to reach the desired setpoint will happen after release of the control switch.**

Time	Position	Applicant's Actions or Behavior
	RO	IF the current potentiometer setpoint of controller 1CS-283, FK-113 BORIC ACID FLOW, needs to be changed to obtain makeup flow, THEN: (N/A)
		• RECORD the current potentiometer setpoint of controller 1CS-283, FK-113 BORIC ACID FLOW, in Section 5.2.3. • SET controller 1CS-283, FK-113 BORIC ACID FLOW, for the desired flow rate.
	RO	PLACE control switch RMW MODE SELECTOR to the BOR position.
Procedure Note:		• **Boration may be manually stopped at any time by turning control switch RMW CONTROL to STOP.** • **During makeup operations following an alternate dilution, approximately 10 gallons of dilution should be expected due to dilution water remaining in the primary makeup lines.**
	RO	• START the makeup system as follows: o TURN control switch RMW CONTROL to START momentarily. o ENSURE the RED indicator light is LIT. o IF expected system response is not obtained, THEN TURN control switch RMW CONTROL to STOP. • ENSURE boration automatically terminates when the desired quantity of boron has been added.
	RO	IF controller 1CS-283, FK-113 BORIC ACID FLOW, was changed in Step 5.2.2.5, THEN: (N/A)
		• REPOSITION controller 1CS-283, FK-113 BORIC ACID FLOW, to the position recorded in Step 5.2.2.5.a. • INDEPENDENTLY VERIFY controller 1CS-283, FK-113 BORIC ACID FLOW, position.

Op Test No.: <u>NRC</u>	Scenario # 4	Event # 1	Page <u>17</u> of <u>70</u>
Event Description:	**Power Reduction**		

Time	Position	Applicant's Actions or Behavior
	RO	- Monitor Tavg and rod control for proper operation. - Establish VCT pressure between 20-30 psig. - Turn control switch RMW MODE SELECTOR to AUTO. - START the makeup system as follows: o TURN control switch RMW CONTROL to START momentarily. o ENSURE the RED indicator light is LIT. o IF expected system response is not obtained, THEN TURN control switch RMW CONTROL to STOP. (Ref. 4.0.31)
	SRO	GP-006, Section 6.2 continued
	SRO	DIRECTS BOP to start power reduction at 4 MW/Min. May direct initiation of a boration before the power reduction begins.
Procedure Note:		**Routine load changes must be coordinated with the Load Dispatcher to meet system load demands** **GVPC is the preferred method of Load Control. Megawatt Control is normally used only during GV and TV testing** **Controls and indications in following steps are on the TCS Load Control screen** **If Oper Entry is selected with the Turbine in GO, the value currently in the Ramp Rate Entry Window will become the load rate in effect. It may be desirable to place the turbine in HOLD to avoid undesirable ramp rates**
	BOP	Requests PEER check prior to manipulations of TCS Load Control screen

Op Test No.:	NRC	Scenario #	4	Event #	1	Page	18	of	70	
Event Description:						**Power Reduction**				

Time	Position	Applicant's Actions or Behavior
	BOP	On TCS Load Control screen, Load Control section, perform the following:
		a. IF GVPC indicator is TRUE, THEN go to Step 5.c c. Select Ramp Rate Selection, Select button d. Select the desired ramp rate OR Oper Entry on Load Ramp Rate Selection menu • ENTER the desired rate, **NOT** to exceed 5 MW/MIN, in the DEMAND display. **(4 DEH Units/minute)** e. IF Oper Entry is selected, THEN enter the desired loading rate in the Ramp Rate Entry window and depress Enter. • ENTER the desired rate, NOT to exceed 5 MW/MIN, in the DEMAND display. **(4 DEH Units/minute)** • DEPRESS the ENTER push-button.
Procedure Note:		**The unloading of the unit can be stopped at any time by selecting the Hold button. The load reduction can be resumed by selecting the Go button**
Evaluator Note:		**There is no procedural guidance directing when the boration to lower power is required. The crew may elect to perform the boration prior to placing the Turbine in GO.**
	BOP	Reduce turbine load as follows: a. Enter desired Target Load (120 MW if shutting down) in Target Entry window and depress Enter b. Select the Go button c. Check that Demand window indication counts down towards desired Target Load d. Check that load ramps towards desired Target Load
Procedure Note:		**Once a raise/lower command button is activated, it will remain in the visually depressed state as an indication the button cannot be activated again for approximately two seconds. After two seconds, command buttons automatically return to their default visual state indicating the button may be activated again**

Op Test No.:	NRC	Scenario #	4	Event #	1	Page	19 of 70
Event Description:				**Power Reduction**			

Time	Position	Applicant's Actions or Behavior
	BOP	IF AT ANY TIME, a small incremental change of Target Load value (1 or 5 megawatts) is desired, THEN select any of the following buttons: • ▲1 MW • ▲▲5 MW • ▼ 1 MW • ▼▼ 5 MW
	BOP	Ensure Generator load is lowering
Evaluator Note:		**As the crew demonstrates a satisfactory load reduction, cue Simulator Operator to insert Trigger 2** **Event 2: Normal Service Water Pump 'A' sheared shaft (AOP-022)**

Op Test No.:	NRC	Scenario #	4	Event #	2	Page	20	of	70	
Event Description:	**Normal Service Water Pump 'A' sheared shaft (AOP-022)**									

Time	Position	Applicant's Actions or Behavior

Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 2 "Normal Service Water Pump 'A' sheared shaft (AOP-022)"**	
Indications Available:		• **ALB 002-4-5, SERV WTR LEAKAGE** • **ALB 002-5-5, SERV WTR HEADER A HIGH-LOW FLOW** • **ALB-002-6-1, SERV WTR SUPPLY HDR A LOW PRESS** • **ALB 002-6-6, SERV WTR HEADER B HIGH-LOW FLOW** • **ALB-002-7-1, SERV WTR SUPPLY HDR B LOW PRESS** • **ALB-002-7-2, SERV WTR PUMPS DISCHARGE LOW PRESS**	
	RO	Responds to ALB-002 alarms – reports low NSW header pressure with pump running indication.	
Evaluator Note:		**The ESW Pumps will auto start on low header pressure after 20 second time delay.**	
AOP-022		Loss Of Service Water	
	SRO	ENTERS and directs actions of AOP-022, Conducts a Crew Update Makes PA announcement for AOP entry.	
Immediate Action	RO	CHECK ESW flow lost to ANY RUNNING CSIP - MORE THAN 1-minute:	(NO)
	SRO	RNO: GO TO Step 2.	
Immediate Action	RO	CHECK ESW flow lost to ANY RUNNING EDG - MORE THAN 1-minute:	(NO)
	SRO	RNO: GO TO Step 3.	

Op Test No.:	NRC	Scenario #	4	Event #	2	Page	21	of	70
Event Description:	**Normal Service Water Pump ‘A’ sheared shaft (AOP-022)**								

Time	Position	Applicant’s Actions or Behavior	
Simulator Communicator:		**There are several points in the AOP where an AO may be dispatched to check for leaks and proper operation of equipment. Report no leaks, no breaker problems but when dispatched to the pump, after 1 to 2 minutes report that the coupling appears to have failed and request maintenance assistance.**	
Simulator Operator:		**IF REQUESTED TO OPEN KNIFE SWITCH ON THE ‘A’ NSW PUMP BREAKER: go to rf SWS100 and “open the knife switch” then have Communicator report back when completed**	
	SRO	GO TO the appropriate step as indicated by the parameter LOST: • NSW Pump failure • NSW Pump loss of flow GO TO 3.0/ Step 6 (Page 9)	(YES)
	SRO	PERFORM the following for a loss of NSW flow:	
	RO	a. CHECK loss of NSW Header due to NSW Pump FAILED or LOSS OF FLOW. b. START standby NSW Pump as follows: • ENSURE discharge valve for affected pump is CLOSING by placing affected pump control switch to STOP. • START standby NSW Pump in priming mode by momentarily placing standby NSW Pump control switch to START. • WHEN discharge valve for affected pump is fully SHUT, THEN PLACE and HOLD control switch for running pump to START to fully OPEN pump discharge valve. c. CHECK ANY NSW Pump - RUNNING.	(YES) (YES)
	SRO	d. GO TO Section 3.2 (page 37)	

Evaluator Note:		**The following alarms will annunciate due to loss of cooling in containment and subsequent start of ESW:** • **ALB-028-5-1, CONTAINMENT AIR HIGH VACUUM** • **ALB-028-8-5, COMPUTER ALARM VENTILATION SYSTEM** **The BOP should identify these alarms and identify Tech Specs 3.6.1.4, 3.6.1.1, 3.6.3, 3.6.5 and 3.9.4 to be referenced**	
	BOP	MAY go to MANUAL and shut FK-7624, Norm Purge Exh Flow, in order to raise CNMT pressure to exit T.S. 3.6.1.4 (ALB-028-5-1, 3.c and AD-OP-ALL-1000) NOTE: informs CRS prior to taking manual control for need of actions	
	SRO	T.S. 3.6.1.4 – Restore within 1 hour LCO or HSB within next 6 hours: due to High Vac in CNMT	
	SRO	CHECK Turbine trip required by ANY of the following conditions - EXIST:	
	RO	• No NSW Pump can be operated • Non-isolable leak exists in the NSW system • Major isolable leak exists on the Turbine Building NSW Header AND time does not permit a controlled plant shutdown	(NO)
	SRO	RNO: OBSERVE Note prior to Step 13 AND GO TO Step 13.	
Procedure Note:		**Steps 13 through 19 address leaks on NSW turbine building header. Leaks on individual components supplied by the Turbine Building header are addressed by Steps 20 and 21.**	

Op Test No.:	NRC	Scenario #	4	Event #	2	Page	23	of	70
Event Description:	**Normal Service Water Pump 'A' sheared shaft (AOP-022)**								

Time	Position	Applicant's Actions or Behavior	
	CREW	CHECK for minor isolable leak on Turbine Building header – ANY EXISTING.	(NO)
	SRO	RNO: GO TO Step 20.	
	CREW	CHECK for leak in an individual component - ANY EXISTING.	(NO)
	SRO	RNO: GO TO Step 22.	
	CREW	CHECK for leak on WPB header - ANY EXISTING.	(NO)
	SRO	RNO: GO TO Step 24.	
	RO	CHECK that NSW Pump(s) - MALFUNCTIONED.	(YES)
	CREW	PERFORM the following for affected NSW Pump(s): • CHECK NSW Pump breaker(s) - MALFUNCTIONED.	(NO)
	SRO	RNO: GO TO Step 25.b.	

Op Test No.: NRC	Scenario # 4	Event # 2	Page 24 of 70
Event Description:	**Normal Service Water Pump 'A' sheared shaft (AOP-022)**		

Time	Position	Applicant's Actions or Behavior	
Procedure Note:		**If Service Water Chamber level indication is not available, a substituted conservative value of LESS THAN 31 INCHES Cooling Tower Basin level (LI-1931) may indicate that Service Water Chamber level is low.**	
	CREW	• CHECK adequate pump suction inventory EXISTS: ○ LI-9300.1, Service Water PMP A CHMBR LVL, GREATER THAN 51% (ERFIS LSW9300) ○ LI-9302, Service Water PMP B CHMBR LVL, GREATER THAN 51% (ERFIS LSW9302) ○ LI-1931, Cooling Tower Basin Level, GREATER THAN 31 inches	(YES) (YES) (YES)
	CREW	• Locally VERIFY the following for the affected NSW Pump per OP-139, Service Water System: ○ Proper cooling and seal water supply to NSW Pumps. ○ Proper operation of NSW strainer backwash. • Locally CHECK NSW Pump(s) for signs of damage (shaft shear or other obvious problems).	(YES) (YES) (YES)
	SRO	INITIATE appropriate corrective action for the loss of NSW. • Completes an Emergent Issue Checklists and contacts WCC for the failure of "A" NSW Pump assistance. (WR, LCOTR and Maintenance support)	
Simulator Communicator:		**Acknowledge communications**	
	SRO	CHECK Reactor thermal power changed by less than 15% in any one hour period	(YES)

Op Test No.:	NRC	Scenario #	4	Event #	2	Page	25	of 70
Event Description:		**Normal Service Water Pump 'A' sheared shaft (AOP-022)**						

Time	Position	Applicant's Actions or Behavior
	RO	IF ESW Pump(s) were placed in service by this procedure, THEN NOTIFY Chemistry to sample the return to the Auxiliary Reservoir per CRC-155
	SRO	Exit AOP-022
Evaluator Note:		**With NSW restored to the Turbine and load lowering, cue Simulator Operator to insert Trigger 3** **Event 3: Pressurizer PORV 445A Leakage (AOP-016)**

Op Test No.:	NRC	Scenario #	4	Event #	3	Page	26	of	70	
Event Description:		**Pressurizer PORV 445A Leakage (AOP-016)**								

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 3 "Pressurizer PORV 445A Leakage (AOP-016)"**
Indications Available:		• **ALB-009-8-2, PRESSURIZER RELIEF DISCHARGE HIGH TEMP** • **TI-463, PRZ PORV discharge line temperature rising** • **LI-470.1, Pressurizer relief tank level rising** • **PI-472.1, Pressurizer relief tank pressure rising** • **TI-471.1, Pressurizer relief tank temperature rising**
Evaluator Note:		**Responding to the annunciator will direct the operator to shut 1RC-117, PRZ PORV Isolation valve to stop leakage from PRZ PORV PCV-445A. With the condition clear the crew may not enter AOP-016.**
APP-ALB-009	RO	Responds to alarm and evaluates APP-ALB-009-8-2
Procedure Note:		Past experience has shown that this alarm may come in due to valve stem leakoff from one of the PORV Block Valves. The block valves share a common leak-off line with the PORVs. This can be checked using ERFIS points TVL5647 and TVL5646
	RO	• CONFIRM alarm using: o TI-463, PRZ PORV discharge line temperature o LI-470.1, Pressurizer relief tank level o PI-472.1, Pressurizer relief tank pressure o TI-471.1, Pressurizer relief tank temperature o Reports TI-463, LI-470.1, PI-472.1, TI-471.1 reading or trending high. • VERIFY Automatic Functions: None

Op Test No.:	NRC	Scenario #	4	Event #	3	Page	27	of	70
Event Description:	**Pressurizer PORV 445A Leakage (AOP-016)**								

Time	Position	Applicant's Actions or Behavior	
Procedure Caution:		Any PORV isolations that are shut due to decreasing RCS Pressure should NOT be reopened without further evaluation	
	RO	PERFORM Corrective Actions: Monitors TI-401 indications and identifies temperature is lowering	
	RO	• IF a PORV is open, THEN CHECK PRZ pressure using PI-444, PI-445.1, PI-455.1, PI-456, and PI-457.	(NO)
Procedure Note:		For minor leakage, it may be necessary to have Engineering assistance to develop proper strategies	
	RO	• IF all PORV's indicate closed and RCS pressure is NOT normal: • IF all PORV's indicate closed and RCS pressure is normal: o THEN SHUT one PORV isolation at the time. o IF PRZ PORV discharge line temperature is not affected, THEN REOPEN the isolation valve.	(NO) (YES)
Evaluator Note:		**ERFIS Point TRC-0463 can be used to evaluate if PORV is leaking. ERFIS Quick Plot "QP PRT" can be used to monitor this parameter.**	
	RO	• Shuts PORV isolations as directed by SRO o After shutting 1RC-117, PRT Relief Line Temperature stops rising and PRT pressure stabilizes • Determines/reports PRZ PORV-445A leaking. • Informs SRO leakage from PRZ PORV PCV-445A is isolated	
	SRO	Directs RO to reopen 1RC-115 and or 1RC-113 if shut.	

Op Test No.:	NRC	Scenario #	4	Event #	3	Page	28	of 70
Event Description:		**Pressurizer PORV 445A Leakage (AOP-016)**						

Time	Position	Applicant's Actions or Behavior	
Evaluator Note:		**Any Tech Spec evaluation can be conducted with a follow up question after the scenario.**	
	SRO	Evaluates Reactor Coolant System TS 3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE. ACTION a**.** - With one or more PORV(s) inoperable because of excessive seat leakage within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s): otherwise be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.	
	SRO	Completes an Emergent Issue Checklists for leakage from PRZ PORV PCV-445A.	
Simulator Communicator:		**Acknowledge communications**	
Evaluator Note:		**The following write up is if AOP-016 is used for the response to the leakage from PRZ PORV PCV-445A.**	
	CREW	Identifies entry conditions to AOP-016, Excessive Primary Plant Leakage are met	
AOP-016		Excessive Primary Plant Leakage	
	SRO	ENTERS and directs actions of AOP-016, Conducts a Crew Update Makes PA announcement for AOP entry	
Procedure Note:		**This procedure contains no immediate actions.**	
	RO	CHECK RHR in operation	(NO)

Op Test No.:	NRC	Scenario #	4	Event #	3	Page	29	of	70
Event Description:	**Pressurizer PORV 445A Leakage (AOP-016)**								

Time	Position	Applicant's Actions or Behavior	
	SRO	REFER TO PEP-110, Emergency Classification And Protective Action Recommendations, AND ENTER the EAL Matrix.	
	RO	CHECK RCS leakage within VCT makeup capability May report that the leak is exceeding Tech Spec SG leakage.	(YES)
Procedure Note:		If CSIP suction is re-aligned to the RWST, negative reactivity addition should be anticipated.	
	RO	MAINTAIN VCT level GREATER THAN 5%	(YES)
	SRO	RNO: GO TO Step 10.	
	RO	CHECK valid CNMT Ventilation Isolation monitors (REM-3561A, B, C and D) ALARM CLEAR CHECK RM 3502A, RCS Leak Detection Radiation Monitor, ALARM CLEAR CHECK ALL valid Area Radiation Monitors ALARM CLEAR CHECK valid Stack Monitors ALARM CLEAR	(YES) (YES) (YES) (YES)
	SRO	DETERMINE if unnecessary personnel should be evacuated from affected areas, as follows:	
		CHECK that a valid RMS Secondary Monitor HIGH ALARM CHECK that an RCS leak outside Containment, other than SG tube leakage, has caused a valid RMS alarm.	(NO) (NO)
	SRO	RNO: GO TO Step 15.	

Op Test No.:	NRC	Scenario #	4	Event #	3	Page	30	of 70
Event Description:		**Pressurizer PORV 445A Leakage (AOP-016)**						

Time	Position	Applicant's Actions or Behavior
	BOP	NOTIFY Chemistry to stop any primary sampling activities.
Simulator Communicator:		**Acknowledge request to stop primary sampling activities.**
Procedure Note:		• The following qualitative flow balance is to quickly determine if RCS leakage exceeds Tech Spec limits, EAL classification thresholds, or RCS makeup capability. • RCS influent and effluent flow rates are compared and PRZ level rate of change is used to determine the RCS flow balance.
	RO	PERFORM a qualitative RCS flow balance, as follows: a. ESTIMATE leak rate considering the following parameters: • PRZ level rate of change (~55 gal/% at 653°F) • Charging flow • Total seal injection flow • Letdown flow • Total seal return flow Reports estimate to SRO of ~ 15 gpm
		b. OPERATE the following letdown orifice valves as necessary to maintain charging flow on scale: • 1CS-7, 45 gpm Letdown Orifice A • 1CS-8, 60 gpm Letdown Orifice B • 1CS-9, 60 gpm Letdown Orifice C (No changes required)
Procedure Note:		Performance of surveillance tests to determine if leakage exceeds Tech Spec limits, or to more accurately quantify leakage is up to CRS discretion.
	SRO	Determines that more accurate quantification is not needed due to excessive leakage indications present.

Op Test No.:	NRC	Scenario #	4	Event #	3	Page	31	of	70	
Event Description:		**Pressurizer PORV 445A Leakage (AOP-016)**								

Time	Position	Applicant's Actions or Behavior	
Evaluator Note:		**Any Tech Spec evaluation can be conducted as a follow up question after the scenario.**	
	SRO	EVALUATE RCS leakage (refer to Tech Spec 3.4.6.2). **(N/A < 10 gpm based on changes in plant parameters)**	
	SRO	DETERMINE leak location from one or more of the following: MCB indications and Valid Radiation Monitors • From PRZ PORV PCV-445A	
	BOP	NOTIFY Health Physics of the following: a. Leak location: • Source inside or outside CNMT • To closed system, SG or to atmosphere b. Applicable radiation levels. NOTIFY HP of leakage from PRZ PORV PCV-445A	
Simulator Communicator:		**Acknowledge RCS leakage is coming from PRZ PORV PCV-445A.**	
	SRO	WHEN leakage location has been determined, THEN PERFORM the applicable Attachment: Leakage From Pressurizer PORV Attachment 5 page 27	
	SRO	Transitions to Attachment 5:	
	RO	• CHECK the PRZ PORVs SHUT. • CHECK that the leaking PORV has been identified.	(YES) (NO)

Op Test No.:	NRC	Scenario #	4	Event #	3	Page	32	of	70
Event Description:		**Pressurizer PORV 445A Leakage (AOP-016)**							

Time	Position	Applicant's Actions or Behavior	
	SRO	PERFORM ONE of the following based on severity of leak:	
	RO	• SHUT AND REOPEN ONE PORV Block Valve at a time to identify the affected PORV. • IF leakage is significant AND RCS pressure is normal, THEN: o SHUT ALL PORV Block Valves. o REOPEN ONE PORV Block Valve at a time to identify the affected PORV.	(YES) (NO)
Evaluator Note:		**Any Tech Spec evaluation can be conducted with a follow up question after the scenario.**	
	SRO	REFER TO Tech Spec 3.4.4.	
		Evaluates Reactor Coolant System TS 3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE. ACTION a**.** - With one or more PORV(s) inoperable because of excessive seat leakage within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s): otherwise be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.	
	SRO	VERIFY valves manipulated for leak isolation are documented per the following: • OMM-001, Operations Administrative Requirements • OPS-NGGC-1303, Verification Practices	
	SRO	Exit AOP-016	
	SRO	Completes an Emergent Issue Checklists for leakage from PRZ PORV PCV-445A. Contacts WCC for assistance. (WR, LCOTR and Maintenance support).	

Op Test No.:	NRC	Scenario #	4	Event #	3	Page	33	of	70

Event Description: **Pressurizer PORV 445A Leakage (AOP-016)**

Time	Position	Applicant's Actions or Behavior

Simulator Communicator:		**Acknowledge communications**
Evaluator Note:		**After Pressurizer PORV 445A Leakage has stabilized, cue Simulator Operator to insert Trigger 4** **Event 4: Generator Voltage Regulator Failure.**

Op Test No.:	NRC	Scenario #	1	Event #	4	Page	34	of 70
Event Description:			**Generator Voltage Regulator Failure**					

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 4 "Generator Voltage Regulator Failure"**
Indications Available:		• **ERFIS Quick Plot "QP VAC"** • **EI-565, Generator Megawatts** • **EI-567, Megavars** • **ALB 022-9-4, COMPUTER ALARM GEN/EXCITER SYSTEMS** • **ALB 022-4-3, GEN VOLT/FREQ RATIO LIMITER ACTIVE OR UNDER-FREQ**
Evaluator Note:		**ALB-022-9-4 is a computer alarm.** **ALB-022-4-3 provides direction for corrective actions.** **The crew may refer to AOP-006, Turbine Generator Trouble but no actions will result.**
ALB-022	BOP	RESPONDS to alarm on APP-ALB-022-4-3
Evaluator Note:		**Operator may use AD-OP-ALL-1000 guidance to take manual control of voltage regulator to avoid a trip or transient prior to receiving ALB-022-4-3.**
	BOP	CONFIRM alarm using: • EI-525, Generator Frequency. • EI-520, Generator Phase Volts. (YES-Reports voltage regulation problem) • EI-540, Gen Exciter Field Volts. • EI-541, Gen Exciter Field Current.
	BOP	VERIFY Automatic Functions: • VOLTAGE Regulator Limiter decreases Generator excitation • IF Voltage Limiter is unable to control excitation increase, a Generator Lockout occurs

Op Test No.:	NRC	Scenario # 1	Event # 4	Page	35	of	70
Event Description:			Generator Voltage Regulator Failure				

Time	Position	Applicant's Actions or Behavior	
	BOP	PERFORM Corrective Actions:	
		CHECK for the following at MCB: • EI-525, Generator Frequency, stable at 60 Hz. • EI-520, Generator Phase Volts, stable at 22 KV. • EI-540, Exciter Field Voltage stable. • EI-541, Exciter Field Current stable.	(YES) (NO) (NO) (YES)
Procedure Note:		An automatic transfer to MANUAL voltage control is indicated by CS-1538, Operation Mode switch, white light being lit. The CS-1538, Operation Mode switch, amber light will be off.	
	BOP	OPERATE CS-1539, Voltage Setpoint Reference switch, to restore Generator voltage to 22 KV and reduce MVARS.	
		IF CS-1539, Voltage Setpoint Reference switch, is ineffective AND an automatic voltage regulator control failure is suspected, THEN PERFORM the following to transfer and maintain voltage manually: • PLACE CS-1538, Operation Mode switch, in MANUAL mode. • OPERATE CS-1539, Voltage Setpoint Reference switch, to stabilize the Generator Stator Voltage at 22KV and reduce MVARS. • DISPATCH operator to 286 TB switchgear room to check the Excitation Control Terminal (ECT) (1EE-E258:137) on the ABB Automatic Voltage Regulator (AVR) cabinet for any event or alarm indications. o IF any event or alarm indications are present, THEN NOTIFY Maintenance.	(YES)
Simulator Communicator:		**If dispatched to 286' Switchgear to inspect ABB Automatic Voltage Regulator locally, wait approximately 2 minutes and report that there are no abnormal indications at the ABB Automatic Voltage Regulator.**	
	SRO	Directs BOP to maintain a MVAR output controlling band of 75 to 160 MVAR gross output per OP-153.01.	

Op Test No.:	NRC	Scenario #	1	Event #	4	Page	36	of 70
Event Description:			**Generator Voltage Regulator Failure**					

Time	Position	Applicant's Actions or Behavior
	BOP	- IF AVR in Manual, THEN NOTIFY Load Dispatcher within 30 minutes of an Automatic Voltage Regulator status change. (The notification shall include an explanation of the status change and an estimate of expected duration.) [R – Reference 5]
Simulator Communicator:		**Acknowledge report from Control Room**
	SRO	REFERENCE AOP-028, Grid Instability. [R - Reference 6]
	BOP	VERIFY Main Generator is operating per the Generator Capability Curve.
	SRO	Completes an Emergent Issue Checklist and contacts WCC for assistance. (WR, Maintenance support)
Simulator Communicator:		**Acknowledge requests for assistance.**
Lead Evaluator:		**After the Generator Voltage Regulator is stabilized, cue Simulator Operator to insert Trigger 5** **Event 5: Feed pressure transmitter failure low on 'A' SG PT-475 (AOP-010)**

Op Test No.:	NRC	Scenario #	4	Event #	5	Page	37	of	70
Event Description:	**Feed pressure transmitter failure low on 'A' SG PT-475 (AOP-010)**								

Time	Position	Applicant's Actions or Behavior	
Simulator Operator:		**On cue from the Lead Evaluator insert Trigger 5** **Feed pressure transmitter failure low on 'A' SG PT-475 (AOP-010)**	
Available Indications		• **ALB 014-1-2, LOOP A HI STEAM LINE ΔP LOW-P1** • **ALB 014-1-4, LOOP A HI STEAM LINE PRESS RATE ALERT** • **ALB 014-4-1A, SG A FW > STM FLOW MISMATCH** • **ALB 014- 4-2A, LOOP A LOW STM LINE PRESS ALERT**	
	BOP	RESPONDS to alarms and ENTERS AOP-010	
AOP-010		Feedwater Malfunctions	
Critical Task # 1 **Immediate Action**	BOP	CHECK Feedwater Regulator valves operating properly. **RNO** PERFORM the following: • PLACE affected Feedwater Regulator valve(s) in MANUAL. Places SG 'A' Feedwater Reg valve in MANUAL • MAINTAIN Steam Generator level(s) between 52 and 62%. Checks SG level and operates manual controller to maintain level between 52%-62% ***Critical Task: Maintain control of SG 'A' level above 25% to prevent an automatic Reactor trip after the controlling level transmitter PT-475 fails low.*** IF Steam Generator level(s) cannot be controlled, THEN TRIP the Reactor AND GO TO EOP-E-0. **(Should be controlled)**	(NO)
Immediate Action	BOP	CHECK ANY Main Feedwater Pump TRIPPED	(NO)

Op Test No.:	NRC	Scenario #	4	Event #	5	Page	38	of	70

Event Description: **Feed pressure transmitter failure low on ‘A’ SG PT-475 (AOP-010)**

Time	Position	Applicant’s Actions or Behavior	
	SRO	RNO: GO TO STEP 6	
		ENTERS and directs actions of AOP-010, Conducts a Crew Update Makes PA announcement for AOP entry	
	SRO	• Directs BOP to maintain controlling band of 52% to 62% per OMM-001 attachment 11.	

Controller	Control Band	Administrative Limit	
		Low	High
Steam Generator Level	52% to 62%	30%	73%

Time	Position	Applicant’s Actions or Behavior	
	BOP	MAINTAIN ALL of the following: • At least ONE Main Feedwater Pump RUNNING • Main Feedwater flow to ALL Steam Generators • ALL Steam Generator levels greater than 30% Maintains all of the above	
	BOP	CHECK Feedwater Regulator Valves operating properly in AUTO: **(NO not ‘A’)** • Response to SG levels • Valve position indication • Response to feed flow/steam flow mismatch	(NO)
		RNO PERFORM the following: • IF automatic SG water level control can be restored by selecting out a failed instrument, THEN USE OP-134.01, Feedwater System, Section 8.10 to swap Steam Flow/Feed Flow Control and Recorder Channels and restore level control to automatic.	(YES)
	BOP	IF swapping Steam Generator A channels. THEN PERFORM the following: • PLACE MAIN FW A REGULATOR FK-478, 1FW-133 in MAN. • IF selecting Channel III, THEN PERFORM the following: • IF selecting Channel IV, THEN PERFORM the	**(N/A)**

Op Test No.:	NRC	Scenario #	4	Event #	5	Page	39	of 70
Event Description:	**Feed pressure transmitter failure low on 'A' SG PT-475 (AOP-010)**							

Time	Position	Applicant's Actions or Behavior	
		following: • PLACE the following selector switches to the position specified: ○ STM GEN A FW FLOW CONTROL AND RECORDER Selector Switch to FT-476. ○ STM GEN A STM FLOW CONTROL AND RECORDER Selector Switch to FT-475.	
	BOP	• PERFORM the following to restore 1FW-133 to AUTO: ○ ENSURE proper indication for steam flow and feed flow on the S/G 1A LEVEL, STEAM FLOW & FEEDWATER FLOW recorder, UR-478. ○ ENSURE associated SG level (LT-476) is trending towards 57%. ○ PLACE MAIN FW A REGULATOR FK-478, 1FW-133 to AUTO.	
	BOP	• REFER to Tech Spec 3.3.1 AND IMPLEMENT OWP-RP or OWP-ESF where appropriate. • IF needed, THEN CONTROL feed flow to SGs using Main Feed Reg Valve Bypass FCVs	(NO)
Procedure Note:		Inability to monitor one or more Safety System Parameters concurrent with a turbine runback of greater than 25%, requires a change of event classification per the HNP Emergency Plan. [C.2, C.3].	
	BOP	CHECK turbine runs back less than 25% turbine load	(YES)
Procedure Note:		A feedwater train consists of a Condensate Pump, Condensate Booster Pump and Main Feedwater Pump.	
	SRO	GO TO the applicable section: EVENT: All Condensate/Feedwater flow malfunctions (other than pump trips) Section 3.1 Page 10	
	BOP	CHECK the following Recirc and Dump Valves operating properly in MODU:	

Op Test No.:	NRC	Scenario #	4	Event #	5	Page	40 of 70
Event Description:	**Feed pressure transmitter failure low on 'A' SG PT-475 (AOP-010)**						

Time	Position	Applicant's Actions or Behavior	
		• Main Feedwater Pumps • Condensate Booster Pumps • Condensate Pumps • 1CE-293, Condensate Recirc • 1CE-142, Condensate Dump To CST Isolation Valve (SLB-4/7-1)	 (YES) (YES) (YES) (YES) (YES)
	BOP	CHECK the Condensate and Feedwater System INTACT.	
Procedure Note:		Pumps should be stopped in the order of higher to lower pressure. (To stop a Condensate Pump, stop a Main Feedwater Pump followed by a Condensate Booster Pump and then the Condensate Pump.)	
	BOP	CHECK pumps for NORMAL OPERATION	(YES)
	SRO	NOTIFY Load Dispatcher of ANY load limitations. (No load limitations so Dispatcher will not be called)	
	SRO	CHECK Reactor thermal power changed by less than 15% in any one hour period.	(NO)
	SRO	EXIT this procedure.	
OWP-ESF-02	SRO	Refer to OWP-ESF-02 to remove channel from service.	
	SRO	Contacts WCC for support, requests WR and LCOTR. Contacts I&C to have channel removed from service.	
Simulator Communicator:		**Respond to crew requests.**	
Evaluator Note:		**Any Tech Spec evaluation may be completed with a follow-up question after the scenario.**	

Op Test No.:	NRC	Scenario #	4	Event #	5	Page	41	of	70
Event Description:	**Feed pressure transmitter failure low on 'A' SG PT-475 (AOP-010)**								

Time	Position	Applicant's Actions or Behavior
	SRO	Enters Instrumentation TS 3.3.1 Functional Unit 14 ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels. STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: a. The inoperable channel is placed in the tripped condition within 6 hours. b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1. 3.3.2 Functional Unit 1.e ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied : a. The inoperable channel is placed in the tripped condition within 6 hours, and b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.
Evaluator Note:		**Channel does NOT have to be removed from service using the OWP to continue the scenario. Once SG level is under control and the TS has been identified, cue Simulator Operator to insert Trigger 6** **Event 6: PT-444 Fails HIGH (AOP-019).**

Op Test No.: NRC	Scenario # 4	Event # 6	Page 42 of 70
Event Description:	**PT-444 Fails HIGH (AOP-019)**		
Time	Position	Applicant's Actions or Behavior	

Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 6 "PT-444 Fails HIGH (AOP-019)"**	
Indications Available:		• **ALB-009-3-2 PRESSURIZER HIGH PRESS DEVIATION CONTROL** • **ALB-009-5-1 PRESSURIZER HIGH-LOW PRESS** • **ALB-009-8-1 PRESSURIZER RELIEF TANK HIGH-LOW LEVEL PRESS OR TEMP** • **ALB-009-8-2 PRESSURIZER RELIEF DISCHARGE HIGH TEMP**	
	CREW	Identifies entry conditions to AOP-019, Malfunction Of RCS Pressure Control are met	
AOP-019		Malfunction Of RCS Pressure Control	
Immediate Action	RO	CHECK that a bubble exists in the PRZ.	(YES)
Immediate Action	RO	VERIFY ALL PRZ PORVs AND associated block valves properly positioned for current PRZ pressure and plant conditions. **(SHUTS 1RC-114)** **RNO** IF ANY PRZ PORV will NOT shut when required, THEN SHUT its associated block valve	(NO) **(N/A)**
	RO	CHECK BOTH PRZ Spray Valves properly positioned for current PRZ pressure and plant conditions.	(NO)

Op Test No.: NRC	Scenario # 4 Event # 6	Page 43 of 70
Event Description:	**PT-444 Fails HIGH (AOP-019)**	

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td>Critical Task # 2
Immediate Action</td><td></td><td>RNO
CONTROL PRZ spray valves using ONE of the following methods (listed in order of preference):
• AFFECTED Spray Valve controller in MANUAL (if only one is obviously malfunctioning)
OR
• PK-444A, Master Pressure Controller
(Manually Controls PK-444A to restore pressure)
OR
• Both individual spray valve controllers

<i>Critical Task: Maintain control of PRZ Pressure above 1960 psig to prevent an automatic Reactor trip after the pressure transmitter PT-444 fails high.</i></td><td></td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">ENTERS and directs actions of AOP-019,
Conducts a Crew Update
Makes PA announcement for AOP entry</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">• Directs RO to maintain PRZ Pressure controlling band of 2210 to 2260 PSIG per OMM-001 attachment 11.
<table>
<tr><th rowspan="2">Controller</th><th rowspan="2">Control Band</th><th colspan="2">Administrative Limit</th></tr>
<tr><th>Low</th><th>High</th></tr>
<tr><td>Pressurizer Pressure</td><td>2210 – 2260 PSIG</td><td>2050 PSIG</td><td>2350 PSIG</td></tr>
</table></td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">GO TO Section 3.1, Pressure Control Malfunctions While Operating With a Pressurizer Bubble.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Procedure Note:</td><td colspan="2">Loss of RCS pressure control may require initiation of the SHNPP Emergency Plan.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">REFER TO PEP-110, Emergency Classification And Protective Action Recommendations, AND ENTER the EAL Matrix.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td colspan="2">MONITOR PRZ pressure by observing other reliable indication.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
</table>

Op Test No.: NRC Scenario # 4 Event # 6 Page 44 of 70

Event Description: **PT-444 Fails HIGH (AOP-019)**

Time	Position	Applicant's Actions or Behavior	
	SRO	CHECK plant in MODE 1 OR 2.	(YES)
Evaluator Note:		**ERFIS Quick Plot "ITREND" can be used to monitor this parameter.**	
	RO	CHECK PRZ pressure CONTROLLED.	(YES)
		CHECK PRZ pressure 2335 PSIG OR LESS.	(YES)
Procedure Note:		• If PT-445 is failed low, normal plant operation is not affected. However, PORVs 1RC-118 (PCV-445A SA) and 1RC-116 (PCV-445B) will NOT open on high PRZ pressure when in AUTO. • Auto actuation is NOT required for PORV operability.	
	RO	CHECK ALL of the following PRZ PORV block valves OPEN: • 1RC-117 (for PCV-445A SA) • 1RC-115 (for PCV-445B) • 1RC-113 (for PCV-444B SB)	(NO) (YES) (YES)
Procedure Note:		• Attachment 2 lists the controller outputs corresponding to heater, spray, and PRZ PORV operation that are applicable during normal operation.	
	RO	CHECK that a malfunction of one or more of the following has occurred: • PT-444 • PK-444A • PRZ heater(s) • PRZ spray valve(s) or controller(s)	(YES) (NO) (NO) (NO)
	RO	• CHECK PK-444A controlling properly in AUTO.	(NO)
	RO	RNO: PERFORM the following: • VERIFY PK-444A in MANUAL • ADJUST PK-444A output as necessary, to attempt to restore and maintain PRZ pressure.	(YES)

Op Test No.: NRC	Scenario # 4	Event # 6	Page	45 of 70

Event Description: **PT-444 Fails HIGH (AOP-019)**

Time	Position	Applicant's Actions or Behavior	
	RO	CONTROL PRZ pressure as follows:	
Procedure Note:		If individual spray valve controllers are already in MAN, do NOT return to AUTO.	
	RO	CHECK BOTH PRZ spray valve controllers in AUTO AND BOTH spray valves operating as desired.	(YES)
Procedure Note:		Cycling a heater control switch to OFF and back to AUTO will restore normal heater function if the anti-pumping circuit has disabled the heater.	
	RO	CHECK ALL PRZ heaters operating as desired.	(YES)
	RO	CHECK at least one of the following conditions present: • PRZ pressure is UNCONTROLLED • Status of a normal spray valve or a PRZ heater bank is UNCONTROLLED	(YES)
	SRO	RNO: GO TO Step 22.	
	SRO	REFER TO Tech Spec 3.2.5 AND IMPLEMENT action where appropriate. (DNB Parameters, Limit is 2185 psig – restore within 2 hours)	
	SRO	PERFORM the following: • REFER TO Attachment 3, Pressure Control Malfunction Symptoms—Bubble in Pressurizer. • DIRECT Maintenance to investigate and repair the PRZ Pressure Control System component malfunction	

Op Test No.: NRC	Scenario # 4	Event # 6	Page	46 of 70

Event Description: **PT-444 Fails HIGH (AOP-019)**

Time	Position	Applicant's Actions or Behavior
Simulator Communicator:		**Respond to crew requests.**
	SRO	Contacts WCC for support, requests WR and LCOTR. Contacts I&C to have channel removed from service.
Examiner Note:		**After the TS have been identified and the plant has stabilized, cue Simulator Operator to insert Trigger 7** **Event 7: Loss of Offsite Power with a Feed line break inside CNMT (EOP-E-0).**

Op Test No.: NRC	Scenario # 4	Event # 7		Page 47 of 70		
Event Description:	**Loss of Offsite Power with a Feed line break inside CNMT (EOP-E-0)**					
Time	Position	Applicant's Actions or Behavior				

Evaluator Note:		**A Loss of Offsite power will occur coincident with a Feedline Break inside Containment from the 'A' SG. The loss of power to the RCPs will result in an automatic trip of the Reactor and the Feedline Break will result in an auto actuation of SI requiring entry into EOP-E-0. The crew will initiate a MSL Isolation. The crew should diagnose that a LOCA is NOT in progress. The TDAFW pump will trip immediately after starting and four (4) minutes after the reactor trips the 'A' MDAFW Pump will trip requiring the crew to transition to EOP-FR-H.1, Response To Loss Of Secondary Heat Sink.**
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 7** **"Loss of Offsite Power with a Feed line break inside CNMT (EOP-E-0) "**
Indications Available:		• **Multiple alarms due to a Reactor trip** • **Containment press/temp and humidity rising** • **Containment Sump level rising** • **Momentary loss of MCR lighting**
	CREW	Identifies re-entry conditions to EOP-E-0, Reactor Trip Or Safety Injection are met
EOP-E-0		Reactor Trip Or Safety Injection
	SRO	Enters EOP-E-0 Holds crew update
	RO/BOP	Performs E-0 Immediate Actions.

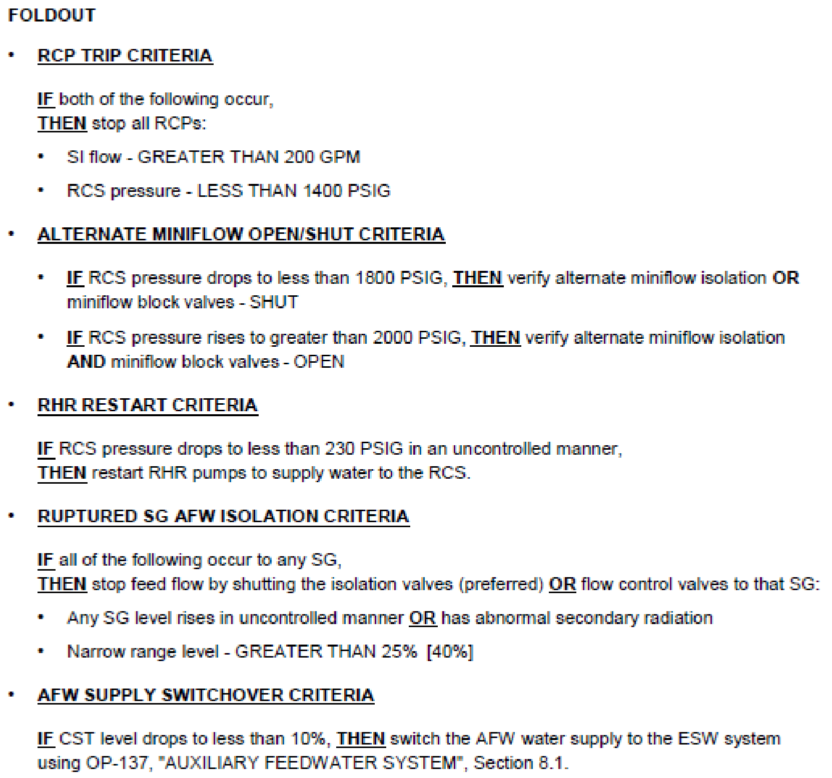
Op Test No.:	NRC	Scenario #	4	Event #	7	Page	48	of	70

Event Description: **Loss of Offsite Power with a Feed line break inside CNMT (EOP-E-0)**

Time	Position	Applicant's Actions or Behavior

<table>
<tr><td>Immediate Actions</td><td>RO</td><td>VERIFY Reactor Trip:<table><tr><td>REACTOR TRIP CONFIRMATION</td></tr><tr><td>Reactor Trip <u>AND</u> Bypass BKRs - OPEN
Rod Bottom Lights (Zero Steps) - LIT
Neutron Flux - DROPPING</td></tr></table></td><td>(YES)
(YES)
(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td>Immediate Actions</td><td>BOP</td><td>Check Turbine Trip – ALL THROTTLE VALVES SHUT<table><tr><td>TURB STOP VLV 1</td><td>TSLB-2-11-1</td></tr><tr><td>TURB STOP VLV 2</td><td>TSLB-2-11-2</td></tr><tr><td>TURB STOP VLV 3</td><td>TSLB-2-11-3</td></tr><tr><td>TURB STOP VLV 4</td><td>TSLB-2-11-4</td></tr></table></td><td>(YES)
(YES)
(YES)
(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td>Immediate Actions</td><td>BOP</td><td>Perform The Following:
AC Emergency Buses – AT LEAST ONE ENERGIZED
AC Emergency Buses – BOTH ENERGIZED</td><td>(YES)
(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td>Immediate Actions</td><td>RO</td><td>Safety Injection – ACTUCATED (BOTH TRAINS)<table><tr><td>BPLP 4-1,"SI ACTUATED" - LIT (CONTINUOUSLY)</td></tr></table></td><td>(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td colspan="2">Evaluator Note:</td><td colspan="2">The Main Feedwater Pumps will lose power when Off-site power is lost. The TD AFW Pump will trip once the turbine comes up to speed. The crew should identify the trip by the following annunciator:
ALB-017-7-3, Aux Feedwater Pump Turbine Trip</td></tr>
</table>

Op Test No.:	NRC	Scenario #	4	Event #	7	Page	49	of 70
Event Description:	**Loss of Offsite Power with a Feed line break inside CNMT (EOP-E-0)**							

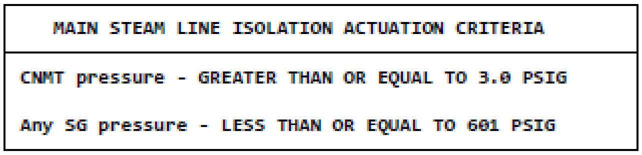
Time	Position	Applicant's Actions or Behavior
Simulator Communicator:		**IF contacted to investigate the cause of the TDAFW pump trip report the mechanical overspeed leakage is damaged and will not reset. No other sign of damage at the pump.** **WHEN / IF WCC is contacted report that Mechanical Maintenance is investigating the damage and that repairs will be made as quickly as possible.** **IF asked about the "B" MD AFW pump status report that it is still waiting on parts to complete emergent repairs.**
Procedure Note:		Steps 1 through 4 are immediate action steps Foldout applies. (Immediate actions should be completed prior implementing Foldout Page items.)
	SRO	Reviews Foldout page
Evaluator Note:		 **FOLDOUT** • **RCP TRIP CRITERIA** **IF** both of the following occur, **THEN** stop all RCPs: • SI flow - GREATER THAN 200 GPM • RCS pressure - LESS THAN 1400 PSIG • **ALTERNATE MINIFLOW OPEN/SHUT CRITERIA** • **IF** RCS pressure drops to less than 1800 PSIG, **THEN** verify alternate miniflow isolation **OR** miniflow block valves - SHUT • **IF** RCS pressure rises to greater than 2000 PSIG, **THEN** verify alternate miniflow isolation **AND** miniflow block valves - OPEN • **RHR RESTART CRITERIA** **IF** RCS pressure drops to less than 230 PSIG in an uncontrolled manner, **THEN** restart RHR pumps to supply water to the RCS. • **RUPTURED SG AFW ISOLATION CRITERIA** **IF** all of the following occur to any SG, **THEN** stop feed flow by shutting the isolation valves (preferred) **OR** flow control valves to that SG: • Any SG level rises in uncontrolled manner **OR** has abnormal secondary radiation • Narrow range level - GREATER THAN 25% [40%] • **AFW SUPPLY SWITCHOVER CRITERIA** **IF** CST level drops to less than 10%, **THEN** switch the AFW water supply to the ESW system using OP-137, "AUXILIARY FEEDWATER SYSTEM", Section 8.1.

Op Test No.:	NRC	Scenario #	4	Event #	7	Page	50 of 70
Event Description:	**Loss of Offsite Power with a Feed line break inside CNMT (EOP-E-0)**						

Time	Position	Applicant's Actions or Behavior	
	SRO	Assigns Foldout items: Alternate Miniflow Open/Shut Criteria, RHR Restart Criteria, AFW Supply Switchover Criteria Directs Shift Manager to Evaluate EAL Matrix (Refer to PEP-110)	
	RO	Ensure CSIPs – ALL RUNNING	(YES)
	RO	Ensure RHR pumps – ALL RUNNING	(YES)
	RO	Safety Injection flow – GREATER THAN 200 GPM	(YES)
	RO	RCS pressure – LESS THAN 230 PSIG	(NO)
	SRO	RNO: GO TO Step 12.	
	BOP	MAIN Steam isolation – ACTUATED.	(NO)
	SRO	RNO: Perform the following:	

Op Test No.:	NRC	Scenario #	4	Event #	8	Page	51	of	70
Event Description:				**Main Steam line Isolation fails**					

Time	Position	Applicant's Actions or Behavior	

Event 8	BOP	Check MAIN Steam isolation – REQUIRED
		
		MAIN STEAM LINE ISOLATION ACTUATION CRITERIA
		CNMT pressure - GREATER THAN OR EQUAL TO 3.0 PSIG
		Any SG pressure - LESS THAN OR EQUAL TO 601 PSIG
		• IF Main Steam Isolation is required THEN perform the following:
		  • Manually actuate Main Steam Line Isolation.
		  • Go to Step 13.
		Identifies that the MSLI did not automatically actuate and attempts to manually from the MCB.
		(Manually actuation of MSLI from MCB switch fails)
			(YES)

Event 8	BOP	Ensure All MSIVs AND Bypass Valves – SHUT Identifies that the MSIV's are not shut and attempts to manually shut by placing MCB in Shut. **(MSIVs fail to close from the MCB)**	(NO)
	BOP	Any SG pressure - 100 PSIG LOWER THAN PRESSURE IN TWO OTHER SGs	(NO)
	SRO	RNO: GO TO Step 16.	
	RO	CHECK CNMT Pressure – HAS REMAINED LESS THAN 10 PSIG	(NO)
	SRO	RNO: Perform the following:	
	BOP	• Ensure CNMT spray – ACTUATED • Stop all RCPs	(YES) (YES)

Op Test No.:	NRC	Scenario #	4	Event #	7	Page	52	of	70
Event Description:	**Loss of Offsite Power with a Feed line break inside CNMT (EOP-E-0) Continued**								

Time	Position	Applicant's Actions or Behavior	
Evaluator Note:		**Depending on the pace of the crew the four minutes timer for the trip of the 'A' MDAFW Pump may have elapsed. Evaluation of AFW flow is a continuous action step and once the time has elapsed and the pump trips the crew should return to this step (EOP-E-0, Step 17).** **The following steps assume the 'A' MDAFW Pump has tripped and will transition the crew to EOP-FR-H.1.** **The crew should identify the trip by the following annunciator:** **ALB-017-5-4, Aux Feedwater Pump A Trip or Close Ckt Trouble**	
Simulator Communicator:		**IF contacted to investigate the cause of the 'A' MDAFW pump trip report the breaker is tripped on overcurrent. No signs of damage at the pumps.** **WHEN / IF WCC is contacted report that Electrical Maintenance is investigating the breaker and that repairs will be made as quickly as possible.**	
	BOP	Ensure AFW flow – AT LEAST 200 KPPH ESTABLISHED	(NO)
	SRO	RNO: Perform the following:	

	BOP	• IF any SG level greater than 25% [40%], THEN go to Step 18.
		• IF no SG level greater than 25% [40%], THEN perform the following:
		  • Manually start AFW pumps
		  • Ensure AFW valves - PROPERLY ALIGNED
		(Manually alignment of the AFW system is not successful)
			(NO)
			(NO)

	SRO	• IF at least 200 KPPH can NOT be established THEN perform the following:	(NO)

Op Test No.: NRC	Scenario # 4	Event # 8		Page 53 of 70
Event Description:	**'B' CCW Pump fails to auto start on SI**			
Time	Position	Applicant's Actions or Behavior		

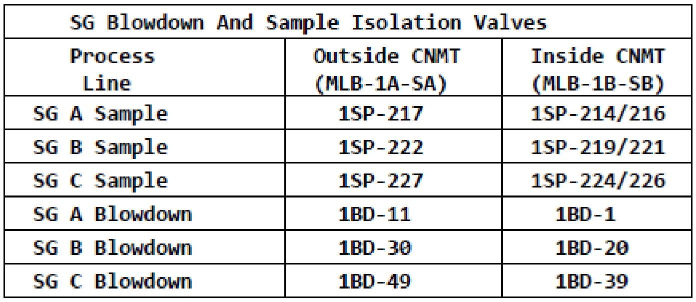
Evaluator Note:		**The RO will perform all board actions until the BOP completes Attachment 3. The BOP is permitted to properly align plant equipment in accordance with Attachment 3 without SRO approval. The Scenario Guide still identifies tasks by board position because the time frame for completion of Attachment 3 is not predictable.** **To follow BOP actions E-0 Attachment 3 is located in the back of this guide.**
	BOP	• Ensure alignment of components from actuation of ESFAS Signals Attachment 3, "Safeguards Actuation Verification", while continuing with implementation of EOPs.
Event 8	BOP	Ensure Two CCW Pumps – RUNNING Identifies that the 'B' CCW Pump is NOT running and manually starts pump.
	BOP	Directs TB AO – Place air compressor 1A and 1B in the Local Control mode. Directs RAB AO – Locally unlock and turn on the breakers for the CSIP Suction and Discharge Cross-Connect valves
Simulator Operator:		**When contacted to place A/B air compressors in Local Control mode, run CAEP :\air\ACs_to_local.txt.**
Simulator Communicator:		**When CAEP is complete, report that the air compressors are running in local control mode.**
Simulator Operator:		**When contacted to Unlock and Turn ON the breakers for the CSIP suction and discharge cross-connect valves, run CAEP *:*\cvc\E-0 Att 2 CSIP suct & disc valve power.txt.**
Simulator Communicator:		**When the CAEP is complete, report task to the MCR.**

Op Test No.:	NRC	Scenario #	4	Event #	9	Page	54	of 70
Event Description:	**'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1)**							

Time	Position	Applicant's Actions or Behavior
	SRO	- Go to FR-H.1, "RESPONSE TO LOSS OF SECONDARY HEAT SINK", Step 1.
EOP-FR-H.1		EOP-FR-H.1, Response To Loss Of Secondary Heat Sink
Procedure Caution:		- This procedure should NOT be performed if total feed flow capability of 200 KPPH is available and total feed flow has been reduced due to operator action as directed by the EOPs. (The following EOPs direct feed flow to be reduced below 200 KPPH: ECA-2.1, "UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS" FR-S.1, "RESPONSE TO NUCLEAR POWER GENERATION/ATWS" FR-P.1, "RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK" FR-P.2, "RESPONSE TO ANTICIPATED PRESSURIZED THERMAL SHOCK" FR-Z.1, "RESPONSE TO HIGH CONTAINMENT PRESSURE") - Feed flow should NOT be established to any faulted SG while a non-faulted SG is available.
	SRO	Perform The Following:
		- Initiate Monitoring Of Critical Safety Function Status Trees - Directs Shift Manager to Evaluate EAL Matrix
	SRO	Check Secondary Heat Sink Requirements:

Op Test No.: NRC Scenario # 4 Event # 9 Page 55 of 70

Event Description: **'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1)**

Time	Position	Applicant's Actions or Behavior	
	RO	• RCS pressure - GREATER THAN ANY NON-FAULTED SG PRESSURE • RCS temperature – GREATER THAN 350°F [330°F] • Stop any running RHR pumps.	(YES) (YES)
	SRO	Check If Bleed And Feed Is Required:	
		• SG wide range levels - ANY TWO LESS THAN 15% [30%]	(NO)
	SRO	RNO: Perform the following:	
		• Observe NOTE prior to Step 4 and go to Step 4.	
Procedure Note:		**Foldout applies.**	
	SRO	Assigns Foldout items: RCS Bleed and Feed Initiation Criteria, Cold Leg Recirculation Switchover Criteria, AFW Supply Switchover Criteria and RHR Restart Criteria Check If Bleed And Feed Is Required:	
	BOP	• Check SG blowdown AND SG sample isolation valves in table – SHUT 	(YES)

SG Blowdown And Sample Isolation Valves

Process Line	Outside CNMT (MLB-1A-SA)	Inside CNMT (MLB-1B-SB)
SG A Sample	1SP-217	1SP-214/216
SG B Sample	1SP-222	1SP-219/221
SG C Sample	1SP-227	1SP-224/226
SG A Blowdown	1BD-11	1BD-1
SG B Blowdown	1BD-30	1BD-20
SG C Blowdown	1BD-49	1BD-39

Op Test No.: NRC	Scenario # 4	Event # 9	Page 56 of 70			
Event Description:	**'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1)**					

Time	Position	Applicant's Actions or Behavior	
	SRO	ESTABLISH AFW Flow to at least ONE SG:	
	BOP	• OBSERVE MCB indications to determine cause of AFW failure: ○ CST level ○ MDAFW pump power supplies ○ TDAFW pump steam supply valves ○ TDAFW pump speed controller ○ TDAFW pump control power ○ AFW valve alignment	(NO) (YES) (YES) (NO) (NO) (NO)
		• TRY to restore AFW flow at the MCB. (Refer to EOP-FR-H.1 Attachment 1 for guidance of rate of feed flow.) (Refer to OP-137, Auxiliary Feedwater System, for guidance regarding AFW pump operations, precautions and limitations and valve operation.)	
	CREW	Contacts AO's to investigate failures	
Simulator Communicator:		**During the remainder of the scenario any communications for a request to restore MFW or AFW** **Maintenance is looking at the situation and will make repairs as soon as they can.** **When ANY pump is available the WCC will contact the MCR.**	
	SRO	Check If AFW Flow Established:	
		• Total feed flow to SGs – GREATER THAN 200 KPPH	(NO)
	SRO	RNO: Go to Step 6c.	
	SRO	Check AFW flow - ESTABLISHED TO ANY SG	(NO)

Op Test No.:	NRC	Scenario #	4	Event #	9	Page	57	of 70
Event Description:	**'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1)**							

Time	Position	Applicant's Actions or Behavior	
	SRO	RNO: Perform the following:	
		• Continue attempts to restore AFW flow at the MCB. • TRY to restore AFW flow locally. (Refer to OP-137, Auxiliary Feedwater System, for guidance regarding AFW pump operations, precautions and limitations and valve operation.) • Observe NOTE prior to Step 7 and continue with Step 7.	
Procedure Note:		**After stopping all RCPs and placing steam dump in the steam pressure mode, RCS pressure and temperature will rise as natural circulation is established. A large loop ΔT prior to PRZ PORV opening confirms natural circulation.**	
	SRO	Stop Heat Input From RCP Operations:	
		• Stop All RCPs. • Check steam dump to condenser - AVAILABLE:	(YES) (NO)
	SRO	RNO: Use intact SG(s) PORV for steam dumping in subsequent steps.	
		• Go to Step 8.	
	RO	CHECK SI - ACTUATED	(YES)
	SRO	Perform The Following To Verify Proper Sequencer And Component Operations While Continuing With This Procedure:	
	RO	• Sequencer Load Block 9 (Manual Loading Permissive) - ACTUATED (BOTH TRAINS) • Energize AC buses 1A1 AND 1B1	(YES)
	SRO	• Ensure Automatic Actions From SI Actuation While Continuing With This Procedure. • (Refer to E-0, "REACTOR TRIP OR SAFETY INJECTION", Attachment 3.)	

Op Test No.: NRC Scenario # 4 Event # 9 Page 58 of 70

Event Description: **'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1)**

Time	Position	Applicant's Actions or Behavior	
Procedure Caution:		**SI reset can NOT occur until sixty seconds after SI signal actuation.**	
	RO	- Reset SI	
	SRO	- Manually Realign Safeguards Equipment Following A Loss Of Offsite Power. - (Refer to E-0, "REACTOR TRIP OR SAFETY INJECTION", Attachment 6.)	

	RO	- Reset Phase A
		- Open Instrument Air AND Nitrogen Valves To CNMT:
		`1IA-819 (ISOL VALVE CONT. BLDG 236' PENETRATION (M-80))`
		`1SI-287 (ACCUMULATOR & PRZ PORV N2 SUPPLY ISO VLV)`	

	SRO	Establish Main FW Flow To At Least One SG:	
		- Check condensate system – IN SERVICE	(NO)
	SRO	RNO: Place condensate system in service. (Refer to OP-134, "CONDENSATE SYSTEM", Section 5.0.)	
		- IF condensate system can NOT be placed in service, THEN go to Step 16.	(NO)
Simulator Communicator:		**If contacted by the by the crew for a time for the return of Offsite Power acknowledge the request and report that Offsite Power to the Harris should be restored within 4 hour.**	
Procedure Note:		**The EDMP should NOT be used unless other sources are unavailable.**	

Op Test No.:	NRC	Scenario # 4	Event # 9	Page	59 of 70
Event Description:	**'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1)**				

Time	Position	Applicant's Actions or Behavior	
	SRO	Prepare To Depressurize Two SGs:	
		• Identify 2 SGs to be fed.	(YES)
	BOP	• Shut the following valves for the SG that is NOT to be fed. ○ MSIVs ○ MSIV bypass valves ○ SG main steam drain isolations before MSIV:	(NO) (YES) (YES)
	SRO	RNO: Shut the following valves for the SGs to be fed.	
		○ MSIVs ○ MSIV bypass valves ○ SG main steam drain isolations before MSIV:	(NO) (YES) (YES)
	SRO	Align EDMP to SGs as follows:	
	BOP	Direct local installation of connections/hoses using ISG-HS, "HEAT SINK", Attachment 5 Steps 3 through 7.	
		Contacts AO's to perform ISG-HS task	
Simulator Communicator:		**Acknowledge request**	
	SRO	Check local installation - COMPLETE	(NO)
		RNO: WHEN local installation of connection/hoses is complete, THEN go to Step 16.c.3. • Continue with Step 19.	
	SRO	Check For Loss Of Secondary Heat Sink:	
		• SG wide range levels - ANY TWO LESS THAN 15% [30%]	(NO)
		RNO: Return to Step 1.	

Op Test No.: NRC	Scenario # 4	Event # 9	Page	60	of	70
Event Description:	**'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1)**					

Time	Position	Applicant's Actions or Behavior	
Evaluator Note:		**The SRO will loop back to the beginning of the procedure and evaluate the status of infield actions and foldout criteria until the RCS Bleed and Feed Initiation Criteria is met at which time the crew will continue EOP-FR-H.1 returning to step 20.**	
Procedure Caution:		**Perform Steps 20 through 30 without delay to establish RCS heat removal by RCS bleed and feed.**	
	RO	Actuate Safety Injection.	
	SRO	Ensure RCS Feed Path:	
Critical Task #3	RO	• SI flow - GREATER THAN 200 GPM • Check CSIPs - BOTH RUNNING • Observe NOTE prior to Step 23 and go to Step 23. *Critical to initiate RCS Bleed and Feed for Successful High-Head SI Pump Injection before RCS temperature rises above 730°F and RVLIS Full Range Level lowers below 39%*	(YES) (YES)
Procedure Note:		**SI reset can NOT occur until sixty seconds after SI signal actuation.**	
	RO	Reset SI	
	SRO	Manually Realign Safeguards Equipment Following A Loss Of Offsite Power. (Refer to E-0, "REACTOR TRIP OR SAFETY INJECTION", Attachment 6.)	
	SRO	Reset Phase A AND Phase B Isolation:	
	RO	• Reset Phase A (if actuated) • Reset Phase B (if actuated)	(YES) (YES)

Op Test No.:	NRC	Scenario #	4	Event #	9	Page	61 of 70
Event Description:	**'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1)**						

Time	Position	Applicant's Actions or Behavior	
	SRO	Check Sequencers - RESET (BOTH TRAINS)	(NO)
		RNO: For any Sequencer that is NOT reset, perform the following:	
Procedure Note:		**Manual actuation of Load Block 9 cannot occur for 150 SECONDS after sequencer operation.**	
	BOP	• Check Sequencer Load Block 9 (Manual Loading Permissive) - ACTUATED	(YES)
		Energize AC buses 1A1 AND 1B1	(YES)
	RO	Open Instrument Air AND Nitrogen Valves To CNMT:	(YES)

```
1IA-819 (ISOL VALVE CONT. BLDG
        236' PENETRATION (M-80))

1SI-287 (ACCUMULATOR & PRZ PORV
        N2 SUPPLY ISO VLV)
```

Op Test No.: NRC Scenario # 4 Event # 10 Page 62 of 70

Event Description: **Pressurizer PORV 445B fails to open**

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>SRO</td><td colspan="2">Establish RCS Bleed Path:</td></tr>
<tr><td></td><td>RO</td><td>• Establish ALL RCS bleed paths listed in table by performing the following:
○ Ensure PRZ PORV Block
○ Open all PRZ PORVs (safety and non-safety regardless of operability status).
<table>
<tr><th colspan="3">RCS Bleed Paths Based On PRZ PORV AND Associated Block Valve</th></tr>
<tr><th>Bleed Path</th><th>Block Valve</th><th>PRZ PORV</th></tr>
<tr><td>"A" Train PRZ PORV</td><td>1RC-117</td><td>1RC-118 (PCV-445A SA)</td></tr>
<tr><td>"B" Train PRZ PORV</td><td>1RC-113</td><td>1RC-114 (PCV-444B SB)</td></tr>
<tr><td>Non Safety PRZ PORV</td><td>1RC-115</td><td>1RC-116 (PCV-445B)</td></tr>
</table>
(PRZ PORV 445B (1RC-116) fails to open)</td><td>(YES)
(NO)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">Ensure Adequate RCS Bleed Path:</td></tr>
<tr><td></td><td>RO</td><td>• PRZ PORVs - ALL OPEN
(PRZ PORV 445B (1RC-116) fails to open)
• PRZ PORV block valves – ALL OPEN</td><td>(NO)
(YES)</td></tr>
<tr><td>Critical Task #3</td><td></td><td>RNO: Open all RCS vent valves to commence venting:
• 1RC-900
• 1RC-901
• 1RC-902
• 1RC-903
• 1RC-904
• 1RC-905
<i>Critical to initiate RCS Bleed and Feed for Successful High-Head SI Pump Injection before RCS temperature rises above 730°F and RVLIS Full Range Level lowers below 39%</i></td><td>(YES)
(YES)
(YES)
(YES)
(YES)
(YES)
(YES)</td></tr>
</table>

Op Test No.:	NRC	Scenario #	4	Event #	7	Page	63	of 70
Event Description:	**'A' MDAFW pump trips after the Reactor trips (EOP-FR-H.1) Continued**							

Time	Position	Applicant's Actions or Behavior
	SRO	Ensure Automatic Actions From SI Actuation While Continuing With This Procedure. (Refer to E-0, "REACTOR TRIP OR SAFETY INJECTION", Attachment 3.)
	SRO	Maintain RCS Heat Removal:
		• Maintain SI flow. • Maintain RCS bleed paths.
Lead Evaluator:		**Terminate the scenario after RCS Heat Removal has been established.** **Announce 'Crew Update' - End of Evaluation - I have the shift.** **Have crew remain in the Simulator without discussing the exam. Examiners will formulate any follow-up questions.**

Simulator Operator:	**When directed by Lead Evaluator go to FREEZE**

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 1 of 7
SAFEGUARDS ACTUATION VERIFICATION

NOTE

- General guidance for verification of safeguards equipment is contained in Attachment 4 of this procedure.
- ERFIS displays of safeguards equipment status are not reliable while any associated safety-related electrical buses are de-energized.

☐ 1. **Ensure** Two CSIPs - RUNNING

☐ 2. **Ensure** Two RHR Pumps - RUNNING

☐ 3. **Ensure** Two CCW Pumps - RUNNING

☐ 4. **Ensure** All ESW **AND** ESW Booster Pumps - RUNNING

☐ 5. **Ensure** SI Valves - PROPERLY ALIGNED

(**Refer** to Attachment 1.)

☐ 6. **Ensure** CNMT Phase A Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 4.)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 2 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 7. **Ensure** SG Blowdown **AND** SG Sample Isolation Valves In Table 1 - SHUT

Table 1: SG Blowdown And Sample Isolation Valves

Process Line	Outside CNMT (MLB-1A-SA)	Inside CNMT (MLB-1B-SB)
SG A Sample	1SP-217	1SP-214/216
SG B Sample	1SP-222	1SP-219/221
SG C Sample	1SP-227	1SP-224/226
SG A Blowdown	1BD-11	1BD-1
SG B Blowdown	1BD-30	1BD-20
SG C Blowdown	1BD-49	1BD-39

8. IF Main Steam Line Isolation Actuated OR Is Required By Any Of The Following, THEN **Ensure** MSIVs **AND** MSIV Bypass Valves - SHUT

☐ • Steam line pressure - LESS THAN 601 PSIG

☐ • CNMT pressure - GREATER THAN 3.0 PSIG

9. IF CNMT Spray Actuation Signal Actuated OR Is Required, THEN **Ensure** The Following:

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 9.)

☐ • CNMT spray pumps - RUNNING

☐ • CNMT spray valves - PROPERLY ALIGNED

☐ • Phase B isolation valves - SHUT

☐ • All RCPs - STOPPED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 3 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 10. **Ensure** Both Main FW Pumps - TRIPPED

☐ 11. **Ensure** FW Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 6.)

☐ 12. **Ensure** Both MDAFW pumps - RUNNING

13. __IF__ Any Of The Following Conditions Exist, **__THEN__** **Ensure** The TDAFW Pump - RUNNING

- ☐ Undervoltage on either 6.9 KV emergency bus
- ☐ Level in two SGs - LESS THAN 25%
- ☐ Manual actuation to control SG level

14. **Ensure** AFW Valves - PROPERLY ALIGNED

- ☐ __IF__ no AFW Isolation Signal, **__THEN__** **ensure** isolation **__AND__** flow control valves - OPEN

__NOTE__

An AFW Isolation signal signal requires a Main Steam Line Isolation coincident with one SG pressure 100 PSIG below the other two SGs.

- ☐ __IF__ AFW Isolation Signal present, **__THEN__** **ensure** MDAFW **__AND__** TDAFW isolation **__AND__** flow control valves to affected SG - SHUT

☐ 15. **Ensure** Both EDGs - RUNNING

☐ 16. **Ensure** CNMT Fan Coolers - ONE FAN PER UNIT RUNNING IN SLOW SPEED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 4 of 7
SAFEGUARDS ACTUATION VERIFICATION

☐ 17. **Ensure** CNMT Ventilation Isolation Valves - SHUT

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 7.)

☐ 18. **Ensure** Control Room Area Ventilation - MAIN CONTROL ROOM ALIGNED FOR EMERGENCY OPERATION

(**Refer to** OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 5, Sheets 1 and 2, Sections for MAIN CONTROL BOARD, SLB-5 and SLB-6.)

19. **Ensure** Essential Service Chilled Water System Operation:

☐ • **Ensure** both WC-2 chillers - RUNNING

☐ • **Ensure** both P-4 pumps - RUNNING

☐ (**Refer** to AOP-026, "LOSS OF ESSENTIAL SERVICE CHILLED WATER SYSTEM" for loss of any WC-2 chiller.)

20. **Ensure** CSIP Fan Coolers - RUNNING

☐ AH-9 A SA
☐ AH-9 B SB
☐ AH-10 A SA
☐ AH-10 B SB

NOTE

Security systems are powered by bus 1A1 (normal supply) or bus 1B1 (alternate supply). Backup power will be available for approximately 30 MINUTES after the supplying bus is de-energized. (Refer to OP-115, "CENTRAL ALARM STATION ELECTRICAL SYSTEMS", Section 8.9 and 8.10.)

☐ 21. **Ensure** AC buses 1A1 **AND** 1B1 - ENERGIZED

☐ 22. **Place** Air Compressor 1A **AND** 1B In The LOCAL CONTROL Mode.

(**Refer to** Attachment 7.)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 5 of 7
SAFEGUARDS ACTUATION VERIFICATION

CAUTION

The maximum calculated dose rate in the vicinity of MCC 1A35-SA and MCC 1B35-SB is between 10 MREM/HR and 150 MREM/HR.

☐ 23. Dispatch An Operator To **Unlock** And **Close** The Breakers For The CSIP Suction **AND** Discharge Cross-Connect Valves:

(**Refer** to Attachment 2.)

MCC 1A35-SA		MCC 1B35-SB	
VALVE	CUBICLE	VALVE	CUBICLE
1CS-170	4A	1CS-171	4D
1CS-169	4B	1CS-168	7D
1CS-218	14D	1CS-220	9D
1CS-219	14E	1CS-217	12C

24. **Check** If C CSIP Should Be Placed In Service:

☐ • **IF** two charging pumps can **NOT** be verified to be running, **AND** C CSIP is available, **THEN place** C CSIP in service in place of the non-running CSIP using OP-107, "CHEMICAL AND VOLUME CONTROL SYSTEM, Section 8.5 or 8.7.

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 6 of 7
SAFEGUARDS ACTUATION VERIFICATION

25. **Start** The Spent Fuel Pump Room Ventilation System:

    a. At AEP-1, **ensure** the following ESCWS isolation valves - OPEN

        1) SLB-11 (Train A)

        ☐ • AH-17 SUP CH 100 (Window 9-1)

        ☐ • AH-17 RTN CH 105 (Window 10-1)

        2) SLB-9 (Train B)

        ☐ • AH-17 SUP CH 171 (Window 9-1)

        ☐ • AH-17 RTN CH 182 (Window 10-1)

    b. At AEP-1, **start** one SFP PUMP ROOM FAN COOLER:

    ☐ • AH-17 1-4A SA

    ☐ • AH-17 1-4B SB

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 7 of 7
SAFEGUARDS ACTUATION VERIFICATION

NOTE

- Fuel pool levels and temperatures should be monitored approximately every 1 to 2 HOURS.
- Following the initial check of fuel pool levels and temperature, monitoring responsibilities may be assumed by the plant operations staff (including the TSC or STA).
- Only fuel pools containing fuel are required to be monitored.

26. **Check** Status Of Fuel Pools:

☐ a. **Operate** spent fuel cooling pumps to maintain fuel pool temperatures between 85°F to 105°F.

b. **Monitor** fuel pool levels **AND** temperatures:

- ☐ **Refer to** AOP-041, "SPENT FUEL POOL EVENT" Attachments 7, 8, 9, 10 and 11 for SFP parameter monitoring methods.
- ☐ **Refer to** Curves H-X-24, H-X-25 and H-X-26 for SFP time to 200°F.
- ☐ Levels - GREATER THAN LO ALARM (284 FT, 0 IN)
- ☐ Temperatures - LESS THAN HI TEMP ALARM (105°F)

NOTE

If control room ventilation was previously aligned to an emergency outside air intake for post-accident operations, then follow-up actions will be required to restore the alignment.

27. **Consult** Plant Operations Staff Regarding Alignment Of The Control Room Ventilation System:

- ☐ Site Emergency Coordinator - Control Room
- ☐ Site Emergency Coordinator - Technical Support Center

(Refer to PEP-230, "CONTROL ROOM OPERATIONS".)

- END -

Facility: Harris Nuclear Plant

Task No.: 004055H101

Task Title: BTRS End of Life Dilution Operation (OP-108)

JPM No.: 2020 NRC Exam Simulator JPM a

K/A Reference: 004 A4.07 RO 3.9 SRO 3.7

ALTERNATE PATH – YES

Examinee: ________________________

NRC Examiner: _________________

Facility Evaluator: ________________________

Date: ________

Method of testing:

Simulated Performance: ______

Actual Performance: X

Classroom ______ Simulator X Plant ________

READ TO THE EXAMINEE I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.

Initial Conditions:	• The plant is operating at 100% power End of Life. • RCS temperature is ~0.3°F low. • The BTRS system has been filled and vented • EST-394, ASME System Pressure Test For The CVCS – C/D Thermal Regeneration Demineralizers is not required • EST-702, Moderator Temperature Coefficient - EOL is complete • BTRS is aligned OP-108, Boron Thermal Regeneration System per Attachments 1 and 2 • Chemistry has recommended placing the BTRS Demineralizer B resin bed in service. • All other parameters are normal.

Initiating Cue:	• The CRS has directed you to rinse in BTRS Demineralizer B resin bed for 10 minutes for a chemistry sample using End of Life Dilution Operation per OP-108, Section 8.9. • The initial conditions for the aligning the system are complete. • The RAB AO is standing by to support placing BTRS in service. • OP-108, Section 8.9.2 step 1 and Attachment 3 are complete.

Evaluator Note:	***To reduce student prep time, consider supplying student with a copy of the procedure and pre-briefing student prior to entry into the Simulator.***

Task Standard:
- Verify open the BTRS bypass, 1CS-98 and verify shut the BTRS inlet, 1CS-570 due to improper BTRS valve alignment

Required Materials: None

General References: OP-108, Boron Thermal Regeneration System, Rev. 25

Handout: OP-108, Rev. 25, pages 1- 7, Prerequisites, P&L's
OP-108, Rev. 25, pages 43 – 47, Section, 8.9, End of Life Dilution Operation, **with the Initial Conditions signed off if desired**

Time Critical Task: No

Validation Time: 15 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 5	Critical to open inlet valve to demin to establish correct lineup.
Step 16	Critical to have in correct position to flush the new resin bed prior to initiating dilution flow.
Step 20	Critical to make adjustment in order to initiate dilution flow.
Step 22	Critical to identify that the BTRS has malfunctioned and bypass the BTRS system to prevent a unexpected dilution event and initiate a request for repairs to be made.

2020 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-145
- Password “NRC3sros”
- Go to run
- Silence and Acknowledge annunciators

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner. (The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

To recreate the IC setup for this JPM:

- Initial Simulator IC was IC-20
- Shut 1CS-638 and 1CS-647 to remove the A Demin from service and place BTRS in Standby in accordance with OP-108, Section 8.9.2, step 23.a - d
- Borate the RCS to get approximately - 0.3°F mismatch between Tave and Tref
- Remove the jumper from TB B1494 by lifting the leads using the malfunction below.
    - imf cvc154 (n 00:00:00 00:00:00) LIFTED
- Silence Acknowledge and Reset Annunciators
- Freeze and Snap these conditions to your exam IC

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ____________

OP-108, Section 8.9.1, Initial Conditions

Performance Step: 1 Initial Conditions

1. BTRS aligned per Attachments 1 and 2.
2. ASME Section XI Pressure Testing Program Manager (Engineering) has been notified to perform EST-394, if needed.
3. BTRS filled and vented per Section 8.10.

Standard: Reviews initial cue and determines the initial conditions for starting the system are complete

Comment:

OP-108, Section 8.9.2 Procedure Steps, Step 1

Performance Step: 2 VERIFY Part 1 of Attachment 3 complete.

Standard: Reviews Attachment 3 and determines I&C has installed the required jumpers in Term Box B1494.

Comment:

OP-108, Section 8.9.2 NOTE prior to step 2

Performance Step: 3 NOTE: For End of Life Dilution Operation as many as three beds may be used, one at a time.

Standard: Operator reads and placekeeps note

Comment:

OP-108, Section 8.9.2 Step 2

Performance Step: 4 CONTACT Chemistry, and determine BTRS demin bed to be used for dilution.

Standard: Reviews initial cue and determines the BTRS demin bed to be placed in service is the BTRS Demineralizer B resin bed.

Comment:

Evaluator Cue:	**If the candidate contacts Chemistry confirm the BTRS demin bed to be used is the BTRS Demineralizer B resin bed.**

OP-108, Section 8.9.2 Step 3

✓ **Performance Step: 5** OPEN inlet isolation valve for demin bed to be used.

- 1CS-638 BTRS Demineralizer A Isolation
- **1CS-628 BTRS Demineralizer B Isolation**
- 1CS-618 BTRS Demineralizer C Isolation
- 1CS-608 BTRS Demineralizer D Isolation

Standard: Locates MCB switch for 1CS-628 and places switch to open:

- 1CS-628 BTRS Demineralizer B Isolation - OPEN

Comment:

OP-108, Section 8.9.2 Step 4

Performance Step: 6

VERIFY SHUT inlet isolation valves for demin beds that will NOT be used.

- **1CS-638 BTRS Demineralizer A Isolation**
- 1CS-628 BTRS Demineralizer B Isolation
- **1CS-618 BTRS Demineralizer C Isolation**
- **1CS-608 BTRS Demineralizer D Isolation**

Standard:

Locates MCB switch for 1CS-638 and places switch to shut:

Locates MCB switches for BTRS demin inlet isolation valves and verifies shut (switch position and green light)

- **1CS-638 BTRS Demineralizer A Isolation**
- **1CS-618 BTRS Demineralizer C Isolation**
- **1CS-608 BTRS Demineralizer D Isolation**

Comment:

OP-108, Section 8.9.2 Step 5

Performance Step: 7

OPEN outlet isolation valve for demin bed to be used.

- 1CS-647 1A Demin Lower Isolation Valve
- **1CS-637 1B Demin Lower Isolation Valve**
- 1CS-627 1C Demin Lower Isolation Valve
- 1CS-617 1D Demin Lower Isolation Valve

Standard:

Contacts the RAB AO to open:

- 1CS-637 1B Demin Lower Isolation Valve

Comment:

Simulator Operator Communication:	**Use Sim Drawing CVC\btr01 and OPEN 1CS-637 when open then:** **Report 1CS-637, 1B Demin Lower Isolation Valve is OPEN**

OP-108, Section 8.9.2 Step 6

Performance Step: 8 **VERIFY SHUT** outlet isolation valves for demin beds that will NOT be used.

- **1CS-647 1A Demin Lower Isolation Valve**
- 1CS-637 1B Demin Lower Isolation Valve
- **1CS-627 1C Demin Lower Isolation Valve**
- **1CS-617 1D Demin Lower Isolation Valve**

Standard: Contacts the RAB AO to verify shut:

- **1CS-647 1A Demin Lower Isolation Valve**
- **1CS-627 1C Demin Lower Isolation Valve**
- **1CS-617 1D Demin Lower Isolation Valve**

Comment:

Simulator Operator Communication:	**Wait 1 minute and report** **1CS-647 1A Demin Lower Isolation Valve** **1CS-627 1C Demin Lower Isolation Valve** **1CS-617 1D Demin Lower Isolation Valve** **Are all shut.**

OP-108, Section 8.9.2 Step 7

Performance Step: 9 OPEN 1IA-1221-I2 IA Isol Valve to 1CS-570, BTRS INLET.

Standard: Contacts the RAB AO to open:

- 1IA-1221-I2 IA Isol Valve to 1CS-570, BTRS INLET

Comment:

Simulator Operator Communication:	**1IA-1221-I2 IA Isol Valve to 1CS-570, BTRS INLET is Open**

OP-108, Section 8.9.2 Step 8

Performance Step: 10 OPEN 1CS-669 BTRS Outlet Isolation Valve.

Standard: Contacts the RAB AO to open:

- OPEN 1CS-669 BTRS Outlet Isolation Valve.

Comment:

Simulator Operator Communication:	**Wait 1 minute and report** **1CS-669 BTRS Outlet Isolation Valve is Open**

OP-108, Section 8.9.2 Step 9

Performance Step: 11 POSITION the control switch for 1CS-570, BTRS INLET to AUTO.

Standard: Locates the MCB control switch for 1CS-570 and verifies it is in the AUTO position.

Comment:

OP-108, Section 8.9.2 Step 10

Performance Step: 12 POSITION the control switch for 1CS-98, BTRS BYPASS to AUTO.

Standard: Locates the MCB control switch for 1CS-98 and verifies it is in the AUTO position.

Comment:

OP-108, Section 8.9.2 Step 11

Performance Step: 13 VERIFY HC-387, BTRS DEMIN BYPASS 1CS-606, has a 100% demand signal.

Standard: Locates the MCB control switch for HC-387, BTRS DEMIB BYPASS 1CS-606 and verifies the horizontal demand meter output signal is at 100% demand.

Comment:

OP-108, Section 8.9.2 NOTES prior to Step 12

Performance Step: 14 NOTE: Flowing borated water through the bed and into the RHT for extended periods of time will exhaust the bed prematurely. This can be minimized by limiting the time letdown is diverted to that necessary for Chemistry to obtain a sample and securing flow through the system until the results are obtained.

NOTE: Blockage in BTRS while 1CS-120 is aligned to RHT during flushes will be seen as flow to the VCT. 1CS-47, LD Hx Relief Vlv, relieves to the VCT. This has previously been misdiagnosed as a 1CS-120 issue.

Standard: Operator reads and placekeeps notes

Comment:

OP-108, Section 8.9.2 CAUTION prior to Step 12

Performance Step: 15 CAUTION: Failure to divert letdown to the holdup tank when a new resin bed is being placed in service could result in a change in RCS chemistry.

Standard: Operator reads and placekeeps caution

Comment:

OP-108, Section 8.9.2 Step 12

✓ **Performance Step: 16**

IF any, one (1) of the following is true:

- BTRS has been shut down for greater than 30 days,

OR

- A new BTRS resin bed has been placed in service,

OR

- Fill and vent has been performed,

THEN PLACE 1CS-120, LETDOWN TO VCT/HOLD UP TANK LCV-115A, to the RHT position.

Standard:

Reviews initiating cues and determines that the condition “a new BTRS resin bed has been placed in service” is true and places 1CS-120, LETDOWN TO VCT/HOLD UP TANK LCV-115A, to the RHT position.

Comment:

OP-108, Section 8.9.2 Step 13

Performance Step: 17

IF flow was diverted to the RHT is Step 8.9.2.12, THEN NOTIFY the RMS Tech that flushing operations are in progress and will lower VCT level. This will increase radiation levels in the room.

Standard:

Contacts the RMS Tech and notifies them of the flushing operations per the note.

Comment:

Simulator Operator Communication:	**Acknowledges flushing operations are in progress.**

Evaluator Cue:	**If an Auto makeup of the Reactor Water Makeup system occurs cue the candidate that the another operator will monitor the Auto makeup for proper operation.**

OP-108, Section 8.9.2 Step 14

Performance Step: 18 PERFORM the following steps:

- POSITION the BTRS FUNCTION SELECTOR switch to DIL position.
- VERIFY that the white DIL light is NOT LIT.

Standard: Locates the control switch for the BTRS FUNCTION SELECTOR, and verifies the switch in the DIL position and verifies the white DIL light is not lit.

Comment:

OP-108, Section 8.9.2 NOTE prior to Step 15

Performance Step: 19 NOTE: HC-387 operates the opposite of what may be expected. To move the output from right to left (100% to 0%), HC-387 must be rotated to the right (clockwise). Operating Experience shows the white DIL light comes on at about 70% output of HC-387 and goes off at about 100% output.

Standard: Operator reads and placekeeps note

Comment:

OP-108, Section 8.9.2 Step 15

✓ **Performance Step: 20** PERFORM the following steps:

- POSITION HC-387, BTRS DEMIN BYPASS 1CS-606, at a less than 100% demand signal.
- VERIFY that the WHITE DIL light illuminates.

Standard: Locates the control switch for HC-387, BTRS DEMIN BYPASS 1CS-606, reduces the demand signal to less than 100% and determines the WHITE DIL light **does NOT** illuminates.

Comment:

OP-108, Section 8.9.2 NOTE prior to Step 16

Performance Step: 21 NOTE: If the White DIL light is not illuminated, an improper BTRS valve lineup may be the cause, and a loss of letdown flow may occur.

Standard: Operator reads and placekeeps note

Comment:

OP-108, Section 8.9.2 Step 17 – Alternate Path Begins Here

✓ **Performance Step: 22** IF the DIL light does not illuminate, PERFORM the following steps:

- VERIFY OPEN 1CS-98, BTRS BYPASS.
- VERIFY SHUT 1CS-570, BTRS INLET.
- INITIATE a work request to have the BTRS repaired.

Standard:

- Locates the control switch for 1CS-98, BTRS BYPASS and takes control switch to OPEN.
- Locates the control switch for 1CS-570, BTRS INLET and takes control switch to SHUT.
- Notifies the CRS to initiate a work request to repair the BTRS system

Comment:

Evaluator Cue:	**When the CRS is notified that a work request needs to be initiated to repair the BTRS system. Evaluation on this JPM is complete.** **Direct Simulator Operator to place the Simulator in Freeze.**

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

STOP TIME: ______________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam JPM CR a
BTRS End of Life Dilution Operation

In accordance with OP-108, Boron Thermal Regeneration System

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	• The plant is operating at 100% power End of Life • RCS temperature is ~0.3°F low • The BTRS system has been filled and vented • EST-394, ASME System Pressure Test For The CVCS – C/D Thermal Regeneration Demineralizers is not required • EST-702, Moderator Temperature Coefficient - EOL is complete • BTRS is aligned OP-108, Boron Thermal Regeneration System per Attachments 1 and 2 • Chemistry has recommended placing the BTRS Demineralizer B resin bed in service • All other parameters are normal

Initiating Cue:	• The CRS has directed you to rinse in BTRS Demineralizer B resin bed for 10 minutes for a chemistry sample using End of Life Dilution Operation per OP-108, Section 8.9 • The initial conditions for the aligning the system are complete • The RAB AO is standing by to support placing BTRS in service • OP-108, Section 8.9.2 step 1 and Attachment 3 are complete

Facility:	Harris Nuclear Plant	Task No.:	004016H101
Task Title:	Place Excess Letdown In Service	JPM No.:	2020 HNP NRC Exam Simulator JPM CR b
K/A Reference:	004 A4.06 3.6 RO 3.1 SRO	**ALTERNATE PATH - NO**	

Examinee: ________________________ NRC Examiner: __________________

Facility Evaluator: ________________________ Date: ________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom ______ Simulator X Plant ________

READ TO THE EXAMINEE

I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.

Initial Conditions:	- The unit is operating at 100% power MOL - Normal letdown needs to be secured for maintenance due to a problem with PCV-145 - PCV-145 is in manual

Initiating Cue:	- You are the OATC and have been directed by the CRS to establish Excess Letdown to the VCT per OP-107, Section 8.2. - Excess letdown has not been in service during this refueling cycle

Evaluator Note:	**The candidates should be briefed outside of the Simulator prior to performing this JPM. Provide them with a copy of the procedure and inform them that ALL initial conditions are satisfied.** **This will allow them to review the Precautions and Limitations associated with OP-107 and have time for a task preview of the steps to accomplish establishing Excess Letdown. Expect that the candidates will take about 10-15 minutes to complete this review.**

Task Standard: Excess letdown is established with proper flow and temperature

Required Materials: None

General References: OP-107, Rev. 117

Handout: OP-107, Rev. 117, pages 1 – 17, Prerequisites, P&L's
OP-107, Rev. 117, pages 46 – 52, Section 8.2, Excess Letdown Heat Exchanger Operation

Time Critical Task: No

Validation Time: 15 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 12	Excess Letdown flow cannot be established if 1CS-466, EXCESS LETDOWN TO VCT/RCDT, is NOT positioned to the RCDT.
Step 13	Excess Letdown flow cannot be established if 1CS-461, EXCESS LETDOWN valve is NOT opened.
Step 14	Excess Letdown flow cannot be established if 1CS-460, EXCESS LETDOWN valve is NOT opened
Step 17	Exceeding procedural parameters limits for outlet temperatures or pressure could damage the Excess Letdown Heat Exchanger OR the RCDT for this flow path.
Step 19	Exceeding procedural parameters limits for outlet temperatures or pressure could damage the Excess Letdown Heat Exchanger and for this flow path the excess pressure would go to the RCDT.
Step 22	Exceeding procedural parameters limits for outlet temperatures or pressure could damage the Excess Letdown Heat Exchanger and for this flow path the high pressure will lift the Letdown relief which discharges to the PRT.

2020 NRC Exam JPM b - SIMULATOR SETUP

Simulator Operator

- IC will be saved in IC-151 once the Audit Exam is completed (Reset to IC-151)
- IC was not saved on the simulator (Password "NRC3sros")
- Place RED Off Normal placard on PCV-145
- Go to RUN
- Silence and Acknowledge annunciators

GO TO FREEZE and inform the lead examiner the Simulator is ready. **DO NOT GO TO RUN** until directed by the lead examiner. (The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

To recreate the IC setup for this JPM:

- Initial Simulator IC was IC-19
- Initialize to IC-19, go to RUN
- Place PCV-145 in manual
- Silence Acknowledge and Reset Annunciators
- FREEZE and Snap these conditions to your exam IC

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ____________

Performance Step: 1 **OBTAIN PROCEDURE**

Standard: Obtains OP-107 and reviews P & L's and Section 8.2 for Excess Letdown Heat Exchanger Operation. Reviews and verifies initial conditions are satisfied.

Evaluator Cue:	**Initial conditions have been established**

Comment:

OP-107, Section 8.2.2, Note prior to step 1

Performance Step: 2

NOTE: Normally Excess Letdown will go to the VCT. However, if plant conditions warrant, the RCDT may be selected. When the Excess Letdown line has been flushed, the VCT position can then be re-selected.

NOTE: If Excess Letdown is to remain in service for sufficient time for dilution or boration to be necessary then VCT level should be lowered to accommodate the expected level increase before placing Excess Letdown in service

NOTE: Placing Excess Letdown in service will result in increased dose rates in the Seal Water Heat Exchanger Room.

Standard: Operator reads and placekeeps notes

Comment:

OP-107, Section 8.2.2, Caution prior to step 1

Performance Step: 3 Caution: Excess Letdown operation during times of BTRS operation may result in damage to the RCP seals (due to increased contaminants and higher pH water). This should not prevent any AOP or EOP actions. The Responsible Engineer for RCP or CVCS may provide additional guidance if needed.

Standard: Operator reads and placekeeps caution

Comment:

OP-107, Section 8.2.2, Step 1

Performance Step: 4 **INFORM** Radwaste Control Room to monitor Seal Water Filter ∆P while Excess Letdown is in service.

Standard: Contacts RW Control Room operator to monitor Seal Water Filter ∆P while Excess Letdown is in service

Simulator Operator:	**Acknowledge request to monitor Seal Water Filter ∆P while Excess Letdown is in service**

Comment:

OP-107, Section 8.2.2, Step 2.a

Performance Step: 5 **PLACE** the excess letdown heat exchanger in operation as follows:

VERIFY 1CC-188, CCW TO EXCESS LETDOWN HEAT EXCHANGER, is open.

Standard: Locates MCB switch for 1CC-188, CCW TO EXCESS LETDOWN HEAT EXCHANGER, verifies it is open

Comment:

OP-107, Section 8.2.2, Step 2.b

Performance Step: 6 **VERIFY** 1CC-202 SB, CCW FM EXCESS LTDN & RCDT HEAT EXCHANGERS, is open.

Standard: Locates MCB switch for 1CC-202 SB, CCW FM EXCESS LTDN & RCDT HEAT EXCHANGERS, verifies it is open.

Comment:

OP-107, Section 8.2.2, Step 2.c

Performance Step: 7 **VERIFY** 1CC-176, CCW TO EXCESS LTDN & RCDT HEAT EXCHANGERS, is open.

Standard: Locates MCB switch for 1CC-176, CCW TO EXCESS LTDN & RCDT HEAT EXCHANGERS, verifies it is open.

Comment:

OP-107, Section 8.2.2, Note prior to step 3

Performance Step: 8 NOTE: Flushing the excess letdown line to the RCDT is required if the boron concentration in the excess letdown line from the RCS isolation valves to 1CS-466 is unknown or differs from RCS concentration. The volume of this line is 74 gallons. Two volumes (148 gallons) should be adequate to prevent unexpected reactivity changes in the RCS when flow is aligned to the VCT.

Standard: Operator reads and placekeeps note

Comment:

OP-107, Section 8.2.2, Caution prior to step 3

Performance Step: 9 Caution: 1CS-464, HC-137 EXCESS LTDN FLOW is rated for 1500 psid. Anytime that 1CS-464 is exposed to greater than 1500 psid, leakby should be expected.

Standard: Operator reads and placekeeps caution

Comment:

OP-107, Section 8.2.2, Step 3.a

Performance Step: 10 **IF** excess letdown flow is to be aligned to the RCDT,
THEN PERFORM the following:

NOTIFY Radwaste Control Room of expected RCDT level change.

Standard: Contacts RW Control Room and informs the operator to expect RCDT level change.

Simulator Operator:	**RW Operator acknowledges**

Comment:

OP-107, Section 8.2.2, Step 3.b

Performance Step: 11 **VERIFY** 1CS-464, HC-137 EXCESS LTDN FLOW is shut (potentiometer to zero).

Standard: Operator verifies 1CS-464, HC-137 EXCESS LTDN FLOW is shut (potentiometer to zero).

Comment:

OP-107, Section 8.2.2, Step 3.c

✓ **Performance Step: 12** **PLACE** 1CS-466, EXCESS LETDOWN TO VCT/RCDT, to the RCDT position.

Standard: Operator locates MCB switch and places 1CS-466, EXCESS LETDOWN TO VCT/RCDT, to the RCDT position.

Comment:

OP-107, Section 8.2.2, Step 4

✓ **Performance Step: 13** **PLACE** 1CS-461, EXCESS LETDOWN to OPEN.

Standard: Operator locates MCB switch and places 1CS-461, EXCESS LETDOWN valve to OPEN.

Comment:

OP-107, Section 8.2.2, Step 5

✓ **Performance Step: 14** **PLACE** 1CS-460, EXCESS LETDOWN to OPEN.

Standard: Operator locates switch and places 1CS-460, EXCESS LETDOWN valve to OPEN.

Comment:

OP-107, Section 8.2.2, Note prior to Step 6

Performance Step: 15 NOTE: Seal Water Flow should be observed on FR-154A and FR-154B when adjusting 1CS-464, HC-137 EXCESS LTDN FLOW for the following reasons:

- RCP No 1 seal leakoff flow will be affected, and
- The possibility exists of lifting the 150 psi safety on the excess letdown/No. 1 seal return line.

Standard: Operator reads and placekeeps note

Comment:

OP-107, Section 8.2.2, Caution prior to Step 6

Performance Step: 16 Caution: Do **NOT** exceed 174°F outlet temperature as indicated on TI-139.

Caution: Do **NOT** exceed 150 psig as indicated on PI-138.

Standard: Operator reads and placekeeps cautions

Comment:

OP-107, Section 8.2.2, Step 6

✓ **Performance Step: 17**

ADJUST 1CS-464, HC-137 EXCESS LTDN FLOW as necessary to establish excess letdown flow, and not exceed the following parameters:

- 174°F outlet temperature as indicated on TI-139
- 150 psig as indicated on PI-138

Standard:

Operator adjusts 1CS-464, HC-137 EXCESS LTDN FLOW to establish excess letdown flow while not exceeding 174°F outlet temperature as indicated on TI-139 and 150 psig as indicated on PI-138 until $\geq$ 148 gallons have been flushed to the RCDT.

Examiner Cue: **(NOTE: This should be enough time for the candidate to determine that an adequate flush has been completed.)**	After adjustments to 1CS-464 have been made establishing Excess letdown to RCDT cue the applicant: **“Time compression is being used; approximately 10 minutes have elapsed since 1CS-464 has been opened.”**

Comment:

OP-107, Section 8.2.2, Step 7.a

Performance Step: 18

IF excess letdown flow is to be aligned to the VCT, THEN PERFORM the following:
VERIFY 1CS-464, HC-137 EXCESS LTDN FLOW is shut (potentiometer to zero).

Standard:

Locates and verifies 1CS-464, HC-137 EXCESS LTDN FLOW is SHUT

Comment:

OP-107, Section 8.2.2, Step 7.b

✓ **Performance Step: 19** PLACE 1CS-466, EXCESS LETDOWN TO VCT/RCDT, to the VCT position.

Standard: Locates MCB switch and places 1CS-466, EXCESS LETDOWN TO VCT/RCDT, to the VCT position.

Comment:

OP-107, Section 8.2.2, Note prior to Step 7.c

Performance Step: 20 NOTE: Seal Water Flow should be observed on FR-154A and FR-154B when adjusting 1CS-464, HC-137 EXCESS LTDN FLOW for the following reasons:

- RCP No 1 seal leakoff flow will be affected, and
- The possibility exists of lifting the 150 psi safety on the excess letdown/No. 1 seal return line.

Standard: Operator reads and placekeeps note

Comment:

OP-107, Section 8.2.2, Caution prior to Step 7.c

Performance Step: 21 Caution: Do **NOT** exceed 174°F outlet temperature as indicated on TI-139.

Caution: Do **NOT** exceed 150 psig as indicated on PI-138.

Standard: Operator reads and placekeeps cautions

Comment:

OP-107, Section 8.2, Step 7.c

✓ **Performance Step: 22**

ADJUST 1CS-464, HC-137 EXCESS LTDN FLOW as necessary to establish excess letdown flow and not exceed the following parameters:
- 174°F outlet temperature as indicated on TI-139.
- 150 psig as indicated on PI-138.

Standard:

Locates MCB control for 1CS-464, HC-137 EXCESS LTDN FLOW to establish flow and adjusts excess letdown flow while not exceeding 174°F outlet temperature as indicated on TI-139 or 150 psig as indicated on PI-138.

Comment:

Examiner Cue:	**NOTE: It may be necessary to ask the candidate if Excess Letdown has been placed in service IF they do not report to the CRS after Excess Letdown has clearly been established.** **After Excess Letdown has been established and reported to the CRS then:** **Announce: I have the shift, END OF JPM** **Contact Simulator Operator to place the Simulator in Freeze.**

STOP TIME: ______________

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Simulator JPM b

Establish Excess Letdown to the VCT
In accordance with OP-107, Section 8.2, Excess Letdown Heat Exchanger Operation

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT ______

Examiner's Signature: ______________________ Date: ______________

Initial Conditions:	• The unit is operating at 100% power MOL • Normal letdown needs to be secured for maintenance due to a problem with PCV-145 • PCV-145 is in manual

Initiating Cue:	• You are the OATC and have been directed by the CRS to establish Excess Letdown to the VCT per OP-107, Section 8.2. • Excess letdown has not been in service during this refueling cycle

Facility: Harris Nuclear Plant

Task No.: 301135H601

Task Title: Take Corrective Action For Failure of CSIP Mini-Flow Valves to Re-Position

JPM No.: 2020 NRC Exam Simulator JPM c

K/A Reference: 006 A4.07 RO 4.4 SRO 4.4

ALTERNATE PATH - YES

Examinee: ______________________

NRC Examiner: ______________

Facility Evaluator: ______________________

Date: ________

Method of testing:

Simulated Performance: ______

Actual Performance: X

Classroom ______ Simulator X Plant ________

READ TO THE EXAMINEE
I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.

Initial Conditions:	• The Unit was at 100% power when a technician's error resulted in a Reactor Trip and Safety Injection • The crew is performing EOP-E-0, Reactor Trip or Safety Injection, and are at step 37

Initiating Cue:	• You are the OATC • Beginning at Step 37, you are to continue performing EOP-E-0

Task Standard: Obtain adequate flow through a running CSIP.

Required Materials: E-0, Reactor Trip or Safety Injection, Rev. 15

General References: E-0, Reactor Trip or Safety Injection, Rev. 15

Handout: None

Time Critical Task: No

Validation Time: 10 minutes

PERFORMANCE STEP	**CRITICAL STEP JUSTIFICATION**
Step 2	Resetting SI removes the active signal to allow termination of SI (allows component re-positioning).
Step 4	Stopping one CSIP prevents unnecessary PRZ overfill to a solid condition.
Step 9	Shutting FK-122.1 prevents CSIP runout when establishing a charging flowpath.
Step 10	Opening 1CS-235 and 1CS-238 establishes a charging flowpath.
Step 11	Opening FK-122.1 to a minimum of 10% establishes minimal charging flow prior to isolating the BIT to ensure the running CSIP is not deadheaded.
Step 12	Shutting 1SI-3 and 1SI-4 isolates flow through the BIT to prevent CSIP runout.
Step 14	Establishing a flow rate of >60 gpm is required by procedure.

Performance Step	**ALTERNATE PATH JUSTIFICATION**
Steps 7 - 14	1CS-214 (common miniflow isolation) failing to open prevents normal miniflow for the running CSIP to be established. The candidate must establish minimal charging flow prior to isolating the BIT to ensure that the running CSIP is not deadheaded.

2020 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-146
- Password “NRC3sros”
- Go to run
- Silence and Acknowledge annunciators
- It may be necessary to roll the Generator 86 relays at the start of this JPM or between runs. To accomplish this run the AMS file “Roll Gen 86 Relays” to get the 86 relays to the trip condition.
- **NOTE: The ERFIS screen that normally displays Tavg needs to be switched to Turn on code “ITREND” for RCS temperature and pressure.**

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner. (The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

To recreate the IC setup for this JPM:

- Initial Simulator IC was IC-19
- Pre-load failure of control switch 1CS-214
    - IDI XA2I162 (n 00:00:00 00:00:00) ASIS

Insert:

- SIS01A (1 00:00:00 00:00:00) INADVERTENT_INIT
- SIS01B (1 00:00:00 00:00:00) INADVERTENT_INIT
- Go To RUN and initiate Trigger 1 – Inadvertent SI Train A and B
- Perform / markup E-0 through Step 37 (SI Termination Criteria).
- Set up ERFIS Plot to include RCS Pressure
- Adjust AFW flow to approx. 80 KPPH/SG
- Secure TDAFWP by closing 1MS-70 and 1MS-72
- Energize 1A1 and 1B1
- Silence Acknowledge and Reset Annunciators
- FREEZE (with PZR Level at approx. 60%) and Snap these conditions to your exam IC
- **NOTE: The ERFIS screen that normally displays Tavg needs to be switched to Turn on code “ITREND” for RCS temperature and pressure.**

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ____________

Performance Step: 1 **OBTAIN PROCEDURE**

Standard: Obtains copy of EOP-E-0 and reviews steps that will be performed prior to initiation of step.

Comment:

E-0, Step 37

✓ **Performance Step: 2** Reset Safety Injection.

Standard: **(✓) Locates Train A and Train B SI reset MCB switch and takes respective train switch to reset position and then allows switch to return to normal position.**

Verifies that SI is reset by observation of Bypass Permissive Lights

- SI Actuated light stays on until both A and B train reset is completed.
- When train A or B is reset the SI Reset Auto SI Blocked light blinks on and off
- When both train A and B are reset the SI Actuated light extinguishes and the SI Reset Auto SI Blocked Light stays ON

Comment:

E-0, Step 38

Performance Step: 3 Manually Realign Safeguards Equipment Following A Loss Of Offsite Power. (Refer to Attachment 6)

Standard: Acknowledges requirement to manually realign Safeguards Equipment following a loss of Offsite Power

(Notes that no loss of power has occurred)

Comment:

E-0, Step 39

✓ **Performance Step: 4** Stop All But One CSIP.

Standard: Observes that A and B CSIP are running.

- Locates MCB switch for the CSIP control and secures ONE CSIP.

Comment:

E-0, Step 40

Performance Step: 5 RCS Pressure - STABLE OR RISING

Standard: Verifies RCS pressure is rising by trends on ERFIS, OSI PI or MCB RCS pressure meters. (may report trend to CRS)

Evaluator Cue:	**(IF reported that RCS pressure is rising: acknowledge report)**

Comment:

E-0, Step 41

Performance Step: 6

Open Normal Miniflow Isolation Valves:
- CSIP A: 1CS-182
- CSIP B: 1CS-196
- CSIP C: 1CS-210
- COMMON: 1CS-214

Standard:

Locates MCB switch for each of the following valves and takes switch to OPEN position
- CSIP A: 1CS-182
- CSIP B: 1CS-196
- CSIP C: 1CS-210

Locates MCB switch for 1CS-214 and after attempting to open valve determines that the valve will NOT OPEN

Determines RNO for step 41 is needed

Comment:

E-0, Step 41 RNO - ALTERNATE PATH begins here

Performance Step: 7

- If normal miniflow for running CSIP established, THEN GO TO Step 42. (NO)
- IF normal miniflow for running CSIP can NOT be established, THEN Observe NOTE prior to Step 45 AND GO TO Step 45. (YES)

Standard:

Determines that RNO action is to go to step 45 and proceed with actions there.

Comment:

E-0, Step 45 – NOTE prior to step (ALTERNATE PATH)

Performance Step: 8

NOTE: The following step contains a Safety Injection termination sequence for which CSIP normal miniflow is not available. The charging flow control valve is opened a minimal amount prior to isolating the BIT to ensure the running CSIP is not deadheaded.

Standard:

Operator reads and placekeeps note

Comment:

E-0, Step 45.a (ALTERNATE PATH)

✓ **Performance Step: 9**

Establish Minimum Charging Flow AND Isolate BIT Flow:
Shut charging flow control valve: FK-122.1

Standard:

Locates MCB control for FK-122.1, places FK-122.1 in MANUAL and reduces output to 0 (shuts valve)

Comment:

E-0, Step 45.b (ALTERNATE PATH)

✓ **Performance Step: 10**

Open charging line isolation valves:

- 1CS-235
- 1CS-238

Standard:

Locates MCB control switches for each valve and takes switches to OPEN

- 1CS-235 (red light on)
- 1CS-238 (red light on)

Comment:

E-0, Step 45.c (ALTERNATE PATH)

✓ **Performance Step: 11** Set charging flow controller demand position to 30%.

Standard: Locates MCB control for 1FK-122.1 and adjusts FK-122.1 open to 30%. (critical to establish an indication of a positive increase in charging flow)

Comment:

E-0, Step 45.d (ALTERNATE PATH)

✓ **Performance Step: 12** Shut BIT outlet valves:

- 1SI-3
- 1SI-4

Standard: Locates MCB control switches for each valve and takes switches to SHUT

- 1SI-3 (green light on)
- 1SI-4 (green light on)

Comment:

E-0, Step 45.e (ALTERNATE PATH)

Performance Step: 13 Ensure cold leg AND hot leg injection valves – SHUT

- 1SI-52
- 1SI-86
- 1SI-107

Standard: Locates MCB control for 1SI-52, 1SI-86 and 1SI-107 and verifies that all three valves are shut (green lights on)

Comment:

E-0, Step 45.f (ALTERNATE PATH)

✓ **Performance Step: 14** Establish and maintain at least 60 GPM flow through CSIP.

Evaluator Note:	**Total flow through the running CSIP consists of Charging Flow (FI-122A.1) in addition to the three RCP Seal Injection Flows (FI-130A, FI-127A and FI-124A).** **With FK-122.1 set to ~30% flow will be >60 GPM**

Standard: Totals flow of Charging flow through FI-122A.1 and RCP Seal Injection flows (3) through FI-130A, FI-127A, and FI-124A. IF the total is < 60 gpm THEN Locates MCB for CSIP flow (FI-122) and adjusts Charging Flow Controller FK-122.1 until total flow maintained is ≥ 60 gpm.

Comment:

Evaluator Cue:	**After applicant adjusts/ensures Charging Flow + Seal Injection flow is verified to be maintaining ≥60 gpm flow – Evaluation on this JPM is complete.** **Announce: I have the shift. END OF JPM** **Contact the Simulator Operator and place the Simulator in Freeze.**

STOP TIME: ________________

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Simulator JPM c

Take Corrective Action For Failure of CSIP Mini-Flow Valves to Re-Position

In accordance with EOP-E-0, Reactor Trip or Safety Injection

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	• The Unit was at 100% power when a technician's error resulted in a Reactor Trip and Safety Injection • The crews is performing EOP-E-0, Reactor Trip or Safety Injection and are at step 37

Initiating Cue:	• You are the OATC • Beginning at Step 37, you are to continue performing EOP-E-0

Facility:	Harris Nuclear Plant	Task No.:	003001H101
Task Title:	Start a RCP with Spray Valve Failure	JPM No.:	2020 NRC Exam Simulator JPM CR d
K/A Reference:	002 A1.01 RO 3.8 SRO 4.1	**ALTERNATE PATH - YES**	

Examinee: ______________________ NRC Examiner: ____________

Facility Evaluator: ______________________ Date: ________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom ______ Simulator X Plant ________

READ TO THE EXAMINEE I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	• You are the extra RO during a plant startup. • GP-004, Reactor Startup is in progress and is currently on HOLD. • The plant has been stabilized with Shutdown Banks withdrawn. • Two hours ago the “B” RCP was removed from service for maintenance. • Maintenance has been completed and the “B” RCP is ready for operation. • The previous crew has verified that all initial conditions to start the RCP are met and have initialed all steps

Initiating Cue:	• The CRS has instructed you to start “B” RCP, in accordance with OP-100, Reactor Coolant System, Section 5.1, Reactor Coolant Pump Start-up. • The initial conditions have been verified.

Evaluators Note:	***To reduce student prep time, consider supplying student with a copy of the procedure and pre-briefing student prior to entry into the Simulator.***

Task Standard:	Start a RCP and respond to a failed open PZR spray valve when the pump is started in accordance with AOP-019, Malfunction of RCS Pressure Control
Required Materials:	OP-100 mark up with Attachment 3 included.
General References:	OP-100, Reactor Coolant System, Rev. 47 and AOP-019, Malfunction of RCS Pressure Control, Rev. 25
Handout:	OP-100, Rev. 47, pages 1 – 8, Prerequisites, P&L's OP-100, Rev. 47, pages 9 – 12, Section 5.1, Reactor Coolant Pump Start-up, **with the Initial Conditions signed off if desired** OP-100, Rev. 47, page 94, Attachment 3, #1 Seal Performance Parameters
Time Critical Task:	No
Validation Time:	10 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 9	System interlock requires proper oil pressure be established prior to starting pump
Step 13	Administrative requirements state the System flow cannot be established until the lift oil pump breaker is closed for > 2 mins to prevent pump damage

PERFORMANCE STEP	Alternate Path Critical Step Justification
Step 21	Entry conditions are met for AOP-019, Malfunction of RCS Pressure Control when PZR Spray valve controller PK-444D.1, PZR Spray Loop B, 1RC-103 fails open upon starting the 'B' RCP. AOP-019 requires the operator to perform the immediate actions including the RNO response to control a PZR Spray valve (shut valve) when that valve is NOT properly positioned for current PZR pressure or plant conditions. Performing these actions correctly will prevent an unnecessary Safety Injection from occurring.

2020 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-147
- Password “NRC3sros”
- Go to RUN
- CRT displays – CRT 2: QP VCT, CRT 3: QP TAVG and CRT 4: QP SGLVL
- Set Source Range Audio Multiplier to 1000 to establish audible counts
- Silence and Acknowledge annunciators

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner. (The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

To recreate the IC setup for this JPM:

- Initial Simulator IC was IC-7, Mode 3 HSD, MOL conditions – RCS pressure 2235, RCS temp. 557°F, all rods in
- GO to run
- Secure the “B” RCP
- Wait approximately 5 minutes for the simulator to stabilize
- Create a conditional Trigger to open PZR spray valve PK-444D.1 with a 45 second delay and 45 second ramp after the control switch for the ‘B’ RCP is taken to start

  To create the conditional trigger:
  - Go to malfunctions
  - Find PRS14B Pressurizer Spray Valve 444D Failure (with manual control)
    - Open the malfunction window
    - Set delay to 45 seconds
    - Set ramp time to 45 seconds
    - Set initial severity to 30 (that way the meter will not go to 0 – adjust this to whatever percent open 1RC-103 is at after securing the ‘B’ RCP and the simulator is stable)
    - Set the malfunction to Trigger 1
  - Go to triggers
    - Click on Trigger 1
    - Click on ‘Assign File’
    - Choose RCP_B_START
    - (source file should now have RCP_B_START)

- Silence Acknowledge and Reset Annunciators

- Freeze and Snap these conditions to your exam IC

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

Evaluator Note:	The candidates should be briefed outside of the Simulator prior to performing this JPM. Provide them with a copy of the procedure (with initial conditions initialed as completed). This will allow them to review the Precautions and Limitations associated with OP-100 and have time for a task preview of the steps to accomplish starting the RCP. Expect that the candidates will take about 20 minutes to complete this review. **During the performance of the JPM the candidate may use either MCB indication or ERFIS indications when reviewing RCP pump indications.**

START TIME: ____________

Obtain Procedure

Performance Step: 1 Procedure obtained and begins the task of starting the RCP

Standard: Reviews initial cue and determines the initial conditions for starting the system are complete

Comment:

OP-100, Section 5.1.2, Caution prior to Step 1

Performance Step: 2 CAUTION: Only one RCP is to be started at any one time. If the motor is allowed to coast to a stop between starts, two successive starts are permitted. A third start may be made when the winding and the core have cooled by running for 20 minutes, or by standing idle for 45 minutes.

Standard: Operator reads and placekeeps caution

Comment:

OP-100, Section 5.1.2, Step 1.a

Performance Step: 3 **VERIFY** the following before pump start:

IF jogging RCPs per GP-001, **THEN VERIFY** RCS Pressure is greater than 325 psig.

Standard: Step 1.a is marked N/A

Comment:

OP-100, Section 5.1.2, Step 1.b

Performance Step: 4 **VERIFY** # 1 Seal ΔP is greater than 200 psid.

Standard: Locates PI-156A1 and verifies that the ‘A’ RCP #1 Seal ΔP is greater than 200 psid.

Comment:

OP-100, Section 5.1.2, Note prior to Step 1.c

Performance Step: 5 NOTE: VCT Outlet Temp TE-116 should be used for seal injection water temperature.

Standard: Operator reads and placekeeps note

Comment:

OP-100, Section 5.1.2, Step 1.c

Performance Step: 6

VERIFY Seal Injection flow is between 8 and 13 gpm at a temperature between 60 and 130°F.

Standard:

Locates seal injection flow indication FI-156A and verifies flow between 8-13 gpm and also verifies VCT temperature indicator TI-116.1 reading between 60-130°F. The candidate my use ERFIS points rather than MCB indications.

Comment:

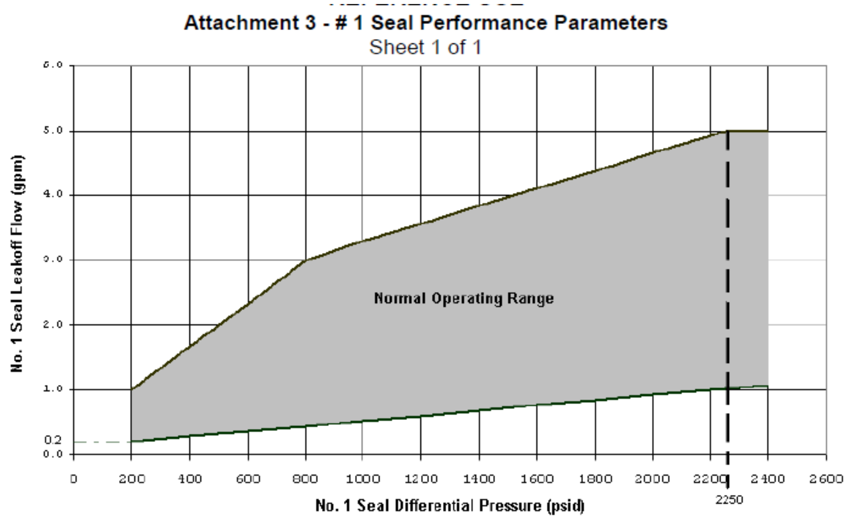
OP-100, Section 5.1.2, Step 1.d

Performance Step: 7

VERIFY # 1 Seal Leakoff is in the normal operating range of Attachment 3.

Standard:

Locates #1 Seal Leakoff flow indicator FR-154A and verifies that it meets Attachment 3 requirements (see below).



Attachment 3 - # 1 Seal Performance Parameters
Sheet 1 of 1

Evaluator Note:	**Seal Leakoff flow is also available via multiple programs on the Plant Computer (ERFIS, OSI-PI, etc). It is acceptable for the candidate to complete this step using ANY of the available indications.**

Comment:

OP-100, Section 5.1.2, Caution prior to Step 2

Performance Step: 8 CAUTION: RCPs shall not be started with one or more of the RCS cold leg temperatures less than or equal to 325°F unless the secondary water temperature is less than 50°F above each of the RCS cold leg temperatures. This caution is only applicable to the first RCP to be started.

Standard: Operator reads and placekeeps caution

- Understands the caution is N/A.

Comment:

OP-100, Section 5.1.2, Step 2

✓ **Performance Step: 9** **START** the RCP Oil Lift Pump.

Standard: Locates 'B' RCP Oil Lift Pump switch and starts the oil lift pump.
(Critical to start the RCP Oil Lift Pump)

START Time ____________

Comment:

OP-100, Section 5.1.2, Step 3

Performance Step: 10 **VERIFY** the amber permissive light on the lift pump control switch is lit indicating proper lift oil pressure has been achieved.

Standard: Locates amber permissive light on the 'B' RCP Oil Lift Pump Switch and verifies it is lit.

Comment:

OP-100, Section 5.1.2, Step 4

Performance Step: 11	**ALLOW** the RCP Oil Lift Pump to run for a minimum of 2 minutes before starting an RCP**.**
Standard:	Waits minimum of 2 minutes after starting the oil lift pump prior to starting the 'A' RCP.
Comment:	

OP-100, Section 5.1.2, Note prior to Step 5

Performance Step: 12	When an RCP is started, the RCP ammeter will go off scale high, and then decrease to the normal hot or cold running amps after 15 to 25 seconds.
Standard:	Operator reads and placekeeps note
Comment:	

Evaluator Note:	**When the 'B' RCP start switch is taken to "start" a timer starts and runs for 45 seconds after which 1RC-103 will ramp open over 45 seconds lowering RCS pressure and requiring the operator to enter into AOP-019.** **Annunciators ALB-009-5-1, PZR High-Low Press and ALB-009-3-3, PZR Cont Low Press and Heaters On will alarm ~60 seconds after 1RC-103 begins to fail open.** **IF no actions are taken a SI will occur ~4:30 minutes from event onset.**

OP-100, Section 5.1.2, Step 5

✓ **Performance Step: 13** **START** the RCP.

Standard: Locates control switch for ‘B’ RCP and starts ‘B’ RCP

Comment: **START Time ____________ (≥ 2minutes since lift pump start)**

The two minute minimum is not critical but ensures start permissives are met for the RCP start.

OP-100, Section 5.1.2, Step 6

Performance Step: 14 **VERIFY** the following normal operating parameters:

- Running amps: Hot 460 to 540 amps Cold 715 amps
- RCS flow: Greater than or equal to 98%
- # 1 Seal ΔP Greater than 200 psid
- # 1 Seal leakoff in the normal operating range of Attachment 3
- Motor Winding temperature <300°F

Standard: Locates and verifies each parameter is in the normal operating range (ERFIS or MCB indications may be used)

Comment: **Note: Hot running motor amp range of 460 to 540 amps will apply.**

Evaluators Note:	**The actions to secure the 'B' RCP oil lift pump do not have to be performed since the RCS pressure reduction will take precedence over this step.**

OP-100, Section 5.1.2, Note prior to Step 7

Performance Step: 15

NOTE: The oil lift pump should be run at least 1 minute after starting an RCP.

After at least 1 minute, STOP the RCP OIL LIFT PUMP.

Standard: Waits at least 1 minute then secures the 'B' RCP oil lift pump.

Comment: **Secure Time ____________ ($\geq$ 1 minute since RCP start)**

ALTERNATE PATH

Performance Step: 16

Identifies RCS pressure lowering and Spray valve 1RC-103 failure

Annunciators:

- **ALB-009-5-1, PZR High-Low Press**
- **ALB-009-3-3, PZR Cont Low Press and Heaters On**

Standard:

Identifies RCS pressure lowering

Identifies PZR Spray Loop A PCV-444D (1RC-103) red indication light and valve demand increasing (or at 100%)

Acknowledges alarms and reports conditions to CRS

May review APP or directly enter AOP-019 based on current plant indications

Announces "AOP-019 Entry Conditions met, taking immediate actions for AOP-019."

Comment:

Evaluator Cue:	**CRS acknowledges report**

Evaluator Note:	**Securing 'B' RCP is an action contained in AOP-019 but this action is not performed immediately. Stopping the 'B' RCP would be performed at step 14 in Section 3.1 of the procedure unless other trip limits on the RCP are exceeded prior to reaching this step.**

AOP-019, Malfunction of RCS Pressure Control

Performance Step: 17

- Steps 1 through 3 are immediate actions

Standard: Performs immediate actions from memory without accessing or reading from the AOP

Comment:

AOP-019, Step 1

Performance Step: 18 **CHECK** that a bubble exists in the PRZ. **(YES)**

Standard: States that a bubble exists in the PRZ

Comment:

AOP-019, Step 2

Performance Step: 19 **VERIFY** ALL PRZ PORVs **AND** associated block valves properly positioned for current PRZ pressure and plant conditions. **(YES)**

Standard: Verifies ALL PRZ PORVs **AND** associated block valves properly Positioned by observing green shut lights indicated for all PZR PORV and all red open lights on for PZR PORV Block Valve control switches.

Comment:

AOP-019, Step 3

Performance Step: 20 **CHECK** BOTH PRZ Spray Valves properly positioned for current PRZ pressure and plant conditions.

PCV-444C PZR Spray Loop A (1RC-107) - SHUT **(YES)**
PCV-444D PZR Spray Loop B (1RC-103) - OPEN **(NO)**

Standard: Identifies that the PZR Spray valves are NOT properly positioned for current plant conditions.

- Takes RNO actions

Comment: **1RC-107 is shut which is its proper position.**
1RC-103 should not be full open (or going full open for this condition).

AOP-019, Step 3 RNO

✓ **Performance Step: 21** CONTROL PRZ Spray Valves using ONE of the following methods (listed in order of preference):

- AFFECTED Spray Valve controller in MANUAL (if only one is obviously malfunctioning)

OR

- PK-444A, Master Pressure Controller, in MANUAL

OR

- BOTH individual Spray Valve controllers in MANUAL

Standard: Places PCV-444D PZR Spray Loop B controller to manual and lowers the output to zero (0).

Stops RCS pressure reduction caused from open spray valve.

Stabilizes RCS pressure.

Comment: **(Critical to stop the RCS pressure reduction using one of the methods listed to prevent an unnecessary automatic SI from occurring.)**

AOP-019, Malfunction of RCS Pressure Control

Performance Step: 22 Obtain copy of AOP-019

Standard: Announces immediate actions of AOP-019 are complete and obtains a copy of AOP-019 to continue actions associated with the procedure.

Evaluator Cue: **CRS acknowledges report**

Comment:

Examiner Cue:	**After the candidate has shut 1RC-103 and has obtained a copy of AOP-019: Evaluation on this JPM is complete.** **Announce END OF JPM** **Direct Simulator Operator to place the Simulator in FREEZE.**

STOP TIME: ____________

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 HNP NRC Exam Simulator JPM d
Start a RCP with Spray Valve Failure

In accordance with OP-100, Reactor Coolant System
In accordance with AOP-019, Malfunction Of RCS Pressure Control

Examinee’s Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner’s Signature: __________________________________ Date: ____________________

Initial Conditions:	• You are the extra RO during a plant startup. • GP-004, Reactor Startup is in progress and is currently on HOLD. • The plant has been stabilized with Shutdown Banks withdrawn. • Two hours ago the "B" RCP was removed from service for maintenance. • Maintenance has been completed and the "B" RCP is ready for operation. • The previous crew has verified that all initial conditions to start the RCP are met and have initialed all steps

Initiating Cue:	• The CRS has instructed you to start "B" RCP, in accordance with OP-100, Reactor Coolant System, Section 5.1, Reactor Coolant Pump Start-up. • The initial conditions have been verified.

Facility:	Harris Nuclear Plant	Task No.:	022001H101
Task Title:	Return the Containment Fan Coolers to normal following a Safety Injection actuation	JPM No.:	2020 NRC Exam Simulator JPM CR e
K/A Reference:	022 A4.01 RO 3.6 SRO 3.6	**ALTERNATE PATH - NO**	

Examinee: NRC Examiner:

Facility Evaluator: Date:

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom ______ Simulator X Plant ________

READ TO THE EXAMINEE

I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.

Initial Conditions:	• An automatic SI initiation occurred due to a combination of equipment failure and technician error • SI has been terminated

Initiating Cue:	You have been directed to return Containment Fan Coolers to the normal alignment per ES-1.1, SI Termination, Attachment 1 step 6.a using OP-169, Containment Cooling And Ventilation, Section 8.4. The A-SA train will be used for normal operation.

Task Standard: Containment Fan Coolers are returned to NORMAL lineup.

Required Materials: None

General References: ES-1.1, SI Termination, Rev 3
OP-169, Containment Cooling And Ventilation, Rev 28

Handout: ES-1.1 Attachment 1 Sheet 3 of 7
OP-169, Rev. 28, pages 1 – 6, Prerequisites, P&L's
OP-169, Rev. 28, pages 7 – 9, Section 5.1, Start Up of Containment Fan Cooler Units (Normal Cooling Mode)
OP-169, Rev. 28, pages 27 – 28, Section, 8.4, Returning System to Normal from SI Operation

Time Critical Task: No

Validation Time: 15 Minutes

Performance Step	**CRITICAL STEP JUSTIFICATION**
Step 13	To comply with OP-169, Precaution and Limitation #11 After any fan cooler is started in low speed, the fan should be allowed to come up to speed for approximately 15 seconds before shifting to fast speed. This reduces the starting current required for high speed operation.
Step 15	The fan must be stopped in order to change fan speed from low speed to high speed in order to be in the correct operating mode for the current plant condition..

2020 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-148
- Password “NRC3sros”
- Go to RUN and wait ~ 10 seconds then silence and acknowledge alarms.

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner. (The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

To recreate the IC setup for this JPM:

- Initial Simulator IC was IC-19
- Insert a Manual SI or MALF for Inadvertent SI
- Perform / markup E-0 through transition to ES-1.1
- Perform / markup ES-1.1 actions up to step 35 which is Realign Plant Systems for Normal Ops using Attachment 1 (restoration of Containment Fan Coolers is directed)
- Leave Fan Coolers in SI Mode
- Silence Acknowledge and Reset Annunciators
- FREEZE and Snap these conditions to your exam IC

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ____________

OP-169, 8.4.1

Performance Step: 1 Review applicable procedure.

Standard: Verifies Initial Conditions are met.

- SI Reset (YES)
- Instrument Air restored to dampers (YES)

Comment:

OP-169, 8.4.2, Caution prior to step 1

Performance Step: 2 CAUTION: Failure of equipment to secure in this section will result in the associated EDG being inoperable. Tech Spec 3.8.1.1 is applicable until the breaker for the affected load is opened.

Standard: Operator reads and placekeeps caution

Comment:

OP-169, 8.4.2 Step 1

Performance Step: 3 CIRCLE the train to be used for normal operation. A-SA (B-SB)

Evaluator Cue:	**The CRS designates Train "A" for normal operation.**

Standard:

- Circles the A-SA components for alignment in step 2

Comment:

OP-169, 8.4.2 Step 2

Performance Step: 4 PLACE the following control switches for the selected train's Air Handling Units to STOP:

- AH-2 A-SA (~~AH-1 A-SB~~)
- AH-2 B-SA (~~AH-1 B-SB~~)
- AH-3 A-SA (~~AH-4 A-SB~~)
- AH-3 B-SA (~~AH-4 B-SB~~)

Standard:

- Places AH-2 A-SA control switch in STOP.
- Places AH-2 B-SA control switch in STOP.
- Places AH-3 A-SA control switch in STOP.
- Places AH-3 B-SA control switch in STOP.

Comment:

OP-169, 8.4.2 Step 3

Performance Step: 5 CHECK the following post-accident discharge nozzle dampers SHUT on Status Light Box 5 (~~6~~) for the selected train of fans:

a. CV-D3 for AH-2 (~~CV-D1 for AH-1~~)

b. CV-D5 for AH-3 (~~CV-D7 for AH-4~~)

Standard: Verifies CV-D3 and CV-D5 indicate SHUT on Status Light Box 5.

Comment:

OP-169, 8.4.2 Step 4

Performance Step: 6 PLACE the train secured in Step 8.4.2.2 in operation per Section 5.1.

Standard: Proceeds to Section 5.1.

Evaluator Cue:	**Provide OP-169, Section 5.1 to the candidate at this time.**

Comment:

OP-169, Note prior to Section 5.1

Performance Step: 7 NOTE: Where the Operator has a choice between Train A or Train B, this procedure will list Train A number and letter identification first with Train B in parentheses.

Standard: Operator reads and placekeeps note

Comment:

OP-169, 5.1.1, Initial Conditions

Performance Step: 8 Verify Initial Conditions:

- Attachments 1 and 2 are completed.
- ESW train is in service which corresponds to the AH unit train to be started.

Standard:

- Acknowledges cue for Attachments 1 and 2.
- Verifies ESW Train “A” in service.

Evaluator Cue:	**Attachments 1 and 2 have been completed.**

Comment:

OP-169, 5.1.2, Note prior to step 1

Performance Step: 9 NOTE: When changing Containment Cooling modes, or swapping Containment Fan Cooler Trains, care must be taken to prevent the following:

- Entering Technical Specification 3.6.1.4 at -1.0 inwg Containment pressure (1 hour action).
- Opening the Containment Vacuum Breakers at -2.25 inwg Containment pressure.

This may be accomplished by performing the evolution slowly, monitoring CNMT pressure effects using ERFIS point PCP7611. Also, placing the Containment Normal Purge Exhaust flow controller (FK-7624) in manual and shutting CP-B9, will allow CNMT pressure to slowly rise, thus compensating for the CNMT pressure drop that will occur during each fan start.

Standard: Operator reads and placekeeps note

Comment:

OP-169, 5.1.2, Caution prior to step 1

Performance Step: 10 CAUTION: Failure of equipment to secure in this section will result in the associated EDG being inoperable. Tech Spec 3.8.1.1 is applicable until the breaker for the affected load is opened.

Standard: Operator reads and placekeeps caution

Comment:

OP-169, 5.1.2, step 1

Performance Step: 11 IF CNMT Normal Purge is in service AND IF desired for CNMT pressure control, THEN PERFORM the following:

a. PLACE FK-7624, NORM PURGE EXH FLOW, in MANUAL.
b. Using FK-7624, SHUT CP-B9, NORM CONT PURGE MODULATING VALVE (SLB-7 / 5-3).
c. IF CNMT Normal Purge needs to be restored at any time during the performance of this procedure section, THEN PERFORM the following:
    (1) IF CNMT Normal Purge has NOT tripped, THEN RESTORE FK-7624 to AUTO.
    (2) IF CNMT Normal Purge has tripped, THEN STARTUP CNMT Normal Purge per OP-168.

Standard:

- Checks CNMT Normal Purge secured and N/A’s steps 1.a, 1.b, and 1.c

Evaluator Cue:	**CNMT Normal Purge will be restored by another operator per ES-1.1 Attachment 1 step 12.**

Comment:

OP-169, 5.1.2, Note prior to step 2

Performance Step: 12 NOTE: In winter months, the operating train should be secured per Section 7.1 prior to starting the idle train, to minimize the potential for entering Technical Specification 3.6.1.4 at -1.0 inwg Containment pressure (1 hour action), or opening the Containment Vacuum Breakers at -2.25 inwg Containment pressure.

Standard: Operator reads and placekeeps note

Comment:

OP-169, 5.1.2, step 2

✓ **Performance Step: 13** Place the control switches for both fans in each Containment cooler unit AH-2 A-SA (~~AH-1 B-SB~~) and AH-3 A-SA (~~AH-4 B-SB~~) to LO-SPD.

Standard:

- Places control switch for AH-2 A-SA in LO-SPD.
- Places control switch for AH-2 B-SA in LO-SPD.
- Places control switch for AH-3 A-SA in LO-SPD.
- Places control switch for AH-3 B-SA in LO-SPD.

Comment:

OP-169, 5.1.2, Notes prior to step 3

Performance Step: 14 NOTE: After any fan cooler is started in low speed, the fan should be allowed to come up to speed for approximately 15 seconds before shifting to fast speed. This reduces the starting current required for high speed operation.

NOTE: The following switch sequence must be performed without delay, one fan at a time, to prevent fan coast down before being started in fast speed. This sequence is functionally related (obtain a single result in close sequence or time), allowing signoff to be delayed until running in HI-SPD.

Standard: Operator reads and placekeeps notes

Comment:

OP-169, 5.1.2, step 3

✓ **Performance Step: 15**

Place the control switch for the fans started in Step 5.1.2.2, START in HI-SPD as follows:

- AH-2 A-SA (~~AH-1 A-SB~~)
  - (1) **PLACE** AH-2 A-SA (~~AH-1 A-SB~~) control switch to STOP
  - (2) **PLACE** AH-2 -SA (~~AH-1 A-SB~~) control switch to HI-SPD
- AH-2 B-SA (~~AH-B-SB~~)
  - (1) **PLACE** AH-2 B-SA (~~AH-1 B-SB~~) control switch to STOP
  - (2) **PLACE** AH-2 B-SA (~~AH-1 B-SB~~) control switch to HI-SPD
- AH-3 A-SA (~~AH-4 A-SB~~)
  - (1) **PLACE** AH-3 A-SA (~~AH-4 A-SB~~) control switch to STOP
  - (2) **PLACE** AH-3 A-SA (~~AH-3 A-SB~~) control switch to HI-SPD
- AH-3 B-SA (~~AH-4 B-SB~~)
  - (1) **PLACE** AH-3 B-SA (~~AH-4 B-SB~~) control switch to STOP
  - (2) **PLACE** AH-3 B-SA (~~AH-4 B-SB~~) control switch to HI-SPD

Standard:

- Places control switch for AH-2 A-SA in STOP, then HI-SPD
- Places control switch for AH-2 B-SA in STOP, then HI-SPD
- Places control switch for AH-3 A-SA in STOP, then HI-SPD
- Places control switch for AH-3 B-SA in STOP, then HI-SPD

Comment:

OP-169, 5.1.2, step 4, 5, and 6

Performance Step: 16

- IF FK-7624 was taken to MANUAL in Step 5.1.2.1.a, THEN RESTORE FK-7624 to AUTO.
- IF CNMT Normal Purge is not in service, AND it is desired to place CNMT Normal Purge in service, THEN STARTUP CNMT Normal Purge per OP-168.
- IF both trains of Containment Fan Cooler fans are running (such as during a train swap evolution), THEN PROCEED to Section 7.1 to secure the desired train.

Standard: Reviews steps 4, 5, and 6 and marks these steps N/A

Returns to Section 8.4.2 and proceeds with step 5

Comment:

OP-169, 8.4.2, step 5

Performance Step: 17 PLACE the following control switches for the standby train to STOP:

- AH-1 A-SB (~~AH-2 A-SA~~)
- AH-1 B-SB (~~AH-2 B-SA~~)
- AH-4 A-SB (~~AH-3 A-SA~~)
- AH-4 B-SB (~~AH-3 B-SA~~)

Standard:

- Places control switch for AH-1 A-SB in STOP
- Places control switch for AH-1 B-SB in STOP
- Places control switch for AH-4 A-SB in STOP
- Places control switch for AH-4 A-SB in STOP

Comment:

OP-169, 8.4.2, step 6

Performance Step: 18 CHECK the following post-accident discharge nozzle dampers SHUT on Status Light Box 5 (~~6~~) for the standby train of fans:

a. CV-D1 for AH-1 (~~CV-D3 for AH-2~~) (Shut)

b. CV-D7 for AH-4 (~~CV-D5 for AH-3)~~ (Shut)

Standard: Checks CV-D1 for AH-1 and CV-D7 for AH-4 indicate SHUT on Status Light Box 6.

Comment:

OP-169, 8.4.2, step 7

Performance Step: 19 If containment temperature is greater than 118 ˚F or if additional cooling is desired, refer to Section 8.1, Start-Up of Fan Cooler Units (Maximum Cooling mode).

Standard: Verifies containment temperature is less than 118 ˚F.

(Maybe > 118° but trending DOWN at this time.)

Marks step 7 as N/A

	Evaluator Cue:	**If requested to perform section 8.1 cue the candidate that another operator will complete section 8.1.**

Comment:

Evaluator Cue:	**After containment temperature is verified at or trending to less than 118 ˚F: Evaluation on this JPM is complete.** **Announce END OF JPM** **Direct Simulator Operator to place the Simulator in FREEZE.**

STOP TIME: ______________

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Simulator JPM CR e

Return the Containment Fan Coolers to normal following a Safety Injection actuation

In accordance with OP-169, Containment Cooling And Ventilation

In accordance with EOP-ES-1.1, SI Termination

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	• An automatic SI initiation occurred due to a combination of equipment failure and technician error • SI has been terminated

Initiating Cue:	You have been directed to return Containment Fan Coolers to the normal alignment per ES-1.1, SI Termination, Attachment 1 step 6.a using OP-169, Containment Cooling And Ventilation, Section 8.4. The A-SA train will be used for normal operation.

Facility: Harris Nuclear Plant | Task No.: 064005H101

Task Title: Shutdown EDG A-SA From MCB For Maintenance – Field Flash Stays Energized | JPM No.: 2020 NRC Exam Simulator JPM f

K/A Reference: 064 A4.06 RO 3.9 SRO 3.9 | **ALTERNATE PATH - YES**

Examinee: ______________________ NRC Examiner: __________________

Facility Evaluator: ______________________ Date: ________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom ______ Simulator X Plant ________

READ TO THE EXAMINEE I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	- The Unit is operating 100% power - The 'A' EDG is running in parallel with the grid to support testing of the governor - OST-1013, 1A-SA Emergency Diesel Generator Operability Test Monthly Interval Modes 1-2-3-4-5-6 is NOT in progress - Testing of the governor is complete - The previous shift has reduced the EDG load from 6.3 MW to 2.3 MW and 1 MVAR over the last 30 minutes per OP-155, Emergency Diesel Generator Section 7.1

Initiating Cue:	- You are the BOP. The CRS directs you to continue shutting down the 'A' EDG using OP-155 Section 7.1.2 starting at Step 4.

***Evaluator NOTE*:**	**The candidate should be briefed outside of the simulator prior to performing this JPM. Provide them with a copy of OP-155, pages 1 – 14, 42 – 46, 177 – 182. This will allow them to review the Precautions and Limitations associated with OP-155 and have time for a task review of the steps. Expect the candidate to take about 10 - 15 minutes to complete this review.**

Task Standard:	‘A’ EDG secured using the Emergency Stop switch to de-energize the field flash following failure to de-energize via the normal stop switch per OP-155.
Required Materials:	None
General References:	OP-155, Diesel Generator Emergency Power System, Rev. 91
Handout:	OP-155, Rev. 91, pages 1 – 14, Prerequisites, P&L’s OP-155, Rev. 91, pages 42 – 46, Section 7.1, Unloading and Shutdown of Emergency Diesel Generators From the MCB, **signed off up to 7.1.2 Step 4.** OP-155, Rev. 91, pages 177 – 182, Attachment 7 - Emergency Diesel Generator Post Run Checklist
Time Critical Task:	No
Validation Time:	10 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 4	Controlled load reduction to 0.5 MW prevents reverse powering the ‘A’ EDG.
Step 5	Opening Breaker 106 separates the ‘A’ EDG from the grid which removes load from the EDG and allows the stack exhaust temperatures to lower limiting thermal stresses on the EDG.
Step 15	Emergency stopping the EDG de-energizes the field flashing circuit voltage to prevent the voltage regulator from catching fire if not de-energized.

PERFORMANCE STEP	ALTERNATE PATH JUSTIFICATION
Steps 14 & 15	Generator continues to produce voltage following normal shutdown requiring operator to emergency stop the EDG.

2020 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-149
- Password “NRC3sros”
- Put reactivity data sheets for IC-19 and MOL on status board
- Go to run
- Silence and Acknowledge annunciators

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner. (The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

To recreate the IC setup for this JPM:

- Initial Simulator IC was IC-19
- Go To RUN
- Start and load the ‘A’ EDG to approximately 2.3 MW and 1 MVAR.
- To simulate that the EDG has been stopped and field flashing is still occurring by still having voltage on meter EI-6955A and EI-5945A, create a CAEP as follows:

```
TRG 1 "EDG_A_Start_Switch_to_Stop"

iao xd1d010m (1 00:00:05 00:00:00) 7.2 00:00:05 7.08
iao xd1d022m (1 00:00:05 00:00:00) 210 00:00:05 167
iao xd1d023m (1 00:00:05 00:00:00) 56 00:00:05 51

TRG 2 "EDG_A_Emergency_Stop_Switch_to_Stop"

trg= 2 dao xd1d010m
trg= 2 dao xd1d022m
trg= 2 dao xd1d023m
```

- Silence Acknowledge and Reset Annunciators
- FREEZE and Snap these conditions to your exam IC

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ____________

OP-155 Section 7.1.2, Note prior to step 4

Performance Step: 1 NOTE: The EDG should be completely unloaded from 35% load in less than 5 minutes to minimize carbon buildup.

Standard: Operator reads and placekeeps note

Comment:

OP-155 Section 7.1.2 Step 4.a

Performance Step: 2 PERFORM the following:

a. ENSURE load has been less than 6.2 to 6.4 MW for at least 20 minutes

Standard: Refers to Initial Conditions and determines the ‘A’ EDG load has been below 6.2 MW for the last 30

Comment:

OP-155, Section 7.1.2 Step 4.b

Performance Step: 3 b. ENSURE Generator Winding Temperature is less than 135°C

Standard: Contacts AO at the ECP or obtains winding temperature from ERFIS

Simulator Operator:	**If contacted, Generator Winding Temperature is 125°C.**

Comment:

Evaluator Note:	**Terminate the JPM if the candidate reverse powers the 'A' EDG**

OP-155, Section 7.1.2 Step 4.c

Performance Step: 4 c. REDUCE load to 0.5 MW

Standard: **Time Started Load Reduction: ______**

Locates the Governor Control and Auto Voltage Adjust control switches and adjust the control switches to reduce the 'A' EDG load to 0.5 MW while maintaining load within the Attachment 9 – Emergency Diesel Generator Capacity Curve limits

Comment:

OP-155, Section 7.1.2 Step 4.d

√ **Performance Step: 5** d. PLACE DIESEL GEN A-SA BREAKER 106 SA to TRIP.

Standard: Locates Diesel Gen A-SA Breaker 106 SA control switch and places the switch in the trip position in ≤ 5 minutes from resuming load reduction

Comment:

OP-155, Section 7.1.2 Step 5. a

Performance Step: 6 ENSURE the following:

a. DIESEL GEN A-SA BREAKER 106 SA to indicates OPEN.

Standard: Locates Diesel Gen A-SA Breaker 106 SA control switch and determines the Green light is Lit and the Red light is extinguished.

Comment:

OP-155, Section 7.1.2 Step 5.b

Performance Step: 7 b. EI-6957A1 SA, A Power, indicates zero..

Standard: Locates EI-6957A1 SA, A Power indication and determines the meter is on the lower peg indicating zero.

Comment:

OP-155, Section 7.1.2 Step 5.c

Performance Step: 8 c. EI-6951A SA, A Amps, indicates zero.

Standard: Locates EI-6951A SA, A Amps indication and determines the meter is on the lower peg indicating zero.

Comment:

OP-155, Section 7.1.2 Step 6

Performance Step: 9 RECORD time DIESEL GEN A-SA BREAKER 106 SA is opened on Attachment 7.

Standard: Refers to Attachment 7 - Emergency Diesel Generator Post Run Checklist, and records the time Breaker 106 SA is open in Step 17.j.

Comment:

OP-155, Section 7.1.2 Step 7

Performance Step: 10 IF performing monthly EDG test, THEN PERFORM the following:

a. MARK the remainder of this section “N/A.”
b. CONTINUE EDG shutdown per OST-1013 (OST-1073).

Standard: Refers to initial conditions and determines step is not applicable and marks step 7a and 7b N/A.

Comment:

OP-155, Note prior to Section 7.1.2 Step 8

Performance Step: 11 To determine that cylinder temperatures are less than 450°F, the stack exhaust temperature will be monitored until temperature is less than 500°F.

Standard: Operator reads and placekeeps note

Comment:

OP-155, Section 7.1.2 Step 8

Performance Step: 12 At the ECP, POSITION temperature selector switch to positions 17 and 18 to monitor stack exhaust temperatures.

Standard:

- Contacts AO at the ECP and requests stack exhaust temperatures.

Simulator Operator:	**Stack Exhaust Temperatures are 475°F on positions 17 and 18.**

Comment:

OP-155, Section 7.1.2 Step 9

Performance Step: 14 At the MCB, WHEN stack exhaust temperatures are less than 500°F, THEN POSITION DIESEL GENERATOR A-SA CONTROL SWITCH TO STOP.

Standard: Locates Diesel Generator A-SA control switch and places the switch in the stop position

Comment:

Alternate Path begins here
OP-155, Section 7.1.2 Step 10

Performance Step: 15

At the MCB, CHECK the following: (Reference 2.7.4, 2.7.7, 2.8.11)
   a. EI-6955A SA, A VOLTS, voltage is decreasing.
   b. EI-6954A SA, A FLD VOLTS, voltage is decreasing.

Standard:

Locates EI-6955A SA, A Volts and EI-6954A SA, A FLD Volts indications and determines the meters are unchanged and the generator is still producing voltage.

Comment:

OP-155, Section 7.1.2 Step 11 (ALTERNATE PATH)

✓ **Performance Step: 16**

IF voltage is NOT decreasing, THEN EMERGENCY STOP the EDG to prevent the voltage regulator from catching fire.

Standard:

(✓) Locate Diesel Generator A-SA Emergency Stop control switch and places the switch in the emergency stop position

Reports to CRS that EDG A had to be Emergency Stopped due to field flashing still occurring.

Comment:

Evaluator Communication:	**Acknowledge any communications.**

Evaluator Note:	**After the A EDG has been Emergency stopped and communications are completed:** **Cue – END OF JPM – I have the shift.** **Direct Simulator Operator to go to FREEZE**

STOP TIME: ______________

Simulator Operator:	***When directed by the Lead Examiner go to FREEZE.***

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Simulator JPM f

Shutdown EDG A-SA From MCB For Maintenance – Field Flash Stays Energized

In accordance with OP-155, Diesel Generator Emergency Power System

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT ______

Examiner's Signature: ______________________ Date: ______________

Initial Conditions:	• The Unit is operating 100% power • The ‘A’ EDG is running in parallel with the grid to support testing of the governor • OST-1013, 1A-SA Emergency Diesel Generator Operability Test Monthly Interval Modes 1-2-3-4-5-6 is NOT in progress • Testing of the governor is complete • The previous shift has reduced the EDG load from 6.3 MW to 2.3 MW and 1 MVAR over the last 30 minutes per OP-155, Emergency Diesel Generator Section 7.1

Initiating Cue:	• You are the BOP. The CRS directs you to continue shutting down the ‘A’ EDG using OP-155 Section 7.1.2 starting at Step 4 .

Facility:	Harris Nuclear Plant	Task No.:	015001H101
Task Title:	Power Range NI Gain Adjustment	JPM No.:	2020 NRC Exam Simulator JPM CR g
K/A Reference:	015 A4.02 RO 3.9 SRO 3.9		**ALTERNATE PATH - NO**

Examinee: ________________________ NRC Examiner: ______________

Facility Evaluator: ________________________ Date: ________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom ______ Simulator X Plant ________

READ TO THE EXAMINEE

I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.

Initial Conditions:	• The unit is at 100% equilibrium conditions. • Following maintenance on PR Channel NI-41, all required testing has been completed and the channel is ready to be returned to service. A calorimetric has just been performed per OST-1000, “Power Range Heat Balance, ERFIS Online Calculation, Daily Interval, Mode 1 (Above 15% Power).” The OST requires that an NI gain adjustment be performed. • The calculated power is 99.88%. Indicated power on PR channel NI-41 at the time of the calorimetric was at its current value. • Rod Control is in Automatic.

Initiating Cue:	You are to perform the Power Range NI Gain Adjust for PR channel NI-41 in accordance with OP-105, “Excore Nuclear Instrumentation,” Section 8.3 and Attachment 2.

NOTE: The Simulator Operator will be required to adjust the Pot Setting for NI-41 per the Simulator Setup instructions each time this JPM is administered.

***Evaluator NOTE*:**	**The candidate should be briefed outside of the simulator prior to performing this JPM. Provide them with a copy of OP-105, pages 1-8, 19, 37-48. This will allow them to review the Precautions and Limitations associated with OP-105 and have time for a task review of the steps. Expect the candidate to take about 10 - 15 minutes to complete this review.**

Task Standard:	Gain has been adjusted within limits for PR Channel N-41
Required Materials:	None
General References:	OP-104, Rod Control System, Rev. 45 OP-105, Excore Nuclear Instrumentation Rev. 30
Handout:	OP-104, Rev. 45, pages 1- 8, Prerequisites, P&L's OP-104, Rev. 45, page 54, and Section, 8.15, Placing Rod Control In Manual For Testing/Plant Conditions OP-105, Rev. 30, pages 1 – 7, Prerequisites, P&L's OP-105, Rev. 30, page 19, Section, 8.3, Power Range NI Gain Adjustment OP-105, Rev. 30, pages 37 – 48, Attachment 2, Power Range NI Gain Adjustment
Time Critical Task:	No
Validation Time:	15 minutes

PERFORMANCE STEP	**CRITICAL STEP JUSTIFICATION**
Step 4	Must determine the desired indication, including sign, of N-41 to return the instrumentation to the current power level.
Step 6	Must place Rod control from Automatic control to Manual control to prevent unnecessary reactivity change from occurring due to control rod motion that would occur while adjusting NI gain on channel N-41.
Step 14	Must adjust the gain pot in CW direction until indicated power is within 0.5% of value determined to meet the acceptance criteria prior to relocking the pots. Note: the procedure acceptance is within 0.2% but due to the sensitivity of the pots, acceptance for this step is ± 0.5%

2020 NRC Exam - SIMULATOR SETUP

Simulator Operator

- IC will be saved in IC-152 once the Audit Exam is completed (Reset to IC-152)
- IC was not saved on the simulator (Password "NRC3sros")
- Put reactivity data sheets for IC-19 and MOL on status board
- Go to run
- Place Meter Rate on front of PR channel NI-41 to Fast
- Unlock gain pot on the front of PR channel NI-41
    - **If this is the first performance of the day, swipe the pot by rotating it several turns in each direction to clean it. The pots can become very sensitive over time.**
- Slowly adjust the gain to 2.70 (verify that it indicates approximately 3 % - 4 % below the other 3 PR channels)
- Lock gain pot
- Place Meter Rate on front of PR channel NI-41 to Slow
- Silence and Acknowledge annunciators

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner. (The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

This portion of Simulator setup is now compete and must be completed between each evaluation.

To recreate the IC setup for this JPM:

- Initial Simulator IC was IC-19
- Initialize to IC-19, go to RUN
- Place Rod Control in MAN
- Place Meter Rate on front of PR channel NI-41 to Fast
- Unlock gain pot on the front of PR channel NI-41
- **If this is the first performance of the day, swipe the pot by rotating it several turns in each direction to clean it. During Validation it was noted the pots can become very sensitive over time.**
- Slowly adjust the gain to 2.70 (verify that it indicates approximately 3 % - 4 % below the other 3 PR channels)
- Ensure any alarms caused by this adjustment are acknowledged
- Lock gain pot
- Place Meter Rate on front of PR channel NI-41 to Slow
- Place Rod Control in AUTO
- Silence Acknowledge and Reset Annunciators
- Freeze and Snap these conditions to your exam IC

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ____________

OP-105 Attachment 2 Notes and Caution prior to step 1

Performance Step: 1

NOTE: Calculated power shall be that power calculated by OST-1000, OST-1004 or other applicable plant procedures.

NOTE: If the indicator on the NI drawers is not available, the corresponding ERFIS point may be used. The following computer points update every two seconds, and can be used for initial adjustment:

- **ANM0120 NI-41 PR CHANNEL**
- **ANM0121 NI-42 PR CHANNEL**
- **ANM0122 NI-43 PR CHANNEL**
- **ANM0123 NI-44 PR CHANNEL**

The following computer points are the one minute average of the points above, and are used in recording indicated power and making final determinations on adequacy of the adjustment:

- **ANM0120M NI-41 PR CHANNEL**
- **ANM0121M NI-42 PR CHANNEL**
- **ANM0122M NI-43 PR CHANNEL**
- **ANM0123M NI-44 PR CHANNEL**

Caution: To prevent a possible non-conservative adjustment being made, no significant power decreases should be made between the time of performance of the calorimetric and the following adjustments.

Standard: Reads and place keeps notes and caution

Comment:

OP-105 Attachment 2 Step 1

Performance Step: 2

MARK portions of Attachment 2 N/A for any NI not being adjusted as follows:
IF NI-41 will not be adjusted, THEN MARK the following N/A:
IF NI-42 will not be adjusted, THEN MARK the following N/A:
IF NI-43 will not be adjusted, THEN MARK the following N/A:
IF NI-44 will not be adjusted, THEN MARK the following N/A:

Standard:

Determines NI-42, NI-43 and NI-44 will not be adjusted and marks through the applicable section with N/A.

Comment:

Evaluator Note:	**The candidate should be allowed to complete this step as part of the pre-job brief prior to entering the simulator for evaluation.**

OP-105 Attachment 2 Step 2

Performance Step: 3

DETERMINE the difference, including sign, between the calculated power (from OST-1000 or OST-1004) and the indicated reactor power at the time data was obtained as follows:

CALC PWR - N41 IND PWR = N41 DIFFERENCE
99.88 – 96.0 = +3.88

Standard:

Calculates difference

(Determined by subtracting present indicated value of N-41 from 99.88% calculated power.)

Comment:

Evaluator Note:	**You will be asked to initial for IV during the procedure. State that you can assume that the IV has been performed for each step performed.** **The candidate is responsible to ensure each step is completed correctly**

OP-105 Attachment 2 Step 3

✓ **Performance Step: 4**

DETERMINE the desired indication, including sign, of NIS as follows:

N41 PRESENT IND ± N41 DIFFERENCE = N41 DESIRED IND
96.0 + (+3.88) = 99.88

Standard:

Calculates desired N-41 indication to be 99.9%

Determined by algebraically summing N-41 difference from Step 2 and N-41 present indicated value

Comment:

OP-105 Attachment 2 Step 4

Performance Step: 5

Record the as found setting of the GAIN potentiometer on the front of Power Range Drawer B

Standard:

Records setting as 2.70

Comment:

OP-105 Attachment 2 Step 5

✓ **Performance Step: 6** VERIFY the ROD BANK SELECTOR switch is in MANUAL per OP-104 Section 8.15, Placing Rod Control In Manual For Testing/Plant Conditions, to prevent undesired rod movement during the adjustment.

Standard:

Obtains copy of OP-104 Section 8.15 and places Rod Bank Selector switch in Manual position.

OP-104 Section 8.15

Initial conditions: Plant conditions or testing require Rod Control to be in Manual

Step 1. At the MCB, ROTATE the ROD BANK SELECTOR Switch to MAN

Step 2. VERIFY Rod Speed of 48 steps per minute on SI-408

NOTE: OMM-001, Operations Administrative Requirements, suggest a trip limit of Tavg not within 10° of Tref, whether high or low, in stable plant conditions.

Step 3. MAINTAIN Tavg within 2°F of Tref.

Step 4. IF desired WHEN testing is completed or plant conditions have changed, THEN **PLACE** Rod Control in AUTO per Section 5.5.

Evaluator Cue:	**Provide OP-104, Section 8.15 to the candidate at this time.**

Comment:

OP-105 Attachment 2 Step 6

Performance Step: 7

VERIFY the Feed Reg Bypass Valve Controllers are in manual to prevent undesired valve motion during adjustment.
1FW-140, MN FW A REG BYP FK-479.1
1FW-256, MN FW B REG BYP FK-489.1
1FW-198, MN FW C REG BYP FK-499.1

Standard:

Locates MCB switches for Feed Reg Bypass Valve Controllers and verifies that all 3 are in manual

Comment:

OP-105, Attachment 2, N41 Adjustments (sheet 4 of 12)
Step 1

Performance Step: 8

RECORD N41 DESIRED IND from calculation performed in Step 2 on Sheet 3 in the space provided.

N41 DESIRED IND _______________

Standard:

Records desired indication from calculation performed in Step 2 on sheet 3 in the space provided

N41 desired indication (99.88)

Comment:

OP-105, Attachment 2, N41 Adjustments,
Caution prior to Step 2

Performance Step: 9 Caution: Adjustments should NOT be made to a Power Range channel while another channel has tripped bistables. This may cause a reactor trip due to required logic being completed. (Reference CR 97-03027-5)

Standard: Reads and place keeps caution

Comment:

OP-105, Attachment 2, N41 Adjustments, Step 2

Performance Step: 10 VERIFY that there are no PR trip bistables energized on TSLB-3 or TSLB-4, except for trip bistables that are manually blocked.

Standard: Verifies no PR trip bistables energized on TSLB-3 or TSLB-4, with exception of PR High Flux Lo Setpoint, which is manually blocked

Comment:

OP-105, Attachment 2, N41 Adjustments,
Note prior to Step 3

Performance Step: 11 After the GAIN adjustment, the METER RATE switch may be returned to SLOW to evaluate if the adjustment is adequate.

Standard: Reads and place keeps Note

Comment:

OP-105, Attachment 2, N41 Adjustments, Step 3

Performance Step: 12 At N41 power range drawer A, PLACE the METER RATE switch in FAST.

Standard: Places Meter Rate switch to Fast position

Comment:

OP-105, Attachment 2, N41 Adjustments,
Caution prior to Step 4

Performance Step: 13 Adjustment of GAIN potentiometer should be made slowly to avoid producing a RATE TRIP signal.

Standard: Reads and place keeps Caution

Comment:

OP-105, Attachment 2, N41 Adjustments, Step 4

✓ **Performance Step: 14** At N41 power range drawer B, PERFORM the following:

- **a.** UNLOCK GAIN potentiometer.
- **b.** SLOWLY ADJUST GAIN potentiometer until the indicated power is within 0.2% of the DESIRED IND from Step 1.

Standard: Unlocks and slowly adjusts Gain pot in CW direction until indicated power is within 0.2 % of value previously determined in Step 1 **(STEP 1 was 99.9, band is 99.7 to 99.9)**

Comment: Due to the sensitivity of the pots, acceptance for this step is lower limit of 98.9% and upper limit of 100%

OP-105, Attachment 2, N41 Adjustments, Step 5

Performance Step: 15 IF there is insufficient fine gain adjustment using the drawer B gain potentiometer, THEN PERFORM Attachment 3 AND RETURN to Step 4.b: (Otherwise, this Step is N/A)

Standard: N/A's step since adequate adjustment exists

Comment:

OP-105, Attachment 2, N41 Adjustments, Step 6

Performance Step: 16 LOCK GAIN potentiometer(s) in place.

Standard: Locks Gain pot on N-41 in place

Comment:

OP-105, Attachment 2, N41 Adjustments, Step 7

Performance Step: 17 IF a RATE TRIP signal occurs, THEN RESET the RATE TRIP signal before going to the next channel.
(Otherwise this Step is N/A)

Standard: N/A's step Rate Trip should not have occurred

Comment:

OP-105, Attachment 2, N41 Adjustments, Step 8

Performance Step: 18 RECORD the as left GAIN potentiometer setting.

Standard: Records current as left GAIN potentiometer setting in space provided.

Comment:

OP-105, Attachment 2, N41 Adjustments, Step 9

Performance Step: 19 On Drawer A, PLACE the METER RATE switch in SLOW.

Standard: Takes Drawer A Meter Rate switch to SLOW

Comment:

OP-105, Attachment 2, N41 Adjustments, Step 10

Performance Step: 20 RECORD the new indicated power (on drawer A)

Standard: Records the new indicated power on drawer A in space provided

Comment:

OP-105, Attachment 2, N41 Adjustments, Step 11

Performance Step: 21	VERIFY that new indicated power is within 2% of desired indication from Step 1 above.
Standard:	Verifies that new indicated power is within 2% of desired indication from Step 1 above.
Comment:	

OP-105, Attachment 2, Restoration

Performance Step: 22	NOTE: If placing ROD BANK SELECTOR switch in AUTO use OP-104 Section 5.5, Automatic Rod Control.
Standard:	Reads and place keeps Note
Comment:	

Evaluator Cue:	**After completing Attachment 2 up to the Restoration of Rod Control: Evaluation on this JPM is complete.** **END OF JPM** **Inform Simulator Operator to place the Simulator in Freeze.**

STOP TIME: _______________

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam JPM CR g

Power range NI Gain Adjustment
In accordance with OP-105, Excore Nuclear Instrumentation

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	• The unit is at 100% equilibrium conditions. • Following maintenance on PR Channel N-41, all required testing has been completed and the channel is ready to be returned to service. A calorimetric has just been performed per OST-1000, Power Range Heat Balance, ERFIS Online Calculation, Daily Interval, Mode 1 (Above 15% Power). • The calculated power is 99.88%. Indicated power on PR channel NI-41 at the time of the calorimetric was at its current value. • Rod Control is in Automatic

Initiating Cue:	You are to perform the Power Range NI Gain Adjust for PR channel NI-41 in accordance with OP-105, Excore Nuclear Instrumentation, Section 8.3 and Attachment 2.

Facility:	Harris Nuclear Plant	Task No.:	008010H101
Task Title:	Align CCW to Support RHR System Operations	JPM No.:	2020 NRC Exam Simulator JPM CR h
K/A Reference:	008 A4.01 RO 3.3 SRO 3.1	**ALTERNATE PATH - NO**	

Examinee: ________________________ NRC Examiner: ______________

Facility Evaluator: ________________________ Date: ________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom ______ Simulator X Plant ________

READ TO THE EXAMINEE	
I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	• The Unit is in Mode 4, going to Mode 5 • Preparations are underway to place both trains of RHR in service • Both ESW Trains are in service • CCW Pump "A" is running

Initiating Cue:	• Align CCW to support operation of both RHR trains with B train of CCW supplying the non-essential header in accordance with OP-145, Component Cooling Water. • All Section 3.0 Prerequisites are met.

Examiners Note:	• **The candidate should be briefed outside of the simulator prior to performing this JPM. Provide a copy of OP-145, Rev. 80, pages 1-10, 14-17, and 45-49.** • **Inform them that ALL initial conditions are satisfied.** • **The section 8.9 initial conditions should be signed off and section 5.2 initial conditions signed off. This will allow them to review the Precautions and Limitations associated with OP-145 and have time for a task review of the steps. Expect the candidate to take about 10 - 15 minutes to complete this review.**

Task Standard: Two CCW Pumps running and the required CCW flow rate established through both RHR Heat Exchangers

Required Materials: None

General References: OP-145, Component Cooling Water, Rev. 80

Handout: OP-145, Rev. 80, pages 1 – 11, Prerequisites, P&L's

OP-145, Rev. 80, pages 15 – 19, Section 5.2, Starting a Second CCW Pump, **with the Initial Conditions signed off if desired**

OP-145, Rev 80, pages 48 – 53, Section 8.9, Aligning CCW to Support RHR System Operations, **with the Initial Conditions signed off if desired**

OP-145, Rev 80, pages 224 – 225, Attachment 18, RHR HX Outlet and RHR Pump Cooler Outlet Flows As Found / As Left Data

Time Critical Task: No

Validation Time: 25 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 3	Must direct AO to throttle shut 1CC-508 to prevent pump runout when aligning CCW flow to the RHR Hx due to flow rate not within band.
Step 7	Critical because two CCW Pumps are required to support two RHR Trains and other loads.
Step 11	Critical to align flow through RHR HX "A" or heat exchanger will not provide cooling.
Step 15	Critical to isolate Train "A" from Non-Essential Header.
Step 17	Critical to isolate Train "A" from Non-Essential Header.
Step 26	Critical to align flow through RHR HX "B".

2020 NRC Exam - SIMULATOR SETUP

Simulator Operator – NOTE: The setup time for this JPM may take ≥ 5 minutes

- Reset to IC-150
- Password “NRC3sros”
- Go to RUN and wait ~ 10 seconds then silence and acknowledge alarms.

GO TO FREEZE and inform the lead examiner the Simulator is ready. DO NOT GO TO RUN until directed by the lead examiner. (The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

To recreate the IC setup for this JPM:

- Reset to IC-16
- Place LTOPS in NORMAL and establish feed with AFW to prevent distracting alarms
- Start both ESW Pumps
- Throttle 1CC-575 irf ccw080 (n 0 0) 50 0 0
- Shut 1CC-522 irf ccw122 (n 0 0) 0 0 0
- Open 1CC-512 irf ccw083 (n 0 0) 100 0 0
- Check FI-652.1 in normal band 10,000 gpm to 11,000 gpm
- IF VCT makeup occurs during this set up allow the VCT to fill
- Return CCW to normal lineup per OP-145
- Stabilize the plant including AFW flows
- FREEZE and SNAP

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ___________

OP-145, Section 8.9.1 Initial Conditions

Performance Step: 1 Implements procedure

Standard:

- Reviews Sections 1.0 through 4.0.
- Proceeds to Section 8.9.
- Reviews the initial conditions for 8.9
  1. RHR System operation desired for RCS cooldown (YES)

Comment:

OP-145, 8.9.2.1 Notes and Caution prior to step 1

Performance Step: 2 Note: The purpose of this section is to ensure CCW pump runout does not occur. Maximum flow through one CCW pump is 12,650 gpm. This section will ensure that one CCW pump is not supplying both essential cooling loops and the non-essential loop simultaneously.

Note: Normally it is desirable to place both RHR cooling trains in operation in Mode 4. This will require both CCW pumps to be in operation and one train of non-essential supply and return valves to be shut.

Caution: To prevent pump runout when aligning CCW flow to the RHR Hx, verify flow rate to the Non-essential header with one pump running is less than 8500 gpm, as indicated on FI-652.1 (FI-653.1) prior to opening 1CC-147 (1CC-167).

Standard: Reads and place keeps Notes and Caution

Comment:

OP-145, 8.9.2 step 1

✓ **Performance Step: 3**

PERFORM the following to verify total CCW flow rate is between 7850 gpm and 8500 gpm:

- IF SFP 2&3A is in service, THEN THROTTLE SHUT 1CC-508, SFP HX 2&3A CCW Outlet Isolation Valve.
- IF SFP 2&3B is in service, THEN THROTTLE SHUT 1CC-521, SFP HX 2&3B CCW Outlet Isolation Valve.

Standard:

Determines flow is NOT within band and contacts Aux Operator to throttle shut 1CC-508 to their mark

Simulator Operator / Communicator:	When contacted to throttle shut 1CC-508 use Simulator Drawing CCW07 / open window for 1CC-508 and adjust the percent open in three increments to allow the candidate to monitor the progress on ERFIS – you should be in open communication with the candidate during this evolution • 41% to 25% with a 10 second ramp • 25% to 10% with a 10 second ramp This last adjustment will get flow to be within band and you will be instructed to stop. • 10% to 4% with a 10 second ramp

Evaluator Note:	**FI-652.1 reads 8400 gpm and 8200 gpm on ERFIS** **FI-652.1 Tolerance is ± 200 gpm** **Band 8200 / 8600 gpm outside of this band is not acceptable**

Comment:

OP-145, 8.9.2 step 2

Performance Step: 4

IF both trains of RHR cooling are to be placed in service, START the second CCW pump per Section 5.2.

Standard:

Proceeds to Section 5.2 to start CCW Pump “B”.

Comment:

OP-145, 5.2.1

Performance Step: 5 Verifies Initial Conditions

Standard: Notes all Initial Conditions are signed (including the prestart checks)

Contacts Aux Operator to standby for “B” CCW pump start

Simulator Communicator:	When requested: Report you are standing by.

Comment:

OP-145, 5.2.2 Notes and Caution prior to step 1

Performance Step: 6 Note:

- Starting the second pump could cause ΔP fluctuations across REM-01CC-3501ASA (BSB) which may shut solenoid valves 1CC-23 and 1CC-40.
- Starting the second pump may cause flow oscillations which could shut 1CC-252. Re-opening of 1CC-252 should not be attempted until the second pump is secured.
- APP-ALB-005 Windows 1-3, 2-1, and 3-2 are expected alarms when starting the second CCW Pump.

Caution:

- With one CCW pump running and the standby pump capable of an automatic start, ensure a minimum flowrate of 7850 gpm exists as indicated on FI-652.1 (FI-653.1). If both CCW pumps are running OR the CCW trains are separated, a minimum of 3850 gpm per pump is required. This lower flowrate should only be allowed for short durations to accomplish pump swapping or system realignment.

Standard: Reads and place keeps notes and caution

Makes PA announcement for pump start then:

At the MCB, START CCW Pump Train B-SB.

Comment:

OP-145, 5.2.2 step 1

✓ **Performance Step: 7** At the MCB, **START** CCW Pump Train B-SB (A-SA).

Standard:

Selects CCW Pump "B" to start and releases **(critical)**
Verifies pump start indications (not critical)
Contacts Aux Operator to ensure good start (not critical)

Simulator Communicator:	IF contacted OR asked to report on "B" CCW pump start Report the "B" CCW pump had a good start and you will continue to monitor during pump warm up to full operating conditions.

Comment:

OP-145, 5.2.2 step 2

Performance Step: 8 VERIFY flow is greater than or equal to 3850 gpm on FI-653.1 and FI- 652.1.

Standard: Verifies ≥ 3850 gpm on FI-653.1 and FI-652.1.

Comment:

OP-145, 5.2.2 step 3

Performance Step: 9 VERIFY OPEN, 1CC-23 and 1CC-40, REM 3501 A CCW Inlet Solenoid Valve and REM 3501 B CCW Inlet Solenoid Valve respectively.

Standard: Contacts Aux Operator for verification

Simulator Communicator:	**Report: 1CC-23 and 1CC-40 are OPEN**

Comment:

OP-145, 5.2.2 steps 4 and 5

Performance Step: 10 IF 1CC-23 or 1CC-40 shut as a result of starting the CCW pump, THEN INITIATE a CR.

PERFORM one of the following:

- SECURE a second CCW Pump using Section 7.1
- ALIGN CCW to support RHR cooling using Section 8.9

Standard: N/As step 4 and returns to Section 8.9

Comment:

OP-145, 8.9.2 step 3

✓ **Performance Step: 11** OPEN 1CC-147 (~~1CC-167~~), CCW FROM RHR HEAT EXCHANGER A-SA (~~B-SB~~).

Standard: Locates switch and Places 1CC-147 in OPEN (RED indication).

Comment:

OP-145, 8.9.2 Caution prior to step 4

Performance Step: 12 Caution:

With one CCW pump running and the standby pump capable of an automatic start, ensure a minimum flowrate of 7850 gpm exists as indicated on FI-652.1 (FI-653.1). If both CCW pumps are running OR the CCW trains are separated, a minimum of 3850 gpm per pump is required. This lower flowrate should only be allowed for short durations to accomplish pump swapping or system realignment. (Reference 2.6.6)

Standard: Reads and place keeps Caution

Comment:

OP-145, 8.9.2 step 4

Performance Step: 13 VERIFY RHR HX A (~~B~~) out flow is 5600 to 8150 gpm on FI-688A1 ~~(FI-689A1).~~

Standard: Verifies RHR HX A out flow is 5600 to 8150 gpm on FI-688A1.

Comment:

OP-145, 8.9.2 Notes and Caution prior to step 5

Performance Step: 14 Note: Steps 8.9.2.5 and 8.9.2.6 are written to place the non-essential header on 'B' CCW. If desired to place the non-essential header on 'A' CCW, perform steps in parenthesis.

Note: If a leak occurs, and surge tank level is less than 40% (below the divider plate), make up water for the B CCW header will be supplied by demin water. Makeup water for the A CCW header must be supplied by the Reactor Makeup Water System.

Caution: Shutting both 1CC-99 and 1CC-113 will result in the loss of the Nonessential Header.

Standard: Reads and place keeps Notes and Caution

Comment:

OP-145, 8.9.2 step 5

✓ **Performance Step: 15** IF both CCW pumps are in service, CLOSE 1CC-99 (~~1CC-113~~), CCW HEAT EXCHANGER A(~~B~~) TO NONESSENTIAL SUP.

Standard: Locates switch and Closes only 1CC-99 (GREEN indication).

Comment:

OP-145, 8.9.2 Caution prior to step 6

Performance Step: 16 Caution: Shutting both 1CC-128 and 1CC-127 will result in the loss of the Nonessential Header.

Standard: Reads and place keeps Caution

Comment:

OP-145, 8.9.2 step 6

✓ **Performance Step: 17** IF both CCW pumps are in service, CLOSE 1CC-128 (~~1CC-127~~), CCW NONESSENTIAL RETURN TO HEADER A(~~B~~).

Standard: Locates switch and Closes 1CC-128 (GREEN indication).

Comment:

OP-145, 8.9.2 step 7.a.(1)

Performance Step: 18 VERIFY the following:

a. IF both CCW Pumps are in service, PERFORM the following, recording data on Attachment 18:
  - RECORD AS FOUND (AF) RHR Hx A-SA (~~B-SB~~) CCW outlet flow from FCC0688 (~~FCC0689~~).

Standard:

- Determines current reading on MCB indicator FI-688A1
- Circles appropriate step (8.9.2.7.a(1)) in the A Train column of Attachment 18
- Documents the as found (AF) value in the Reading column of Attachment 18

Comment:

OP-145, 8.9.2 step 7.a.(2)

Performance Step: 19 VERIFY the following:

a. IF both CCW Pumps are in service, PERFORM the following, recording data on Attachment 18:

- RECORD AS FOUND (AF) RHR Pump A (~~B~~) Cooler Outlet flow rate from FIS-646 (~~FIS-647~~).

Standard:

- Contacts local operator to determine the current reading on indicator FIS-646
- Circles appropriate step (8.9.2.7.a(2)) in the A Train column of Attachment 18
- Documents the as found (AF) value in the Reading column of Attachment 18

Simulator Communicator:	**Report: The as found value of FIS-646 is 8.4 gpm**

Comment:

OP-145, 8.9.2 step 7.a.(3)

Performance Step: 20 VERIFY the following:

a. IF both CCW Pumps are in service, PERFORM the following, recording data on Attachment 18:

- IF RHR Hx A-SA (~~B-SB~~) CCW outlet flow is NOT within 7850 - 8050 gpm, THEN PERFORM the following:

Standard:

- Determines steps 7.a (3) is N/A

Simulator Operator:	**If asked to adjust flow then reduce the percent open of 1CC-146 on Sim drawing for CCW03 from 46 to 44 to obtain slightly lower flow rate on FI-688A1**

Comment:

OP-145, 8.9.2 step 7.a.(4)

Performance Step: 21 VERIFY the following:

a. IF both CCW Pumps are in service, PERFORM the following, recording data on Attachment 18:

- RECORD throttle position of 1CC-146 (~~1CC-166~~), in DEGREES OPEN.

Standard:

- Contacts local operator to determine the current position of 1CC-146
- Circles appropriate step (8.9.2.7.a(4)) in the A Train column of Attachment 18
- Documents the as found (AF) position in the Reading column of Attachment 18

Simulator Communicator:	**Report: The position of 1CC-146 is 47.5 degrees OPEN**

Comment:

OP-145, 8.9.2 step 7.a.(5)

Performance Step: 22 VERIFY the following:

a. IF both CCW Pumps are in service, PERFORM the following, recording data on Attachment 18:

- IF CCW cooler outlet flow rate is NOT between 7 gpm and 10 gpm as indicated on FIS-646 (~~FIS-647~~), THEN PERFORM the following substeps:

Standard:

- Determines step 7.a (5) is N/A

Comment:

OP-145, 8.9.2 step 7.a.(6)

Performance Step: 23

VERIFY the following:
a. IF both CCW Pumps are in service, PERFORM the following, recording data on Attachment 18:

- RECORD AS LEFT (AL) RHR Hx A-SA (~~B-SB~~) CCW outlet flow from FCC0688 (~~FCC0689~~).

Standard:

- Determines current reading on MCB indicator FI-688A1
- Circles appropriate step (8.9.2.7.a(6)) in the A Train column of Attachment 18
- Documents the as found (AL) value in the Reading column of Attachment 18

Comment:

OP-145, 8.9.2 step 7.a.(7)

Performance Step: 24

VERIFY the following:

a. IF both CCW Pumps are in service, PERFORM the following, recording data on Attachment 18:

- RECORD AS LEFT (AL) RHR Pump A (~~B~~) Cooler Outlet flow rate from FIS-646 (~~FIS-647~~).

Standard:

- Contacts local operator to determine the current reading on indicator FIS-646
- Circles appropriate step (8.9.2.7.a(7)) in the A Train column of Attachment 18
- Documents the as found (AL) value in the Reading column of Attachment 18

Simulator Communicator:	**Report: The as found value of FIS-646 is 8.4 gpm**

Comment:

OP-145, 8.9.2 step 7.a.(8)

Performance Step: 23 VERIFY the following:
a. IF both CCW Pumps are in service, PERFORM the following, recording data on Attachment 18:

- PERFORM component verifications on Attachment 18.

Standard:

- Directs a second operator verify position of 1CC-146

Comment:

OP-145, 8.9.2 step 7.b

Performance Step: 24 VERIFY the following:
b. IF one CCW Pump is in service, THEN PERFORM the following:

Standard:

- Determines step 7.b is N/A

Comment:

OP-145, 8.9.2 Caution prior to step 8

Performance Step: 25 Caution: Do not supply CCW to both RHR Heat Exchangers simultaneously with only one CCW pump running.

Standard: Reads and place keeps note

Comment:

OP-145, 8.9.2 step 8

✓ **Performance Step: 26** IF both trains of RHR cooling are to be placed in service, OPEN 1CC-167 (~~1CC-147~~), CCW FROM RHR HEAT EXCHANGER B-SB (~~A-SA~~).

Standard: Locates switch and Opens 1CC-167 (RED indication).

Comment:

OP-145, 8.9.2 step 9

Performance Step: 27 VERIFY CCW Pump B-SB (~~A-SA~~) flow rate in the required range, as follows:

- CHECK CCW Pump B-SB (~~A-SA~~) flow rate is between 10,000 and 12,500 gpm on MCB indicator FI-653.1 (~~FI-652.1~~). IF flow rate is not between 10,000 and 12,500 gpm, THEN ADJUST the applicable valve.

Standard: Verifies flow rate is between 10,000 and 12,500 gpm on FI-653.1 and there is NO need for flow adjustment and N/As step 9.b.

Comment:

OP-145, 8.9.2 step 10

Performance Step: 28 Locally VERIFY FI-693, CCW Flow Gross Failed Fuel Detector, is between 8 and 12 gpm.

Standard: Contacts Aux Operator to verify flow on FI-693 between 8 and 12 gpm

Simulator Communicator:	**Flow on FI-693 reads 10 gpm**

Comment:

OP-145, 8.9.2 step 11

Performance Step: 29 WHEN CCW is no longer required for RHR Operation, PERFORM the following steps:

Standard: Step is N/A at this time.

Comment:

Evaluator Cue:	**When Step 8.9.2.11 is read: Evaluation on this JPM is complete.** **Announce END OF JPM** **Direct Simulator Operator to place the Simulator in FREEZE.**

STOP TIME: ______________

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Simulator JPM CR h
Align CCW to Support RHR System Operations

In accordance with OP-145, Component Cooling Water

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

JPM CUE SHEET

Initial Conditions:	• The Unit is in Mode 4, going to Mode 5 • Preparations are underway to place both trains of RHR in service • Both ESW Trains are in service • CCW Pump “A” is running

Initiating Cue:	• Align CCW to support operation of both RHR trains with B train of CCW supplying the non-essential header in accordance with OP-145, COMPONENT COOLING WATER. • All Section 3.0 Prerequisites are met.

Facility: Harris Nuclear Plant

Task No.: 344074H504

Title: Locally Start EDGs per OP-155

JPM No.: 2020 NRC Exam In-Plant JPM i

K/A Reference: APE 068 AA1.10 RO 3.7 SRO 3.9

ALTERNATE PATH - YES

Examinee: ______________________

NRC Examiner: ________________

Facility Evaluator: ______________________

Date: ________

Method of testing:

Simulated Performance: __X__

Actual Performance: _______

Classroom ______ Simulator ______ Plant __X__

READ TO THE EXAMINEE	
I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	• AOP-004 has been entered due to a fire in the MCR • 'A' ('B)' Safety bus is not energized due to a SUT fault • EDG 1A-SA (1B-SB) was in standby operation but did not automatically start • AOP-004 has directed that the 'A' ('B') EDG be locally started and 'A' ('B') safety bus energized • Both safety and non-safety Plant DC Distribution Systems are in operation per OP-156.01 to support EDG operation • The manual transfer to LOCAL has been completed at MTP 1A-SA (1B-SB)

Initiating Cue:	• Your position is the Outside Operator • The CRS has directed you to locally start the 'A' ('B') EDG IAW OP-155 Section 8.14.2.

Evaluator:	**At this time provide the student with a copy of OP-155, Section 8.14, signed off up to 8.14.1, step 4 and the student Initiating Cue for the EDG the JPM will be performed on.** **This should be the NON- protected train EDG based on discussion with Shift Manager.**

Task Standard: EDG ‘A’ (‘B’) is locally started in accordance with OP-155

Required Materials: Standard PPE
Attachments 1 and 2, Pictures of K1 relay **(Optional)**

General References: OP-155 (Rev. 91)
APP-DGP-001 (Rev. 34)

Handout: OP-155, Rev. 91, pages 1 – 14, Prerequisites, P&L’s

OP-155, Rev. 91, pages 88 – 94, Section 8.14, Local Manual Start with an Emergency Bus Deenergized, **signed off up to 8.14.1 Step 4 if desired.**

Time Critical Task: No

Validation Time: 15 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
6	Depressing the STOP pushbutton will reset the starting circuit and allow the EDG to be started. If this pushbutton is NOT depressed the EDG will not start and the bus will remain de-energized.
13	Depressing the START pushbutton will start the EDG. The EDG must be operating to power the emergency bus.
18	Required to reset K1 relay to allow EDG to flash.

PERFORMANCE STEP	ALTERNATE PATH JUSTIFICATION
17	Generator field fails to automatically flash requiring operator action to reset the K1 relay to allow the generator field to flash.

BEFORE YOU START THIS JPM

IN-PLANT JPM SAFETY CONSIDERATIONS:

CAUTION: EQUIPMENT MAY AUTO START OR MAY BE ENERGIZED

- SIMULATE ONLY - DO NOT OPERATE ANY ACTUAL PLANT EQUIPMENT!!!

Before entering the performance location of this JPM, ensure you **AND** the candidate have the proper PPE for the area you are going to go to or will travel through to get there.
Avoid contacting any plant equipment.
Follow ALARA practices in the RCA.
Do NOT remove ladders from their storage locations. Have the candidate simulate obtaining and using a ladder if one would be needed during the actual performance of this task.

NOTE:	**Add one minute for Take a Minute checks.**

Start time begins when the candidate is briefed outside the Blue Heaven conference Room

START TIME: ____________

Proceeds to 1A-SA (1B-SB) EDG

OP-155 Section 8.14.1 Notes prior to Initial Conditions

Performance Step: 1 NOTE: Equipment applicable to B train is shown in parenthesis.

NOTE: If power is NOT available to 1D131-3 (1E231-3), Engine Control Panel, the ECP Temperature Indication System and ENGINE HOURS meter are de-energized.

Standard: Operator reads and placekeeps notes

Comment:

OP-155 Section 8.14.1 Initial Conditions

Performance Step: 2

1. EDG 1A-SA (1B-SB) is not in operation. **(YES)**
2. AOP-004 has directed EDG to be started. **(YES)**
3. Both safety and non-safety Plant DC Distribution System in operation per OP-156.01 to support EDG operation. **(YES)**
4. Attachments 1, 3, 4 (1A-SA) or 2, 3, 5 (1B-SB) are complete. **(YES – the EDG was in standby and ready for an emergency start so all Attachments for these lineups have been previously completed)**

Standard: Reviews Initial Conditions 1 – 4 as complete

Evaluator Cue:	**If CRS is called at the ACP about the initial conditions then cue that the initial conditions are satisfied.**

OP-155 Section 8.14.2 Note prior to Step 1

Performance Step: 3 NOTE: Equipment applicable to B train is shown in parenthesis. Relay 43T-DG6/SA is N/A if transferring B train relays.

Standard: Operator reads and placekeeps notes

Comment:

OP-155 Section 8.14.2 Step 1

Performance Step: 4 IF necessary, THEN, At Main Transfer Panel 1A-SA (1B-SB), PERFORM a manual transfer to LOCAL by placing the following relays in TRANSF:

Relay	Position
43T-DG1SA/1082 (43T-DG1SB/1085)	TRANSF
43T-DG2SA/1082 (43T-DG2SB/1085)	TRANSF
43T-DG3SA/1082 (43T-DG3SB/1085)	TRANSF
43T-DG4SA/1082 (43T-DG4SB/1085)	TRANSF
43T-DG5SA/1082 (43T-DG5SB/1085)	TRANSF
43T-DG6SA/1082	TRANSF

Standard: **Initials step 1 completed**
(Provided in the JPM initial conditions)

Evaluator Cue:	**IF asked or if they are going to perform step 1 then CUE:** **The Main Transfer Panel 1A-SA (1B-SB) relays have been placed in LOCAL by another operator.**

Comment:

OP-155 Section 8.14.2 Step 2

Performance Step: 5 **ENSURE** the following:

a. NO non-emergency trips are active.

b. At GCP, **ENSURE** the UNIT-PARALLEL switch in PARALLEL.

Standard: Operator verifies on the EDG control panel that there are NO non-emergency trips active and the UNIT-PARALLEL switch is in the PARALLEL position

Evaluator Cue:	**(when checked) The non-emergency trip windows are clear**

Comment:

OP-155 Section 8.14.2 Step 3

✓ **Performance Step: 6** IF the FAILED TO START annunciator is in, THEN DEPRESS the STOP pushbutton

(**critical step** is to depress the STOP pushbutton; not the resetting of the annunciator)

Standard: Operator checks annunciator window G-6 clear

Evaluator Cue:	**Annunciator window G-6 "Failed to Start" is lit**

✓ **Standard:** Operator depresses 'RED' STOP pushbutton

Evaluator Cue:	**Annunciator window G-6 is slow flashing**

Standard: Operator depresses the alarm functions reset pushbutton

Evaluator Cue:	**Annunciator window G-6 is clear**

Comment:

OP-155 Section 8.14.2 Step 4

Performance Step: 7 At ECP, **ENSURE** the following OPERATIONAL MODE indicator lights are *LIT*:

a. A CONTROL CIRCUIT
b. B CONTROL CIRCUIT

Standard: Operator checks control circuit lights lit

Evaluator Cue:	**(when checked)** **The control circuit light for A Control Circuit is lit** **The control circuit light for B Control Circuit is lit**

Comment:

OP-155 Section 8.14.2 Step 5

Performance Step: 8 PERFORM a general inspection of the EDG, looking for any obvious reasons that the EDG failed to start

Standard: Operator performs inspection

Evaluator Cue:	**Allow the candidate 1 or 2 minutes to describe the actions for performing the inspection then cue them there are no obvious signs of damage and all indications associated with this EDG are as you see them now.**

Comment:

OP-155 Section 8.14.2 Step 6

Performance Step: 9 ENSURE the Fuel Limit Cylinder has retracted

Standard: Operator Verifies **the** Fuel Limit Cylinder has retracted

Evaluator Cue:	**NOTE: The examinee may want to climb on the EDG to verify where the Fuel Limit Cylinder is and inspect the current position. Direct them to use a flashlight and describe how they would verify the Fuel Limit Cylinder has retracted. (SAFETY FIRST)** ***On the left side of the mechanical governor is where the Fuel Limit Cylinder is located. On the engine side of the cylinder a rod extends and will contact a bolted on flat stock piece attached to the fuel rack rod. The fuel limit cylinder rod extends and contacts this piece to prevent full fuel rack motion. Looking at the back of the cylinder you can see if the rod that protrudes out of it is retracted or extended.*** **Cue: The Fuel Limit Cylinder (rod) has retracted**

Comment:

OP-155 Section 8.14.2, Note before step 7

Performance Step: 10 NOTE: If starting air receiver pressures are low, but still above 100 psig, isolating one of the receivers prior to attempting to start the EDG will maximize the potential number of start attempts

Standard: Operator reads and placekeeps notes

Comment:

OP-155 Section 8.14.2 step 7

Performance Step: 11 **ENSURE** at least one starting air receiver is greater than 100 psig

Standard: Operator verifies at least one starting air receiver is greater than 100 psig

Evaluator Cue:	**Pressures are what you see - (current values)**

Comment:

OP-155 Section 8.14.2, Notes before step 8

Performance Step: 12

NOTE: If the STOP pushbutton was depressed in Step 8.14.2.3, the control circuitry must reset before another start can be attempted. This takes approximately three minutes.

NOTE: The EDG most likely will start in a fast start mode due to the undervoltage. Depending on what failure(s) occurred, however, the EDG may start in the slow start mode.

Standard: Operator reads and placekeeps notes

Evaluator Cue:	**If asked inform the candidate 5 minutes has elapsed**

Comment:

OP-155 Section 8.14.2 step 8

✓ **Performance Step: 13** **DEPRESS** EDG 1A-SA (1B-SB) PUSH TO START pushbutton

Standard: Operator depresses ***(BLACK)*** EDG 1A-SA (1B-SB) PUSH TO START pushbutton

Evaluator Cue:	**1A-SA (1B-SB) EDG has started** **(when checked) The Diesel Engine RPM's are rising and have now stabilized at 450 RPM**

Comment:

OP-155 Section 8.14.2, Caution before step 9

Performance Step: 14

CAUTION: EDG field flashing will occur at 360 to 380 RPM for a normal slow start. EDG field flashing will occur at 190 to 210 RPM for an emergency start. If EDG fails to start field flashing will remain energized resulting in possible fire in GCP control section. Depressing the EDG 1A-SA (1B-SB) STOP pushbutton will de-energize field flashing circuit.

Standard: Operator reads and placekeeps caution

Evaluator Cue:	**Operator may verify proper start of diesel. If operator requests or goes to observe these indication, provide the following information as requested:** • **DG LOCAL CONTROL PANEL AC VOLTMETER – 0 VAC** • **DG GEN FIELD AMMETER – 0 Amps** • **DG frequency is 0 Hz** • **DG FIELD DC VOLTAGE – 0 volts**

Comment:

OP-155 Section 8.14.2 step 9

Performance Step: 15 IF the FAILED TO START annunciator is received, THEN DEPRESS the STOP pushbutton

Standard: Operator checks annunciator and does not depress STOP

Evaluator Cue:	**(when checked) Annunciator G-6 “Failed To Start” is clear**

Comment:

OP-155 Section 8.14.2 notes before step 10

Performance Step: 16

NOTE: If EDG starts but the generator fails to flash:
- The EDG will be operating at 470 rpm with zero field volts.
- The K1 relay must be reset to enable any future field flashing.
- If EDG is left running, the EDG should self flash within 10 seconds, if the K1 relay is reset.

NOTE: Satisfactory field flash conditions are indicated by:
- Generator AC voltage between 6500 and 7200 volts
- Engine speed between 445 and 455 rpm
- Field DC voltage indicates a higher voltage

Standard: Operator reads and placekeeps notes

Evaluator Note:	**IF checking parameters cue these when asked:** • **DG LOCAL CONTROL PANEL AC VOLTMETER – 0 VAC** • **Engine speed is 450 RPM** • **DG 1A-SA FIELD DC VOLTAGE – 0 volts**

Comment:

OP-155 Section 8.14.2 step 10 - ALTERNATE PATH begins

Performance Step: 17 IF EDG starts but the generator fails to flash, THEN PERFORM the following

Standard: Operator identifies the generator failed to flash and implements step 8.14.2.10.

Evaluator Note:	**From the indications provided to the examinee, they should be able to identify that the field has NOT flashed.**

Comment:

Evaluator Note:	**The Attachment 1 pictures will be used once the location of the GCP has been demonstrated. Att. 1 should be shown first. When the operator points out the K1 relay, Att. 2 may be used for close up review of the relay.**

OP-155 Section 8.14.2 step 10.a (ALTERNATE PATH)

✓ **Performance Step: 18** In GCP behind left section door three feet above floor, **RESET** the K1 relay by pushing the reset switch in the direction of the arrow on the K1 Relay reset coil.

Standard: Operator locates and resets relay K1 in the GCP (left section).

Operator should determine the generator field has flashed.

Evaluator Note:	**(When reset) The K1 relay is reset.** **IF checking parameters, provide when asked:** • **DG LOCAL CONTROL PANEL AC VOLTMETER – 6900 VAC** • **Engine speed is 450 RPM** • **DG 1A-SA FIELD DC VOLTAGE – 45 volts**

Comment:

OP-155 Section 8.14.2 step 10.b (ALTERNATE PATH)

Performance Step: 19 **ENSURE** disconnect DS-DP-1A1-SA-13 (DS-DP-1B1-SB-13), Gen 1A-SA (1B-SB) Control Panel, in ON and power is present to panel.

Standard: Operator locates and verifies disconnect DS-DP-1A1-SA-13 (DS-DP-1B1-SB-13) is ON. Operator should also note that steps 8.14.2.10.c through 10.e are now N/A.

Evaluator Note:	**When checked, disconnect DS-DP-1A1-SA-13 (DS-DP-1B1-SB-13) is in the ON position.**

Comment:

OP-155 Section 8.14.2 step 11

Performance Step: 20 **ENSURE** the following:

a. CS-1983SA (CS-2003SB), A (B) EDG Auxiliary Lube Oil Pump, in *AUTO*

b. CS-1984SA (CS-2004SB), A (B) EDG Lube Oil Keep Warm Pump, in *AUTO*.

Standard: Operator verifies AUXILIARY LUBE OIL PUMP control switch in *AUTO*

Evaluator Cue:	**AUXILIARY LUBE OIL PUMP switch is in AUTO**

Standard: Operator verifies LUBE OIL KEEP WARM PUMP control switch in *AUTO*.

Evaluator Cue:	**LUBE OIL KEEP WARM PUMP control switch in AUTO**

Comment:

OP-155 Section 8.14.2 step 12

Performance Step: 21 At ECP, **ENSURE** the following:

a. Engine is running at 445 to 455 RPM.
b. JACKET WATER PRESS rises to 10 to 20 psig.
c. SHUTDOWN SYSTEM ACTIVE light lit.
d. READY TO LOAD light lit.

Standard: Operator verifies Engine is running at 445 to 455 RPM

Evaluator Cue:	**Engine RPM is 450**

Standard: Operator verifies JACKET WATER PRESS increases to 10 to 20 psig

Evaluator Cue:	**JACKET WATER PRESS is 16 psig**

Standard: Operator verifies SHUTDOWN SYSTEM ACTIVE light lit

Evaluator Cue:	**SHUTDOWN SYSTEM ACTIVE** (**Red**) **light is lit**

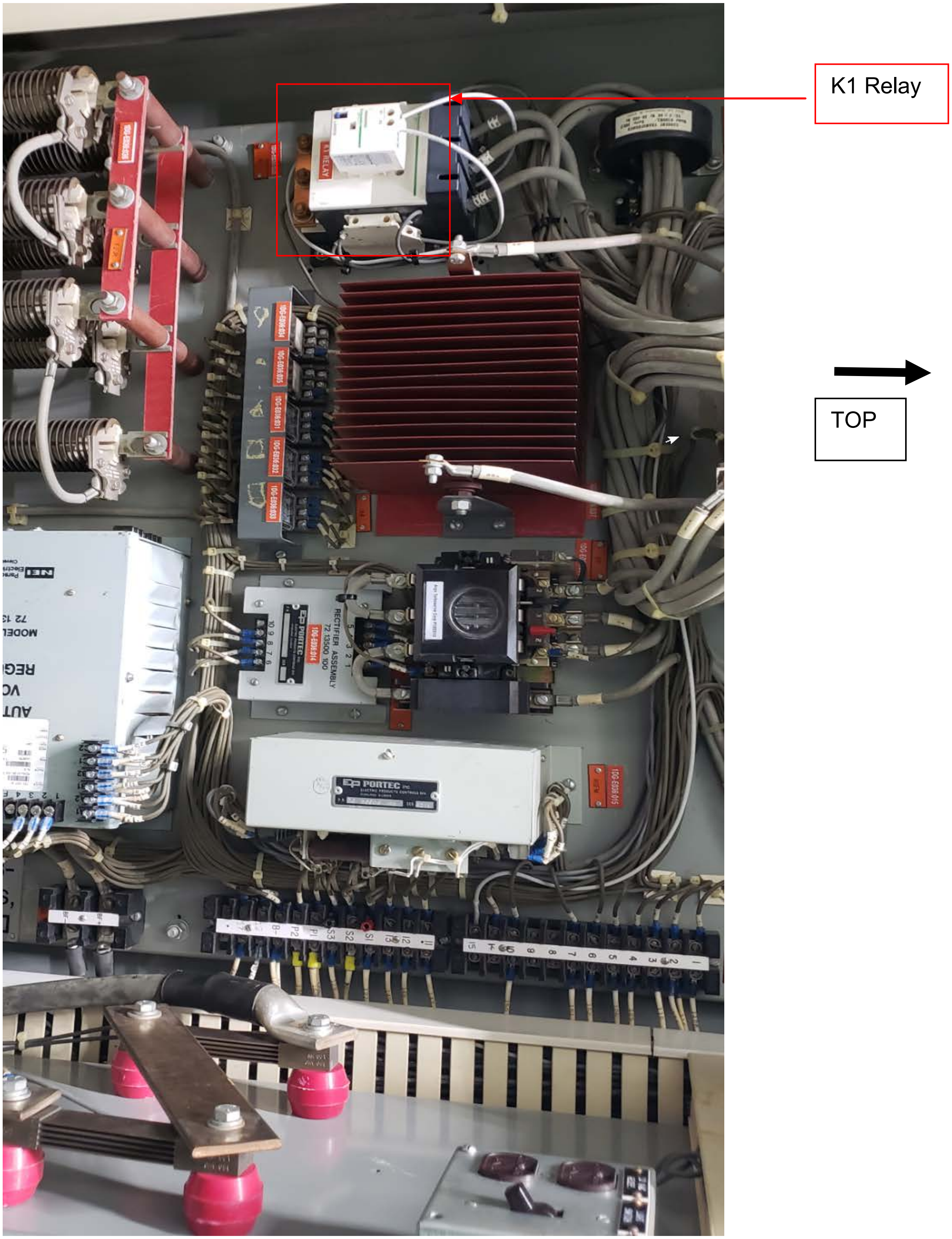
Standard: Operator verifies READY TO LOAD light lit

Evaluator Cue:	**READY TO LOAD** (**Blue**) **light is lit**

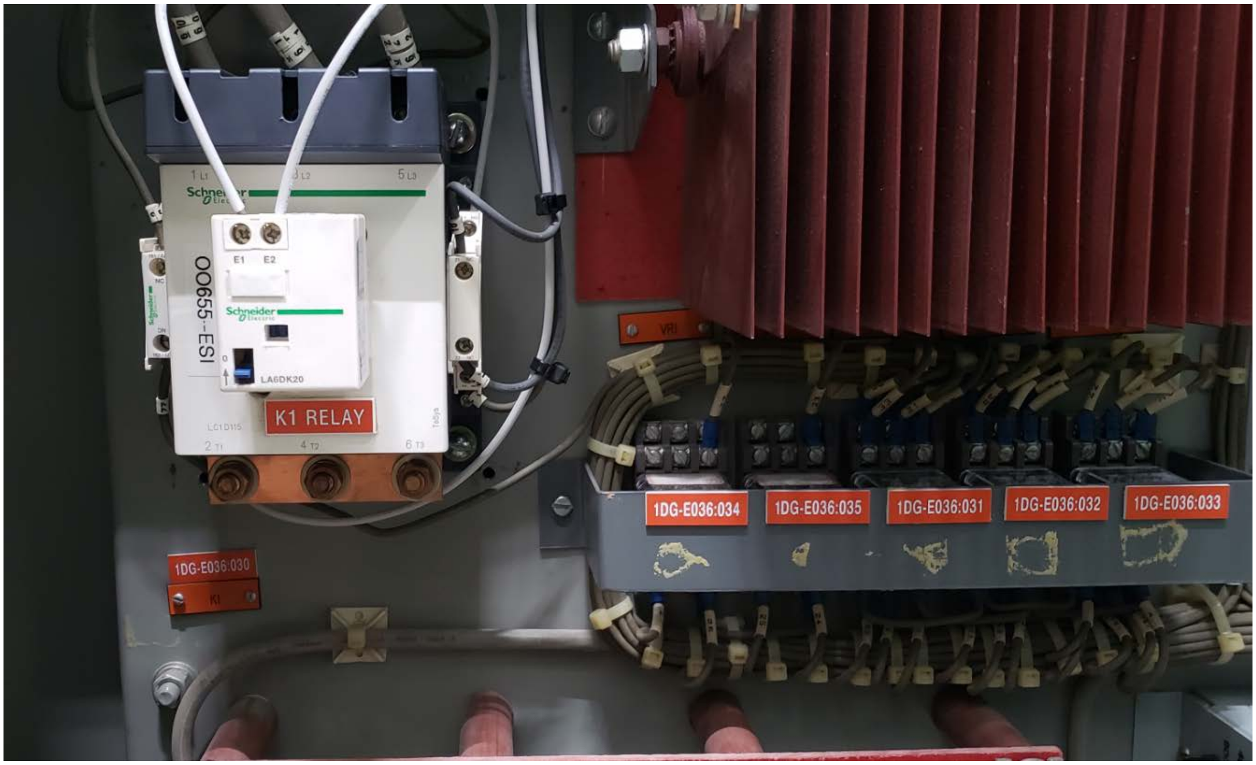
Comment:

STOP TIME: ______________

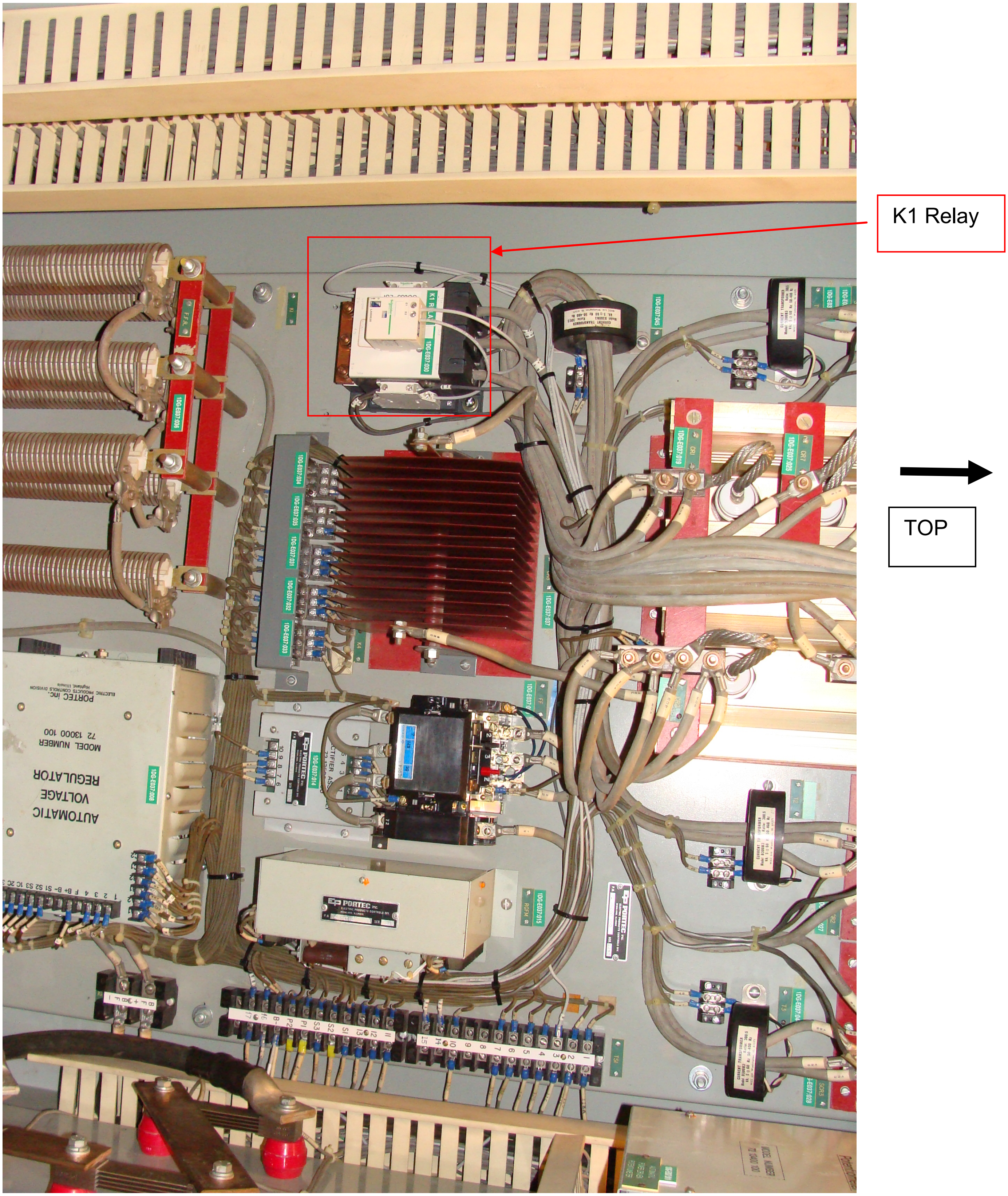
KEY 1A


K1 Relay
TOP
K1 RELAY
RECTIFIER ASSEMBLY
72 13500 100
PORTEC
MODEL 72 13

KEY 2A


OO655--ESI
LA6DK20
K1 RELAY
VRI
1DG-E036:034
1DG-E036:035
1DG-E036:031
1DG-E036:032
1DG-E036:033
1DG-E036:030
K1

KEY 1B



KEY 2B


K1 RELAY
1DG-E037:030
OO655-ESI
E1
E2
Schneider Electric
0
LA6DK20
LC1D115
1 L1
L2
5 L3
2 T1
4 T2
6 T3

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam In-Plant JPM j

Locally Start A-SA or B-SB EDG per OP-155

Examinee's Name:

Date Performed:

Facility Evaluator:

Time to Complete:

Question Documentation

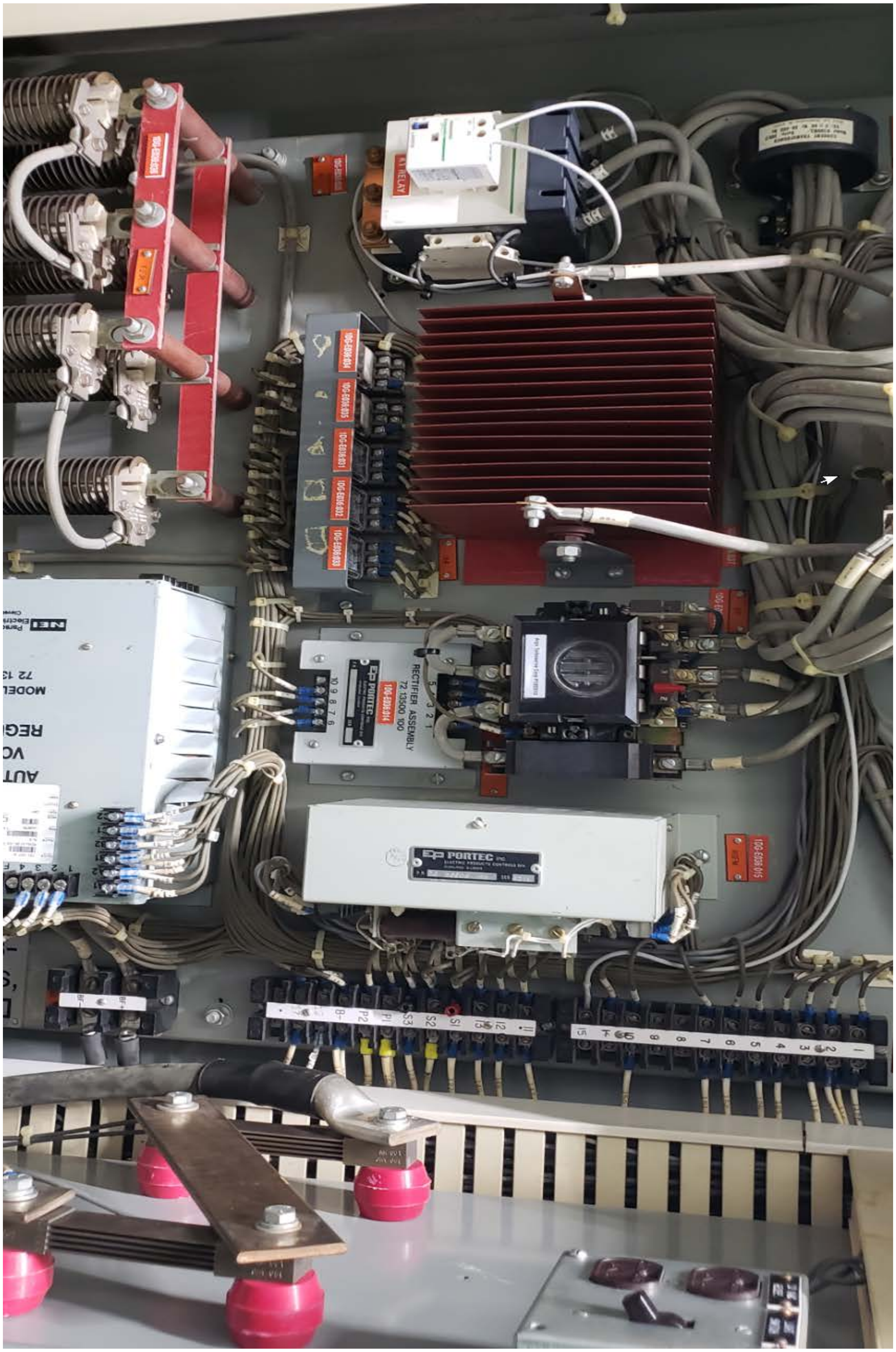
Question:

Response:

Result: PASS ________ FAIL ________

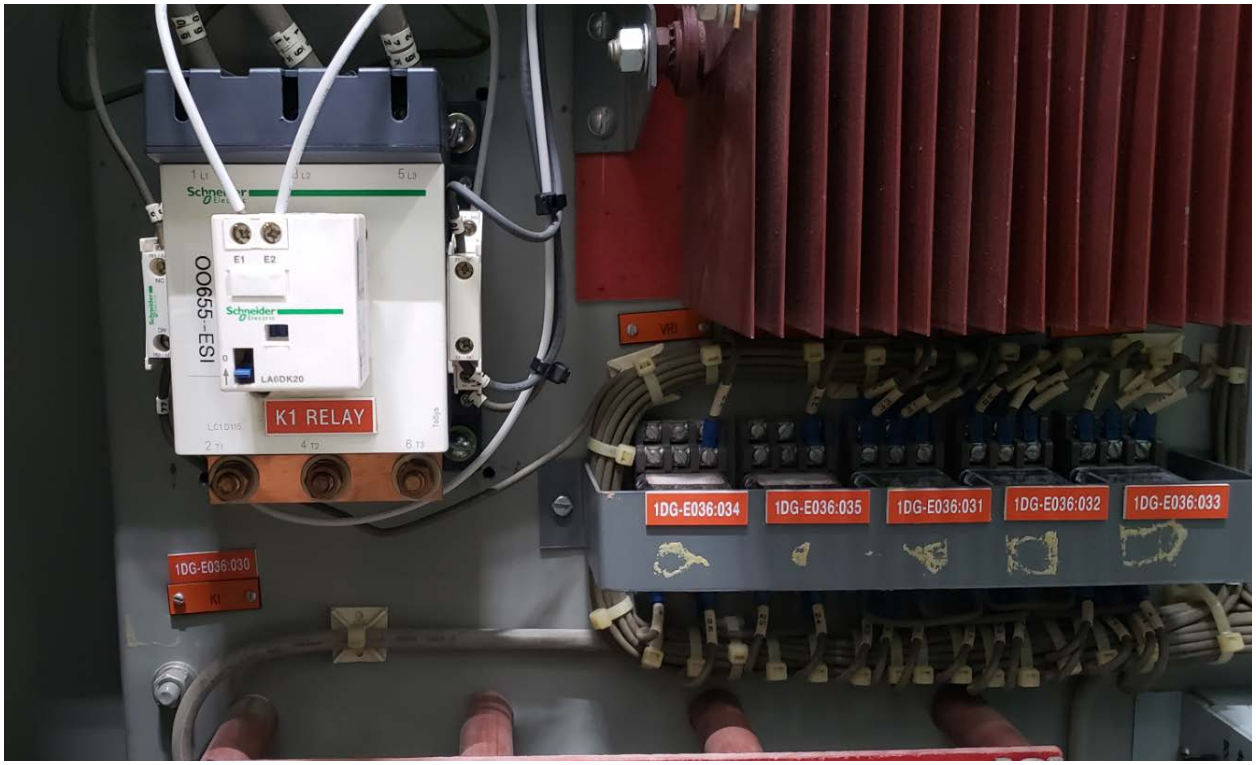
Examiner's Signature: ________________________________ Date: ______________________

Attachment 1A

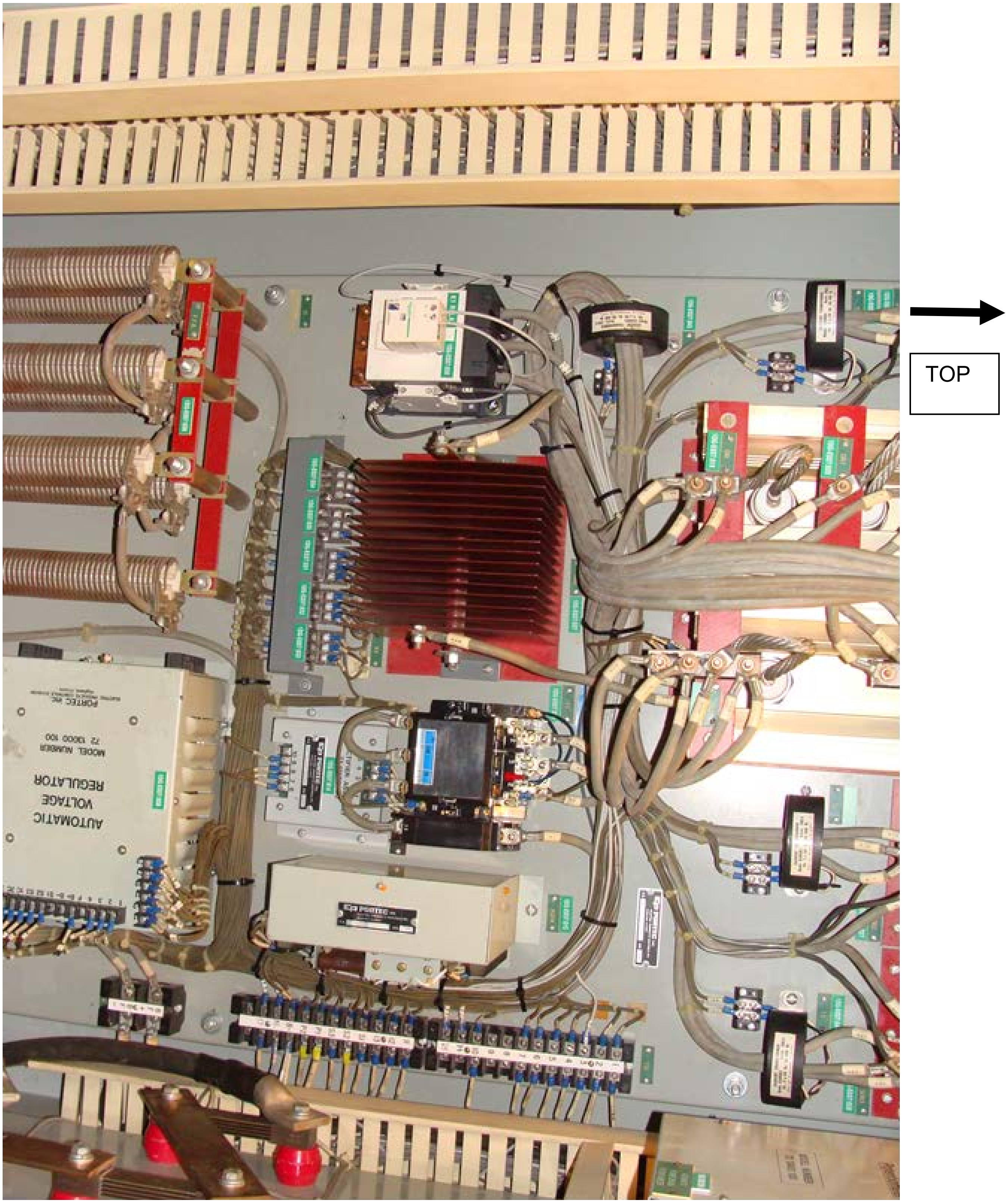


TOP

Attachment 2A



Attachment 1B



Attachment 2B

BEFORE YOU START THIS JPM

IN-PLANT JPM SAFETY CONSIDERATIONS:

CAUTION: EQUIPMENT MAY AUTO START OR MAYBE ENERGIZED

- SIMULATE ONLY - DO NOT OPERATE ANY ACTUAL PLANT EQUIPMENT!!!

Before entering the performance location of this JPM, ensure you **AND** the examiner have the proper PPE for the area you are going to go to or will travel through to get there.
Avoid contacting any plant equipment.
Follow ALARA practices in the RCA.
Do NOT remove ladders from their storage locations. Simulate obtaining and using a ladder if one would be needed during the actual performance of this task.

Initial Conditions:	• AOP-004 has been entered due to a fire in the MCR • ‘A’ (‘B)’ Safety bus is not energized due to a SUT fault • EDG 1A-SA (1B-SB) was in standby operation but did not automatically start • AOP-004 has directed that the ‘A’ (‘B’) EDG be locally started and ‘A’ (‘B’) safety bus energized • Both safety and non-safety Plant DC Distribution Systems are in operation per OP-156.01 to support EDG operation • The manual transfer to LOCAL has been completed at MTP 1A-SA (1B-SB)

Initiating Cue:	• Your position is the Outside Operator • The CRS has directed you to locally start the ‘A’ (‘B’) EDG IAW OP-155 Section 8.14.2.

Facility:	Harris Nuclear Plant	Task No.:	121001H404
Task Title:	Place the ASI System in Standby Alignment (OP-185)	JPM No.:	2020 NRC Exam In-Plant JPM j
K/A Reference:	AA2.67 RO 2.9 SRO 3.1	**ALTERNATE PATH - NO**	
Examinee:	______________________	NRC Examiner:	__________________
Facility Evaluator:	______________________	Date: ________	

Method of testing:

Simulated Performance: __X__ Actual Performance: _______

Classroom ______ Simulator ______ Plant __X__

READ TO THE EXAMINEE I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	• A Normal Plant Heatup is in progress in accordance with GP-002, Normal Plant Heatup From Cold Solid To Hot Subcritical Mode 5 To Mode 3. • Current RCS temperature is 335°F • The 'A' CSIP is in service and providing 9 gpm to all 3 RCP Seals.

Initiating Cue:	• The MCR has directed you to perform OP-185, Alternate Seal Injection, Section 5.1, Automatic Standby Alignment Prior to MODE 3. • Initial Conditions have been met • You are to perform section 5.1.2. **For this task assume you have a set of AO RAB rounds keys.**

Evaluator:	**At this time provide the student with a copy of OP-185, Section 5.1, Marked up through Initial Conditions**

NOTE: Expect that the entry and exit from the RCA will add time to complete this JPM.

Task Standard:	Place the ASI System in Standby Alignment (OP-185)
Required Materials:	Standard PPE Photos of 1CS-828 and 1CS-827 (if available)
General References:	OP-185, Alternate Seal Injection, Rev. 12
Handout:	OP-185, Rev. 12, pages 1 – 4, Prerequisites, P&L's OP-185, Rev. 12, pages 5 – 8, Section 5.1, Automatic Standby Alignment Prior to MODE 3, **signed off up to 5.1.1 Step 7 if desired.**
Time Critical Task:	No
Validation Time:	15 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 3	Required to ensure proper Alternate Seal Injection Standby Alignment Prior to entering Mode 3.
Step 4	Required to ensure proper Alternate Seal Injection Standby Alignment Prior to entering Mode 3.
Step 7	Must locate CS-210.1 switch and place CS-210.1, ASI PUMP, in the AUTO position in order for the ASI pump to work when required.
Step 8	Must locate CS-210.2, SQUIB VALVE 1ASI-21 BYPASS, switch and place the switch to NORMAL for the squib valve to work when required.
Step 9	Must locate CS-210.3, SQUIB VALVE 1ASI-22 BYPASS, switch and place the switch to NORMAL for the squib valve to work when required.
Step 10	Must locate and PLACE breaker PP-1D232-6, Feed to ASI System Control Panel, in the ON. position in order for the ASI pump to work when required.
Step 11	Must locate and PLACE breaker 1D23-1B, Alternate Seal Injection Pump in the ON. position in order for the ASI pump to work when required.

BEFORE YOU START THIS JPM

IN-PLANT JPM SAFETY CONSIDERATIONS:

CAUTION: EQUIPMENT MAY AUTO START OR MAY BE ENERGIZED

- SIMULATE ONLY - DO NOT OPERATE ANY ACTUAL PLANT EQUIPMENT!!!

Before entering the performance location of this JPM, ensure you **AND** the candidate have the proper PPE for the area you are going to go to or will travel through to get there.
Avoid contacting any plant equipment.
Follow ALARA practices in the RCA.
Do NOT remove ladders from their storage locations. Have the candidate simulate obtaining and using a ladder if one would be needed during the actual performance of this task.

NOTE:	**Add one minute for Take a Minute checks.**

Start time begins when the candidate is briefed outside the Waste Process Building 276' Elevation conference Room

START TIME: ____________

OP-185, 5.1.2 Note prior to step 1

Performance Step: 1 The valves in step 5.1.2.2 and 5.1.2.3 are located in the CVCS Filter Valve Gallery.

Standard: Operator reads and placekeeps notes

Comment:

OP-185, 5.1.2.1

Performance Step: 2 IF aligning ASI for OPT-1532 testing, THEN:
a. MARK Steps 5.1.2.2, 5.1.2.3, 5.1.2.10 and 5.1.2.11 N/A.
b. CONTINUE with Step 5.1.2.4.

Standard: Determines steps 5.1.2.2, 5.1.2.3, 5.1.2.10 and 5.1.2.11 are applicable and marks Step 5.1.2.1 N/A

Comment:

OP-185, 5.1.2.2

✓ **Performance Step: 3** Lock Open 1CS-828, ASI Supply Header Upstream Isolation Vlv.

Evaluator Note:	**The location of 1CS-828 and 1CS-827 may be difficult to see when following the candidate into the CVCS filter valve gallery since the area to stand in is small. Have the candidate show you where the valves are located on the valve map outside the CVCS filter gallery before entering the area.** *** There may have be a change in dose conditions from when this JPM was validated. DO NOT ENTER THE AREA TO IDENTIFY THE VALVES IF YOU WILL RECEIVE A DOSE OF $\geq$1milliRem during the performance of this JPM. Instead use the valve map exclusively and conduct a reverse brief on what would be done.**

Standard: Locates 1CS-828, ASI Supply Header Upstream Isolation valve (or **on the valve map** outside the CVCS filter valve gallery) (#51 on the map).

Evaluator Cue:	**Provide feedback that 1CS-828 as found position is locked open.**

Comment:

OP-185, 5.1.2.3

✓ **Performance Step: 4** Lock Open 1CS-827, ASI Supply Header Downstream Isolation Vlv.

Standard: Locates 1CS-827, ASI Supply Header Downstream Isolation valve (or **on the valve map** outside the CVCS filter valve gallery) (#50 on the map).

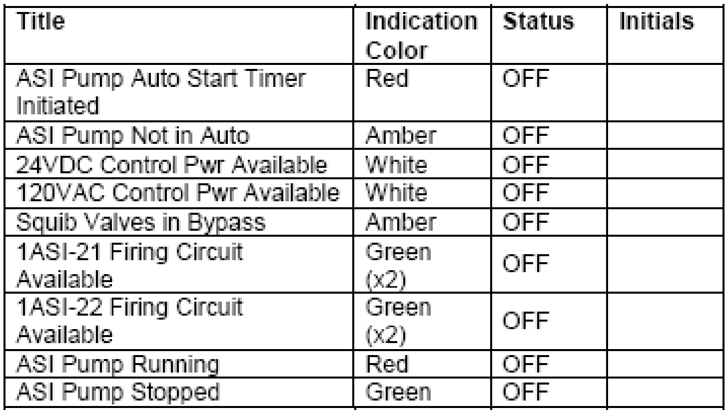
Evaluator Cue:	**Provide feedback that 1CS-827 as found position is locked open**.

Comment:

OP-185, 5.1.2.4

Evaluator Cue:	**Provide feedback as each component is checked that the associated light indication is OFF**.

Performance Step: 5 CHECK the ASI System Control Panel for the following:



Title	Indication Color	Status	Initials
ASI Pump Auto Start Timer Initiated	Red	OFF	
ASI Pump Not in Auto	Amber	OFF	
24VDC Control Pwr Available	White	OFF	
120VAC Control Pwr Available	White	OFF	
Squib Valves in Bypass	Amber	OFF	
1ASI-21 Firing Circuit Available	Green (x2)	OFF	
1ASI-22 Firing Circuit Available	Green (x2)	OFF	
ASI Pump Running	Red	OFF	
ASI Pump Stopped	Green	OFF	

Standard: Locates each indication in step 4 and checks that all lights are off.

Comment:

OP-185 section 5.1.2 Note prior to step 5

Performance Step: 6 The actions in Step 5.1.2.5 will clear ALB-8-2-3, ASI SYSTEM TROUBLE, if no other inputs to the ALB are active.

Standard: Operator reads and placekeeps notes

Comment:

OP-185 section 5.1.2.5.a

✓ **Performance Step: 7** At the ASI System Control Panel, PERFORM the following:

a. PLACE CS-210.1, ASI PUMP, in AUTO.

Evaluator Cue: **The initial switch position of CS-210.1 is OFF**

Standard: Locates CS-210.1 and places CS-210.1, ASI PUMP, in the AUTO position.

NOTE: Both lights are OUT and both lights will STILL BE OUT when CS-210.1 is placed in AUTO

Evaluator Cue: **Once the switch is turned provide feedback:**

CS-210.1 is now in AUTO

Comment:

OP-185 section 5.1.2.5.b (Begin Critical Steps)

✓ **Performance Step: 8** Place CS-210.2, SQUIB VALVE 1ASI-21 BYPASS, in NORMAL

Evaluator Cue:	**The initial switch position of CS-210.2 is in BYPASS**

Standard: Identifies that this step is a critical step. Locates CS-210.2 and determine that switch is in the bypass position. Repositions switch to NORMAL

Evaluator Cue:	**NOTE: Both lights are OUT and both lights will STILL BE OUT when CS-210.2 is placed in NORMAL** **Once the switch is turned:** **CS-210.2 is now in NORMAL.**

Comment:

OP-185 section 5.1.2.5.c

✓ **Performance Step: 9** Place CS-210.3, SQUIB VALVE 1ASI-22 BYPASS, in NORMAL

Evaluator Cue:	**The initial switch position of CS-210.3 is in BYPASS**

Standard: Locates CS-210.3 and determine that switch is in the bypass position. Repositions switch to NORMAL

Evaluator Cue:	**NOTE: Both lights are OUT and both lights will STILL BE OUT when CS-210.3 is placed in NORMAL** **Once the switch is turned:** **CS-210.3 is now in NORMAL.**

Comment:

OP-185 section 5.1.2.6

✓ **Performance Step: 10** PLACE breaker PP-1D232-6, Feed to ASI System Control Panel, to ON.

Evaluator Cue:	**PP-1D232-6, Feed to ASI System Control breaker Panel is OFF**

Standard: Locates PP-1D232-6 and determines that the breaker is OFF. Places breaker to the ON position.

Evaluator Cue:	**Once the breaker is manipulated:** **The breaker is now ON** **NOTE: Candidate may go back to the panel after the breaker is ON to check light conditions. IF they do and they want a response for the light indications then cue:** **24VDC control power available (white light ON)** **120VAC control power available (white light ON)** **ALL 4 Green lights on Firing Circuit Available (green ON)**

Comment:

OP-185 section 5.1.2.7

✓ **Performance Step: 11** PLACE breaker 1D23-1B, Alternate Seal Injection Pump, to ON.

Evaluator Cue:	**1D23-1B, Alternate Seal Injection Pump breaker is OFF** **IF ASKED: (both red and green lights should be OFF)**

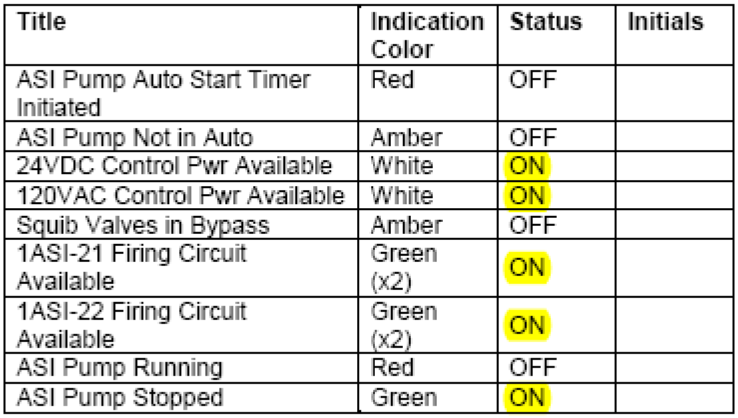
Standard: Locates breaker 1D23-1B, Alternate Seal Injection Pump is OFF and once the breaker is manipulated provide feedback that breaker is now ON.

Evaluator Cue:	**Once the breaker is manipulated:** **The breaker is now ON.** **IF ASKED: green light is LIT on breaker AND above the Auto switch 210.1 on the panel**

Comment:

End Critical Steps
OP-185 section 5.1.2.8

Performance Step: 12 Check the ASI system control Panel for the following:



Title	Indication Color	Status	Initials
ASI Pump Auto Start Timer Initiated	Red	OFF	
ASI Pump Not in Auto	Amber	OFF	
24VDC Control Pwr Available	White	ON	
120VAC Control Pwr Available	White	ON	
Squib Valves in Bypass	Amber	OFF	
1ASI-21 Firing Circuit Available	Green (x2)	ON	
1ASI-22 Firing Circuit Available	Green (x2)	ON	
ASI Pump Running	Red	OFF	
ASI Pump Stopped	Green	ON	

Evaluator Cue:	**As each indicator is read, provide feedback that each light is properly lit as determined from the table in this step.**

Standard: Locates each indication listed in step 8 and verifies that the lights are indicating properly.

Evaluator Cue:	**Once Student reads step 5.1.2.9, Announce End of JPM.**

Comment:

STOP TIME: ______________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam In-Plant JPM j

Place the ASI System in Standby Alignment
In accordance with OP-185

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

BEFORE YOU START THIS JPM

IN-PLANT JPM SAFETY CONSIDERATIONS:

CAUTION: EQUIPMENT MAY AUTO START OR MAYBE ENERGIZED

- SIMULATE ONLY - DO NOT OPERATE ANY ACTUAL PLANT EQUIPMENT!!!

Before entering the performance location of this JPM, ensure you **AND** the examiner have the proper PPE for the area you are going to go to or will travel through to get there.
Avoid contacting any plant equipment.
Follow ALARA practices in the RCA.
Do NOT remove ladders from their storage locations. Simulate obtaining and using a ladder if one would be needed during the actual performance of this task.

Initial Conditions:	• A Normal Plant Heatup is in progress in accordance with GP-002, Normal Plant Heatup From Cold Solid To Hot Subcritical Mode 5 To Mode 3. • Current RCS temperature is 335°F • The 'A' CSIP is in service and providing 9 gpm to all 3 RCP Seals.

Initiating Cue:	• The MCR has directed you to perform OP-185, Alternate Seal Injection, Section 5.1, Automatic Standby Alignment Prior to MODE 3. • Initial Conditions have been met with the exception of step 5.1.1.6. • You are to complete initial conditions and perform section 5.1.2. **For this task assume you have a set of AO RAB rounds keys.**

Facility:	Harris Nuclear Plant	Task No.:	301013H401
Task Title:	Isolate the ECCS Accumulators After a Control Room Evacuation (AOP-004)	JPM No.:	2020 NRC Exam IP JPM k
K/A Reference:	APE 068 AG2.1.30 RO 3.9 SRO 3.4	**ALTERNATE PATH - NO**	
Examinee:	____________________	NRC Examiner:	________________
Facility Evaluator:	____________________	Date:	________

Method of testing:

Simulated Performance: X Actual Performance: _______

Classroom ______ Simulator ______ Plant X

READ TO THE EXAMINEE	
I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	• The control room has been evacuated due to a fire. • A cooldown is in progress in accordance with AOP-004, REMOTE SHUTDOWN. • RCS Pressure is 975 PSIG by PI-402.2.

Initiating Cue:	You are the TB AO and have been assigned to perform AOP-004, Section 3.1, Step 30 – Isolate SI Accumulators.

Task Standard: All accumulators isolated and MOV's de-energized.

Required Materials:

- Standard PPE
- Attachments 1, 2, 3, 4 and 5, ATP Pictures of 1SI-246, 1SI-247, 1SI-248 **(Optional)**
- ***Provide the evaluator with a key for ATP Cabinet (Key #33).***
- ***Discuss with CRS allowing applicants to reset local alarm caused by opening ATP Cabinet door on Sequencer Panel.***

General References: AOP-004, Remote Shutdown, Rev 70

Handout: AOP-004, Rev. 70, page 37, Section 3.1, Step 30 (Pgs. 37)

Time Critical Task: No

Validation Time: 20 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 2	Required action to reenergizes valve motor to allow valve operation
Step 3	Required action to reenergizes valve motor to allow valve operation
Step 4	Repositioning of this valve is required to isolated accumulator water flow path and possible inadvertent injection of nitrogen into the RCS
Step 5	Repositioning of this valve is required to isolated accumulator water flow path and possible inadvertent injection of nitrogen into the RCS

BEFORE YOU START THIS JPM

IN-PLANT JPM SAFETY CONSIDERATIONS:

CAUTION: EQUIPMENT MAY AUTO START OR MAY BE ENERGIZED

- SIMULATE ONLY - DO NOT OPERATE ANY ACTUAL PLANT EQUIPMENT!!!

Before entering the performance location of this JPM, ensure you **AND** the candidate have the proper PPE for the area you are going to go to or will travel through to get there.
Avoid contacting any plant equipment.
Follow ALARA practices in the RCA.
Do NOT remove ladders from their storage locations. Have the candidate simulate obtaining and using a ladder if one would be needed during the actual performance of this task.

NOTE:	**Add one minute for Take a Minute checks.**

Start time begins when the candidate is briefed outside the Blue Heaven conference Room

START TIME: ____________

AOP-004

Performance Step: 1 Obtain locked valve and ATP Cabinet keys.

Standard: Discusses how to obtain keys (ACP Room Key Locker).

Evaluator Note:	**The Evaluator can elect to have the applicant locate the ACP Room Key Locker or to discuss the key acquisition. The key to the ACP Key Locker is in a "break glass" case.**

Evaluator Cue:	• **Provide Handout of AOP-004, Section 3.1, Step 30.** • **Acknowledge discussion and tell applicant to assume that they have the locked valve key.** • **Provide ATP Cabinet key.**

Comment:

AOP-004, Section 3.1, Step 30.a

✓ **Performance Step: 2**

WHEN RCS pressure is 900 to 1000 psig, as indicated on PI-402.2, THEN ISOLATE SI accumulators:
286' RAB / RO with locked valve key
a. UNLOCK AND TURN ON accumulator discharge valve breakers:

- Accumulator A: 1A21-SA-5C (both breakers)
- Accumulator C: 1A21-SA-3D (both breakers)

Standard:

- Locates 1A21-SA-5C, identifies UNLOCKS then places breaker in ON position for both breakers for Accumulator A
- Locates 1A21-SA-3D, identifies UNLOCKS then places breaker in ON position for both breakers for Accumulator C

Evaluator Cue:	**Provide feedback on breaker position:** **Valve indicating lights indicate the valves are OPEN, i.e. Red light ON, Green light OFF.** **Voltage Vision lights indicate valve is energized, i.e. Red lights ON**

Comment:

The locked valve key is on the key ring which is a turnover item for the TB AO watch station. Critical to unlock valve breaker in order to provide power to MOV for operation.

AOP-004, Section 3.1, Step 30.a

✓ **Performance Step: 3**	WHEN RCS pressure is 900 to 1000 psig, as indicated on PI-402.2, THEN ISOLATE SI accumulators: *286' RAB / RO with locked valve key* a. UNLOCK AND TURN ON accumulator discharge valve breakers: • Accumulator B: 1B21-SB-5C (both breakers)
Standard:	• Locates 1B21-SB-5C, identifies UNLOCKS then places breaker in ON position for both breakers for Accumulator B
Evaluator Cue:	**Provide feedback on breaker position.** **Valve indicating lights indicate the valves are OPEN, i.e. Red light ON, Green light OFF.** **Voltage Vision lights indicate valve is energized, i.e. Red lights ON**
Comment:	**The locked valve key is on the key ring which is a turnover item for the TB AO watch station. Critical to unlock valve breaker in order to provide power to MOV for operation.**

Evaluator Note:	**Opening the ATP door actuates an alarm in the control room.** **The Attachment pictures will be used once the location of the ATP has been demonstrated. Att. 1 should be shown first.** **When the operator points out 1SI-246, Att. 2 may be used for close up review of the control switch.** **When the operator points out 1SI-248, Att. 3 may be used for close up review of the control switch.**

AOP-004, Section 3.1, Step 30.b

✓ **Performance Step: 4**

SHUT SI accumulator discharge valves at the Auxiliary Transfer Panels listed:

Cable Vault A / RO with ATP cabinet key

- 1SI-246, Accumulator A Discharge (at ATP A)

Cable Vault A / RO with ATP cabinet key

- 1SI-248, Accumulator C Discharge (at ATP A)

Standard:

- Locates and opens ATP “A” and identifies control switch for 1SI-246 then places switch in SHUT position
- Locates and opens ATP “A” and identifies control switch for 1SI-248 then places switch in SHUT position

Evaluator Cue:	**Provide feedback on switch position.** **Valve indication lights change status at this time, i.e.** **Green light ON, Red light OFF**

Comment: **Critical to close discharge valves to prevent inadvertent discharge during cooldown.**

Evaluator Note:	**Opening the ATP door actuates an alarm in the control room.** **The Attachment pictures will be used once the location of the ATP has been demonstrated. Att. 4 should be shown first.** **When the operator points out 1SI-247, Att. 5 may be used for close up review of the control switch.**

AOP-004, Section 3.1, Step 30.b

✓ **Performance Step: 5**

SHUT SI accumulator discharge valves at the Auxiliary Transfer Panels listed:
Cable Vault B / RO with ATP cabinet key

- 1SI-247, Accumulator B Discharge (at ATP B)

Standard:

- Locates and opens ATP “B” and identifies control switch for 1SI-247 then places switch in SHUT position

Evaluator Cue:	**Provide feedback on switch position.** **Valve indication lights change status at this time, i.e.** **Green light ON, Red light OFF**

Comment: **Critical to close discharge valves to prevent inadvertent discharge during cooldown.**

Evaluator Note:	***The Evaluator can elect to have the candidate discuss the remaining steps since it involves returning to equipment already located and re-opening the breakers that were previously closed.***

AOP-004, Section 3.1, Step 30.c

Performance Step: 6

286' RAB / RO with locked valve key
TURN OFF AND LOCK accumulator discharge valve breakers:

- Accumulator A: 1A21-SA-5C (both breakers)
- Accumulator C: 1A21-SA-3D (both breakers)

Standard:

- Returns to 1A21-SA-5C, identifies OFF then LOCK position for both breakers for Accumulator A.
- Returns to 1A21-SA-3D, identifies OFF then LOCK position for both breakers for Accumulator C.

Evaluator Cue:	**Provide feedback on breaker position.** **Voltage Vision lights indicate valve is de-energized, i.e. Red lights OFF** **Valve indicating lights indicate the valves are SHUT, i.e. Green light ON, Red light OFF**

Comment:

AOP-004, Section 3.1, Step 30.c

Performance Step: 7

286' RAB / RO with locked valve key
TURN OFF AND LOCK accumulator discharge valve breakers:

- Accumulator B: 1B21-SB-5C (both breakers)

Standard:

Returns to 1B21-SB-5C, identifies OFF then LOCK position for both breakers for Accumulator B.

Evaluator Cue:

Provide feedback on breaker position.

Voltage Vision lights indicate valve is de-energized, i.e.

Red lights OFF

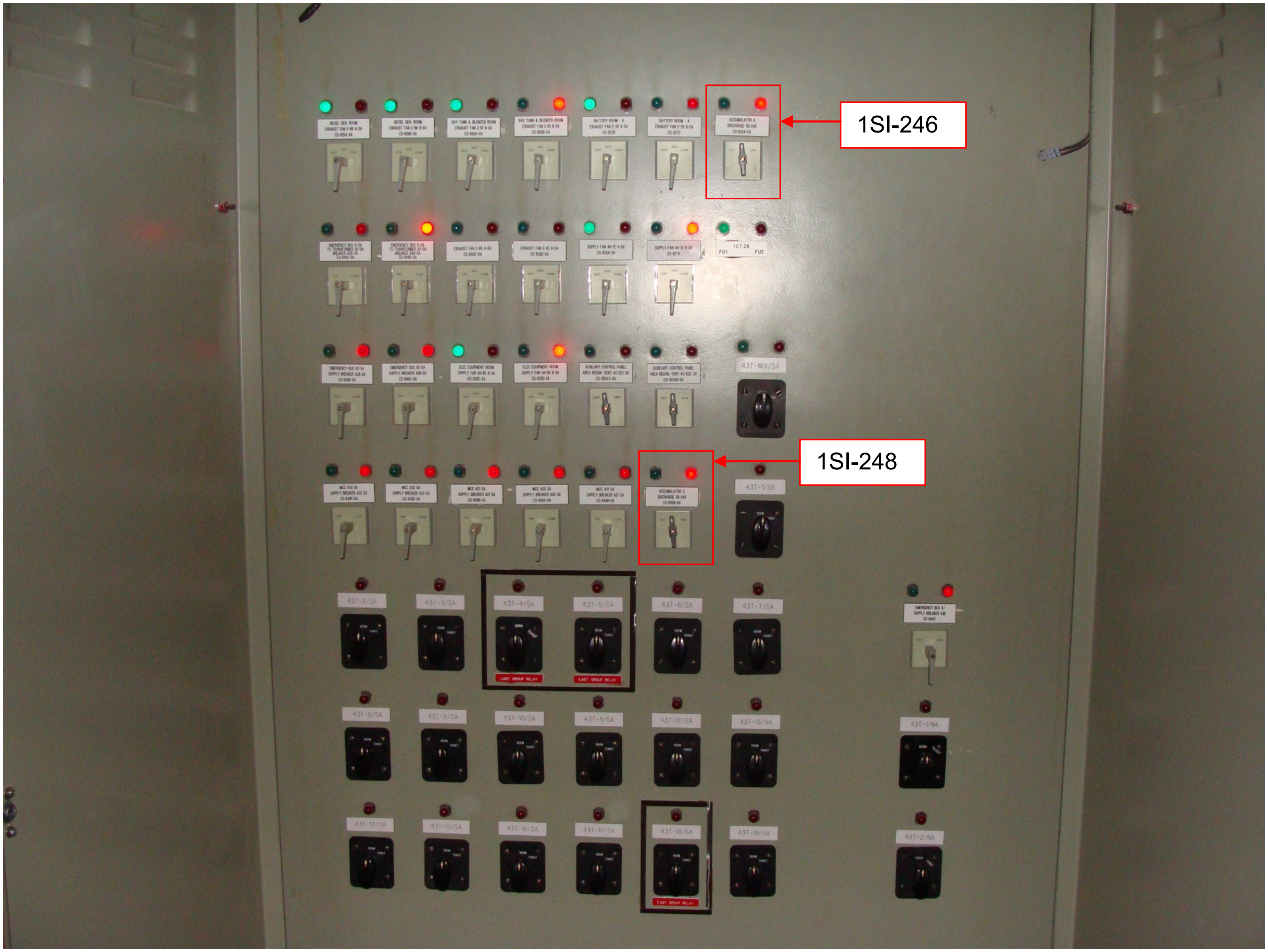
Valve indicating lights indicate the valves are SHUT, i.e. Green light ON, Red light OFF

Comment:

Terminating Cue:	**When all SI Accumulator Discharge Valves are de-energized: Evaluation on this JPM is complete.**

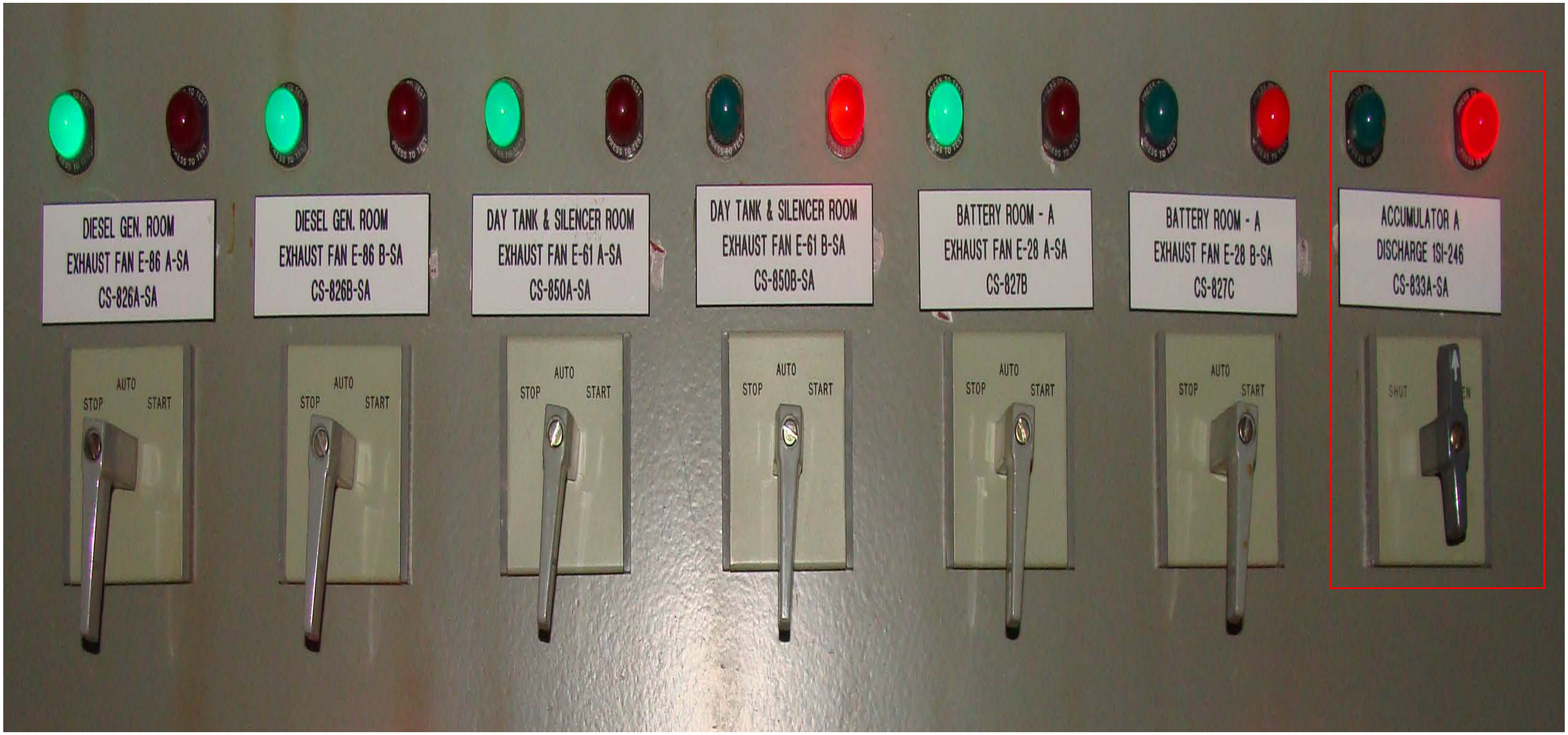
STOP TIME: ______________

KEY



✓ - Denotes Critical Steps

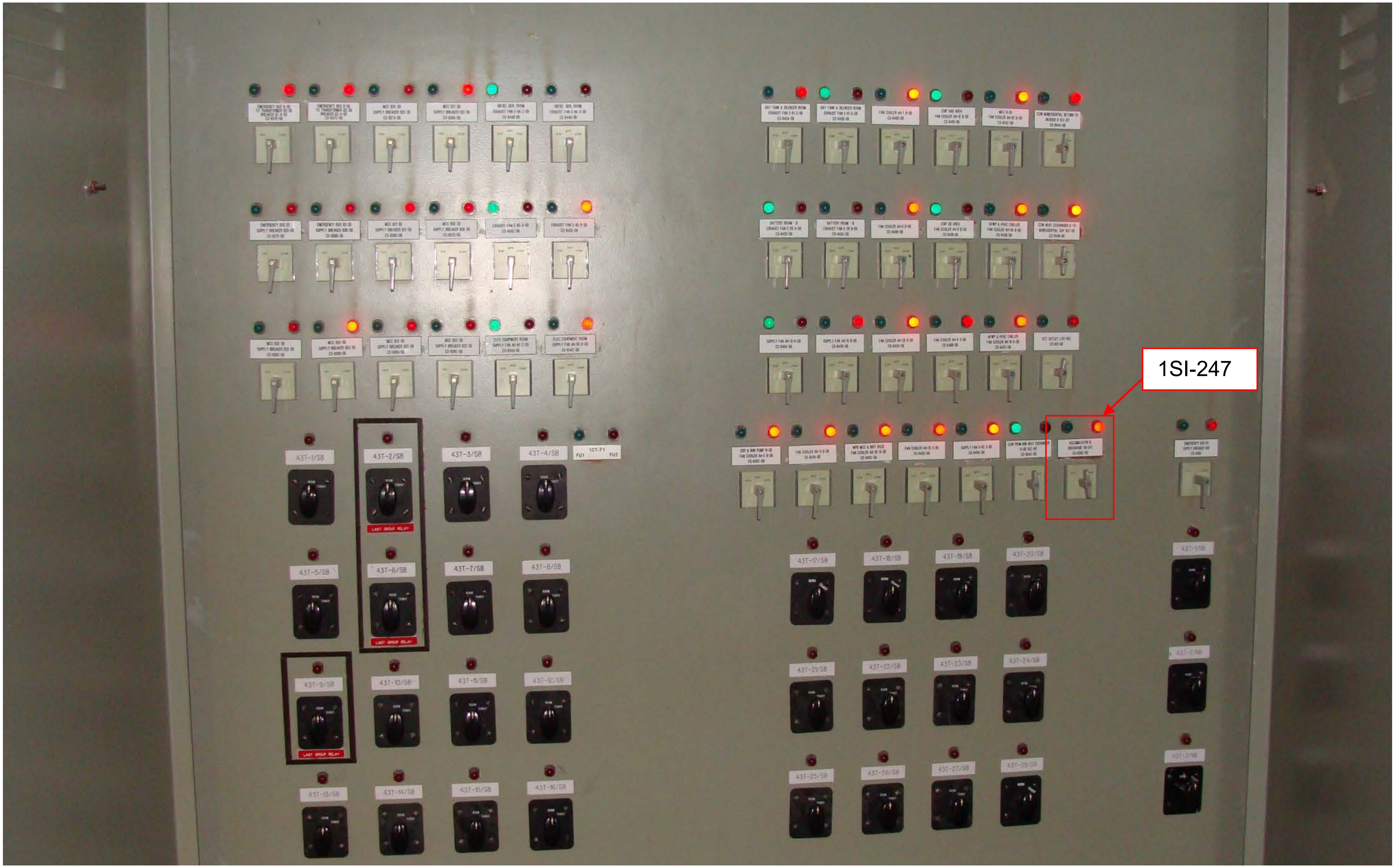
KEY


DIESEL GEN. ROOM
EXHAUST FAN E-86 A-SA
CS-826A-SA
STOP AUTO START
DIESEL GEN. ROOM
EXHAUST FAN E-86 B-SA
CS-826B-SA
STOP AUTO START
DAY TANK & SILENCER ROOM
EXHAUST FAN E-61 A-SA
CS-850A-SA
STOP AUTO START
DAY TANK & SILENCER ROOM
EXHAUST FAN E-61 B-SA
CS-850B-SA
STOP AUTO START
BATTERY ROOM - A
EXHAUST FAN E-28 A-SA
CS-827B
STOP AUTO START
BATTERY ROOM - A
EXHAUST FAN E-28 B-SA
CS-827C
STOP AUTO START
ACCUMULATOR A
DISCHARGE 1SI-246
CS-833A-SA
SHUT

KEY


MCC A32 SA
SUPPLY BREAKER A32 SA
CS-849F-SA
TRIP
CLOSE
MCC A23 SA
SUPPLY BREAKER A23 SA
CS-848D-SA
TRIP
CLOSE
MCC A21 SA
SUPPLY BREAKER A21 SA
CS-848B-SA
TRIP
CLOSE
MCC A35 SA
SUPPLY BREAKER A35 SA
CS-849A-SA
TRIP
CLOSE
MCC A31 SA
SUPPLY BREAKER A31 SA
CS-849B-SA
TRIP
CLOSE
ACCUMULATOR C
DISCHARGE 1SI-248
CS-833B-SA
SHUT
OPEN
43T-1/
NORM

KEY



1SI-247

KEY


CSP & RHR PUMP B-SB
FAN COOLER AH-5 B-SB
CS-840C-SB
FAN COOLER AH-11 B-SB
CS-841A-SB
WPB MCC & INST RACK
FAN COOLER AH-29 1X-SB
CS-843C-SB
FAN COOLER AH-25 X-SB
CS-843D-SB
SUPPLY FAN S-65 X-SB
CS-840A-SB
CCW FROM RHR HEAT EXCHANGER
B-SB 1CC-167
CS-834C-SB
ACCUMULATOR B
DISCHARGE 1SI-247
CS-833C-SB
STOP
AUTO
START
SHUT
OPEN

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam IP JPM k
Isolate the ECCS Accumulators After a Control Room Evacuation
In accordance with AOP-004

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

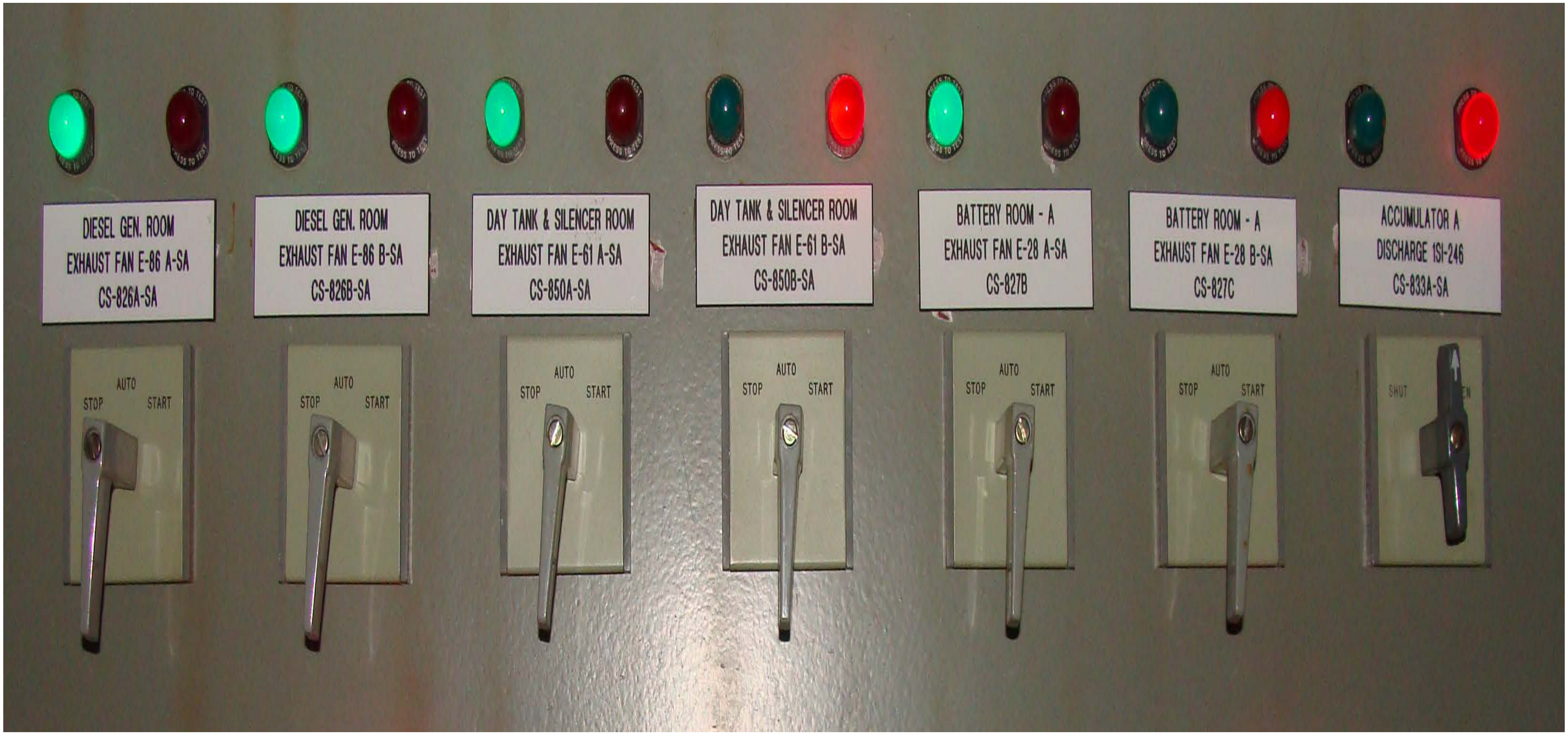
Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

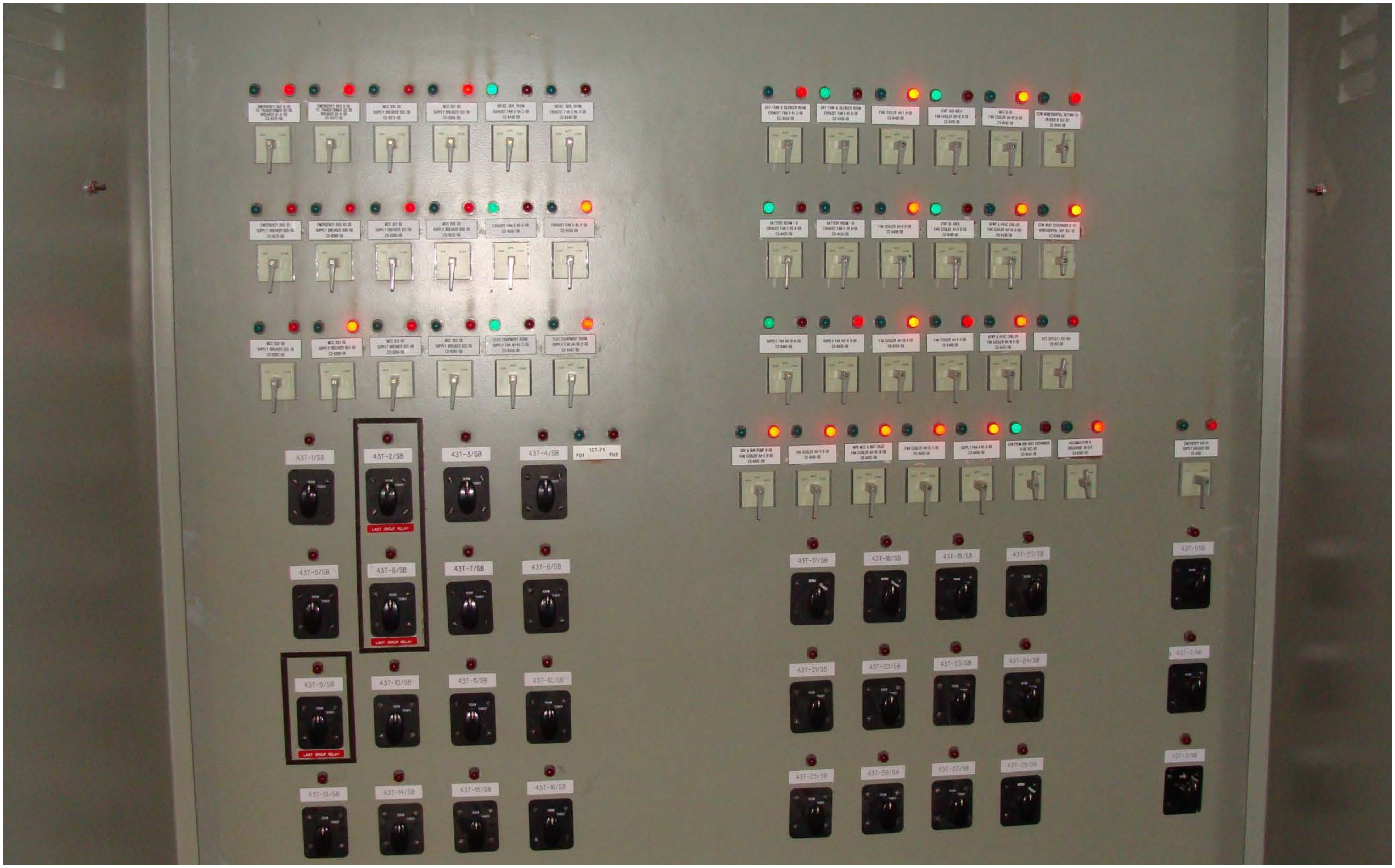
Attachment 1

Attachment 2



Attachment 3

Attachment 4



Attachment 5

BEFORE YOU START THIS JPM

IN-PLANT JPM SAFETY CONSIDERATIONS:

CAUTION: EQUIPMENT MAY AUTO START OR MAYBE ENERGIZED

- SIMULATE ONLY - DO NOT OPERATE ANY ACTUAL PLANT EQUIPMENT!!!

Before entering the performance location of this JPM, ensure you **AND** the examiner have the proper PPE for the area you are going to go to or will travel through to get there.
Avoid contacting any plant equipment.
Follow ALARA practices in the RCA.
Do NOT remove ladders from their storage locations. Simulate obtaining and using a ladder if one would be needed during the actual performance of this task.

Initial Conditions:	• The control room has been evacuated due to a fire. • A cooldown is in progress in accordance with AOP-004, REMOTE SHUTDOWN. • RCS Pressure is 975 PSIG by PI-402.2.

Initiating Cue:	You are the TB AO and have been assigned to perform AOP-004, Section 3.1, Step 30 – Isolate SI Accumulators.

Facility: Harris Nuclear Plant | Task No.: 018003H101

Task Title: Determine AFD with AFD Monitor INOP | JPM No.: 2020 NRC Exam Admin JPM RO A1-1

K/A Reference: G 2.1.25 RO 3.9 SRO 4.2 **ALTERNATE PATH: NO**

Examinee: ____________________ NRC Examiner: ____________________

Facility Evaluator: ____________________ Date: ____________________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom X Simulator ______ Plant ______

READ TO THE EXAMINEE

I will explain the initial conditions, which steps to simulate, discuss, or perform and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.

Initial Conditions:	• The plant was at 90% power, with a load reduction in progress • The load reduction has been stopped to evaluate AFD following oscillations

Initiating Cue:	With the information provided complete OST-1021, Daily Surveillance Requirements to determine Axial Flux Difference. After performing the calculation evaluate the results and circle the response below. When complete return your results to the evaluator.

Task Standard:	All calculations within ± 2% of actual.
Required Materials:	Calculator
General References:	OST-1021, Daily Surveillance Requirements, Rev. 114 OP-163, ERFIS, Rev. 42 Rod Control Manual, Unit One Reactor Operating Data, Rev. 8
Handouts:	OP-163, Rev. 42, pages 1 – 8, Prerequisites, P&L's OP-163, Rev. 42, pages 14 – 15, Section 6.2, (Continuous Use) - Axial Flux Differential (AFD) Monitor Rod Control Manual, Section 2.1, Axial Flux Difference Limits, Rev. 0 OR 2020 NRC Exam Frozen Procedures Folder OST-1021, Rev. 114, pages 44-46, Attachment 5, Axial Flux Difference Log JPM Cue Sheets Pages 16 - 20
Time Critical Task:	No
Validation Time:	10 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 7	If the wrong values are selected then the results will NOT be correct
Step 10	If the wrong Limit is determined a required Tech Spec LCO action could be exceeded
Step 11	If the wrong Limit is determined a required Tech Spec LCO action could be exceeded
Step 12	If operation outside of the acceptable region is allowed to continue fuel damage may result.

Start Time: __________.

OP-163

Performance Step: 1 OBTAIN PROCEDURE (provided in frozen procedure)

Standard: Obtains OP-163 and refers to Section 6.2.

Comment:

OP-163, Section 6.2.2, Step 1.a

Performance Step: 2 REVIEW the automatic or “On Demand” report print-out to verify the following:

- The print-out monitored values are consistent with MCB indications.

Standard: Locates JPM Cue sheet with attached Shift Summary Report

Comment:

OP-163, Section 6.2.2, NOTE prior to Step 1.b

Performance Step: 3 NOTE: There may be rounding off differences between the automatic printout and the latest AFD curve generated by TE-NF-PWR-0809, Target AFD Calculation.

Standard: Operator reads and placekeeps notes

Comment:

OP-163, Section 6.2.2, Step 1.b

Performance Step: 4 REVIEW the automatic or “On Demand” report print-out to verify the following:

- The printout Operating Band Low and Operating Band High values match the latest Axial Flux Difference Limits As A Function of Rated Thermal Power curve as shown in the Rod Manual.

Standard: Locates Rod Manual and reviews Section 2.1, AFD Limits and determines the current limits are

-12.0% to + 8.0% at 100% Reactor Power

-26.0% to + 20.0% at 50% Reactor Power

Comment:

OP-163, Section 6.2.2, Step 2

Performance Step: 5 CHANNEL CHECK the following AFD ERFIS points against MCB indication:

- URE1540 CURRENT CHAN 1 AXIAL FLUX DIFF
- URE1541 CURRENT CHAN 2 AXIAL FLUX DIFF
- URE1542 CURRENT CHAN 3 AXIAL FLUX DIFF
- URE1543 CURRENT CHAN 4 AXIAL FLUX DIFF

Standard: Locates JPM Cue sheet with attached MCB Indication images and compares to information from Shift Summary Report

Comment:

OP-163, Section 6.2.2, NOTE prior to Step 3

Performance Step: 6 NOTE: Only one (1) channel having an unacceptable quality does not make the AFD Monitor inoperable.

Standard: Operator reads and placekeeps notes

Comment:

OP-163, Section 6.2.2, Step 3

✓ **Performance Step: 7** VERIFY the following AFD ERFIS points are restored to processing with acceptable quality codes as defined in Precaution & Limitation Step 4.0.4:

- URE1540 CURRENT CHAN 1 AXIAL FLUX DIFF
- URE1541 CURRENT CHAN 2 AXIAL FLUX DIFF
- URE1542 CURRENT CHAN 3 AXIAL FLUX DIFF
- URE1543 CURRENT CHAN 4 AXIAL FLUX DIFF
- ANM0120M PWR RNG CHANNEL N41 Q4 1-MIN AVG
- ANM0121M PWR RNG CHANNEL N42 Q2 1-MIN AVG
- ANM0122M PWR RNG CHANNEL N43 Q1 1-MIN AVG
- ANM0123M PWR RNG CHANNEL N44 Q3 1-MIN AVG

Standard: Reviews P&L # 4 determines the quality codes are **NOT** acceptable for

- **URE1541 CURRENT CHAN 2 AXIAL FLUX DIFF**
- **URE1542 CURRENT CHAN 3 AXIAL FLUX DIFF**
- **ANM0121M PWR RNG CHANNEL N42 Q2 1-MIN AVG**
- **ANM0122M PWR RNG CHANNEL N43 Q1 1-MIN AVG**

Notifies the CRS the AFD Monitor does met the criteria for Operable status

Evaluator Cue:	**If necessary prompt the candidate to completed OST-1021, Attachment 5 as required.**

Comment:

OST-1021, Attachment 5, Page 2 of 3

Performance Step: 8 LOG current reading for the following instruments:

- NI-41C, PR 41 % Δ FLUX
- NI-42C, PR 42 % Δ FLUX
- NI-43C, PR 43 % Δ FLUX
- NI-44C, PR 44 % Δ FLUX

Standard: Locates JPM Cue sheet with attached MCB Indication images and logs current reading

- **NI-41C, PR 41 % Δ FLUX = 11% +/- 2%**
- **NI-42C, PR 42 % Δ FLUX = 13% +/- 2%**
- **NI-43C, PR 43 % Δ FLUX = 14% +/- 2%**
- **NI-44C, PR 44 % Δ FLUX = 10% +/- 2%**

Comment:

OST-1021, Attachment 5, Page 2 of 3

Performance Step: 9 DETERMINE and LOG Average (AVG) Reactor Power:

- NI-41B, PR 41 % POWER
- NI-42B, PR 42 % POWER
- NI-43B, PR 43 % POWER
- NI-44B, PR 44 % POWER

Standard: Locates JPM Cue sheet with attached MCB Indication images and logs current reading

- **NI-41B, PR 41 % POWER = 90% +/- 2%**
- **NI-42B, PR 42 % POWER = 90% +/- 2%**
- **NI-43B, PR 43 % POWER = 90% +/- 2%**
- **NI-44B, PR 44 % POWER = 90% +/- 2%**

Comment: **Performs calculation to determine AVG Reactor Power and logs value on OST-1021 Attachment 5**

OST-1021, Attachment 5, Page 2 of 3

✓ **Performance Step: 10** DETERMINE and LOG AFD Lower limit:

Standard: **Critical action is to determine required limit.**

Locates Rod Manual and reviews Section 2.1, AFD Limits and determines the current Lower limits is:

-14.5% at 90% Reactor Power (+/- 2%)

Comment: **Must interpolate limit based on current power level**

OST-1021, Attachment 5, Page 2 of 3

✓ **Performance Step: 11** DETERMINE and LOG AFD Upper limit:

Standard: **Critical action is to determine required limit.**

Locates Rod Manual and reviews Section 2.1, AFD Limits and determines the current Upper limits is:

11.0% at 90% Reactor Power (+/- 2%)

Comment: **Must interpolate limit based on current power level**

OST-1021, Attachment 5, Page 2 of 3

✓ **Performance Step: 12** PERFORM evaluation of AFD limits

Standard: Reviews current MCB readings and determines AFD Limits and determines two of four MCB indications are NOT within the curve for Acceptable Operation:

- **NI-42C, PR 42 % Δ FLUX = 13% +/- 2%**
- **NI-43C, PR 43 % Δ FLUX = 14% +/- 2%**

Notifies the CRS two of four MCB indications are NOT within the AFD curve for Acceptable Operation

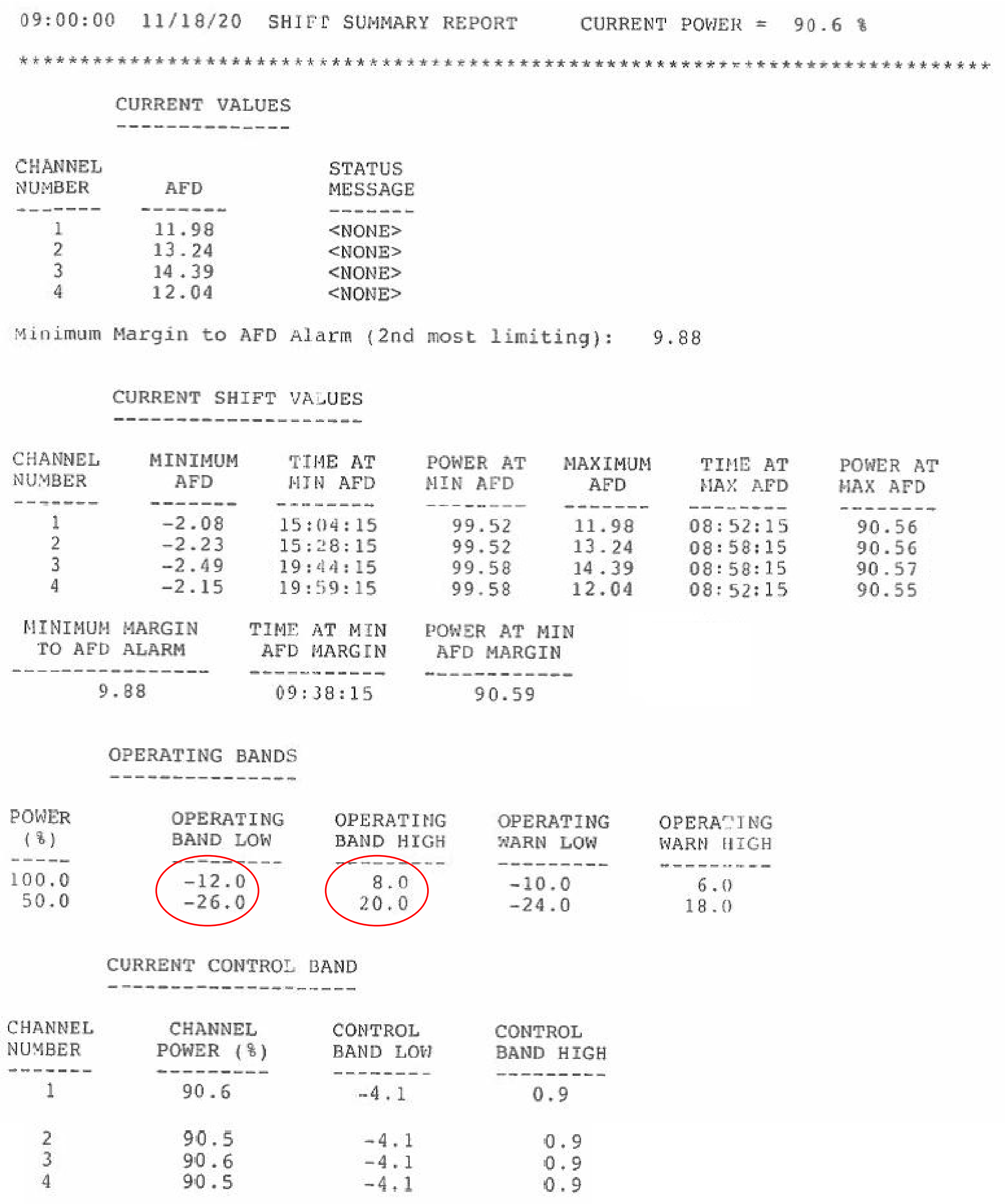
Comment: **Must interpolate limit based on current power level**

Evaluator Cue:	**After the candidate has determined the current values of Axial Flux Difference and its limits have been manually determined.** **END OF JPM**

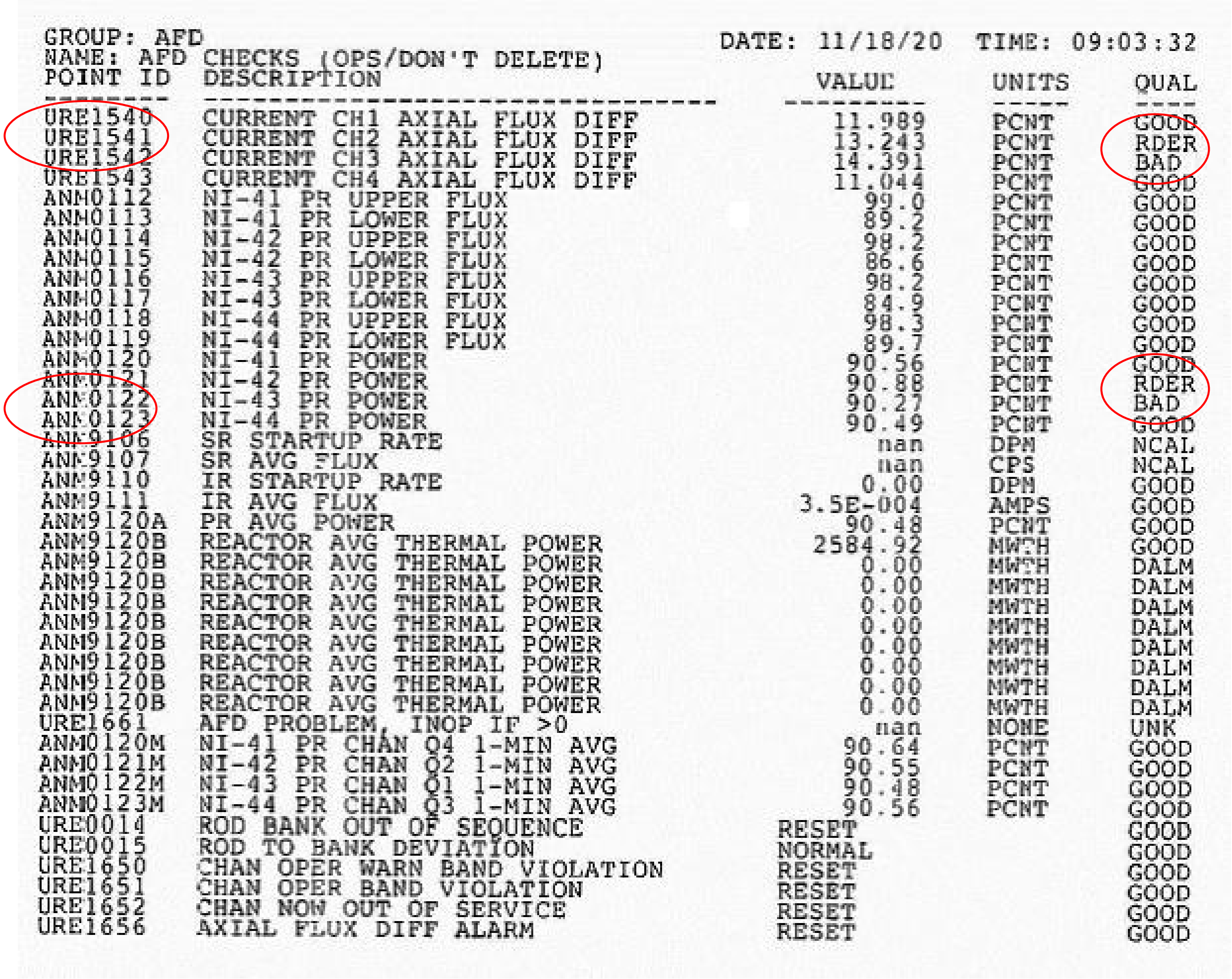
Terminating Cue:	Current value of Axial Flux Difference has been manually determined.

Stop Time: _________

KEY

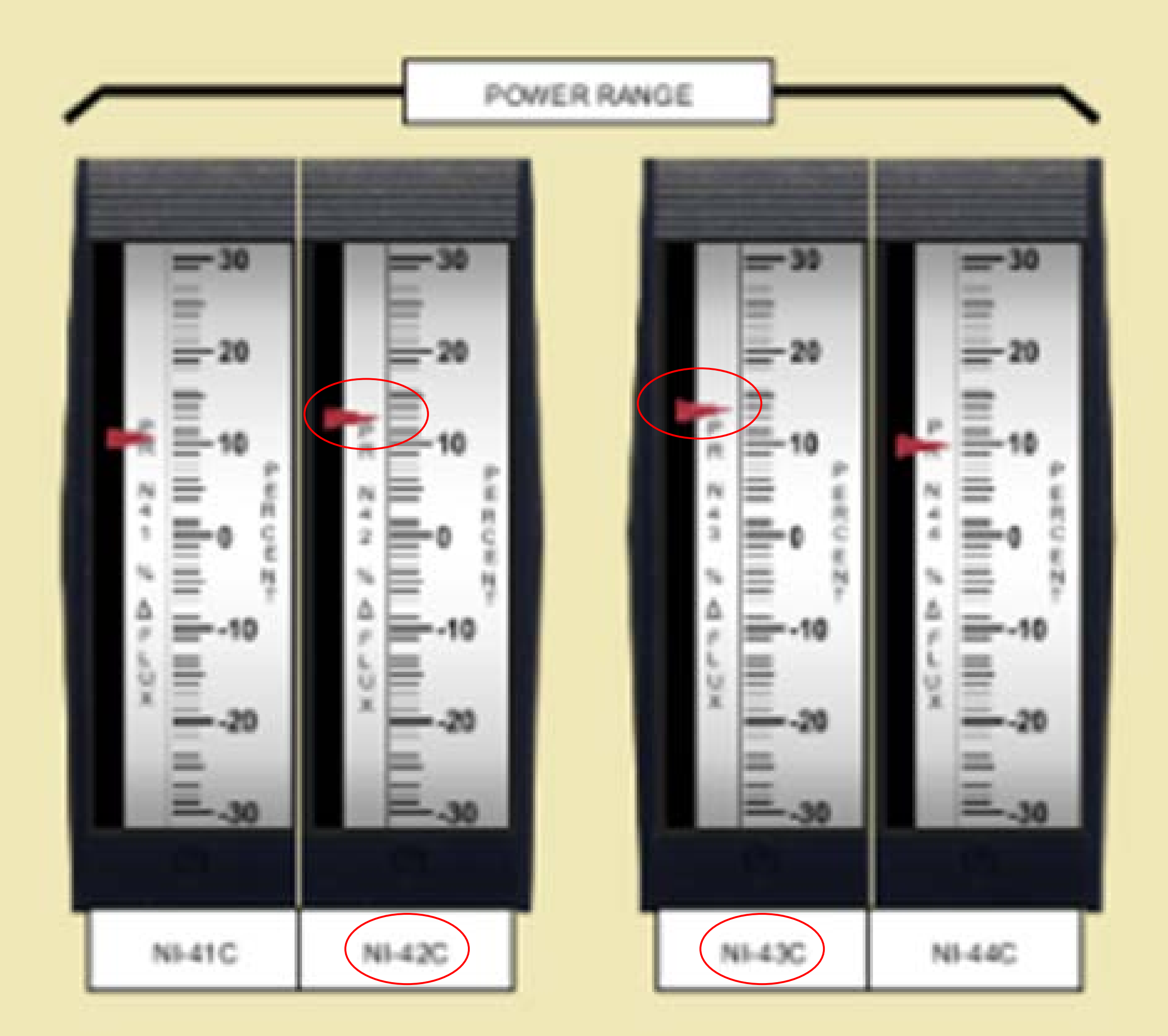


```
09:00:00  11/18/20  SHIFT SUMMARY REPORT     CURRENT POWER =  90.6 %

****************************************************************************

          CURRENT VALUES
          --------------

CHANNEL                     STATUS
NUMBER      AFD             MESSAGE
-------   -------           -------
   1       11.98            <NONE>
   2       13.24            <NONE>
   3       14.39            <NONE>
   4       12.04            <NONE>

Minimum Margin to AFD Alarm (2nd most limiting):   9.88

          CURRENT SHIFT VALUES
          --------------------

CHANNEL   MINIMUM    TIME AT    POWER AT   MAXIMUM    TIME AT    POWER AT
NUMBER      AFD      MIN AFD    MIN AFD      AFD      MAX AFD    MAX AFD
-------   -------   --------   --------   -------   --------   --------
   1       -2.08    15:04:15     99.52     11.98    08:52:15     90.56
   2       -2.23    15:28:15     99.52     13.24    08:58:15     90.56
   3       -2.49    19:44:15     99.58     14.39    08:58:15     90.57
   4       -2.15    19:59:15     99.58     12.04    08:52:15     90.55

 MINIMUM MARGIN    TIME AT MIN   POWER AT MIN
  TO AFD ALARM     AFD MARGIN     AFD MARGIN
----------------   -----------   ------------
      9.88          09:38:15        90.59

          OPERATING BANDS
          ---------------

POWER        OPERATING     OPERATING     OPERATING     OPERATING
 (%)         BAND LOW      BAND HIGH     WARN LOW      WARN HIGH
-----        ---------     ---------     ---------     ---------
100.0          -12.0          8.0          -10.0          6.0
 50.0          -26.0         20.0          -24.0         18.0

          CURRENT CONTROL BAND
          --------------------

CHANNEL     CHANNEL       CONTROL       CONTROL
NUMBER      POWER (%)     BAND LOW      BAND HIGH
-------     ---------     --------      ---------
   1          90.6          -4.1           0.9

   2          90.5          -4.1           0.9
   3          90.6          -4.1           0.9
   4          90.5          -4.1           0.9
```

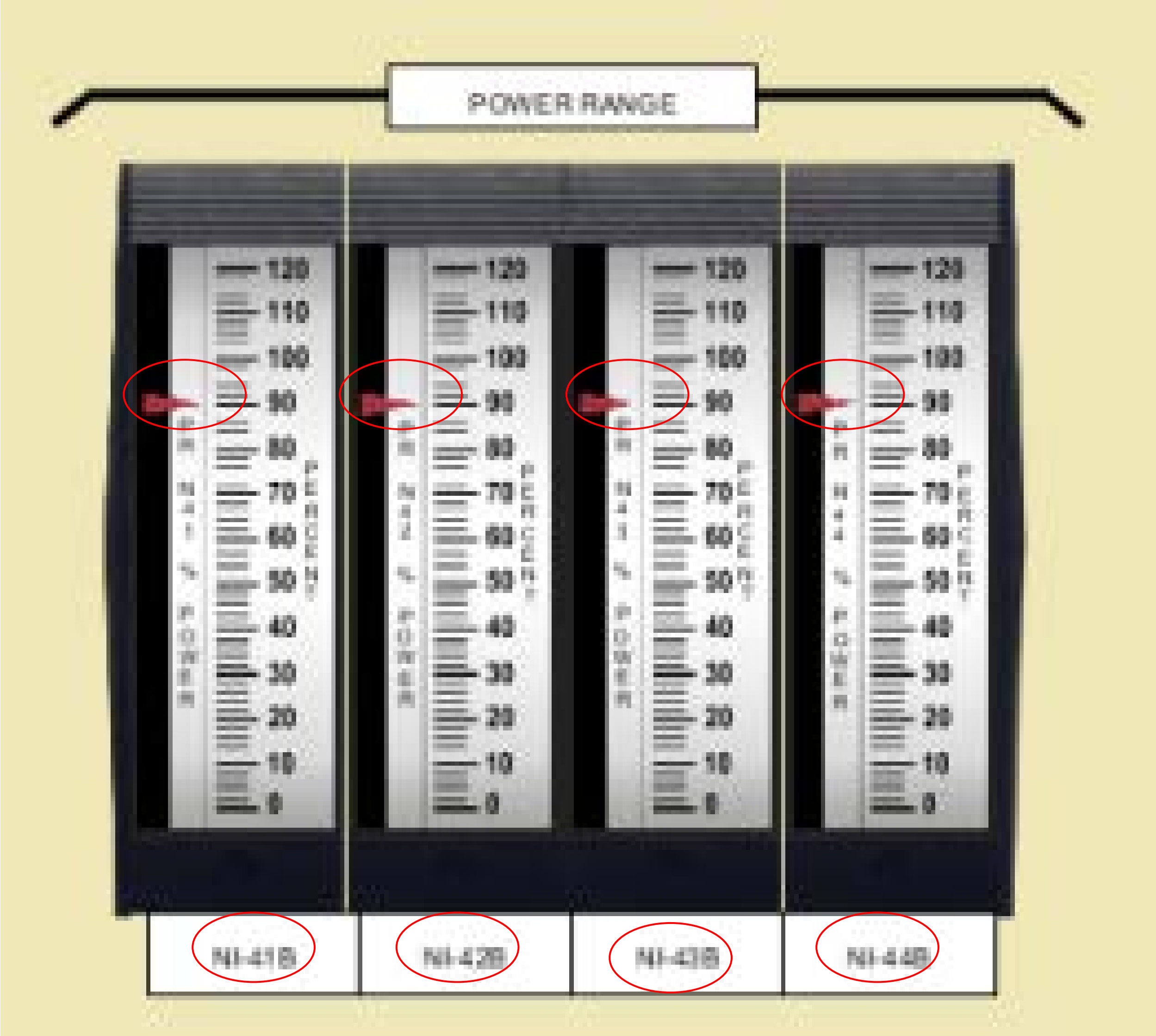


```
GROUP: AFD                                          DATE: 11/18/20   TIME: 09:03:32
NAME: AFD CHECKS (OPS/DON'T DELETE)
POINT ID  DESCRIPTION                                VALUE       UNITS     QUAL
--------  -------------------------------          ----------    -----     ----
URE1540   CURRENT CH1 AXIAL FLUX DIFF                  11.989    PCNT      GOOD
URE1541   CURRENT CH2 AXIAL FLUX DIFF                  13.243    PCNT      RDER
URE1542   CURRENT CH3 AXIAL FLUX DIFF                  14.391    PCNT      BAD
URE1543   CURRENT CH4 AXIAL FLUX DIFF                  11.044    PCNT      GOOD
ANM0112   NI-41 PR UPPER FLUX                            99.0    PCNT      GOOD
ANM0113   NI-41 PR LOWER FLUX                            89.2    PCNT      GOOD
ANM0114   NI-42 PR UPPER FLUX                            98.2    PCNT      GOOD
ANM0115   NI-42 PR LOWER FLUX                            86.6    PCNT      GOOD
ANM0116   NI-43 PR UPPER FLUX                            98.2    PCNT      GOOD
ANM0117   NI-43 PR LOWER FLUX                            84.9    PCNT      GOOD
ANM0118   NI-44 PR UPPER FLUX                            98.3    PCNT      GOOD
ANM0119   NI-44 PR LOWER FLUX                            89.7    PCNT      GOOD
ANM0120   NI-41 PR POWER                                90.56    PCNT      GOOD
ANM0121   NI-42 PR POWER                                90.88    PCNT      RDER
ANM0122   NI-43 PR POWER                                90.27    PCNT      BAD
ANM0123   NI-44 PR POWER                                90.49    PCNT      GOOD
ANM9106   SR STARTUP RATE                                 nan    DPM       NCAL
ANM9107   SR AVG FLUX                                     nan    CPS       NCAL
ANM9110   IR STARTUP RATE                                0.00    DPM       GOOD
ANM9111   IR AVG FLUX                                3.5E-004    AMPS      GOOD
ANM9120A  PR AVG POWER                                  90.48    PCNT      GOOD
ANM9120B  REACTOR AVG THERMAL POWER                   2584.92    MWTH      GOOD
ANM9120B  REACTOR AVG THERMAL POWER                      0.00    MWTH      DALM
ANM9120B  REACTOR AVG THERMAL POWER                      0.00    MWTH      DALM
ANM9120B  REACTOR AVG THERMAL POWER                      0.00    MWTH      DALM
ANM9120B  REACTOR AVG THERMAL POWER                      0.00    MWTH      DALM
ANM9120B  REACTOR AVG THERMAL POWER                      0.00    MWTH      DALM
ANM9120B  REACTOR AVG THERMAL POWER                      0.00    MWTH      DALM
ANM9120B  REACTOR AVG THERMAL POWER                      0.00    MWTH      DALM
ANM9120B  REACTOR AVG THERMAL POWER                      0.00    MWTH      DALM
URE1661   AFD PROBLEM, INOP IF >0                         nan    NONE      UNK
ANM0120M  NI-41 PR CHAN Q4 1-MIN AVG                    90.64    PCNT      GOOD
ANM0121M  NI-42 PR CHAN Q2 1-MIN AVG                    90.55    PCNT      GOOD
ANM0122M  NI-43 PR CHAN Q1 1-MIN AVG                    90.48    PCNT      GOOD
ANM0123M  NI-44 PR CHAN Q3 1-MIN AVG                    90.56    PCNT      GOOD
URE0014   ROD BANK OUT OF SEQUENCE                 RESET                   GOOD
URE0015   ROD TO BANK DEVIATION                    NORMAL                  GOOD
URE1650   CHAN OPER WARN BAND VIOLATION            RESET                   GOOD
URE1651   CHAN OPER BAND VIOLATION                 RESET                   GOOD
URE1652   CHAN NOW OUT OF SERVICE                  RESET                   GOOD
URE1656   AXIAL FLUX DIFF ALARM                    RESET                   GOOD
```

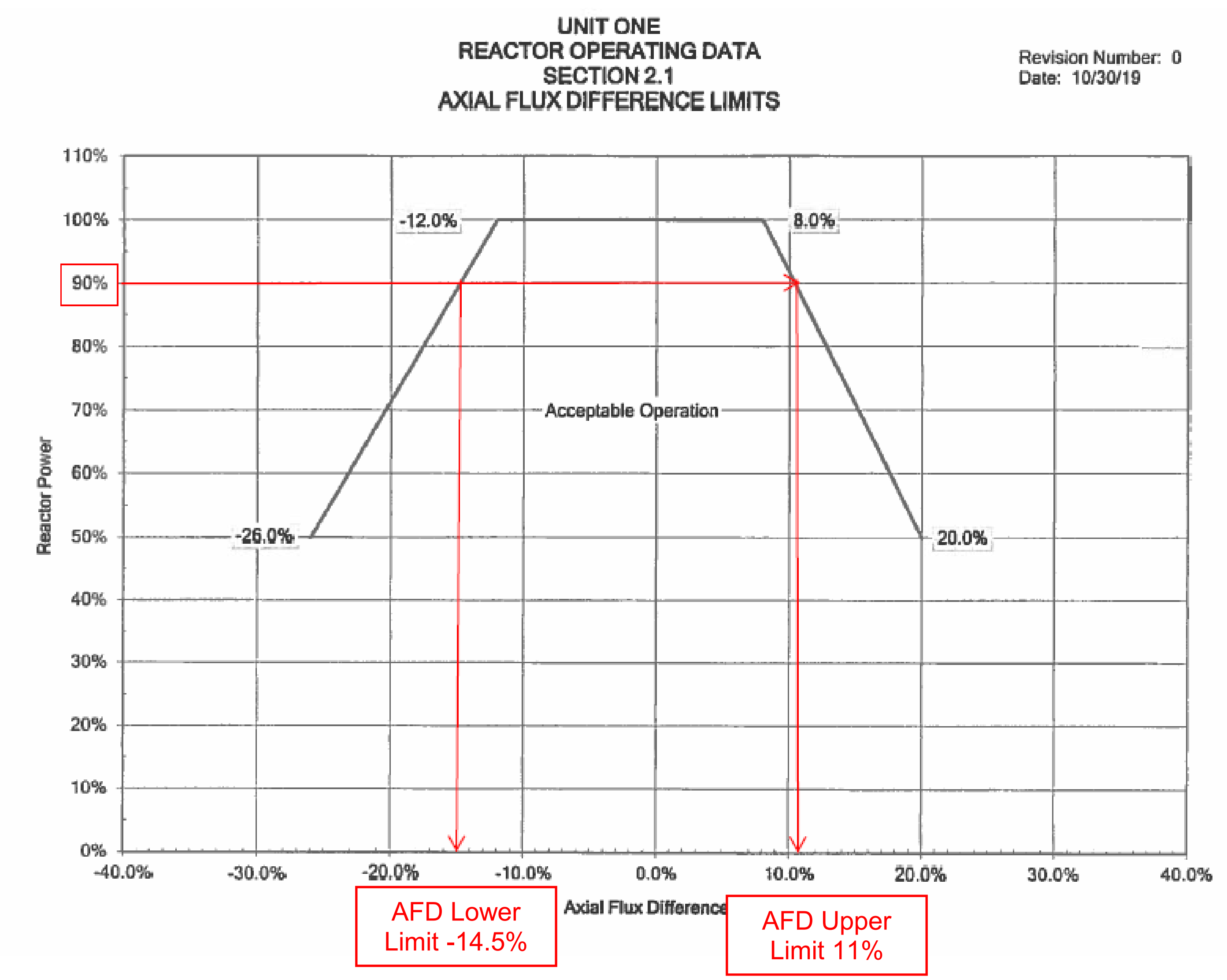
KEY


POWER RANGE
30
20
10
0
-10
-20
-30
PR N41 % Δ FLUX
PR N42 % Δ FLUX
PR N43 % Δ FLUX
PR N44 % Δ FLUX
PERCENT
NI-41C
NI-42C
NI-43C
NI-44C

KEY


POWER RANGE
120
110
100
90
80
70
60
50
40
30
20
10
0
PERCENT
PR N41 % POWER
PR N42 % POWER
PR N43 % POWER
PR N44 % POWER
NI-41B
NI-42B
NI-43B
NI-44B

KEY


UNIT ONE
REACTOR OPERATING DATA
SECTION 2.1
AXIAL FLUX DIFFERENCE LIMITS
Revision Number: 0
Date: 10/30/19
110%
100%
90%
80%
70%
60%
50%
40%
30%
20%
10%
0%
Reactor Power
-12.0%
8.0%
-26.0%
20.0%
Acceptable Operation
-40.0%
-30.0%
-20.0%
-10.0%
0.0%
10.0%
20.0%
30.0%
40.0%
Axial Flux Difference
AFD Lower Limit -14.5%
AFD Upper Limit 11%

KEY

1. The current AFD Limits are **Upper AFD limit 11.0% at 90% Reactor Power (+/- 2%)**
Lower AFD limit 14.5% at 90% Reactor Power (+/- 2%)

Circle the correct response that applies:

2. AFD Monitor Alarm is Operable / [✓ Inoperable]
3. AFD is / [✓ is NOT] within the range of Acceptable Operation

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Admin Exam RO A1-1

Determine Axial Flux Difference (AFD) with AFD Monitor INOP

OP-163, ERFIS

OST-1021, Daily Surveillance Requirements

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	• The plant was at 90% power, with a load reduction in progress • The load reduction has been stopped to evaluate AFD following oscillations

Initiating Cue:	With the information provided complete OST-1021, Daily Surveillance Requirements to determine Axial Flux Difference. After performing the calculation evaluate the results and circle the response below. When complete return your results to the evaluator.

Name: ____________________________________

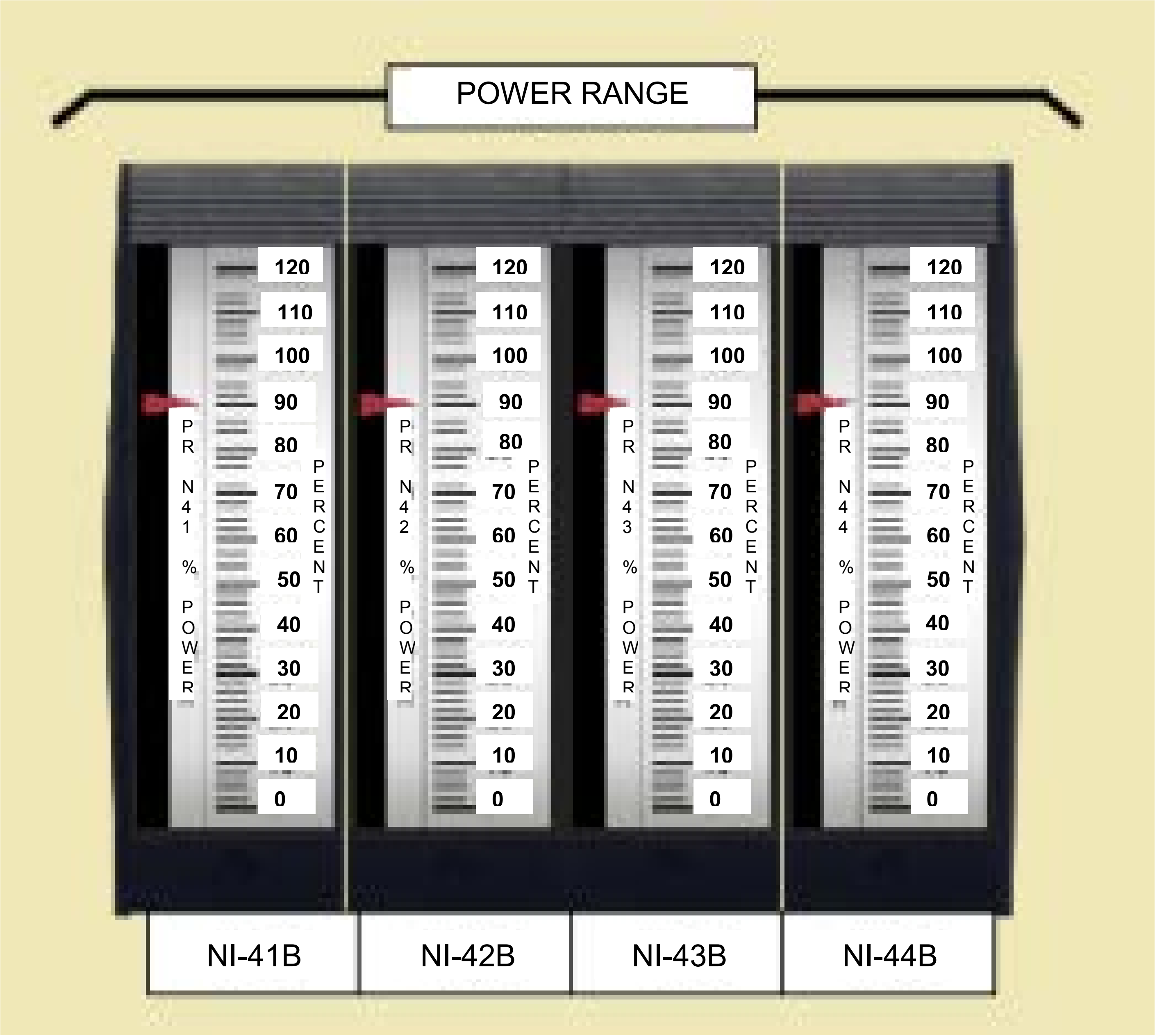
Date: ____________________

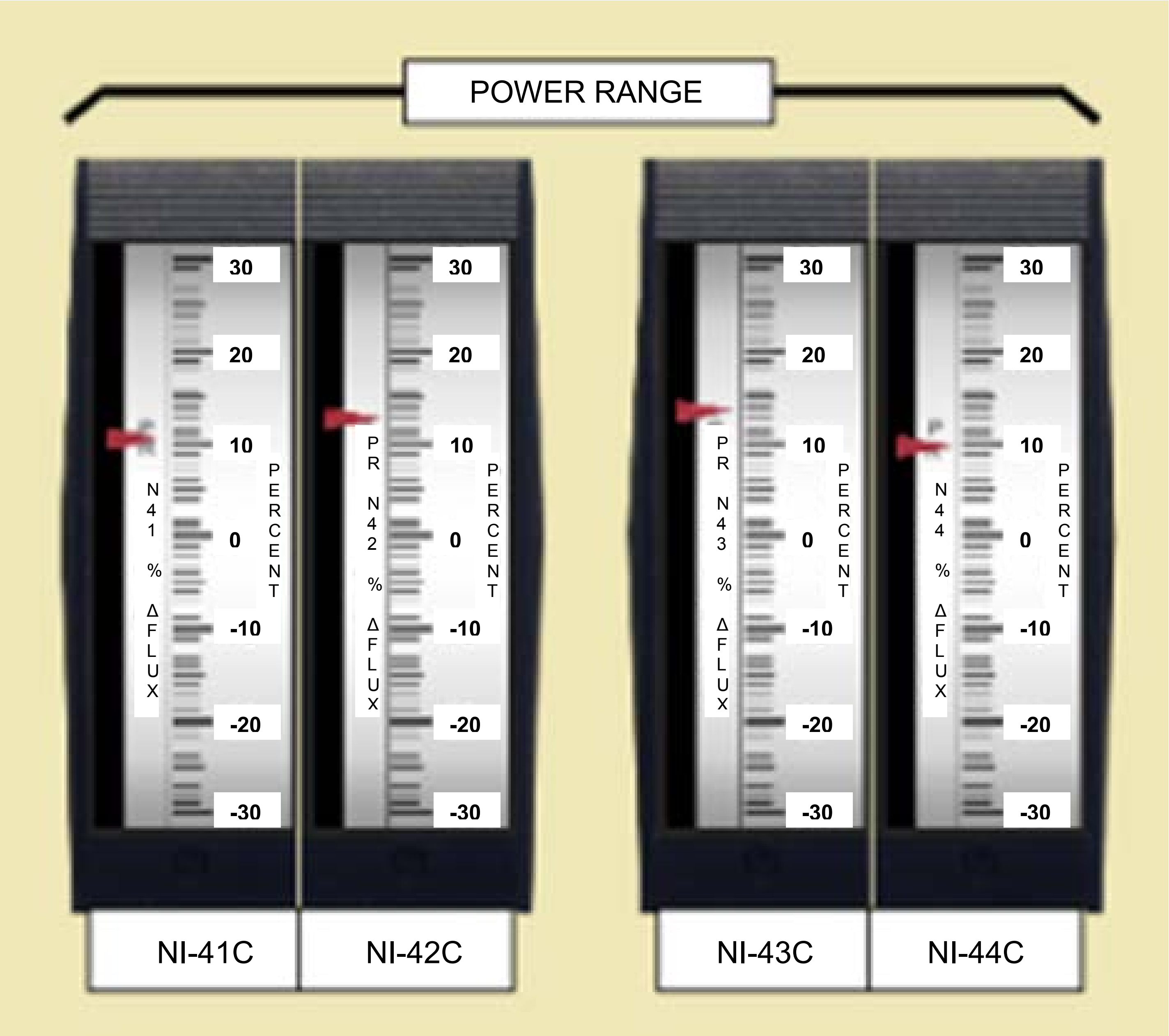
1. The current AFD Limits are ____________________

Circle the correct response that applies:

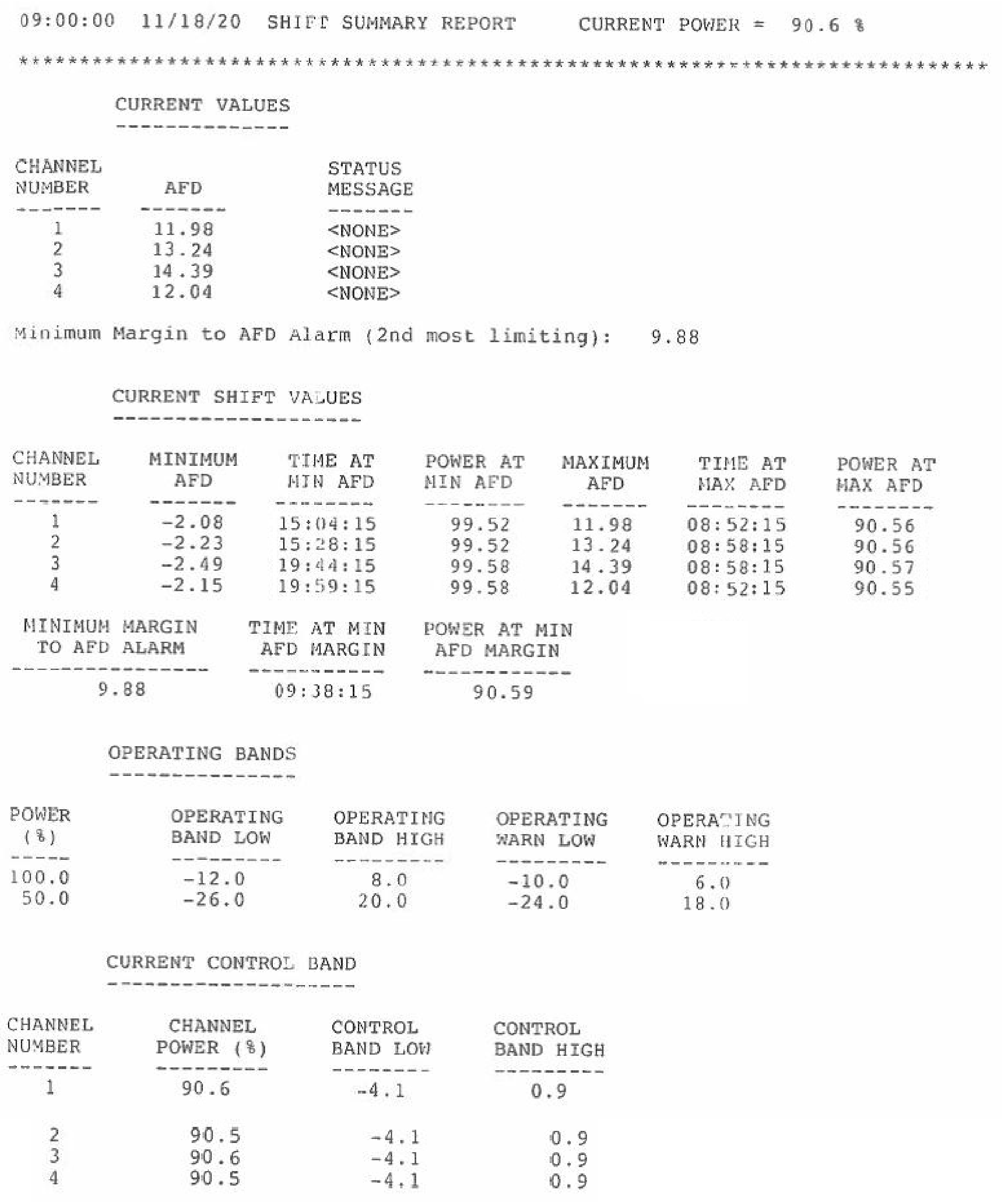
2. AFD Monitor Alarm is Operable / Inoperable

3. AFD is / is NOT within the range of Acceptable Operation


POWER RANGE
PR N41 % POWER
PR N42 % POWER
PR N43 % POWER
PR N44 % POWER
PERCENT
120
110
100
90
80
70
60
50
40
30
20
10
0
NI-41B
NI-42B
NI-43B
NI-44B

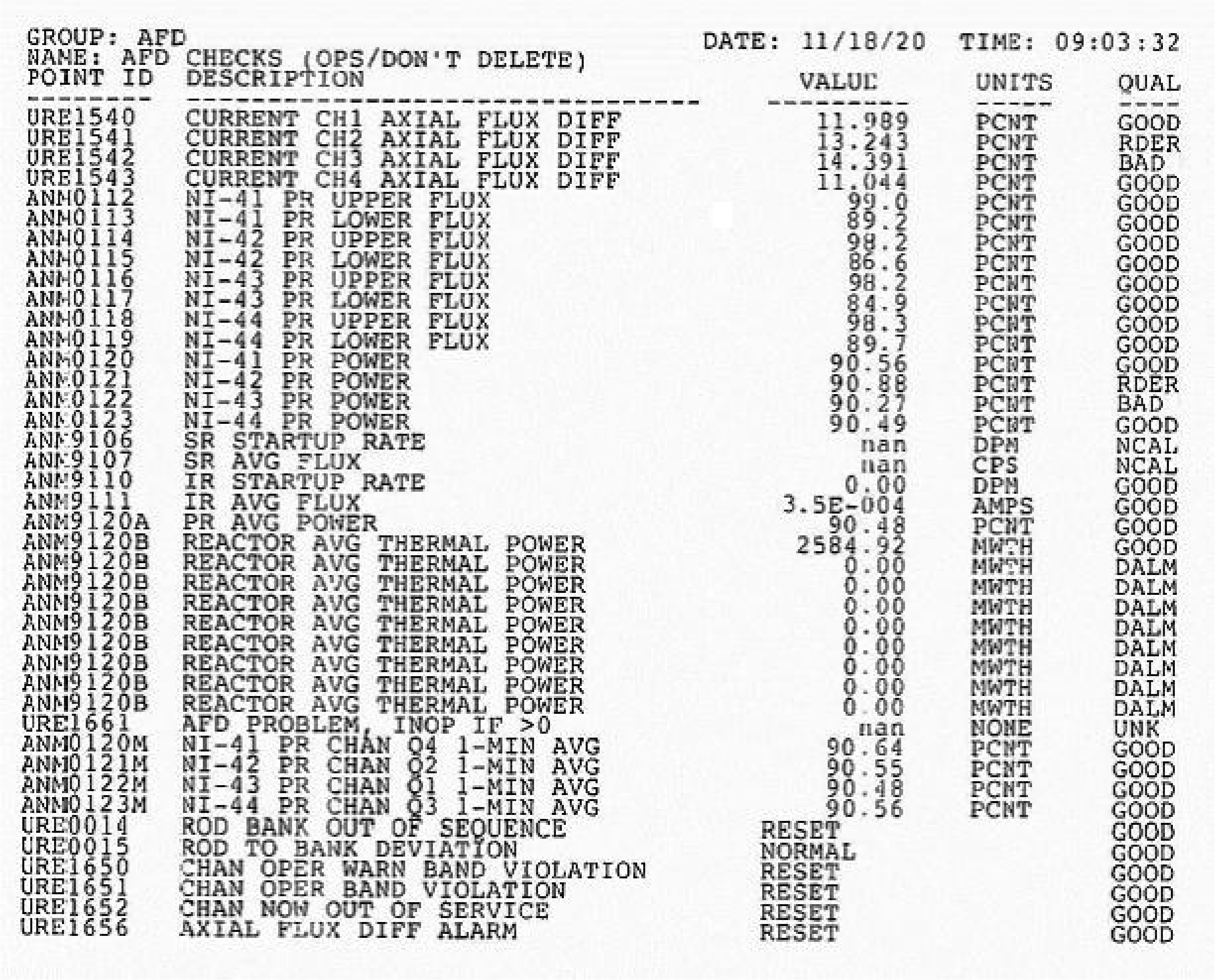

POWER RANGE
30
20
10
0
-10
-20
-30
N41 % ΔFLUX
PR N42 % ΔFLUX
PR N43 % ΔFLUX
N44 % ΔFLUX
PERCENT
NI-41C
NI-42C
NI-43C
NI-44C

JPM CUE SHEET



```
09:00:00  11/18/20  SHIFT SUMMARY REPORT     CURRENT POWER =  90.6 %

*******************************************************************************

        CURRENT VALUES
        --------------

CHANNEL                        STATUS
NUMBER       AFD               MESSAGE
-------    -------             -------
   1        11.98              <NONE>
   2        13.24              <NONE>
   3        14.39              <NONE>
   4        12.04              <NONE>

Minimum Margin to AFD Alarm (2nd most limiting):    9.88

        CURRENT SHIFT VALUES
        --------------------

CHANNEL    MINIMUM    TIME AT    POWER AT   MAXIMUM    TIME AT    POWER AT
NUMBER       AFD      MIN AFD    MIN AFD      AFD      MAX AFD    MAX AFD
-------    -------   --------    --------   -------   --------   --------
   1        -2.08    15:04:15     99.52      11.98    08:52:15     90.56
   2        -2.23    15:28:15     99.52      13.24    08:58:15     90.56
   3        -2.49    19:44:15     99.58      14.39    08:58:15     90.57
   4        -2.15    19:59:15     99.58      12.04    08:52:15     90.55

 MINIMUM MARGIN     TIME AT MIN    POWER AT MIN
  TO AFD ALARM      AFD MARGIN      AFD MARGIN
----------------    -----------    ------------
      9.88           09:38:15          90.59

        OPERATING BANDS
        ---------------

POWER        OPERATING    OPERATING    OPERATING    OPERATING
 (%)         BAND LOW     BAND HIGH    WARN LOW     WARN HIGH
-----        ---------    ---------    ---------    ---------
100.0         -12.0          8.0        -10.0          6.0
 50.0         -26.0         20.0        -24.0         18.0

        CURRENT CONTROL BAND
        --------------------

CHANNEL      CHANNEL      CONTROL      CONTROL
NUMBER      POWER (%)     BAND LOW     BAND HIGH
-------     ---------     --------     ---------
   1          90.6          -4.1          0.9

   2          90.5          -4.1          0.9
   3          90.6          -4.1          0.9
   4          90.5          -4.1          0.9
```

JPM CUE SHEET



```
GROUP: AFD                                              DATE: 11/18/20  TIME: 09:03:32
NAME: AFD CHECKS (OPS/DON'T DELETE)
POINT ID  DESCRIPTION                                VALUE      UNITS    QUAL
--------  --------------------------------     ----------    -----    ----
URE1540   CURRENT CH1 AXIAL FLUX DIFF              11.989    PCNT     GOOD
URE1541   CURRENT CH2 AXIAL FLUX DIFF              13.243    PCNT     RDER
URE1542   CURRENT CH3 AXIAL FLUX DIFF              14.391    PCNT     BAD
URE1543   CURRENT CH4 AXIAL FLUX DIFF              11.044    PCNT     GOOD
ANM0112   NI-41 PR UPPER FLUX                        99.0    PCNT     GOOD
ANM0113   NI-41 PR LOWER FLUX                        89.2    PCNT     GOOD
ANM0114   NI-42 PR UPPER FLUX                        98.2    PCNT     GOOD
ANM0115   NI-42 PR LOWER FLUX                        86.6    PCNT     GOOD
ANM0116   NI-43 PR UPPER FLUX                        98.2    PCNT     GOOD
ANM0117   NI-43 PR LOWER FLUX                        84.9    PCNT     GOOD
ANM0118   NI-44 PR UPPER FLUX                        98.3    PCNT     GOOD
ANM0119   NI-44 PR LOWER FLUX                        89.7    PCNT     GOOD
ANM0120   NI-41 PR POWER                            90.56    PCNT     GOOD
ANM0121   NI-42 PR POWER                            90.88    PCNT     RDER
ANM0122   NI-43 PR POWER                            90.27    PCNT     BAD
ANM0123   NI-44 PR POWER                            90.49    PCNT     GOOD
ANM9106   SR STARTUP RATE                             nan    DPM      NCAL
ANM9107   SR AVG FLUX                                 nan    CPS      NCAL
ANM9110   IR STARTUP RATE                            0.00    DPM      GOOD
ANM9111   IR AVG FLUX                           3.5E-004    AMPS     GOOD
ANM9120A  PR AVG POWER                              90.48    PCNT     GOOD
ANM9120B  REACTOR AVG THERMAL POWER               2584.92    MWTH     GOOD
ANM9120B  REACTOR AVG THERMAL POWER                  0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                  0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                  0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                  0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                  0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                  0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                  0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                  0.00    MWTH     DALM
URE1661   AFD PROBLEM, INOP IF >0                     nan    NONE     UNK
ANM0120M  NI-41 PR CHAN Q4 1-MIN AVG                90.64    PCNT     GOOD
ANM0121M  NI-42 PR CHAN Q2 1-MIN AVG                90.55    PCNT     GOOD
ANM0122M  NI-43 PR CHAN Q1 1-MIN AVG                90.48    PCNT     GOOD
ANM0123M  NI-44 PR CHAN Q3 1-MIN AVG                90.56    PCNT     GOOD
URE0014   ROD BANK OUT OF SEQUENCE             RESET                  GOOD
URE0015   ROD TO BANK DEVIATION                NORMAL                 GOOD
URE1650   CHAN OPER WARN BAND VIOLATION        RESET                  GOOD
URE1651   CHAN OPER BAND VIOLATION             RESET                  GOOD
URE1652   CHAN NOW OUT OF SERVICE              RESET                  GOOD
URE1656   AXIAL FLUX DIFF ALARM                RESET                  GOOD
```

ATTACHMENT 5
Page 1 of 3

<< Axial Flux Difference Log >>

AFD MONITOR OPERABLE

Tech Spec	4.2.1.1.a				
Parameter	Axial Flux Difference				
Instrument	NI-41C	NI-42C	NI-43C	NI-44C	AFD Monitor Checks Performed PER OP-163 (Initials)
Acceptance Criteria	Within AFD COLR Limits				
MODE	1 Above 50% Rated Thermal Power				
0800 - 1100					
2000 - 2300					

ATTACHMENT 5
Page 2 of 3

<< Axial Flux Difference Log >>

AFD MONITOR INOPERABLE

Tech Spec	4.2.1.1.b, 4.2.1.2								
Parameter	Axial Flux Difference								
Instrument	NI-41C	NI-42C	NI-43C	NI-44C	AVG Reactor Power	AFD Limits			
						Lower	Upper	Perform	Verify
Acceptance Criteria	Within AFD COLR Limits								
MODE	1 Above 50% Rated Thermal Power								
0000 - 0005									
0030 - 0035									
0100 - 0105									
0130 - 0135									
0200 - 0205									
0230 - 0235									
0300 - 0305									
0330 - 0335									
0400 - 0405									
0430 - 0435									
0500 - 0505									
0530 - 0535									
0600 - 0605									
0630 - 0635									
0700 - 0705									
0730 - 0735									
0800 - 0805									
0830 - 0835									
0900 - 0905									
0930 - 0935									
1000 - 1005									
1030 - 1035									
1100 - 1105									
1130 - 1135									
1200 - 1205									

Nightshift CRS Review ____________________

ATTACHMENT 5
Page 3 of 3

<< Axial Flux Difference Log >>

AFD MONITOR INOPERABLE

Tech Spec	4.2.1.1.b, 4.2.1.2								
Parameter	Axial Flux Difference								
Instrument	NI-41C	NI-42C	NI-43C	NI-44C	AVG Reactor Power	AFD Limits			
						Lower	Upper	Perform	Verify
Acceptance Criteria	Within AFD COLR Limits								
MODE	1 Above 50% Rated Thermal Power								
1230 - 1235									
1300 - 1305									
1330 - 1335									
1400 - 1405									
1430 - 1435									
1500 - 1505									
1530 - 1535									
1600 - 1605									
1630 - 1635									
1700 - 1705									
1730 -1735									
1800 - 1805									
1830 - 1835									
1900 - 1905									
1930 - 1935									
2000 - 2005									
2030 - 2035									
2100 - 2105									
2130 - 2135									
2200 - 2205									
2230 - 2235									
2300 - 2305									
2330 - 2335									

Dayshift CRS Review ______________________

Facility:	Harris Nuclear Plant	Task No.:	005016H101
Task Title:	AOP-017 Attachment 4 manual makeup calculation	JPM No.:	2020 NRC Exam Admin JPM RO A1-2
K/A Reference:	G2.1.25 RO 3.9 SRO 4.2		**ALTERNATE PATH - NO**

Examinee: ______________________ NRC Examiner: ________________

Facility Evaluator: ______________________ Date: ________________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom X Simulator ______ Plant ________

READ TO THE EXAMINEE	
I will explain the initial conditions, which steps to simulate, discuss, or perform and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	• The plant is in Mode 3 • Instrument air header pressure is 45 psig and stable • Automatic Blender automatic makeup is not available • VCT level is currently 19% and stable

Initiating Cue:	The CRS has directed you to perform a Manual Makeup and to determine the following for these conditions: • Required Boric acid flow rate • The maximum possible makeup flow rate to achieve required boron concentration in the VCT. • Dilution flow rate **Record your results on the applicable procedure** **Show all work.**

Task Standard: Determines Required boric acid flow to be 27.5 gpm (27.0-28.0) and required dilution flow is 72.5 gpm (72.0 – 73.0), using AOP-017 attachment 4 and OP-107.1.

Required Materials: AOP-017, Rev. 40
OP-107.01, Rev. 30
Calculator

General References: AOP-017, Rev. 40
OP-107.01, Rev. 30

Handouts: AOP-017, Rev. 40, pages 47 – 50, Manual Makeup
OP-107.01, Rev. 30, pages 120 – 128, Makeup Concentration Limits
OR
2020 NRC Exam Frozen Procedures Folder

JPM Cue Sheets

Time Critical Task: No

Validation Time: 15 minutes

PERFORMANCE STEP	**CRITICAL STEP JUSTIFICATION**
Step 7	Must calculate correct boric acid and dilution flow rates to ensure correct manual makeup is performed.
Step 8	Must calculate correct boric acid and dilution flow rates to ensure correct manual makeup is performed.

START TIME: ____________

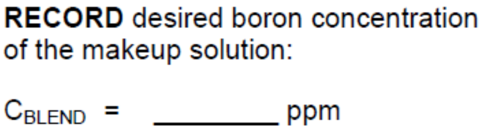
Performance Step: 1 Obtain a copy of the appropriate procedures **(AOP-017)**

Standard: Operator obtains a copy of AOP-017 to determine appropriate attachment is Attachment 4 to complete a manual makeup.

Comment:

AOP-017 ATT.4 Step 1

Performance Step: 2



RECORD desired boron concentration of the makeup solution:

C_{BLEND} = ________ ppm

Standard: References Reactivity sheet and uses the RCS Boron concentration of 1928.

Comment:

AOP-017 ATT.4 NOTE before Step 2

Performance Step: 3 NOTE

If RCS boron concentration is above 1750 ppm, blended makeup at 120 gpm may not be possible, due to the inability of the system to reliably deliver more than 30 gpm boric acid flow. In those cases, either select a conservatively low total makeup flow, or consult Attachment 7 of OP-107.01 to determine the maximum possible makeup flow.

Standard: Reads and placekeeps Note

Comment:

OP-107.01 Attachment 7

Performance Step: 4 Determine maximum total flow with a RCS boron concentration of 1928 ppm

Standard: References OP-107.01 Attachment 7 page 4 and determines the maximum total flow available to meet the 1928 ppm requirement is 100 gpm.

Comment:

AOP-017 ATT. 4 step 2

Performance Step: 5 RECORD desired total makeup flow rate:

MBLEND = ________ gpm

Standard: Records 100 gpm as the desired total makeup flow.

Comment:

AOP-017 ATT. 4 step 3

Performance Step: 6 RECORD most recent Boric Acid Tank boron concentration from Unit Status Board:

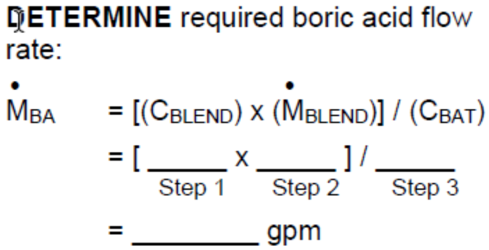
C_{BAT} = ________ ppm

Standard: References Reactivity data sheet and records 7000 ppm

Comment:

AOP-017 ATT. 4 step 4

✓ **Performance Step: 7**



DETERMINE required boric acid flow rate:

$\dot{M}_{BA} = [(C_{BLEND}) \times (\dot{M}_{BLEND})] / (C_{BAT})$

$= [\,\underset{\text{Step 1}}{\_\_\_\_\_} \times \underset{\text{Step 2}}{\_\_\_\_\_}\,] / \underset{\text{Step 3}}{\_\_\_\_\_}$

$= \_\_\_\_\_\_\_\_$ gpm

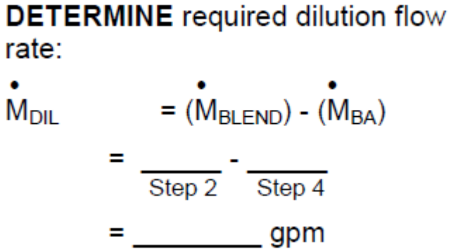
Standard: $[(C_{BLEND}) \times (M_{BLEND})] / (C_{BAT})$

(1928 ppm x 100gpm) / (7000ppm) = 27.5 gpm (27.0-28.0) gpm

Comment:

AOP-017 ATT. 4 step 6

✓ **Performance Step: 8**



DETERMINE required dilution flow rate:

$\dot{M}_{DIL} = (\dot{M}_{BLEND}) - (\dot{M}_{BA})$

$= \underset{\text{Step 2}}{\_\_\_\_\_} - \underset{\text{Step 4}}{\_\_\_\_\_}$

$= \_\_\_\_\_\_\_\_$ gpm

Standard: Calculates

100 gpm – 27.5 gpm = 72.5 gpm (72.0 – 73.0) gpm

Comment:

Evaluator Cue:	**When the BA flow rate and total flow rate has been determined. Evaluation on this JPM is complete.** **END OF JPM**

Stop Time: ________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Admin JPM RO A1-2
AOP-017 Attachment 4 manual makeup calculation
AOP-017
OP-107.01

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	• The plant is in Mode 3 • Instrument air leak resulted header pressure lowering to 45 psig • Automatic Blender automatic makeup is not available • VCT level is currently 19% and stable

Initiating Cue:	The CRS has directed you to perform Manual Makeup and to determine the following for these conditions: • Required Boric acid flow rate • The maximum possible makeup flow rate to achieve required boron concentration in the VCT. • Dilution flow rate **Record your results on the applicable procedure** **Show all work.**

Name: ____

Date: __________________

1. Identify the procedure required to be entered to address the current plant conditions.

   ____

2. Record your results on the procedure section or attachment required to complete the Manual Makeup for the current plant conditions.

   ____

JPM CUE SHEET

REACTIVITY DATA

Plant on-line: Date: **11/16/20** Time: **1535**

Core Burn up: **15** EFPD Date: **TODAY**

Date / Time

RCS Boron: **1928** PPM **NOW / NOW**

PZR Boron: **1929** PPM **NOW / NOW**

BAT Boron: **7000** PPM **NOW / NOW**

RWST Boron: **2450** PPM **NOW / NOW**

Xenon Free SDM Boron Requirements

557°F	1378 ppm	**450°F**	1566 ppm	**300°F**	1668 ppm	**70°F**	1765 ppm
550°F	1397 ppm	**400°F**	1611 ppm	**250°F**	1686 ppm		
500°F	1500 ppm	**350°F**	1644 ppm	**200°F**	1721 ppm		

Attachment 4

Sheet 1 of 4

Manual Makeup

INSTRUCTIONS	RESPONSE NOT OBTAINED

CAUTION

Due to accident analysis, this attachment can only be used in Modes 1 through 4.

☐ 1. **RECORD** desired boron concentration of the makeup solution:

C_{BLEND} = ________ ppm

NOTE

If RCS boron concentration is above 1750 ppm, blended makeup at 120 gpm may not be possible, due to the inability of the system to reliably deliver more than 30 gpm boric acid flow. In those cases, either select a conservatively low total makeup flow, or consult Attachment 7 of OP-107.01 to determine the maximum possible makeup flow.

☐ 2. **RECORD** desired total makeup flow rate:

$\dot{M}_{BLEND}$ = ________ gpm

☐ 3. **RECORD** most recent Boric Acid Tank boron concentration from Unit Status Board:

C_{BAT} = ________ ppm

☐ 4. **DETERMINE** required boric acid flow rate:

$\dot{M}_{BA} = [(C_{BLEND}) \times (\dot{M}_{BLEND})] / (C_{BAT})$

$= [\underset{\text{Step 1}}{\_\_\_\_\_} \times \underset{\text{Step 2}}{\_\_\_\_\_}] / \underset{\text{Step 3}}{\_\_\_\_\_}$

= ________ gpm

Attachment 4

Sheet 2 of 4

Manual Makeup

INSTRUCTIONS

RESPONSE NOT OBTAINED

☐ **5.** **RECORD** in Step 12a the result from the previous step.

☐ **6.** **DETERMINE** required dilution flow rate:

$$\dot{M}_{DIL} = (\dot{M}_{BLEND}) - (\dot{M}_{BA})$$

$$= \underset{\text{Step 2}}{\underline{\qquad\quad}} - \underset{\text{Step 4}}{\underline{\qquad\quad}}$$

$$= \underline{\qquad\qquad} \text{ gpm}$$

☐ **7.** **RECORD** in Step 12b the result from the previous calculation.

☐ **8.** **INDEPENDENTLY VERIFY** boric acid flow rate and dilution flow rate calculations made in steps 4 and 6.

9. **DIRECT** an operator to perform the following:

☐ **a.** **OBTAIN** a radio and a locked valve key.

☐ **b.** **ESTABLISH** communication between 236’ RAB Emergency Boration Valve Gallery and the Control Room.

c. **UNLOCK** the following:

☐ • 1CS-274, RMUW Manual Blend from RMWST

☐ • 1CS-287, Manual Alternate Emergency Boration

Attachment 4

Sheet 3 of 4

Manual Makeup

INSTRUCTIONS	RESPONSE NOT OBTAINED

10. VERIFY the following:

☐ • One Boric Acid Transfer Pump—RUNNING

☐ • One Reactor Makeup Water Pump—RUNNING

☐ • FCV-113A, Boric Acid Filter to Boric Acid Blender Flow Control Valve—OPEN

☐ **11.** Prior to continuing, **CHECK** that all previous steps are complete.

NOTE

- Boric acid flow can be monitored on ERFIS point FCS0113A.
- Dilution water flow can be monitored on ERFIS point FCS0110, or on FI-110 on the MCB.
- Actions in the next step should be performed as closely together as possible to achieve an even makeup.

12. As closely together as possible, **LOCALLY PERFORM** the following:

☐ **a. THROTTLE OPEN** 1CS-287, Manual Alternate Emergency Boration, to obtain _____ gpm boric acid flow rate (from Step 4).

☐ **b. THROTTLE OPEN** 1CS-274, RMUW Manual Blend from RMWST, to obtain _____ gpm dilution water flow rate (from Step 6).

Attachment 4
Sheet 4 of 4

Manual Makeup

INSTRUCTIONS	RESPONSE NOT OBTAINED

✷ ☐ **13.** **MONITOR** the following for expected response:

- Tavg
- Reactor power
- Control rod motion
- VCT level

14. **WHEN** desired VCT level has been reached,
THEN LOCALLY SHUT AND LOCK the following**:**

☐ • 1CS-274, RMUW Manual Blend from RMWST

☐ • 1CS-287, Manual Alternate Emergency Boration

☐ **15.** **EXIT** this attachment.

--END OF ATTACHMENT 4--

Facility: Harris Nuclear Plant

Task No.: 119013H304

Task Title: Determine Clearance Requirements for a CCW Pump

JPM No.: 2020 NRC Exam Admin JPM RO A2

K/A Reference: G 2.2.13 RO 4.1 SRO 4.3

ALTERNATE PATH - NO

Examinee: ______________________ NRC Examiner: ________________

Facility Evaluator: ______________________ Date: ________________

Method of testing:

Simulated Performance: ______ Actual Performance: __X__

Classroom __X__ Simulator ______ Plant ________

READ TO THE EXAMINEE I will explain the initial conditions, which steps to simulate, discuss, or perform and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	The plant is defueled. CCW Pump 1A-SA is required to be placed under a clearance for seal replacement. Cooling water and lube oil systems are NOT required to be placed under clearance. There is NO known isolation boundary leakage.

Initiating Cue:	You have been directed to determine the clearance requirements for CCW Pump 1A-SA using the SFDs, and Plant Procedures, as necessary. **NOTE: LISTING OF CIT'S IS NOT REQUIRED FOR THIS JPM.** **IT IS NOT INTENDED THAT YOU ACTUALLY GENERATE A CLEARANCE. ONLY PROVIDE THE EVALUATOR WITH A LISTING OF THE REQUIRED COMPONENTS, POSITIONS AND THE INSTALLATION SEQUENCE.**

Task Standard: Provide complete electrical and mechanical isolation of CCW Pump 1A-SA

Required Materials:
AD-OP-ALL-0200, Equipment Clearance, pgs. 40, 41 and 42, Rev. 20
OP-145, Component Cooling Water, Rev. 80
SFD 2165 S-1319
CWD 6-B-401 941
Additional copies of page 12 of this JPM available

General References:
AD-OP-ALL-0200, Equipment Clearance, Rev. 20
OP-145, Component Cooling Water, Rev. 80
SFD 2165 S-1319, 1320, and 1321
CWD 6-B-401 941, 942, and 943

OR

2020 NRC Exam Frozen Procedures Folder

Handouts:
JPM Cue Sheets
SFD 2165 S-1319

Time Critical Task: No

Validation Time: 20 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 2	Critical to remove power from the pump for personnel protection.
Step 3	Critical to isolate suction source to allow pump to be depressurized.
Step 4	Critical to isolate discharge path to allow pump to be depressurized.
Step 5	Critical to open vent path to allow pump to depressurize.
Step 6	Critical to open drain path to allow pump to depressurize.

NOTE: Provide applicants a copy of SFD 2165 S-1319, 1320 and 1321 along with CWD 6-B-401 Sheet 941, 942 and 943.

Laptops are to be used for this JPM.

Prior to starting this JPM: Ensure each candidate is familiar with the contents of the frozen procedures and are able to access the files containing OP-145, Component Cooling Water Activities and AD-OP-ALL-0200, Equipment Clearance.

START TIME: ____________

Performance Step: 1 Obtain a copy of the appropriate drawings and procedures **(AD-OP-ALL-0200, OP-145, SFD 2165 S-1319, CWD 6-B-401 Sheet 941)**

Standard: Operator obtains a copy of OP-145 to determine electrical requirements. SFD 2165 S-1319 to determine mechanical requirements. AD-OP-ALL-0200 to determine proper installation sequence for clearance.

Comment:

Evaluator Note:	***SEE JPM ATTACHMENT FOR A COMPLETE LISTING OF EACH COMPONENT AND REQUIRED POSITION. JPM STEPS ARE NOT REQUIRED TO BE PERFORMED IN THE LISTED SEQUENCE.***

✓ **Performance Step: 2** **Determine the electrical supply breaker for CCW Pump 1A-SA**

Standard: Refers to CWD 6-B-401 Sheet 941, OP-145 (or any other valid source) and determines the electrical supply breaker for CCW Pump 1A-SA to be 6.9 KV Emergency Bus 1A-SA, Cubicle 8

(BREAKER RACKED OUT)

Also determines pump has MCB and ACP switch and includes a CIT on CCW Pump 1A-SA switch for each location

Comment:

Evaluator Note:	**CRITICAL TO REMOVE POWER FROM PUMP.**

✓ **Performance Step: 3** **Determine the discharge isolation for CCW Pump 1A-SA**

Standard: Refers to S-1319 and determines the valve to isolate CCW Pump 1A-SA discharge is 1CC-36, CCW Pump A Discharge Isol Valve **(CLOSE)**

Comment:

✓ **Performance Step: 4** **Determine the suction valve for CCW Pump 1A-SA**

Standard: Refers to S-1319 and determines the suction valve for CCW Pump 1A-SA to be 1CC-27, CCW Pump A Suction Valve **(CLOSE)**

Comment:

✓ **Performance Step: 5** **Determine the vent path for CCW Pump 1A-SA**

Standard: Refers to S-1319 and determines the valve to vent CCW Pump 1A-SA is 1CC-28, CCW Pump A Suction Pressure Tap

(OPEN WITH CAP REMOVED)

Comment:

Evaluator Note:	***EITHER STEP 5 <u>OR</u> STEP 6 IS CRITICAL TO DEPRESSURIZE THE SYSTEM. ONE <u>OR</u> THE OTHER MUST BE PERFORMED, BUT <u>NOT BOTH</u>. HOWEVER, IF BOTH ARE PERFORMED, THIS IS ALSO ACCEPTABLE.***

✓ **Performance Step: 6** **Determine the drain path for CCW Pump 1A-SA**

Standard: Refers to S-1319 and determines the valves to drain CCW Pump 1A-SA suction piping to be 1CC-29, CCW Pump A Suction Drain Valve and discharge piping to be 1CC-30, CCW Pump A Discharge Drain Valve, and 1CC-31, CCW Pump A Discharge Line Drain Isol Valve

(ALL OPEN)

Comment:

Evaluator Note:	***EITHER STEP 5 OR STEP 6 IS CRITICAL TO DEPRESSURIZE THE SYSTEM. ONE OR THE OTHER MUST BE PERFORMED, BUT NOT BOTH. HOWEVER, IF BOTH ARE PERFORMED, THIS IS ALSO ACCEPTABLE.***

Evaluator Cue:	**When applicant completes and returns clearance list.** **END OF JPM**

Stop Time: _________

KEY
JPM ATTACHMENT

COMPONENT LISTING AND REQUIRED POSITIONS

Critical sequences:

1) Remove power from the CCW Pump 1A-SA
2) Shut 1CC-36, CCW Pump 1A-SA Discharge Isol Valve
2) Shut 1CC-27, CCW Pump 1A-SA Suction Isol Valve
3) Open Vent and/or Drain to depressurize boundary

	COMPONENT	**POSITION**
1)	CCW Pump 1A-SA P.S. - 6.9 KV Emergency Bus 1A-SA, Cubicle 8.	Racked Out
2)	1CC-36, CCW Pump 1A-SA, Discharge Isol Valve	Shut
3)	1CC-27, CCW Pump 1A-SA, Suction Isol Valve	Shut

4) Accept - EITHER **one vent path** OR **the drain path** or **BOTH a vent path and drain path**.

NOTE: Any of the following vent valves will support a vent path for the pump. One or more of these vent paths are required to be identified

VENT PATHS

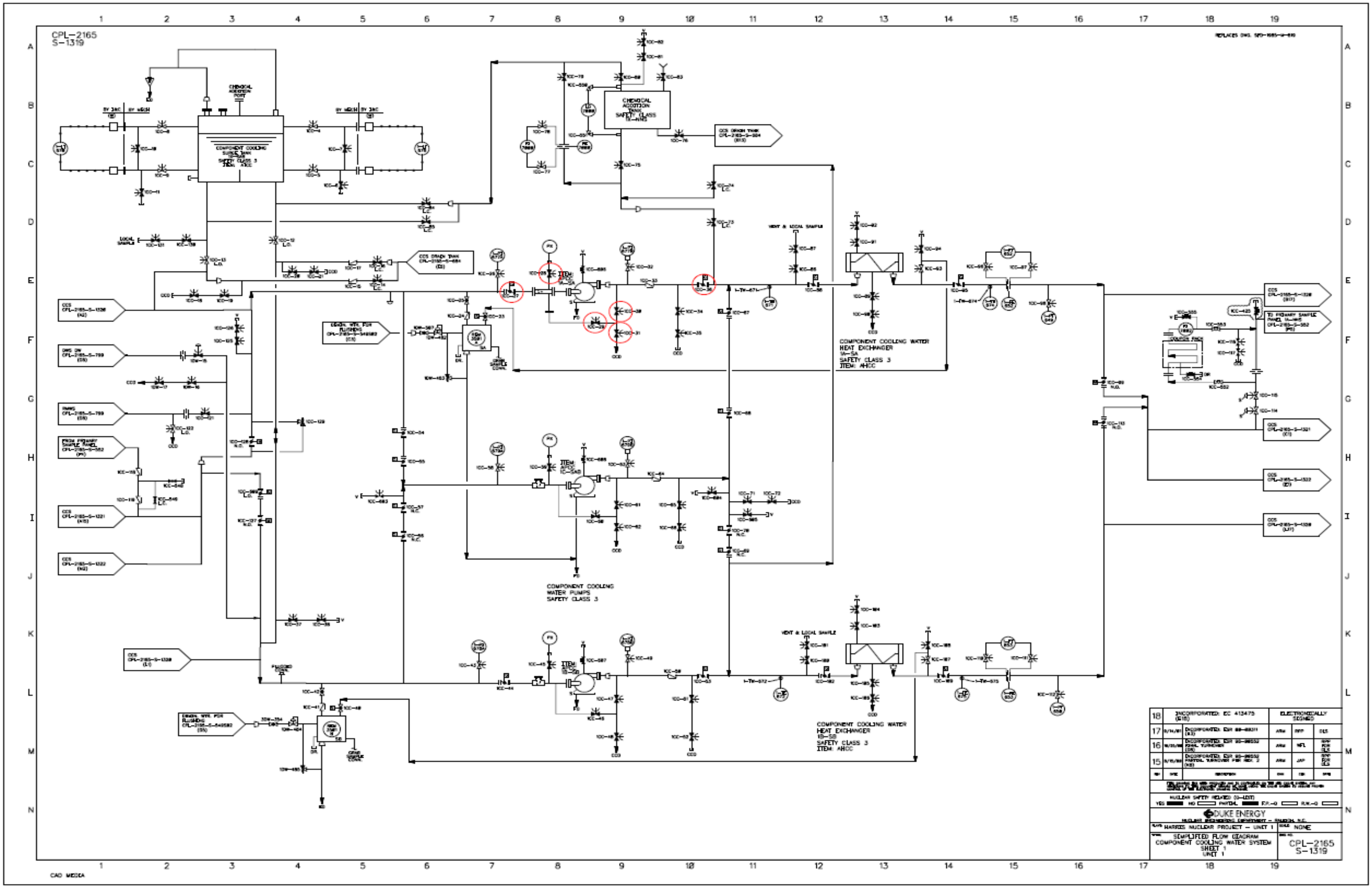
1CC-28, CCW Pump A Suction Pressure Tap	Uncapped/Open
- **OR** –	
1CC-606, CCW Pump 1A Casing Vent Valve	Uncapped/Open
- **OR** –	
1CC-32, PI-677B Root Isolation Valve	Uncapped/Open

DRAIN PATH

1CC-29, CCW Pump A Suction Drain Valve	Open
- **AND** -	
1CC-30, CCW Pump A Discharge Drain Valve	Open
- **AND** -	
1CC-31, CCW Pump A Discharge Drain Isolation Valve	Open
- **OR** –	
1CC-34, CCW Pump A Disch Line Inner Drain Valve	Open
- **AND** -	
1CC-35, CCW Pump A Disch Line Outer Drain Valve	Uncapped/Open

NOTE – CITs are NOT required for satisfactory completion of JPM.

KEY


CPL-2165
S-1319
CHEMICAL ADDITION TANK SAFETY CLASS 3
COMPONENT COOLING WATER PUMPS SAFETY CLASS 3
COMPONENT COOLING WATER HEAT EXCHANGER 1A-SA SAFETY CLASS 3 ITEM: AHCC
COMPONENT COOLING WATER HEAT EXCHANGER 1B-SB SAFETY CLASS 3 ITEM: AHCC
VENT & LOCAL SAMPLE
18 INCORPORATED: EC 413475 (E18)
ELECTRONICALLY SIGNED
NUCLEAR SAFETY RELATED (Q-LIST)
DUKE ENERGY
HARRIS NUCLEAR PROJECT - UNIT 1
NONE
SIMPLIFIED FLOW DIAGRAM COMPONENT COOLING WATER SYSTEM SHEET 1 UNIT 1
CPL-2165 S-1319
CAD MEDIA

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Admin JPM RO A2
Determine Clearance Requirements for a CCW Pump
AD-OP-ALL-0200
OP-145

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	**The plant is defueled. CCW Pump 1A-SA is required to be placed under a clearance for seal replacement. Cooling water and lube oil systems are NOT required to be placed under clearance. There is NO known isolation boundary leakage.**

Initiating Cue:	**You have been directed to determine the clearance requirements for CCW Pump 1A-SA using the SFDs, and Plant Procedures, as necessary. The AOM-Shift has approved using single valve isolation.** **NOTE: LISTING OF CIT'S IS NOT REQUIRED FOR THIS JPM.** **IT IS NOT INTENDED THAT YOU ACTUALLY GENERATE A CLEARANCE. ONLY PROVIDE THE EVALUATOR WITH A LISTING OF THE REQUIRED COMPONENTS, POSITIONS AND THE INSTALLATION SEQUENCE.**

NOTE: Provide a list of components in the proper installation sequence to the examiner using the following page(s).

Additional pages are available upon request.

Name: ____________________________________

Date: ____________________

CCW Pump 1A-SA CLEARANCE COMPONENT LISTING AND REQUIRED POSITIONS

SEQUENCE	**COMPONENT**	**POSITION**

Facility:	Harris Nuclear Plant	Task No.:	344171H404
Task Title:	Given a set of conditions, determine and apply the facility dose limits.	JPM No.:	2020 NRC Exam Admin JPM RO A3
K/A Reference:	G 2.3.7 RO 3.5 SRO 3.6		**ALTERNATE PATH - NO**
Examinee:	______________________	NRC Examiner:	________________
Facility Evaluator:	______________________	Date:	________________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom X Simulator ______ Plant ________

READ TO THE EXAMINEE	
I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	• A fire has occurred in 1-A-SWGRA • The reactor is tripped • The operating crew is performing AOP-036.08, Fire Areas: 1-A-SWGRA, 1-A-SWGRB • Charging flow cannot be controlled from the control room

Initiating Cue:	• You have been assigned to locally control charging in accordance with AOP-036.08, Section 3.1, Step 10.d • This is not considered to be an emergency evolution. Your accumulated TEDE dose for this year is 1550 mrem • You will be performing the evolution under RWP # 23, Operations Activities • Identify the Minimum Operation Activities Task # to perform this evolution • Determine the maximum permissible stay time before the first Stop Work limit requires you to exit the area **(Assume you remain at the valves and 0 dose is received in transit)**

Task Standard: Lowest facility limit determined and stay time calculated within tolerance band.

Required Materials: Calculator

General References: AOP-036.08, Fire Areas: 1-A-SWGRA, 1-A-SWGRB, Rev. 21

AD-RP-ALL-2000, Preparation And Management Of Radiation Work Permits (RWP), Rev. 4

RWP # 23 Operations Activities, Rev. 12
Valve Map 9, RAB 236’ Mechanical Penetration Area
Survey HNP-M-20200621-4, RAB 236’ Mechanical Penetration Area
OR
2020 NRC Exam Frozen Procedures Folder

Handout: JPM Cue Sheets pages 5 - 19

Time Critical Task: No

Validation Time: 10 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 2	Must determine the location of the valve in order to calculate the dose value until an alarm limit is reached
Step 3	Must determine the area classification of the valve location in order to minimum task dose alarm setpoint and dose rate alarm limits
Step 4	Must determine the RWP limits in order to calculate the dose value until an alarm limit is reached
Step 5	Must determine the time allowed in order to exit the area once the alarm limit is reached

(Denote Critical Steps with a check mark)

START TIME: ____________

Performance Step: 1 Determine the general location of the valves from AOP-036.08 on the survey map.

Standard: Uses Valve Map to determine location of the required valves

Locates general area on the Survey Map.

Evaluator Cue:

- **Provide the handout.**
- **Assume that all handout materials are the most recent, approved documents.**

Comment:

✓ **Performance Step: 2** Determine the radiation level in the area of the valves.

Standard: Using Survey HNP-M-20200621-4, determines general radiation level in the area of the valves to be 3 mr/hr.

Comment:

✓ **Performance Step: 3** Determines the RCA classification of the work area.

Standard: Using Survey HNP-M-20200621-4 and RWP # 23 determines the work area is only a RA and Task # 1 Operations Activities (No HRA Access) is the minimum RWP task required to perform the evolution.

Comment:

✓ **Performance Step: 4** Determine the first Stop Work limit.

Standard: Reviews RWP # 23 Task # 1 and determines first Stop Work limit will be reached when the Alarming Dosimeter actuates:

- 8 mr accumulated dose (80% of 10 mr)

or

- 75 mr/hr dose rate.

Comment:

✓ **Performance Step: 5** Calculate maximum stay time.

Standard: (8 mr)(1 hr/3 mr) = 2.67 hours or 2 hours and 40 minutes

≥ 2.1 hours ≤ 2.67 hours or

≥ 2 hours and 6 minutes ≤ 2 hours and 40 minutes.

Evaluator Note: **Tolerance allows for 5% error on the low end without exceeding the upper limit. 5% is permitted in the event the candidate interpolates the radiation level using adjacent areas.**

Comment:

Terminating Cue: **After stay time is reported: Evaluation on this JPM is complete.**

STOP TIME: ______________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC ADM JPM RO A3

Given a set of conditions, determine and apply the facility dose limits.

AD-RP-ALL-2000,

RWP # 23, Operations Activities

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	• A fire has occurred in 1-A-SWGRA • The reactor is tripped • The operating crew is performing AOP-036.08, Fire Areas: 1-A-SWGRA, 1-A-SWGRB • Charging flow cannot be controlled from the control room

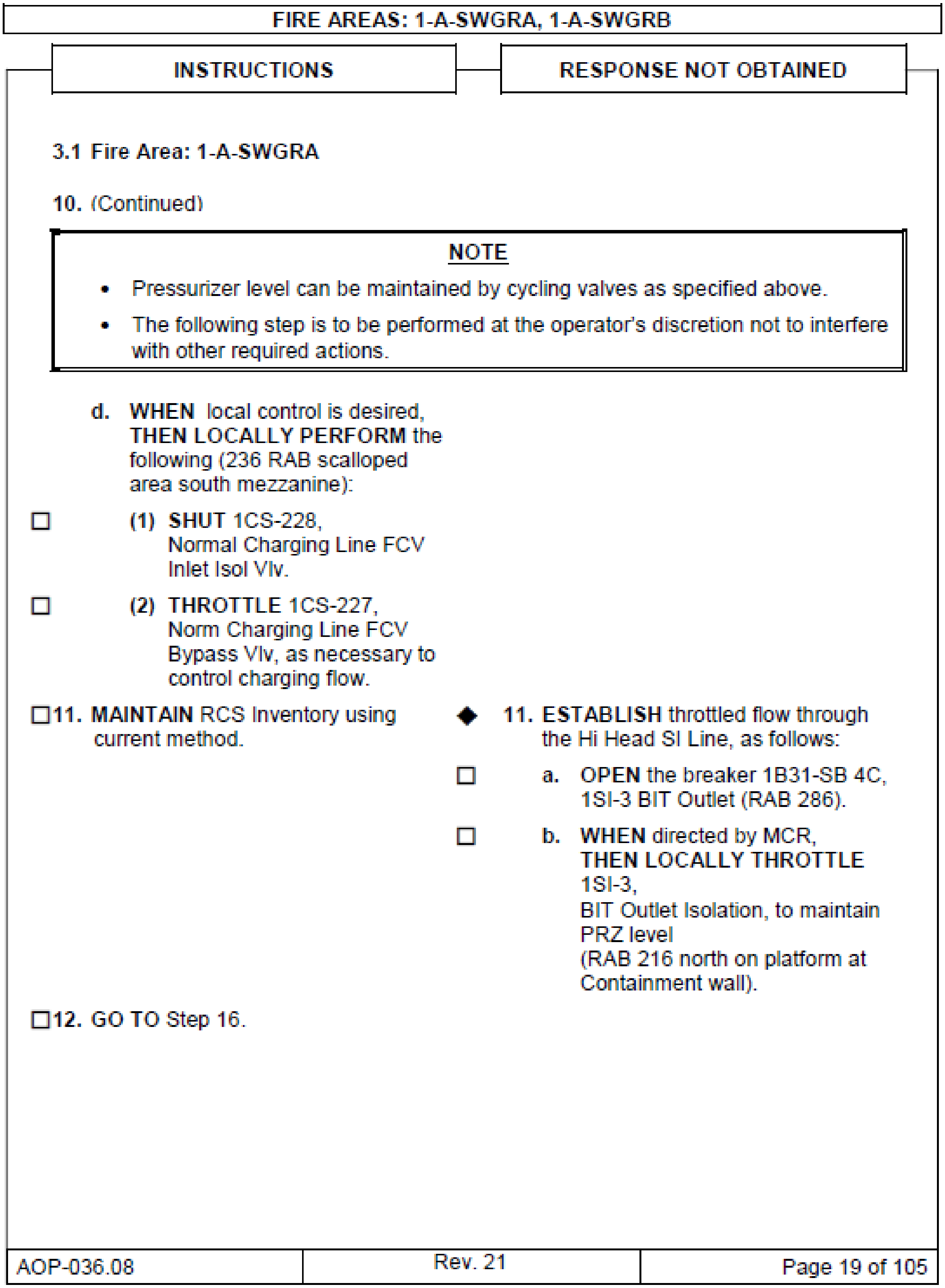
Initiating Cue:	• You have been assigned to locally control charging in accordance with AOP-036.08, Section 3.1, Step 10.d • This is not considered to be an emergency evolution. Your accumulated TEDE dose for this year is 1550 mrem • You will be performing the evolution under RWP # 23, Operations Activities • Identify the Minimum Operation Activities Task # to perform this evolution • Determine the maximum permissible stay time before the first Stop Work limit requires you to exit the area **(Assume you remain at the valves and 0 dose is received in transit)**

Name: ____

Date: ___________________

1. Minimum Operation Activities Task # _______________________

2. Maximum permissible stay time before the first Stop Work limit requires you to exit the area for the identified Task # is ______________________



FIRE AREAS: 1-A-SWGRA, 1-A-SWGRB

INSTRUCTIONS	RESPONSE NOT OBTAINED

3.1 Fire Area: 1-A-SWGRA

10. (Continued)

> **NOTE**
>
> - Pressurizer level can be maintained by cycling valves as specified above.
> - The following step is to be performed at the operator's discretion not to interfere with other required actions.

d. **WHEN** local control is desired, **THEN LOCALLY PERFORM** the following (236 RAB scalloped area south mezzanine):

☐ **(1)** **SHUT** 1CS-228, Normal Charging Line FCV Inlet Isol Vlv.

☐ **(2)** **THROTTLE** 1CS-227, Norm Charging Line FCV Bypass Vlv, as necessary to control charging flow.

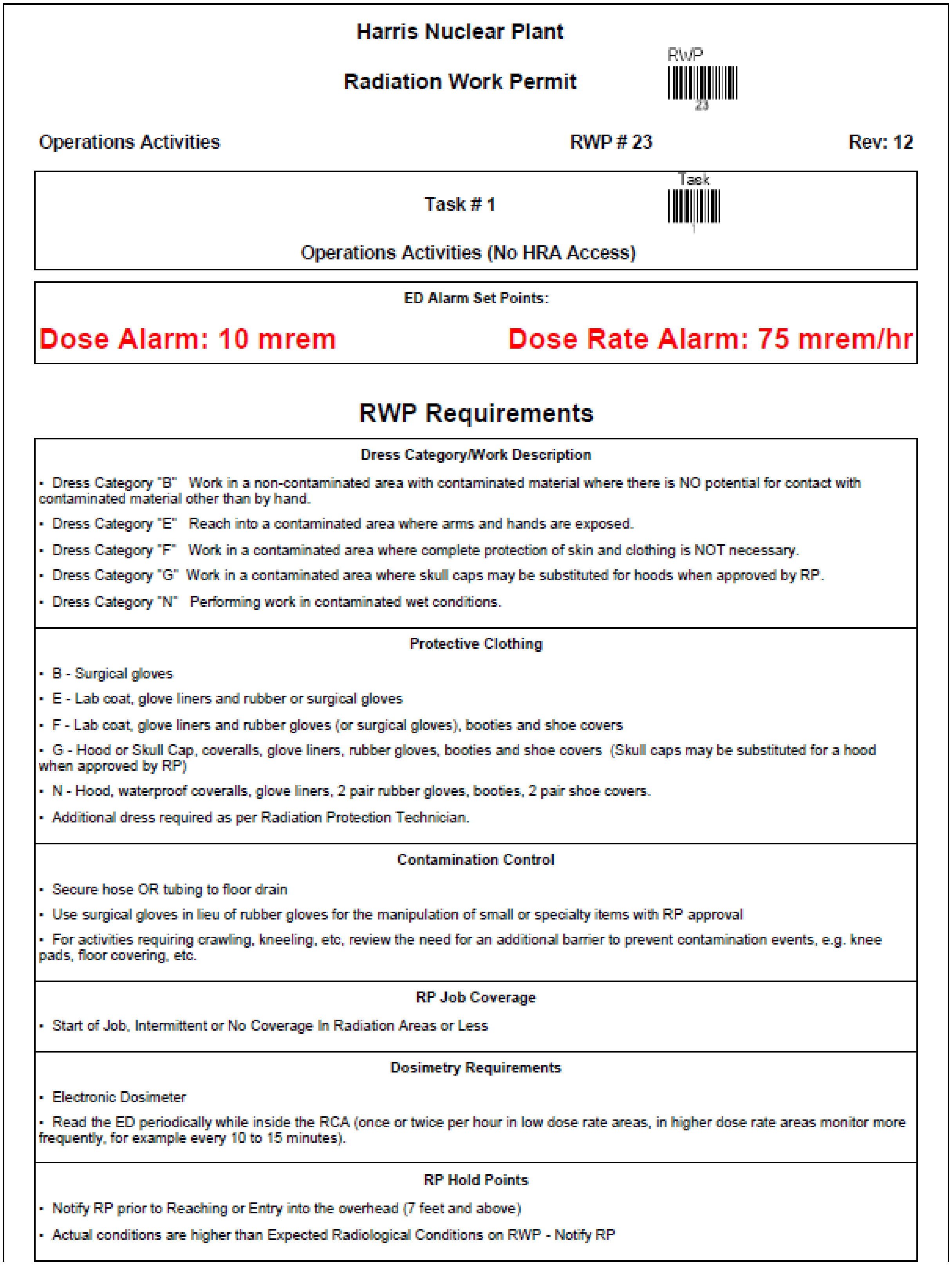
☐**11.** **MAINTAIN** RCS Inventory using current method.

◆ **11.** **ESTABLISH** throttled flow through the Hi Head SI Line, as follows:

☐ **a.** **OPEN** the breaker 1B31-SB 4C, 1SI-3 BIT Outlet (RAB 286).

☐ **b.** **WHEN** directed by MCR, **THEN LOCALLY THROTTLE** 1SI-3, BIT Outlet Isolation, to maintain PRZ level (RAB 216 north on platform at Containment wall).

☐**12.** GO TO Step 16.



Harris Nuclear Plant

Radiation Work Permit

RWP
23

Operations Activities RWP # 23 Rev: 12

Task
1

Task # 1

Operations Activities (No HRA Access)

ED Alarm Set Points:

Dose Alarm: 10 mrem **Dose Rate Alarm: 75 mrem/hr**

RWP Requirements

Dress Category/Work Description

- Dress Category "B" Work in a non-contaminated area with contaminated material where there is NO potential for contact with contaminated material other than by hand.
- Dress Category "E" Reach into a contaminated area where arms and hands are exposed.
- Dress Category "F" Work in a contaminated area where complete protection of skin and clothing is NOT necessary.
- Dress Category "G" Work in a contaminated area where skull caps may be substituted for hoods when approved by RP.
- Dress Category "N" Performing work in contaminated wet conditions.

Protective Clothing

- B - Surgical gloves
- E - Lab coat, glove liners and rubber or surgical gloves
- F - Lab coat, glove liners and rubber gloves (or surgical gloves), booties and shoe covers
- G - Hood or Skull Cap, coveralls, glove liners, rubber gloves, booties and shoe covers (Skull caps may be substituted for a hood when approved by RP)
- N - Hood, waterproof coveralls, glove liners, 2 pair rubber gloves, booties, 2 pair shoe covers.
- Additional dress required as per Radiation Protection Technician.

Contamination Control

- Secure hose OR tubing to floor drain
- Use surgical gloves in lieu of rubber gloves for the manipulation of small or specialty items with RP approval
- For activities requiring crawling, kneeling, etc, review the need for an additional barrier to prevent contamination events, e.g. knee pads, floor covering, etc.

RP Job Coverage

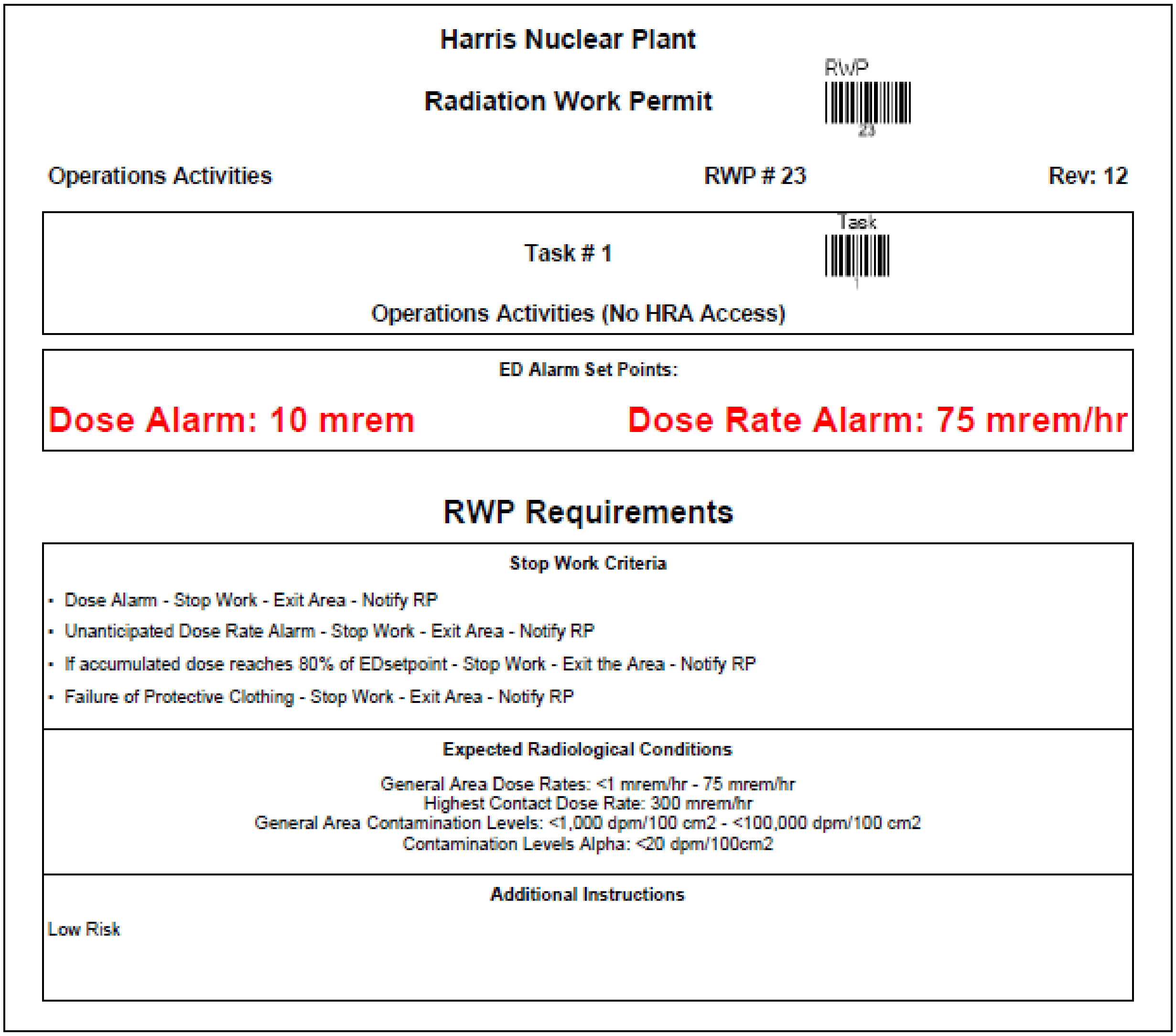
- Start of Job, Intermittent or No Coverage In Radiation Areas or Less

Dosimetry Requirements

- Electronic Dosimeter
- Read the ED periodically while inside the RCA (once or twice per hour in low dose rate areas, in higher dose rate areas monitor more frequently, for example every 10 to 15 minutes).

RP Hold Points

- Notify RP prior to Reaching or Entry into the overhead (7 feet and above)
- Actual conditions are higher than Expected Radiological Conditions on RWP - Notify RP



Harris Nuclear Plant

Radiation Work Permit

Operations Activities RWP # 23 Rev: 12

Task # 1

Operations Activities (No HRA Access)

ED Alarm Set Points:

Dose Alarm: 10 mrem

Dose Rate Alarm: 75 mrem/hr

RWP Requirements

Stop Work Criteria

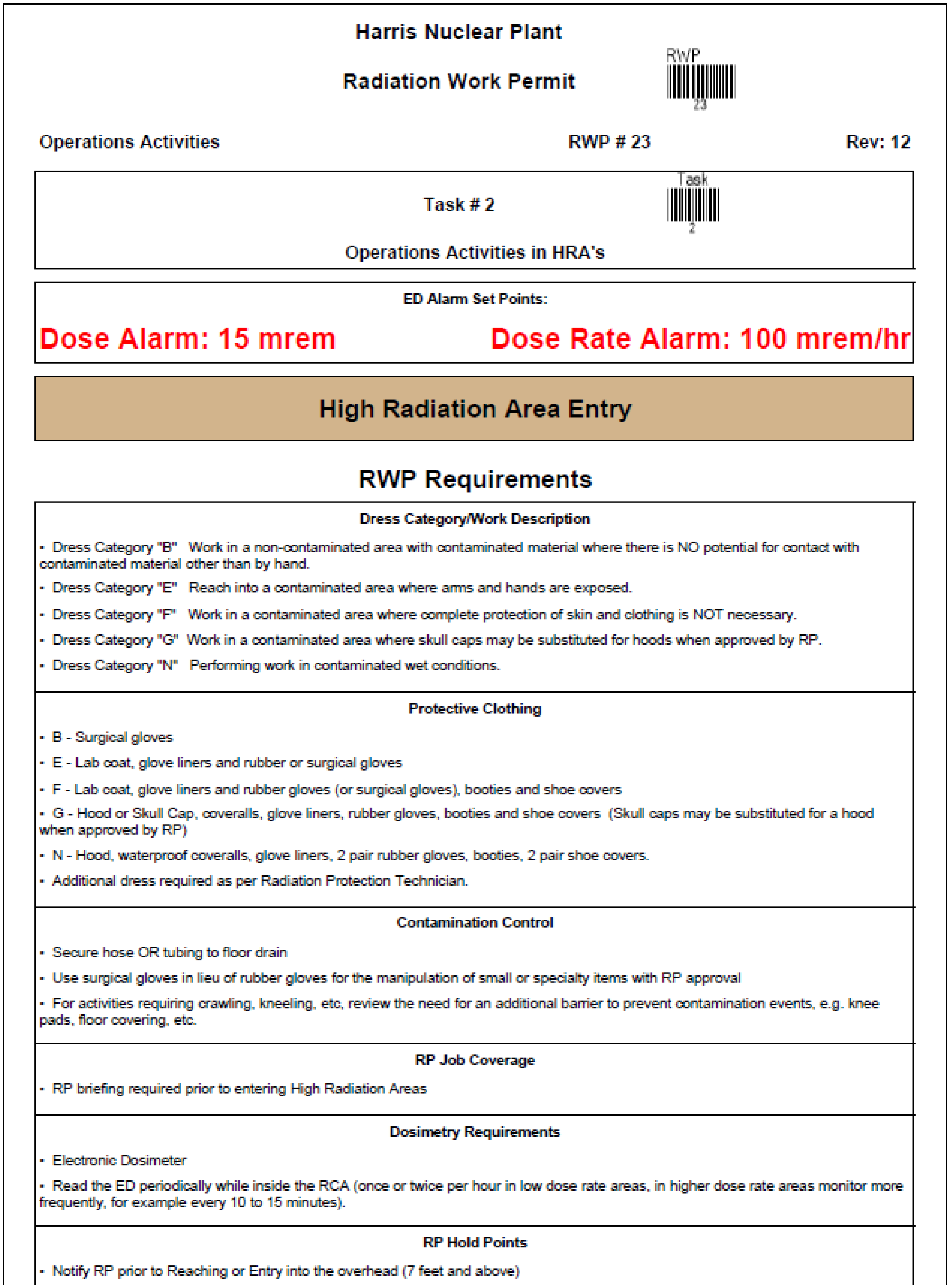
- Dose Alarm - Stop Work - Exit Area - Notify RP
- Unanticipated Dose Rate Alarm - Stop Work - Exit Area - Notify RP
- If accumulated dose reaches 80% of EDsetpoint - Stop Work - Exit the Area - Notify RP
- Failure of Protective Clothing - Stop Work - Exit Area - Notify RP

Expected Radiological Conditions

General Area Dose Rates: <1 mrem/hr - 75 mrem/hr
Highest Contact Dose Rate: 300 mrem/hr
General Area Contamination Levels: <1,000 dpm/100 cm2 - <100,000 dpm/100 cm2
Contamination Levels Alpha: <20 dpm/100cm2

Additional Instructions

Low Risk



Harris Nuclear Plant

Radiation Work Permit

RWP

23

Operations Activities **RWP # 23** **Rev: 12**

Task

Task # 2

2

Operations Activities in HRA's

ED Alarm Set Points:

Dose Alarm: 15 mrem **Dose Rate Alarm: 100 mrem/hr**

High Radiation Area Entry

RWP Requirements

Dress Category/Work Description

- Dress Category "B" Work in a non-contaminated area with contaminated material where there is NO potential for contact with contaminated material other than by hand.
- Dress Category "E" Reach into a contaminated area where arms and hands are exposed.
- Dress Category "F" Work in a contaminated area where complete protection of skin and clothing is NOT necessary.
- Dress Category "G" Work in a contaminated area where skull caps may be substituted for hoods when approved by RP.
- Dress Category "N" Performing work in contaminated wet conditions.

Protective Clothing

- B - Surgical gloves
- E - Lab coat, glove liners and rubber or surgical gloves
- F - Lab coat, glove liners and rubber gloves (or surgical gloves), booties and shoe covers
- G - Hood or Skull Cap, coveralls, glove liners, rubber gloves, booties and shoe covers (Skull caps may be substituted for a hood when approved by RP)
- N - Hood, waterproof coveralls, glove liners, 2 pair rubber gloves, booties, 2 pair shoe covers.
- Additional dress required as per Radiation Protection Technician.

Contamination Control

- Secure hose OR tubing to floor drain
- Use surgical gloves in lieu of rubber gloves for the manipulation of small or specialty items with RP approval
- For activities requiring crawling, kneeling, etc, review the need for an additional barrier to prevent contamination events, e.g. knee pads, floor covering, etc.

RP Job Coverage

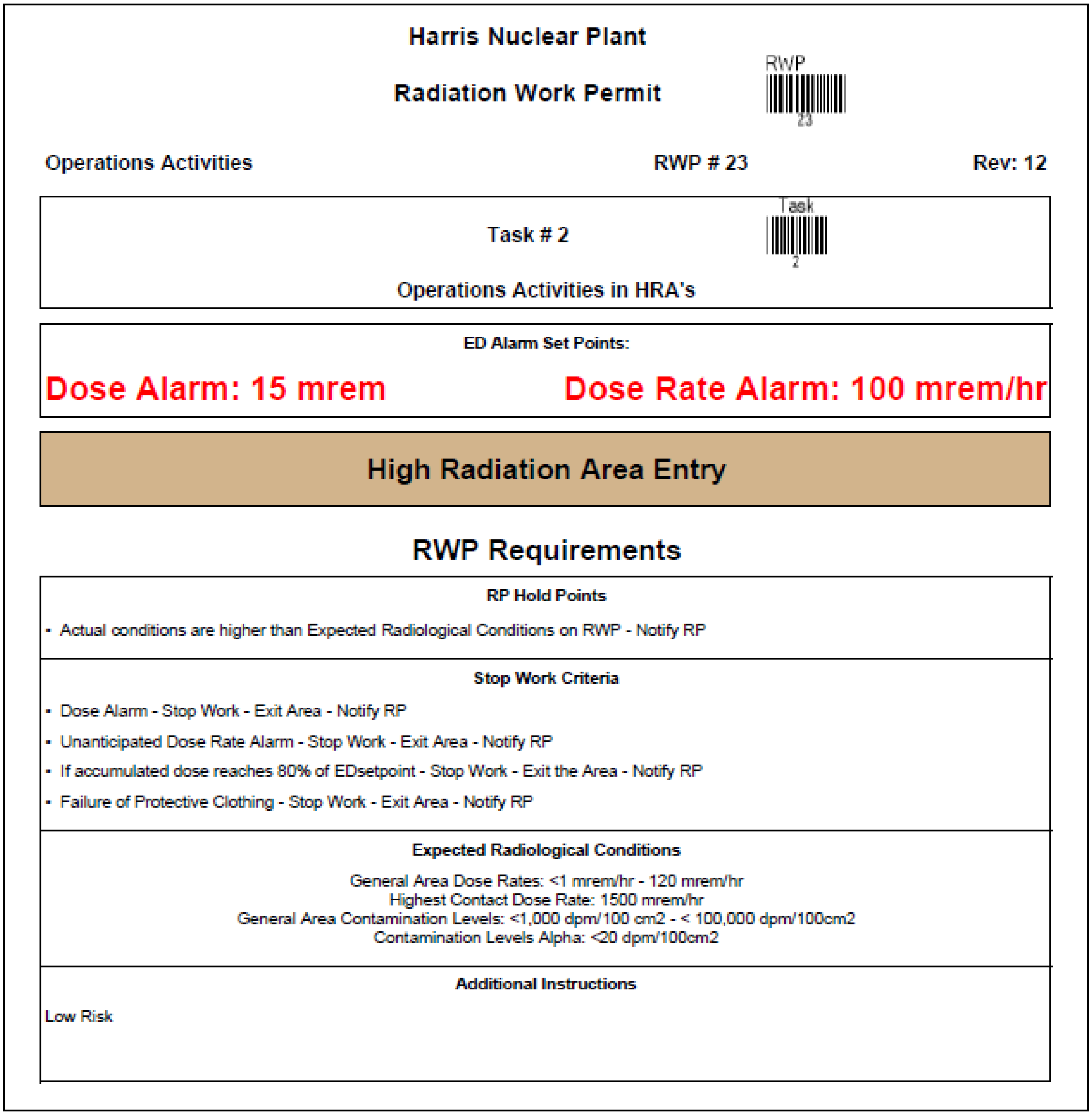
- RP briefing required prior to entering High Radiation Areas

Dosimetry Requirements

- Electronic Dosimeter
- Read the ED periodically while inside the RCA (once or twice per hour in low dose rate areas, in higher dose rate areas monitor more frequently, for example every 10 to 15 minutes).

RP Hold Points

- Notify RP prior to Reaching or Entry into the overhead (7 feet and above)



Harris Nuclear Plant

Radiation Work Permit

Operations Activities **RWP # 23** **Rev: 12**

Task # 2

Operations Activities in HRA's

ED Alarm Set Points:

Dose Alarm: 15 mrem **Dose Rate Alarm: 100 mrem/hr**

High Radiation Area Entry

RWP Requirements

RP Hold Points

- Actual conditions are higher than Expected Radiological Conditions on RWP - Notify RP

Stop Work Criteria

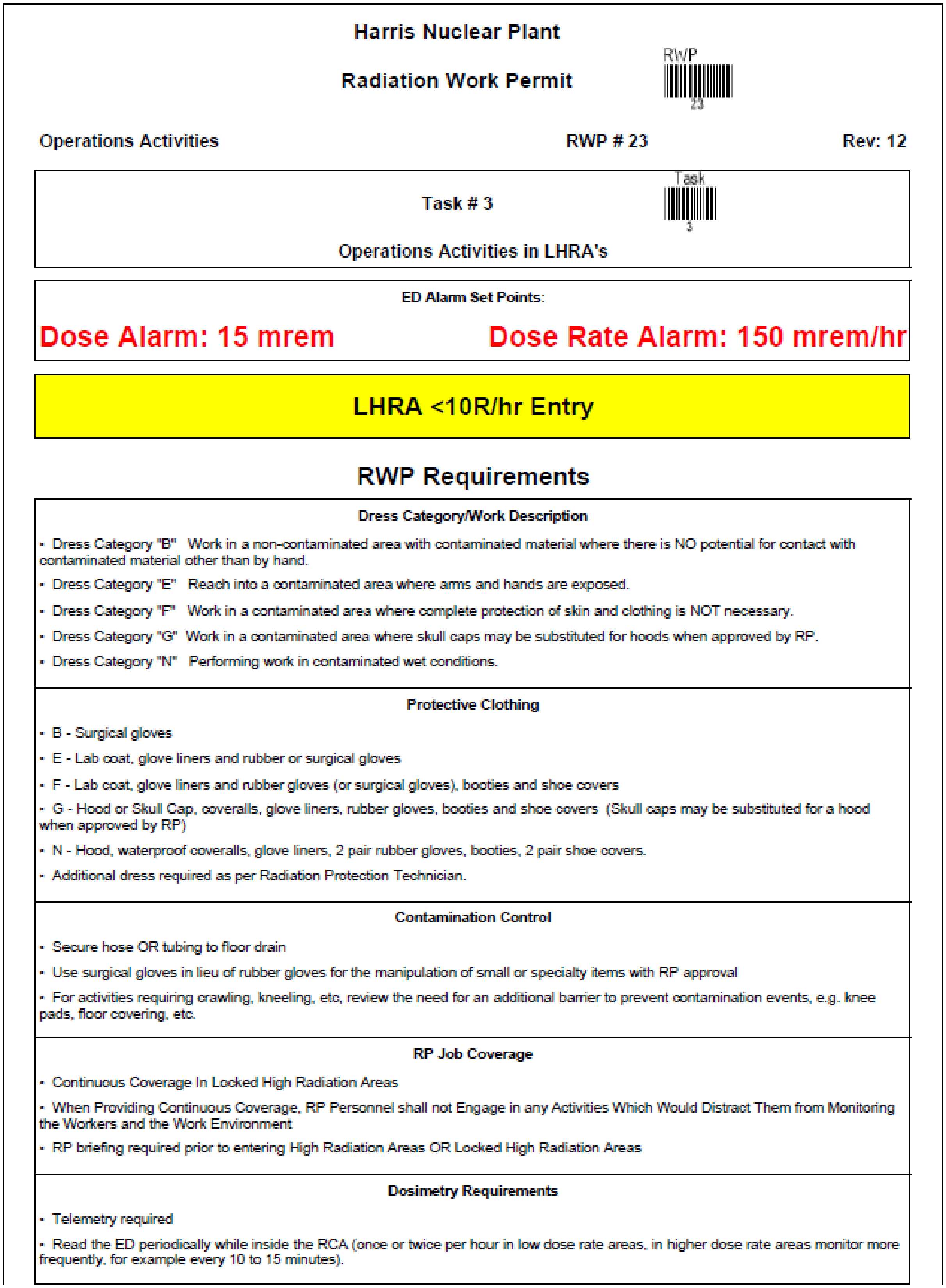
- Dose Alarm - Stop Work - Exit Area - Notify RP
- Unanticipated Dose Rate Alarm - Stop Work - Exit Area - Notify RP
- If accumulated dose reaches 80% of EDsetpoint - Stop Work - Exit the Area - Notify RP
- Failure of Protective Clothing - Stop Work - Exit Area - Notify RP

Expected Radiological Conditions

General Area Dose Rates: <1 mrem/hr - 120 mrem/hr
Highest Contact Dose Rate: 1500 mrem/hr
General Area Contamination Levels: <1,000 dpm/100 cm2 - < 100,000 dpm/100cm2
Contamination Levels Alpha: <20 dpm/100cm2

Additional Instructions

Low Risk



Harris Nuclear Plant

Radiation Work Permit

Operations Activities **RWP # 23** **Rev: 12**

Task # 3

Operations Activities in LHRA's

ED Alarm Set Points:

Dose Alarm: 15 mrem **Dose Rate Alarm: 150 mrem/hr**

LHRA <10R/hr Entry

RWP Requirements

Dress Category/Work Description

- Dress Category "B" Work in a non-contaminated area with contaminated material where there is NO potential for contact with contaminated material other than by hand.
- Dress Category "E" Reach into a contaminated area where arms and hands are exposed.
- Dress Category "F" Work in a contaminated area where complete protection of skin and clothing is NOT necessary.
- Dress Category "G" Work in a contaminated area where skull caps may be substituted for hoods when approved by RP.
- Dress Category "N" Performing work in contaminated wet conditions.

Protective Clothing

- B - Surgical gloves
- E - Lab coat, glove liners and rubber or surgical gloves
- F - Lab coat, glove liners and rubber gloves (or surgical gloves), booties and shoe covers
- G - Hood or Skull Cap, coveralls, glove liners, rubber gloves, booties and shoe covers (Skull caps may be substituted for a hood when approved by RP)
- N - Hood, waterproof coveralls, glove liners, 2 pair rubber gloves, booties, 2 pair shoe covers.
- Additional dress required as per Radiation Protection Technician.

Contamination Control

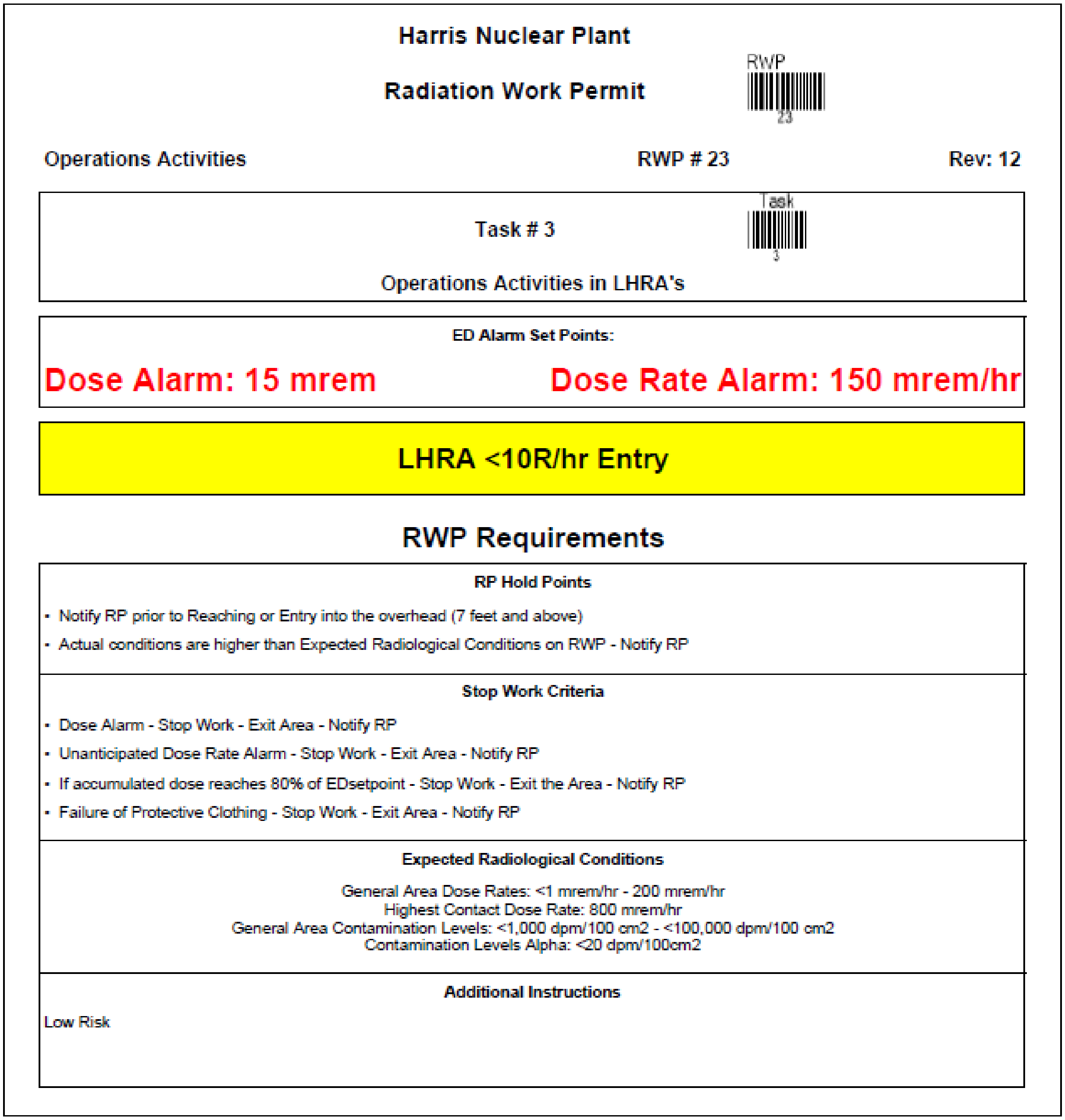
- Secure hose OR tubing to floor drain
- Use surgical gloves in lieu of rubber gloves for the manipulation of small or specialty items with RP approval
- For activities requiring crawling, kneeling, etc, review the need for an additional barrier to prevent contamination events, e.g. knee pads, floor covering, etc.

RP Job Coverage

- Continuous Coverage In Locked High Radiation Areas
- When Providing Continuous Coverage, RP Personnel shall not Engage in any Activities Which Would Distract Them from Monitoring the Workers and the Work Environment
- RP briefing required prior to entering High Radiation Areas OR Locked High Radiation Areas

Dosimetry Requirements

- Telemetry required
- Read the ED periodically while inside the RCA (once or twice per hour in low dose rate areas, in higher dose rate areas monitor more frequently, for example every 10 to 15 minutes).



Harris Nuclear Plant

Radiation Work Permit

Operations Activities **RWP # 23** **Rev: 12**

Task # 3

Operations Activities in LHRA's

ED Alarm Set Points:

Dose Alarm: 15 mrem **Dose Rate Alarm: 150 mrem/hr**

LHRA <10R/hr Entry

RWP Requirements

RP Hold Points

- Notify RP prior to Reaching or Entry into the overhead (7 feet and above)
- Actual conditions are higher than Expected Radiological Conditions on RWP - Notify RP

Stop Work Criteria

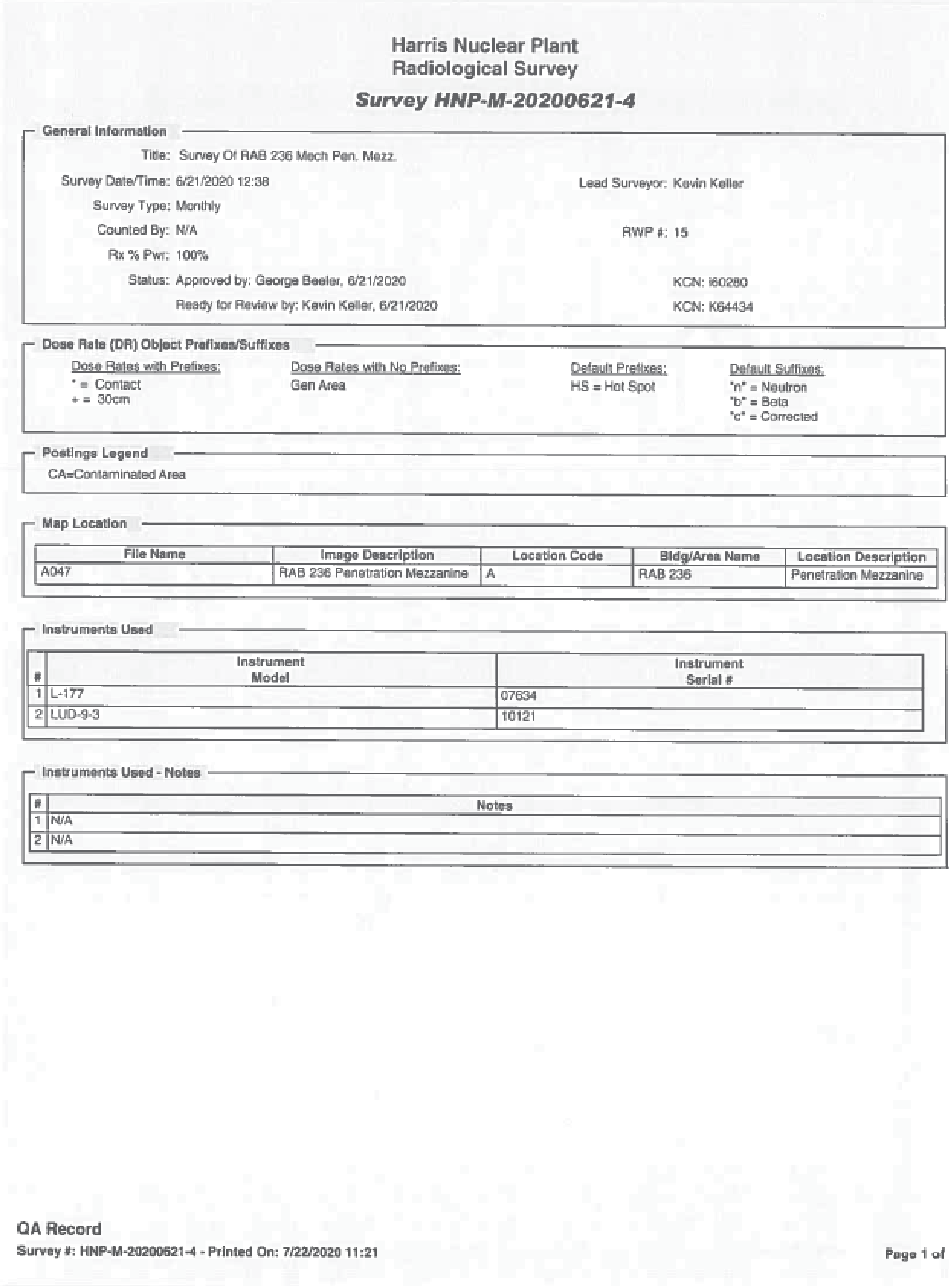
- Dose Alarm - Stop Work - Exit Area - Notify RP
- Unanticipated Dose Rate Alarm - Stop Work - Exit Area - Notify RP
- If accumulated dose reaches 80% of EDsetpoint - Stop Work - Exit the Area - Notify RP
- Failure of Protective Clothing - Stop Work - Exit Area - Notify RP

Expected Radiological Conditions

General Area Dose Rates: <1 mrem/hr - 200 mrem/hr
Highest Contact Dose Rate: 800 mrem/hr
General Area Contamination Levels: <1,000 dpm/100 cm2 - <100,000 dpm/100 cm2
Contamination Levels Alpha: <20 dpm/100cm2

Additional Instructions

Low Risk



Harris Nuclear Plant
Radiological Survey
Survey HNP-M-20200621-4

General Information

Title: Survey Of RAB 236 Mech Pen. Mezz.
Survey Date/Time: 6/21/2020 12:38
Lead Surveyor: Kevin Keller
Survey Type: Monthly
Counted By: N/A
RWP #: 15
Rx % Pwr: 100%
Status: Approved by: George Beeler, 6/21/2020
KCN: I60280
Ready for Review by: Kevin Keller, 6/21/2020
KCN: K64434

Dose Rate (DR) Object Prefixes/Suffixes

Dose Rates with Prefixes:	Dose Rates with No Prefixes:	Default Prefixes:	Default Suffixes:
* = Contact + = 30cm	Gen Area	HS = Hot Spot	"n" = Neutron "b" = Beta "c" = Corrected

Postings Legend

CA=Contaminated Area

Map Location

File Name	Image Description	Location Code	Bldg/Area Name	Location Description
A047	RAB 236 Penetration Mezzanine	A	RAB 236	Penetration Mezzanine

Instruments Used

#	Instrument Model	Instrument Serial #
1	L-177	07634
2	LUD-9-3	10121

Instruments Used - Notes

#	Notes
1	N/A
2	N/A

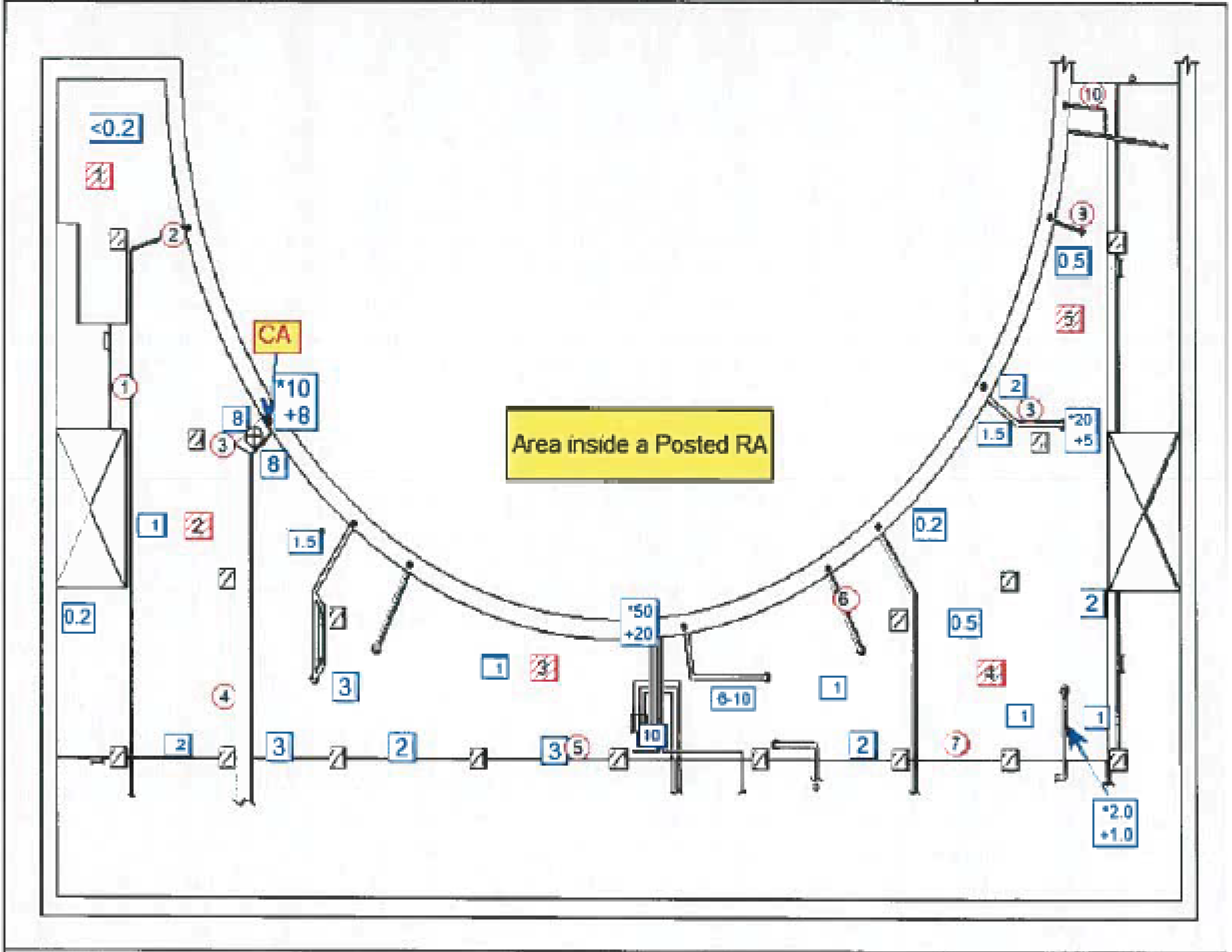
QA Record

Survey #: HNP-M-20200621-4 - Printed On: 7/22/2020 11:21

Page 1 of

Harris Nuclear Plant
Radiological Survey

RAB 236 Penetration Mezzanine	**Survey #: HNP-M-20200621-4**	**Date/Time: 6/21/2020 12:38**



Comments:

Type: Monthly

Symbol Legend (for example only)

Dose Rate
*150 ← Contact Reading
+ 75 ← 30 cm Reading
20 ← General Area
HS-50 Hot Spot
RCA Posting
Drip Bag
⑮ Smear
15 Air Sample
RM
15 Wipe

RWP #: 15
Reactor Power = 100%
Tag Name: Monthly Survey

Unless otherwise noted, dose rates in mrem/hr.

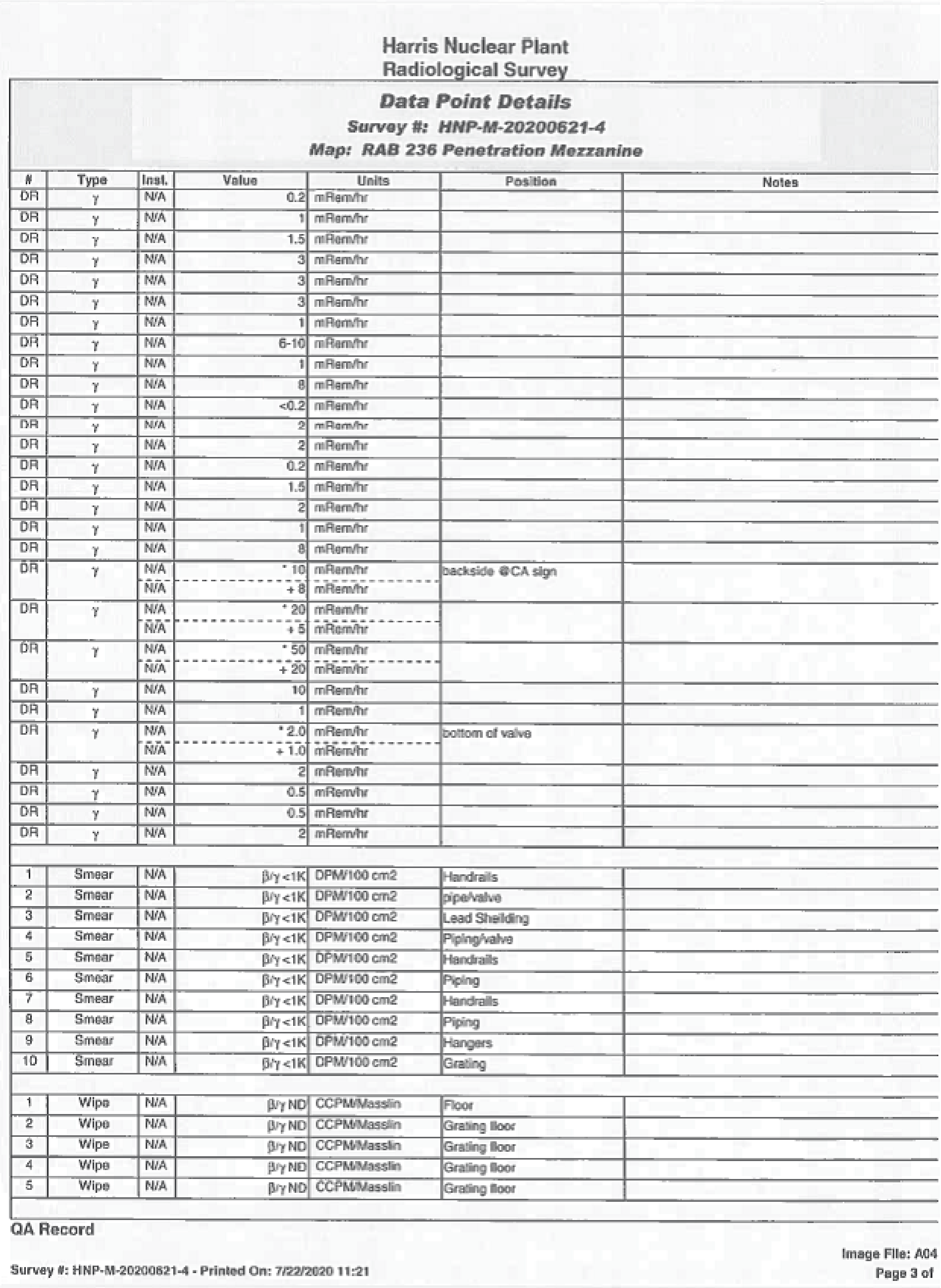
Lead Surveyor: Kevin Keller	Status: Approved by: George Beeler, 6/21/2020
Location Code: A	Bldg/Area Name: RAB 236
Location Description: Penetration Mezzanine	

QA Record

Image File: A04

Survey #: HNP-M-20200621-4 - Printed On: 7/22/2020 11:21

Page 2 of



Harris Nuclear Plant
Radiological Survey

Data Point Details

Survey #: HNP-M-20200621-4

Map: RAB 236 Penetration Mezzanine

#	Type	Inst.	Value	Units	Position	Notes
DR	γ	N/A	0.2	mRem/hr		
DR	γ	N/A	1	mRem/hr		
DR	γ	N/A	1.5	mRem/hr		
DR	γ	N/A	3	mRem/hr		
DR	γ	N/A	3	mRem/hr		
DR	γ	N/A	3	mRem/hr		
DR	γ	N/A	1	mRem/hr		
DR	γ	N/A	6-10	mRem/hr		
DR	γ	N/A	1	mRem/hr		
DR	γ	N/A	8	mRem/hr		
DR	γ	N/A	<0.2	mRem/hr		
DR	γ	N/A	2	mRem/hr		
DR	γ	N/A	2	mRem/hr		
DR	γ	N/A	0.2	mRem/hr		
DR	γ	N/A	1.5	mRem/hr		
DR	γ	N/A	2	mRem/hr		
DR	γ	N/A	1	mRem/hr		
DR	γ	N/A	8	mRem/hr		
DR	γ	N/A	* 10	mRem/hr	backside @CA sign	
		N/A	+ 8	mRem/hr		
DR	γ	N/A	* 20	mRem/hr		
		N/A	+ 5	mRem/hr		
DR	γ	N/A	* 50	mRem/hr		
		N/A	+ 20	mRem/hr		
DR	γ	N/A	10	mRem/hr		
DR	γ	N/A	1	mRem/hr		
DR	γ	N/A	* 2.0	mRem/hr	bottom of valve	
		N/A	+ 1.0	mRem/hr		
DR	γ	N/A	2	mRem/hr		
DR	γ	N/A	0.5	mRem/hr		
DR	γ	N/A	0.5	mRem/hr		
DR	γ	N/A	2	mRem/hr		
1	Smear	N/A	β/γ <1K	DPM/100 cm2	Handrails	
2	Smear	N/A	β/γ <1K	DPM/100 cm2	pipe/valve	
3	Smear	N/A	β/γ <1K	DPM/100 cm2	Lead Sheilding	
4	Smear	N/A	β/γ <1K	DPM/100 cm2	Piping/valve	
5	Smear	N/A	β/γ <1K	DPM/100 cm2	Handrails	
6	Smear	N/A	β/γ <1K	DPM/100 cm2	Piping	
7	Smear	N/A	β/γ <1K	DPM/100 cm2	Handrails	
8	Smear	N/A	β/γ <1K	DPM/100 cm2	Piping	
9	Smear	N/A	β/γ <1K	DPM/100 cm2	Hangers	
10	Smear	N/A	β/γ <1K	DPM/100 cm2	Grating	
1	Wipe	N/A	β/γ ND	CCPM/Masslin	Floor	
2	Wipe	N/A	β/γ ND	CCPM/Masslin	Grating floor	
3	Wipe	N/A	β/γ ND	CCPM/Masslin	Grating floor	
4	Wipe	N/A	β/γ ND	CCPM/Masslin	Grating floor	
5	Wipe	N/A	β/γ ND	CCPM/Masslin	Grating floor	

QA Record

Image File: A04
Page 3 of

Survey #: HNP-M-20200621-4 - Printed On: 7/22/2020 11:21

Harris Nuclear Plant
Radiological Survey

Data Point Details

Survey #: HNP-M-20200621-4

Map: RAB 236 Penetration Mezzanine

#	Type	Inst.	Value	Units	Position	Notes
	Posting		CA		inside shielding/ pipe chase	
	Text		Area inside a Posted RA			

QA Record

Survey #: HNP-M-2020[illegible]621-4 - Printed On: 7/22/2020 11:21

Image File: A
Page 4

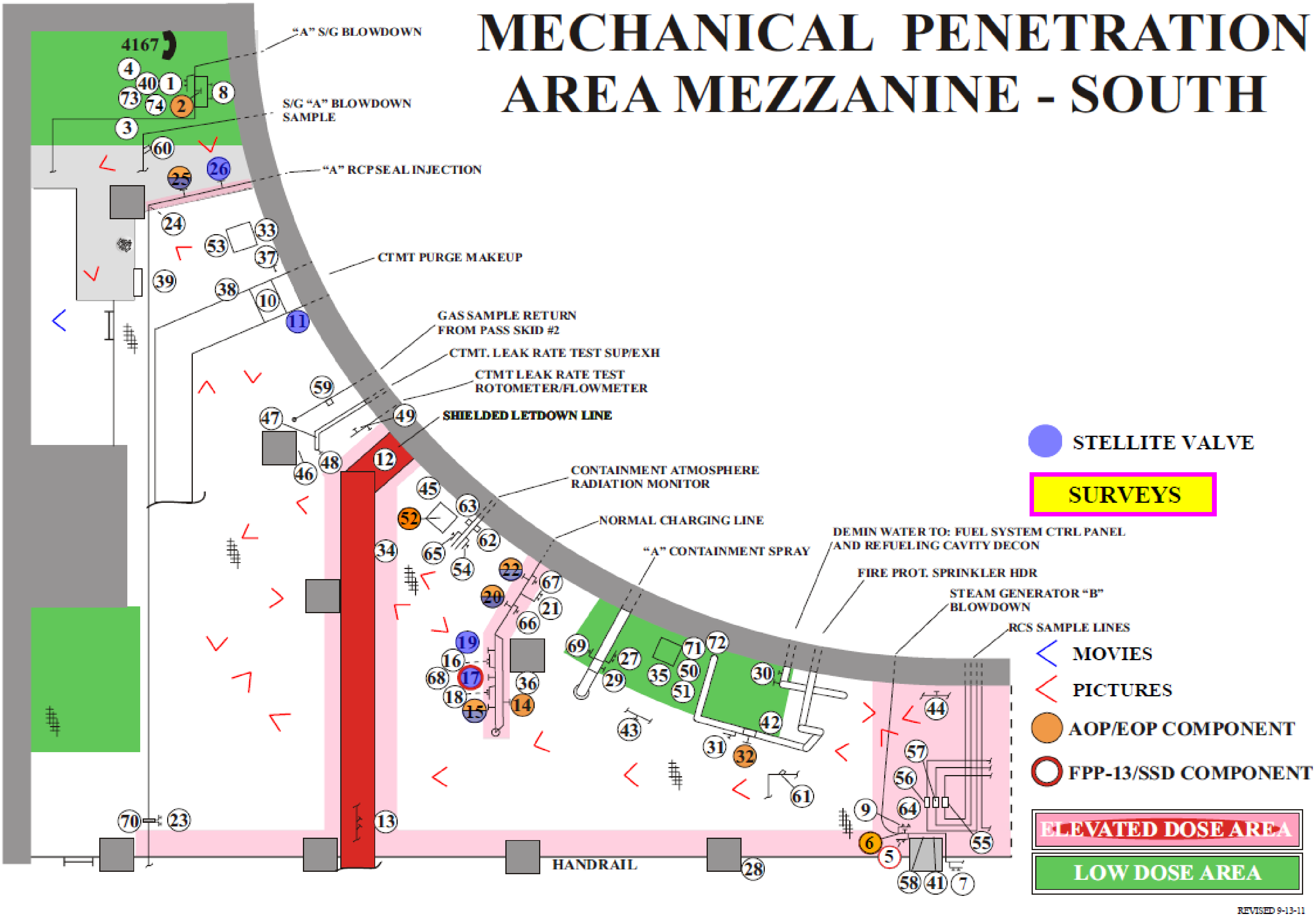
RAB 236' MECHANICAL PENETRATION AREA MEZZANINE - SOUTH

ITEM	DESCRIPTION	ELEV.(ft)	ITEM	DESCRIPTION	ELEV.(ft.)	ITEM	DESCRIPTION	ELEV.(ft.)	ITEM	DESCRIPTION	ELEV.(ft.)
1	1BD-009/10	1'	30	1DW-63/64	2'	61	1SP-0222	7'			
2	1BD-011	2'	31	1FP-0346	3'	62	1SP-0915	6'			
3	1BD-012/151	3'	32	1FP-0347	4'	63	1SP-917	6'			
4	1BD-012-HD1/	1'-5'	33	IFP-2924	9'	64	1SP-1139 to 1142	1'			
	HD2/HI1/HI2/		34	1FP-2925	5'	65	1SP-1184	5'			
	HV1/HV2/HV3		35	1IA-0213/214	3'	66	1VL-15	3'			
5	1BD-028/29	3'	36	1IA-1039-I2	3'	67	1VL-16	3'			
6	1BD-030	3'	37	1IA-1044-I1	1'	68	1VL-17/18/19	2'			
7	1BD-031/153	4'	38	1IA-1044-I2/I3	1'/3'	69	1VL-20	4'			
8	1BD-270/271	3'	39	1IA-1098	10'	70	FT-01CS-0130SW	8'			
9	1BD-272/273	3'	40	1IA-1099/1393-I1	2'/3'	71	PDT-01CB-7680	6'-9'			
10	1CP-6/7	2'	41	1IA-1331/1391-I1	2'/5'		ASA-CV/HI1/				
11	1CP-8	4'	42	1IA-1390-I7	9'		HV1/LI1/LI2/				
12	1CS-011	2'	43	1IA-1390-I8	9'		LV1				
13	1CS-014/15	9'	44	1IA-1392	9'	72	PDT-01CB-7680	9'			
14	1CS-227	6'	45	1IA-1393	8'		A1SA-CV/HI1				
15	1CS-228	2'	46	1IA-1908	4'	73	PI-01BD-8405A	4'			
16	1CS-229/230	0.5'	47	1LT-3	9'		1SA				
17	1CS-231	2'	48	1LT-4	4'	74	PT-01BD-8405A1SA	4'			
18	1CS-232/233	0.5'	49	1LT-5/6	3'						
19	1CS-234	2'	50	1SA-076 to 80	1'-6'						
20	1CS-235	3'	51	1SA-537/538	3'						
21	1CS-236/237	2'	52	1SI-107	2'						
22	1CS-238	3'	53	1SI-359	2'						
23	1CS-336/337	8'	54	1SP-0015	5'						
24	1CS-338/339	5'	55	1SP-0041	2'						
25	1CS-340	6'	56	1SP-0060	2'						
26	1CS-341	6'	57	1SP-0085	2'					Revised 9-13-11	
27	1CT-45/46	1'	58	1SP-0086/87	1'				🟦	STELLITE VALVE	
28	1CT-48-HV2/LV2	4'	59	1SP-0208	3'				🟧	AOP/EOP COMPONENT	
29	1CT-50	4'	60	1SP-0217	6'				🟥	FPP-13/SSD COMPONENT	

RAB 236’

MAP 9

MECHANICAL PENETRATION AREA MEZZANINE - SOUTH



Facility:	Harris Nuclear Plant	Task No.:	018003H101
Task Title:	Determine AFD with AFD Monitor INOP and Evaluate Tech Specs	JPM No.:	2020 NRC Exam Admin JPM SRO A1-1

K/A Reference: G 2.1.25 RO 3.9 SRO 4.2 **ALTERNATE PATH - NO**

Examinee: ______________________ NRC Examiner: ______________

Facility Evaluator: ______________________ Date: ______________

Method of testing:

Simulated Performance: _____ Actual Performance: X

Classroom X Simulator _____ Plant _____

READ TO THE EXAMINEE

I will explain the initial conditions, which steps to simulate, discuss, or perform and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.

Initial Conditions:	- The plant was at 90% power, with a load reduction in progress - The load reduction has been stopped to evaluate AFD following oscillations

Initiating Cue:	With the information provided complete Attachment 5 of OST-1021, Daily Surveillance Requirements to determine Axial Flux Difference. After performing the calculation evaluate the results and circle the response below. List the Technical Specifications and the associated LCO action(s) that apply. When complete return your results to the evaluator.

Task Standard:	All calculations within ± 2% of actual. Correct Tech Spec and LCO action is identified.
Required Materials:	Calculator
General References:	OST-1021, Daily Surveillance Requirements, Rev. 114 OP-163, ERFIS, Rev. 42 Rod Control Manual, Unit One Reactor Operating Data, Rev. 8 Technical Specifications, Rev 185
Handouts:	OP-163, Rev. 42, pages 1 – 8, Prerequisites, P&L's OP-163, Rev. 42, pages 14 – 15, Section 6.2, (Continuous Use) - Axial Flux Differential (AFD) Monitor Rod Control Manual, Section 2.1, Axial Flux Difference Limits, Rev. 0 Technical Specification 3.2.1, Power Distribution Limits - Axial Flux Difference OR 2020 NRC Exam Frozen Procedures Folder OST-1021, Rev. 114, pages 44-46, Attachment 5, Axial Flux Difference Log JPM Cue Sheets Pages 16 - 20
Time Critical Task:	No
Validation Time:	10 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 7	If the wrong values are selected then the results will NOT be correct
Step 10	If the wrong Limit is determined a required Tech Spec LCO action could be exceeded
Step 11	If the wrong Limit is determined a required Tech Spec LCO action could be exceeded
Step 12	If operation outside of the acceptable region is allowed to continue fuel damage may result.
Step 13	If the wrong Tech Spec Action is selected an LCO action could be exceeded

Start Time: __________.

OP-163

Performance Step: 1 OBTAIN PROCEDURE (provided in frozen procedure)

Standard: Obtains OP-163 and refers to Section 6.2.

Comment:

OP-163, Section 6.2.2, Step 1.a

Performance Step: 2 REVIEW the automatic or "On Demand" report print-out to verify the following:

- The print-out monitored values are consistent with MCB indications.

Standard: Locates JPM Cue sheet with attached On Demand and Shift Summary Report

Comment:

OP-163, Section 6.2.2, NOTE prior to Step 1.b

Performance Step: 3 NOTE: There may be rounding off differences between the automatic printout and the latest AFD curve generated by TE-NF-PWR-0809, Target AFD Calculation.

Standard: Operator reads and placekeeps notes

Comment:

OP-163, Section 6.2.2, Step 1.b

Performance Step: 4 REVIEW the automatic or “On Demand” report print-out to verify the following:

- The printout Operating Band Low and Operating Band High values match the latest Axial Flux Difference Limits As A Function of Rated Thermal Power curve as shown in the Rod Manual.

Standard: Locates Rod Manual and reviews Section 2.1, AFD Limits and determines the current limits are

-12.0% to + 8.0% at 100% Reactor Power

-26.0% to + 20.0% at 50% Reactor Power

Comment:

OP-163, Section 6.2.2, Step 2

Performance Step: 5 CHANNEL CHECK the following AFD ERFIS points against MCB indication:

- URE1540 CURRENT CHAN 1 AXIAL FLUX DIFF
- URE1541 CURRENT CHAN 2 AXIAL FLUX DIFF
- URE1542 CURRENT CHAN 3 AXIAL FLUX DIFF
- URE1543 CURRENT CHAN 4 AXIAL FLUX DIFF

Standard: Locates JPM Cue sheet with attached MCB Indication images and compares to information from Shift Summary Report

Comment:

OP-163, Section 6.2.2, NOTE prior to Step 3

Performance Step: 6 NOTE: Only one (1) channel having an unacceptable quality does not make the AFD Monitor inoperable.

Standard: Operator reads and placekeeps notes

Comment:

OP-163, Section 6.2.2, Step 3

✓ **Performance Step: 7** VERIFY the following AFD ERFIS points are restored to processing with acceptable quality codes as defined in Precaution & Limitation Step 4.0.4:

- URE1540 CURRENT CHAN 1 AXIAL FLUX DIFF
- URE1541 CURRENT CHAN 2 AXIAL FLUX DIFF
- URE1542 CURRENT CHAN 3 AXIAL FLUX DIFF
- URE1543 CURRENT CHAN 4 AXIAL FLUX DIFF
- ANM0120M PWR RNG CHANNEL N41 Q4 1-MIN AVG
- ANM0121M PWR RNG CHANNEL N42 Q2 1-MIN AVG
- ANM0122M PWR RNG CHANNEL N43 Q1 1-MIN AVG
- ANM0123M PWR RNG CHANNEL N44 Q3 1-MIN AVG

Standard: Reviews P&L # 4 determines the quality codes are **NOT** acceptable for

- **URE1541 CURRENT CHAN 2 AXIAL FLUX DIFF**
- **URE1542 CURRENT CHAN 3 AXIAL FLUX DIFF**
- **ANM0121M PWR RNG CHANNEL N42 Q2 1-MIN AVG**
- **ANM0122M PWR RNG CHANNEL N43 Q1 1-MIN AVG**

Notifies the CRS the AFD Monitor does met the criteria for Operable status

Evaluator Cue:	**If necessary prompt the candidate to completed OST-1021, Attachment 5 as required.**

Comment:

OST-1021, Attachment 5, Page 2 of 3

Performance Step: 8 LOG current reading for the following instruments:

- NI-41C, PR 41 % Δ FLUX
- NI-42C, PR 42 % Δ FLUX
- NI-43C, PR 43 % Δ FLUX
- NI-44C, PR 44 % Δ FLUX

Standard: Locates JPM Cue sheet with attached MCB Indication images and logs current reading

- **NI-41C, PR 41 % Δ FLUX = 11% +/- 2%**
- **NI-42C, PR 42 % Δ FLUX = 13% +/- 2%**
- **NI-43C, PR 43 % Δ FLUX = 14% +/- 2%**
- **NI-44C, PR 44 % Δ FLUX = 10% +/- 2%**

Comment:

OST-1021, Attachment 5, Page 2 of 3

Performance Step: 9 DETERMINE and LOG Average (AVG) Reactor Power:

- NI-41B, PR 41 % POWER
- NI-42B, PR 42 % POWER
- NI-43B, PR 43 % POWER
- NI-44B, PR 44 % POWER

Standard: Locates JPM Cue sheet with attached MCB Indication images and logs current reading

- **NI-41B, PR 41 % POWER = 90% +/- 2%**
- **NI-42B, PR 42 % POWER = 90% +/- 2%**
- **NI-43B, PR 43 % POWER = 90% +/- 2%**
- **NI-44B, PR 44 % POWER = 90% +/- 2%**

Comment: **Performs calculation to determine AVG Reactor Power and logs value on OST-1021 Attachment 5**

OST-1021, Attachment 5, Page 2 of 3

✓ **Performance Step: 10** DETERMINE and LOG AFD Lower limit:

Standard: **Critical action is to determine required limit.**

Locates Rod Manual and reviews Section 2.1, AFD Limits and determines the current Lower limits is:

-14.5% at 90% Reactor Power (+/- 2%)

Comment: **Must interpolate limit based on current power level**

OST-1021, Attachment 5, Page 2 of 3

✓ **Performance Step: 11** DETERMINE and LOG AFD Upper limit:

Standard: **Critical action is to determine required limit.**

Locates Rod Manual and reviews Section 2.1, AFD Limits and determines the current Upper limits is:

11.0% at 90% Reactor Power (+/- 2%)

Comment: **Must interpolate limit based on current power level**

OST-1021, Attachment 5, Page 2 of 3

✓ **Performance Step: 12** PERFORM evaluation of AFD limits

Standard: Reviews current MCB readings and determines AFD Limits and determines two of four MCB indications are NOT within the curve for Acceptable Operation:

- **NI-42C, PR 42 % Δ FLUX = 13% +/- 2%**
- **NI-43C, PR 43 % Δ FLUX = 14% +/- 2%**

Notifies the CRS two of four MCB indications are NOT within the AFD curve for Acceptable Operation

Comment: **Must interpolate limit based on current power level**

Technical Specifications

✓ **Performance Step: 13** OBTAIN AND EVALUATE TECHNICAL SPECIFICATIONS

Standard: Obtains Technical Specifications and refers to LCO 3.2.1

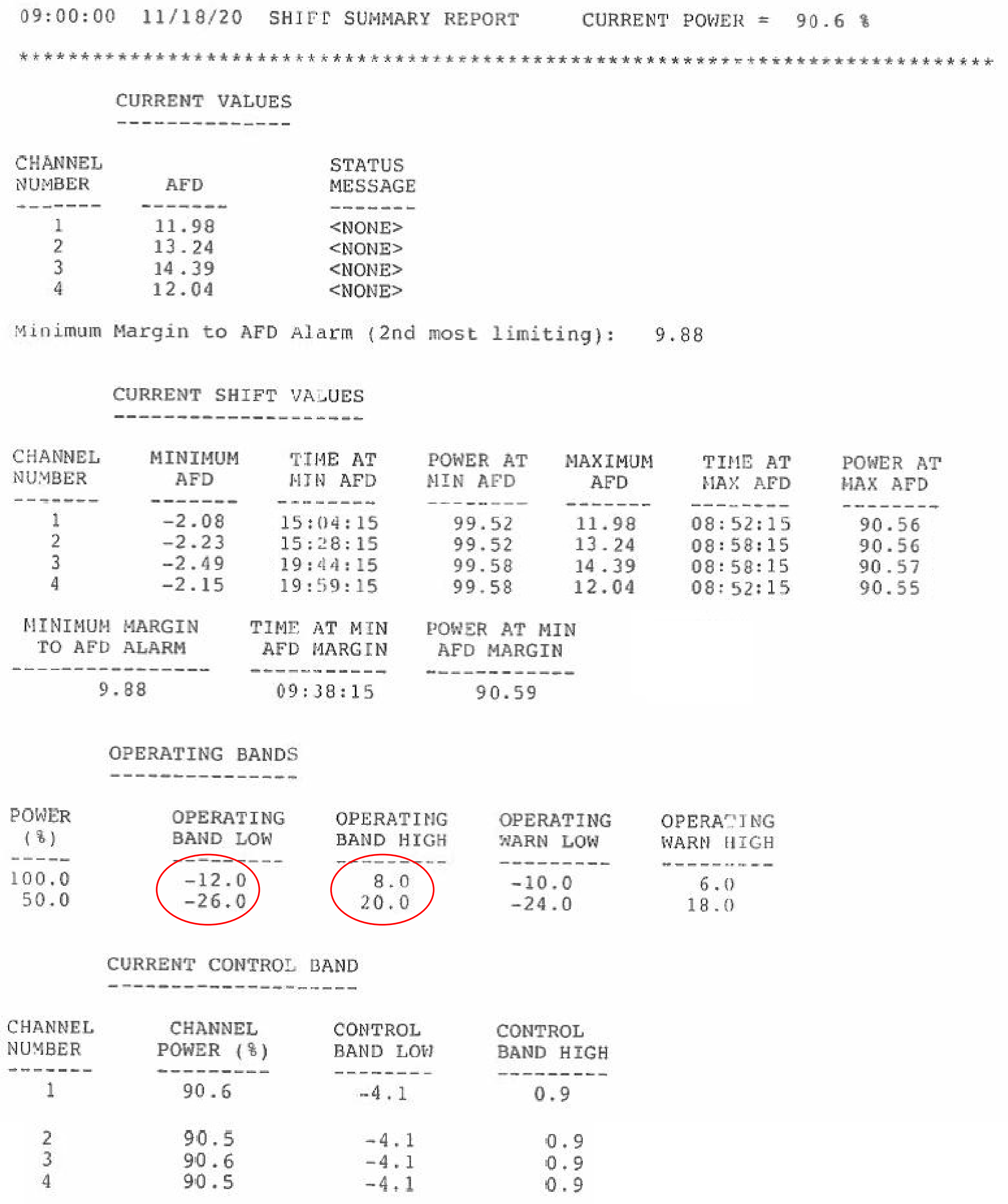
Determines that ACTION a. is applicable. (See page 14)

Evaluator Note:	**After the candidate has determined the current values of Axial Flux Difference and its limits have been manually determined and performed a Technical Specification evaluation.** **END OF JPM**

Terminating Cue:	Current value of Axial Flux Difference has been manually determined and the Technical Specifications evaluation completed.

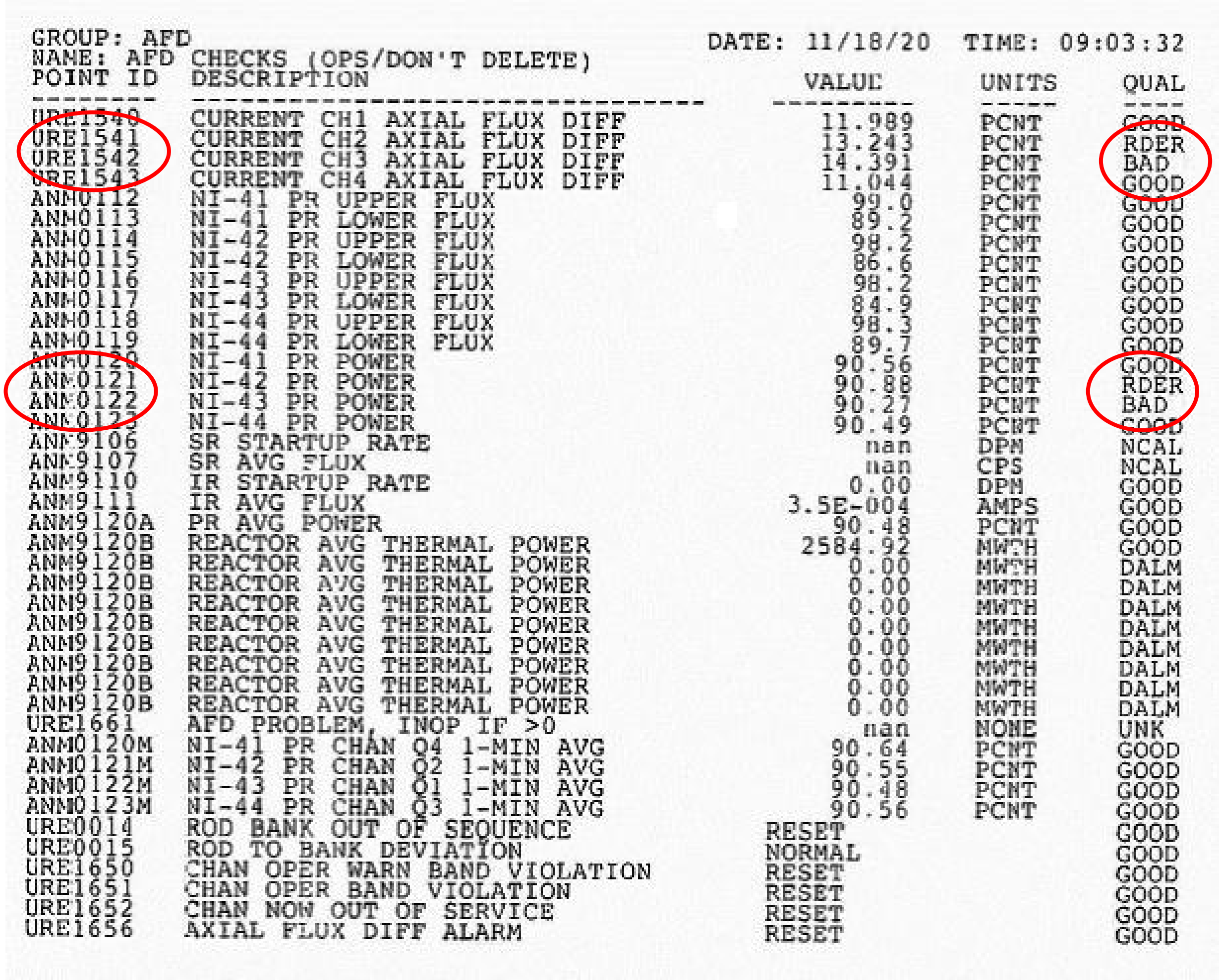
Stop Time: _________

KEY



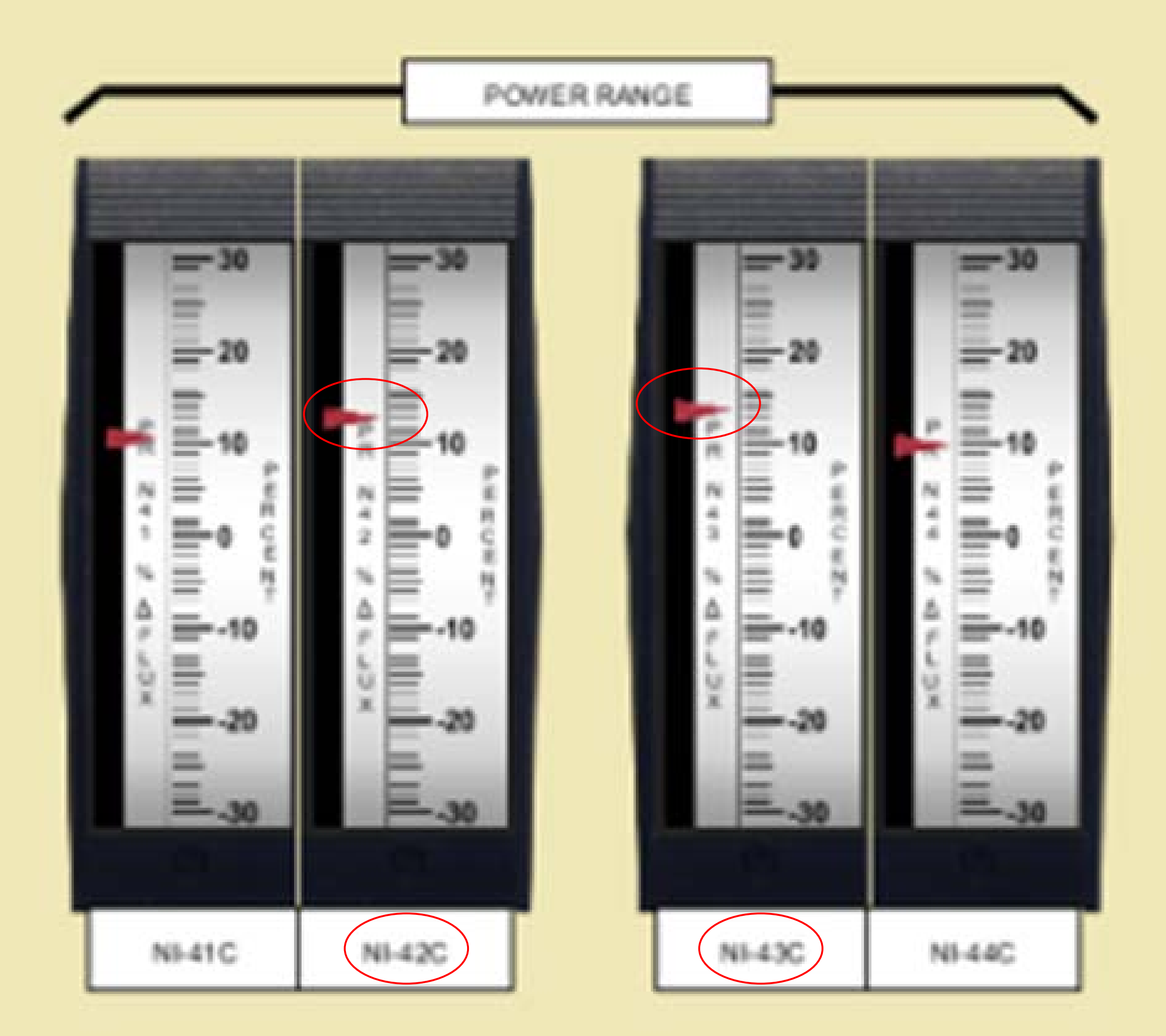
```
09:00:00  11/18/20  SHIFT SUMMARY REPORT     CURRENT POWER =  90.6 %

*******************************************************************************

          CURRENT VALUES
          --------------

CHANNEL                          STATUS
NUMBER       AFD                 MESSAGE
-------    -------               -------
   1        11.98                <NONE>
   2        13.24                <NONE>
   3        14.39                <NONE>
   4        12.04                <NONE>

Minimum Margin to AFD Alarm (2nd most limiting):   9.88

          CURRENT SHIFT VALUES
          --------------------

CHANNEL    MINIMUM    TIME AT    POWER AT   MAXIMUM    TIME AT    POWER AT
NUMBER       AFD      MIN AFD    MIN AFD      AFD      MAX AFD    MAX AFD
-------    -------    --------   --------   -------    --------   --------
   1        -2.08     15:04:15    99.52      11.98     08:52:15    90.56
   2        -2.23     15:28:15    99.52      13.24     08:58:15    90.56
   3        -2.49     19:44:15    99.58      14.39     08:58:15    90.57
   4        -2.15     19:59:15    99.58      12.04     08:52:15    90.55

 MINIMUM MARGIN     TIME AT MIN    POWER AT MIN
  TO AFD ALARM       AFD MARGIN     AFD MARGIN
----------------   -----------    ------------
      9.88            09:38:15        90.59

          OPERATING BANDS
          ---------------

POWER         OPERATING      OPERATING      OPERATING      OPERATING
 (%)          BAND LOW       BAND HIGH      WARN LOW       WARN HIGH
-----         ---------      ---------      ---------      ---------
100.0           -12.0           8.0           -10.0           6.0
 50.0           -26.0          20.0           -24.0          18.0

          CURRENT CONTROL BAND
          --------------------

CHANNEL       CHANNEL        CONTROL        CONTROL
NUMBER       POWER (%)       BAND LOW       BAND HIGH
-------      ---------       --------       ---------
  1            90.6            -4.1            0.9

  2            90.5            -4.1            0.9
  3            90.6            -4.1            0.9
  4            90.5            -4.1            0.9
```

KEY

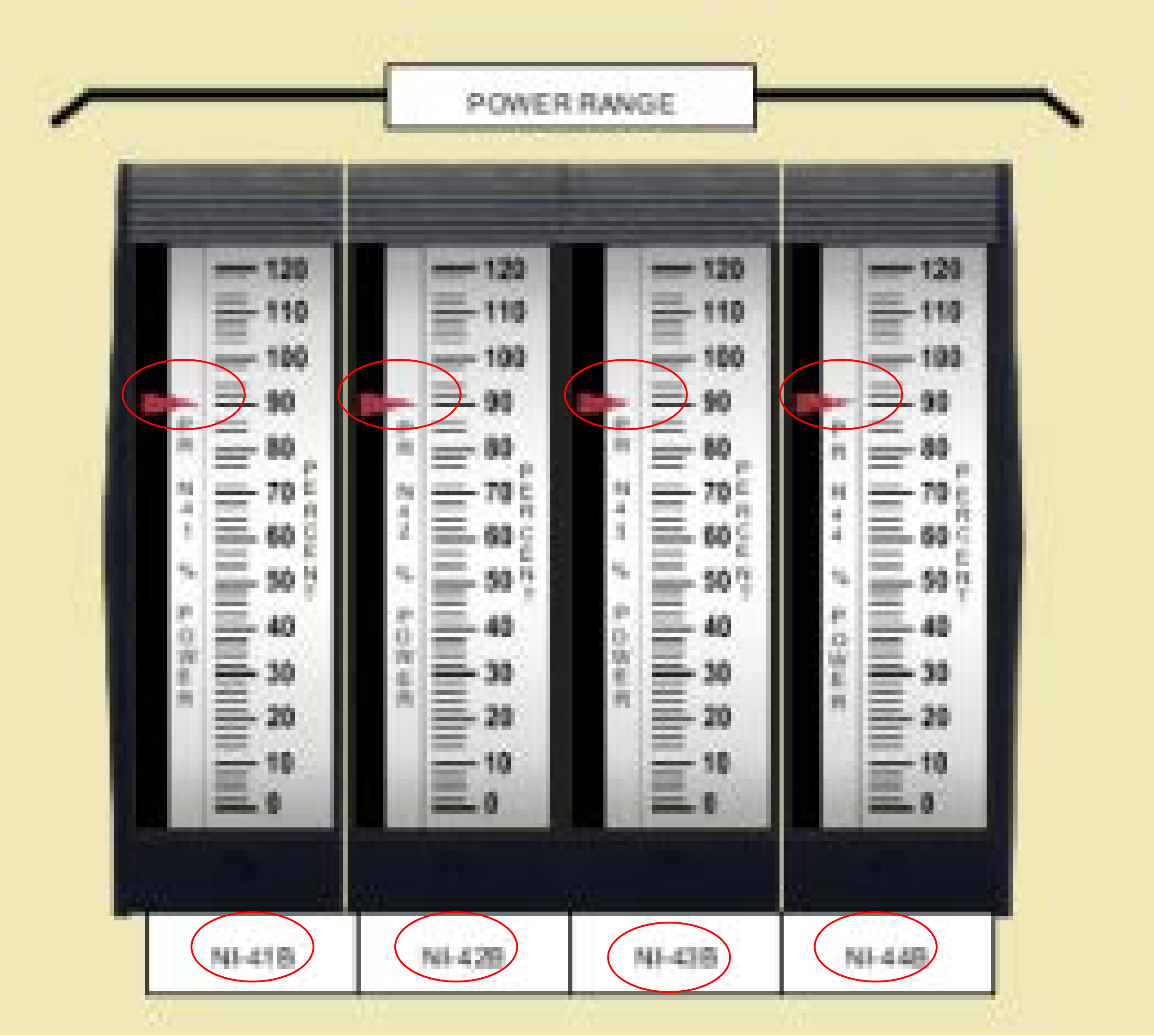


```
GROUP: AFD                                          DATE: 11/18/20   TIME: 09:03:32
NAME: AFD CHECKS (OPS/DON'T DELETE)
POINT ID  DESCRIPTION                                  VALUE      UNITS    QUAL
--------  --------------------------------          ----------    -----    ----
URE1540   CURRENT CH1 AXIAL FLUX DIFF                   11.989    PCNT     GOOD
URE1541   CURRENT CH2 AXIAL FLUX DIFF                   13.243    PCNT     RDER
URE1542   CURRENT CH3 AXIAL FLUX DIFF                   14.391    PCNT     BAD
URE1543   CURRENT CH4 AXIAL FLUX DIFF                   11.044    PCNT     GOOD
ANM0112   NI-41 PR UPPER FLUX                             99.0    PCNT     GOOD
ANM0113   NI-41 PR LOWER FLUX                             89.2    PCNT     GOOD
ANM0114   NI-42 PR UPPER FLUX                             98.2    PCNT     GOOD
ANM0115   NI-42 PR LOWER FLUX                             86.6    PCNT     GOOD
ANM0116   NI-43 PR UPPER FLUX                             98.2    PCNT     GOOD
ANM0117   NI-43 PR LOWER FLUX                             84.9    PCNT     GOOD
ANM0118   NI-44 PR UPPER FLUX                             98.3    PCNT     GOOD
ANM0119   NI-44 PR LOWER FLUX                             89.7    PCNT     GOOD
ANM0120   NI-41 PR POWER                                 90.56    PCNT     GOOD
ANM0121   NI-42 PR POWER                                 90.88    PCNT     RDER
ANM0122   NI-43 PR POWER                                 90.27    PCNT     BAD
ANM0123   NI-44 PR POWER                                 90.49    PCNT     GOOD
ANM9106   SR STARTUP RATE                                  nan    DPM      NCAL
ANM9107   SR AVG FLUX                                      nan    CPS      NCAL
ANM9110   IR STARTUP RATE                                 0.00    DPM      GOOD
ANM9111   IR AVG FLUX                                 3.5E-004    AMPS     GOOD
ANM9120A  PR AVG POWER                                   90.48    PCNT     GOOD
ANM9120B  REACTOR AVG THERMAL POWER                    2584.92    MWTH     GOOD
ANM9120B  REACTOR AVG THERMAL POWER                       0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                       0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                       0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                       0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                       0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                       0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                       0.00    MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                       0.00    MWTH     DALM
URE1661   AFD PROBLEM, INOP IF >0                          nan    NONE     UNK
ANM0120M  NI-41 PR CHAN Q4 1-MIN AVG                     90.64    PCNT     GOOD
ANM0121M  NI-42 PR CHAN Q2 1-MIN AVG                     90.55    PCNT     GOOD
ANM0122M  NI-43 PR CHAN Q1 1-MIN AVG                     90.48    PCNT     GOOD
ANM0123M  NI-44 PR CHAN Q3 1-MIN AVG                     90.56    PCNT     GOOD
URE0014   ROD BANK OUT OF SEQUENCE                  RESET                  GOOD
URE0015   ROD TO BANK DEVIATION                     NORMAL                 GOOD
URE1650   CHAN OPER WARN BAND VIOLATION             RESET                  GOOD
URE1651   CHAN OPER BAND VIOLATION                  RESET                  GOOD
URE1652   CHAN NOW OUT OF SERVICE                   RESET                  GOOD
URE1656   AXIAL FLUX DIFF ALARM                     RESET                  GOOD
```

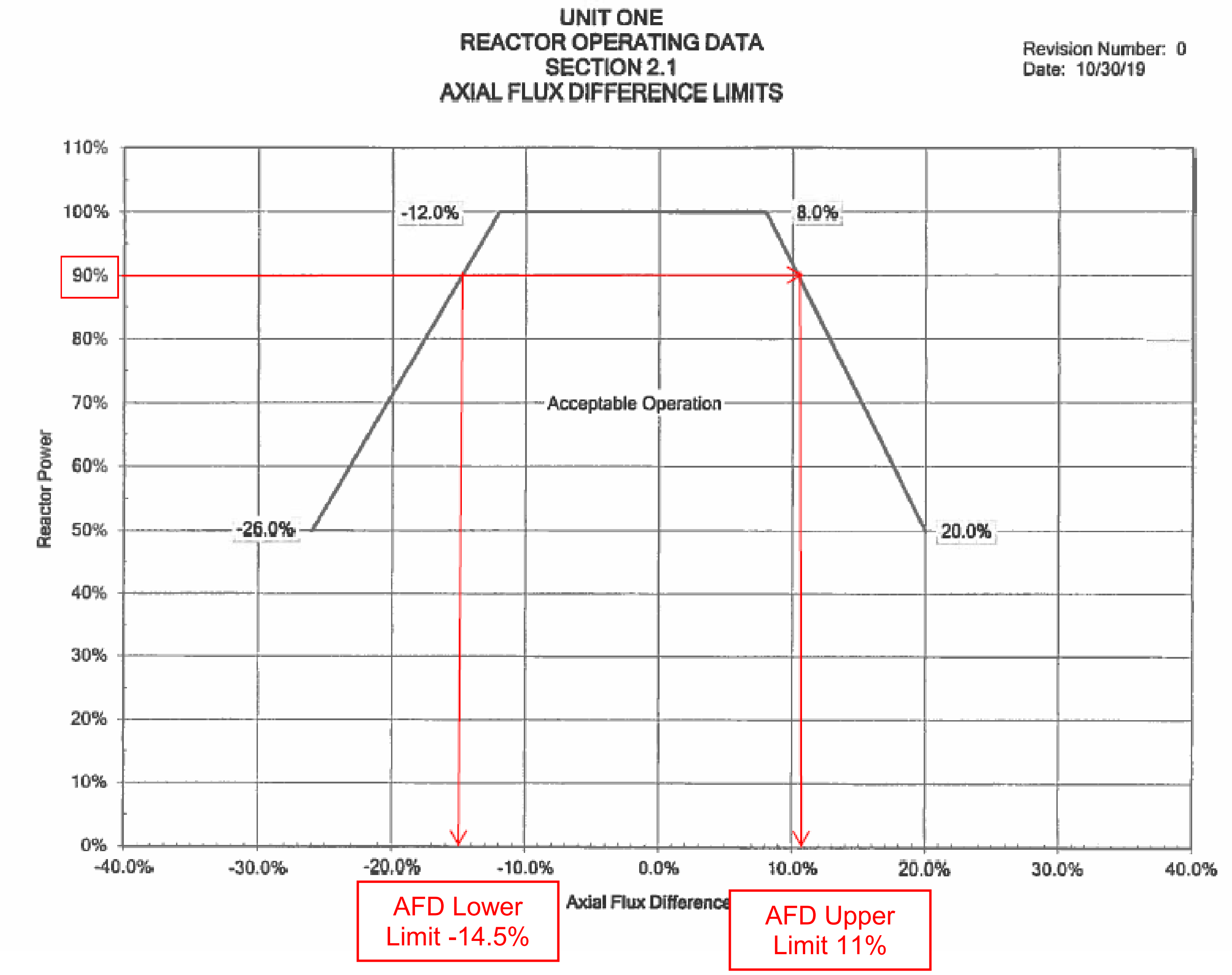
KEY


POWER RANGE
30
20
10
0
-10
-20
-30
PR N41 % Δ FLUX
PR N42 % Δ FLUX
PR N43 % Δ FLUX
PR N44 % Δ FLUX
PERCENT
NI-41C
NI-42C
NI-43C
NI-44C

KEY


POWER RANGE
120
110
100
90
80
70
60
50
40
30
20
10
0
PERCENT
PR N41 % POWER
PR N42 % POWER
PR N43 % POWER
PR N44 % POWER
NI-41B
NI-42B
NI-43B
NI-44B

KEY


UNIT ONE
REACTOR OPERATING DATA
SECTION 2.1
AXIAL FLUX DIFFERENCE LIMITS
Revision Number: 0
Date: 10/30/19
110%
100%
90%
80%
70%
60%
50%
40%
30%
20%
10%
0%
Reactor Power
-12.0%
8.0%
-26.0%
20.0%
Acceptable Operation
-40.0%
-30.0%
-20.0%
-10.0%
0.0%
10.0%
20.0%
30.0%
40.0%
Axial Flux Difference
AFD Lower Limit -14.5%
AFD Upper Limit 11%

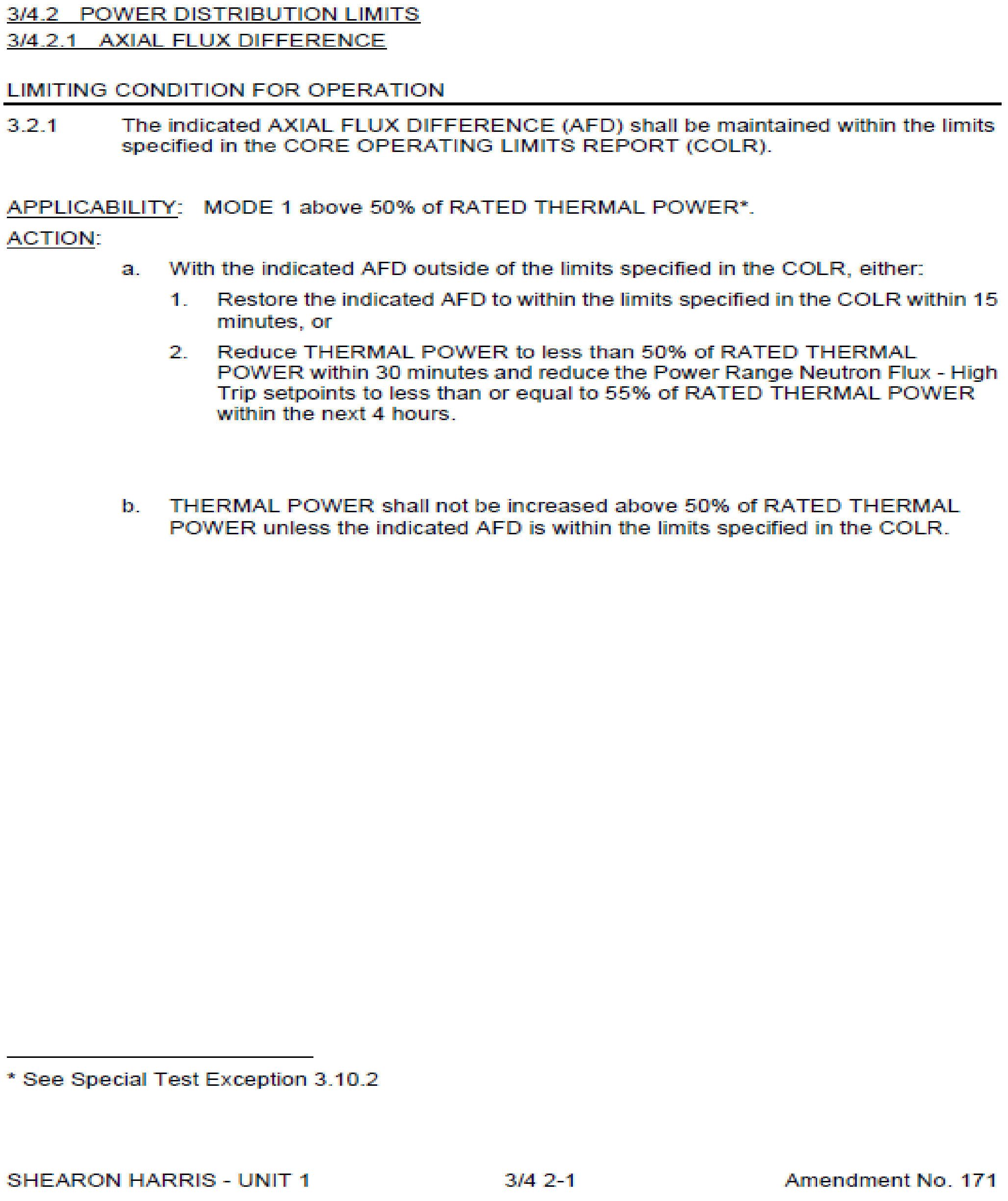
KEY

1. The current AFD Limits are **Upper AFD limit 11.0% at 90% Reactor Power (+/- 2%)**
Lower AFD limit 14.5% at 90% Reactor Power (+/- 2%)

Circle the correct response that applies:

2. AFD Monitor Alarm is Operable / [✓ Inoperable]

3. Technical Specification(s) and applicable LCO's that apply



3/4.2 POWER DISTRIBUTION LIMITS
3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the limits specified in the COLR, either:
   1. Restore the indicated AFD to within the limits specified in the COLR within 15 minutes, or
   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

---

* See Special Test Exception 3.10.2

SHEARON HARRIS - UNIT 1 3/4 2-1 Amendment No. 171

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Admin Exam SRO A1-1

Determine Axial Flux Difference (AFD) with AFD Monitor INOP and Evaluate Technical Specifications

OP-163, ERFIS

OST-1021, Daily Surveillance Requirements

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	- The plant was at 90% power, with a load reduction in progress - The load reduction has been stopped to evaluate AFD following oscillations

Initiating Cue:	With the information provided complete Attachment 5 of OST-1021, Daily Surveillance Requirements to determine Axial Flux Difference. After performing the calculation evaluate the results and circle the response below. List the Technical Specifications and the associated LCO action(s), IF any, that apply. When complete return your results to the evaluator.

Name: ____

Date: ___________________

1. The current AFD Limits are ___________________

Circle the correct response that applies:

2. AFD Monitor Alarm is Operable / Inoperable

Technical Specification(s) and applicable LCO's that apply: _______________________

____

____

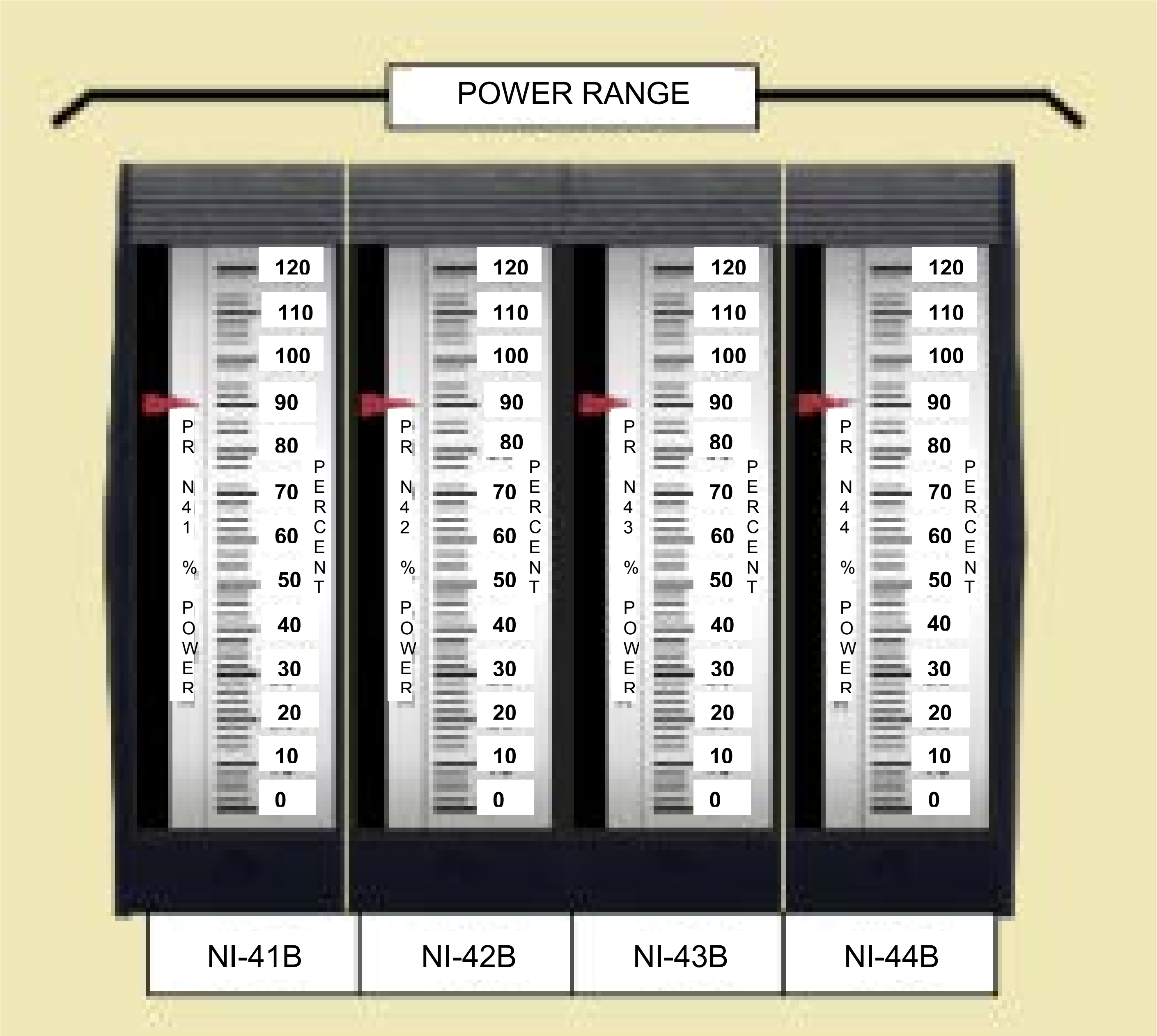
____

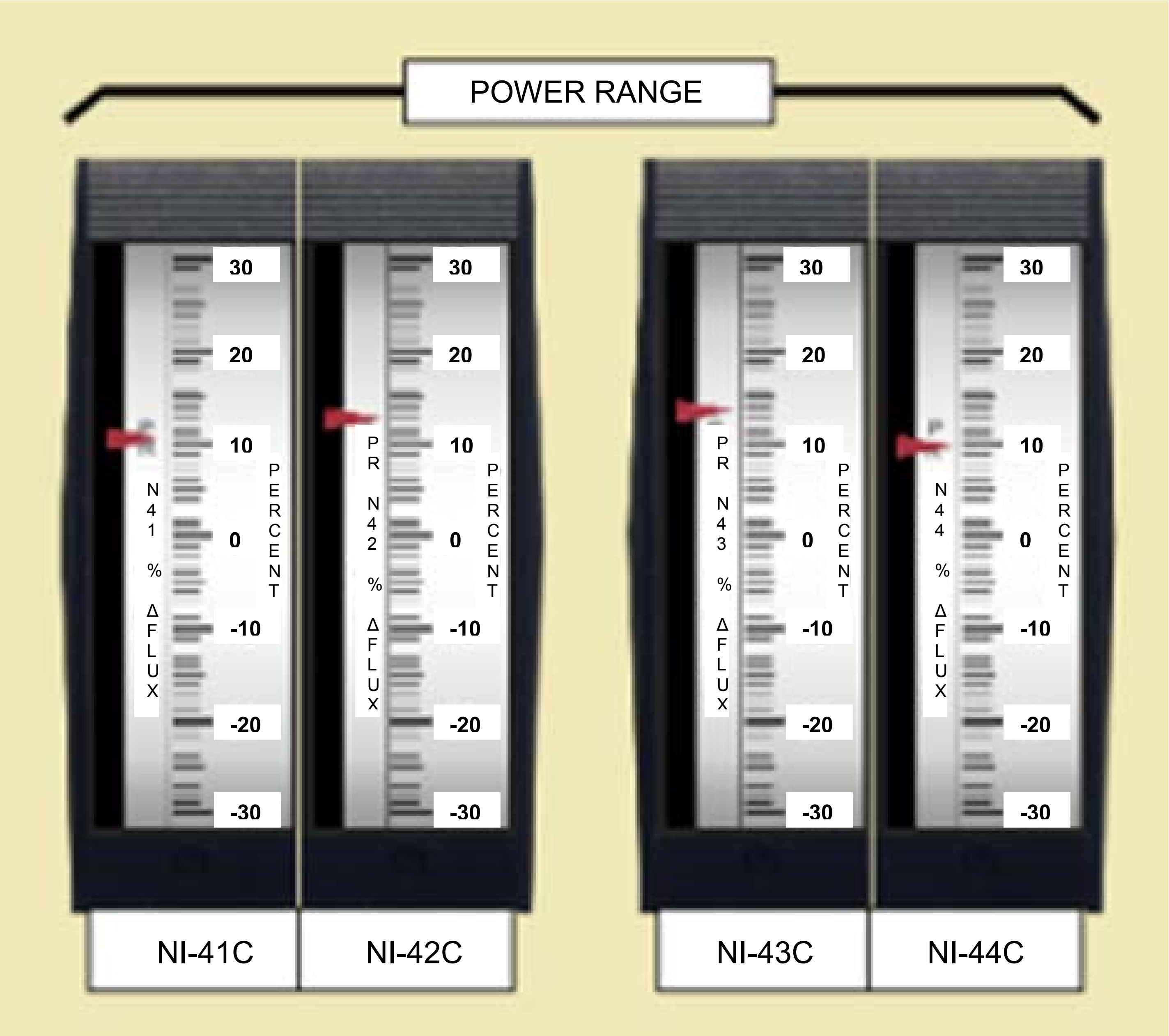
____

____

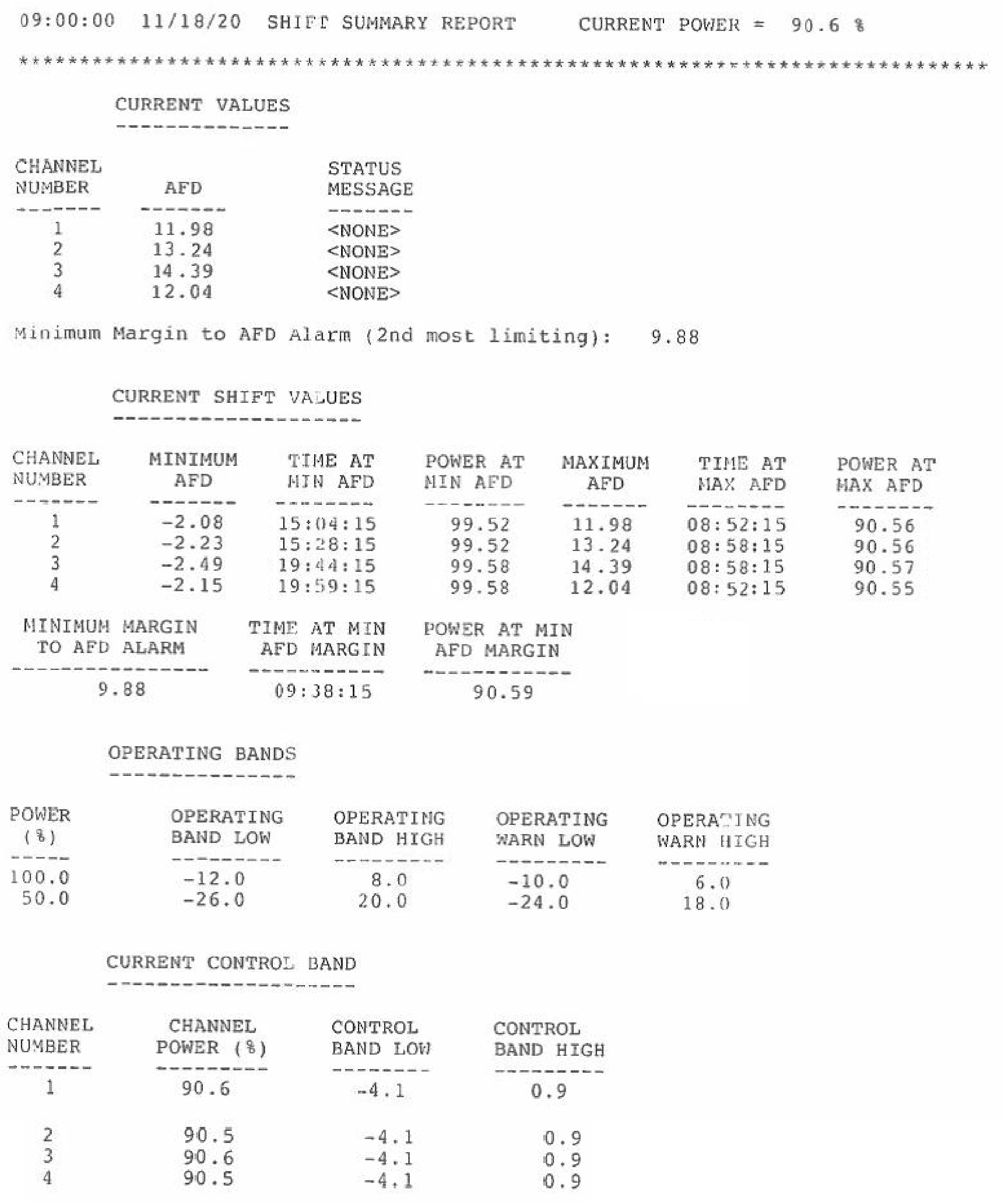
____

____

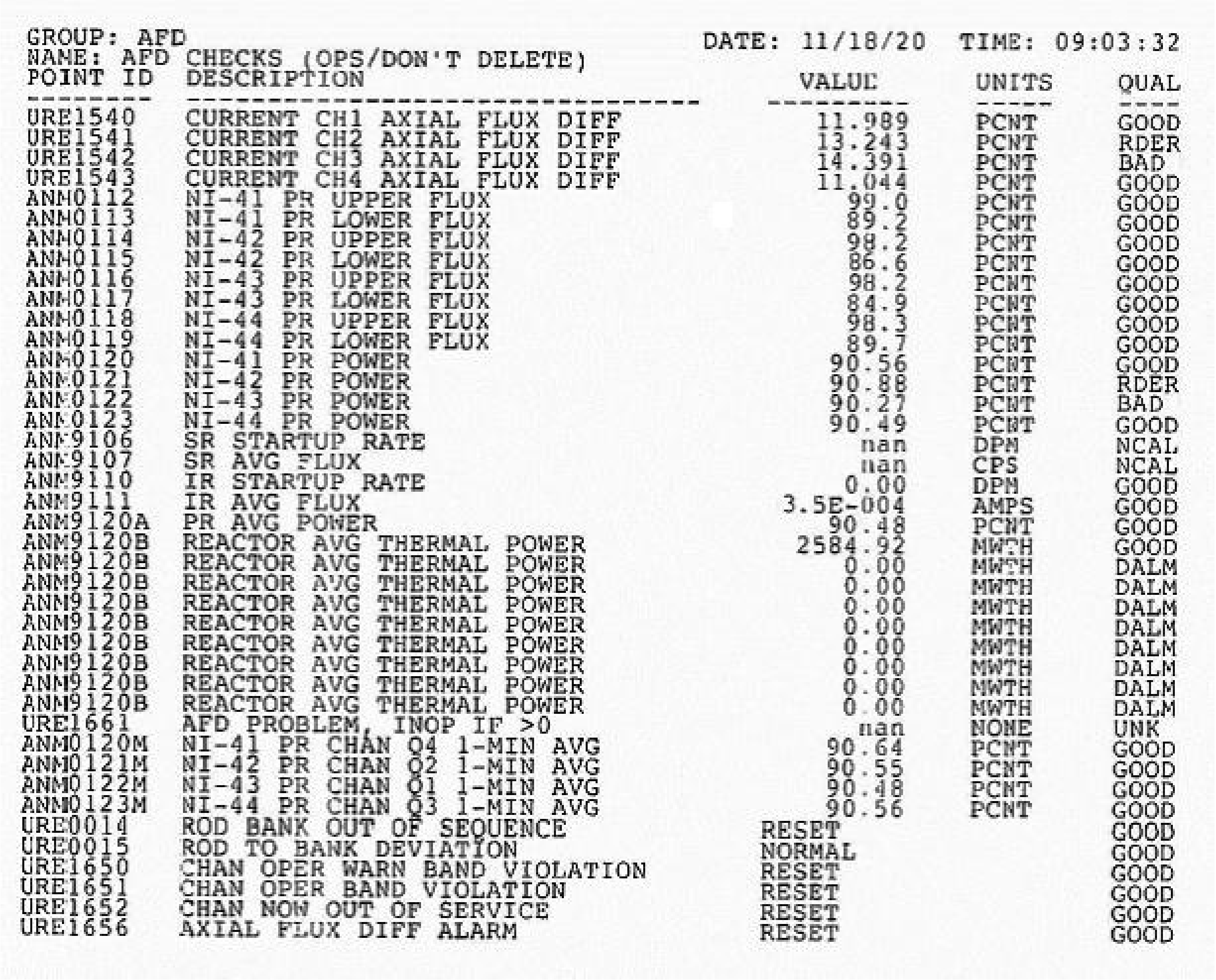

POWER RANGE
120
110
100
90
80
70
60
50
40
30
20
10
0
PR N41 % POWER
PR N42 % POWER
PR N43 % POWER
PR N44 % POWER
PERCENT
NI-41B
NI-42B
NI-43B
NI-44B


POWER RANGE
30
20
10
0
-10
-20
-30
N41 % Δ FLUX
PR N42 % Δ FLUX
PR N43 % Δ FLUX
N44 % Δ FLUX
PERCENT
NI-41C
NI-42C
NI-43C
NI-44C

JPM CUE SHEET



```
09:00:00  11/18/20  SHIFT SUMMARY REPORT    CURRENT POWER =  90.6 %
*******************************************************************************

        CURRENT VALUES
        --------------

CHANNEL                       STATUS
NUMBER      AFD               MESSAGE
-------   -------             -------
   1       11.98              <NONE>
   2       13.24              <NONE>
   3       14.39              <NONE>
   4       12.04              <NONE>

Minimum Margin to AFD Alarm (2nd most limiting):   9.88

        CURRENT SHIFT VALUES
        --------------------

CHANNEL   MINIMUM    TIME AT    POWER AT   MAXIMUM    TIME AT    POWER AT
NUMBER      AFD      MIN AFD    MIN AFD      AFD      MAX AFD    MAX AFD
-------   -------   --------   --------   -------   --------   --------
   1       -2.08    15:04:15     99.52     11.98    08:52:15     90.56
   2       -2.23    15:28:15     99.52     13.24    08:58:15     90.56
   3       -2.49    19:44:15     99.58     14.39    08:58:15     90.57
   4       -2.15    19:59:15     99.58     12.04    08:52:15     90.55

 MINIMUM MARGIN     TIME AT MIN   POWER AT MIN
  TO AFD ALARM      AFD MARGIN     AFD MARGIN
----------------    -----------   ------------
      9.88           09:38:15        90.59

        OPERATING BANDS
        ---------------

POWER        OPERATING     OPERATING     OPERATING     OPERATING
 (%)         BAND LOW      BAND HIGH     WARN LOW      WARN HIGH
-----        ---------     ---------     ---------     ---------
100.0         -12.0           8.0         -10.0           6.0
 50.0         -26.0          20.0         -24.0          18.0

        CURRENT CONTROL BAND
        --------------------

CHANNEL      CHANNEL      CONTROL       CONTROL
NUMBER      POWER (%)     BAND LOW      BAND HIGH
-------     ---------     --------      ---------
   1          90.6          -4.1           0.9

   2          90.5          -4.1           0.9
   3          90.6          -4.1           0.9
   4          90.5          -4.1           0.9
```



```
GROUP: AFD                                              DATE: 11/18/20  TIME: 09:03:32
NAME: AFD CHECKS (OPS/DON'T DELETE)
POINT ID  DESCRIPTION                                 VALUE      UNITS    QUAL
--------  --------------------------------        ---------      -----    ----
URE1540   CURRENT CH1 AXIAL FLUX DIFF                11.989      PCNT     GOOD
URE1541   CURRENT CH2 AXIAL FLUX DIFF                13.243      PCNT     RDER
URE1542   CURRENT CH3 AXIAL FLUX DIFF                14.391      PCNT     BAD
URE1543   CURRENT CH4 AXIAL FLUX DIFF                11.044      PCNT     GOOD
ANM0112   NI-41 PR UPPER FLUX                          99.0      PCNT     GOOD
ANM0113   NI-41 PR LOWER FLUX                          89.2      PCNT     GOOD
ANM0114   NI-42 PR UPPER FLUX                          98.2      PCNT     GOOD
ANM0115   NI-42 PR LOWER FLUX                          86.6      PCNT     GOOD
ANM0116   NI-43 PR UPPER FLUX                          98.2      PCNT     GOOD
ANM0117   NI-43 PR LOWER FLUX                          84.9      PCNT     GOOD
ANM0118   NI-44 PR UPPER FLUX                          98.3      PCNT     GOOD
ANM0119   NI-44 PR LOWER FLUX                          89.7      PCNT     GOOD
ANM0120   NI-41 PR POWER                              90.56      PCNT     GOOD
ANM0121   NI-42 PR POWER                              90.88      PCNT     RDER
ANM0122   NI-43 PR POWER                              90.27      PCNT     BAD
ANM0123   NI-44 PR POWER                              90.49      PCNT     GOOD
ANM9106   SR STARTUP RATE                               nan      DPM      NCAL
ANM9107   SR AVG FLUX                                   nan      CPS      NCAL
ANM9110   IR STARTUP RATE                              0.00      DPM      GOOD
ANM9111   IR AVG FLUX                             3.5E-004      AMPS     GOOD
ANM9120A  PR AVG POWER                                90.48      PCNT     GOOD
ANM9120B  REACTOR AVG THERMAL POWER                 2584.92      MWTH     GOOD
ANM9120B  REACTOR AVG THERMAL POWER                    0.00      MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                    0.00      MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                    0.00      MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                    0.00      MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                    0.00      MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                    0.00      MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                    0.00      MWTH     DALM
ANM9120B  REACTOR AVG THERMAL POWER                    0.00      MWTH     DALM
URE1661   AFD PROBLEM, INOP IF >0                       nan      NONE     UNK
ANM0120M  NI-41 PR CHAN Q4 1-MIN AVG                  90.64      PCNT     GOOD
ANM0121M  NI-42 PR CHAN Q2 1-MIN AVG                  90.55      PCNT     GOOD
ANM0122M  NI-43 PR CHAN Q1 1-MIN AVG                  90.48      PCNT     GOOD
ANM0123M  NI-44 PR CHAN Q3 1-MIN AVG                  90.56      PCNT     GOOD
URE0014   ROD BANK OUT OF SEQUENCE                RESET                   GOOD
URE0015   ROD TO BANK DEVIATION                   NORMAL                  GOOD
URE1650   CHAN OPER WARN BAND VIOLATION           RESET                   GOOD
URE1651   CHAN OPER BAND VIOLATION                RESET                   GOOD
URE1652   CHAN NOW OUT OF SERVICE                 RESET                   GOOD
URE1656   AXIAL FLUX DIFF ALARM                   RESET                   GOOD
```

ATTACHMENT 5
Page 1 of 3

<< Axial Flux Difference Log >>

AFD MONITOR OPERABLE

Tech Spec	4.2.1.1.a				
Parameter	Axial Flux Difference				
Instrument	NI-41C	NI-42C	NI-43C	NI-44C	AFD Monitor Checks Performed PER OP-163 (Initials)
Acceptance Criteria	Within AFD COLR Limits				
MODE	1 Above 50% Rated Thermal Power				
0800 - 1100					
2000 - 2300					

<< Axial Flux Difference Log >>

AFD MONITOR INOPERABLE

Tech Spec	4.2.1.1.b, 4.2.1.2								
Parameter	Axial Flux Difference								
Instrument	NI-41C	NI-42C	NI-43C	NI-44C	AVG Reactor Power	AFD Limits			
						Lower	Upper	Perform	Verify
Acceptance Criteria	Within AFD COLR Limits								
MODE	1 Above 50% Rated Thermal Power								
0000 - 0005									
0030 - 0035									
0100 - 0105									
0130 - 0135									
0200 - 0205									
0230 - 0235									
0300 - 0305									
0330 - 0335									
0400 - 0405									
0430 - 0435									
0500 - 0505									
0530 - 0535									
0600 - 0605									
0630 - 0635									
0700 - 0705									
0730 - 0735									
0800 - 0805									
0830 - 0835									
0900 - 0905									
0930 - 0935									
1000 - 1005									
1030 - 1035									
1100 - 1105									
1130 - 1135									
1200 - 1205									

Nightshift CRS Review ____________________

<< Axial Flux Difference Log >>

AFD MONITOR INOPERABLE

Tech Spec	4.2.1.1.b, 4.2.1.2								
Parameter	Axial Flux Difference								
Instrument	NI-41C	NI-42C	NI-43C	NI-44C	AVG Reactor Power	AFD Limits			
						Lower	Upper	Perform	Verify
Acceptance Criteria	Within AFD COLR Limits								
MODE	1 Above 50% Rated Thermal Power								
1230 - 1235									
1300 - 1305									
1330 - 1335									
1400 - 1405									
1430 - 1435									
1500 - 1505									
1530 - 1535									
1600 - 1605									
1630 - 1635									
1700 - 1705									
1730 -1735									
1800 - 1805									
1830 - 1835									
1900 - 1905									
1930 - 1935									
2000 - 2005									
2030 - 2035									
2100 - 2105									
2130 - 2135									
2200 - 2205									
2230 - 2235									
2300 - 2305									
2330 - 2335									

Dayshift CRS Review ________________________

Facility:	Harris Nuclear Plant	Task No.:	301079H401
Task Title:	During a Loss of Shutdown Cooling, determine the time that the RCS will reach Core Boiling and Boil-Off	JPM No.:	2020 NRC Exam Admin JPM SRO A1-2

K/A Reference: G2.1.20 RO 4.6 SRO 4.6 **ALTERNATE PATH - NO**

Examinee: ________________________ NRC Examiner: ________________

Facility Evaluator: ________________________ Date: ________________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom X Simulator ______ Plant ________

READ TO THE EXAMINEE

I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.

Initial Conditions:	The unit was operating at 100% power for the last 17 months. On 10/31/20 at 0000 the plant was shut down for a refueling outage. • While the Reactor cavity was being filled the 'A' RHR pump tripped. Motor repairs are not expected to be completed until 11/25/20. • The Reactor cavity fill was completed to the normal refueling levels with the 'B' RHR pump. • No fuel has been moved due to problems with the Manipulator Crane The current date and time is 11/20/20 at 1200 • Fuel still remains in the vessel due to complications with the Manipulator crane. • The 'B' RHR pump just tripped. • Core exit thermocouples are rising; they are currently reading 105°F

Initiating Cue:	You are directed to determine: 1. The time to reach core boiling 2. Core boil-off time **and** 3. The action(s) required to maintain level Mark up your curves to indicate where you are determining these times. Write your estimates of "time to boil" and "time to boil-off" and the required action(s) on the lines at the bottom of this page (below). Calculate your times in hours and minutes

Task Standard:	Candidate obtains curves and correctly identifies the time to reach core boiling and core boil-off time
Required Materials:	Curve Book Straight Edge
General References:	AOP-020, Loss Of RCS Inventory Or Residual Heat Removal While Shutdown, Rev. 39 Curve H-X-8, RCS Boiling Curves From Mid Loop, Rev. 3 Curve H-X-9, RCS Boiling Curves At Vessel Flange, Rev. 3 Curve H-X-10, RCS Boiloff Curves From Mid Loop To TAF, Rev. 3 Curve H-X-11, RCS Boiloff Curves From Vessel Flange To TAF, Rev. 3 OR 2020 NRC Exam Frozen Procedures Folder
Handout:	JPM Cue Sheets pages 8 - 13
Time Critical Task:	No
Validation Time:	10 minutes

PERFORMANCE STEP	**CRITICAL STEP JUSTIFICATION**
Step 3	Step required in order to accurately determine “time to boil” using the appropriate curve in order to determine the required actions in accordance with the appropriate plant procedure to maximize the available RCS inventory.
Step 4	Step required in order to accurately determine “time to boil-off” using the appropriate curve in order to determine the required actions in accordance with the appropriate plant procedure to maximize the available RCS inventory.
Step 5	Step required in order to determine the required actions in accordance with the appropriate plant procedure to maximize the available RCS inventory.

START TIME: ____________

Performance Step: 1 OBTAIN CURVES NEEDED FOR CALCULATION
(Curve Book will be provided to the candidate)

Standard: Refers to curves H-X-8 through H-X-11

Comment:

Performance Step: 2 Refers to provided data and determines that curve H-X-9 is required to calculate “time to boil” and curve H-X-11 is required to calculate “boil-off” time

Standard: Reviews curves and determines which ones are appropriate to determine the “time to boil” and “boil-off time”

Comment:

✓ **Performance Step: 3** Based on time since shutdown (10/31/20 – 11/20/20) 20 days 12 hours since shutdown and current RCS temperature of 105°F using curve H-X-9 determine “time to boil”.
(Interpolate 125°-150° lines)

Standard: Reviews curve H-X-9
Determines that “time to boil” is **~30 minutes**
(± 2 minutes, 28 – 32 min is acceptable)

Comment:

✓ **Performance Step: 4** Based on time since shutdown (10/31/20 – 11/20/20) 20 days 12 hours since shutdown and current RCS temperature of 105°F using curve H-X-11 determine "time to boil-off"

Standard: Reviews curve H-X-11

Determines that "time to boil-off" is **4 hrs and 50 mins (4.8 hrs)** (± 15 minutes) or (4 hours 35 minutes to 5 hours 5 minutes)

Comment:

✓ **Performance Step: 5** Determine the action required to maintain level for the plant conditions

Standard: Reviews AOP-020

Determines that the crew is required to REFER TO Table 1 below AND ADJUST CSIP flow to maintain level in accordance with current plant conditions.

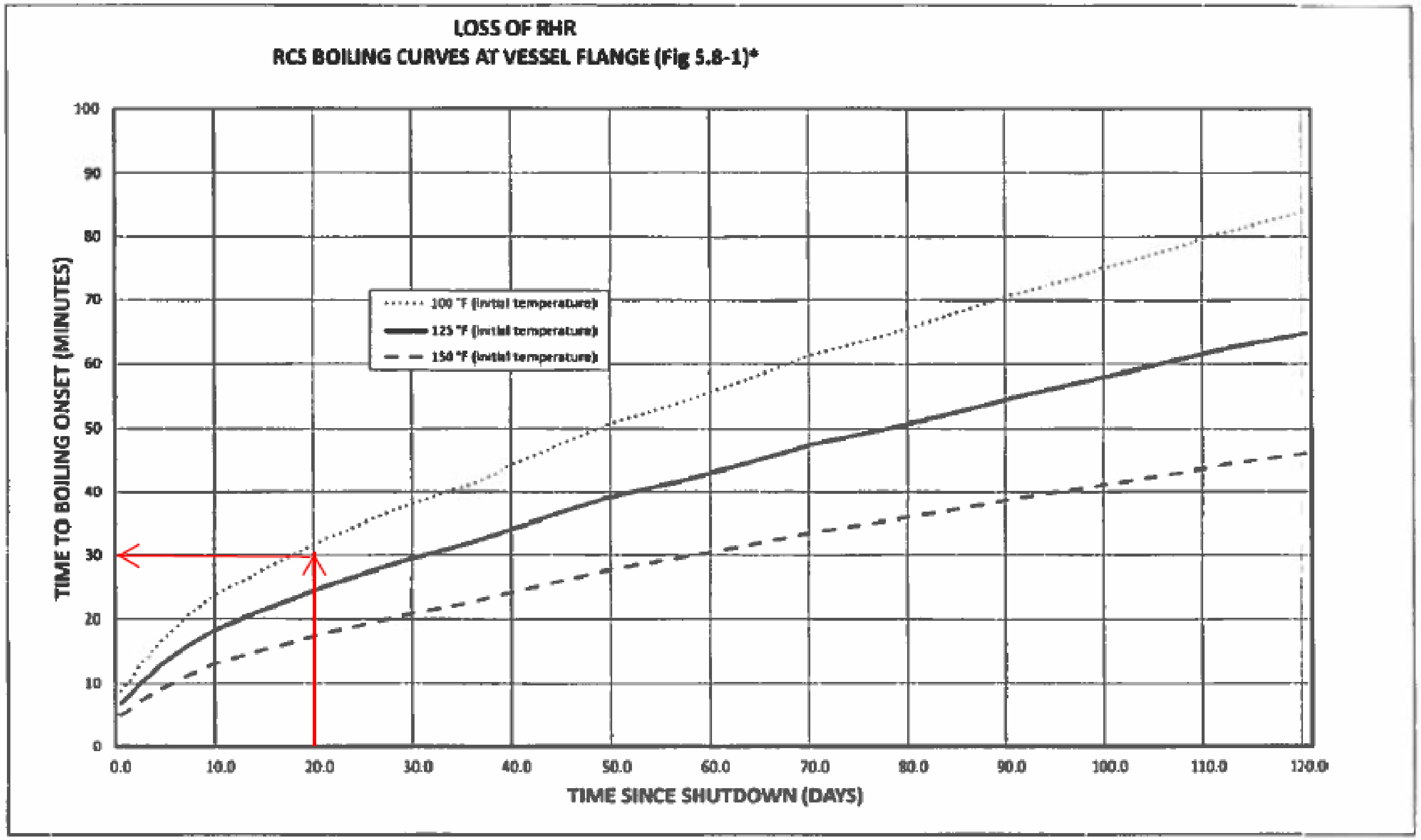
Comment:

Terminating Cue: **After completing the "time to boil", "time to boil-off" calculation and determining the action required, the evaluation on this JPM is complete.**

END OF JPM

STOP TIME: ____________

KEY



Curve No. H-X-9 Rev. No. 3

Originator Gregory A. Brown Date 6·22·12

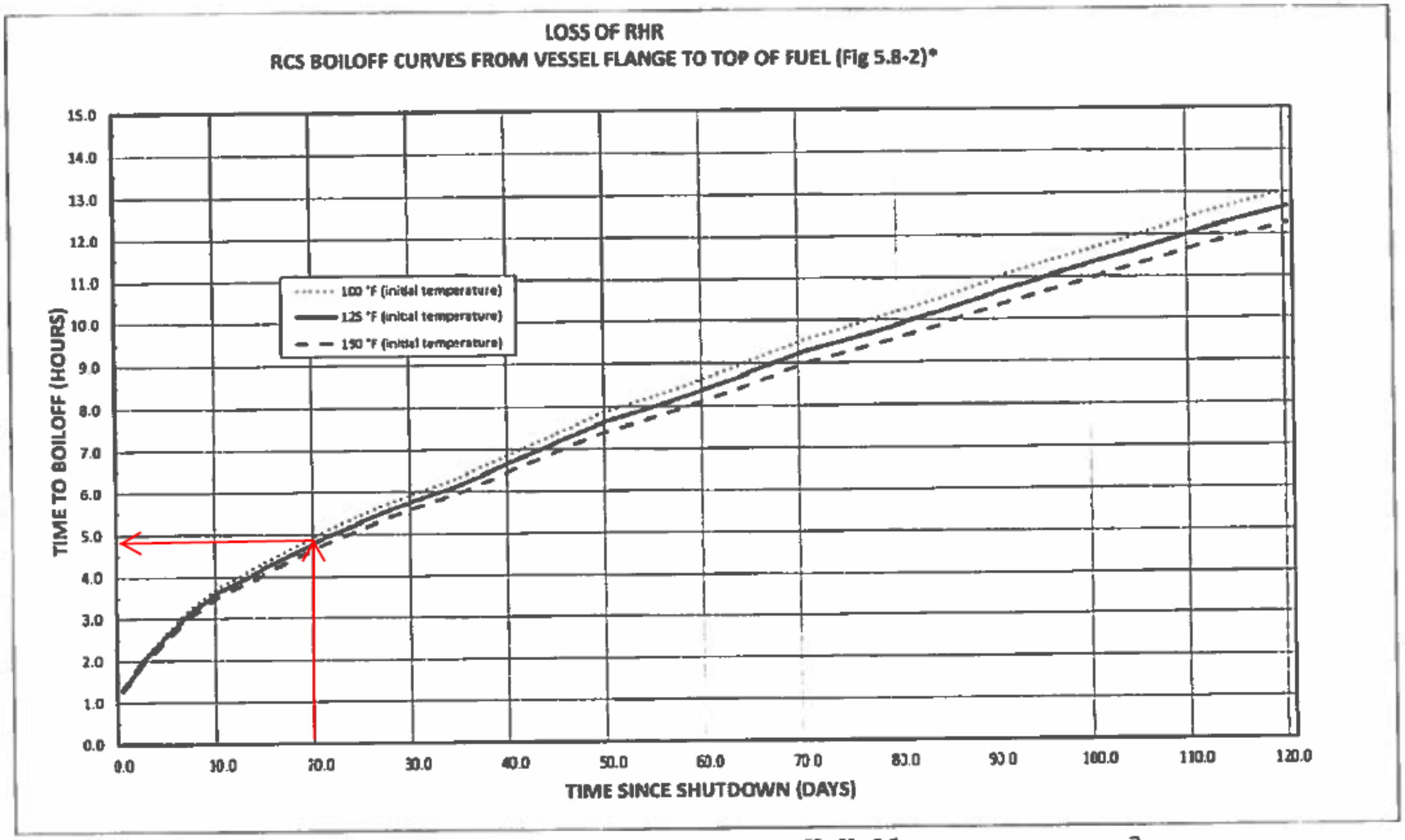
Supervisor Pat Date 6/25/12

Shift Manager Date 6/26/12

*Westinghouse CN-PCSA-10-22, Revision 0, "Loss of RHR Evaluation for the Harris Unit 1 NSSS Measurement Uncertainty Recapture Uprate," Nov 3, 2010

Initial conditions: Reactor cavity filled for refueling without fuel movement due to Manipulator Crane and Source Range problems. Core cooling is lost at 1200 and 20 days after shutdown. Core Exit Thermocouples are rising and are currently 105°F. Estimated time to boiling onset will be approximately 30 minutes from the time of the loss of cooling event.

KEY



Curve No. H-X-11 Rev. No. 3

Originator Gregory A. Brown Date 6-22-12

Supervisor Date 6/25/12

Shift Manager Date 6/26/12

*Westinghouse CN-PCSA-10-22, Revision 0, "Loss of RHR Evaluation for the Harris Unit 1 NSSS Measurement Uncertainty Recapture Uprate," Nov 3, 2010

Initial conditions: Reactor cavity filled for refueling without fuel movement due to Manipulator Crane and Source Range problems. Core cooling is lost at 1200 and 20 days after shutdown. Core Exit Thermocouples are rising and are currently 105°F. Estimated time to boiling onset will be approximately 4.9 hours from the time of the loss of cooling event.

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Admin JPM SRO A1-2

During a Loss of Shutdown Cooling, determine the time that the RCS will reach Core Boiling, Boil-Off and Required Actions

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	The unit was operating at 100% power for the last 17 months. On 10/31/20 at 0000 the plant was shut down for a refueling outage. - While the Reactor cavity was being filled the 'A' RHR pump tripped. Motor repairs are not expected to be completed until 11/25/20. - The Reactor cavity fill was completed to the normal refueling levels with the 'B' RHR pump. - No fuel has been moved due to problems with the Manipulator Crane The current date and time is 11/20/20 at 1200 - Fuel still remains in the vessel due to complications with the Manipulator crane. - The 'B' RHR pump just tripped. - Core exit thermocouples are rising; they are currently reading 105°F

Initiating Cue:	You are directed to determine: 1. The time to reach core boiling 2. Core boil-off time **and** 3. The action(s) required to maintain level Mark up your curves to indicate where you are determining these times. Write your estimates of "time to boil" and "time to boil-off" and the required action(s) on the lines at the bottom of this page (below). Calculate your times in hours and minutes

Name ____

Record your calculations here and return your curves to the examiner:

TIME TO BOIL (hours / minutes) ________________________

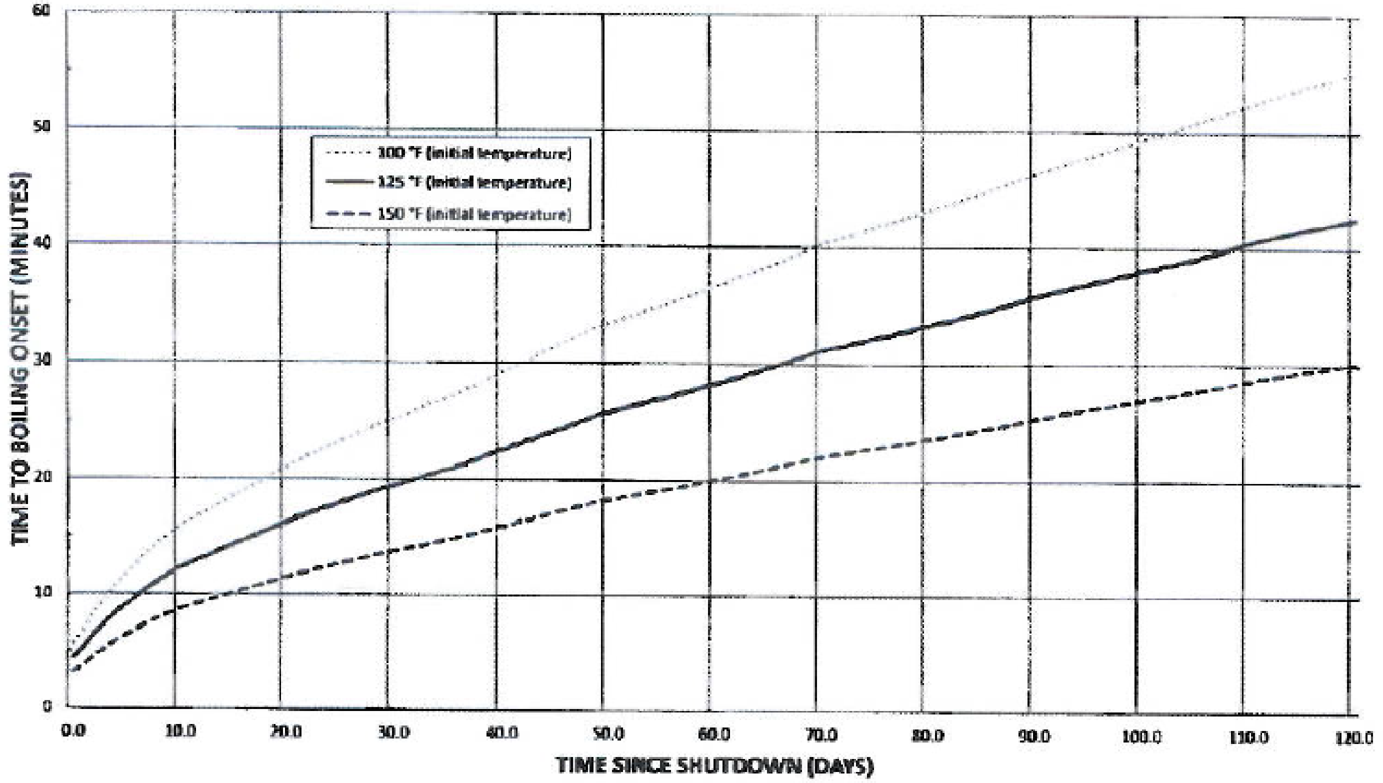
TIME TO BOIL-OFF (hours / minutes) ____________________

REQUIRED ACTION(S) TO MAINTAIN LEVEL ______________________________

____

____

LOSS OF RHR
RCS BOILING CURVES FROM MIDLOOP (Fig 5.8-3)*



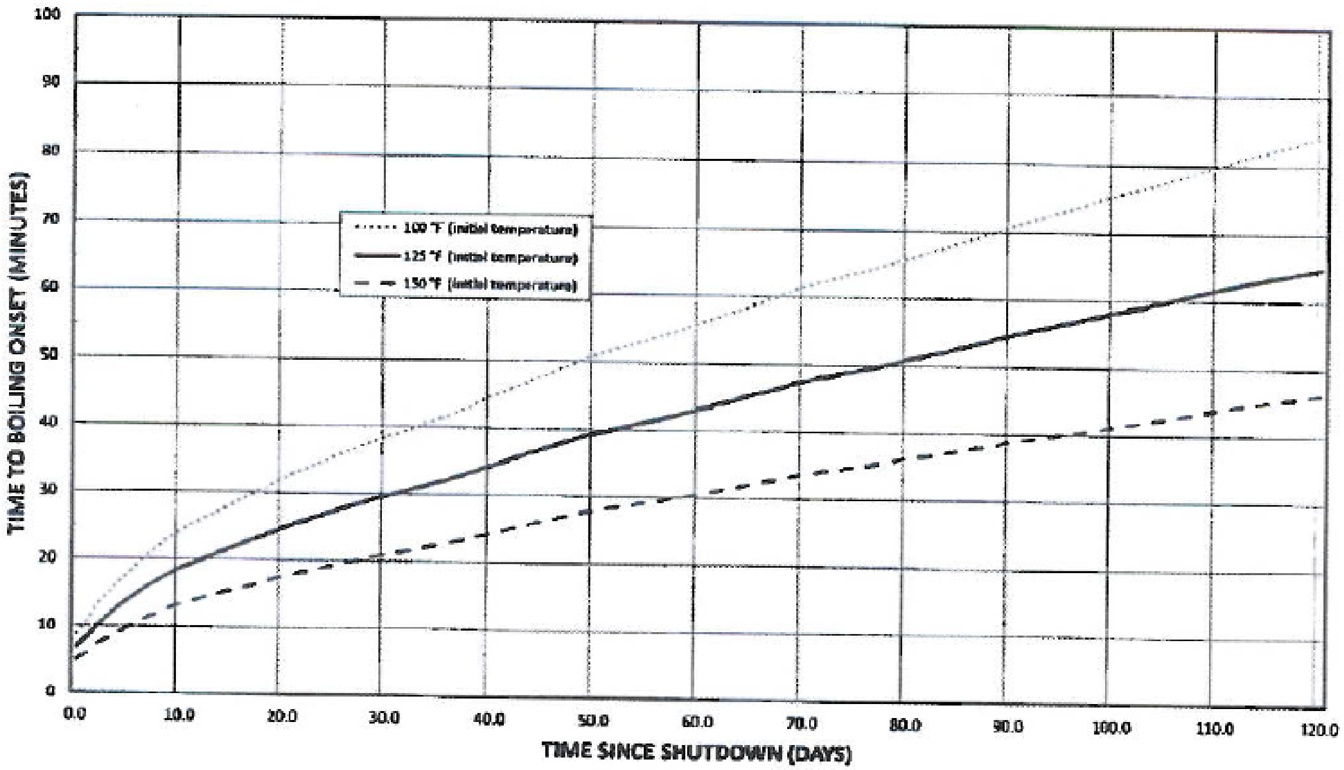
Curve No. H-X-8 Rev. No. 3

Originator Gregory A. Brown Date 6-22-12

Supervisor Date 6/25/12

Shift Manager Date 6/26/12

*Westinghouse CN-PCSA-10-22, Revision 0, "Loss of RHR Evaluation for the Harris Unit 1 NSSS Measurement Uncertainty Recapture Uprate," Nov 3, 2010



LOSS OF RHR
RCS BOILING CURVES AT VESSEL FLANGE (Fig 5.8-1)*

Curve No. H-X-9 Rev. No. 3

Originator Gregory A. Brown Date 6·22·12

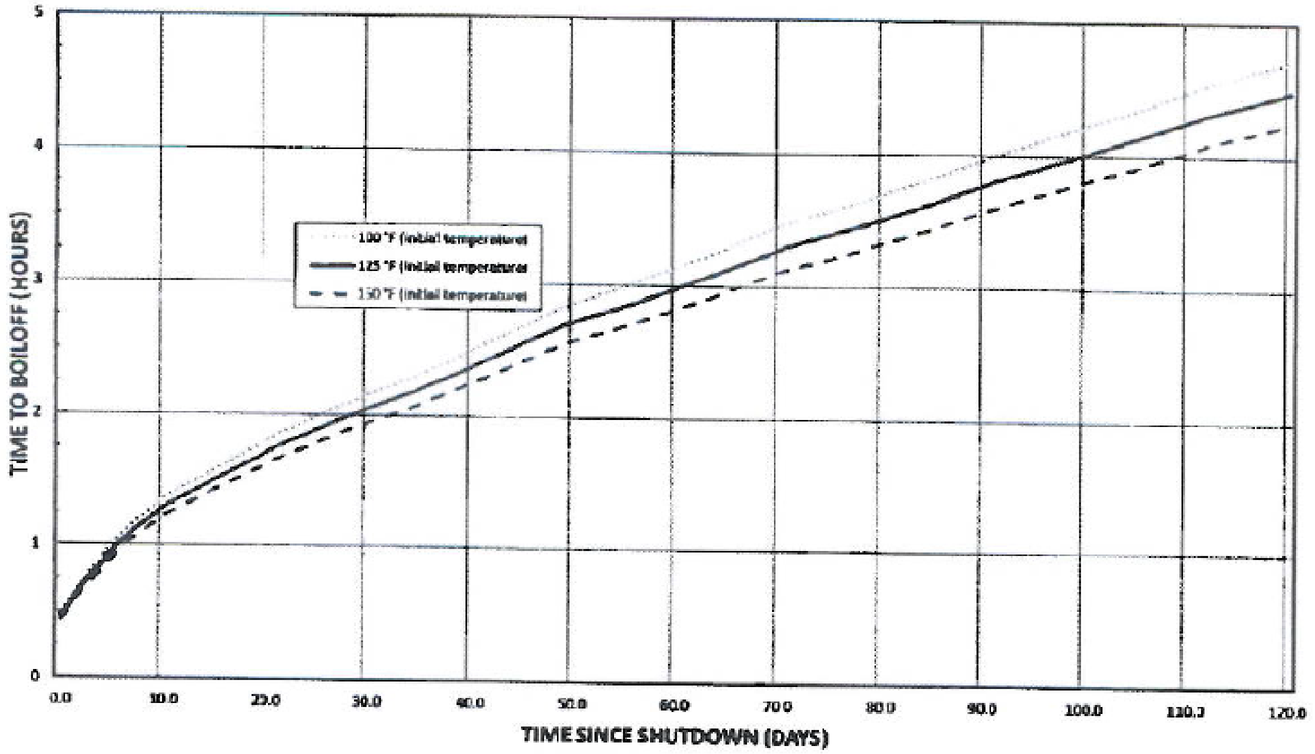
Supervisor Pat Chrisue Date 6/25/12

Shift Manager Date 6/26/12

*Westinghouse CN-PCSA-10-22, Revision 0, "Loss of RHR Evaluation for the Harris Unit 1 NSSS Measurement Uncertainty Recapture Uprate," Nov 3, 2010

JPM CUE SHEET

LOSS OF RHR
RCS BOILOFF CURVES FROM MID LOOP TO TOP OF FUEL (Fig 5.8-4)*



Curve No. H-X-10 Rev. No. 3

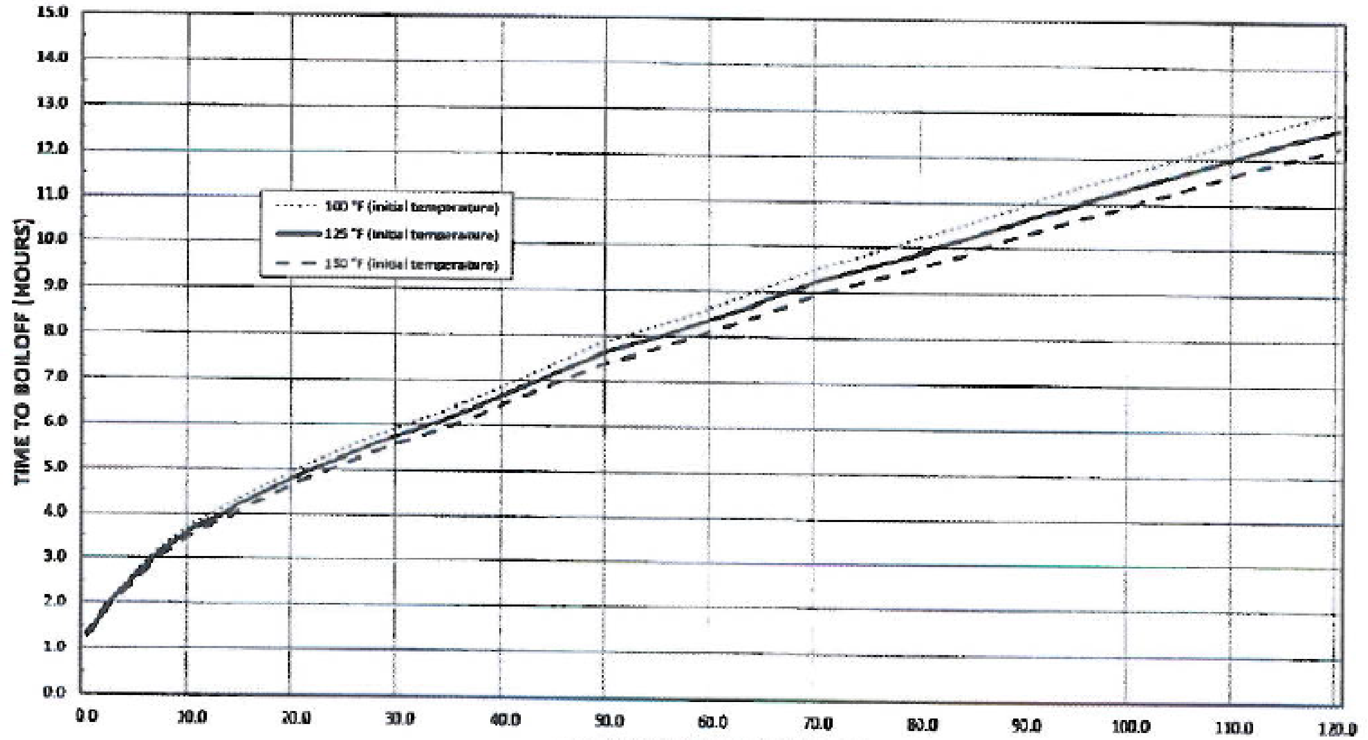
Originator Gregory A. Brown Date 6-22-12

Supervisor Pat Chesnut Date 6/25/12

Shift Manager G-S-L Date 6/26/12

*Westinghouse CN-PCSA-10-22, Revision 0, "Loss of RHR Evaluation for the Harris Unit 1 NSSS Measurement Uncertainty Recapture Uprate." Nov 3, 2010

LOSS OF RHR
RCS BOILOFF CURVES FROM VESSEL FLANGE TO TOP OF FUEL (Fig 5.8-2)*



Curve No. H-X-11 Rev. No. 3

Originator Gregory A. Brown Date 6-22-12

Supervisor Date 6/25/12

Shift Manager Date 6/26/12

*Westinghouse CN-PCSA-10-22, Revision 0, "Loss of RHR Evaluation for the Harris Unit 1 NSSS Measurement Uncertainty Recapture Uprate," Nov 3, 2010

Facility: Harris Nuclear Plant

Task No.: 002001H201

Task Title: Review (for approval) a completed surveillance for PORV block valves and Evaluate Tech Specs

JPM No.: 2020 NRC Exam Admin SRO JPM A2

K/A Reference: G 2.1.25 RO 3.7 SRO 4.1

ALTERNATE PATH - NO

Examinee: ______________________ NRC Examiner: ________________

Facility Evaluator: ______________________ Date: ________________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom X Simulator ______ Plant ________

READ TO THE EXAMINEE I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	• Today is 11/19/20 • The unit is operating at 100% power • PRZ PORV PCV-445B (1RC-116) has a failure in the SHUT circuit • 1RC-115 has been closed and power is removed • TS 3.4.4 Action b is in effect. LCOTR T-20-00431 has been initiated • The control room crew has completed OST-1017, Pressurizer PORV Block Valve Full Stroke Test Quarterly Interval Modes 1-2-3-4

Initiating Cue:	You are the CRS. Review the completed OST for approval. Identify any discrepancies and any required action(s), if applicable.

Task Standard:	Both errors and the correct Technical Specification actions identified.
Required Materials:	None
General References:	OST-1017, Pressurizer PORV Block Valve Full Stroke Test Quarterly Interval Modes 1-2-3-4, Rev. 22
Handout:	Completed OST-1017 with errors that align with the JPM content.
Time Critical Task:	No
Validation Time:	15 Minutes

PERFORMANCE STEP	**CRITICAL STEP JUSTIFICATION**
NOTE:	There are 2 items that will make the surveillance UNSAT. Either of which when identified would require a performance retest.
Step 2	The stopwatch is beyond the calibration date –all timing data collected with the use of this out of calibration device is non reliable therefore the test is invalid until a satisfactory stop watch calibration check is performed.
Step 3	The shut time for valve 1RC-113 has exceeded the limit – if not identified an inoperable component could fail when needed to perform it’s intended action.
Step 4	If the wrong Tech Spec Action is selected an LCO action could be exceeded

START TIME: ____________

Performance Step: 1 Obtain procedure.

Standard: Reviews Sections 3.0, 4.0, 5.0, 6.0.

Evaluator Cue: **Provide handout for 2020 NRC JPM SRO A2.**

Evaluator Note:

- **The steps of reviewing the procedure can be completed in any order.**
- **There are three errors in the procedure. Only the errors are documented in the JPM.**

Comment:

✓ **Performance Step: 2** Review the completed OST-1017.

Standard: Identifies Stopwatch beyond calibration date per Prerequisite 3.0.4.

Comment:

✓ **Performance Step: 3** Review the completed OST-1017.

Standard: Identifies SHUT time for 1RC-113 exceeds LIMITING VALUE, **(✓) should be retest in accordance with Attachment 3 of OST-1017 or (✓) declared Inoperable** and an AR should be initiated.

Evaluator Cue: **If a retest of 1RC-113 is determine inform the candidate the second stroke time results are the same as the first test.**

Comment:

✓ **Performance Step: 4** Obtain and Evaluate Technical Specifications

Standard: Obtains Technical Specifications and refers to LCO 3.4.4

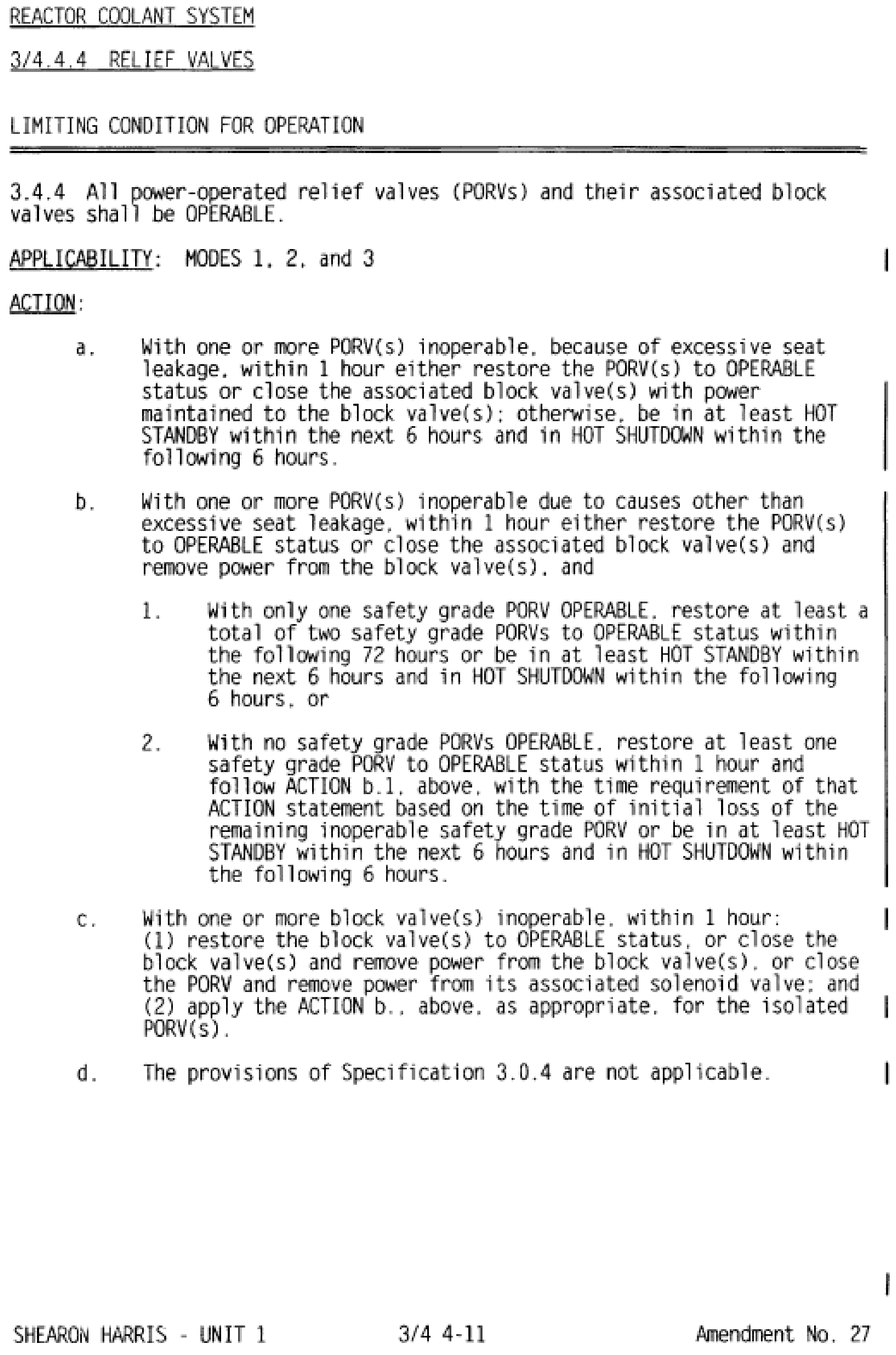
Determines that ACTION c. is applicable and ACTION b.1 would become applicable as directed by ACTION c(2) once the associated PRZ PORV PCV-444B SB is declared Inoperable. (See page 5)

Evaluator Note:	**After the candidate has identified the 2 errors in the procedure and performed a Technical Specification evaluation.** **END OF JPM**

Terminating Cue:	Current status of OST-1017 has been determined and the Technical Specifications evaluation completed.

STOP TIME: ______________

KEY



REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3

ACTION:

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

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and

   1. With only one safety grade PORV OPERABLE, restore at least a total of two safety grade PORVs to OPERABLE status within the following 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

   2. With no safety grade PORVs OPERABLE, restore at least one safety grade PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable safety grade PORV or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply the ACTION b., above, as appropriate, for the isolated PORV(s).

d. The provisions of Specification 3.0.4 are not applicable.

SHEARON HARRIS - UNIT 1 3/4 4-11 Amendment No. 27

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Admin SRO JPM A2
Review (for approval) a completed surveillance procedure for PORV block valves. OST-1017

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

Initial Conditions:	• Today is 11/19/20 • The unit is operating at 100% power • PRZ PORV PCV-445B (1RC-116) has a failure in the SHUT circuit • 1RC-115 has been closed and power is removed • TS 3.4.4 Action b is in effect. LCOTR T-20-00431 has been initiated • The control room crew has completed OST-1017, Pressurizer PORV Block Valve Full Stroke Test Quarterly Interval Modes 1-2-3-4

Initiating Cue:	You are the CRS. Review the completed OST for approval. Identify any discrepancies and the required actions, if applicable.

NAME: ______________________________________

DATE: ____________________

- IF discrepancies were identified from your review of OST-1017 list them all on the lines below.

____

____

____

____

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____

C
CONTINUOUS
USE

HARRIS NUCLEAR PLANT

PLANT OPERATING MANUAL

VOLUME 3

PART 9

PROCEDURE TYPE: OPERATION SURVEILLANCE TEST

NUMBER: **OST-1017**

TITLE:

PRESSURIZER PORV BLOCK VALVE FULL STROKE TEST QUARTERLY INTERVAL MODES 1-2-3-4

1.0 PURPOSE

This OST demonstrates the operability of each PORV block valve by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of action b. or c. in Tech Spec 3.4.4, per Tech Spec Surveillance 4.4.4.2. This OST also partially satisfies the Inservice Testing Program requirements.

2.0 REFERENCES

2.1. Plant Operating Manual Procedures

1. OP-100
2. ISI-801

2.2. Technical Specifications

1. 3.4.4, Relief Valves
2. 6.8.4.m (Inservice Testing Program)
3. 4.4.4.2, Relief Valves Surveillance Requirements

2.3. Final Safety Analysis Report

1. 5.4.13, Safety and Relief Valves

2.4. Drawings

1. 5-S-1301, Reactor Coolant System Sheet 2

2.5. Others

1. HNP IST Program

3.0 PREREQUISITES

1. Verify that the Reactor Coolant System is aligned in a manner that will support the performance of this OST. OP
2. Coordinate the performance of this OST with other plant evolutions such that the minimum equipment operating requirements of Tech Specs are met. OP
3. Obtain any tools and equipment required per Section 5.0. OP
4. Complete the Calibration Data Sheet and verify instrumentation is within calibration. OP
5. Verify instrumentation needed for the performance of this test is free of deficiencies that may affect instrument indication. OP
6. Verify all prerequisites are met, then obtain CRS permission to perform this OST.

Signature

Today
Date

4.0 PRECAUTIONS AND LIMITATIONS

1. Test only one PORV block valve at a time.
2. If testing is suspended for any reason return the block valve being tested to the as found position.
3. Do not test block valves which are closed with power removed in order to meet requirements of ACTION b. or c. in Tech Spec 3.4.4.
4. If any valve stroke time falls outside its Code Criteria, the valve will be immediately retested per the retest instructions or declared inoperable.

5.0 TOOLS AND EQUIPMENT

1. Calibrated Stopwatch

6.0 ACCEPTANCE CRITERIA

This OST will be completed satisfactorily when the following are verified:

1. Stroke times of PORV block valves tested are within the stated acceptance criteria as listed on Attachment 2, Valve Test Data.
2. Full open and full closed position indication observed by position indication lights is proper for each PORV block valve tested.
3. The PORV block valves complete a full cycle of travel.

7.0 PROCEDURE

NOTE: The following two steps should be signed off at the completion of the test.

1. If, during the performance of this test, a valve stroke time exceeds its Code Criteria, immediately retest the valve per Attachment 3. Otherwise this step is N/A. N/A OP ______

2. If, during the performance of this test, a valve exhibits abnormal or erratic action, document the condition in the comments section of Attachment 4. Otherwise this step is N/A. N/A OP ______

3. Complete the As Found positions in Step 7.0.5. OP ______

CAUTION

PORV block valves closed with power removed in order to meet the requirements of ACTION b. or c. of Tech Spec 3.4.4 are not required to be tested. The Comments Section of Attachment 2 must reference the applicable EIR number for PORV block valves not tested.

4. Refer to Attachment 2 and test all valves listed per the following:

NOTE: Steps 7.0.4.a through 7.0.4.l are for testing of all valves listed on Attachment 2. Initialing for Steps is done when all valves on Attachment 2 are tested.

	1RC-113	1RC-117
a. Verify the valve to be tested is aligned to the Pretest Position and initial the space provided on Attachment 2.	OP	OP
b. Simultaneously start the stopwatch and place the control switch for the valve to be tested to the position opposite the pretest position.	OP	OP
c. Stop the stopwatch when the valve has completed its travel as indicated by a singular position indicating light for the demanded position (no dual indication).	OP	OP

7.0 PROCEDURE (continued)

		1RC-113	1RC-117
d.	Record valve stroke time in space provided on Attachment 2.	OP	OP
e.	Place the control switch for the valve in test to the Posttest Position shown on Attachment 2 and start the stopwatch.	OP	OP
f.	Stop the stopwatch when the valve has completed its travel as indicated by a singular position indicating light for the demanded position (no dual indication).	OP	OP
g.	Record valve stroke time in space provided on Attachment 2.	OP	OP
h.	Verify that the valve has traveled to its Posttest Position as indicated by a singular position indicating light for the demanded position (no dual indication) and initial in the space provided on Attachment 2.	OP	OP
i.	Initial for the Full Stroke Test on Attachment 2 as verification of satisfactory valve operation (as previously performed per Steps 7.0.4.b through 7.0.4.h above).	OP	OP
j.	Repeat Steps 7.0.4.a through 7.0.4.i above to test all required remaining valves on Attachment 2.	OP	
k.	Perform independent verification of valves as required per Attachment 2.	NR	
l.	Review all data taken on Attachment 2 and ensure all stroke times are within the stated Acceptance Criteria. Inform the CRS of any out of tolerance reading and take appropriate action.	OP	

5. Restore components to their as found condition as follows:

		As Found	Restored	Verified
a.	1RC-113	OPEN	OPEN	NR
b.	1RC-115	SHUT	SHUT	NR Today
c.	1RC-117	OPEN	OPEN	NR

6. Complete Attachment 4, Certification and Reviews and inform the CRS when this test is completed or found to be unsatisfactory. OP

8.0 DIAGRAMS/ATTACHMENTS

Attachment 1 - Calibration Data

Attachment 2 - Valve Test Data

Attachment 3 - Valve Retest Data Sheet

Attachment 4 - Certifications and Reviews

Attachment 1 - Calibration Data

Sheet 1 of 1

Instrument	Instrument I.D.	Calibration Due Date
Stopwatch	CT 2359	11/01/20

Attachment 2 - Valve Test Data

Sheet 1 of 1

All spaces next to valve number shall be filled in with initials, data or N/A as applicable.

PRETEST ALIGNMENT			FULL STROKE TEST				FAIL SAFE TEST		POSTTEST ALIGNMENT			ACCEPTANCE CRITERIA (SEC)					
			Verification of Travel by Ind Lights (INIT)		Stroke Time (SEC)							CODE CRITERIA				LIMITING VALUE	
												OPEN		SHUT			
Valve Number	Pretest Position	Init	Stem	Ind Lights	OPEN	SHUT	Fail Safe Position	Position Verified	Posttest Position	Pos Init	Verf Init	Low	High	Low	High	OPEN	SHUT
1RC-113	OPEN	OP	N/A	OP	13.34	24.41	N/A	N/A	OPEN	OP	NR	11.67	15.77	13.49	18.25	20.58	23.80
1RC-115	OPEN	N/A	~~N/A~~				~~N/A~~	~~N/A~~	~~OPEN~~		N/A ①	N/A	N/A	13.56	18.34	N/A	23.92
1RC-117	OPEN	OP	N/A	OP	14.12	16.35	N/A	N/A	OPEN	OP	NR	11.94	16.14	14.26	19.28	21.06	25.15

Today

Comments: ① LCOTR 2020285, 1RC-115 closed with power removed C/O 241035

Attachment 3 - Valve Retest Data Sheet

Sheet 1 of 2

NOTE: This entire Attachment is N/A if no valve is retested due to exceeding the Code Criteria.

Determine if the stroke time exceeds the Limiting Value.

1. If the stroke time exceeds the Limiting Value, declare the valve inoperable and initiate an AR. (N/A if stroke time is less than the Limiting Value)
2. If the stroke time is less than the Limiting Value, but outside the Code Criteria limits, perform the following Steps:
    a. If the cause is known to be mechanical failure, or if a retest cannot be performed expeditiously, declare the valve inoperable and initiate an AR (except for PMTRs).
    b. If retesting the valve is desired, perform the following:

NOTE: If necessary, separate marked up sheets of this OST may be used to document necessary manipulations. These sheets would be attached to this procedure and noted in the comments Section of Attachment 4. (Certifications and Reviews)

(1) Determine which Steps need to be performed to set up conditions for testing the valve. CRS concurrence must be obtained and documented in the Comments section of Attachment 4. (Certifications and Reviews)

(2) Perform the Steps determined in the previous Step and document stroke times/valve positioning on Sheet 2.

(3) If retest results are still outside the Code Criteria, declare the valve inoperable and initiate an AR (except for PMTRs).

(4) If retest results are within the Code Criteria, perform the following:

(a) Declare the valve operable.

(b) Initiate a CR identifying test findings for the first and second tests.

(c) Send test results to Responsible Engineer (IST) for evaluation and documentation on the AR.

Attachment 3 - Valve Retest Data Sheet
Sheet 2 of 2

(1) Fill out PRETEST ALIGNMENT, POSTTEST ALIGNMENT, and ACCEPTANCE CRITERIA values for the valve(s) being tested using the values in the initial test Attachment.

PRETEST ALIGNMENT (1)			FULL STROKE TEST		POSTTEST ALIGNMENT (1)			ACCEPTANCE CRITERIA (SEC) (1)					
			Stroke Time (SEC)					CODE CRITERIA				LIMITING VALUE	
								OPEN		SHUT			
Valve Number	Pretest Position	Init	OPEN	SHUT	Posttest Position	Pos Init	Verf Init	Low	High	Low	High	OPEN	SHUT

Attachment 4 - Certifications and Reviews

Sheet 1 of 1

This OST was performed as a:

Periodic Surveillance Requirement: ✓

Postmaintenance Operability Test: ______

Redundant Subsystem Test: ______

Plant Conditions: 100 %

Mode: 1

OST Completed By: OP Rater

Date: Today

Time: Now

OST Performed By:

Initials	Name (Print)	Initials	Name (Print)
OP	OP Rater		
IVR	Indy Verifer		
L	RU Lucky		

General Comments/Recommendation/Corrective Actions/Exceptions:

① LCOTR 2020285, 1RC-115 closed with power removed C/O 241035

Pages Used: All

OST Completed with NO EXCEPTIONS/EXCEPTIONS:

______ CRS

Date: ______

Reviewed By: ______ Responsible Engineer (IST)

Date: ______

After receiving the final review signature, this OST becomes a QA RECORD and should be submitted to Records Management.

Revision Summary

General (Revision 22)

This revision, PRR 2108005, incorporates the following:

PRR 2108005 - Revise using the markup in [Y:\Engineering\Equipment Performance\IST\4th Interval Update\4th Interval Procedure Markups] as an aid to meet the requirements of the IST Program 4th Interval.

Description of Changes

Page	Section	Change Description
1	Cover	Deleted the cover PLP-100 Case Note. Basis: AD-OP-ALL-0106, Conduct of Infrequently Performed Tests or Evolutions, now describes IPTE determination and implementation.
2	2.5.1	Revised to HNP IST Program.
9	Att. 2 1RC-115	Replaced OPEN CRITERIA times and LIMITING VALUE with NA.
12	Att. 4	Deleted ANII signature.

Facility:	Harris Nuclear Plant	Task No.:	341021H102
Task Title:	Review and complete Operations Actions of AP-545, Attachment 3, Section II. Pre-Entry Planning Actions	JPM No.:	2020 NRC Exam Admin JPM SRO A3

K/A Reference: G.2.3.13 RO 3.4 SRO 3.8 **ALTERNATE PATH - NO**

Examinee: ________________________ NRC Examiner: ________________

Facility Evaluator: ________________________ Date: ________

Method of testing:

Simulated Performance: ______ Actual Performance: X

Classroom X Simulator ______ Plant ______

READ TO THE EXAMINEE	
I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	The plant is operating at 100% power FIN is preparing AP-545, Attachment 3, RCB Entry Permit to identify the source of Containment sump in-leakage

Initiating Cue:	You are the WCC SRO and have been asked to review and complete the Operations Actions of the Attachment 3, Section II. Pre-Entry Actions, using the attached data sheet. Note any problems or required actions in the spaces provided.

Task Standard:	Completes the Operations portion of AP-545, Attachment 3, Section II. Identifies the MIDS system is tagged out, the PAL is Operable, but OST-1082 will be required to performed if either airlock is operated.
Required Materials:	None
General References:	AP-545, Containment Entries, Rev 61 OR 2020 NRC Exam Frozen Procedures Folder
Handout:	JPM Information Sheet Partially completed AP-545, Attachment 3 JPM Cue Sheet for LCOTR T-20-00346 and T-20-00311
Time Critical Task:	No
Validation Time:	15 minutes

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 2	Must ensure the MIDS system is tagged out to prevent inadvertent exposure to a source of radiation that has not been evaluated during entry inside containment.
Step 3	Must determine operability status for the PAL and the EAL to comply with Technical Specifications
Step 4	Must determine surveillance requirements for the PAL and the EAL to comply with Technical Specifications

START TIME: ___________

	Performance Step: 1	Reviews AP-545, Attachment 3, Section I:Entry Description, for the completed RCB Entry Permit
	Standard:	Ensures proper conditions, signatures/initials, entry location and may verify the current revision of the procedure
	Comment:	
✓	**Performance Step: 2**	Reviews AP-545, Attachment 3, Section II: Pre-Entry Planning Actions, Operations Actions, Establish a clearance for all Incore Detector movement
	Standard:	**Reviews information sheet and determines OPS-1-16-1050-MIDSCLEAR-1292 has been established to tag-out the MIDS system and document the information on AP-545, Attachment 3 on Sheet 2 of 4.**
	Comment:	
✓	**Performance Step: 3**	Reviews AP-545, Attachment 3, Section II: Pre-Entry Planning Actions, Operations Actions, Determine operability of entry location
	Standard:	**Reviews provided LCOTR information and Technical Specification 4.6.1.3 to determine the PAL is Operable based on the last performance of the surveillance for the door.** **Documents the information on AP-545, Attachment 3 by checking the Operable box on Sheet 2 of 4.**
	Comment:	

✓ **Performance Step: 4** Reviews AP-545, Attachment 3, Section II: Pre-Entry Planning Actions, Operations Actions, Determine if TS surveillance requirement 4.6.1.3.b (OST-1082) is met for the door to be used for entry.

Standard: **Reviews LCOTR information and Technical Specification 4.6.1.3 to determine the PAL is NOT WITHIN PERIODICITY and documents the information on AP-545, Attachment 3 by circling NOT WITHIN PERIODICITY on Sheet 2 of 4. Documents that OST-1082 is required to be performed for the PAL on the JPM Cue sheet.**

Comment:

Performance Step: 5 Reviews AP-545, Attachment 3, Section II: Pre-Entry Planning Actions, Operations Actions, Determine if TS surveillance requirement 4.6.1.3.b (OST-1082) is met for the door to be used for entry.

Standard: **Reviews LCOTR information and Technical Specification 4.6.1.3 to determine the EAL is NOT WITHIN PERIODICITY and documents the information on JPM cue sheet if OST-1082 is NOT WITHIN PERIODICITY for either door, and that door is subsequently used for emergency exit, NOTIFY the WCC SRO or CRS that OST-1082 is required to be performed.**

Comment:

Performance Step: 6 Reviews AP-545, Attachment 3, Section II: Pre-Entry Planning Actions, Operations Actions, Establish maximum cooling mode.

Standard: **Reviews information sheet and determines Containment Cooling is in the Maximum Cooling Mode and initials AP-545, Attachment 3 on Sheet 2 of 4.**

Comment:

Performance Step: 7 Reviews AP-545, Attachment 3, Section II: Pre-Entry Planning Actions, Operations Actions, RCB elevator breaker operation.

Standard: **Reviews information sheet and determines RCB elevator breaker operation is not required and initials action as N/A on AP-545, Attachment 3 on Sheet 2 of 4.**

Comment:

Evaluator Note and Terminating Cue:	**When the procedure is returned: Evaluation on this JPM is complete.**

STOP TIME: ______________

KEY

CONTAINMENT ENTRIES	AP-545
	Rev. 61
	Page 27 of 42

ATTACHMENT 3
Page 2 of 4

<< RCB Entry Permit >>

II. Pre-Entry Planning Actions	INITIAL WHEN COMPLETED
RWO Lead(s) Actions	
• Contact all affected personnel to ensure they have completed Attachment 5,Attachment 6, and Attachment 7, as necessary.	e
• Ensure AD-RP-ALL-2011 ALARA briefing held.	e
• Discuss a communications plan including immediate RCB exit notification method (for example: pagers, PA or ASCOM phones) with entry team(s) to include method and expected frequency of communications.	e
• Review the material control chits and adjust Attachment 5,Attachment 6, and Attachment 7, as necessary (N/A if not applicable.)	e
• Designate and brief material control gatekeeper(s), if required to support the entry. (N/A if not applicable.)	N/A e
• Notify Security of the date and time of the entry.	e
Work Week Manager, Outage & Scheduling, Actions	
• Evaluate the impact of in-core detector maintenance on other work. (N/A if not applicable.)	N/A e
• The Work Week Manager has verified that there are no planned activities which will affect reactivity or reactor power (e.g., Feed Regulator Valve in Manual, Control Rod testing).	e
Operations Actions:	
• Establish a clearance for all Incore Detector movement: # OPS-1-16-1050-MID-1292	e
• Determine operability of entry location. ☑ Operable ☐ Inoperable	e
• Coordinate with the WCC SRO to determine if TS surveillance requirement 4.6.1.3.b (OST-1082) is met for the door to be used for entry: **PAL** – OST-1082 is WITHIN PERIODICITY / [✓ NOT WITHIN PERIODICITY] (circle one) **EAL** – OST-1082 is WITHIN PERIODICITY / NOT WITHIN PERIODICITY (circle one) If OST-1082 is NOT WITHIN PERIODICITY for either door, and that door is subsequently used for emergency exit, NOTIFY the WCC SRO or CRS that OST-1082 is required to be performed.	e
• Establish maximum cooling mode, if required. (Note: ESW temperature at suction is less than surface temperature and provides better cooling than NSW, AR 405289) N/A if not applicable.	e
• **IF** requested by RPM, **THEN close** the RCB elevator breakers per OP-113	N/A e
RC Actions	
RRSA Level per AD-RP-ALL-2006: ☑ N/A ☐ Medium ☐ High _______________ / _______________ RP/ALARA Technician Print/Sign	e
• Obtain Containment pre-entry atmosphere information. (radiological, oxygen, hydrogen (%LEL), carbon monoxide and temperature, as a minimum) by one of the following methods: 1. From the Normal Containment Purge Exhaust duct below CP-B9 (near RAB261 EAL) **OR** a. Verify with Operations that the Normal Containment Purge Exhaust is in service b. Remove a rubber plug from the duct work located below CP-B9 c. Obtain an atmospheric sample with a direct reading instrument (MX8 or equivalent) from the exhaust duct. d. Replace the rubber plug in the duct work. 2. Obtain a sample during the initial entry with a direct reading multi-gas instrument **OR** 3. Per CRC-244 **OR** CRC-821 within 24 hours of the start of the entry. 4. Record results in Section V - RCB Entry Comments of this Attachment	e
• Ensure that the RCB elevator has been locked out. Permission to use this elevator must be obtained from the RPM or designee.	e
Chemistry Actions:	
• Determine if RCS lithium hydroxide additions are in progress or planned.	e
• Sample Containment atmosphere, as requested. N/A if not applicable.	e
• Verify that all required chemicals and chemical cabinets have been requested. N/A if not applicable.	N/A e
• Notify Duty RP Supervisor, or designee, of any recently performed, in progress, or planned samples that could affect dose rates in Containment.	e
Maintenance Actions:	
• Designate a qualified door operator for the duration of the entry. N/A if not applicable.	e

✓ - Denotes Critical Steps

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Admin JPM SRO A3 – Review and complete Operations Actions of AP-545, Attachment 3, RCB Entry Permit, Section II. Pre-Entry Planning Actions

AP-545, Containment Entries, Attachment 3, RCB Entry Permit

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT ______

Examiner's Signature: ______________________________ Date: ____________________

JPM CUE SHEET

Initial Conditions:	The plant is operating at 100% power FIN is preparing AP-545, Attachment 3, RCB Entry Permit to identify the source of Containment sump in-leakage

Initiating Cue:	You are the WCC SRO and have been asked to review and complete the Operations Actions of the Attachment 3 Section II. Pre-Entry Actions using the attached data sheet. Note any problems or required actions in the spaces provided.

NAME ____ DATE _____________________

IF any action(s) were identified in the review of AP-545 list them on the lines below

____

____

____

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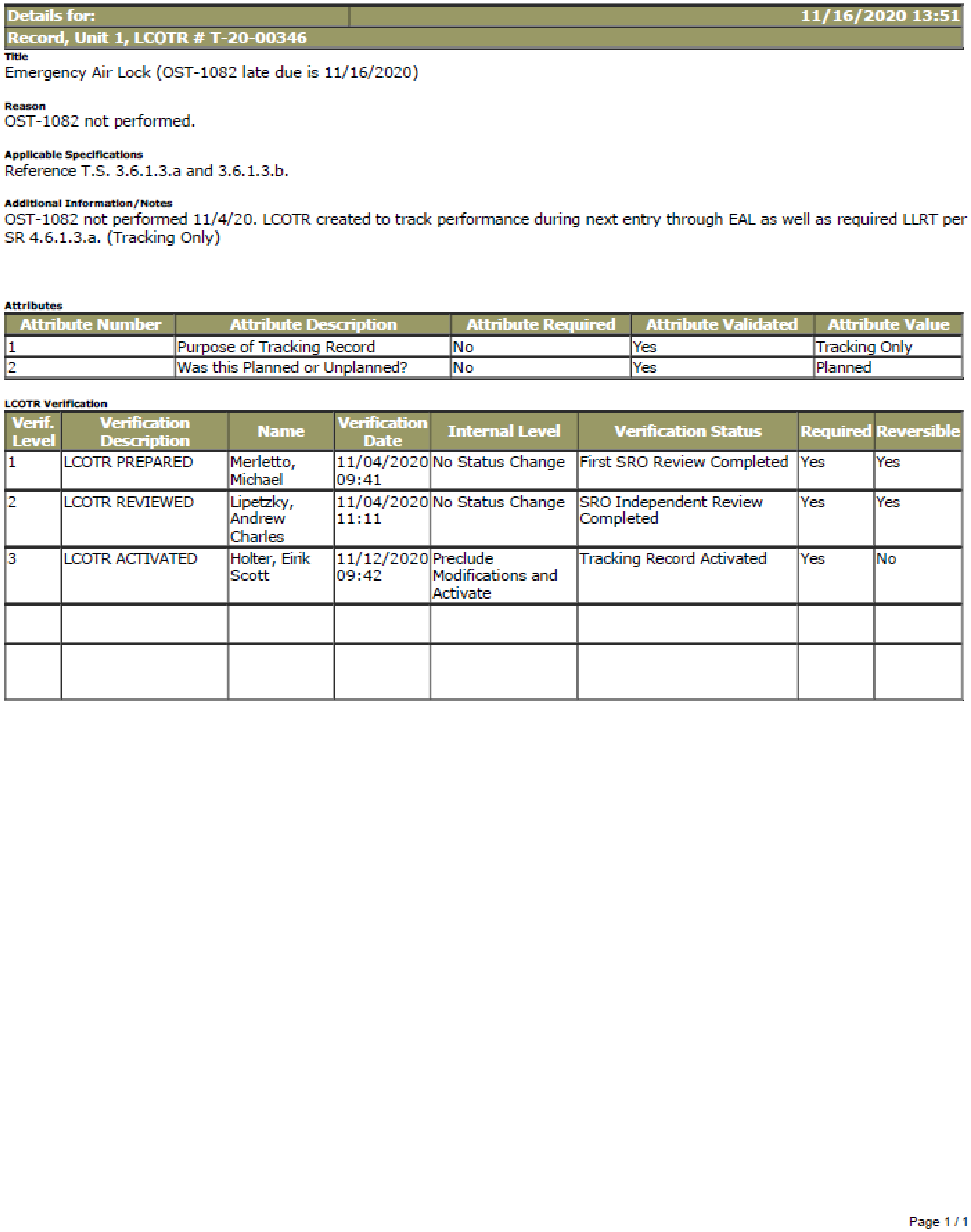
2020 NRC JPM SRO A3 Information Sheet

- Reactor Power is 98%
- Entry Date: November 19, 2020
- Entry Time: 0900
- Containment Temperature: 97˚F
- Allowable Reactor Power Band: 97% to 100%
- Entry Description: CNMT Entry to look for CNMT sump in-leakage. Remote robots will be used inside the bio-shield
- Entry Type: Planned
- Entry RWO Lead: FIN SRO.
- Entry Location: PAL
- RCB Elevator Operation is not required
- Clearance OPS-1-16-1050- MIDSCLEAR-1292 is hanging
- Containment Fan Coolers are in Maximum Cooling mode in accordance with OP-169
- LCOTR T-20-00346 and T-20-00311 are provided



Details for: **11/16/2020 13:51**

Record, Unit 1, LCOTR # T-20-00346

Title

Emergency Air Lock (OST-1082 late due is 11/16/2020)

Reason

OST-1082 not performed.

Applicable Specifications

Reference T.S. 3.6.1.3.a and 3.6.1.3.b.

Additional Information/Notes

OST-1082 not performed 11/4/20. LCOTR created to track performance during next entry through EAL as well as required LLRT per SR 4.6.1.3.a. (Tracking Only)

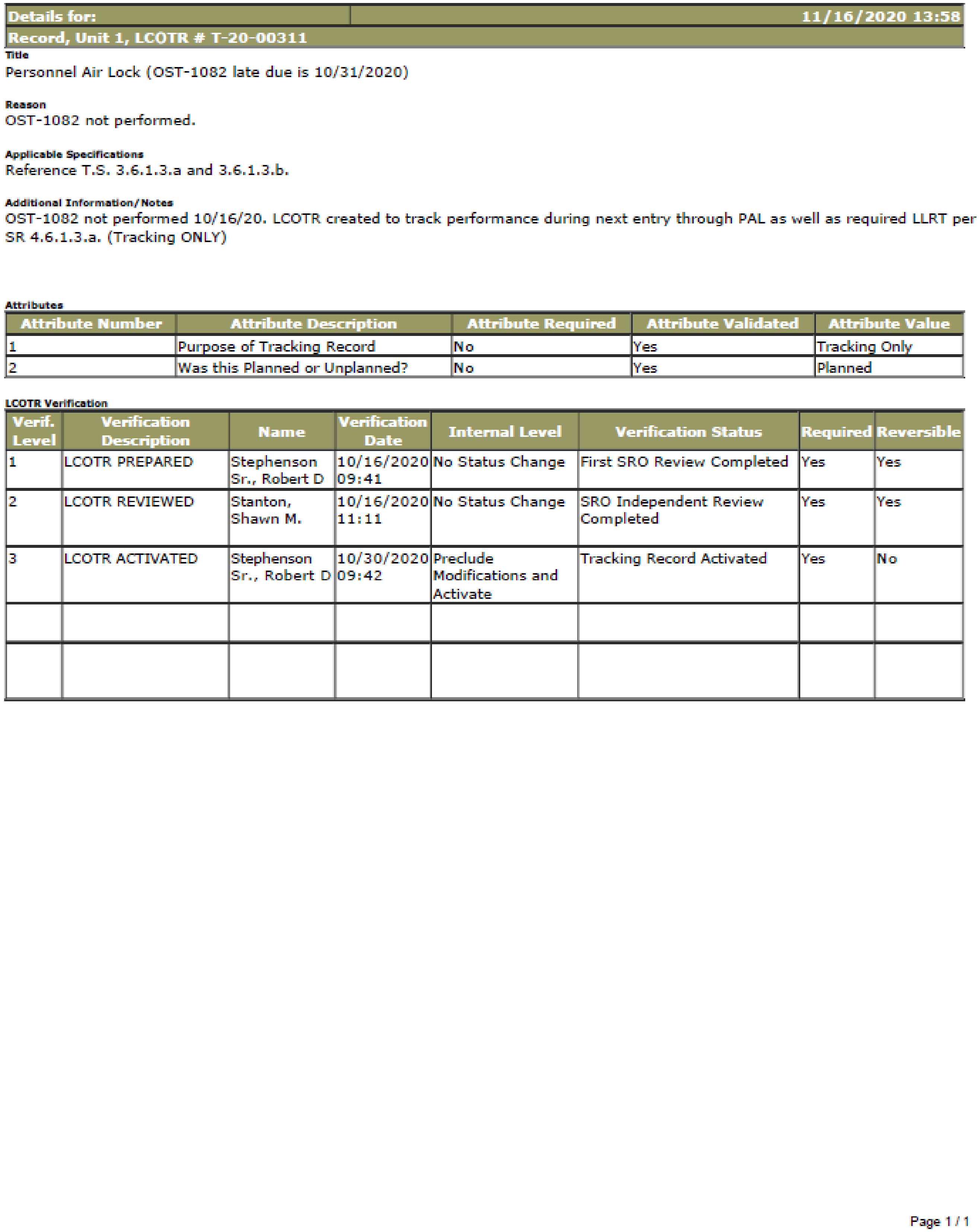
Attributes

Attribute Number	Attribute Description	Attribute Required	Attribute Validated	Attribute Value
1	Purpose of Tracking Record	No	Yes	Tracking Only
2	Was this Planned or Unplanned?	No	Yes	Planned

LCOTR Verification

Verif. Level	Verification Description	Name	Verification Date	Internal Level	Verification Status	Required	Reversible
1	LCOTR PREPARED	Merletto, Michael	11/04/2020 09:41	No Status Change	First SRO Review Completed	Yes	Yes
2	LCOTR REVIEWED	Lipetzky, Andrew Charles	11/04/2020 11:11	No Status Change	SRO Independent Review Completed	Yes	Yes
3	LCOTR ACTIVATED	Holter, Eirik Scott	11/12/2020 09:42	Preclude Modifications and Activate	Tracking Record Activated	Yes	No

Page 1 / 1



Details for:	**11/16/2020 13:58**
Record, Unit 1, LCOTR # T-20-00311	

Title

Personnel Air Lock (OST-1082 late due is 10/31/2020)

Reason

OST-1082 not performed.

Applicable Specifications

Reference T.S. 3.6.1.3.a and 3.6.1.3.b.

Additional Information/Notes

OST-1082 not performed 10/16/20. LCOTR created to track performance during next entry through PAL as well as required LLRT per SR 4.6.1.3.a. (Tracking ONLY)

Attributes

Attribute Number	Attribute Description	Attribute Required	Attribute Validated	Attribute Value
1	Purpose of Tracking Record	No	Yes	Tracking Only
2	Was this Planned or Unplanned?	No	Yes	Planned

LCOTR Verification

Verif. Level	Verification Description	Name	Verification Date	Internal Level	Verification Status	Required	Reversible
1	LCOTR PREPARED	Stephenson Sr., Robert D	10/16/2020 09:41	No Status Change	First SRO Review Completed	Yes	Yes
2	LCOTR REVIEWED	Stanton, Shawn M.	10/16/2020 11:11	No Status Change	SRO Independent Review Completed	Yes	Yes
3	LCOTR ACTIVATED	Stephenson Sr., Robert D	10/30/2020 09:42	Preclude Modifications and Activate	Tracking Record Activated	Yes	No

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<< RCB Entry Permit >>

Section I: Entry Description

Scheduled Entry Date: 11 / 19 / 2020 Time: 0400

Plant Mode: 1 Containment Temperature: 97 °F

Allowable Reactor Power Band for RCB Entry: 97 % to 100 %

The SM Signature below verifies that:

1. The CRS and MCR personnel have been briefed for the RCB entry that reactor power is planned to be constant while personnel are inside the bio-shield and the reactor power level is consistent with the conditions briefed on in Attachment 3, RCB Entry Permit, for the duration of the RCB entry.

Nuclear Shift Manager: ____________ Date/Time: ____________

Brief Entry Description:

CNMT Entry to look for CNMT sump in-leakage. Remote robots will be used inside the bio-shield

Type of Entry (review entry description and reference Section 3.0, Definitions):

Check only one: Planned ☑ Emergent ☐

RWO Lead(s): ____________

Chief RWO Lead (if required): N/A

Entry Location: PAL ☑ EAL ☐ Both PAL and EAL ☐

High RRSA Entry Approval: (required for entries inside the bio-shield; for SOER 01-1 entries on Elevation 286' or above when the reactor is critical (≥ 1E-08 amps); or as designated by the RPM. (N/A as necessary)

RPM: N/A Date/Time: N/A

PGM/SOM/Duty Manager - Harris Plant: N/A Date/Time: N/A

RCB Elevator Operation Approval (N/A as necessary):

RPM: N/A Date/Time: N/A

Copy of Approval Entry Permit Delivered to MCR:

RWO Lead: ____________ Date/Time: ____________

QA Record (or equivalent form)

ATTACHMENT 3
Page 2 of 4

<< RCB Entry Permit >>

II. Pre-Entry Planning Actions	INITIAL WHEN COMPLETED
RWO Lead(s) Actions	
• Contact all affected personnel to ensure they have completed Attachment 5,Attachment 6, and Attachment 7, as necessary.	e
• Ensure AD-RP-ALL-2011 ALARA briefing held.	e
• Discuss a communications plan including immediate RCB exit notification method (for example: pagers, PA or ASCOM phones) with entry team(s) to include method and expected frequency of communications.	e
• Review the material control chits and adjust Attachment 5,Attachment 6, and Attachment 7, as necessary (N/A if not applicable.)	e
• Designate and brief material control gatekeeper(s), if required to support the entry. (N/A if not applicable.)	N/A e
• Notify Security of the date and time of the entry.	e
Work Week Manager, Outage & Scheduling, Actions	
• Evaluate the impact of in-core detector maintenance on other work. (N/A if not applicable.)	N/A e
• The Work Week Manager has verified that there are no planned activities which will affect reactivity or reactor power (e.g., Feed Regulator Valve in Manual, Control Rod testing).	e
Operations Actions:	
• Establish a clearance for all Incore Detector movement: #	
• Determine operability of entry location. ☐ Operable ☐ Inoperable	
• Coordinate with the WCC SRO to determine if TS surveillance requirement 4.6.1.3.b (OST-1082) is met for the door to be used for entry: **PAL** – OST-1082 is WITHIN PERIODICITY / NOT WITHIN PERIODICITY (circle one) **EAL** – OST-1082 is WITHIN PERIODICITY / NOT WITHIN PERIODICITY (circle one) If OST-1082 is NOT WITHIN PERIODICITY for either door, and that door is subsequently used for emergency exit, NOTIFY the WCC SRO or CRS that OST-1082 is required to be performed.	
• Establish maximum cooling mode, if required. (Note: ESW temperature at suction is less than surface temperature and provides better cooling than NSW, AR 405289) N/A if not applicable.	
• **IF** requested by RPM, **THEN close** the RCB elevator breakers per OP-113	
RC Actions	
RRSA Level per AD-RP-ALL-2006: ☑ N/A ☐ Medium ☐ High ____________________/____________________ RP/ALARA Technician Print/Sign	e
• Obtain Containment pre-entry atmosphere information. (radiological, oxygen, hydrogen (%LEL), carbon monoxide and temperature, as a minimum) by one of the following methods: 1. From the Normal Containment Purge Exhaust duct below CP-B9 (near RAB261 EAL) **OR** a. Verify with Operations that the Normal Containment Purge Exhaust is in service b. Remove a rubber plug from the duct work located below CP-B9 c. Obtain an atmospheric sample with a direct reading instrument (MX6 or equivalent) from the exhaust duct. d. Replace the rubber plug in the duct work. 2. Obtain a sample during the initial entry with a direct reading multi-gas instrument **OR** 3. Per CRC-244 **OR** CRC-821 within 24 hours of the start of the entry. 4. Record results in Section V - RCB Entry Comments of this Attachment	e
• Ensure that the RCB elevator has been locked out. Permission to use this elevator must be obtained from the RPM or designee.	e
Chemistry Actions:	
• Determine if RCS lithium hydroxide additions are in progress or planned.	e
• Sample Containment atmosphere, as requested. N/A if not applicable.	e
• Verify that all required chemicals and chemical cabinets have been requested. N/A if not applicable.	N/A e
• Notify Duty RP Supervisor, or designee, of any recently performed, in progress, or planned samples that could affect dose rates in Containment.	e
Maintenance Actions:	
• Designate a qualified door operator for the duration of the entry. N/A if not applicable.	e

<< RCB Entry Permit >>

III. RCB Entry Actions

	INITIAL WHEN COMPLETED
Operations Actions	
• N/A	N/A
RWO Lead(s) Actions	
• Turn on the RCB lights.	N/A

Control STA 1A	Position	Initials	Control STA 1B	Position	Initials
LP-101	ON		LP-103	ON	
LP-102	ON		LP-104	ON	
LP-105	ON		LP-107	ON	
LP-106	ON		LP-123	ON	

• Personnel are designated to perform and document closeout inspection(s).	
• For entry(s) expected beyond 45 minutes, is heat stress evaluation completed? Contact Job Supervisor for information. (N/A if not applicable.)	
• When ready to enter Containment, request permission from the CRS to enter Containment.	
• Notify the CRS by **direct verbal contact** of the time the PAL/EAL door is opened to allow updating the eSoms LCOTR with the time so that Operations personnel responsible for LLRT are aware of the Date and Time of the initial entry.	
• Notify the RCC Lead when personnel initially enter the RCB.	
RC Actions	
• Validate Containment atmosphere information. (i.e., H_2, O_2, and temperature, as a minimum)	
• Survey for radiological conditions in respective work/inspection area(s).	

IV. RCB Closeout Actions

	INITIAL WHEN COMPLETED
Operations Actions	
• **IF** requested by RPM, **THEN open** the RCB elevator breakers per OP-113.	
RC Actions	
• **IF** requested by RPM, **THEN lock** the RCB elevator breakers per OP-113.	
RWO Lead(s) Actions	
• Turn off the RCB lights.	N/A

Control STA 1A	Position	Initials	Control STA 1B	Position	Initials
LP-101	OFF		LP-103	OFF	
LP-102	OFF		LP-104	OFF	
LP-105	OFF		LP-107	OFF	
LP-106	OFF		LP-123	OFF	

• Coordinate the performance of a closeout inspection(s) per procedure Section 5.5.	
• Obtain completed Attachment 4(s) from personnel designated to perform and document closeout inspection(s).	
• Notify the CRS of the time the PAL/EAL door is closed to allow updating the eSoms LCOTR with the time.	
• SM notified of closeout inspection completion and all personnel have exited Containment.	
• Verified PAL/EAL doors have been secured by Security/Radiation Control	
• This form and Attachment 4, Attachment 5, Attachment 6, and Attachment 7 (if used) are reviewed for completion.	
• If the Containment entry revises the "Sq. Ft. Aluminum Margin Remaining" on Attachment 7 or the "Loose Material Margin Remaining" on Attachment 13, write a PRR to revise AP-545 for updated values. N/A if not applicable.	

QA Record (or equivalent form)

ATTACHMENT 3
Page 4 of 4

<< RCB Entry Permit >>

V. RCB Entry Comments

Pre-Entry Atmospheric Information:

1. Airborne Radioactivity ___________________µCi/ml from REM-01LT-3502A or Chemistry Sample (circle one)
2. Atmospheric Quality: MX6 or equivalent Serial #: _______________

   CP-B9, Initial Entry or Chemistry Sample (circle one)

__________%O2 (Range: 19.5% to 23.5%) __________%LEL (<10%) __________ppm CO (<35 ppm)

Performed By (Print Name)	Initials	Date

VI. RCB Entry Permit Cancellation

The below signatures indicate that the RCB entry has been completed and all personnel have exited the RCB.

Completed by (RWO Lead): ____ Date: ____________

SM review: ____ Date: ____________

NOTE

After reviews are complete, submit forms to Radiation Protection per Section 5.6.

QA Record (or equivalent form)

`Facility: Harris Nuclear Plant
Task No.: 345001H602

Task Title: Classify an Event
JPM No.: 2020 NRC Exam Admin JPM SRO A4

K/A Reference: G2.4.38 RO 2.4 SRO 4.4
G2.4.41 RO 2.9 SRO 4.6

ALTERNATE PATH - NO

Examinee: ________________________
NRC Examiner: ________________

Facility Evaluator: ________________________
Date: __________________

Method of testing:

Simulated Performance: ______
Actual Performance: X

Classroom X Simulator ______ Plant ________

READ TO THE EXAMINEE I will explain the initial conditions, which steps to simulate, discuss or perform, and provide initiating cues. When you complete the task successfully, the objective for this Job Performance Measure will be satisfied.	
Initial Conditions:	**This is a TIME CRITICAL JPM.** Given the following plant conditions: • A shutdown for refueling is underway • RCS Temperature is 193°F Fuel movement is taking place in the Spent Fuel Pool (SFP) when the Bridge Crane operator noted that the pool water level is rapidly lowering. The Control Room was notified and an AO was dispatched to investigate the possible leakage source. • The SFP Area radiation monitors are all reading slightly <1.0 mr/hr The following occurs at 1115: • A loss of offsite power occurs The time is now 1131: • The leak was identified on the 'A' SFP suction strainer and is now isolated • Offsite power has been restored • Spent fuel pool A level is at 280.6' • Several SFP Area radiation monitors have increased to 2.5 mr/hr

Initiating Cue:	Evaluate the EAL Matrix and determine the **<u>HIGHEST</u>** classification required for these plant conditions. **NOTE: DO NOT use SEC judgment.** Write out the HIGHEST EAL classification in blank provided then return your assessment page to the Evaluator.

Task Standard: Event classified as an Site Area Emergency (RU1.1) within 15 minutes.

Required Materials: None

General References: CSD-EPHNP-0101-01, EAL Technical Basis Document, Rev 01
CSD-EPHNP-0101-02, EAL Matrix, Rev 00

OR
2020 NRC Exam Frozen Procedures Folder

Handouts: CSD-EPHNP-0101-01, EAL Technical Basis Document, Rev 01
CSD-EPHNP-0101-02, EAL Matrix, Rev 00
Attached Initial Conditions

Time Critical Task: **YES** – 15 minutes for classification.

Validation Time: 15 minutes for classification

PERFORMANCE STEP	CRITICAL STEP JUSTIFICATION
Step 2	Classification of the event is critical for determining State and County notifications, public information notices, site information notices, and event reportability to the Nuclear Regulatory Commission.
Step 4	Timely classification of the event is critical for determining State and County notifications, public information notices, site information notices, and event reportability to the Nuclear Regulatory Commission.

Evaluator Cue:	**Start Time for this portion of JPM begins when the individual has been briefed.**

START TIME: ____________

Performance Step: 1 OBTAINS EAL Technical Basis Document and EAL Matrix.

Standard : Obtains EAL Technical Basis Document and EAL Matrix.

Comments:

✓ **Performance Step: 2** Identify EAL Classification for events in progress

Standard : The candidate should evaluate three potential classifications for these conditions at a minimum:

CU1.2 Unusual Event
RCS water level cannot be monitored
AND EITHER
- UNPLANNED increase in any Table C-1 sump or tank due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

CU2.1 Unusual Event
AC power capability, Table C-6, to emergency 6.9 KV buses 1A-SA and 1B-SB reduced to a single power source for ≥ 15 min. (Note 1)
AND
Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS

RU2.1 Unusual Event is MET for these conditions
UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication (LI-01SF-5101A/LI-01SF-5102A/LI-01SF-5103A, LI-403 or RCS standpipe)
AND
UNPLANNED rise in corresponding area radiation levels as indicated by any Table R-2 area radiation monitors

Comments:

Performance Step: 3 Verify Classification

Standard : Reviews EAL Technical Basis Document to verify classification

Comments:

✓ **Performance Step: 4** Verify Classification Completion Time

Standard : Stop minus start time less than or equal to 15 minutes

Comments:

Examiners Cue:	**After the candidate returns this JPM Classification, document the stop time and then announce.** **END of JPM.**

STOP TIME:

START TIME ____________

STOP TIME ____________

Stop minus start time less than or equal to 15 minutes ____________

ATTACHMENT 1
EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication (LI-5101A/LI-5102A/LI-5103A, LI-403 or RCS standpipe)

AND

UNPLANNED rise in corresponding area radiation levels as indicated by **any** Table R-2 area radiation monitors

Table R-2 Refueling Pathway Area Radiation Monitors

Containment
- RM-1CR-3561A-SA Containment Ventilation Isolation
- RM-1CR-3561B-SB Containment Ventilation Isolation
- RM-1CR-3561C-SA Containment Ventilation Isolation
- RM-1CR-3561D-SB Containment Ventilation Isolation

Fuel Handling Building
- RM-1FR-3564A-SA Spent Fuel Pool SW, SE, SW
- RM-1FR-3564B-SB Spent Fuel Pool SW, SE, SE
- RM-1FR-3565A-SA Spent Fuel Pool SW, SE, SW
- RM-1FR-3565B-SB Spent Fuel Pool SW, SE, SE
- RM-1FR-3566A-SA Spent Fuel Pool NE, NW, NE
- RM-1FR-3566B-SB Spent Fuel Pool NW, NE, NW
- RM-1FR-3567A-SA Spent Fuel Pool NW, NE, NW
- RM-1FR-3567B-SB Spent Fuel Pool NE, NW, NE

Mode Applicability:

All

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY - The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway.

Basis:

The spent fuel pool low water level alarm setpoint is actuated at a setpoint of 284 ft. (ref. 1, 2, 3). Water level restoration instructions are performed in accordance with AOPs (ref. 4, 5).

The listed SFP level and refueling cavity level instruments provide indication of REFUELING PATHWAY level drop (ref. 7, 8).

ATTACHMENT 1
EAL Bases

The specified radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 4, 5, 6). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING CAVITY level are not classifiable under this EAL.

When the spent fuel pool and reactor cavity are connected, there could exist the possibility of uncovering irradiated fuel. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the reactor vessel and spent fuel pool.

This IC addresses a decrease in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an unplanned loss of water level.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RA2.

HNP Basis Reference(s):

1. APP-ALB-023/4-17, SPENT FP HI/LO LEVEL
2. APP-ALB-023/4-18, SFP C HI/LO LEVEL
3. APP-ALB-023/5-18, SFP D HI/LO LEVEL
4. AOP-013, Fuel Handling Accident
5. AOP-031, Loss of Refueling Cavity Integrity
6. AOP-005, Radiation Monitoring System
7. AOP-20, Loss of RCS Inventory or Residual Heat Removal While Shutdown – Basis Document
8. EC 89579
9. NEI 99-01 AU2

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2020 NRC Exam Admin JPM SRO A4
Classify an Event
CSD-EPHNP-0101-01, EAL Technical Basis Document
CSD-EPHNP-0101-02, EAL Matrix

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

JPM CUE SHEET

Initial Conditions:	**This is a TIME CRITICAL JPM.** Given the following plant conditions: • A shutdown for refueling is underway • RCS Temperature is 193°F Fuel movement is taking place in the Spent Fuel Pool (SFP) when the Bridge Crane operator noted that the pool water level is rapidly lowering. The Control Room was notified and an AO was dispatched to investigate the possible leakage source. • The SFP Area radiation monitors are all reading slightly <1.0 mr/hr The following occurs at 1115: • A loss of offsite power occurs The time is now 1131: • The leak was identified on the 'A' SFP suction strainer and is now isolated • Offsite power has been restored • Spent fuel pool A level is at 280.6' • Several SFP Area radiation monitors have increased to 2.5 mr/hr

Initiating Cue:	Evaluate the EAL Matrix and determine the **HIGHEST** classification required for these plant conditions. **NOTE: DO NOT use SEC judgment.** Write out the **HIGHEST** EAL classification in blank provided then return your assessment page to the Evaluator.

Name: ____

Date: ________________

Highest EAL Classification for the plant conditions: ________________