Facility: HARRIS Date of Exam: MARCH 2018

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

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

T1G1 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
007EK2.03	Reactor Trip - Stabilization - Recovery / 1	3.5	3.6	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Reactor trip status panel
009EK2.03	Small Break LOCA / 3	3	3.3	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	S/Gs
015AG2.1.23	RCP Malfunctions / 4	4.3	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to perform specific system and integrated plant procedures during all modes of plant operation.
025AA2.07	Loss of RHR System / 4	3.4	3.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Pump cavitation
026AA1.01	Loss of Component Cooling Water / 8	3.1	3.1	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	CCW temperature indications
027AG2.2.22	Pressurizer Pressure Control System Malfunction / 3	4.0	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of limiting conditions for operations and safety limits.
038EA1.10	Steam Gen. Tube Rupture / 3	3.7	3.7	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Control room radiation monitoring indicators and alarms
054AK1.01	Loss of Main Feedwater / 4	4.1	4.3	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	MFW line break depressurizes the S/G (similar to a steam line break)
055EK1.02	Station Blackout / 6	4.1	4.4	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Natural circulation cooling
056AG2.4.9	Loss of Off-site Power / 6	3.8	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.
057AK3.01	Loss of Vital AC Inst. Bus / 6	4.1	4.4	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Actions contained in EOP for loss of vital ac electrical instrument bus

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
058AA2.03	Loss of DC Power / 6	3.5	3.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	DC loads lost; impact on ability to operate and monitor plant systems
062AA2.06	Loss of Nuclear Svc Water / 4	2.8	3.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	The length of time after the loss of SWS flow to a component before that component may be damaged
065AK3.04	Loss of Instrument Air / 8	3	3.2	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Cross-over to backup air supplies
077AA1.03	Generator Voltage and Electric Grid Disturbances / 6	3.8	3.7	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Voltatge regulator controls
WE04EK1.3	LOCA Outside Containment / 3	3.5	3.9	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Annunciators and conditions indicating signals, and remedial actions associated with the (LOCA Outside Containment).
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.
WE11EK3.4	Loss of Emergency Coolant Recirc. / 4	3.6	3.8	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

T1G2 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
001AK1.22	Continuous Rod Withdrawal / 1	3.2	3.6	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Delta flux (I)
036AK3.02	Fuel Handling Accident / 8	2.9	3.6	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Interlocks associated with fuel handling equipment
037AA2.01	Steam Generator Tube Leak / 3	3	3.4	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Unusual readings of the monitors; steps needed to verify readings
051AA2.02	Loss of Condenser Vacuum / 4	3.9	4.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Conditions requiring reactor and/or turbine trip
068AA1.28	Control Room Evac. / 8	3.8	4	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	PZR level control and pressure control
076AK2.01	High Reactor Coolant Activity / 9	2.6	3	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Process radiation monitors
we02EG2.4.20	SI Termination / 3	3.8	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of operational implications of EOP warnings, cautions and notes.
WE03EK2.2	LOCA Cooldown - Depress. / 4	3.7	4.0	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	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.
WE08EK3.3	RCS Overcooling - PTS / 4	3.7	3.8	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Manipulation of controls required to obtain desired operating results during abnormal and emergency situations.

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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												
003K6.02	Reactor Coolant Pump	2.7	3.1	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	RCP seals and seal water supply
004K2.06	Chemical and Volume Control	2.6	2.7	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Control instrumentation
004K6.31	Chemical and Volume Control	3.1	3.5	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Seal injection system and limits on flow range
005A1.05	Residual Heat Removal	3.3	3.3	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Detection of and response to presence of water in RHR emergency sump
006K3.02	Emergency Core Cooling	4.3	4.4	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Fuel
007A1.03	Pressurizer Relief/Quench Tank	2.6	2.7	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Monitoring quench tank temperature
007A4.01	Pressurizer Relief/Quench Tank	2.7	2.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	PRT spray supply valve
008K4.07	Component Cooling Water	2.6	2.7	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Operation of the CCW swing-bus power supply and its associated breakers and controls
010K5.02	Pressurizer Pressure Control	2.6	3.0	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Constant enthalpy expansion through a valve
012G2.4.1	Reactor Protection	4.6	4.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of EOP entry conditions and immediate action steps.
012K3.01	Reactor Protection	3.9	4.0	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	CRDS

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
013K2.01	Engineered Safety Features Actuation	3.6	3.8	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	ESFAS/safeguards equipment control
013K5.02	Engineered Safety Features Actuation	2.9	3.3	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Safety system logic and reliability
022A3.01	Containment Cooling	4.1	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Initiation of safeguards mode of operation
026A4.05	Containment Spray	3.5	3.5	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	Containment spray reset switches
039A3.02	Main and Reheat Steam	3.1	3.5	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Isolation of the MRSS
039K4.06	Main and Reheat Steam	3.3	3.6	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Prevent reverse steam flow on steam line break
059A2.12	Main Feedwater	3.1	3.4	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Failure of feedwater regulating valves
061K3.02	Auxiliary/Emergency Feedwater	4.2	4.4	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	S/G
062A1.01	AC Electrical Distribution	3.4	3.8	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Significance of D/G load limits
062G2.4.45	AC Electrical Distribution	4.1	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to prioritize and interpret the significance of each annunciator or alarm.
063K1.02	DC Electrical Distribution	2.7	3.2	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	AC electrical system

T2G1 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
064K1.04	Emergency Diesel Generator	3.6	3.9	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	DC distribution system
073K5.02	Process Radiation Monitoring	2.5	3.1	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Radiation intensity changes with source distance
076K2.04	Service Water	2.5	2.6	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Reactor building closed cooling water
078A4.01	Instrument Air	3.1	3.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	Pressure gauges
103A2 03	Containment	3.5	3.8	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Phase A and B isolation
103A2.05	Containment	2.9	3.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Emergency containment entry

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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												
002K6.07	Reactor Coolant	2.5	2.8	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Pumps
015K2.01	Nuclear Instrumentation	3.3	3.7	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	NIS channels, components and interconnections
016K5.01	Non-nuclear Instrumentation	2.7	2.8	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Separation of control and protection circuits
033A1.02	Spent Fuel Pool Cooling	2.8	3.3	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Radiation monitoring systems
034A4.01	Fuel Handling Equipment	3.3	3.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	Radiation levels
035A3.02	Steam Generator	3.7	3.5	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	MAD valves
071G2.1.30	Waste Gas Disposal	4.4	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to locate and operate components, including local controls.
072K4.01	Area Radiation Monitoring	3.3	3.6	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Containment ventilation isolation
079K1.01	Station Air	3.0	3.1	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	IAS
086A2.04	Fire Protection	3.3	3.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Failure to actuate the FPS when required, resulting in fire damage

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
G2.1.2	Conduct of operations	4.1	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of operator responsibilities during all modes of plant operation.
G2.1.36	Conduct of operations	3.0	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of procedures and limitations involved in core alterations
G2.1.45	Conduct of operations	4.3	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to identify and interpret diverse indications to validate the response of another indication
G2.2.18	Equipment Control	2.6	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process for managing maintenance activities during shutdown operations.
G2.2.35	Equipment Control	3.6	4.5	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to determine Technical Specification Mode of Operation
G2.2.38	Equipment Control	3.6	4.5	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of conditions and limitations in the facility license.
G2.3.4	Radiation Control	3.2	3.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of radiation exposure limits under normal and emergency conditions
G2.4.11	Emergency Procedures/Plans	4.0	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of abnormal condition procedures.
G2.4.25	Emergency Procedures/Plans	3.3	3.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of fire protection procedures.
G2.4.31	Emergency Procedures/Plans	4.2	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of annunciators alarms, indications or response procedures

SRO T1G1 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
008AG2.4.2	Pressurizer Vapor Space Accident / 3	4.5	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.
054AG2.4.35	Loss of Main Feedwater / 4	3.8	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects
055EA2.04	Station Blackout / 6	3.7	4.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Instruments and controls operable with only dc battery power available
065AA2.04	Loss of Instrument Air / 8	2.2	2.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Typical conditions which could cause a compressor trip (e.g. high temperature)
we04EG2.4.3	LOCA Outside Containment / 3	2.7	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of events related to system operations/status that must be reported to internal orginizations or outside agencies.
WE05EA2.2	Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	3.7	4.3	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

ES-401, REV 11 **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												
001AG2.2.44	Continuous Rod Withdrawal / 1	4.2	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions
033AG2.4.46	Loss of Intermediate Range NI / 7	4.2	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to verify that the alarms are consistent with the plant conditions.
060AA2.05	Accidental Gaseous Radwaste Rel. / 9	3.7	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	That the automatic safety actions have occurred as a result of a high ARM system signal
074EA2.07	Inad. Core Cooling / 4	4.1	4.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	The difference between a LOCA and inadequate core cooling from trends and indicators

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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												
004G2.4.35	Chemical and Volume Control	3.8	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects
006G2.2.36	Emergency Core Cooling	3.1	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions of operations
013A2.04	Engineered Safety Features Actuation	3.6	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Loss of instrument bus
059A2.06	Main Feedwater	2.7	2.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Loss of steam flow to MFW system
064A2.16	Emergency Diesel Generator	3.3	3.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Loss of offsite power during full-load testing of ED/G

SRO T2G2 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
002A2.04	Reactor Coolant	4.3	4.6	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Loss of heat sinks
011G2.4.50	Pressurizer Level Control	4.2	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.
041A2.02	Steam Dump/Turbine Bypass Control	3.6	3.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Steam valve stuck open

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												
G2.1.35	Conduct of operations	2.2	3.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the fuel handling responsibilities of SRO's
G2.1.43	Conduct of operations	4.1	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to use procedures to determine the effects on reactivity of plant changes
G2.2.19	Equipment Control	2.3	3.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of maintenance work order requirements.
G2.2.22	Equipment Control	4.0	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of limiting conditions for operations and safety limits.
G2.3.15	Radiation Control	2.9	3.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of radiation monitoring systems
G2.4.11	Emergency Procedures/Plans	4.0	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of abnormal condition procedures.
G2.4.23	Emergency Procedures/Plans	3.4	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

Facility: Harris Nuclear Plant

Date of Examination: March 5, 2018

Examination Level: RO ☒ SRO ☐

Operating Test Number: 05000400/2018301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, P, R	Perform a manual Shutdown Margin Calculation (OST-1036) (JPM ADM-019-d) *K/A G 2.1.25* **2018 NRC RO A1-1**
Conduct of Operations	M, R	Determine Rod Misalignment Using Thermocouples (AOP-001) (JPM ADM-062-b) *K/A G 2.1.7* **2018 NRC RO A1-2**
Equipment Control	M, R	Perform a Quadrant Power Tilt Ratio (QPTR) calculation to determine control rod misalignment (OST-1039) (JPM ADM-010-i) *K/A G 2.2.12* **2018 NRC RO A2**
Radiation Control	D, R	Using Valve Maps And Survey Maps Determine Stay Times For A Clearance (PD-RP-ALL-0001) (JPM ADM-051-d) *K/A G2.3.4* **2018 NRC RO A3**
Emergency Plan	N/A	NOT SELECTED FOR RO **2018 NRC RO 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	**(4)**
	(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)	**(2)**
	(N)ew or (M)odified from bank (≥ 1)	**(2)**
	(P)revious 2 exams (≤ 1; randomly selected)	**(1)**

2018 NRC RO Admin JPM Summary

2018 NRC RO A1-1 - Perform a manual Shutdown Margin Calculation (OST-1036) (JPM ADM-019-c) **PREVIOUS** from the 2014 Exam. (Randomly selected from the Admin JPM bank)

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 plant is operating at 92% power and the CRS will direct the candidate to complete OST-1036, Shutdown Margin Calculation Modes 1-5, Section 7.3, for the current plant conditions.

2018 NRC RO A1-2 - Determine Rod Misalignment Using Thermocouples (AOP-001) (JPM ADM-020-b-SRO) **MODIFIED**

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 95% power with a load reduction in progress when a control rod is observed indicating 24 steps higher than group demand. The candidate must perform Attachment 2 of AOP-001, Malfunction of Rod Control and Indication System, to calculate the temperature difference between the affected thermocouple and its symmetric thermocouples. .

NOTE: Modified because the thermocouples A08, E10, F03, G01, K11, L14, have been returned to operable status. Additionally a different control rod was selected and the thermocouple indications were modified to result in a different final value.

2018 NRC RO A2 - Perform a Quadrant Power Tilt Ratio (QPTR) calculation to determine control rod misalignment (OST-1039) (JPM ADM-010-i) **MODIFIED**

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

The candidate must perform a QPTR calculation in accordance with surveillance procedure OST-1039, Calculation of Quadrant power Tilt Ratio, Weekly Interval and as required by the AOP-001, Malfunction of Rod Control and Indication System for a misaligned rod at 90% power. The candidate should calculate a QPTR value between 1.02 and 1.09.

NOTE: Modified due to the Cycle 21 Nuclear Instrument current equivalent reading being significantly changed from the Cycle 19 values. The change in these values result in a QPTR reading that is different from the previous answer.

2018 NRC RO Admin JPM Summary

2018 NRC RO A3 - – Using Valve Maps And Survey Maps Determine Stay Times For A Clearance (PD-RP-ALL-0001) (JPM-ADM-051-d) - **DIRECT**

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

> The candidate will be supplied a survey map of a location in the RAB and a clearance mission to complete in this radioactive area. The location also contains one or more hot spots. They must determine the individual stay times for two Auxiliary Operators (AO) without exceeding the annual administrative dose limits. 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 hanging a clearance on. The given information will supply the accumulated annual whole body doses for the two AOs, one of which recently worked for another utility. They must perform their calculations based on Duke Energy Administrative Dose Limits.

2018 NRC RO A4 – Not selected

Facility: Harris Nuclear Plant

Date of Examination: March 5, 2018

Examination Level: RO ☐ SRO ☒

Operating Test Number: 05000400/2018301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, R	Perform Review of Daily Surveillance Requirements Log (OST-1021) (JPM ADM-014-f-SRO) *K/A G 2.1.18* **2018 NRC SRO A1-1**
Conduct of Operations	M, R	Determine Rod Misalignment Using Thermocouples and Evaluate Technical Specifications (AOP-001) (JPM ADM-062-b-SRO) *K/A G 2.1.7* **2018 NRC SRO A1-2**
Equipment Control	M, R	Perform a Quadrant Power Tilt Ratio (QPTR) calculation to determine control rod misalignment and Evaluate Technical Specifications (OST-1039) (JPM ADM-010-i-SRO) *K/A G 2.2.12* **2018 NRC SRO A2**
Radiation Control	N, R	Complete review and approval of OP-120.07, Attachment 3 Waste Gas Decay Tank Release Log (OP-120.07) (JPM ADM-075-a-SRO) *K/A G2.3.4* **2018 NRC SRO A3**
Emergency Plan	N, R	Classify an Event (EP-EAL) (JPM ADM-076-a-SRO) *K/A G2.4.41* **2018 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)	**(0)**

2018 NRC SRO Admin JPM Summary

2018 NRC SRO A1-1 - Perform Review of Daily Surveillance Requirements Log (OST-1021) (JPM ADM-014-f-SRO) **DIRECT**

K/A G2.1.18 - Ability to make accurate, clear, and concise logs, records, status boards, and reports. (CFR: 41.10 / 45.12 / 45.13) RO 3.6 SRO 3.8

The candidate must perform the CRS review of the control board readings log, identify all errors (4) and determine the Technical Specification application, as necessary.

2018 NRC SRO A1-2 - Determine Rod Misalignment Using Thermocouples and Evaluate Technical Specifications (AOP-001)
(JPM ADM-020-b-SRO) **MODIFIED**

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 95% power with a load reduction in progress when a control rod is observed indicating 24 steps higher than group demand. The candidate must perform Attachment 2 of AOP-001, Malfunction of Rod Control and Indication System, to calculate the temperature difference between the affected thermocouple and its symmetric thermocouples. For this JPM the SRO will need to determine Technical Specification requirements for a failed control rod position indicator.

NOTE: Modified because the thermocouples A08, E10, F03, G01, K11, L14, have been returned to operable status. Additionally a different control rod was selected and the thermocouple indications were modified to result in a different final value.

2018 NRC SRO A2 - Perform a Quadrant Power Tilt Ratio (QPTR) calculation to determine control rod misalignment and Evaluate Technical Specifications . (OST-1039)
(JPM ADM-010-i-SRO) **MODIFIED**

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

The candidate must perform a QPTR calculation in accordance with surveillance procedure OST-1039, Calculation of Quadrant power Tilt Ratio, Weekly Interval and as required by the AOP-001, Malfunction of Rod Control and Indication System for a misaligned rod at 90% power. The candidate should calculate a QPTR value between 1.02 and 1.09.

NOTE: Modified due to the Cycle 21 Nuclear Instrument current equivalent reading being significantly changed from the Cycle 19 values. The change in these values results in a QPTR reading that is different from the previous answer. The change in this reading will require the candidate to apply a different action statement from the Technical Specification when compared to the previous JPM.

2018 NRC SRO Admin JPM Summary

2018 NRC SRO A3 - Complete review and approval of OP-120.07, Attachment 3 Waste Gas Decay Tank Release Log
(OP-120.07) (JPM-ADM-075-a-SRO) **NEW**

K/A G 2.3.6 Ability to approve release permits.

(CFR: 41.13 / 43.4 / 45.10) RO 2.0 SRO 3.8

The candidate will be provided with the pre-release data provided by the RWCR AO and Shift Chemistry Technician along with the completed OP-120.07 Attachment 3 for authorization to commence the release of a Waste Gas Decay tank. They must determine that three items (Vent Stack 5 flow Rate, Sample Request Date/Time and RCDT Vent position) dispositions are not correct and the release should not be approved to commence.

2018 NRC SRO A4 - Classify an Event (EP-EAL) (JPM-ADM-076-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:	Shearon Harris	Date of Examination:	March 5, 2018
Exam Level: RO ☒ SRO-I ☒ **SRO-U (Bold)** ☒		Operating Test Number:	05000400/2018301

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.	**Initiate Emergency Boration Following a Reactor Trip (AOP-002) (JPM-CR-037-f)** ***K/A APE024 AA1.17***	**A, L, M, S**	**1**
b.	**Manually Load Safeguards Equipment On AC Emergency Buses After a LOSP event (EOP-ECA-0.2) (JPM-CR-301-a)** ***K/A 006 A4.04***	**A, EN, L, N, 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, S	3
d.	Initiate RCS Feed and Bleed (EOP-FR-H.1) (JPM-CR-068-d) *K/A EPE E05 EA1.1*	A, D, L, S	4P
e.	Perform Containment Ventilation Isolation Valve ISI Test (OST-1056) (JPM-CR-288-b) *K/A 028 A4.01*	D, EN, S	5
f.	**Restoration of Offsite Power to Emergency Buses (EOP ECA-0.0) (JPM-CR-291-b)** ***K/A 055 EA1.07***	**A, D, P, S**	**6**
g.	Take an Excore NI Channel Out Of Service at Power (OWP-RP-26) (JPM CR-019-c) **RO Only** *K/A 015 A4.03*	D, S	7
h.	Respond to an Instrument Air Header Rupture at 50% power (AOP-017) (JPM-CR-234-d) *K/A APE 065 AA2.06*	D, S	8

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for **SRO-U**		
i. **Manually isolate the SG "B" PORV and SHUT the SG "B" TDAFW Pump steam supply MOV (AOP-016) (JPM IP-257-b)** ***K/A 037 AAG2.1.30***	**D, E, L**	**3**
j. **Reset TD AFW Pump Mechanical Overspeed (OP-137) (JPM-IP-001-c)** ***K/A 061 A2.04***	**D, E, L, P, R**	**4S**
k. Perform Local Actions for Placing an OT∆T Channel in Test (OWP-RP-01) (JPM IP-209-d) *K/A 012 A4.04*	D, E	7

* 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	(9, 8, 3)	
(E)mergency or abnormal in-plant	≥ 1/≥ 1/≥ 1	(3, 3, 2)	
(EN)gineered safety feature	≥ 1/≥ 1/≥ 1	(2, 2, 1)	(control room system)
(L)ow-Power/Shutdown	≥ 1/≥ 1/≥ 1	(5, 5, 4)	
(N)ew or (M)odified from bank including 1(A)	≥ 2/≥ 2/≥ 1	(2, 2, 2)	
(P)revious 2 exams	≤ 3/≤ 3/≤ 2	(2, 2, 2)	(randomly selected)
(R)CA	≥ 1/≥ 1/≥ 1	(1, 1, 1)	
(S)imulator			

2018 NRC Control Room/In-Plant JPM Summary

Simulator JPMs

JPM a – Initiate Emergency Boration following a Reactor Trip (AOP-002) (JPM-CR-037-f) SRO Upgrade - Alternate Path - Modified

K/A APE024 AA1.17 Ability to operate and / or monitor the following as they apply to Emergency Boration: Emergency borate control valve and indicators
(CFR 41.7 / 45.5 / 45.6) RO 3.9 SRO 3.9

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

Turnover: The plant was operating at 100% power when automatic reactor trip occurred due to an inadvertent main turbine trip. The crew has just completed the immediate actions of EOP E-0, Reactor Trip or Safety Injection and have transitioned to ES-0.1, Reactor Trip Response

Task: Initiate Emergency Boration following a Reactor Trip.

Verifiable actions: The candidate will attempt to start a Boric Acid pump and determine that the pump has tripped. The pump failure will require the candidate to establish an alternate flow path by opening either of the two RWST valves to the suction of the CSIP, shutting either of the two VCT outlet valves and then raise flow to > 90 gpm using a FCV with the flow rate indication on a meter on the MCB.

Alternate Path – YES. The only available Boric Acid Pump will fail when started requiring the candidate to utilize an alternate boration flow path and also establish a flow rate to the CSIP of > 90 gpm using FK-122.1 in manual.

JPM completion: After the candidate has established and verified at least 90 gpm charging flow from the RWST to the RCS on FI-122A.1, evaluation on this JPM is complete.

Modification: Modified by placing one Boric Acid pump out of service and failing the remaining pump so that no boric acid pumps are available. This change requires the candidate to complete step 1 then step 6 (boration from the RWST) of AOP-002, vice step 2 through 5 (boration from the Boric Acid Storage Tank).

JPM b – Manually Load Safeguards Equipment On AC Emergency Buses After a LOSP event (EOP-ECA-0.2) (JPM-CR-301-a) SRO Upgrade - Alternate Path - New

K/A 006 A4.04 Ability to manually operate and/or monitor in the control room: RHRS
(CFR: 41.7 / 45.5 to 45.8) RO 3.7 SRO 3.6

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

Turnover: The plant was operating at 100% power when a LOCA occurred. As a result of the LOCA a Reactor Trip / Safety Injection have been actuated. Offsite Power was lost during the Reactor Trip and both Diesel Generators failed to start.
EOP-ECA-0.0, Loss of All AC Power was entered and offsite power was restored to both Emergency Busses. The crew has transitioned to EOP-ECA-0.2, Loss of All AC Power Recovery With SI Required. Steps 1-3 have been completed.

Simulator JPMs (continued)

JPM b (continued)

Turnover: (Continued) The CRS has directed you to continue EOP-ECA-0.2 starting at step 4 in preparation to Manually Load Safeguards Equipment On AC Emergency Buses.

Task: Manually Load Safeguards Equipment On AC Emergency Buses After A LOSP.

Verifiable actions: The candidate will be required to perform EOP-ECA-0.2 steps 4 and 5.a-e which will check the status of CCW flow to the RCP Thermal Barrier Hx to isolate CCW to the RCP Seals prior to starting the CCW pumps. Once CCW to the RCP Thermal barrier Hx is isolated the OATC will coordinate the restoration of control power to the CCW pumps with an AO locally in the switchgear. The first CCW pump will automatically start on low pressure once the control switch is returned to the auto position. With control power restored to the CCW pumps the OATC will start the standby CCW pump and both RHR pumps.

Alternate Path – YES. When checking the status of the RCP Thermal Barrier Hx CCW will not be isolated to the heat exchanger. The outside containment isolation valve 1CC-251 will be open when checked in step 4.d. This will require the candidate to implement an alternate method of isolating the heat exchanger. Attempts to shut the 1CC-251 from the MCB or locally will fail. This will require shutting the inside Containment isolation valve 1CC-249 from the MCB.

JPM completion: When Both RHR pumps and two CCW pumps are running the SRO will notify the OATC that the task is complete. Another operator will align Containment Fan coolers and continue implementing the procedure.

JPM c – Take Corrective Action For Failure of CSIP Mini-Flow Valves to Re Position (EOP-E-0) (JPM-CR-225-e) - Direct

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 an automatic Reactor Trip occurred due to a spurious Safety Injection signal. The crews is performing EOP-E-0, Reactor Trip or Safety Injection and are 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.

Simulator JPMs (continued)

JPM c – continued

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.

JPM d – Initiate RCS Feed and Bleed (EOP-FR-H.1)
(JPM-CR-068-d) – Direct

K/A EPE E05 EA1.1 Ability to operate and / or monitor the following as they apply to the (Loss of Secondary Heat Sink): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
(CFR: 41.7 / 45.5 / 45.6) RO 4.1 SRO 4.0

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

Turnover: The plant status is 'A' MDAFW pump is under clearance, the Reactor tripped from 100% power due to a loss of off-site power followed by a Small Break LOCA. Emergency Bus 1B-SB locked out on an electrical fault and the TDAFW pump failed when it started. The crew is performing EOP-FR-H.1, Response To Loss Of Secondary Heat Sink. The foldout criteria for initiation of RCS Feed and Bleed have just been met.

Task: Perform the actions to initiate RCS feed and bleed.

Verifiable actions: The candidate will be required to locate and operate the SI, and Phase A RESET switches, control switches for 1IA-819, 1SI-287, 1RC-900, 1RC-902 and 1RC-904 while monitoring progress using MCB indications.

Alternate Path – YES. The PRZ PORVs will not open when the control switches are operated on the MCB. The candidate will be required to verify adequate RCS bleed path by implementing the RNO action and open the RCS vents with power available.

JPM completion: When the RCS Vent Valves with power available are OPENED and the CRS has been informed that RCS Feed and bleed has been established, evaluation on this JPM is complete.

JPM e – Perform Containment Ventilation Isolation Valve ISI Test (OST-1056)
(JPM-CR-288-b) – Direct

K/A 028 A4.01 - Ability to manually operate and/or monitor in the control room: HRPS controls
(CFR: 41.7 / 45.5 to 45.8) RO 4.0 SRO 4.0

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

Turnover: The plant is at 100% power steady state Middle of Life (MOL). OST-1056, Containment Ventilation Isolation Valve ISI Test Quarterly Interval MODE1-6 is being performed to test the operability of the Containment ventilation isolation valves per the ISI program. The Airborne Radioactive Removal & Normal Purge Systems were shutdown in accordance with OP-168, Containment Ventilation And Vacuum Relief. The CRS has directed the BOP to continue OST-1056 at Section 7.2 step 2.

Simulator JPMs (continued)

JPM e (continued)

Task: Perform stroke time tests using a stop watch on four pairs of Containment Isolation valves. Timing of each valve is then documented and compared to acceptance criteria.

Verifiable actions: The candidate will be required to perform stroke timing of containment ventilation valves and document the results on Attachment 2 of OST-1056

Alternate Path – NO. There are no failures with this task

JPM completion: When OST-1056, Section 7.2 and documentation of timings on Attachment 2 are complete and the CRS is informed, evaluation on this JPM is complete.

JPM f – Restoration of Offsite Power to Emergency Buses (EOP-ECA-0.0) (JPM-CR-291-b) **SRO Upgrade - Alternate Path – Previous** from the 2016 Exam. (Randomly selected from the Simulator JPM bank)

K/A 055 EA1.07 Ability to operate and monitor as they apply to station blackout: Restoration of power from offsite
(CFR: 41.7 / 45.5 / 45.6) RO 4.3 SRO 4.5

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

Turnover: The plant was operating at 100% power. ‘A’ EDG is under clearance due to the generator field not flashing during OST-1013. The failure of a major line on the Duke grid resulted in the cascading trip of several units and low grid frequency. A loss of offsite power occurred. ‘B’ EDG failed to start and the problem is being investigated. The crew is implementing EOP-ECA-0.0. The load dispatcher has stabilized the grid and has given permission to restore offsite power to 6.9 KV buses and to reset any tripped Start Up XFMR lockout relays (there are currently no lockout relays tripped).

Task: Restore offsite power to a (one) AC emergency bus using EOP ECA-0.0, Attachment 1.

Verifiable actions: The candidate will be manipulating electrical supply breaker switches on the MCB to restore power to the dead Emergency Bus.

Alternate Path – YES - During the lineup for power restoration on the A-SA Emergency Bus the supply breaker from offsite (Breaker 105) will fail to close. The candidate will be required to continue Attachment 1 using the guidance for the B-SB Emergency Bus to complete restoration of offsite power to a (one) AC emergency bus.

JPM completion: Emergency Bus 1B-SB is being powered via offsite power and the 480 V breakers powering emergency equipment is energized and the CRS is informed, evaluation on this JPM is complete.

Simulator JPMs (continued)

JPM g – Place an Excore NI Channel Out Of Service at Power (OWP-RP-26)
(JPM CR-019-c) RO Only – Direct

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

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

Turnover: Prior to taking watch, with the unit is operating at 100% power steady state conditions, Nuclear Instrument 44 has failed low. The CRS has directed the candidate to remove NI-44 from service in accordance with OWP-RP-26, Reactor Protection.

Task: NI-44 removed from service.

Verifiable actions: The candidate will be required to place rod control to manual. The candidate will then remove the detector from service at the detector current comparator drawer, the miscellaneous control and indication panel, and the comparator and rate drawer. The candidate will then contact I&C to lift leads from the circuit. They will then check the bi-stable status panels for proper responses. The candidate will also have to remove the channel from scan on the ERFIS computer.

Alternate Path – NO. There are no failures with this task.

JPM completion: When N44 has been removed from service in accordance with OWP-RP-26 and the CRS is informed, evaluation on this JPM is complete.

JPM h – Respond to a Rupture in the Instrument Air Header at 50% power (AOP-017)
(JPM-CR-234-d)

K/A APE 065 AA2.06 Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is decreasing (CFR: 43.5 / 45.13) RO 3.6 SRO 4.2

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

Turnover: The unit is operating at ~50% power. Soon after taking the watch an Instrument Air leak will develop. The candidate will be expected to respond to the low pressure annunciators and enter AOP-017.

Task: Trips the Reactor, carries out immediate actions of EOP-E-0, and then continues the actions directed by AOP-017 for low air pressure

Verifiable actions: The candidate will be expected to manually trip the Reactor perform the immediate actions of EOP-E-0 then be directed to continue with AOP-017. They will have to contact Auxiliary Operators to vent and depressurize the remaining air from the system. Continuing with the procedure requires the candidate to locate and place multiple MCB controls to manual and zero demand.

Alternate Path – NO. There are no additional failures with this task.

JPM completion: When the candidate completes AOP-017, Attachment 2 and CRS is informed, evaluation on this JPM is complete.

2018 NRC Control Room/In-Plant JPM Summary

In-Plant JPMs

JPM i – Manually isolate the SG "C" PORV and SHUT the SG "C" TDAFW Pump steam supply MOV (AOP-016)
(JPM IP-257-b) SRO Upgrade - Direct

K/A 063 G2.1.30 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: A shutdown was in progress due to escalating tube leakage in SG 'C' when a loss of off-site power caused a reactor trip. The crew is performing AOP-016, Excessive Primary Leakage, Attachment 11, Plant Shutdown Actions for Primary-To-Secondary Leakage Action Level 2 and 3 in parallel with EOP-E-0, Reactor Trip Or Safety Injection. The CRS has directed you to perform AOP-016, Attachment 11, Step 11, for SG 'C'.

Task: SG PORV manual isolation closed and MS-72 closed manually.

Verifiable actions: The candidate will be required to perform local actions for AOP-016, Attachment 11, Step 11 RNO. The JPM cues include information of the proper status of the Valve operator and the expected candidate actions.

Alternate Path – NO. There are no additional failures with this task.

JPM completion: When 1MS-63 and 1MS-72 are closed and the MCR is informed, evaluation on this JPM is complete.

JPM j – Reset the Turbine Driven AFW Pump Mechanical Overspeed (OP-137)
(JPM-IP-001-c) SRO Upgrade - Direct – Previous from the 2014 Exam. (Randomly selected from the In-Plant JPM bank)

K/A 061 A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: pump failure or improper operation

(CFR: 41.5 / 43.5 / 45.3 / 45.13) RO 3.4 SRO 3.8

NOTE: This JPM is inside the RCA.

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

Turnover: The unit has tripped from 100% power. The Turbine Driven AFW pump started and has tripped on overspeed. The pump is needed for plant cooldown efforts. The cause of the overspeed trip has been identified and corrected by Maintenance. The CRS has directed the candidate to reset the Turbine Driven AFW mechanical overspeed trip linkage. 1MS-70 and 1MS-72 (steam supply valves to the TDAFW pump) are indicating shut from the MCB. The CRS also notifies the candidate that the Trip and Throttle Valve will be reopened from the Control Room.

Task: The Turbine-driven AFW pump turbine trip and throttle valve is latched.

In-Plant JPMs (continued)

JPM j (continued)

Verifiable actions: The candidate will be required to align the Aux Feedwater Overspeed Trip mechanism with the tappet nut correctly oriented and the connecting rod locked in the latched position. Additionally the candidate must identify the local indication for the Turbine Overspeed Trip status.

Alternate Path – NO. There are no additional failures with this task.

JPM completion: When the mechanical overspeed linkage is reset and the MCR is informed the Trip and Throttle valve maybe opened from the MCB, evaluation on this JPM is complete.

JPM k – Perform Local Actions for Placing an OTΔT Channel in Test (OWP-RP-01)
(JPM IP-209-d) Direct

K/A 012 A4.04 Ability to manually operate and/or monitor in the control room: Bi-stable, trips, reset and test switches
(CFR: 41.7 / 45.5 to 45.8) RO 3.3 SRO 3.3

Evaluated position: Reactor Operator in the Reactor Auxiliary Building (RO RAB)

Turnover: The unit is operating at 100% power when Loop 1 Hot Leg temperature input to T_{avg} and OTΔT failed low. To comply with Technical Specifications, the CRS is directing you to perform the local actions of OWP-RP-01 for troubleshooting and tripping bi-stables for Loop 1 T_{avg} and OTΔT. Inform the Control Room when all switches have been positioned to allow the Control Room to complete the actions required in the Control Room.

Task: Place the PIC Cabinet Master Test switches and bi-stables in the Test position..

Verifiable actions: The candidate will be required to reposition multiple test switches on PIC card within the PIC 1 cabinet on the RAB 304’ elevation. The candidate will be required to identify the individual PIC card and test switch and operate the toggle switch. The candidate will be provided a copy of OWP-RP-01, to complete the task.

Alternate Path – NO. There are no additional failures with this task.

JPM completion: When the required switches in PIC 1 have been placed in the TEST position and the MCR is informed, evaluation on this JPM is complete.

2018 NRC RO 001/BANK/C/A//ES-0.1/NONE/2009A NRC RO 01/007 EK2.03/

Given the following plant conditions:

- A Reactor trip has just occurred from full power due to a loss of Offsite Power
- The crew has entered EOP-ES-0.1, Reactor Trip Response

The BOP is directed to control temperature in accordance with EOP-ES-0.1, Table 1

The following plant conditions currently exist:

- All SG levels are 21% and lowering
- Total AFW flow to the Steam Generators is 208 KPPH
- Loop Low T_{avg} Bistable lights are lit on TSLB 3
- Loop Low-Low T_{avg} Bistable lights are lit on TSLB 3
- Group 1 Condenser Steam Dumps red AND green lights are lit on SLB 1

Which ONE of the following describes the action required in accordance with EOP-ES-0.1, Table 1?

A✓ Close all Main Steam Isolation Valves

B. Place Steam Dumps in the Steam Pressure Mode

C. Raise AFW flow to the SGs to raise SG water levels

D. Reduce AFW flow to the SGs to stop the RCS Cooldown

Plausibility and Answer Analysis

Reason answer is correct: With Loop Low-Low T_{avg} Bistable lights illuminated on TSLB 3 (P-12), the Steam Dumps should be closed and are not which results in an uncontrolled cooldown of the RCS. EOP-ES-0.1 states if the cooldown continues, then shut the MSIV and bypass valves.

A Correct.

B Incorrect. Plausible since EOP-ES-0.1 has guidance to place the Steam Dumps in the Steam Pressure Mode in several locations. The candidate may have the misconception that this will assist with present plant conditions, but this is incorrect because P-12 affects Steam Dumps in both the T_{avg} *and Steam Pressure Modes.*

C Incorrect. Plausible since Steam Generator Water Levels are 21% and lowering so the candidate may believe raising AFW flow to be correct because EOP-ES-0.1 has guidance to maintain SGWL 25 to 50%. However, with RCS Temperature less than 557°F, AFW flow should be greater than 200 KPPH but minimized to prevent further cooldown.

D Incorrect. Plausible since RCS Temperature is less than 557°F, the candidate may have the misconception that reducing AFW flow will slow the cooldown rate because EOP-ES-0.1 has guidance to control RCS temperature at 555°F to 559°F but greater than 200 KPPH is required for heatsink.

000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1

007EK2.03; Knowledge of the interrelations between a reactor trip and the following:
Reactor trip status panel

(CFR 41.7 / 45.7)

Importance Rating:	3.5 3.6
Technical Reference:	EOP-ES-0.1, Step 4, Pg 6, Rev. 3
References to be provided:	None
Learning Objective:	EOP-LP-3.22, Obj. 3.d
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 002/BANK/C/A//E-1, ES-1.2,FR-H.1/NONE/2013 NRC RO 03/009 EK2.03/

Given the following plant conditions:
- All RCPs are running
- RCS pressure is 920 psig and slowly lowering
- SI flow is 100 GPM
- Containment pressure is 3.2 psig and slowly rising
- SG pressures are 1120 psig and stable

Which ONE of the following completes the statements below, in accordance with EOP-E-1, Loss of Reactor Or Secondary Coolant?

RCP trip criteria ___(1)___ met.

___(2)___ is the MINIMUM pressure above which the crew will transition to EOP-ES-1.2, Post LOCA Cooldown and Depressurization, where the SGs will be required for RCS cooldown.

A. (1) is

(2) 230 psig

B. (1) is

(2) 400 psig

C.✓ (1) is NOT

(2) 230 psig

D. (1) is NOT

(2) 400 psig

Plausibility and Answer Analysis

Reason answer is correct: *A Small Break LOCA is in progress. EOP-E-0, Reactor Trip or Safety Injection foldout for RCP trip criteria is NOT met with RCS pressure < 1400 psig and SI flow < 200 gpm. Therefore the RCPs should NOT be tripped. A procedure transition from EOP-E-0 to EOP-E-1 to EOP-ES-1.2 will take place for this event. In order to transistion to EOP-ES-1.2, Post LOCA Cooldown, the crew will evaluate if the break is large enough to allow the low pressure high volume RHR system to provide core cooling or if the Steam generators are required for cooling. If RCS pressure is less than 230 psig and RHR flow is greater than 1000 gpm the crew will remain in EOP-E-1 and initiate a RCS cooldown to CSD with SG PORV's. This method would be used instead of using the Steam Dumps since the MSIV's would be shut due to high Containment pressure. In this situtuation the RCS pressure is still greater than the pressure required to place RHR in service therefore the SG's would be required for subsequent heat removal until the RHR system is placed in service in the shutdown cooling mode.*

A. Incorrect. The first part is plausible because the RCS pressure is below the value (<1400 psig) in which the RCPs would be NOT be required if adequate SI flow (>200 gpm) was injecting into the core. The second part is correct.

B. Incorrect. The first part is plausible because the RCS pressure is below the value (<1400 psig) in which the RCPs would be NOT be required if adequate SI flow (>200 gpm) was injecting into the core, however with only 100 gpm flow from the SI system the RCPs are required to remain in operation. The second part is plausible because 400 psig is the lift setpoint for PRZ PORV's when in LTOP configuration.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible because 400 psig is the lift setpoint for PRZ PORV's when in LTOP configuration.

000009 009 Small Break LOCA / 3

009EK2.03; Knowledge of the interrelations between the small break LOCA and the following: S/Gs

(CFR 41.7 / 45.7)

Importance Rating:	3.0 3.3
Technical Reference:	EOP-E-1, Foldout, Pg 3, Rev. 4 EOP-E-1, step 13, RNO
References to be provided:	None
Learning Objective:	EOP-LP-3-11, Obj. 4.e
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 003/NEW/C/A//AOP-018/NONE//015 AG2.1.23/

Given the following plant conditions:

- A Reactor startup is progress with the unit in Mode 2
- Annunciator ALB-008, Window 3-3, RCP-A Seal #1 Leakoff High Low Flow, has alarmed

Subsequently:

- The BOP reports that RCP 'A' #1 seal leakoff is approximately 8.5 gpm and rising

Which ONE of the following sequence of actions is the OATC required to perform for these conditions in accordance with AOP-018, Reactor Coolant Pump Abnormal Conditions?

A. Trip the Reactor, secure RCP 'A' and then Go To EOP-E-0, perform the immediate actions

B.✓ Trip the Reactor, Go To EOP-E-0, perform the immediate actions and then secure RCP 'A' as time permits

C. Secure RCP 'A', shut 1CS-355, RCP 'A' #1 Seal Water Return and then shut 1RC-107 PRZ Spray Loop A valve

D. Secure RCP 'A', wait 3 to 5 minutes then shut 1RC-107 PRZ Spray Loop A valve and then shut 1CS-355, RCP 'A' #1 Seal Water Return

Plausibility and Answer Analysis

Reason answer is correct: RCP 'A' seal has failed as evidenced by the magnitude of seal leakoff ($\geq$ 8 gpm) and continuing degrading conditions. The OATC will be required to trip the Reactor, GO TO EOP-E-0 (to perform the immediate actions) and then perform steps 3-9 of AOP-018 section 3.3 when time permits. AOP-018 Section 3.3 step 3 stops the affected RCP.

A. Incorrect. This is plausible because these are the correct actions but not in the correct order.

B. Correct.

C. Incorrect. Plausible since this would be correct if the Reactor trip breakers were open and if the operator waited for 3 - 5 minutes after securing the RCP. Since the Shutdown banks are withdrawn the Reactor trip breakers are closed. Therefore the Reactor must be tripped prior to securing the affected RCP.

D. Incorrect. Plausible since this would be the correct actions if the Reactor trip breakers were open, but in the wrong sequence. Since the Shutdown banks are withdrawn the Reactor trip breakers are closed. Therefore the Reactor must be tripped prior to securing the affected RCP.

000015 Reactor Coolant Pump (RCP) Malfunctions / 4

015AG2.1.23; 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:	AOP-018, Section 3.3, Pg 10, Rev. 49
References to be provided:	None
Learning Objective:	AOP-LP-3.18, Obj. 3
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 004/BANK/C/A//AOP-020/NONE/2012 NRC RO 04/025 AA2.07/

Given the following plant conditions:
- The unit is in Mode 5
- RHR 'A' and 'B' trains are in service for Shutdown Cooling

Subsequently:
- The OATC reports FI-605A1, RHR Hx 'A' Header Flow AND PDI-5450A, RHR Pump 'A' Diff Pressures are oscillating

Which ONE of the following completes the statement below for actions required to be taken in accordance with AOP-020, Loss of RCS Inventory or Residual Heat Removal while Shutdown?

Stop ___(1)___ AND shut ___(2)___ .

A. (1) BOTH RHR Pumps

(2) 1RH-2, RCS Loop 'A' to RHR Pump 'A' ONLY

B.✓ (1) BOTH RHR Pumps

(2) 1RH-2, RCS Loop 'A' to RHR Pump 'A' AND 1RH-40, RCS Loop 'C' to RHR Pump 'B'

C. (1) RHR Pump 'A' ONLY

(2) 1RH-2, RCS Loop 'A' to RHR Pump 'A' ONLY

D. (1) RHR Pump 'A' ONLY

(2) 1RH-2, RCS Loop 'A' to RHR Pump 'A' AND 1RH-40, RCS Loop 'C' to RHR Pump 'B

Plausibility and Answer Analysis

Reason answer is correct: AOP-020 requires all running RHR pumps to have normal operating indications and RPV level above 82" below the flange, if not both RHR pumps are stopped and isolated from the RCS by at least one isolation valve

A. Incorrect. The first part is correct. The second part is plausible since stopping both RHR pumps prevents damage to the non cavitating pump while isolation of the cavitating pump maintains the loop suction filled and vented.

B. Correct.

C. Incorrect. The first part is plausible since only the RHR pump 'A' is demonstrating indications of pump cavitation, however this is incorrect because AOP-020 requires both RHR pumps to be secured. The second part is plausible since isolation of the affected RHR pump is required to be shut in order to maintain the loop suction filled and vented, however this is incorrect because AOP-020 requires both RHR loops to be isolated.

D. Incorrect. The first part is plausible since only the RHR pump 'A' is demonstrating indications of pump cavitation, however this is incorrect because AOP-020 requires both RHR pumps to be secured. The second part is plausible since an RHR pump is capable of operating on its recirculation flow path with the discharge valve shut, candidate may apply this knowledge to the loop suction valve.

000025 Loss of Residual Heat Removal System / 4

025AA2.07; Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Pump cavitation

(CFR: 43.5 / 45.13)

Importance Rating:	3.4 3.7
Technical Reference:	AOP-020, Section 3.1, Step 1 RNO, Pg 5, Rev 38
References to be provided:	None
Learning Objective:	AOP-LP-3-20, Obj. 3.b and 3.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 005/NEW/C/A//AOP-014/NONE//026 AA1.02/

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

At 1000 a loss of Component Cooling Water occurred
- The crew has entered AOP-014, Loss of Component Cooling Water, due to a loss of all CCW flow to both trains
- RCP Seal Injection flow to each RCP is approximately 9 gpm
- RCP Temperatures are as listed below:
  - 'A' RCP Motor bearing is 142° and continuing to rise at 4°F/minute
  - 'B' RCP Radial bearing is 176° and continuing to rise at 7°F/minute
  - 'C' RCP Stator Winding is 253°F and continuing to rise at 8°F/minute

Of the four times listed below which ONE of the following describes the MAXIMUM time that the unit would be allowed to operate BEFORE a Reactor Trip would be required in accordance with AOP-014?

A. 1002

B✓ 1006

C. 1008

D. 1010

Plausibility and Answer Analysis

Reason answer is correct: The provided heatup rates of the RCPs would require that the Reactor is tripped at 2006 based on the 'C' RCP exceeding the Stator Winding temperature trip limit of 300°F

1002 - 'B' RCP Radial bearing temperture exceeds the Motor bearing temperture trip limit of 190°F

1006 - 'C' RCP Stator Winding temperature rising at 9°F/min starting at 253°F would reach the trip setpoint of 300°F [253° + (8°F x 6 min) = 301°F]

1008 - 'B' RCP Radial Bearing temperature rising at 7°F/min starting at 176°F would reach the trip setpoint of 230°F [176 + (7° x 8 min) = 232°F]

1010 - RCPs have operated for 10 minutes without CCW flow to either motor oil cooler (1000 + 10 min = 1010)

1012 - 'A' RCP Motor Bearing temperature rising at 4°F/min starting at 142°F would reach the trip setpoint of 190°F [142° + (4° x 12 min) = 190°F]

A. Incorrect. Plausible if the student has a misconception that the trip limit for Radial Bearing temperature is 190°F and not 230°F. At 1002 the 'B' RCP Radial bearing temperature exceeded 190°F. Starting temperture of 176°F + (2 min x 7°F) = 190°F

B. Correct.

C. Incorrect. Plausible since at 1008 the 'B' RCP Radial Bearing trip limit of 230°F was exceeded.

D. Incorrect. Plausible since a loss of CCW to an RCP or RCP Motor has operated for 10 minutes without CCW flow to either motor oil cooler

000026 Loss of Component Cooling Water /8

026AA1.02 Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: Loads on the CCWS in the control room

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.2 3.3
Technical Reference:	AOP-014, Attachment 1, Pg 40-41, Rev. 37
References to be provided:	None
Learning Objective:	AOP-LP-3.14 Obj 2.e
Question Origin:	New
Comments:	Discuss AOP-014, Loss Of CCW, actions are based on the change in temperature for the parameters associated with components cooled by CCW and not the CCW Temperature indications. Phonecon 10/23/2017: HNP discussed being unable to create a T1/G1 question based on plant abnormal procedures for the K/A topic of Loss Of Component Cooling Water associated with CCW temperature indications, so selected a new K/A, keeping 026 and determined this item was better tied to a different randomly selected K/A: New K/A 026AA1.02: Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: Loads on the CCWS in the control room.
Tier/Group:	T1/G1

2018 NRC RO 006/BANK/FUNDAMENTAL//TS 2.1.2/NONE//027 AG2.2.22/

Given the following plant conditions:
- The RCS is operating in solid plant conditions

Subsequently:
- A controller failure has caused 1CS-38, Letdown Pressure Control valve, to shut
- The crew has just entered AOP-019, Malfunction of RCS Pressure Control

Which ONE of the following completes the statement below?

In accordance with Technical Specification 2.1.2, Safety Limits - Reactor Coolant System Pressure, RCS pressure shall not exceed ___(1)___ psig. If this is violated, the RCS pressure shall be reduced below the safety limit within a MAXIMUM of ___(2)___ minutes.

A. (1) 2485

(2) 5

B. (1) 2485

(2) 15

[✓ C] (1) 2735

(2) 5

D. (1) 2735

(2) 15

Plausibility and Answer Analysis

Reason answer is correct: In accordance with Technical Specification 2.1.2 the RCS pressure shall not exceed 2735 in Modes 1, 2, 3, 4, and 5 except for hydrostatic testing. The action for exceeding this limit in Modes 3, 4 and 5 is to restore below the limit within 5 minutes and comply with the requirements of specification 6.7.1.

A. Incorrect. The first part is plausible since this is the pressure that the PRZ safeties lift. The second part is correct.

B. Incorrect. The first part is plausible since this is the pressure that the PRZ safeties lift. The second part is plausible since the 15 minute action is correct for restoring T_{avg} *above the minimum temperature for criticality and the candidate may misapply this time limitation to the 5 minute requirement of Technical Specification 2.1.2.*

C. Correct.

D. Incorrect. The first part is correct.The second part is plausible since the 15 minute action is correct for restoring T_{avg} *above the minimum temperature for criticality and the candidate may misapply this time limitation to the 5 minute requirement of Technical Specification 2.1.2.*

000027 Pressurizer Pressure Control System Malfunction / 3

027AG2.2.22; Knowledge of limiting conditions for operations and safety limits.

(CFR: 41.5 / 43.2 / 45.2)

Importance Rating:	4.0 4.7
Technical Reference:	Technical Specification, 2.1.2
References to be provided:	None
Learning Objective:	TS-LP-2.0, Obj. 2
Question Origin:	Bank
Comments:	Discuss whether 1CS-38 can be considered part of the PRZ Pressure control system during solid plant operations with Dan Bacon for question development. Phonecon 10/23: Dan agrees that using 1CS-38 as a component in the PRZ Control system during solid plant operations is acceptable to meet this K/A
Tier/Group:	T1/G1

2018 NRC RO 007/NEW/C/A//AOP-016/NONE//038 EA1.10/

Given the following plant conditions:
- The unit is operating at 100% power
- Charging is in MANUAL at 85 gpm

Subsequently:
- At 0900 Annunciator ALB-010-4-5, Rad Monitor System Trouble, has alarmed
- The crew entered AOP-016, RCS Leakage, and have identified a SG tube leak is in progress

The following plant parameter changes are occurring as follows:

(Rad readings are mR/Hr)	0900	0901	0902	0903	0904
RM-01MS-3591 SB, MSL "A"	2.91E-1	7.67E-1	1.95E-0	3.21E-0	5.98E-0
RM-01MS-3592 SB, MSL "B"	2.22E-1	2.29E-1	3.38E-1	4.45E-1	4.82E-1
RM-01MS-3593 SB, MSL "C"	3.82E-1	3.82E-1	3.82E-1	3.82E-1	3.83E-1
Pressurizer Level	60.0%	59.2%	57.9%	56.2%	53.6%

Which ONE of the following completes the statement below concerning these conditions?

The trends indicate that the SG Tube Rupture is occuring on the ___(1)___ .
The EARLIEST time that a manual Reactor trip is required in accordance with AOP-016 is ___(2)___ .

A. (1) "A" SG Only

(2) 0902

B.✓ (1) "A" SG Only

(2) 0904

C. (1) "A" and "B" SG

(2) 0902

D. (1) "A" and "B" SG

(2) 0904

Plausibility and Answer Analysis

Reason answer is correct: Steam Line Radiation monitor indications on "A" SG have gone up by a factor of 10 over the last 4 minutes indicating that the "A" SG has the SG Tube Rupture. Pressurizer level has lowered > 2% in a one minute interval from 0903 to 0904. In accordance with AOP-016-BD, a drop of 2% or greater per minute in the Pressuizer indicates that makeup capability has been exceeded (Pressurizer has ~55 gal/% level at 653°F).

A. Incorrect. The first part is correct. The second part is plausible since Pressurizer level has dropped 2% since the initation of the event (from 0900 to 0902). But the level drop of 2% or greater must occur in a 1 minute interval. Makeup capability has not yet been exceeded.

B. Correct.

C. Incorrect. Plausible since both "A" SG and "B" SG radiation levels have risen since the initiation of the leak. Based on the magnitude of the radiation level change on the "B" SG the radiation level rise is the result of the "shine" phenomena coming from the "A" SG steam line radiation and is expected. The second part is plausible since Pressurizer level has dropped 2% since the initation of the event (from 0900 to 0902). But the level drop of 2% or greater must occur in a 1 minute interval. Makeup capability has not yet been exceeded.

D. Incorrect. Plausible since both "A" SG and "B" SG radiation levels have risen since the initiation of the leak. Based on the magnitude of the radiation level change on the "B" SG the radiation level rise is the result of the "shine" phenomena coming from the "A" SG steam line radiation and is expected. The second part is plausible since Pressurizer level has dropped 2% since the initation of the event (from 0900 to 0902). The second part is correct.

000038 Steam Generator Tube Rupture / 3

038EA1.10; Ability to operate and monitor the following as they apply to a SGTR: Control room radiation monitoring indicators and alarms

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.7 3.7
Technical Reference:	AOP-016, Section 3.0, Step 4 RNO, Pg 4, Rev 56 AOP-016, Attachment 1, Step 4, Pg 14, Rev 56 AOP-016-BD, Section 3.0, Step 4, Pg 8, Rev 30
References to be provided:	None
Learning Objective:	EOP-LP-3.02 Objective 5
Question Origin:	New
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 008/BANK/C/A//E-0/NONE//054 AK1.01/

Given the following plant conditions:
- The Unit is operating at 75% power

Subsequently:
- Containment pressure is 2.5 psig, and rising
- RCS T_{AVG} is 582°F, and rising
- RCS Pressure is 2195 psig, and lowering
- SG 'A' Steam Flow is 4.0 MPPH, stable
- SG 'A' feed flow is 4.7 MPPH, and rising
- SG 'A' Narrow Range level is 32%, and lowering
- SG 'A' Pressure is 1003 psig, and lowering

The CRS directs the OATC to manually trip the Reactor

Which ONE of the following completes the statements below concering this event?

RCS temperature will ___(1)___ after the Reactor trip.

The crew will respond to a ___(2)___ .

A. (1) lower below no-load T_{AVG} in an uncontrolled manner

(2) Loss Of Reactor Coolant

B✓ (1) lower below no-load T_{AVG} in an uncontrolled manner

(2) Faulted Steam Generator

C. (1) stabilize at no-load T_{AVG} shortly after the Main Feedwater Regulating Valves (FRVs) close

(2) Loss Of Reactor Coolant

D. (1) stabilize at no-load T_{AVG} shortly after the Main Feedwater Regulating Valves (FRVs) close

(2) Faulted Steam Generator

Plausibility and Answer Analysis

Reason answer is correct: A rising Contaiment pressure and SG feedwater flow with a lowering SG level is indicative of a Feedwater rupture. The Feedwater rupture will cause SG pressure to lower uncontrollably and cooldown the RCS. Any SG with pressure lowering in an uncontrolled SG manner indicates a Faulted Steam Generator.

A. Incorrect. The first part is correct. The second part is plausible since RCS pressure is lowering and Containment pressure is rising the candidate may determine that a LOCA is in progress. This is incorrect because SG Feedflow is rising concurrent with SG level lowering which is indication of a fault on the Feedwater line to the SG.

B. Correct.

C. Incorrect. The first part is plausible since the FRVs automatically shut after the Reactor trip breakers open and RCS temperature lowers below 564°F. The candidate may determine a Feedwater break inside Containment upstream of the Feedwater Isolation Valves would be isolated from the SG. This is incorrect because the SG inventory would continue to be released into Containment following the Reactor Trip. The second part is plausible since RCS pressure is lowering and Containment pressure is rising. The candidate may determine that a LOCA is in progress. This is incorrect because SG feedflow is rising concurrent with SG level lowering which is indication of a fault on the Feedwater line to the SG

D. Incorrect. The first part is plausible since the FRVs automatically shut after the Reactor trip breakers open and RCS temperature lowers below 564°F. The candidate may determine a feedwater break inside Containment upstream of the FW Isolation Valves would be isolated from the SG. This is incorrect because the SG inventory would continue to be released into Containment following the Reactor Trip. The sceond part is correct.

000054 Loss of Main Feedwater /4

054AK1.01; Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW):MFW line break depressurizes the S/G (similar to a steam line break)

(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	4.1 4.3
Technical Reference:	EOP-E-0, Step 25, Pg 30, Rev. 7
References to be provided:	None
Learning Objective:	EOP-LP-3.02, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 009/BANK/C/A//EOP-ECA-0.0/NONE/2012 NRC RO 09/055 EK1 .02/

Given the following plant conditions:

- A loss of Off-site power has occurred with the unit operating at 100%
- EDG 'A' fails to start
- EDG 'B' output breaker closes, but trips open
- EOP-ECA-0.0, Loss Of all AC Power, is being implemented
- ASI system is operating as expected

Which ONE of the following completes the statement below?

In accordance with EOP-ECA-0.0, the RCS cooldown during natural circulation is limited to a MAXIMUM rate of ___(1)___ AND the cooldown is required to ___(2)___ .

A. (1) 100°F / Hr

(2) minimize RCP seal leakage

B. (1) 100°F / Hr

(2) control Pressurizer level

C. (1) 50°F / Hr

(2) minimize RCP seal leakage

D.✓ (1) 50°F / Hr

(2) control Pressurizer level

Plausibility and Answer Analysis

Reason answer is correct: During a LOOSP the ASI system provides seal injection to the RCP seals. To prevent the RCS from going solid a cooldown must be initiated to offset the mass addition from the ASI system.
The maximum cooldown rate to control PRZ level is 50°F / hr.

A. Incorrect. The first part is plausible since a rate of 100°F / hr maximum cooldown rate under normal conditions, however this is incorrect because the cooldown rate during EOP-ECA-0.0 is limited to half the normal rate. The second part is plausible since this is the correct answer if the ASI system fails to operate, however this is incorrect because all other systems operate as designed.

.

B. Incorrect. The first part is plausible since a rate of 100°F / hr maximum cooldown rate under normal conditions, however this is incorrect because the cooldown rate during EOP-ECA-0.0 is limited to half the normal rate. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since this is the correct answer if the ASI system fails to operate, however this is incorrect because all other systems operate as designed.

D. Correct.

000055 Station Blackout / 6

055EK1 .02; Knowledge of the operational implications of the following concepts as they apply to the Station Blackout : Natural circulation cooling

(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	4.1 4.4
Technical Reference:	EOP-ECA-0.0, Step 33.a, Pg 56, Rev. 7
References to be provided:	Steam Tables
Learning Objective:	EOP-LP-3.7 Obj. 6
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 010/BANK/C/A//AOP-020/NONE/2009A NRC RO 75/056 AG2.4.9/

Given the following plant conditions:
- The unit is in Mode 5
- The RCS is in solid plant operation
- Both Trains of RHR are aligned in the Shutdown Cooling Mode

Subsequently an earthquake occurs:
- Offsite power is lost and a large RCS leak has developed
- 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 is the action required by AOP-020 to mitigate the event?

A. Start the 'B' CSIP with flow through 1SI-3 and 1SI-4, BIT Outlet Valves

B. Start the 'B' CSIP with flow through 1SI-52, Alternate High Head SI to Cold Leg Valve

C. Align 'A' RHR Pump for Low Head SI with flow through 1SI-359, Low Head SI Trains to Hot Leg Valve

D✓ Align 'A' RHR Pump for Low Head SI with flow through 1SI-340, Low Head SI Train A to Cold 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.

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 because 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 with two CSIPs, however this is incorrect because only one CSIP is Operable in this mode.

C. Incorrect. Plausible since alignment of one train of RHR for Low Head SI is directed in AOP-020, however this is incorrect because flow is through 1SI-340, Low Head SI Train A to Cold Leg Valve not 1SI-359, which is a possible alignment when implementing EOP-ES-1.4.

D. Correct.

000056 Loss of Offsite Power / 6

056AG2.4.9; Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.8 4.2
Technical Reference:	AOP-020, Section 3.6, pg 66, Rev 38
References to be provided:	None
Learning Objective:	AOP-LP-3.20, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 011/BANK/C/A//AOP-024-BD/NONE/2012 NRC RO 11/057 AK3.01/

Given the following plant conditions:
- The unit is operating at 100% power
- PRZ Level Control is selected to position 459/460

Subsequently:
- Instrument Bus S-II is lost
- The crew is implementing AOP-024, Loss of Uninterruptible Power Supply

Which ONE of the following completes the statements below?

Letdown flow will be isolated because ___(1)___ went SHUT.

In accordance with AOP-024, the OATC will control Charging in manual with FK-122.1, Charging Flow, to ___(2)___ with letdown isolated.

A. (1) 1CS-1, Letdown Isolation LCV-460

(2) prevent gas binding of the CSIP

B✓ (1) 1CS-1, Letdown Isolation LCV-460

(2) minimize the PRZ level rise

C. (1) 1CS-2, Letdown Isolation LCV-459

(2) prevent gas binding of the CSIP

D. (1) 1CS-2, Letdown Isolation LCV-459

(2) minimize the PRZ level rise

Plausibility and Answer Analysis

Reason answer is correct: Instrument Bus SII supplies Pressurizer Level channel 460 and the affect of losing power is the channel will fail low, which causes its associated letdown isolation valve 1CS-1 to shut. With letdown isolated to reduce charging flow to the minimum amount the FK-122 must be placed in manual. This is required to minimize the rise in PRZ level while selecting a valid channel since seal injection will continue into the RCS with no letdown to lower the level.

A. Incorrect. The first part is correct. The second part is plausible since with letdown isolated the VCT level will lower as the CSIP continues to take suction from it which expands the VCT vapor space and increases the potential for gas intrusion at the pump impeller, however this is incorrect because the CSIP suction will automatically shift from the VCT to the RWST at 5% therefore placing charging flow to manaul and minimum is not required.

B. Correct.

C. Incorrect. Part 1 is plausible since 1CS-2 shutting will isolate letdown flowif it is determined that LT-459 has loss power.

Part 2 is plausible if the candidate believes the VCT Level transmitters are effected with the lost of power to S-II, since with letdown isolated the VCT level will lower as the CSIP continues to take suction from it which expands the VCT vapor space and increases the potential for gas intrusion at the pump impeller.

D. Incorrect. Part 1 is plausible since 1CS-2 shutting will isolate letdown flowif it is determined that LT-459 has loss power.

Part 2 is correct.

000057 Loss of Vital AC Instrument Bus / 6

057AK3.01; Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus

(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	4.1 4.4
Technical Reference:	AOP-024-BD, Section 1.0, Pg 5, Rev. 21 AOP-024-BD, Section 2.0, Step 3, Pg 7, Rev. 21
References to be provided:	None
Learning Objective:	AOP-LP-3.24, Obj. 4
Question Origin:	Bank
Comments:	Discuss use of actions contained in AOP verses EOP for loss of vital AC instrument bus with Dan Bacon for question use since HNP does not have any EOP references to the loss of an instrument bus. Phonecon 6/13: Dan agrees that using an AOP for loss of vital AC instrument bus is acceptable to meet this K/A. RO Q29 and ROQ11 have been looked at for double jeopardy and as an exam team we have determined that there is an adequate difference to NOT be a concern.
Tier/Group:	T1/G1

2018 NRC RO 012/BANK/C/A//6-B-401 SH 1922/NONE/2012 NRC RO 12/058 AA2.03/

Given the following plant conditions:
- The unit is in Mode 3
- MDAFW Pumps 'A' and 'B' are in service feeding all 3 SGs
- EDG 'B' is under clearance for maintenance

Subsequently:
- All power from 125 VDC Emergency Bus DP-1B-SB is lost
- One minute later Startup Transformer 1B lockout occurs

Which ONE of the following completes the statement below?

MDAFW Pump 'B' breaker indication on the MCB is ___(1)___ AND the pump motor breaker is ___(2)___ .

A. (1) available

(2) closed

B. (1) available

(2) open

C.✓ (1) not available

(2) closed

D. (1) not available

(2) open

Plausibility and Answer Analysis

Reason answer is correct:125 VDC DP-1B-SB provides control power to the breaker cubicle to provide remote indication of breaker status on the MCB and allows for the operation of the breaker from the MCB. With the loss of control power the indication on the MCB is not available and the ability to remotely operate the breaker is lost as well. The breaker remains shut even with a fault of 6.9KV bus 1B-SB, which would normally open the breaker on UV.

A. Incorrect. (1) Plausible because indication would remain available if the DC bus lost were not the specific supply for DC control power to the 'B' MDAFW. (2) is correct.

B. Incorrect. (1) Plausible because indication would remain available if the DC bus lost were not the specific supply for DC control power to the 'B' MDAFW. (2) Plausible because the breaker normally trips open on UV when its associated 6.9KV bus is de-energized.

C. Correct.

D. Incorrect. (1) is correct. (2) Plausible because the breaker normally trips open on UV when its associated 6.9KV bus is de-energized.

000058 Loss of DC Power / 6

058AA2.03; Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems

(CFR: 43.5 / 45.13)

Importance Rating:	3.5 3.9
Technical Reference:	Drawing 2166-B-401-1922, Rev 12
References to be provided:	None
Learning Objective:	DCP Lesson Plan, Obj. 9
Question Origin:	Bank
Comments:	Discuss if it was required to have a K/A match for both monitor and operate or if the K/A is met by meeting just one of the two topics. Phonecon 10/23: Dan stated that it was acceptable to meet this K/A by addressing either the monitoring or operating portion of this K/A topic based on the question specifics.
Tier/Group:	T1/G1

2018 NRC RO 013/BANK/FUNDAMENTAL//AOP-022-BD/NONE//062 AA2.06/

Which ONE of the following identifies (1) how long the EDG 1A-SA can operate fully loaded without ESW and (2) the reason why?

A.✓ (1) a maximum of one minute

(2) to protect against equipment damage due to overheating

B. (1) a maximum of one minute

(2) to allow adequate time to re-align the equipment to NSW

C. (1) a maximum of five minutes

(2) to protect against equipment damage due to overheating

D. (1) a maximum of five minutes

(2) to allow adequate time to re-align the equipment to NSW

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-022, Loss of Service Water Basis Document the EDG must be secured within 1 minute of operating at full load without cooling water flow. The EDG is considered an essential load and requires the component to be stopped to protect against equipment damage due to overheating.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since NSW is normally the cooling supply for the EDG 1A-SA during non-emergency conditions, however this is incorrect because essitial equipment such as EDG 1A-SA are stopped to protect *against equipment damage due to overheating under these conditions.*

C. Incorrect. The first part is plausible since five minutes is in reference to how long a CSIP has been stopped before a controlled heat is required when restarting the CSIP, however this is incorrect because the requirement for the stopping of a loaded EDG without ESW is 1 minute. Additionally five minutes is a reasonable amount of time to ensure actions can be taken in the procedure prior to reaching the point of checking EDG cooling.

D. Incorrect. The first part is plausible since five minutes is in reference to how long a CSIP has been stopped before a controlled heat is required when restarting the CSIP, however this is incorrect because the requirement for the stopping of a loaded EDG without ESW is 1 minute. Additionally five minutes is a reasonable amount of time to ensure actions can be taken in the procedure prior to reaching the point of checking EDG cooling. The second part is plausible since NSW is normally the cooling supply for the EDG 1A-SA during non-emergency conditions, however this is incorrect because essitial equipment such as EDG 1A-SA are stopped to protect *against equipment damage due to overheating under these conditions.*

000062 Loss of Nuclear Service Water / 4

062AA2.06; Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water:The length of time after the loss of SWS flow to a component before that component may be damaged

(CFR: 43.5 / 45.13)

Importance Rating:	2.8 3.1
Technical Reference:	AOP-022-BD, Section 2.0, Step 2, Pg 10, Rev. 14
References to be provided:	None
Learning Objective:	AOP-LP-022, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 014/BANK/C/A//AOP-017, OP-134.01/NONE/2012 NRC RO 14/065 AK3.04/

Given the following plant conditions:
- The unit is operating at 100% power
- IA header pressure lowers as indicated below:

Time	IA Header Pressure
0750	115 psig
0755	95 psig
0800	85 psig
0805	70 psig
0810	60 psig
0815	45 psig

Which ONE of the following completes the statements below in accordance with AOP-017, Loss Of Instrument Air?

At ___(1)___ an auto shut signal to the Main Feed Reg valves is FIRST generated.

The reason these valves have back up air accumulators is to provide ___(2)___ .

A. (1) 0800

(2) motive force to shut the valves

B. (1) 0800

(2) the ability to maintain the valves open for a short period of time following loss of air

[✓ C.] (1) 0810

(2) motive force to shut the valves

D. (1) 0810

(2) the ability to maintain the valves open for a short period of time following loss of air

Plausibility and Answer Analysis

Reason answer is correct: During a loss of IA when pressure is < 60# the valves receive an auto shut signal. The motive force to shut each valve is provided by air accumulators for each valve. The reason the air accumulators are neccessary is the valves would fail in the last position if there was not a motive force to shut them.

A. Incorrect. Plausible because at 85 psig it has been determined that spurious valve actuations will begin to occur on the Letdown system due to air pressure not being able to overcome spring pressure to maintain the valve open. The second part of the answer is correct.

B. Incorrect. Plausible because at 85 psig it has been determined that spurious valve actuations will begin to occur on the Letdown system due to air pressure not being able to overcome spring pressure to maintain the valve open. The second part of the answer is plausible because other plant valves have hydraulic accumulators (DEH system for GV's and Throttle valves) to allow operation during a loss of power

C. Correct.

D. Incorrect. Plausible since the pressure is correct and because other plant valves have hydraulic accumulators (DEH system for GV's and Throttle valves) to allow operation during a loss of power.

000065 Loss of Instrument Air / 8

065AK3.04; Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Cross-over to backup air supplies

(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	3.0 3.2
Technical Reference:	AOP-017, Note prior to Step 1, Pg 4, Rev 40 OP-134.01, P&L 17, 18, Pg 8, Rev 44
References to be provided:	None
Learning Objective:	CFW Lesson Plan, Obj. 8.j
Question Origin:	Bank
Comments:	HNP uses backup air supply accumulators on the Main Feed Reg Valves which provides motive force to shut the valve when pressure lowers to 60 psig. Discussed with Dan Bacon using this system for a question since HNP does not have a backup air supply system that has cross over capability. Phonecon 6/13: Dan agrees that using the backup air supply accumulators on the Main Feed Reg Valves is acceptable to meet this K/A.
Tier/Group:	T1/G1

2018 NRC RO 015/NEW/FUNDAMENTAL//ALB-022/NONE/2012 NRC RO 15/077 AA1.03/

Given the following plant conditions:
- The plant is operating at 100% power
- Annunciator ALB-022-5-3, Generator Voltage Balance Relay Operated, Alarms
- The voltage regulator remains in AUTO
- MW and MVAR indications are erratic

Which ONE of the following actions will be taken for these conditions in accordance with APP-ALB-022?

A. Place the Generator Voltage Regulator switch to OFF and use CS-1539, Voltage Setpoint Reference switch to restore Main Generator voltage

B✓ Place CS-1538, Operational Mode switch in Manual, then operate CS-1539, Voltage Setpoint Reference switch to stabilize Generator Stator voltage at 22KV.

C. Adjust the Generator Output Voltage to 230KV with CS-1539, Voltage Setpoint Reference switch. If the adjustment does not work, Trip the Reactor and enter EOP-E-0, Reactor Trip Or Safety Injection.

D. Transfer the Voltage Regulator control to Local by placing CS-1540, Local Control Enable switch in the Local, to stabilize Output Voltage and dispatch an operator to 286 TB Switchgear room. If this action does not work, Trip the Reactor and enter EOP-E-0, Reactor Trip Or Safety Injection.

Plausibility and Answer Analysis

Reason answer is correct: Per APP-ALB-022-5-3, if the Generator voltage regulator is in Auto then **PLACE** the CS-1538, Operation Mode switch, in MANUAL mode and **OPERATE** the CS-1539, Voltage Setpoint Reference switch, to stabilize the Generator Stator Voltage at 22KV.

A. Incorrect. Plausible because during a Generator shutdown the Generator Voltage Regulator is placed in MANUAL and the Isophase bus duct cooling fan control switches are placed in the OFF position and the candidate may have the misconception that the MANUAL position for the Voltage regulator corresponds to the OFF position.

B. Correct.

C. Incorrect. Plausible because the voltage regulator is normally adjusted using CS-1539, Voltage Setpoint Reference and 230 KV is the nominal voltage maintained on the Main Transformer high side. Additionally AOP-006 for turbine/generator trouble provides trip criteria when exceeding limits on the turbine.

D. Incorrect. Plausible because the AVR has the capability to be controlled locally from the switchgear, but event and alarm indications are available locally and AOP-006 for turbine/generator trouble provides trip criteria when exceeding limits on the turbine.

000077 Generator Voltage and Electric Grid Disturbances / 6

077AA1 .03; Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Voltage regulator controls

(CFR: 41.5 and 41.10 / 45.5, 45.7, and 45.8)

Importance Rating:	3.8 3.7
Technical Reference:	APP-ALB-022, Window 5-3, Page 38, Rev 85
References to be provided:	None
Learning Objective:	Main Generator Lesson Plan, Obj. 3.c
Question Origin:	New
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 016/NEW/C/A//E-1, ECA-1.2/NONE//WE04 EK1.3/

Given the following plant conditions:
- A Reactor Trip and Safety Injection has occured
- EOP-E-1, Loss of Reactor Or Secondary Coolant, is being implemented and SI has been reset
- RCS Pressure is 1425 psig and stable
- Containment pressure 2.1 psig
- PZR level is off scale low
- Subcooling is 8°F
- Rad monitor, RM-1RR-3598, RHR Pump 1A is in high alarm and rising
- Window 6-2, RAB Equip A/B Sump Alert Lvl, is lit on MLB-4A-SA and MLB-4B-SB

Which ONE of the following identifies the FIRST procedure to be entered from EOP-E-1 to mitigate the event in progress?

A. EOP-ES-1.2, Post LOCA Cooldown And Depressurization

B. EOP-ES-1.3, Transfer To Cold Leg Recirculation

C. EOP-ECA-1.1, Loss Of Emergency Coolant Recirculation

D✓ EOP-ECA-1.2, LOCA Outside Containment

Plausibility and Answer Analysis

Reason answer is correct: EOP-E-1 step 12 evaluates the status of the plant and directs entry into EOP-ECA-1.2 if high radiation levels are due to a loss of RCS inventory outside of containment.

A. Incorrect. The plausible since RCS pressure is above the Low Pressure ECCS system injection pressure and subcooling is less than 10˚F and PRZ level is less than 10% SI termination criteria is not satisfied, however this is incorrect because the leak is outside of containment and must be addressed first prior to depressurizing the RCS.

B. Incorrect. The plausible since this is the expected procedure transition when the ECCS system operates as designed to collect the coolant in the Emergency Recirculation sump to allow for long term cooling of the RCS, however this is incorrect because the coolant is bypassing the Emergency Recirculation sump and other actions must be taken to ensure the coolant is going to the desired location in the ECCS system.

C. Incorrect. The plausible since the coolant is bypassing the Emergency Recirculation sump and the ECCS system is not capable of returning the coolant collect outside of Containment to the RCS as designed the candidate will eventually transition to this procedure, however this is incorrect because the candidate with first attempt other actions to isolate the leakage outside of Containment to ensure the coolant is going to the desired location in the ECCS system.

D. Correct.

W/E04 LOCA Outside Containment / 3

WE04EK1.3; Knowledge of the operational implications of the following concepts as they apply to the (LOCA Outside Containment): Annunciators and conditions indicating signals, and remedial actions associated with the (LOCA Outside Containment).

(CFR: 41.8 / 41.10, 45.3)

Importance Rating:	3.5 3.9
Technical Reference:	EOP-E-1, Step 12.e, Pg 16, Rev 4 EOP-ECA-1.2, Pg 2, Rev 0
References to be provided:	None
Learning Objective:	EOP-LP-3.03, Obj. 2.d
Question Origin:	New
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 017/BANK/C/A//CSFST/NONE/2012 NRC RO 17/WE05 EK2.2/

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
- SG NR levels are 41% and lowering
- Containment pressure is 3.3 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)___.

A.✓ (1) 39%

(2) 195 KPPH

B. (1) 39%

(2) 205 KPPH

C. (1) 24%

(2) 195 KPPH

D. (1) 24%

(2) 205 KPPH

Plausibility and Answer Analysis

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 conditions).

A. Correct

B. Incorrect. The first part is correct. The second part is plausible since one of the parameters monitors for heat sink is below the required value the candidate may misapply this information and determine that both SG level and AFW flow need to be above the adverse Containment requirement to preclude entry into EOP-FR-H.1.

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

D. Incorrect. The first part is plausible see C (1). The second part is plausible see answer B (2).

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, Obj. 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 018/BANK/FUNDAMENTAL//FR-Z.1, BKGRD/NONE/2011 NRC RO 17/WE11 EK3.4/

Why does EOP-ECA-1.1, Loss of Emergency Coolant Recirculation take precedence over EOP-FR-Z.1, Response to High Containment Pressure for operation of the Containment Spray Pumps?

The reason is based on _______ .

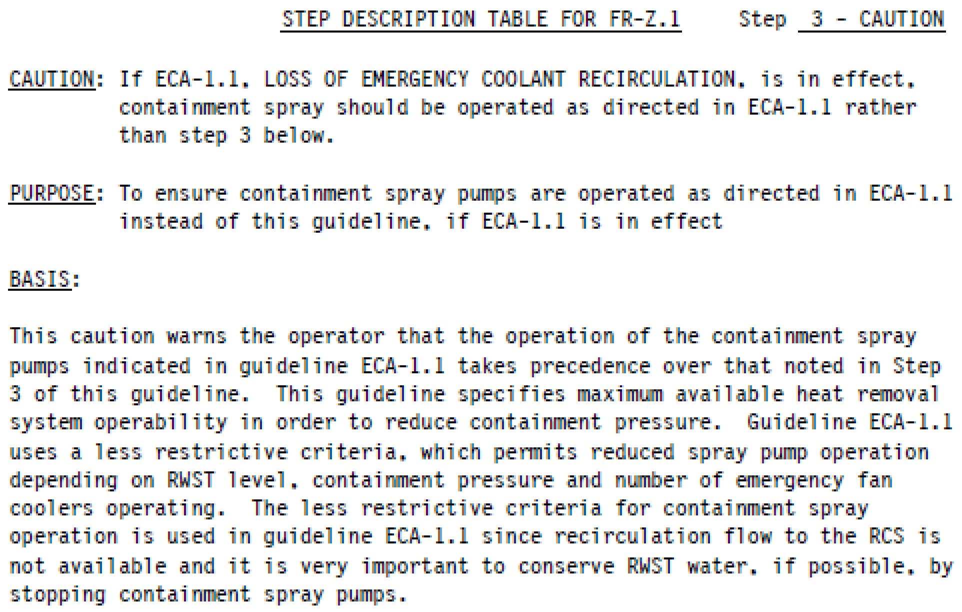
A. maintaining Containment iodine removal

B. maintaining Containment heat removal

C. limiting Containment pressure

D✓ conserving RWST inventory

Reason answer is correct:



STEP DESCRIPTION TABLE FOR FR-Z.1 Step 3 - CAUTION

CAUTION: If ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, containment spray should be operated as directed in ECA-1.1 rather than step 3 below.

PURPOSE: To ensure containment spray pumps are operated as directed in ECA-1.1 instead of this guideline, if ECA-1.1 is in effect

BASIS:

This caution warns the operator that the operation of the containment spray pumps indicated in guideline ECA-1.1 takes precedence over that noted in Step 3 of this guideline. This guideline specifies maximum available heat removal system operability in order to reduce containment pressure. Guideline ECA-1.1 uses a less restrictive criteria, which permits reduced spray pump operation depending on RWST level, containment pressure and number of emergency fan coolers operating. The less restrictive criteria for containment spray operation is used in guideline ECA-1.1 since recirculation flow to the RCS is not available and it is very important to conserve RWST water, if possible, by stopping containment spray pumps.

A. Incorrect. *Plausible since containment iodine removal is a function of the containment spray pumps and is a factor in determination of securing containment spray pumps in E-0/E-1.*

B. Incorrect. *Plausible since EOP-ECA-1.1 uses the number of of cnmt fan coolers operating (heat removal capability) in evaluating the number of containment spray pumps to run.*

C. Incorrect. *Plausible since EOP-ECA-1.1 uses containment pressure in evaluating the number of containment spray pumps to run.*

D. Correct.

W/E11 Loss of Emergency Coolant Recirculation / 4

WE11EK3.4; Knowledge of the reasons for the following responses as they apply to the (Loss of Emergency Coolant Recirculation): RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

(CFR: 41.5 / 41.10, 45.6, 45.13)

Importance Rating:	3.6 3.8
Technical Reference:	ERG-BKGRD-FR-Z.1, Step 3 Caution, Pg 8, Rev. 1c
References to be provided:	None
Learning Objective:	EOP-LP-2.3/3.3, Obj. 1.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 019/NEW/C/A//GFES/NONE//001 AK1.22/

Given the following plant conditions:

- The unit is operating at 70% power
- NI Gain Adjustment is in progress
- N-43 has been completed and the BOP is performing adjustment on N-44

Subsequently the following conditions are observed:

- T_{avg} is rising
- T_{ref} remains constant
- Control Rods are in motion
- PRZ pressure and level are rising

Which ONE of the following completes the statements below?

A(An) ___(1)___ event in progress.

As a result of the above, Axial Flux Distribution (AFD) will ___(2)___ the original value.

(Assuming NO operator action)

A. (1) Inadvertant dilution

(2) LOWER then return to

B. (1) Inadvertant dilution

(2) be LESS negative than

C. (1) Continuous rod withdrawal

(2) LOWER then return to

D✓ (1) Continuous rod withdrawal

(2) be LESS negative than

Plausibility and Answer Analysis

Reason answer is correct: The procedure for performing NI Gain adjustments places the rod control system in manual to preclude a spurious rod motion event due to the adjustment. The combination of rising T_{avg}, PRZ level and PRZ pressure combined with rod motion are indications that the RCS is responding to a Continuous rod withdrawl event. Control rod motion out of the core exposes more fuel to nuclear reactions resulting in higher flux being genereated in the upper region of the core resulting in less negative AFD.

A. Incorrect. The first part is plausible since RCS Temperature/Pressurizer response are correct for an inadvertant dilution, however this is incorrect because the rods are placed in manual to support NI gain adjustments therefore an inadvertant dilution would not cause rod motion. The second part is plausible since rod motion should not occur for an inadverdant dilution event with rod contol in manual, however this is incorrect for the conditions of the question because outward rod motion is in progress which will result in AFD becoming less negative.

B. Incorrect. The first part is plausible since RCS Temperature/Pressurizer response are correct for an inadvertant dilution, however this is incorrect because the rods are placed in manual to support NI gain adjustments therefore an inadvertant dilution would not cause rod motion. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since rod motion should not occur with rod contol in manual therefore AFD should remain the same, however this is incorrect for the conditions of the question because outward rod motion is in progress which will result in AFD becoming less negative.

D. Correct.

QUESTIONS REPORT

for 2018 NRC RO SRO WRITTEN EXAM REV 75 Submittal

000001 Continuous Rod Withdrawal / 1

001AK1.22; Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal: Delta flux (Δ I)

(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	3.2 3.6
Technical Reference:	GFES, Reactor Theory
References to be provided:	None
Learning Objective:	TAA-LP-3.25, Obj. 1.a
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 020/NEW/FUNDAMENTAL//FHP-400/NONE/EARLY/036 AK3.02/

Given the following plant conditions:
- The unit is in Mode 6
- Fuel movement is in progress

Subsequently the Manipulator Crane upward motion suddenly stops and the Overload indicator is illuminated

Which ONE of the following completes the statement below?

The Manipulator Crane Overload Interlock stopped upward motion when the hoist detected ___(1)___ pounds above the weight of the mast and the ___(2)___ .

A.✓ (1) 150

(2) Fuel Assembly

B. (1) 150

(2) Rod Control Cluster Assembly

C. (1) 430

(2) Fuel Assembly

D. (1) 430

(2) Rod Control Cluster Assembly

Plausibility and Answer Analysis

Reason answer is correct: Overload Interlock - Prevents fuel damage by stopping the hoist up travel when hoist load is 150 lbs above weight of mast and fuel assembly. (variable set points depending on Load Select switch position).

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since the manipulator setpoint is varied based on the weight of the fuel assembly with or without an RCCA, however this is incorrect because the overload interlock is associated with the manipulator main hoist.

C. Incorrect. The first part is plausible since it is the weight of the Slack Cable interlock therefore the candidate may have the misconception that the overload interlock is set to 430 lbs vice 150 lbs. The second part is plausible since the manipulator setpoint is varied based on the weight of the fuel assembly with or without an RCCA, however this is incorrect because the overload interlock is associated with the manipulator main hoist.

D. Incorrect. The first part is plausible since it is the weight of the Slack Cable interlock therefore the candidate may have the misconception that the overload interlock is set to 430 lbs vice 150 lbs. The second part is correct.

000036 Fuel-Handling Incidents / 8

036AK3.02; Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents: Interlocks associated with fuel handling equipment

(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	2.9 3.6
Technical Reference:	FHP-400, Attachment 3, Pg 69, Rev 3
References to be provided:	None
Learning Objective:	FHS Lesson Plan, Obj. 6
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 021/BANK/C/A//AOP-016/NONE/2006 NRC RO 33/037 AA2.01/EARLY

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

Subsequently
- REM-01TV-3534, Condenser Vacuum Pump Rad monitor, is indicating 1.32×10^{-06} µCi/cc and rising
- AOP-016, Excessive Primary Plant Leakage, is in progress
- The OATC has completed a leak rate estimate calcuation to quantify the leak rate

Which ONE of the following indications will serve to verify the value of actual primary to secondary leak rate?

A. Local surveys of Steam Generator Blowdown Lines

B. Trend on Turbine Building Vent Stack Effluent, RM-1TV-3536-1

C. Alarm status of Main Steam Line Radiation Monitors RM-01MS-3591 SB, 3592 SB, or 3593 SB

D✓ Condenser Vacuum Pump Effluent Monitor indication and a conversion factor supplied by Chemistry after sampling

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-016 Primary-To-Secondary leak rate is estimated using ONE of the methods listed below
Condenser Vacuum Pump Rad Monitor, REM-01TV-3534 and Curve H-X-15
Condenser Vacuum Pump Rad Monitor, REM-01TV-3534 and Conversion factor provided by Chemistry after sampling has commenced

A. Incorrect. Plausible since local surveys of SGBD lines are directed to be performed in order to determine the which SG is leaking, however this is incorrect as this method does not determine the amount of leakage from Primary to Secondary.

B. Incorrect. Plausible since the Turbine Building Vent Stack is one of the radiation monitors used to determine if an Offsite Dose Calculation is required to be performed, however this is incorrect as this method does not determine the amount of leakage from Primary to Secondary.

C. Incorrect. Plausible since Main Steam line radiation monitor levels are directed to be monitored in order to determine the which SG is leaking, however this is incorrect as this method does not determine the amount of leakage from Primary to Secondary.

D. Correct.

000037 Steam Generator Tube Leak / 3

037AA2.01; Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Unusual readings of the monitors; steps needed to verify readings

(CFR: 43.5 / 45.13)

Importance Rating:	3.0 3.4
Technical Reference:	AOP-016, Attachment 1, Pg 14, Rev. 56
References to be provided:	None
Learning Objective:	AOP-LP-3.16, Obj. 3
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 022/BANK/C/A//ALB-018/NONE/2011 NRC RO 22/051 AA2.02/

Given the following plant conditions:
- The plant is operating at 45% power
- Condenser vacuum is degrading as indicated below:

Time	Zone 1 Pressure	Zone 2 Pressure
0900	4.4" Hg absolute	4.8" Hg absolute
0905	4.8" Hg absolute	5.3" Hg absolute
0910	5.4" Hg absolute	5.9" Hg absolute
0915	7.2" Hg absolute	7.8" Hg absolute
0920	7.6" Hg absolute	8.4" Hg absolute

What is the EARLIEST time that an automatic Turbine/Reactor Trip will be generated?

A. 0905

B.✓ 0910

C. 0915

D. 0920

Plausibility and Answer Analysis

Reason answer is correct: With the 1st stage pressure below 60% the Low vacuum trip of the main turbine occurs automatically when condenser vacuum is above 5" HgA on condenser zone 1. This condition is first satisfied at 0910

A. Incorrect. Plausible because the value is above the 5" HgA setpoint if detemined the low vacuum trip occurs automatically from zone 2.

B. Correct.

C. Incorrect. Plausible because the value is above the 7.5" HgA setpoint if detemined the low vacuum trip occurs automatically from zone 2, but this occurs when 1st stage pressure is above 60%.

D. Incorrect. Plausible because the value is above the 7.5" HgA setpoint if detemined the low vacuum trip occurs automatically from zone 1, but this occurs when 1st stage pressure is above 60%.

000051 Loss of Condenser Vacuum / 4

051AA2.02; Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip

(CFR: 43.5 / 45.13)

Importance Rating:	3.9 4.1
Technical Reference:	APP-ALB-018, Window 1-1, Pg 3, Rev 21
References to be provided:	None
Learning Objective:	AOP-LP-3.12, Obj. 4.a
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 023/NEW/C/A//AOP-004/NONE//068 AA1.28/

Given the following plant conditions:

- The unit was operating at 100% when a fire occurred in the MCR
- The crew has entered AOP-004, Remote Shutdown, and has relocated to the ACP
- The crew has secured the CSIP's and the ASI pump has started

Which ONE of the following actions by the operators at the ACP completes the statements below to control PZR level and pressure?

To support automatic control of 'A' and 'B' PRZ heaters, PK-444A2.A, Pressurizer Pressure controller, will be placed in automatic with the setpoint adjusted to __(1)__ .

To prevent PRZ level from exceeding the high level band required by the procedure the operators will __(2)__ .

A. (1) 67%

(2) secure the ASI pump

B✓ (1) 67%

(2) perform a plant cooldown

C. (1) 75%

(2) secure the ASI pump

D. (1) 75%

(2) perform a plant cooldown

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-004 step 16.b (2) PK-444A2.A will be placed in automatic with a setpoint adjusted to 67% (2235 psig) to support automatic control of 'A' and 'B' PRZ heaters.

Master pressure controller at the ACP, PK- 444A2. A controller setpoint of 0% equates to 1700 psig setpoint and 100% to 2500 psig setpoint (1700-2500 psig range on PT-444). Therefore, 67% = 2235 psig nominal setpoint:
2500 - 1700 = 800 psig total meter span
2235 - 1700 = 535 psig
535/800 = 67%

A note prior to step 21 states that with RCS temperature stable, PRZ level will rise slowly with the ASI pump in service....Pressurizer level may not rise to 75% for several hours. Maintaining RCS Tcold between 555°F and 559°F is preferred. High Pressurizer level may require a plant cooldown below this band. Step 21 is a continuous action step of the procedure which is to maintain Pressurizer level between

25% and 75%. To reduce Pressurizer level the RCS is cooled down to shrink the RCS by controlling AFW flow and dumping steam using SG PORVs. The ASI pump is NOT secured until after a RCS cooldown is used to shrink the RCS and PRZ level is >75% and cannot be controlled with RCS cooldown. The question asks what the operators do to PREVENT PRZ level from exceeding the high level band not what to do AFTER the PRZ level high level band is exceeded.

A. Incorrect. The first part is correct. The second part is plausible since the ASI system will be running (due to fire in MCR the CSIP is secured IAW AOP-004 step 9). With the ASI pump operating and RCS temperature maintained stable the PRZ level will slowly rise. To maintain PRZ level in band the AOP directs the operators to cool the RCS. Attachment 11, Actions if ASI Pump Starts also has the operators maintain Pressurizer level between 25% and 75% by COOLDOWN to shrink the RCS. The ASI pump is secured ONLY if PRZ level is >75% and cannot be controlled with RCS cooldown.

B. Correct.

C. Incorrect. The first part is plausible since 75% is the upper control band limit for Pressurizer level (25% - 75% is the control band) but if the setpoint is adjusted to 75% PRZ pressure would be set to 2300 psig which would not support automatic control of 'A' and 'B' PRZ heaters. The second part is plausible since the ASI system will be running (due to fire in MCR the CSIP is secured IAW AOP-004 step 9). With the ASI pump in operating and RCS temperature maintained stable the PRZ level will slowly rise. To maintain PRZ level in band the AOP directs the operators to cool the RCS. Attachment 11, Actions if ASI Pump Starts also has the operators maintain Pressurizer level between 25% and 75% by COOLDOWN to shrink the RCS. The ASI pump is secured ONLY if PRZ level is >75% and cannot be controlled with RCS cooldown.

D. Incorrect. The first part is plausible since 75% is the upper control band limit for Pressurizer level (25% - 75% is the control band) but if the setpoint is adjusted to 75% PRZ pressure would be set to 2300 psig which would not support automatic control of 'A' and 'B' PRZ heaters. The second part is correct.

000068 Control Room Evacuation / 8

068 AA1.28; Ability to operate and / or monitor the following as they apply to the Control Room Evacuation: PZR level control and pressure control

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.8 4.0
Technical Reference:	AOP-004, Section 3.1, Step 12.b, Pg 23, Rev. 68 AOP-004, Section 3.1, Step 21.a, Pg 32, Rev. 68
References to be provided:	None
Learning Objective:	AOP-LP-3.4, Obj. 4
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 024/BANK/FUNDAMENTAL//AOP-032/NONE/2012 NRC RO 24/076 AK2.01/

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

Subsequently:
- ALB-026-2-1, Gross Failed Fuel Detector Trouble, is received
- Chemistry reports RCS activity is rising, and the latest sample is 1.4 microcuries per gram dose equivalent I-131

Which ONE of the following identifies (1) the source of radiation detected by the Gross Failed Fuel Detector AND (2) what action will be directed by AOP-032, High RCS Activity, to reduce RCS activity levels?

A.✓ (1) neutron radiation from delayed neutrons

(2) place the Cation demineralizer in service

B. (1) neutron radiation from delayed neutrons

(2) place the second Mixed bed demineralizer in service

C. (1) gamma radiation from decay of fission products

(2) place Cation demineralizer in service

D. (1) gamma radiation from decay of fission products

(2) place the second Mixed bed demineralizer in service

Plausibility and Answer Analysis

Reason answer is correct: The GFFD is a BF3 proportional counter that measures neutron radiation from long lived delayed neutron emitters. IF alarm is due to high neutron activity then refer to AOP-032, High RCS Activity. IAW AOP-032 if a Cation demin is not in service then place one in service IAW OP-107.02. OP-107.02 directs throttling flow through the in service Mixed-bed demin such that some flow is diverted throught the cation deminerlizer (not to exceed 60 gpm).

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since Mixed bed demineralizers are normally utilized during operation, but they are utilized for removal of particles, not reduction of activity (they do not remove Cesium like the Cation bed demin does). If the Mixed bed demineralizer was effective in removing activity, maximizing flow would result in maximum purification and reduction of activity

C. Incorrect. The first part is plausible because increased decay gamma activity will be present in the fuel following a fuel failure. The second part of the answer is correct.

D. Incorrect. The first part is plausible see answer C (1). The second part is plausible see answer B (2).

000076 High Reactor Coolant Activity / 9

076AK2.01; Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors

(CFR 41.7 / 45.7)

Importance Rating:	2.6 3.0
Technical Reference:	GFFD Student Text, Pg 1, Rev. 2, AOP-032, Section 2.0, Pg 3 and 7, Rev. 20
References to be provided:	None
Learning Objective:	GFFD Lesson Plan, Obj. 3.a and 4.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 025/NEW/FUNDAMENTAL//EOP-ES-1.1/NONE//WE02 EG2.4.20/

Given the following plant conditions:

- The crew is performing EOP-ES-1.1, SI Termination, and have secured one CSIP and realigned the pump to the normal Charging line

Concerning SI reinitiation Criteria during these conditions, which ONE of the following completes the statement below?

EOP-ES-1.1 cautions that __(1)__ flow through the Charging and SI lines may cause __(2)__ as indicated by oscillating discharge pressure.

A. (1) isolating

(2) a relief valve to lift

B. (1) isolating

(2) cavitation

C✓ (1) simultaneous

(2) runout

D. (1) simultaneous

(2) cavitation

Plausibility and Answer Analysis

Reason answer is correct: The caution in EOP-ES-1.1 prior to Checking SI Reinitiation Criteria states simultaneous flow through the charging and SI lines may cause CSIP runout (as indicated by oscillating discharge pressure).

A. Incorrect. The first part is plausible since the SI Reinitiation Foldout isolates flow to the charging line prior to establishing SI flow through the BIT, however this is incorrect because the miniflow valve are open during this alignment and the CSIP will not be impacted. The second part is plausible for this action may result in discharge pressure oscillations, however it is not correct in accordance with EOP-ES-1.1. This distractor was chosen to support plausibility vice balance of the distractors due to implausibilty of runout occuring with the system isolated.

B. Incorrect. The first part is plausible since the SI Reinitiation Foldout isolates flow to the charging line prior to establishing SI flow through the BIT, however this is incorrect because the miniflow valve are open during this alignment and the CSIP will not be impacted. The second part is plausible for this action if the candidate misapplies the generic fundamental concept of pump cavitation which results in discharge pressure oscillations, however it is not correct in accordance with EOP-ES-1.1.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible for this action if the candidate misapplies the generic fundamental concept of pump cavitation which results in discharge pressure oscillations, however this is incorrect as cavitation is the result of low NPSH.

W/E02 SI Termination / 3

WE02 EG2.4.20; 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-ES-1.1, Step 14 Caution, Pg 18, Rev. 2
References to be provided:	None
Learning Objective:	EOP-LP-3.1, Obj. 5
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 026/BANK/C/A//USERS GUIDE, ES-1.2/NONE/2009A NRC RO 26/WE03 EK2.2/

Given the following plant conditions:

- A LOCA has occurred
- RCS pressure is 1350 psig and stable
- Containment pressure is 3.5 psig and stable
- The crew is performing actions contained in EOP-ES-1.2, Post LOCA Cooldown and Depressurization

Which ONE of the following describes the method that will be used to perform the cooldown of the RCS?

Perform the cooldown using ______ .

A✓ S/G PORVs at less than 100°F per hour

B. S/G PORVs at the maximum achievable rate

C. Condenser Steam Dumps at less than 100°F per hour

D. Condenser Steam Dumps at maximum achievable rate

Plausibility and Answer Analysis

Reason answer is correct: SG PORVs will be used for the cooldown because the condenser is not available due to an auto closure of the MSIV's when Containment pressure reached 3 psig and EOP-ES-1.2 limits cooldown to 100°F/hour.

A. Correct.

B. Incorrect. SG PORVs will be used for the cooldown because the condenser is not available. The Cooldown Rate is incorrect. Other EOPs perform a max rate cooldown such as in EOP-E-3 but EOP-ES-1.2 limits cooldown to 100°F/hour.

C. Incorrect. Condenser steam dumps are not available because at 3 psig in Containment a MSLI actuated to shut all MSIVs. Rate is correct.

D. Incorrect. Condenser steam dumps are not available because at 3 psig in Containment a MSLI actuated to shut all MSIVs. Plausible because it is the normal method of cooldown. The Cooldown Rate is incorrect. Other EOPs such as EOP-E-3 perform a max rate cooldown but EOP-ES-1.2 limits cooldown to 100°F/hour.

W/E03 LOCA Cooldown - Depressurization / 4

WE03EK2.2; Knowledge of the interrelations between the (LOCA Cooldown and Depressurization) 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:	3.7 4.0
Technical Reference:	EOP-Users Guide, Step 6.19, Pg 47, Rev. 49 EOP-ES-1.2, Step 10, Pg 12, Rev 2.
References to be provided:	None
Learning Objective:	LP-EOP-3.5, Obj. 5.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 027/BANK/FUNDAMENTAL//FR-P.1, BKGRD/NONE/2012 NRC RO 26/WE08 EK3.3/

Given the following plant conditions:

- The crew is implementing EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock
- RCS subcooling does not support SI termination but it does support starting an RCP

Which ONE of the following describes the basis for why it is desirable to re-start an RCP under these conditions?

A. Establish PRZ Spray control

B. Supply additional heat input into the RCS

C. Promote heat transfer from the SGs to the RCS

D.✓ Promote mixing of the Safety Injection water and RCS

Plausibility and Answer Analysis

Reason answer is correct: Per the WOG Background document for EOP-FR-P.1 to reduce the likelihood of a PTS condition the start of an RCP should be attempted in order to mix cold SI water with warm RCS water.

A. Incorrect. Plausible since the preferred method of RCS pressure control is the use of PRZ Spray which requires an RCP running, however this is incorrect because this is not the desired reason for the actions under the conditions of the question stem.

B. Incorrect. Plausible since the start of a RCP will provide pump heat which reduces the low temperature concern of a PTS condition, however this is incorrect because this is not the desired reason for the actions under the conditions of the question stem.

C. Incorrect. Plausible since if SG where at a higher pressure and temperature than the RCS it will be a heat source not a heat sink and the more effective heat transfer method is to force circulation of the RCS vice natural circulation, however this is incorrect because this is not the desired reason for the actions under the conditions of the question stem.

D. Correct.

W/E08 RCS Overcooling - Pressurized Thermal Shock / 4

WE08EK3.3; Knowledge of the reasons for the following responses as they apply to the (Pressurized Thermal Shock): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

(CFR: 41.5 / 41.10, 45.6, 45.13)

Importance Rating:	3.7 3.8
Technical Reference:	ERG-BKGRD-FR-P.1, Step 6, Pg 28, Rev.2
References to be provided:	None
Learning Objective:	EOP-LP-3.14, Obj. 4.d
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 028/BANK/C/A//AOP-018, ALB-008/AOP-018 ATTACHMENT 2//003 K6.02/

Given the following plant conditions:

- The unit is operating at 100% power

Subsequently:

- ALB-008-5-3, RCP-C Seal #1 Leakoff High Low Flow, is in alarm
- ALB-008-5-4, RCP-C Seal #2 Leakoff High Flow, is in alarm
- ALB-008-5-5A, RCP-C Standpipe High Level, is in alarm
- RCP ‘C’ #1 seal leakoff is indicating < 1 gpm
- RCP ‘C’ #1 seal ΔP is indicating > 400 psid
- VCT level is slowly lowering
- RCDT level is slowly rising

Which ONE of the following describes the condition of the 'C' RCP seal package?

(Reference provided)

A. Seal injection flow has been lost

B. #1 seal has failed

C✓ #2 seal has failed

D. #3 seal has failed

Plausibility and Answer Analysis

Reason answer is correct: AOP-018 Attachment 2 Specific Symptoms of Seal Malfunctions #2 Seal failed. #2 seal leakoff flow greater than 1.1 gpm (greater than 12.0 gpm if #1 seal is failed) with a corresponding reduction in #1 seal leakoff flow. #3 seal leakoff should remain fairly constant. In the absence of additional guidance, if the No.2 seal flow exceeds 1.1 gpm, follow the procedures for shutdown of the RCP. The alarm setpoint for ALB-008 window 5-4, RCP-C Seal #2 Leakoff High Flow, is 1.0 gpm. With this condition present and a #1 seal leakoff flow < 1.0 gpm the cause of window 5-4 is failure of # 2 seal.

A. Incorrect. Plausible since window 5-3, Seal #1 leakoff high/ low flow, is in alarm and # 1 seal leakoff is < 1 gpm, however this is incorrect because seal injection flow is provided by the CSIP and VCT level continues to lower indicating the CSIP is running taking a suction from the it's normal source, which is the VCT.

B. Incorrect. Plausible since window 5-3, Seal #1 leakoff high/ low flow, is in alarm , however this is incorrect because the # 1 seal leakoff is < 1 gpm and the #1 seal ∆P is > 400 psid, which is indication of seal blockage vice seal failure.

C. Correct.

D. Incorrect. Plausible since in the RCDT level is rising and 400 cc/hr from the #3 seal leak off is aligned to the RCDT, however this is incorrect because the RCP Standpipe high level is in alarm and therefore abnormally frequent filling of the standpipe is not required, which is an indication that the #3 seal is failed.

003 Reactor Coolant Pump / 4

003K6.02; Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: RCP seals and seal water supply

(CFR: 41.7 / 45.5)

Importance Rating:	2.7 3.1
Technical Reference:	AOP-018 Attachment 2, Pg 24 and 25, Rev. 49 APP-ALB-008, Window 5-4, Pg 32, Rev. 25
References to be provided:	AOP-018 Attachment 2, Pg 24 and 25, Rev. 49
Learning Objective:	LP-AOP-3.18, Obj. 3.b
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 029/NEW/C/A//AOP-024/NONE/EARLY/004 K2.06/

Given the following plant conditions:
- The unit is operating at 100%
- Pressurizer Level Control is selected to 459/460

Which ONE of the following identifies the power supply to the controlling PRZ Level channel?

A.✓ SI

B. SII

C. SIII

D. SIV

Plausibility and Answer Analysis

Reason answer is correct: The normal position for PRZ level control is 459/460 or position 2. In this position Level Channel 459 is the controlling channel providing input to the PRZ level control system. The power supply to level channel 459 is Instrument Bus (IDP) 1A-SI.

A. Correct.

B. Incorrect. Plausible since level channel 460 is powered from SII and the control switch is in the 459/460 position, however this is incorrect because 460 will only provide input to the protection circuitry for PRZ Level.

C. Incorrect. Plausible since level channel 461 is powered from SIII and it is available to provide input to the PRZ level control circuitry, however this is incorrect because the control switch is in the 459/460 position, therefore channel 461 will not provide input to the control or protection circuitry for PRZ Level.

D. Incorrect. Plausible since the logic circuitry for certain plant systems (Safety Injection for example) only use channels supplied by 3 of the 4 instrument bus power supplies therefore the candidate may misapply this knowledge to the PRZ level control system and determine that SIV is a power supply to the controlling PRZ level channel, however this is incorrect because the PRZ level control system is only provided inputs powered from SI, SII or SIII.

004 Chemical and Volume Control / 2

004K2.06; Knowledge of bus power supplies to the following:Control instrumentation

(CFR: 41.7)

Importance Rating:	2.6 2.7
Technical Reference:	AOP-024, Attachment 1, Pg 28, Rev 57 AOP-024-BD, Section 1.0, Pg 5, Rev 21
References to be provided:	None
Learning Objective:	PRZLC Lesson Plan, Obj 3.a
Question Origin:	New
Comments:	RO Q11 and RO Q29 have been looked at for double jeopardy and as an exam team we have determined that there is an adequate difference to NOT be a concern.
Tier/Group:	T2/G1

2018 NRC RO 030/BANK/C/A//OP-185/NONE//004 K6.31/

Given the following plant conditions:
- At 0700 the unit is operating at 100% power
  - 'A' CSIP is running

- At 0701 the following occurs:
  - 'A' CSIP experiences a speed changer malfunction. As a result, discharge pressure and flow lowers.
  - Seal injection flow to all RCPs lowers to 3.4 gpm

- At 0705:
  - Seal injection flow to all RCPs remains at 3.4 gpm

Which ONE of the following identifies the operational status and impact on the Alternate Seal Injection (ASI) pump based on these conditions?

The ASI pump is ______ .

A. not running but is in standby for a subsequent auto start

B.✓ not running and will not start automatically due to lockout

C. running and operators must take local readings of RCP Seal Injection flow

D. running and must be secured promptly to prevent overpressurizing the discharge piping

Plausibility and Answer Analysis

Reason answer is correct: The loss of flow to (2) out of three (3) flow switches will initiate three (3) timers; The first timer will automatically detonate the two (2) squib valves (1ASI-21 & 1ASI-22) after a 2.5 minutes time delay. The second timer will start the ASI pump after 2.75 minutes elapsed time. The third timer will stop the pump from running if the flow to two (2) out of three (3) flow switches hasn't been restored above 4.0 gpm, within 3.5 minutes elapsed time. The third timer will also prevent the ASI pump from being started after 3.5 minutes has elapsed and flow is not restored. If flow is restored above 4.0 gpm prior to 2.5 minutes, all three timers will reset and no initiation of the pump or squib valves will occur. However, if flow is restored above 4.0 gpm after the system timer has actuated local actions will be required to restore the system based on the time elapsed. For example if the 2.5 minute timer actuates and flow is restored before the 2.75 minute timer actuates, the squib valves will be fired by the 2.5 minute timer, but the ASI pump will not start because the 2.75 and 3.5 minute timers will have reset.

A. Incorrect. Plausible since the values for the ASI pump timers are 2.5 and 2.75 minutes, the candidate may misapply the values of the timers to the seal injection values and determine that the ASI system has not met the requirements to automatically start and therefore remain in standby, however this is incorrect because 4 gpm is the minimum seal injection value that must be restored to prevent the ASI system from automatically actuating during a low seal injection flow condition.

B. Correct.

C. Incorrect. Plausible since the 2.5 and 2.75 minute timers have elapsed, the squib valves and the ASI pump should have actuated to restore seal injection flow, however this is incorrect because 4 minutes have elapsed and the ASI system will automatically lockout the pump if seal injection flow is not restored before the 3.5 minute timer elapses.

D. Incorrect. Plausible since the 2.5 and 2.75 minute timers have elapsed, the squib valves and the ASI pump should have actuated and seal injection flow has not been restored the candidate may determine that because the ASI pump is a positive displacement pump damage to the system piping may occur if the ASI pump continues to run, however this is incorrect because 4 minutes have elapsed and the ASI system will automatically lockout the pump if seal injection flow is not restored before the 3.5 minute timer elapses.

004 Chemical and Volume Control / 2

004K6.31; Knowledge of the effect of a loss or malfunction on the following CVCS components: Seal injection system and limits on flow range

(CFR: 41.7 / 45.7)

Importance Rating:	3.1 3.5
Technical Reference:	OP-185, P&L #1, Section 5.2 Note, Pg 4 and 9, Rev. 12
References to be provided:	None
Learning Objective:	ASI Lesson Plan, Obj. 6
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 031/BANK/FUNDAMENTAL//EOP-ES-1.3/NONE/2013 NRC RO 31/005 A1.05/

------------------------Which ONE of the following identifies (1) the MINIMUM Containment Wide Range sump level required to place the RHR system in Cold Leg Recirculation in accordance with EOP-ES-1.3, Transfer To Cold Leg Recirculation AND (2) the basis for this level?

A. (1) 137.5 inches

(2) ensures the recirculation sump pH level is acceptable

B. (1) 137.5 inches

(2) ensures the recirculation sump strainers are completely submerged

C. (1) 142 inches

(2) ensures the recirculation sump pH level is acceptable

D.✓ (1) 142 inches

(2) ensures the recirculation sump strainers are completely submerged

Plausibility and Answer Analysis

Reason answer is correct: Following a LOCA large enough to require transfer to the recirculation phase of the ECCS, containment water level is expected to be above the strainer modules ("top hats") and the vortex breakers located inside the recirculation sump (Design calculations for the limiting case predict a water level of 223 FEET, 8 INCHES, approximately 142 INCHES as read on the wide range containment sump level instruments or 87% as read on the recirculation sump level instruments.). Degraded sump performance could only occur if water level fell below this level or the strainer modules experienced excessive clogging.

A. Incorrect. The first part is plausible because both the RHR pumps and the Containment Spray pumps take a suction from the containment sump and the 137.5 inch level is the correct amount to allow the Containment Spray pumps to take a suction from the recirculation sump. The second part is plausible since both the Containment Spray and RHR systems will be in operation for a LOCA event in which the RWST is depleted to the point that shifting suctions the to the Containment Sump is required. With Containment Spray in operations the Spray Additive tank will be part of the suction supply to the pump. It is reasonable to believe that the sodium hydroxide in the tank requires a minimum amount of water level to ensure that the chemicals when mixed with the LOCA water in the Containment sumps combine to maintain the correct pH level.

B. Incorrect. The first part is plausible because both the RHR pumps and the Containment Spray pumps take a suction from the containment sump and the 137.5 inch level is the correct amount to allow the Containment Spray pumps to take a suction from the recirculation sump. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since both the Containment Spray and RHR systems will be in operation for a LOCA event in which the RWST is depleted to the point that shifting suctions the to the Containment Sump is required. With Containment Spray in operations the Spray Additive tank will be part of the suction supply to the pump. It is reasonable to believe that the sodium hydroxide in the tank requires a minimum amount of water level to ensure that the chemicals when mixed with the LOCA water in the Containment sumps combine to maintain the correct pH level.

D.Correct.

005 Residual Heat Removal / 4

005A1.05; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Detection of and response to presence of water in RHR emergency sump

(CFR: 41.5 / 45.5)

Importance Rating:	3.3 3.3
Technical Reference:	EOP-ES-1.3, Attachment 1, Pg 34, Rev. 2
References to be provided:	None
Learning Objective:	EOP-LP-2.3/3.3, Obj. 5.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 032/NEW/C/A//FR-C.2, BKGRD, CSFST/NONE/EARLY/006 K3.02/

Given the following plant conditions:

- A Small Break LOCA occurred
- 6.9 KV Bus 1B-SB is de-energized
- The 'A' CSIP has tripped
- Core Exit Thermocouple (CET) temperatures are 735°F
- RCS pressure is 985 psig
- Containment pressure is 15 psig
- RVLIS Full Range level is 46%

Which ONE of the following complete the statement below?

Based on the conditions above, the Reactor Vessel water level is __(1)__ the Top Of Active Fuel AND the Core Cooling Critical Safety Function Status Tree is __(2)__ .

A. (1) above

(2) Orange

B. (1) above

(2) Red

C✓ (1) below

(2) Orange

D. (1) below

(2) Red

Plausibility and Answer Analysis

Reason answer is correct: With no high head safety injection available, the cooling of the core becomes degraded. Degraded core cooling reduces RCS inventory below the top of active fuel when RVLIS Full Range indication is less than 63%. CSF-2 monitors for the symptoms of inadequate or degraded core cooling. With no RCPs in service the CSF-2 will be orange for these conditions because the CET temperature is greater than 730°F and RVLIS level is above 39%.

A. Incorrect. The first part is plausible since the EOP-FR-C.2 will monitor RCS void fraction vice the actual RCS water level with the RCPs in service. The RVLIS Dynamic Head Range level of 60% with 3 RCPs, 33% with 2 RCPs and 25% with 1 RCP in service requires no additional actions from the Functional Restoration procedure and the crew returns the procedure and step in effect and therefore the candidate may have a misconception that RCS water level is above the top of active fuel. This is incorrect however because all RCPs are secured when Containment pressure rises above 10 pounds (Phase B) and CCW to Containment is isolated. The second part is correct.

B. Incorrect. The first part is plausible since the EOP-FR-C.2 will monitor RCS void fraction vice the actual RCS water level with the RCPs in service. The RVLIS Dynamic Head Range level of 60% with 3 RCPs, 33% with 2 RCPs and 25% with 1 RCP in service requires no additional actions from the Functional Restoration procedure and the crew returns the procedure and step in effect and therefore the candidate may have a misconception that RCS water level is above the top of active fuel. This is incorrect however because all RCPs are secured when Containment pressure rises above 10 pounds (Phase B) and CCW to Containment is isolated. The second part is plausible since the CET temperature is above 730°F and RVLIS is below the dynamic range value of 60% for all RCPs, however this is incorrect because the RCPs were previously stopped and therefore RVLIS must be less than 39% in order to meet the Red conditions for CSF -2.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since the CET temperature is above 730°F and RVLIS is below the dynamic range value of 60% for all RCPs, however this is incorrect because the RCPs were previously stopped and therefore RVLIS must be less than 39% in order to meet the Red conditions for CSF -2.

006 Emergency Core Cooling 2/3

006K3.02; Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: Fuel

(CFR: 41.7 / 45.6)

Importance Rating:	4.3 4.4
Technical Reference:	ERG-BKGRD-FR-C.2, Introduction, Pg 1, Rev. 2 EOP-CSFST, CSF-2, Pg 2, Rev. 13
References to be provided:	None
Learning Objective:	EOP-LP-3.10, Obj. 1
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 033/MODIFIED/C/A//ALB-009/NONE//007 A1.03/

Given the following plant conditions:
- The unit is operating at 50% power
- The crew is responding to a leaking PRZ PORV

Time	TI-471.1	TI-463
0800	110°F	115°F
0805	115°F	155°F
0810	125°F	195°F
0815	145°F	235°F
0820	155°F	255°F

Which ONE of the following is the first time that annunciator ALB-009-8-1, PRT High-Low Level Press or Temp, will alarm?

Temperature Indicator Noun Name:
TI-471.1, PRZ Relief Tank Temperature
TI-463, PRZ PORV Line Temperature

A. 0805

B✓ 0810

C. 0815

D. 0820

Plausibility and Answer Analysis

Reason answer is correct: Annunciator ALB-009-8-1 has multiple inputs that causes the annunciator to alarm. One of which is high temperature which has a setpoint of 120°F as sensed by TI-471.1, PRZ Relief Tank Temperature. At 0810 the PRT temperature is above the temperature at which the annunciator will go into alarm.

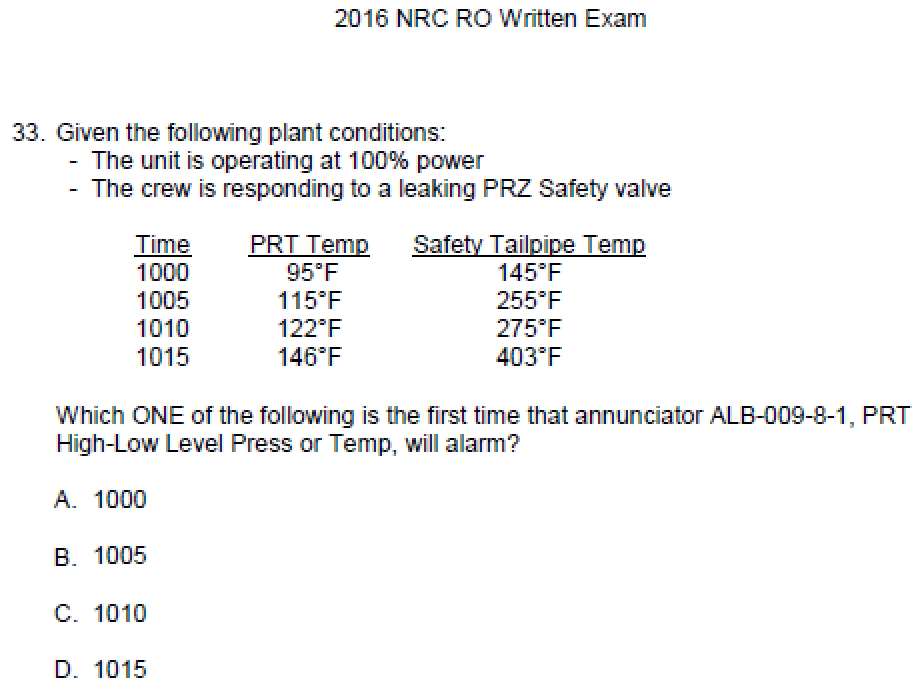
A. Incorrect. Plausible since at this PRT input temperature annunciator ALB-009-8-2, Pressurizer Relief Discharge High Temp goes into alarm. The alarm comes on when the PRZ PORV discharge line temperature exceeds 140°F as sensed by TI-463.

B. Correct.

C. Incorrect. Plausible since the candidate may mis-apply the setpoint of 140°F for the PRZ PORV discharge line high temperature annunciator ALB-009-8-2, Pressurizer Relief Discharge High Temp to TI-471.1, PRZ Relief Tank Temperature, however this is incorrect as PRZ PORV discharge line temperature is sensed by TI-463.

D. Incorrect. Plausible since at this temperature annunciator ALB-009-8-3, Pressurizer Safety Relief Discharge High Temp goes into alarm. The alarm comes on when the PRZ Safety valve discharge line temperature exceeds 250°F , however this is incorrect as it is sensed by TI-465, TI-467, or TI-469.

Original question:



2016 NRC RO Written Exam

33. Given the following plant conditions:
   - The unit is operating at 100% power
   - The crew is responding to a leaking PRZ Safety valve

Time	PRT Temp	Safety Tailpipe Temp
1000	95°F	145°F
1005	115°F	255°F
1010	122°F	275°F
1015	146°F	403°F

Which ONE of the following is the first time that annunciator ALB-009-8-1, PRT High-Low Level Press or Temp, will alarm?

A. 1000

B. 1005

C. 1010

D. 1015

007 Pressurizer Relief/Quench Tank / 5

007A1.03; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Monitoring quench tank temperature

(CFR: 41.5 / 45.5)

Importance Rating:	2.6 2.7
Technical Reference:	APP-ALB-009-8-1, Pg 29, Rev. 18
References to be provided:	None
Learning Objective:	PRZ Lesson Plan, Obj. 5.d
Question Origin:	Modified - 2016 NRC RO 33
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 034/NEW/FUNDAMENTAL//OP-100, 5-S-1301/NONE//007 A4.01/

Given the following plant conditions:
- The unit is operating at 100%
- ALB-009-8-1, Pressurizer Relief Tank High-Low Level Press or Temp, has just alarmed
- PRT level is at the low level alarm setpoint

Which ONE of the following completes the statements concerning PRT fill?

1RC-167, RMW to PRT shutoff valve __(1)__ to restore normal PRT level.

1RC-167 __(2)__ receive a signal to an automatic shut if a Phase A signal occurs.

A. (1) will open

(2) will

B. (1) will open

(2) will not

C. (1) must be manually opened

(2) will

D✓ (1) must be manually opened

(2) will not

Plausibility and Answer Analysis

Reason answer is correct: 1RC-167, RMW to PRT shutoff valve is an AOV that is operated by a switch on the Main Control Board. In accordance with OP-100, Reactor Coolant System Section 8.0 Pressuirzer Relief Tank Make Up, after verifying that 1RC-161, RMW TO PRT containment isolation valve is OPEN the operator then OPENS 1RC-167 until the desired level is reached then shuts 1RC-167.

1RC-167 does NOT receive a Phase A isolation signal. PRT make up Containment isolation valve 1RC-161 receives a shut signal on a Phase A.

A. Incorrect. The first part is plausible since other RCS systems auto-makeup when low level is reached (such as auto fill make up to the RCP stand pipes). The second part is plausible since this is the inside Containment valve to make up to the PRT but since there is a check valve between the PRT and 1RC-167 there is no Phase A isolation signal supplied to shut 1RC-167.

B. Incorrect. The first part is plausible since other RCS systems auto-makeup when low level is reached (such as auto fill make up to the RCP stand pipes). The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since this is the inside Containment valve to make up to the PRT but since there is a check valve between the PRT and 1RC-167 there is no Phase A isolation signal supplied to shut 1RC-167.

D. Correct.

007 Pressurizer Relief/Quench Tank / 5

007A4.01; Ability to manually operate and/or monitor in the control room: PRT spray supply valve

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	2.7 2.7
Technical Reference:	OP-100, Section 8.0, Pg 17, Rev. 44 Simplified Drawing 2165-S-1301 Sheet 2
References to be provided:	None
Learning Objective:	PRZ Lesson Plan, Obj. 5.a
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 035/BANK/FUNDAMENTAL//OP-145/NONE//008 K4.07/

Which ONE of the following completes the statements below?

The 'C' CCW pump can be powered from either __(1)__ .

A __(2)__ is a design feature that prevents two CCW pumps from being aligned to the same power supply.

A. (1) 6.9 KV Emergency Bus 1A-SA or 6.9 KV Emergency Bus 1B-SB

(2) manual transfer switch

B.✓ (1) 6.9 KV Emergency Bus 1A-SA or 6.9 KV Emergency Bus 1B-SB

(2) key-operated interlock

C. (1) 480V Emergency Bus 1A2-SA or 480V Emergency Bus 1B2-SB

(2) manual transfer switch

D. (1) 480V Emergency Bus 1A2-SA or 480V Emergency Bus 1B2-SB

(2) key-operated interlock

Plausibility and Answer Analysis

Reason answer is correct: The 'C' CCW pump can be powered from either the 6.9kV Bus 1A-SA or 1B-SB and a key interlock prevents racking in if 'A' or 'B' CCW pump is racked in on the same bus.

A. Incorrect. Plausible since the power source is correct. The second part is plausible since the 'C' CSIP has a manual transfer switch which is allows for rapid pump swaps per OP-107 if required.

B. Correct.

C. Incorrect. Plausible since the RHR, Containment Spray Pump and Chiller P-4 which are all safety related equipment, are powered from these buses. The second part is plausible since the 'C' CSIP has a manual transfer switch which is allows for rapid pump swaps per OP-107 if required.

D. Incorrect. Plausible since the RHR, Containment Spray Pump and Chiller P-4 which are all safety related equipment, are powered from these buses. The key interlock prevents racking the breaker in if either 'A' or 'B' CCW pump is racked in on the bus.

008 Component Cooling Water / 8

008K4.07; Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Operation of the CCW swing-bus power supply and its associated breakers and controls

(CFR: 41.7)

Importance Rating:	2.6 2.7
Technical Reference:	OP-145, P&L #10, Pg 8, Rev. 62
References to be provided:	None
Learning Objective:	CCW Lesson Plan, Obj. 2.e
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 036/NEW/C/A//AOP-019/NONE//010 K5.02/

Given the following plant conditions:
- The unit is operating at 100% power
- The PRZ pressure master controller, PK-444A, is in AUTOMATIC
- The PRZ pressure master controller setpoint rapidly fails to 61%

Which ONE of the following completes the statements below?

Both spray valves will ___(1)___ .

The fluid enthalpy across the spray valves ___(2)___ .

A. (1) open

(2) lowers

[✓ B.] (1) open

(2) remains constant

C. (1) close

(2) lowers

D. (1) close

(2) remains constant

Plausibility and Answer Analysis

Reason answer is correct: The master PRZ controller is set to 66.88% during normal operation. With the controller in Automatic reducing the setpoint to 61% will raise the output signal from 25 % to 54.4 %. An output signal above 40.62% will send a signal for PRZ Spray valve to begin to throttle open until the output signal in 71.87% at which the valves are full open. The fluid in the PRZ spray line is subcooled water and the enthalpy remains constant as the fluid is throttled through the PRZ Spray valves.

A. Incorrect. The first part is correct. The second part is plausible since the PRZ system undergoes a phase change in order to providing the quenching of the PRZ Vapor space and reduce PRZ pressure, however this is incorrect because the PRZ Spray fluid is subcooled RCS fluid from the A and B loops and the constant enthalpy expansion through the valve results in no enthalpy change.

B. Correct.

C. Incorrect. The first part is plausible since the setpoint change is approximaitely 5% if the candidate misapplies the setpoint thumbrule as a 1 for 1 change the 5% change will raise the output to 30% which is less than 40.62% and therefore the PRZ spray valve would remain shut, however this is incorrect because the change should raise the output signal from 25% to 54.4% resulting in the PRZ Spray valves opening. The second part is plausible since the PRZ system undergoes a phase change in order to providing the quenching of the PRZ Vapor space and reduce PRZ pressure, however this is incorrect because the PRZ Spray fluid is subcooled RCS fluid from the A and B loops and the constant enthalpy expansion through the valve results in no enthalpy change.

D. Incorrect. The first part is plausible since the setpoint change is approximaitely 5% if the candidate misapplies the setpoint thumbrule as a 1 for 1 change the 5% change will raise the output to 30% which is less than 40.62% and therefore the PRZ spray valve would remain shut, however this is incorrect because the change should raise the output signal from 25% to 54.4% resulting in the PRZ Spray valves opening. The second part is correct.

010 Pressurizer Pressure Control / 3

010K5.02; Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: Constant enthalpy expansion through a valve

(CFR: 41.5 / 45.7)

Importance Rating:	2.6 3.0
Technical Reference:	AOP-019, Attachment 2, Pg 20, Rev. 25
References to be provided:	None
Learning Objective:	PZRPC Lesson Plan, Obj. 4.b
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 037/BANK/C/A//E-0/NONE//012 G2.4.1/

Given the following plant conditions:
- The unit is operating at 90% power
- Power Range NI-41 is under clearance and OWP-RP-23 is complete

Subsequently the following occur:
- Multiple system annuciators are received
- Power Range NI-43 fails high

Which ONE of the following completes the statement below?

The FIRST operator action required to be performed is to _______ immediate actions.

A. place Rod Control in manual in accordance with, AOP-001, Rod Control Malfunctions

B. place Rod Control in manual in accordance with, AOP-024, Loss Of Uninterruptible Power Supply

C✓ verify that the Reactor is tripped in accordance with EOP-E-0, Reactor Trip Or Safety Injection

D. manually trip the Reactor using either MCB switches in accordance with EOP-FR-S.1, Response To Nuclear Power Generation/ATWS

Plausibility and Answer Analysis

Reason answer is correct: With OWP-RP-23 in place the bistable for NI-41 is in the tripped condition. NI-43 failing high will satisfy the 2/4 logic for the high flux Reactor trip and the Reactor should trip. EOP-E-0 immediate actions require the operator to verify a Reactor trip occurs.

A. Incorrect. Plausible since the NI-43 has failed high and Rod Control is auctioneered high the system will generate a continuous withdrawal signal and placing the rods in manual is immediate action number two of AOP-001, however this is incorrect because a reactor trip signal has been generated and entry into EOP-E-0 is required.

B. Incorrect. Plausible since multiple systems are in alarm and a failure of NI-43 are indicative of the failure of UPS power supply S-III and placing the rods in manual is immediate action number one of AOP-024, however this is incorrect because a reactor trip signal has been generated and entry into EOP-E-0 is required.

C. Correct.

D. Incorrect. Plausible since the first immdiate action of EOP-FR-S.1 is to verify the Reactor is tripped and manually insert negative reactivity if the reactor is not tripped automatically or manually, however this is incorrect because the actions directed by EOP-FR-S.1 are only required if the action to trip the Reactor in EOP-E-0 are not successful.

012 Reactor Protection / 7

012G2.4.1; Knowledge of EOP entry conditions and immediate action steps.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	4.6 4.8
Technical Reference:	EOP-E-0, Step 1 RNO, Pg 4, Rev. 7
References to be provided:	None
Learning Objective:	RPS Lesson Plan, Obj. 10
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 038/BANK/C/A//AOP-024-BD, ALB-015/NONE//012 K3.01/

Given the following plant conditions:
- The unit is operating at 100% power
- SSPS Train A, General Warning Light is in due to maintenance, but no other bistables have been affected

Subsequently a loss of Instrument Bus S-IV occurs

Which ONE of the following describes the resultant condition of the Control Rod Drive Stationary Gripper Coils?

A. De-energized based on the loss of one instrument bus ONLY.

B✓ De-energized based on both trains having a General Warning condition.

C. Energized due to the General Warning condition blocking the Reactor trip signal.

D. Enegerized due to redundant Instrument Bus power supplies.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-024 if all of the following conditions exist, a reactor trip or safeguards actuation may occur

- Power lost to any Instrument Bus
- Any ESFAS or RPS instrument loops out of service

Because the A SSPS train is in Test a General Warning condition is present 1 of 2 conditions for the General Warning RPS Trip logic to open the Reactor Trip breakers is satisfied. Once power is lost to Instrument Bus IV the B SSPS train will have lost AC power to the output relay cabinet resulting in a General Warning condition. Now the RPS logic for the 2 of 2 General Warning conditions are present and the logic to open the Reactor Trip breakers are satisfied which will remove power to the CRDM Stationary Gripper coils allowing control rods to be inserted into the Reactor.

A. Incorrect. Plausible since the loss of Instrument Bus IV results in the loss of the channel IV RPS inputs which will generate a signal to open the Reactor Trip breakers and de-energize the stationary coils , however this is incorrect because the loss of only an instrument bus does not satisfy the 2/4 logic to open the Reactor Trip breakers.

B. Correct.

C. Incorrect. Plausible since the RPS is designed to have the output signal of certain parameters blocked (i.e. SRNI and IRNI signals), however this is incorrect because the General Warning condition does not have the capability to be blocked.

D. Incorrect. Plausible since the single failure design criteria of RPS allow for the loss of one power supply from causing or preventing an actuation of the system, however this is incorrect because testing is also in progress and under these conditions will generate a signal to open the Reactor Trip breakers.

012 Reactor Protection / 7

012K3.01; Knowledge of the effect that a loss or malfunction of the RPS will have on the following:CRDS

(CFR: 41.7 / 45.6)

Importance Rating:	3.9 4.0
Technical Reference:	AOP-024-BD, Section 1.0, Step 15, Pg 4, Rev. 21 APP-ALB-015, Window 1-3, Pg 5, Rev. 30
References to be provided:	None
Learning Objective:	RPS Lesson Plan, Obj. 11.a
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 039/PREVIOUS/FUNDAMENTAL//AOP-024-BD/NONE/2016 NRC RO 39/013 K2.01/

Which ONE of the following completes the statement below?

Instrument Buses ___(1)___ AND ___(2)___ provide power to the ESFAS Slave Relays.

A. (1) SI

(2) SII

B. (1) SII

(2) SIII

C.✓ (1) SI

(2) SIV

D. (1) SIII

(2) SIV

Plausibility and Answer Analysis

Reason answer is correct: *Train ESFAS slave relays are powered from Instrument Bus SI (SIV). A loss of SI or SIV will result in a loss of ESFAS functions driven by slave relays for that train.*

A. Incorrect *Plausible since the ESFAS relays are powered from the safety instrument buses. If SI or SIII is lost, the MCB controller for MDAFW pump flow control valves will not be operable; flow control valves will not shut on an AFW isolation signal and will not open on an auto open signal. Train ESFAS slave relays are powered from Instrument Bus SI (SIV). A loss of SI or SIV will result in a loss of ESFAS functions driven by slave relays for that train. A loss of SI will cause a loss of 'A' Train ONLY the question is asking for BOTH 'A' and 'B' Train.*

B.Incorrect *Plausible since the ESFAS relays are powered from the safety instrument buses. If power is lost to Instrument Bus SII (B Train and TDAFW) or SIII (A Train) the associated AFW pump suction pressure instrument will read low. If the AFW pump is running, it will not trip on Lo-Lo suction pressure nor will it be prevented from being started. Additionally, if power is lost to Instrument Bus SII (B Train) or SIII (A Train), the associated CNMT Spray Additive Tank level indicators will read empty but their associated CNMT Spray Chemical Addition Valve will not automatically shut. If necessary, the valve(s) may be manually operated.*

C.Correct

D. Incorrect *Plausible since the ESFAS slave relays are powered from Instrumtment Bus SI (SIV). To answer this question it would take BOTH SI and SIV and only one of the two (SIV) are listed. If power is lost to Instrument Bus SII (B Train and TDAFW) or SIII (A Train) the associated AFW pump suction pressure instrument will read low. If the AFW pump is running, it will not trip on Lo-Lo suction pressure nor will it be prevented from being started. Train ESFAS slave relays are powered from Instrument Bus SI (SIV). A loss of SI or SIV will result in a loss of ESFAS functions driven by slave relays for that train.*

013 Engineered Safety Features Actuation / 2

013K2.01; Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

(CFR: 41.7)

Importance Rating:	3.6 3.8
Technical Reference:	AOP-024-BD, Section 1.0, Pg 2, Rev. 20
References to be provided:	None
Learning Objective:	ESFAS Lesson Plan, Obj. 2
Question Origin:	Previous 2016 NRC RO 39 radomly selected
Comments:	None
Tier/Group:	T2G1

2018 NRC RO 040/BANK/C/A//OWP-ESF/NONE//013 K5.02/

Given the following plant conditions:

- The unit is operating at 100% power
- Containment Pressure Channel I pressure indication was oscillating and has been removed from service in accordance with OWP-ESF, Engineered Safety Feature Actuation.

Which ONE of the following identifies the logic from the remaining channels for HI-1 and HI-3 Containment Pressure actuations AFTER Channel I is REMOVED from service?

	HI-1 SI Actuation	HI-3 CNMT Spray Actuation
A.	1 / 2	2 / 3
B.	1 / 2	1 / 3
C.✓	2 / 3	2 / 3
D.	2 / 3	1 / 3

Plausibility and Answer Analysis

Reason answer is correct: HI-1 uses only CNMT pressure channels II, III & IV and is normally a 2/3 logic (at 3 psig in the Containment) so the logic remains unchanged when channel I is tripped. For HI-3, all four channel of CNMT pressure are used with a normal 2/4 logic at 10 psig in the Containment. Unlike all other RPS/ESFAS bistables, when a HI-3 bistable is removed from service, the channel is bypassed.

A. Incorrect. The first part is plausible if the candidate thinks that the HI-1 actuation uses all 4 channels of CNMT pressure. The second part is correct.

B. Incorrect. The first part is plausible if the candidate thinks the HI-1 actuation uses all 4 channels of CNMT pressure. The second part is plausible if the candidate thinks the HI-3 actuation is tripped when it is removed from service.

C. Correct.

D. Incorrect. First part is correct The second part is plausible if the candidate thinks the HI-3 actuation is tripped when it is removed from service

013 Engineered Safety Features Actuation / 2

013K5.02; Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Safety system logic and reliability

(CFR: 41.5 / 45.7)

Importance Rating:	2.9 3.3
Technical Reference:	OWP-ESF-01, Pg 4 and 8, Rev. 21
References to be provided:	None
Learning Objective:	ESFAS Lesson Plan, Obj. 10.b
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 041/BANK/C/A//USERS GUIDE, E-0/NONE/2009B NRC RO 41/022 A3.01/

Given the following plant conditions:

- The unit experiences a Reactor Trip and SI concurrent with a Loss Of Offsite Power
- During the performance of EOP-E-0, Reactor Trip Or Safety Injection, Attachment 3, the BOP notes the following alignment for Containment Fan Coolers:
  - 'A' Train - one fan per unit running in fast speed
  - 'B' Train - one fan per unit running in slow speed

Which ONE of the following identifes the actions required by the BOP?

A✓ Shift the two 'A' Train fans from fast to slow speed.

B. Shift the two 'B' Train fans from slow to fast speed.

C. Start two additional 'A' Train fans in fast speed and secure the 'B' Train fans.

D. Start two additional 'B' Train fans in slow speed and secure the 'A' Train fans.

Plausibility and Answer Analysis

Reason answer is correct: During an SI all fans operating in high (i.e. fast) speed trip and one fan in each Fan Cooler will receive an automatic LOW speed start signal through the Sequencer at load block 2. If this signal fails in accordance with EOP-E-0, Attachment 3 step 16, the BOP should secure the fast speed fans and shift them to slow speed so that one fan per unit is running in slow speed.

A. Correct.

B. Incorrect. Plausible since this alignment would be used following a loss of offsite power, however the SI alignment requires that the fans operate in low speed.

C. Incorrect. Plausible because the normal fan alignment is to have all fans in one train running, however the SI alignment requires that the fans operate in low speed.

D. Incorrect. Plausible because the normal fan alignment is to have all fans in a train running, however the SI alignment requires that the fans operate in low speed.

022 Containment Cooling / 5

022A3.01; Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation

(CFR: 41.7 / 45.5)

Importance Rating:	4.1 4.3
Technical Reference:	EOP-User's Guide, Section 6.7, Pg 37, Rev. 49 EOP-E-0, Attachment 3, Step 16, Pg 61, Rev. 7
References to be provided:	None
Learning Objective:	CCS Lesson Plan, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 042/BANK/FUNDAMENTAL//OP-112/NONE//026 A4.05/

Given the following plant conditions:
- An automatic Containment Spray Actuation Signal (CSAS) has occurred.
- Containment Pressure has lowered to <3 psig.

Which ONE of the following is correct in order to reset the CSAS?

A. Containment Spray pump control switches A-SA AND B-SB must be positioned to STOP prior to resetting.

B. Either Containment Spray Train A OR B reset switch must be taken to RESET to completely reset the CSAS.

C.✓ Both Containment Spray Trains A AND B reset switches must be taken to RESET to completely reset the CSAS.

D. Containment Spray pump discharge valves 1CT-50 AND 1CT-88 must be positioned to SHUT prior to resetting.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with OP-112, Containment Spray System section 7.0, to place system for standby operation following Manual or Automatic initiation at the MCB the Containment Spray Train A and Train B reset switch is taken to RESET. Then the Spray pumps are secured and the valves realigned to complete the standy alignment.

A. Incorrect. Plausible since stopping both CNMT spray pumps is the step to be performed AFTER the reset switches are taken to reset.

B. Incorrect. Plausible since Phase A requires only one switch to be operated for the manaul actuation of the ESF signals. Additionally a student may have a misconception about OP-112 step 7.1.2 since there is only one step but the step is repeated to reset Train A and then Train B.

C. Correct.

D. Incorrect. Plausible since shutting the CNMT spray pump discharge valves are required to be performed in the next steps of the OP-112 to align the system for standby operation.

026 Containment Spray / 5

026A4.05; Ability to manually operate and/or monitor in the control room: Containment spray reset switches

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	3.5 3.5
Technical Reference:	OP-112, Section 7.0, Pg 11, Rev. 45
References to be provided:	None
Learning Objective:	CSS Lesson Plan, Obj. 4.b
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 043/PREVIOUS/FUNDAMENTAL//STUDENT TEXT/NONE/2016 NRC RO 44/039 A3.02/

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

Plausibility and Answer Analysis

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 to modulates open if header pressure is > 5 psig and therefore the valve would be shut on a turbine trip, however the ventilating valve 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 to modulates open if header pressure is > 5 psig and therefore the valve would be shut on a turbine trip, however the ventilating valve 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, however this is incorrect.

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:	MT Student text MSR Student text
References to be provided:	None
Learning Objective:	MT Lesson Plan, Obj. 9 MSR Lesson Plan, Obj. 4.e
Question Origin:	Previous 2016 NRC RO 44 radomly selected
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 044/NEW/C/A//AOP-042/NONE//039 K4.06/

Given the following plant conditions:

- The unit is operating at 97% power to perform OST-1080, Auxiliary Feedwater Pump 1X-SAB Full Flow Test Quarterly Interval Mode 1, 3
- The TDAFW Pump is running

Subsequently:

- The RAB AO reports there is a steam line leak in a steam supply line to the TDAFW pump
- Reactor power rises to 100.1% and stablilizes

Which ONE of the following completes the statements below?

The AOP that should be entered FIRST to address this event is __(1)__.

Reverse steam flow is prevented on the steam supply lines going to the TDAFW pump through the use of __(2)__ valve(s).

Procedure Title:
AOP-038, Rapid Downpower
AOP-042, Secondary Steam Leak / Efficiency Loss

A. (1) AOP-038

(2) check

B. (1) AOP-038

(2) a trip

C.✓ (1) AOP-042

(2) check

D. (1) AOP-042

(2) a trip

Plausibility and Answer Analysis

Reason answer is correct: Entry conditions have been met for AOP-042 with a notification to the MCR of a secondary steam leak. Addtionally, AOP-042 will have the operators attempt to isolate the leak (could be done by shutting 1MS-70 or 1MS-72 or both). The Main Steam Lines supplying motive steam to the TDAFW pumps have check valves installed downstream of the individual steam line isolation valves (1MS-70 from "B" SG and 1MS-72 from "C" SG).

A. Incorrect. The first part is plausible since AOP-038 is used to rapidly lower Reactor power (one of the entry condtions is any condition requiring > 5 MW/min load reductions). With Reactor power just above 100% (currently at 100.01%) it is > 100% but would not require a reduction of load > 5 MW/min to reduce power to < 100%. Additionally, AOP-042 is written to address the steam leak where AOP-038 is not. The second part is correct.

B. Incorrect. The first part is plausible since AOP-038 is used to rapidly lower Reactor power (one of the entry condtions is any condition requiring > 5 MW/min load reductions). With Reactor power just above 100% (currently at 100.01%) it is > 100% but would not require a reduction of load > 5 MW/min to reduce power to < 100%. Additionally, AOP-042 is written to address the steam leak where AOP-038 is not. The second part is plausible since the Trip and Throttle valve automatically shuts to prevent the Turbine Driven AFW pump from overspeeding by isolating steam flow from the Main Steam supply lines.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since the Trip and Throttle valve automatically shuts to prevent the Turbine Driven AFW pump from overspeeding by isolating steam flow from the Main Steam supply lines.

039 Main and Reheat Steam / 4

039K4.06; Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Prevent reverse steam flow on steam line break

(CFR: 41.7)

Importance Rating:	3.3 3.6
Technical Reference:	AOP-042, Section 2.0, Pg 3, Rev. 6 Simplified Drawing 2165-S-0542, Rev 27 (Red circles show check valves installed in the MS lines, Blue circle shows where lines join and are not seperate lines to the TDAFW pump)
References to be provided:	None
Learning Objective:	MSSS Lesson Plan, Obj. 5 AOP-LP-3.42, Obj. 1
Question Origin:	New
Comments:	Discuss the use of Turbine Driven AFW Pump Main Steam supply lines as a possible topic to meet K/A as check valves in the TDAFW Pump steam supply lines prevent back flow during a steam break, but the MS system has no such design features. Phonecon 6/13: Dan agrees that using the Turbine Driven AFW Pump Main Steam supply lines is acceptable to meet this K/A.
Tier/Group:	T2/G1

2018 NRC RO 045/NEW/FUNDAMENTAL//AOP-010/NONE//059 A2.12/

Given the following plant conditions:

- The unit is operating at 7% power
- Main Feedwater Regulating Bypass valves are shut and all Main Feedwater Regulating valves are in automatic operation
- 'B' Main Feedwater Regulating Valve is oscillating causing 'B' SG level to fluctuate from 54% to 58%

Which ONE of the following completes the statements below concerning this failure?

In accordance with AOP-010, Feedwater Malfunctions, the operator should ___(1)___ .

An Automatic Reactor trip would occur if 'B' SG level reaches a setpoint of ___(2)___ .

A. (1) place 'B' Feed Reg valve in manual

(2) 30%

B✓ (1) place 'B' Feed Reg valve in manual

(2) 25%

C. (1) initiate AFW flow to maintain SG level

(2) 30%

D. (1) initiate AFW flow to maintain SG level

(2) 25%

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-010 step 1 RNO actions the operator should place the affected Feedwater Reg valve in Manual. In accordance with EOP-E-0 an automatic Reactor Trip will occur from SG Low-Low Water Level of 25%.

A. Incorrect. The first part is correct. The second part is plausible since AOP-010 and OMM-001, Attachment 13 trip limits for low SG level is 30%

B. Correct.

C. Incorrect. The first part is plausible since this would be the action to take IF feed flow was not maintained to all three steam generators. (AOP-010, Step 8 RNO 8.b.1) however this is incorrect because only the B SG level is effected. The second part is plausible since AOP-010 and OMM-001, Attachment 13 trip limits for low SG level is 30%

D. Incorrect. The first part is plausible since this would be the action to take IF feed flow was not maintained to all three steam generators. (AOP-010, Step 8 RNO 8.b.1) however this is incorrect because only the B SG level is effected. The second part is correct.

059 Main Feedwater / 4

059A2.12; Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of feedwater regulating valves

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.1 3.4
Technical Reference:	EOP-E-0, Attachment 10, Pg 81, Rev. 7 AOP-010, Step 1 RNO, Pg 4, Rev. 39
References to be provided:	None
Learning Objective:	AOP-LP-3.10, Obj. 2
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 046/PREVIOUS/C/A//ALB-014, AOP-010/NONE/2014 NRC RO 48/061 K3.02/

Given the following plant conditions:
- A unit heat up is in progress in accordance with GP-002, Normal Plant Heatup From Cold Solid To Hot Subcritical Mode 5 To Mode 3
- 'A' MDAFW pump is feeding the SGs

Subsequently the following annunciator alarms:
- ALB-017-5-4, Aux Feedwater Pump 'A' Trip Or Close CKT Trouble

Which ONE of the following completes the statements below?

SG levels will lower to ___(1)___ where the 'B' MDAFW pump will automatically start, to restore SG levels.

Entry in to AOP-010, Feedwater Malfunctions, ___(2)___ required.

(Assume NO Operator actions)

A. (1) 20%

(2) is

B. (1) 20%

(2) is NOT

C. (1) 25%

(2) is

D.✓ (1) 25%

(2) is NOT

Plausibility and Answer Analysis

Reason answer is correct:
With any SG level less than 25% (2 out of 3 NR detectors), the following occurs:
(1) Reactor trip
(2) Both MDAFW Pumps start
(3) All three MDAFW FCVs receive an auto open signal

AOP-010 Entry Conditions are any Main Feedwater or Condensate System malfunction causing a flow transient and may also be entered as directed by other approved procedures.

A. Incorrect. The first part is plausible since the AMSAC system generates a start signal to the AFW pumps when it is actuated at 20% SG level, however the AMSAC system is not is service until power is above 35%. The second part is plausible since AOP-010 is the abnormal procedure used to address SG level problems, however the procedure is designed to address issues caused by the loss of Main feedwater or Condensate.

B. Incorrect. The first part is plausible since the AMSAC system generates a start signal to the AFW pumps when it is actuated at 20% SG level, however the AMSAC system is not is service until power is above 35%. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since AOP-010 is the abnormal procedure used to to address SG level problems, however the procedure is designed to address issues caused by the loss of Main feedwater or Condensate.

D. Correct.

061 Auxiliary/Emergency Feedwater / 4

061K3.02; Knowledge of the effect that a loss or malfunction of the AFW will have on the following: S/G

(CFR: 41.7 / 45.6)

Importance Rating:	4.2 4.4
Technical Reference:	APP-ALB-014, Window 5-4B, Pg 32, Re.v 26 AOP-010, Section 2.0, Pg 3, Rev. 39
References to be provided:	None
Learning Objective:	AFW Lesson Plan, Obj. 7.a AOP-LP-3.10, Obj. 1
Question Origin:	Previous 2016 NRC RO 48 radomly selected
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 047/NEW/C/A//OP-155/OP-155, ATTACHMENT 9//062 A1.01/

Given the following plant conditions:

- OST-1073, 1B-SB Emergency Diesel Generator Operability Test Monthly Interval Modes 1-2-3-4-5-6, is in progress
- EDG 1B-SB indications are as follows:
  - Megawatts 6.4 MW
  - Megavars 3.4 MVar

Subsequently after an electrical transient, EDG 1B-SB indications are as follows:

- Megawatts 7.2 MW
- Megavars 3.8 MVar

Which ONE of the following competes the statement below?

The Emergency Diesel ___(1)___ Overload limit is exceeded and the operator must reduce load ___(2)___ .

(Reference Provided)

A✓ (1) Engine

(2) immediately

B. (1) Engine

(2) within 2 hours

C. (1) Generator

(2) immediately

D. (1) Generator

(2) within 2 hours

Plausibility and Answer Analysis

Reason answer is correct: In accordance with OP-155 P&L 28, the continuous and overload ratings are not to be exceeded. At 7.2 Megawatts and 3.8 Megavars the Engine 2 Hour Overload Limit of 7150 KW has been exceeded. Based on this condition the operator should immediately take action to reduce the diesel load to within the requirements of Attachment 9.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since operation at the Generator and Engine overload limit is allowed for 2 hours within a 24 hour period the candidate may misinterpet the capacity curve and determine these limits maybe exceeded for 2 hours in all cases, however this is incorrect because the Engine overload limit of 7150 KW is exceeded and operation above this limit is not allowed.

C. Incorrect. The first part is plausible since the Generator continuous load limit is being exceeded for these conditions, however this is incorrect because only the Engine overload limit is being exceeded for these conditions and operation above this limit is not allowed. The second part is correct.

D. Incorrect. The first part is plausible since the Generator continuous load limit is being exceeded for these conditions, however this is incorrect because only the Engine overload limit is being exceeded for these conditions and operation above this limit is not allowed. The second part is plausible since operation at the Generator and Engine overload limit is allowed for 2 hours within a 24 hour period the candidate may misinterpet the capacity curve and determine these limits maybe exceeded for 2 hours in all cases, however this is incorrect because the Engine overload limit of 7150 KW is exceeded and operation above this limit is not allowed.

062 AC Electrical Distribution / 6

062A1 .01; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significance of D/G load limits

(CFR: 41.5 / 45.5)

Importance Rating:	3.4 3.8
Technical Reference:	OP-155, Precaution and Limitation 28, Pg 10, Rev. 86 OP-155, Attachment 9, Pg 192, Rev. 86
References to be provided:	None
Learning Objective:	Diesel Lesson Plan, Obj. 6
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 048/NEW/C/A//AOP-025/NONE//062 G2.4.45/

Given the following plant conditions:

- The unit was operating at 100% power with 'A' Safety Train Equipment in service

Subsequently:

- A fault occurred on Unit Aux Transformer 'A'
- The bus transfer for 6.9KV Aux Bus 'A' and 'D' failed to occur
- The 'A' EDG failed to start
- The crew is performing EOP-ES-0.1, Reactor Trip Response, and AOP-025, Loss of One Emergency AC Bus (6.9KV) or One Emergency DC Bus (125V)

It has been two minutes since the unit has tripped and the following annunciators are locked in:

- ALB-005-8-2, CCW Pump B Disch Header Low Press
- ALB-023-1-18, Chiller WC2-A Trouble

Which ONE of the following identifies THE PRIORITY that the condition which caused these annunciators will be addressed in accordance with AOP-025 AND whether the annunciator is EXPECTED or NOT EXPECTED for the plant conditions?

A. 1. CCW Pump B Disch Header Low Press — Expected
   2. Chiller WC2-A Trouble — NOT Expected

B.✓ 1. CCW Pump B Disch Header Low Press — NOT Expected
   2. Chiller WC2-A Trouble — Expected

C. 1. Chiller WC2-A Trouble — Expected
   2. CCW Pump B Disch Header Low Press — NOT Expected

D. 1. Chiller WC2-A Trouble — NOT Expected
   2. CCW Pump B Disch Header Low Press — Expected

Plausibility and Answer Analysis

Reason answer is correct: Priority: Priorities of AOP-025, Check for power, check for availability of cooling water to an EDG that has started on low bus voltage, then check CCW cooling to plant equipment, then check a Chiller running to provide cooling to ventilation in safety related areas. Expectation of annunciators: The CCW Pump 'B' Discharge Header Low Pressure is NOT expected because the 'B' CCW pump should have started on low pressure. The standby pump auto-starts on low discharge pressure of 52 psig sensed on its respective discharge header (PT 649 or 650). It is expected that the Chiller WC2-A Trouble annunciator will be locked in since the 1A-SA Emergency Bus is de-energized by the openng of breaker 105 and the failure of the EDG 1A-SA failure to start. Therefore the annuciator should be locked in. AOP-025 addresses the CCW Pump before addressing the Chiller.

A. Incorrect. 1. The priority of addressing the annunciators is correct.
2. Expectation of annunciators: The CCW Pump low pressure annunciator is plausible if there is a misconception that the 'B' CCW pump must be manually started (like a CSIP) since the 'A' CCW pump is without power system pressure will lower to < 52 psig at which time the 'B' CCW pump will auto start and the low pressure annunciator should clear when system pressure is restored. Annunciators: Chiller Trouble annunciator not expected - plausible since the 480 volt control power is provided by 2 phases of the AC from the power supply the breakers remain closed when the high side supply breaker is open. Because the breaker status is unchanged the the input for a breaker trip does not occur and the trouble alarm is not recieved. This is incorrect because the Chiller WC2-A is a 6.9KV breaker with DC control power which will allow the breaker to trip open on the loss of power resulting in a trouble alarm.

B. Correct.

C. Incorrect. 1. The priority of addressing the Chiller Trouble annunciator prior to the CCW low flow is plausible since this both issues are addressed in AOP-025 but they are listed in the opposite order of what is in the procedure.
2. Expectation of annunciators is correct.

D. Incorrect.
1. The priority of addressing the Chiller Trouble annunciator prior to the CCW low flow is plausible since this both issues are addressed in AOP-025 but they are listed in the opposite order of what is in the procedure.
2. Expectation of annunciators: Chiller Trouble annunciator not expected - plausible since the 480 volt control power is provided by 2 phases of the AC from the power supply the breakers remain closed when the high side supply breaker is open. Because the breaker status is unchanged the the input for a breaker trip does not occur and the trouble alarm is not recieved. This is incorrect because the Chiller WC2-A is a 6.9KV breaker with DC control power which will allow the breaker to trip open on the loss of power resulting in a trouble alarm. CCW annunciator expected - plausible since the 'A' CCW pump would have lost power and the annunciator would have initially have alarmed but should be clear after 2 minutes have elapsed from the event initiation. The 'B' CCW pump auto started on low system pressure (52 psig). After the 'B' CCW pump started it would have cleared the low pressure. The pump start and pressure restoration would have been completed prior to the 2 minutes that have elapsed.

062 AC Electrical Distribution / 6

062G2.4.45; 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:	AOP-025, Section 3.1, Step 5 and 10, Pg 8 and 9, Rev. 42
References to be provided:	None
Learning Objective:	AOP-LP-3.25, Obj. 5.a
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 049/BANK/FUNDAMENTAL//6-G-0042 SH 01/NONE/2011 NRC RO 50/063 K1.02/

Which ONE of the following describes the normal power source for a safety-related 125-V DC bus?

480-V MCC through a ______ .

A✓ battery charger to the DC bus

B. 7.5-KVA inverter to the DC bus

C. battery charger, through the DC battery, and then to the DC bus

D. 7.5-KVA inverter, through the DC battery, and then to the DC bus

Plausibility and Answer Analysis

Reason answer is correct: 125VDC Bus is supplied directly by battery chargers

A. Correct.

B. Incorrect. Plausible because the 7.5-KVA inverter has a DC power supply and is safety related equipment.

C. Incorrect. Plausible because the battery charger supplies the DC battery

D. Incorrect. Plausible because the 7.5-KVA inverter has a DC power supply and is safety related equipment.

063 DC Electrical Distribution / 6

063K1.02; Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: AC electrical system

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating:	2.7 3.2
Technical Reference:	Control Wire Drawing 2166-G-042, Sheet 1, 250V DC, 125V DC, & 120V UPS One Line Diagram
References to be provided:	None
Learning Objective:	DCP Lesson Plan, Obj. 2.b
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 050/BANK/FUNDAMENTAL//OP-155, AOP-025-BD/NONE//064 K1 .04/

Given the following plant conditions:
- The unit is operating at 100% power
- A loss of DC Bus 1A-SA occurs

Which ONE of the following completes the statements below regarding operation of the 'A' EDG?

The Governor and Generator Excitation circuits will be __(1)__ .

The EDG Output breaker __(2)__ be closed from the MCB.

A. (1) de-energized

(2) can

B.✓ (1) de-energized

(2) can NOT

C. (1) energized

(2) can

D. (1) energized

(2) can NOT

Plausibility and Answer Analysis

Reason answer is correct: Part 1: OP-155, Diesel Generator Emergency Power System, has a section to restore the Governor after DC power is lost. Additionally, notes are in the procedure for loss of DC power to the EDG. The unavailability of 125 VDC will prevent the generator to flash and electronic speed control is NOT available if there is a loss of 125 VDC control power. AOP-025 Basis document section 3.3 step 2 also provides indication that the governor will lose power with a loss of DC. Part 2: The EDG Output breaker is a 6.9 KV breaker that is remotely operated via 125 VDC power. The effects of losing 125 VDC to a 6.9 KV breaker are as follows:

No power to closing coil (can't shut breaker remotely)
No power to trip coil (can't open breaker remotely)
No power to charge the closing spring (depending on the status of the breaker before the loss of 125 VDC, may only get 1 close and 1 open cycle out of breaker before manually changing closing spring

A. Incorrect. Plausible since the Governor and Excitation circuits will de-energize, however to operate the EDG Output breaker from the MCB requires DC Control Power.

B. Correct

C. Incorrect. Plausible if the candidate has a misconception that the Governor, Excitation, and Control power circuits are supplied with power from the AC Electrical Distribution System similar to the Main Generator.

D. Incorrect. Plausible if the candidate has a misconception that the Governor and Excitation circuits are supplied by the AC Electrical Distribution System similar to the Main Generator.

064 Emergency Diesel Generator / 6

064K1 .04; Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: DC distribution system

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating:	3.6 3.9
Technical Reference:	OP-155, Procedure Note, Pg 91, Rev. 86 AOP-025-BD, Section 2.0, Pg 53, Rev. 19
References to be provided:	None
Learning Objective:	AOP-LP-3.25, Obj. 5
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 051/BANK/FUNDAMENTAL//AOP-005-BD/NONE/2004 NRC RO 40/073 K4.01/

Given the following plant conditions:

- The unit is operating at 40% power
- REM-*1WC-3544, WPB CCW Hx Inlet Monitor, is in HIGH alarm

As a result of the high alarm, which ONE of the following will automatically close?

A. 3WC-321, Waste Evap B Condenser CCW FCV-329 (3WC-320) Outlet Isolation

B✓ 3WC-4, WPB CCW Surge Tank Overflow

C. 3WC-197, WG Compressor A CCW Outlet

D. 3WC-7, WPB CCW Surge Tank Drain

Plausibility and Answer Analysis

Reason answer is correct: A high rad alarm on this monitor will cause the surge tank overflow to isolate to prevent the release of radioactivity.

A. Incorrect. Plausible since this valve does automatically close if RCS leakage to the CCW thermal barrier HX is occurring, but closes on high flow.

B. Correct.

C. Incorrect. Plausible since the valve is normally throttled open and if shut would stop any flow from the system to a potential leak path but the valve does not have an auto close signal it must be manually shut.

D. Incorrect. Plausible since this valve does have an automatic action associated with the surge tank, but the valve closes on a low level.

073 Process Radiation Monitoring / 7

073K4.01; Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint

(CFR: 41.7)

Importance Rating:	4.0 4.3
Technical Reference:	AOP-005-BD, Section 1.0, Pg 3, Rev. 12
References to be provided:	None
Learning Objective:	AOP-LP-3.05, Obj. 4
Question Origin:	Bank
Comments:	HNP was not able to create a valid question for process radiation monitors for K/A 073K5.02, Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Radiation intensity changes with source distance. Requested new K/A. Phonecon 6/13: HNP previously suppressed this K/A dealing with HNP Process Radiation Monitors associated with a source distance relationship with liquid or gaseous monitors, so selected a new K/A, keeping 073 and randomly selecting from the remaining items for this K/A: New K/A 073K4.01: Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint
Tier/Group:	T2/G1

2018 NRC RO 052/BANK/C/A//AOP-022-BD/NONE//076 K2.04/

Given the following plant conditions:
- The unit is operating at 100% power
- NSW Pump 'A' is operating
- NSW Pump 'B' is in standby
- Breaker 104, AUX BUS D TO EMERGENCY BUS A-SA, trips

After one minute, which ONE of the following describes the expected Service Water system alignment?

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

No ESW pumps are running.

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

No ESW pumps are running.

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

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

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

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

Plausibility and Answer Analysis

Reason answer is correct: Power will lost to the 1A-SA Emergency AC Bus until supplied by the 'A' EDG. ESW Pump 'A' will start on Program A (Undervoltage - Load Block 3) and supply the 'A' ESW header. NSW Pump 'A' is powered from Auxiliary Bus 1D and will remain running.

A. Incorrect. Plausible since the candidate may not recognize that the sequencer started the ESW Pump 'A' in Load Block 3.

B. Incorrect. Plausible since the candidate may have a misconception that NSW Pump 'A' lost power and that NSW Pump 'B' auto started. Also, ESW Pump 'A' was started by the sequencer in Load Block 3.

C. Correct.

D. Incorrect. Plausible since the candidate may have a misconception that NSW Pump 'A' lost power and that NSW Pump 'B' auto started. Also, the second part of the distractor is correct in that ESW Pump 'A' will be supplying the 'A' ESW header.

076 Service Water / 4

076K2.04; Knowledge of bus power supplies to the following: Reactor building closed cooling water

(CFR: 41.7)

Importance Rating:	2.5 2.6
Technical Reference:	AOP-022-BD, Section 1.0, Pg 2, Rev. 14 HNP Electrical Load List, Pg 3, Rev. 0
References to be provided:	None
Learning Objective:	SWS Lesson Plan, Obj. 2.a
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 053/NEW/FUNDAMENTAL//AOP-017, AOP-017-BD/NONE//078 A4.01/

Given the following plant conditions:

- The unit is operating at 100% power
- Annunciator ALB-002-8-1, Instrument Air Low Press, alarms
- The OATC checks MCB pressure indication PI-9751.1, Instrument Air Header Pressure, and identifies that air pressure is slowly lowering

Which ONE of the following completes the statements below?

The MCB pressure indicator PI-9751.1 __(1)__ always indicative of pressure throughout the Instrument Air system.

At __(2)__ psig RCS letdown flowpath valves will begin to fail to mid-position.

A. (1) is

(2) 75

B. (1) is

(2) 85

C. (1) is not

(2) 75

D✓ (1) is not

(2) 85

Plausibility and Answer Analysis

Reason answer is correct: In the case of the Instrument Air system the system is piped to multiple locations in various buildings. The indication in the MCR is the pressure near and downstream of the IA compressors. This pressure may not be indicative of the pressure throughout the entire system. AOP-017 basis document informs the operator that in a loss of IA event the location of the pressure indicator may not indicate the pressure in all parts of the system and it may be lower than indicated on the MCB. AOP-017 Attachment 7 states at 85 psig RCS letdown flowpath valves begin to fail to mid-position.

A. Incorrect. The first part is plausible since MCB indications are what is used to accurately determine and monitor system parameters. The second part is plausible since 75 psig is less than the pressure at which the letdown orifice isolation valve will begin to shut, however this is incorrect because most air valves in the direct letdown path require 85 psig for full stroke, and may begin to fail to mid-position as pressure falls below that value.

B. Incorrect. The first part is plausible since MCB indications are what is used to accurately determine and monitor system parameters. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since 75 psig is less than the pressure at which the letdown orifice isolation valve will begin to shut, however this is incorrect because most air valves in the direct letdown path require 85 psig for full stroke, and may begin to fail to mid-position as pressure falls below that value.

D. Correct.

078 Instrument Air / 8

078A4.01; Ability to manually operate and/or monitor in the control room: Pressure gauges

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating: 3.1 3.1

Technical Reference: AOP-017, Section 3.0, Step 1 Note, Pg 4, Rev. 40
AOP-017, Attachment 7, Pg 57, Rev. 40
AOP-017-BD, Section 2.0, Pg 7, Rev. 15
AOP-017-BD, Section 3.0, Pg 14, Rev. 15

References to be provided: None

Learning Objective: ISA Lesson Plan, Obj. 6 and 9

Question Origin: New

Comments: None

Tier/Group: T2/G1

2018 NRC RO 054/NEW/C/A//EOP-E-1/NONE//103 A2.03/

Given the following plant conditions:
- The unit is operating at 100% power
- A LOCA in Containment results in peak pressure to rise to 7 psig

Which ONE of the following completes the statements below?

1SI-287, Accumulator and PRZ PORV N2 Supply Isolation Valve, will__(1)__ .

EOP-E-1, Loss of Reactor Or Secondary Coolant, requires operating the __(2)__ .

A. (1) automatically shut

(2) BOTH Phase A and Phase B reset switches

B.✓ (1) automatically shut

(2) Phase A reset switch ONLY

C. (1) remain open

(2) BOTH Phase A and Phase B reset switches

D. (1) remain open

(2) Phase A reset switch ONLY

Plausibility and Answer Analysis

Reason answer is correct: A Phase 'A' (T) signal, Containment Phase A isolation, shuts the following valves associated with the SI Accumulators:

1SI-179, Accumulator fill line
1SI-287, Accumulator nitrogen supply line
1SI-263, Accumlator test line isolation
1SI-264, Accumlator test line to RWST
and 1SP-78, 1SP-81, 1SP-84 and 1SP-85, Accumulator sample valves

In accordance with EOP-E-1 prior to opening 1SI-287 Phase A and Phase B isolation signals if actuated will be reset (step 17). Containment pressure peaked above 10 psig which would have caused a Containment Isolation Phase A signal at 3 psig and a Phase B signal at 10 psig. ONLY the Phase A signal will need to be reset to restore control of 1SI-287.

A. Incorrect. The first part is correct. The second part is plausible since EOP-E-1 has the operator reset BOTH the Phase A and Phase B signals <u>if actuated</u> but ONLY the Phase A signal reset is required to restore control of Phase A valves.

B. Correct.

C. Incorrect. The first part is plausible since the Cold Leg Accumulator discharge valves all receive OPEN signals when a Safety Injection signal is generated. Since Containment pressure exceeded 3 psig a SI signal would have occurred. The second part is plausible since EOP-E-1 has the operator reset BOTH the Phase A and Phase B signals <u>if actuated</u> but ONLY the Phase A signal reset is required to restore control of Phase A valves.

D. Incorrect. The first part is plausible since the Cold Leg Accumulator discharge valves all receive OPEN signals when a Safety Injection signal is generated. Since Containment pressure exceeded 3 psig a SI signal would have occurred. The second part is correct.

103 Containment / 5

103A2.03; Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.5 3.8
Technical Reference:	OMM-004, Attachmen4, Pg 48, Rev. 41 EOP-E-1, Step 17 and 18, Pg 18 and 20, Rev. 4
References to be provided:	None
Learning Objective:	SIS Lesson Plan, Obj. 7.b
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 055/NEW/C/A//TECH SPEC 3.6.1.3/NONE//103 A2.05/

The Containment Personnel Airlock inner door is stuck shut and can not be opened.

In accordance with OP-113, Penetration Pressurization System And Containment Air Locks, access to the Containment requires __(1)__ in order to gain access to Containment through the EMERGENCY Air Lock doors.

The action required in accordance with Technical Specification 3.6.1.3, Containment Systems - Containment Air Locks, for the failure of the Containment PERSONNEL Airlock is to verify the outer door is closed __(2)__ .

A. (1) operating a hydraulic control station

(2) immediately

B. (1) turning a manual handwheel

(2) immediately

C. (1) operating a hydraulic control station

(2) within 1 hour

D✓ (1) turning a manual handwheel

(2) within 1 hour

Plausibility and Answer Analysis

Reason answer is correct:

A. Incorrect. Part 1 is plausible since the Personnel Airlock is operated with a hydraulic control station that automatically opens the door. Part 2 is plausible since immediate action (to evaluate overall Containment leakage rate) is required if one or more Containment air locks are inoperable for reasons other than Tech Spec 3.6.1.4.a or 3.6.1.4.b

B. Incorrect. Part 1 is correct. Part 2 is plausible since immediate action (to evaluate overall Containment leakage rate) is required if one or more Containment air locks are inoperable for reasons other than Tech Spec 3.6.1.4.a or 3.6.1.4.b

C. Incorrect. Part 1 is plausible since the Personnel Airlock is operated with a hydraulic control station that automatically opens the door. Part 2 is correct.

D. Correct.

103 Containment / 5

103A2.05; Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency containment entry

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	2.9 3.9
Technical Reference:	OP-113, Section 8.3, Pg 23, Rev. 23 Technical Specification 3.6.1.3
References to be provided:	None
Learning Objective:	Containment Lesson Plan, Obj. 4.c and 9.d
Question Origin:	New
Comments:	K/A match by meeting the higher order part of the K/A
Tier/Group:	T2/G1

2018 NRC RO 056/NEW/C/A//AOP-018-BD/NONE//002 K6.07/

Given the following plant conditions:
- The unit is operating at 40% power
- PRZ Backup Heaters are energized

Subsequently:
- The 'B' RCP trips due to an electrical fault

Which ONE of the following completes the statements below?

As 'B' RCP coasts down, Loops 'A' and 'C' steam flow will rise by _______ .

(Assume that Reactor power and Tavg have remained constant)

A. 16.5%

B. 25%

C. 33.3%

D.✓ 50%

Plausibility and Answer Analysis

Reason answer is correct: In accordance with the AOP-018, Reactor Coolant Pump Abnormal Conditions, if an RCP is stopped at power, Reactor power and T_{avg} should remain constant. The steam flow from the RCS loop with the stopped RCP will be insignificant, so the steam flow in the other two RCS loops will increase by 50%. These changes in steam flow and reactor coolant flow will cause the ΔT in the unaffected RCS loops to increase by 50% for a constant Reactor power.

A. Incorrect. Plausiblie since each RCP provides 1/3 of the total flow during normal operation and the flow is now split between the two remaining RCPs.

B. Incorrect. Plausible since the candidate may have the misconception that half of the 50% rise in flow is shared between the two RCPs.

C. Incorrect. Plausible since each RCP provides 1/3 of the total flow during normal operation, however this is incorrect because the remaining loops share half of the load from the 1/3 flow that was being provided by the loss of the RCP.

D. Correct.

002 Reactor Coolant / 2/4

002K6.07; Knowledge of the effect or a loss or malfunction on the following RCS components: Pumps

(CFR: 41.7 / 45.7)

Importance Rating:	2.5 2.8
Technical Reference:	AOP-018-BD, Section 1.0, Step 7, Pg 2, Rev 25
References to be provided:	None
Learning Objective:	AOP-LP-3.18, Obj. 4
Question Origin:	New
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 057/NEW/FUNDAMENTAL//AOP-024/NONE//015 K2.01/

Which ONE of the following identifies the power supply to NI-36, Intermediate Range Nuclear Instrument?

A. SI

B✓ SII

C. SIII

D. SIV

Plausibility and Answer Analysis

Reason answer is correct: The power supply to NI-36, Intermediate Range Nuclear Instrument, is Instrument Bus (IDP) 1B-SII.

A. Incorrect. Plausible since Instrument buses provide power to all nuclear instruments and are typically labelled based on the channel they are supplied from i.e. Channel I is supplied from IDP 1A-SI, however this convention cannot be applied to NI-36 as HNP only has 4 Instrument buses.

B. Correct.

C. Incorrect. Plausible since Instrument buses provide power to all nuclear instruments and are typically labelled based on the channel they are supplied from i.e. Channel III is supplied from IDP 1A-SIII, however this convention cannot be applied to NI-36 as HNP only has 4 Instrument buses.

D. Incorrect. Plausible since Instrument buses provide power to all nuclear instruments and are typically labelled based on the channel they are supplied from i.e. Channel IV is supplied from IDP 1B-SIV, however this convention cannot be applied to NI-36 as HNP only has 4 Instrument buses.

015 Nuclear Instrumentation / 7

015K2.01; Knowledge of bus power supplies to the following: NIS channels, components, and interconnections

(CFR: 41.7)

Importance Rating:	3.3 3.7
Technical Reference:	AOP-024, Attachment 2, Sheet 1, Pg 29, Rev. 57
References to be provided:	None
Learning Objective:	NIS Lesson Plan, Obj 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 058/BANK/FUNDAMENTAL//DBD-301/NONE//016 K5.01/

Which ONE of the following describes the process instrumentation interface with Solid-State Protection System (SSPS) for separation of indication/control circuits from protection circuits?

A. Separate detectors are used to generate indication/control and protection signals.

B✓ Isolation amplifiers are used to separate indication/control from protection signals.

C. Separate instrument channels are used for indication/control and protection signals.

D. Redundant lag filters are used to separate indication/control from protection signals.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with the Reactor Control and Protection system design basis document where redundant protective channels are combined to provide non-protective functions, the required signals are derived through isolation amplifiers. These devices are designed so that open or short circuit conditions, as well as application of 120 vac or 140 vdc, to the isolated side of the circuit will have no effect on the input or protection side of the circuit.

A. Incorrect. Plausible since systems such as the Safeguards Sequencer system use separate relays to detect the loss of power to each safety bus in order to determine which program the sequencer should run for the accident conditions, however this is incorrect because isolation amplifiers are used in the SSPS system to separate the control and protectino signals.

B. Correct.

C. Incorrect. Plausible since systems such as the PRZ Pressure control system have separate channels for control and protection, however this is incorrect because isolation amplifiers are used in the SSPS system to separate the control and protectino signals.

D. Incorrect. Plausible since systems such as the Rod control system use redundant half wave rectifiers to filter the command signals from the Slave Cycler to individual rod coils in order to move a control rod in the Reactor, however this is incorrect because isolation amplifiers are used in the SSPS system to separate the control and protectino signals.

016 Non-Nuclear Instrumentation / 7

016K5.01; Knowledge of the operational implication of the following concepts as they apply to the NNIS: Separation of control and protection circuits

(CFR: 41.5 / 45.7)

Importance Rating:	2.7 2.8
Technical Reference:	DBD-301, Step 4.6, Pg 35, Rev. 7
References to be provided:	None
Learning Objective:	PSPR-LP-3.5, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 059/BANK/C/A//FSAR OP-170 AOP-005/NONE/2012 NRC RO 62/033 A1.02/

Given the following conditions:

- The unit is operating at 100% power
- The suction pipe from Spent Fuel Pool 'B' to the Spent Fuel Pool Cooling Pump completely severs
- Spent Fuel Pool area radiation monitor RM-*1FR-3566A-SA is in HIGH alarm and monitor RM-*1FR-3567B-SB is in ALERT

Which ONE of the following completes the statements below?

Level in the Spent Fuel Pool will stabilize at a MAXIMUM of ___(1)___ above the fuel assemblies.

These Radiation monitor conditions will cause ___(2)___ train(s) of Fuel Handling Ventilation Emergency Exhaust to automatically start.

[✓ A.] (1) 18'

(2) 'A' ONLY

B. (1) 18'

(2) both 'A' and 'B'

C. (1) 23'

(2) 'A' ONLY

D. (1) 23'

(2) both 'A' and 'B'

Plausibility and Answer Analysis

Reason answer is correct: The location and design of the suction pipe prevents the Spent Fuel Pool from draining less than 18 feet above the top of the spent fuel assemblies which is 5 feet below the normal level of 23' above the fuel. Any one of a train's ARMs in HIGH alarm in the FHB will start 1 train of FHB emergency exhaust. Since only the 'A' Train is in HIGH alarm the 'A' train will auto start.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible because in some actuations it requires a 2 / 2 logic but for Fuel Handling Ventilation it only requires a 1 / 2 (one Rad monitor) in HIGH alarm from either train to stop both Normal Supply fan A and B

C. Incorrect. The first part is plausible because 23 feet is the correct level the Fuel Pool is required to be maintained above the Fuel by Tech Specs and the A train Exhaust fan will auto start . The second part is correct.

D. Incorrect. The first part is plausible because 23 feet is the correct level the Fuel Pool is required to be maintained above the Fuel by Tech Specs. The second part is plausible because in some actuations it requires a 2 / 2 logic but for Fuel Handling Ventilation it only requires a 1 / 2 (one Rad monitor) in HIGH alarm from either train to stop both Normal Supply fan A and B

033 Spent Fuel Pool Cooling / 8

033A1.02; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including: Radiation monitoring systems

(CFR: 41.5 / 45.5)

Importance Rating:	2.8 3.3
Technical Reference:	FSAR Chapter 9, Section 9.1.3.2, Pg 9.1.3-3, Amend. 56 OP-170, Section 8.1, Pg 23, Rev. 37 AOP-005-BD, Section1.0, Pg 3, Rev. 8
References to be provided:	None
Learning Objective:	FPC Lesson Plan, Obj. 7.a
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 060/BANK/FUNDAMENTAL//AOP-013/NONE//034 A4.01/

Given the following plant conditions:

- The unit is in Mode 6 with fuel movements in progress
- A fuel assembly has been slightly damaged during removal from the core
- Radiation levels are rising steadily and are currently as follows:

REM-01LT-3502A-SA, CNMT RCS Leak Detection Monitor, is in HIGH ALARM

Which ONE of the following completes the statements below?

Normal Containment Purge ___(1)___ automatically isolate.

Containment Pre-Entry Purge ___(2)___ automatically isolate.

A. (1) will

(2) will

B✓ (1) will

(2) will NOT

C. (1) will NOT

(2) will

D. (1) will NOT

(2) will NOT

Plausibility and Answer Analysis

Reason answer is correct: AOP-013, Fuel Handling Accident section 3.2, Fuel Handling Accident In Containment step 7 has the operator check REM-01LT-3502A-SA, Cnmt RCS Leak Detection Monitor, to verifiy that the HIGH alarm clear. If the high alarm is NOT clear the operator verifies that Normal Containment Purge is isolated.

A. Incorrect. Plausible since the misconception could be that when REM-01LT-3502A-SA, Cnmt RCS Leak Detection Monitor, is in HIGH ALARM it effects both Normal and Pre-Entry Purge.

B. Correct.

C. Incorrect. Plausible if the candidate has a misconception about the auto actions associated with the given rad monitor in HIGH alarm will cause. REM-01LT-3502A-SA, Cnmt RCS Leak Detection Monitor, is in HIGH ALARM will ONLY cause the Normal Containment Purge to isolate. Since there isn't a name "Normal or Pre-Entry Purge" associated with this rad monitors name this misconception could occur. There is a separate Rad monitor associated with Pre-Entry Purge, REM-01LT-3502B, Cnmt Pre-Entry Purge Monitor. If this monitor goes into HIGH ALARM then the Pre-Entry Purge would isolate but NOT the Normal Purge.

D. Incorrect. Plausible if there is a misconception that it would require a 2 of 2 conicidence (both REM-01LT-3502A and 3502B) to cause the Normal or Containment Pre-Entry Purge to isolate and at this time ONLY one Rad monitor is in high alarm.

034 Fuel-Handling Equipment / 8

034A4.01; Ability to manually operate and/or monitor in the control room: Radiation levels

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating: 3.3 3.7

Technical Reference: AOP-013, Section 3.2, Step 7, Pg 12 and 13, Rev. 16

References to be provided: None

Learning Objective: AOP-LP-3.13, Obj. 4

Question Origin: Bank

Comments: None

Tier/Group: T2/G2

2018 NRC RO 061/BANK/C/A//STEAM TABLES/STEAM TABLES//035 A3.02/

What should RCS temperature stabilize at if a Reactor Trip occurred with a Main Steam Line Isolation and all three SG PORV in automatic with their controller setpoints set at 89%?

A. 557°F

B. 558°F

C. 561°F

D✓ 564°F

Plausibility and Answer Analysis

Reason answer is correct: The SG PORV's controller has a range of 0-1300 psig. With the setting of 89% the PSAT (RCS Temperature) is calculated to be 564°F

0.89 x 1300 = 1157 + 15 = 1172 psia PSAT for 1172 is approximately 564°F

A. Incorrect. Plausible since this is the normal temp after Rx trip with Stm Dumps available but a MS Line Isolation has closed the MSIV's so Steam Dumps are not available.

B. Incorrect. Plausible since this is the normal RCS temp with the SG PORV setpoint at 85% which is what the normal at power PORV setpoint is adjusted to.

C. Incorrect. Plausible since this is the temperature you would calculate if you subracted 15 psi instead of added it.

D. Correct.

035 Steam Generator / 4

035A3.02; Ability to monitor automatic operation of the S/G including: MAD valves

(CFR: 41.7 / 45.5)

Importance Rating:	3.7 3.6
Technical Reference:	GP-005, Section 6.0, Step 5.e, Pg 20 and 56, Rev. 102 Steam Tables
References to be provided:	None
Learning Objective:	MSSS Lesson Plan, Obj. 5.a
Question Origin:	Bank
Comments:	Phonecon 6/9: HNP was not familiar with the term 'MAD' Valve and requested clarification of the term. Per discussion with Dan Bacon 'MAD' was determined to be an acronym for Manual/Automatic Depressurization and a question should be written to address the automatic operation of SG PORVs:
Tier/Group:	T2/G2

2018 NRC RO 062/NEW/C/A//ALB-511/NONE//071 G2.1.30/

Given the following plant conditions:

- The Gaseous Waste Processing System is receiving purge flow from the VCT gas space
- ALB-511-2-2, OARC-1119 Product Gas HI-HI Oxygen O_2 Shutdown, is received
- 3WG-400, Recombiner B TCV-1114B, is shut
- 3WG-401, O_2 Addition Valve HACV-1118B, is shut
- 1CS-137, Volume Control Tank Purge Valve, is shut

Which ONE of the following completes the statement below?

The WG Catalytic H_2 Recombiner B has stopped due to a ___(1)___ Trip.

To restore purge flow to the VCT gas space 3WG-401, O_2 Addition Valve HACV-1118B, must be reset from the ___(2)___ .

A. (1) Fast

(2) Radwaste Control Room

B✓ (1) Fast

(2) Recombiner B Control Panel

C. (1) Slow

(2) Radwaste Control Room

D. (1) Slow

(2) Recombiner B Control Panel

Plausibility and Answer Analysis

Reason answer is correct: Nitrogen is aligned to the VCT to purge oxygen from the VCT until the oxygen level in the VCT is below 4%. Once the oxygen level is below 4% hydrogen is aligned to replace the nitrogen in the VCT gas space to support normal online operation.

A. Incorrect. The first part is correct. The second part is plausible since the Radwaste Control Room has switch indications for 1CS-137, along with keylock switches to shut effluent relief isolation valves, however this is incorrect because 3WG-401, O_2 Addition Valve HACV-1118B is controlled locally at the Recombiner Control Panel.

B. Correct.

C. Incorrect. The first part is plausible since 3WG-401, O_2 Addition Valve HACV-1118B, will shut when a Slow Trip of the recombiner occurs, however this is incorrect because 1CS-137, VCT Purge Isolation valve will remain open. The second part is plausible since the Radwaste Control Room has switch indications for 1CS-137, along with keylock switches to shut effluent relief isolation valves, however this is incorrect because 3WG-401, O_2 Addition Valve HACV-1118B is controlled locally at the Recombiner Control Panel.

D. Incorrect. The first part is plausible since 3WG-401, O_2 Addition Valve HACV-1118B, will shut when a Slow Trip of the recombiner occurs, however this is incorrect because 1CS-137, VCT Purge Isolation valve will remain open. The second part is correct.

071 Waste Gas Disposal / 9

071G2.1.30; Ability to locate and operate components, including local controls.

(CFR: 41.7 / 45.7)

Importance Rating:	4.4 4.0
Technical Reference:	APP-ALB-511, Window 2-2, Pg 19 and 21, Rev. 21
References to be provided:	None
Learning Objective:	GWPS Lesson Plan, Obj 9
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 063/BANK/C/A//AOP-005-BD/NONE//072 K4.01/

Given the following plant conditions:

- The unit is operating at 100% power
- Pressurizer level and RCS pressure are lowering
- Containment temperature and pressure are rising

Containment Radiation Monitors read as follows:

- RM-01CR-3561ASA - high (RED) alarm
- RM-01CR-3561BSB - under clearance
- RM-01CR-3561CSA - does not respond to changing conditions
- RM-01CR-3561DSB - high (RED) alarm

Which ONE of the following completes the statements below?

Containment Ventilation Isolation Train A ___(1)___ automatically actuate.

Containment Ventilation Isolation Train B ___(1)___ automatically actuate.

A✓ (1) will

(2) will

B. (1) will

(2) will NOT

C. (1) will NOT

(2) will

D. (1) will NOT

(2) will NOT

Plausibility and Answer Analysis

Reason answer is correct: CVI occurs on 2/4 rad monitors in high alarm and generates both a Train A and a Train B CVI signal.

A. Correct.

B. Incorrect. Plausible since a Train B rad monitor is in high alarm, but both trains will receive a CVI signal.

C. Incorrect. Plausible since a Train A rad monitor is in high alarm, but both trains will receive a CVI signal.

D. Incorrect. Plausible since only a single rad monitor on each train is in alarm, but both trains receive a CVI signal when any 2/4 rad monitors alarm.

072 Area Radiation Monitoring / 7

072K4.01; Knowledge of ARM system design feature(s) and/or interlock(s) which provide for the following: Containment ventilation isolation

(CFR: 41.7)

Importance Rating:	3.3 3.6
Technical Reference:	AOP-005-BD, Section 1.0, Pg 2, Rev. 12
References to be provided:	None
Learning Objective:	AOP-LP-3.5, Obj. 1
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 064/BANK/FUNDAMENTAL//AOP-017, ALB-002/NONE/2009A NRC RO 64/079 K1.01/

Given the following plant conditions:
- A rupture in the Instrument Air system has occurred
- Instrument Air header pressure is 85 psig and lowering slowly

Which ONE of the following describes the status of 1SA-506, Instrument Air from Service Air Isolation Valve, and the status of ALB-002-8-1, Instrument Air Low Pressure Alarm?

	1SA-506	Alarm status
A.	OPEN	Lit
B.	OPEN	NOT Lit
C.	CLOSED	Lit
D✓	CLOSED	NOT Lit

Plausibility and Answer Analysis

Reason answer is correct: *1SA-506 shuts at 90 psig. The alarm however will not come in until 75 psig*

A Incorrect. 1SA-506 closes at 90 psig. Plausible if candidate confuses the setpoint for autoclosure with one of the other IA setpoints (101, 96, 95, 90, 85, 75, 60, and 35 are all significant Instrument Air Pressure Setpoints. See Attachment 7 of AOP-017). Alarm will NOT be lit. Plausible if candidate believes an alarm will alert operators to the condition prior to the automatic action occurring.

B Incorrect. 1SA-506 will shut automatically at 90 psig. Plausible if candidate confuses the setpoint for autoclosure with one of the other IA setpoints (101, 96, 95, 90, 85, 75, 60, and 35 are all significant Instrument Air Pressure Setpoints. See Attachment 7 of AOP-017)

C Incorrect. 1SA-506 will close automatically to isolate the Service Air System from the Instrument Air system, however the alarm is not lit until 75 psig. Plausible if candidate believes an alarm will alert operators to the condition prior to the automatic action occurring.

D. Correct.

079 Station Air / 8

079K1.01; Knowledge of the physical connections and/or cause-effect relationships between the SAS and the following systems: IAS

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating: 3.0 3.1

Technical Reference: AOP-017, Attachment 7, Pg 57, Rev. 40
APP-ALB-002, Window 8-4, Pg 43, Rev. 53

References to be provided: None

Learning Objective: AOP-LP-3.17, Obj. 2

Question Origin: Bank

Comments: None

Tier/Group: T2/G2

2018 NRC RO 065/BANK/FUNDAMENTAL//AOP-036.02/NONE//086 A2.04/

Given the following plant conditions:

- A fire occurs in the Electrical Penetration Area A on RAB 261' elevation
- Automatic fire suppression failed to actuate resulting in damage to the following:
  - The Electrical Penetration Area A Ionization detector
  - MCB level indicators LI-9010A1 SA, CST Level

Subsequently AOP-036, Safe Shutdown Following a Fire, is being implemented

Which ONE of the following completes the statements below?

In accordance with FPP-012-02-RAB261, RAB Elevation 261 Fire Pre-Plan, the alternate method of detecting a fire in Electrical Penetration Area A is a ___(1)___ detector.

In accordance with AOP-036.02, Fire Area: 1-A-BAL-A, 1-A-BAL-G, 1-A-BAL-H, CST level is determined by monitoring the AFW Pump ___(2)___ pressure.

A. (1) thermal

(2) discharge

B✓ (1) thermal

(2) suction

C. (1) ultraviolet

(2) discharge

D. (1) ultraviolet

(2) suction

Plausibility and Answer Analysis

Reason answer is correct: The RAB 261 elevation is monitored by Ionization and Thermal detector in accordance FPP-012-02-RAB261. Identifying the Thermal detectors as the alternate to the Ionization detection system predicts the impact the fire and the resultant damage it has caused. In accordance with AOP-036.02, the CST level is checked greater than 10% using LI-CE-9010A1SA, LI-CE-9010B1SB or Attachment 3, AFW Pump Suction Pressure vs. CST Level.

A. Incorrect. The first part is correct. The second part is plausible since pump discharge pressure oscillations are an indication of pump cavitation which can occur as a result of inadequate NPSH and NPSH is an indication of the level of the suction source of a pump, however this is incorrect because Attachment 3 compares the suction pressure of the AFW pump to determine the CST Level.

B. Correct.

C. Incorrect. The first part is plausible since the fire protection system consists of Thermal, Ionization, and Ultraviolet detectors, however this is incorrect because the Ultraviolet detectors are used primarily around Fuel Oil Storage Systems. The second part is plausible since pump discharge pressure oscillations are an indication of pump cavitation which can occur as a result of inadequate NPSH and NPSH is an indication of the level of the suction source of a pump, however this is incorrect because Attachment 3 compares the suction pressure of the AFW pump to determine the CST Level.

D. Incorrect. The first part is plausible since the fire protection system consists of Thermal, Ionization, and Ultraviolet detectors, however this is incorrect because the Ultraviolet detectors are used primarily around Fuel Oil Storage Systems. The second part is correct.

086 Fire Protection / 8

086A2.04; Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure to actuate the FPS when required, resulting in fire damage

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.3 3.9
Technical Reference:	FPP-012-02, Fire Pre-Plan A27, Pg 62, Rev. 13 AOP-036.02, Section 3.1, Step 9, Pg 15, Rev. 19
References to be provided:	None
Learning Objective:	AOP-LP-3.36, Obj. 3
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 066/BANK/FUNDAMENTAL//AD-OP-ALL-1000/NONE//G2.1.2/

An RO assigned as the OATC is responsible for which ONE of the following actions in accordance with AD-OP-ALL-1000, Conduct of Operations?

A✓ Uses diverse and redundant indications for verification of plant or equipment status.

B. Ensures engineered safety feature equipment operates without operator action.

C. Prioritizes focus and support for activities that ensure reactor core protection and accident mitigation strategies.

D. Documents all plant deficiencies discovered during the shift and reports them to the CRS.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AD-OP-ALL-1000 the RO uses diverse and redundant indications for verification of plant or equipment status.

A. Correct.

B. Incorrect. Plausible since the RO position is responsible for using diverse and redundant indications for verification of plant or equipment status, however this is incorrect because he/she is allowed to operate ESF equipment that does not properly respond to changes in plant conditions.

C. Incorrect. Plausible since the RO position is responsible for operating the Reactor safely and efficiently, however this is incorrect because the CRS is the position repsonsible for proritizing the activities required to support safe operation of the Reactor.

D. Incorrect. Plausible since the RO position is responsible to initiate prompt corrective action upon the reciept of abnormal conditions, however this is incorrect because the AO position is required to document all deficencies discovered during the shift and report them to shift supervision.

2.1 Conduct of Operations

G2.1.2; Knowledge of operator responsibilities during all modes of plant operation.

(CFR: 41.10 / 45.13)

Importance Rating:	4.1 4.4
Technical Reference:	AD-OP-ALL-1000, Section 4.9, Step 3, Pg 13, Rev. 8
References to be provided:	None
Learning Objective:	PP-LP-3.0, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 067/BANK/FUNDAMENTAL//TS 3.9/NONE/2012 NRC RO 66/G2.1.36/

Which ONE of the following would require suspension of Core Alterations?

A. The Containment equipment hatch can be closed with 8 bolts.

B✓ Source Range audible indication inside Containment becomes unavailable.

C. Reactor Cavity water level is 23 feet 3 inches above the vessel flange with only one RHR loop operable and in operation.

D. Train 'A' Fuel Handling Building Emergency Exhaust unit is OPERABLE and in operation; Train 'B' has just been declared INOPERABLE.

Plausibility and Answer Analysis

Reason answer is correct: TS 3.9.2 states that 1 SR channel must have audible indication in the MCR and CNMT during Mode 6. If audible indication is not available in CNMT, TS 3.9.2 states: immediately suspend all operations involving core alterations or positive reactivity additions.

Plausible because these are all Tech Spec immediate actions for a certain event. They cause LCO actions to be completed immediately but not all apply to core alterations or are already in an alignment that is allowed.

A. Incorrect. TS 3.9.4.this meets the requirement and will not require suspension because the equipment door is cabable of being closed and only needs to be held in place by a minimum of four bolts.

B. Correct.

C. Incorrect. TS 3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation in MODE 6 with irradiated fuel in the vessel when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet

D. Incorrect. Alterations involving movement of fuel within the storage pool or crane operation with loads over the storage pool are not allowed but Core Alterations are not affected. (TS 3.9.12)

2.1 Conduct of Operations

G2.1.36; Knowledge of procedures and limitations involved in core alterations.

(CFR: 41.10 / 43.6 / 45.7)

Importance Rating: 3.0 4.1

Technical Reference: Technical Specification 3.9.2

References to be provided: None

Learning Objective: LP-TS-2.0, Obj. 2

Question Origin: Bank

Comments: None

Tier/Group: T3

2018 NRC RO 068/NEW/C/A//CURVES/NONE//G2.1.45/

The operating crew is raising power to 100% in accordance with GP-005, Power Operation (Mode 2 to Mode 1).

PR NIs were indicating 15% Reactor power when the Generator was synched to the grid.

Current indications are:
- Average PR NIs are indicating 20% Reactor power
- T_{avg} is 563°F

In accordance with GP-005, which ONE of the following is a diverse indication that could be used by the crew as an indication of true Reactor power?

(Reference Provided)

A. Pressurizer Level is 31.5%

B.✓ Average Percent RCS Loop ΔT is 22%

C. Average Axial Flux (% Delta-I) is -1.0

D. First Stage Pressure is 110 psig

Plausibility and Answer Analysis

Reason answer is correct: Until a calorimetric is performed at 30% power, true reactor power shall be assumed equal to the highest of the following indicators: average Power Range NI value, average percent RCS Loop ΔT, or Main Turbine load. Average Percent RCS Loop ΔT is 22% which is slightly higher is an expected plant parameter and until the 30% power calorimetric is performed should be used as a diverse indication of true Reactor power.

A Incorrect. *Plausible since Pressurizer level would read approximately 31.5% using Curve H-X-14 Rev. 2 as a reference but it is NOT part of the diverse indications for true Reactor power in accordance with GP-005.*

B *Correct.*

C. *Incorrect.* *Plausible since the Average Axial Flux for 20% power using curve F-20-2 is approximately -1.0 but but it is NOT part of the diverse indications for true Reactor power in accordance with GP-005.*

D. *Incorrect.* *Plausible since the First Stage Pressure could be used as a indication of true reactor power below 30% prior to the calorimetric being performed but this reading is NOT the highest of the indicators for true Reactor power.*

2.1 Conduct of Operations

G2.1.45; Ability to identify and interpret diverse indications to validate the response of another indication.

(CFR: 41.7 / 43.5 / 45.4)

Importance Rating:	4.3 4.3
Technical Reference:	GP-005, Precaution and Limitation 1, Pg 4, Rev. 102
References to be provided:	Curves F-20-2, G-4, H-12, and H-X-14
Learning Objective:	LP-IE-17.3, Obj. 1
Question Origin:	NEW
Comments:	None
Tier/Group:	T3

2018 NRC RO 069/BANK/FUNDAMENTAL//OMP-003/NONE/2012 NRC RO 68/G2.2.18/

Given the following plant conditions:
- Mode 6, Refueling is in progress

Which ONE of the following lists the Key Safety Functions per OMP-003, Outage Shutdown Risk Management?

A. Subcriticality, Core Cooling, Integrity, Inventory, Containment

B. Decay Heat Removal, Reactivity Control, Integrity, Containment, Residual Heat Removal

C. Residual Heat Removal, Inventory, Cavity Level, Electrical Power, Auxiliary Feed Water

D✓ Containment, Inventory, Decay Heat Removal, Electrical Power, Reactivity Control

Plausibility and Answer Analysis

Reason answer is correct: Per OMP-003 the key safety functions when shutdown are Containment, Inventory, Decay Heat Removal, Electrical Power and Reactivity Control (CIDER)

A. Incorrect. Plausible because the catagories listed are those for the CSFST 's

B. Incorrect. Plausible because the catagories listed are the key safety functions while determining on-line risk

C. Incorrect. Plausible because the categories listed are subcategories of the key safety functions while shutdown.

D. Correct. CIDER

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:	OMP-003, Step 5.2.1.12, Pg 15, Rev. 41
References to be provided:	None
Learning Objective:	None
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 070/NEW/C/A//TECH SPEC 3.1.2.5/NONE//G2.2.35/

Given the following plant conditions:

- Reactor Vessel Head Closure bolts are fully tensioned
- RCS T_{avg} is 135°F
- RWST level is 20%

Which ONE of the following identifies (1) the current plant OPERATIONAL MODE and (2) the status of the RWST Limiting Condtion For Operation in accordance with Technical Specification 3.1.2.5, Reactivity Control Systems, Borated Water Source - Shutdown?

A. (1) Mode 5

(2) NOT met

B.✓ (1) Mode 5

(2) met

C. (1) Mode 6

(2) NOT met

D. (1) Mode 6

(2) met

Plausibility and Answer Analysis

Reason answer is correct: Mode 5 is defined in Tech Spec Table 1.2 as a plant condition where the average coolant temperature is = 200°F T_{avg}. With an RCS T_{avg} of 135°F Tech Spec Mode 5 applies. With the unit in Mode 5 per Tech Spec 3.1.2.5 in order for the RWST to be operable it must have a minimum volume of 106,000 gallons which is 12%, with boron concentration between 2400 and 2600 ppm and a minimum solution temperature of 40°F.

A. Incorrect. The first part is correct. The second part is plausible since RWST level is below 23% which is the minimum level for the BAT and below the 23.4% value for the ECCS automatic swap to the Containment Sump. However this is incorrect as the minimum RWST level in mode 5 is 12%.

B. Correct.

C. Incorrect. The first part is plausible since Refueling is Mode 6, which is = 140°F and the temperature of 135°F is < 140°F. However this is incorrect as the reactor vessel head bolts must be less than fully tensioned or the head removed in order to be in Mode 6. The second part is plausible since RWST level is below 23% which is the minimum level for the BAT and below the 23.4% value for the ECCS automatic swap to the Containment Sump. However this is incorrect as the minimum RWST level in mode 5 is 12%.

D. Incorrect. The first part is plausible since Refueling is Mode 6, which is = 140°F and the temperature of 135°F is < 140°F. However this is incorrect as the reactor vessel head bolts must be less than fully tensioned or the head removed in order to be in Mode 6. The second part is correct.

2.2 Equipment Control

G2.2.35; Ability to determine Technical Specification Mode of Operation.

(CFR: 41.7 / 41.10 / 43.2 / 45.13)

Importance Rating:	3.6 4.5
Technical Reference:	Technical Specification Table 1.2, Operational Modes, Technical Specification 3.1.2.5
References to be provided:	None
Learning Objective:	TS-LP-2.0/3.0/5.0/8.0, Obj. 3.a and 4.a
Question Origin:	New
Comments:	None
Tier/Group:	T3

2018 NRC RO 071/BANK/FUNDAMENTAL//FSAR 15.0/NONE//G2.2.38/

Which ONE of the following identifies WHY it is essential to operate the plant in accordance with the limiting conditions required by Technical Specifications?

A. To ensure minimum operator actions are required to mitigate the consequences of a Reactor accident.

B✓ To ensure the assumptions made in the accident analysis remain valid in the event of a Reactor accident.

C. To ensure no radiation is released to the environment as a result of a Reactor accident.

D. To ensure that DNBR will remain greater than the design limit for all Reactor accidents.

Plausibility and Answer Analysis

Reason answer is correct: AREVA Analyses - The accidents presented in Chapter 15 are analyzed at limiting conditions consistent with the Technical Specifications.

A. Incorrect. Plausible since accident analysis is combined with the control system setpoint study, which is designed to automatically maintain prescribed conditions in the plant, to show the plant can meet both safety and operability requirements, however this is incorrect because this is a method used to comply with the limiting conditions of operations and not the reason why the plant is operated within these limited conditions.

B. Correct.

C. Incorrect. Plausible since the basic principle of plant design is to limit the radiological risk to the public, however this is incorrect because the design requirement for radiological risk is the extreme situation having the greatest risk to the public that is least likely to occur and not the reason why the plant is operated within these limited conditions..

D. Incorrect. Plausible since DNBR is one of the criteria designed to be limited by the Reactor Protection and Emergency Core Cooling systems, however this is incorrect because this is a result of compliance with the limiting conditions of operations and not the reason why the plant is operated within these limited conditions.

2.2 Equipment Control

G2.2.38; 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: FSAR Chapter 15.0

References to be provided: None

Learning Objective: TAA-LP-2.10, Obj. 6

Question Origin: Bank

Comments: None

Tier/Group: T3

2018 NRC RO 072/BANK/FUNDAMENTAL//10CFR20/NONE//G2.3.4/

Which ONE of the following is the NRC annual dose limit for a declared pregnant female for the duration of the pregnancy?

A. 200 mrem

B✓ 500 mrem

C. 2 rem

D. 5 rem

Plausibility and Answer Analysis

Reason answer is correct: In accordance with PD-RP-ALL-0001, the NRC annual dose limit for a declared pregnant female is 500 mrem for the pregnancy duration.

A. Incorrect. Plausible since this 10% of an adult limit for Duke Energy imposed on minors (<18 years of age).

B. Correct.

C. Incorrect. Plausible since this is theTotal Effective Doese Equivalent (TEDE) for Duke Energy.

D. Incorrect. Plausible since this is theTotal Effective Doese Equivalent (TEDE) for the NRC Annual Dose.

2.3 Radiation Control

G2.3.4; Knowledge of radiation exposure limits under normal or emergency conditions.

(CFR: 41.12 / 43.4 / 45.10)

Importance Rating:	3.2 3.7
Technical Reference:	PD-RP-ALL-0001, Section 5.2.2, Pg 18, Rev. 7
References to be provided:	None
Learning Objective:	PP-LP-3.7, Obj. 1.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 073/BANK/FUNDAMENTAL//AOP-038/NONE//G2.4.11/

Which ONE of the following ramp rates is the LOWEST that would require using AOP-038, Rapid Down Power, to reduce power?

A. 4 MW/min

B✓ 6 MW/min

C. 10 MW/min

D. 15 MW/min

Plausibility and Answer Analysis

Reason answer is correct: Plant Conditions that require a rapid reduction in power level to preclude a plant trip (or in lieu of a plant trip) may warrant entry into this procedure. Any condition requiring greater than 5 MW/min load reductions.

A. Incorrect. Plausible since this is the power level that GP-006 uses as the maximum power level and there are many AOP's that have the operator determine which shutdown procedure will be implented either GP-006 or AOP-038.

B. Correct.

C. Incorrect. Plausible since AOP-038 identifies a target load reduction rate of 10 Mw/Min when the ASI system is supplying RCP seal injection at the beginning of life, however this is incorrect since entry in to AOP-038 is allowed for any condition requiring greater than 5 MW/min load reduction.

D. Incorrect. Plausible since a notification to chemistry is required if power is changed greater than 15% in any 1 hour period, however this is incorrect since entry in to AOP-038 is allowed for any condition requiring greater than 5 MW/min load reduction.

2.4 Emergency Procedures / Plan

G2.4.11; Knowledge of abnormal condition procedures.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	4.0 4.2
Technical Reference:	AOP-038, Section 2.0, Pg 3, Rev. 44
References to be provided:	None
Learning Objective:	AOP-LP-3.38, Obj. 1
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 074/BANK/FUNDAMENTAL//FPP-002/NONE//G2.4.25/

In accordance with FPP-002, Fire Emergency, when shall the Control Room Operator sound the plant fire alarm?

A. Upon the receipt of any fire alarm in the Control Room

B. ✓ After a second fire alarm is received in an adjacent zone

C. Upon the receipt of an Incipient Fire Detection system alert

D. After receipt of a single fire alarm for areas inside Containment

Plausibility and Answer Analysis

Reason answer is correct: Per FPP-002 When a second alarm is received in the Control room, then sound the plant fire alarm.

A. Incorrect. Plausible since a single fire alarm inside containment will require action by the control room to monitor for diverse indications to confirm a fire exists, however this is incorrect the plant fire alarm is not sounded until confirmation a fire exists.

B. Correct.

C. Incorrect. Plausible since if an Incipient Fire Detection system alarm is received before an Operator or on-shift I&C tech has responded to an Incipient Fire Detection system alert, requires the plant fire alarm to be sounded, however this is incorrect the plant fire alarm is not sounded until confirmation a fire exists if the Incipient Fire Detection system is only in alert status.

D. Incorrect. Plausible since a single fire alarm inside containment has specific actions for the control room to monitor for diverse indications to determine a fire exists, however this is incorrect the plant fire alarm is not sounded until confirmation a fire exists.

2.4 Emergency Procedures / Plan

G2.4.25; Knowledge of fire protection procedures.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.3 3.7
Technical Reference:	FPP-002, Section 6.1.6, Pg 12, Rev. 42
References to be provided:	None
Learning Objective:	PP-LP-3.15, Obj. 1
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 075/BANK/FUNDAMENTAL//ALB-011, 018/NONE//G2.4.31/

Given the following plant conditions:
- The unit is operating at 100% power
- A plant transient results in the crew manually tripping the Turbine

What is the expected color of the following flashing MCB Annunciators?

	ALB-018-2-5 Turbine Trip Manual	ALB-011-3-3 Reactor Trip Turbine Trip P7
A✓	Red	Red
B.	Red	White
C.	White	Red
D.	White	White

Plausibility and Answer Analysis

Reason answer is correct: Three ALBs, two for reactor trip (ALB-011 and ALB-012), one for turbine trip (ALB-018), are provided to function as First Out ALBs. The first alarm received on these ALBs will be red to identify the cause of the trip (i.e., one red for the reactor trip and another red for the turbine trip). Subsequent alarms will be white.

A. Correct.

B. Incorrect. Plausible since the Turbine Trip was generated manually the candidate may determine that only the first annunciator generated for the Turbine Trip System will be Red and subsequent alarms will be white; this however is incorrect because the MCB has a First Out annuciator system for the Reactor system that is separate from the Turbine System and will generate a Red First Out alarm for both systems.

C. Incorrect. Plausible since the Turbine Trip was generated manually the candidate may determine that only the first annunciator generated for the Reactor Trip system will be Red and remaining alarms will be white; this however is incorrect because the MCB has a First Out annuciator system for the Reactor system that is separate from the Turbine System and will generate a Red First Out alarm for both systems.

D. Incorrect. Plausible since the Turbine Trip was generated manually the candidate may determine that only automatic trip signals for both the Reactor and Turbine Trip systems will be Red and subsequent alarms will be white; this however is incorrect because the MCB has a First Out annuciator system for the Reactor system that is separate from the Turbine System and will generate a Red First Out alarm for both systems based on automatic or manual trip signals.

2.4 Emergency Procedures / Plan

G2.4.31; Knowledge of annunciator alarms, indications, or response procedures.

(CFR: 41.10 / 45.3)

Importance Rating:	4.2 4.1
Technical Reference:	APP-ALB-011, Window 3-3, Pg 8, Rev. 8 APP-ALB-018, Window 2-5, Pg 10, Rev. 21
References to be provided:	None
Learning Objective:	MCB Lesson Plan, Obj. 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

QUESTIONS REPORT
for 2018 NRC RO SRO WRITTEN EXAM REV 75 Submittal

2018 NRC RO 001/BANK/C/A//ES-0.1/NONE/2009A NRC RO 01/007 EK2.03/

Given the following plant conditions:
- A Reactor trip has just occurred from full power due to a loss of Offsite Power
- The crew has entered EOP-ES-0.1, Reactor Trip Response

The BOP is directed to control temperature in accordance with EOP-ES-0.1, Table 1

The following plant conditions currently exist:
- All SG levels are 21% and lowering
- Total AFW flow to the Steam Generators is 208 KPPH
- Loop Low T_{avg} Bistable lights are lit on TSLB 3
- Loop Low-Low T_{avg} Bistable lights are lit on TSLB 3
- Group 1 Condenser Steam Dumps red AND green lights are lit on SLB 1

Which ONE of the following describes the action required in accordance with EOP-ES-0.1, Table 1?

A✓ Close all Main Steam Isolation Valves

B. Place Steam Dumps in the Steam Pressure Mode

C. Raise AFW flow to the SGs to raise SG water levels

D. Reduce AFW flow to the SGs to stop the RCS Cooldown

Plausibility and Answer Analysis

Reason answer is correct: With Loop Low-Low T_{avg} Bistable lights illuminated on TSLB 3 (P-12), the Steam Dumps should be closed and are not which results in an uncontrolled cooldown of the RCS. EOP-ES-0.1 states if the cooldown continues, then shut the MSIV and bypass valves.

A Correct.

B Incorrect. Plausible since EOP-ES-0.1 has guidance to place the Steam Dumps in the Steam Pressure Mode in several locations. The candidate may have the misconception that this will assist with present plant conditions, but this is incorrect because P-12 affects Steam Dumps in both the T_{avg} and Steam Pressure Modes.

C Incorrect. Plausible since Steam Generator Water Levels are 21% and lowering so the candidate may believe raising AFW flow to be correct because EOP-ES-0.1 has guidance to maintain SGWL 25 to 50%. However, with RCS Temperature less than 557°F, AFW flow should be greater than 200 KPPH but minimized to prevent further cooldown.

D Incorrect. Plausible since RCS Temperature is less than 557°F, the candidate may have the misconception that reducing AFW flow will slow the cooldown rate because EOP-ES-0.1 has guidance to control RCS temperature at 555°F to 559°F but greater than 200 KPPH is required for heatsink.

000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1

007EK2.03; Knowledge of the interrelations between a reactor trip and the following:
Reactor trip status panel

(CFR 41.7 / 45.7)

Importance Rating:	3.5 3.6
Technical Reference:	EOP-ES-0.1, Step 4, Pg 6, Rev. 3
References to be provided:	None
Learning Objective:	EOP-LP-3.22, Obj. 3.d
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 002/BANK/C/A//E-1, ES-1.2,FR-H.1/NONE/2013 NRC RO 03/009 EK2.03/

Given the following plant conditions:

- All RCPs are running
- RCS pressure is 920 psig and slowly lowering
- SI flow is 100 GPM
- Containment pressure is 3.2 psig and slowly rising
- SG pressures are 1120 psig and stable

Which ONE of the following completes the statements below, in accordance with EOP-E-1, Loss of Reactor Or Secondary Coolant?

RCP trip criteria ___(1)___ met.

___(2)___ is the MINIMUM pressure above which the crew will transition to EOP-ES-1.2, Post LOCA Cooldown and Depressurization, where the SGs will be required for RCS cooldown.

A. (1) is

(2) 230 psig

B. (1) is

(2) 400 psig

C.✓ (1) is NOT

(2) 230 psig

D. (1) is NOT

(2) 400 psig

Plausibility and Answer Analysis

Reason answer is correct: *A Small Break LOCA is in progress. EOP-E-0, Reactor Trip or Safety Injection foldout for RCP trip criteria is NOT met with RCS pressure < 1400 psig and SI flow < 200 gpm. Therefore the RCPs should NOT be tripped. A procedure transition from EOP-E-0 to EOP-E-1 to EOP-ES-1.2 will take place for this event. In order to transistion to EOP-ES-1.2, Post LOCA Cooldown, the crew will evaluate if the break is large enough to allow the low pressure high volume RHR system to provide core cooling or if the Steam generators are required for cooling. If RCS pressure is less than 230 psig and RHR flow is greater than 1000 gpm the crew will remain in EOP-E-1 and initiate a RCS cooldown to CSD with SG PORV's. This method would be used instead of using the Steam Dumps since the MSIV's would be shut due to high Containment pressure. In this situtuation the RCS pressure is still greater than the pressure required to place RHR in service therefore the SG's would be required for subsequent heat removal until the RHR system is placed in service in the shutdown cooling mode.*

A. Incorrect. The first part is plausible because the RCS pressure is below the value (<1400 psig) in which the RCPs would be NOT be required if adequate SI flow (>200 gpm) was injecting into the core. The second part is correct.

B. Incorrect. The first part is plausible because the RCS pressure is below the value (<1400 psig) in which the RCPs would be NOT be required if adequate SI flow (>200 gpm) was injecting into the core, however with only 100 gpm flow from the SI system the RCPs are required to remain in operation. The second part is plausible because 400 psig is the lift setpoint for PRZ PORV's when in LTOP configuration.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible because 400 psig is the lift setpoint for PRZ PORV's when in LTOP configuration.

000009 009 Small Break LOCA / 3

009EK2.03; Knowledge of the interrelations between the small break LOCA and the following: S/Gs

(CFR 41.7 / 45.7)

Importance Rating: 3.0 3.3

Technical Reference: EOP-E-1, Foldout, Pg 3, Rev. 4
EOP-E-1, step 13, RNO

References to be provided: None

Learning Objective: EOP-LP-3-11, Obj. 4.e

Question Origin: Bank

Comments: None

Tier/Group: T1/G1

2018 NRC RO 003/NEW/C/A//AOP-018/NONE//015 AG2.1.23/

Given the following plant conditions:

- A Reactor startup is progress with the unit in Mode 2
- Annunciator ALB-008, Window 3-3, RCP-A Seal #1 Leakoff High Low Flow, has alarmed

Subsequently:

- The BOP reports that RCP 'A' #1 seal leakoff is approximately 8.5 gpm and rising

Which ONE of the following sequence of actions is the OATC required to perform for these conditions in accordance with AOP-018, Reactor Coolant Pump Abnormal Conditions?

A. Trip the Reactor, secure RCP 'A' and then Go To EOP-E-0, perform the immediate actions

B.✓ Trip the Reactor, Go To EOP-E-0, perform the immediate actions and then secure RCP 'A' as time permits

C. Secure RCP 'A', shut 1CS-355, RCP 'A' #1 Seal Water Return and then shut 1RC-107 PRZ Spray Loop A valve

D. Secure RCP 'A', wait 3 to 5 minutes then shut 1RC-107 PRZ Spray Loop A valve and then shut 1CS-355, RCP 'A' #1 Seal Water Return

Plausibility and Answer Analysis

Reason answer is correct: RCP 'A' seal has failed as evidenced by the magnitude of seal leakoff (≥ 8 gpm) and continuing degrading conditions. The OATC will be required to trip the Reactor, GO TO EOP-E-0 (to perform the immediate actions) and then perform steps 3-9 of AOP-018 section 3.3 when time permits. AOP-018 Section 3.3 step 3 stops the affected RCP.

A. Incorrect. This is plausible because these are the correct actions but not in the correct order.

B. Correct.

C. Incorrect. Plausible since this would be correct if the Reactor trip breakers were open and if the operator waited for 3 - 5 minutes after securing the RCP. Since the Shutdown banks are withdrawn the Reactor trip breakers are closed. Therefore the Reactor must be tripped prior to securing the affected RCP.

D. Incorrect. Plausible since this would be the correct actions if the Reactor trip breakers were open, but in the wrong sequence. Since the Shutdown banks are withdrawn the Reactor trip breakers are closed. Therefore the Reactor must be tripped prior to securing the affected RCP.

000015 Reactor Coolant Pump (RCP) Malfunctions / 4

015AG2.1.23; 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:	AOP-018, Section 3.3, Pg 10, Rev. 49
References to be provided:	None
Learning Objective:	AOP-LP-3.18, Obj. 3
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 004/BANK/C/A//AOP-020/NONE/2012 NRC RO 04/025 AA2.07/

Given the following plant conditions:

- The unit is in Mode 5
- RHR 'A' and 'B' trains are in service for Shutdown Cooling

Subsequently:

- The OATC reports FI-605A1, RHR Hx 'A' Header Flow AND PDI-5450A, RHR Pump 'A' Diff Pressures are oscillating

Which ONE of the following completes the statement below for actions required to be taken in accordance with AOP-020, Loss of RCS Inventory or Residual Heat Removal while Shutdown?

Stop __(1)__ AND shut __(2)__ .

A. (1) BOTH RHR Pumps

(2) 1RH-2, RCS Loop 'A' to RHR Pump 'A' ONLY

[✓ B.] (1) BOTH RHR Pumps

(2) 1RH-2, RCS Loop 'A' to RHR Pump 'A' AND 1RH-40, RCS Loop 'C' to RHR Pump 'B'

C. (1) RHR Pump 'A' ONLY

(2) 1RH-2, RCS Loop 'A' to RHR Pump 'A' ONLY

D. (1) RHR Pump 'A' ONLY

(2) 1RH-2, RCS Loop 'A' to RHR Pump 'A' AND 1RH-40, RCS Loop 'C' to RHR Pump 'B

Plausibility and Answer Analysis

Reason answer is correct: AOP-020 requires all running RHR pumps to have normal operating indications and RPV level above 82" below the flange, if not both RHR pumps are stopped and isolated from the RCS by at least one isolation valve

A. Incorrect. The first part is correct. The second part is plausible since stopping both RHR pumps prevents damage to the non cavitating pump while isolation of the cavitating pump maintains the loop suction filled and vented.

B. Correct.

C. Incorrect. The first part is plausible since only the RHR pump 'A' is demonstrating indications of pump cavitation, however this is incorrect because AOP-020 requires both RHR pumps to be secured. The second part is plausible since isolation of the affected RHR pump is required to be shut in order to maintain the loop suction filled and vented, however this is incorrect because AOP-020 requires both RHR loops to be isolated.

D. Incorrect. The first part is plausible since only the RHR pump 'A' is demonstrating indications of pump cavitation, however this is incorrect because AOP-020 requires both RHR pumps to be secured. The second part is plausible since an RHR pump is capable of operating on its recirculation flow path with the discharge valve shut, candidate may apply this knowledge to the loop suction valve.

000025 Loss of Residual Heat Removal System / 4

025AA2.07; Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Pump cavitation

(CFR: 43.5 / 45.13)

Importance Rating:	3.4 3.7
Technical Reference:	AOP-020, Section 3.1, Step 1 RNO, Pg 5, Rev 38
References to be provided:	None
Learning Objective:	AOP-LP-3-20, Obj. 3.b and 3.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 005/NEW/C/A//AOP-014/NONE//026 AA1.02/

Given the following plant conditions:

- The unit is operating at 100% power

At 1000 a loss of Component Cooling Water occurred

- The crew has entered AOP-014, Loss of Component Cooling Water, due to a loss of all CCW flow to both trains
- RCP Seal Injection flow to each RCP is approximately 9 gpm
- RCP Temperatures are as listed below:
  - 'A' RCP Motor bearing is 142° and continuing to rise at 4°F/minute
  - 'B' RCP Radial bearing is 176° and continuing to rise at 7°F/minute
  - 'C' RCP Stator Winding is 253°F and continuing to rise at 8°F/minute

Of the four times listed below which ONE of the following describes the MAXIMUM time that the unit would be allowed to operate BEFORE a Reactor Trip would be required in accordance with AOP-014?

A. 1002

B✓ 1006

C. 1008

D. 1010

Plausibility and Answer Analysis

Reason answer is correct: The provided heatup rates of the RCPs would require that the Reactor is tripped at 2006 based on the 'C' RCP exceeding the Stator Winding temperature trip limit of 300°F

1002 - 'B' RCP Radial bearing temperture exceeds the Motor bearing temperture trip limit of 190°F

1006 - 'C' RCP Stator Winding temperature rising at 9°F/min starting at 253°F would reach the trip setpoint of 300°F [253° + (8°F x 6 min) = 301°F]

1008 - 'B' RCP Radial Bearing temperature rising at 7°F/min starting at 176°F would reach the trip setpoint of 230°F [176 + (7° x 8 min) = 232°F]

1010 - RCPs have operated for 10 minutes without CCW flow to either motor oil cooler (1000 + 10 min = 1010)

1012 - 'A' RCP Motor Bearing temperature rising at 4°F/min starting at 142°F would reach the trip setpoint of 190°F [142° + (4° x 12 min) = 190°F]

A. Incorrect. Plausible if the student has a misconception that the trip limit for Radial Bearing temperature is 190°F and not 230°F. At 1002 the 'B' RCP Radial bearing temperature exceeded 190°F. Starting temperture of 176°F + (2 min x 7°F) = 190°F

B. Correct.

C. Incorrect. Plausible since at 1008 the 'B' RCP Radial Bearing trip limit of 230°F was exceeded.

D. Incorrect. Plausible since a loss of CCW to an RCP or RCP Motor has operated for 10 minutes without CCW flow to either motor oil cooler

000026 Loss of Component Cooling Water /8

026AA1.02 Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: Loads on the CCWS in the control room

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.2 3.3
Technical Reference:	AOP-014, Attachment 1, Pg 40-41, Rev. 37
References to be provided:	None
Learning Objective:	AOP-LP-3.14 Obj 2.e
Question Origin:	New
Comments:	Discuss AOP-014, Loss Of CCW, actions are based on the change in temperature for the parameters associated with components cooled by CCW and not the CCW Temperature indications. Phonecon 10/23/2017: HNP discussed being unable to create a T1/G1 question based on plant abnormal procedures for the K/A topic of Loss Of Component Cooling Water associated with CCW temperature indications, so selected a new K/A, keeping 026 and determined this item was better tied to a different randomly selected K/A: New K/A 026AA1.02: Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: Loads on the CCWS in the control room.
Tier/Group:	T1/G1

2018 NRC RO 006/BANK/FUNDAMENTAL//TS 2.1.2/NONE//027 AG2.2.22/

Given the following plant conditions:
- The RCS is operating in solid plant conditions

Subsequently:
- A controller failure has caused 1CS-38, Letdown Pressure Control valve, to shut
- The crew has just entered AOP-019, Malfunction of RCS Pressure Control

Which ONE of the following completes the statement below?

In accordance with Technical Specification 2.1.2, Safety Limits - Reactor Coolant System Pressure, RCS pressure shall not exceed ___(1)___ psig. If this is violated, the RCS pressure shall be reduced below the safety limit within a MAXIMUM of ___(2)___ minutes.

A. (1) 2485

(2) 5

B. (1) 2485

(2) 15

C✓ (1) 2735

(2) 5

D. (1) 2735

(2) 15

Plausibility and Answer Analysis

Reason answer is correct: In accordance with Technical Specification 2.1.2 the RCS pressure shall not exceed 2735 in Modes 1, 2, 3, 4, and 5 except for hydrostatic testing. The action for exceeding this limit in Modes 3, 4 and 5 is to restore below the limit within 5 minutes and comply with the requirements of specification 6.7.1.

A. Incorrect. The first part is plausible since this is the pressure that the PRZ safeties lift. The second part is correct.

B. Incorrect. The first part is plausible since this is the pressure that the PRZ safeties lift. The second part is plausible since the 15 minute action is correct for restoring T_{avg} *above the minimum temperature for criticality and the candidate may misapply this time limitation to the 5 minute requirement of Technical Specification 2.1.2.*

C. Correct.

D. Incorrect. The first part is correct.The second part is plausible since the 15 minute action is correct for restoring T_{avg} *above the minimum temperature for criticality and the candidate may misapply this time limitation to the 5 minute requirement of Technical Specification 2.1.2.*

000027 Pressurizer Pressure Control System Malfunction / 3

027AG2.2.22; Knowledge of limiting conditions for operations and safety limits.

(CFR: 41.5 / 43.2 / 45.2)

Importance Rating:	4.0 4.7
Technical Reference:	Technical Specification, 2.1.2
References to be provided:	None
Learning Objective:	TS-LP-2.0, Obj. 2
Question Origin:	Bank
Comments:	Discuss whether 1CS-38 can be considered part of the PRZ Pressure control system during solid plant operations with Dan Bacon for question development. Phonecon 10/23: Dan agrees that using 1CS-38 as a component in the PRZ Control system during solid plant operations is acceptable to meet this K/A
Tier/Group:	T1/G1

2018 NRC RO 007/NEW/C/A//AOP-016/NONE//038 EA1.10/

Given the following plant conditions:
- The unit is operating at 100% power
- Charging is in MANUAL at 85 gpm

Subsequently:
- At 0900 Annunciator ALB-010-4-5, Rad Monitor System Trouble, has alarmed
- The crew entered AOP-016, RCS Leakage, and have identified a SG tube leak is in progress

The following plant parameter changes are occurring as follows:

(Rad readings are mR/Hr)	0900	0901	0902	0903	0904
RM-01MS-3591 SB, MSL "A"	2.91E-1	7.67E-1	1.95E-0	3.21E-0	5.98E-0
RM-01MS-3592 SB, MSL "B"	2.22E-1	2.29E-1	3.38E-1	4.45E-1	4.82E-1
RM-01MS-3593 SB, MSL "C"	3.82E-1	3.82E-1	3.82E-1	3.82E-1	3.83E-1
Pressurizer Level	60.0%	59.2%	57.9%	56.2%	53.6%

Which ONE of the following completes the statement below concerning these conditions?

The trends indicate that the SG Tube Rupture is occuring on the __(1)___ .
The EARLIEST time that a manual Reactor trip is required in accordance with AOP-016 is __(2)___ .

A. (1) "A" SG Only

(2) 0902

B.✓ (1) "A" SG Only

(2) 0904

C. (1) "A" and "B" SG

(2) 0902

D. (1) "A" and "B" SG

(2) 0904

Plausibility and Answer Analysis

Reason answer is correct: Steam Line Radiation monitor indications on "A" SG have gone up by a factor of 10 over the last 4 minutes indicating that the "A" SG has the SG Tube Rupture. Pressurizer level has lowered > 2% in a one minute interval from 0903 to 0904. In accordance with AOP-016-BD, a drop of 2% or greater per minute in the Pressuizer indicates that makeup capability has been exceeded (Pressurizer has ~55 gal/% level at 653°F).

A. Incorrect. The first part is correct. The second part is plausible since Pressurizer level has dropped 2% since the initation of the event (from 0900 to 0902). But the level drop of 2% or greater must occur in a 1 minute interval. Makeup capability has not yet been exceeded.

B. Correct.

C. Incorrect. Plausible since both "A" SG and "B" SG radiation levels have risen since the initiation of the leak. Based on the magnitude of the radiation level change on the "B" SG the radiation level rise is the result of the "shine" phenomena coming from the "A" SG steam line radiation and is expected. The second part is plausible since Pressurizer level has dropped 2% since the initation of the event (from 0900 to 0902). But the level drop of 2% or greater must occur in a 1 minute interval. Makeup capability has not yet been exceeded.

D. Incorrect. Plausible since both "A" SG and "B" SG radiation levels have risen since the initiation of the leak. Based on the magnitude of the radiation level change on the "B" SG the radiation level rise is the result of the "shine" phenomena coming from the "A" SG steam line radiation and is expected. The second part is plausible since Pressurizer level has dropped 2% since the initation of the event (from 0900 to 0902). The second part is correct.

000038 Steam Generator Tube Rupture / 3

038EA1.10; Ability to operate and monitor the following as they apply to a SGTR: Control room radiation monitoring indicators and alarms

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.7 3.7
Technical Reference:	AOP-016, Section 3.0, Step 4 RNO, Pg 4, Rev 56 AOP-016, Attachment 1, Step 4, Pg 14, Rev 56 AOP-016-BD, Section 3.0, Step 4, Pg 8, Rev 30
References to be provided:	None
Learning Objective:	EOP-LP-3.02 Objective 5
Question Origin:	New
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 008/BANK/C/A//E-0/NONE//054 AK1.01/

Given the following plant conditions:
- The Unit is operating at 75% power

Subsequently:
- Containment pressure is 2.5 psig, and rising
- RCS T_{AVG} is 582°F, and rising
- RCS Pressure is 2195 psig, and lowering
- SG 'A' Steam Flow is 4.0 MPPH, stable
- SG 'A' feed flow is 4.7 MPPH, and rising
- SG 'A' Narrow Range level is 32%, and lowering
- SG 'A' Pressure is 1003 psig, and lowering

The CRS directs the OATC to manually trip the Reactor

Which ONE of the following completes the statements below concering this event?

RCS temperature will __(1)__ after the Reactor trip.

The crew will respond to a __(2)__ .

A. (1) lower below no-load T_{AVG} in an uncontrolled manner

(2) Loss Of Reactor Coolant

B✓ (1) lower below no-load T_{AVG} in an uncontrolled manner

(2) Faulted Steam Generator

C. (1) stabilize at no-load T_{AVG} shortly after the Main Feedwater Regulating Valves (FRVs) close

(2) Loss Of Reactor Coolant

D. (1) stabilize at no-load T_{AVG} shortly after the Main Feedwater Regulating Valves (FRVs) close

(2) Faulted Steam Generator

Plausibility and Answer Analysis

Reason answer is correct: A rising Contaiment pressure and SG feedwater flow with a lowering SG level is indicative of a Feedwater rupture. The Feedwater rupture will cause SG pressure to lower uncontrollably and cooldown the RCS. Any SG with pressure lowering in an uncontrolled SG manner indicates a Faulted Steam Generator.

A. Incorrect. The first part is correct. The second part is plausible since RCS pressure is lowering and Containment pressure is rising the candidate may determine that a LOCA is in progress. This is incorrect because SG Feedflow is rising concurrent with SG level lowering which is indication of a fault on the Feedwater line to the SG.

B. Correct.

C. Incorrect. The first part is plausible since the FRVs automatically shut after the Reactor trip breakers open and RCS temperature lowers below 564°F. The candidate may determine a Feedwater break inside Containment upstream of the Feedwater Isolation Valves would be isolated from the SG. This is incorrect because the SG inventory would continue to be released into Containment following the Reactor Trip. The second part is plausible since RCS pressure is lowering and Containment pressure is rising. The candidate may determine that a LOCA is in progress. This is incorrect because SG feedflow is rising concurrent with SG level lowering which is indication of a fault on the Feedwater line to the SG

D. Incorrect. The first part is plausible since the FRVs automatically shut after the Reactor trip breakers open and RCS temperature lowers below 564°F. The candidate may determine a feedwater break inside Containment upstream of the FW Isolation Valves would be isolated from the SG. This is incorrect because the SG inventory would continue to be released into Containment following the Reactor Trip. The sceond part is correct.

000054 Loss of Main Feedwater /4

054AK1.01; Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW):MFW line break depressurizes the S/G (similar to a steam line break)

(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	4.1 4.3
Technical Reference:	EOP-E-0, Step 25, Pg 30, Rev. 7
References to be provided:	None
Learning Objective:	EOP-LP-3.02, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 009/BANK/C/A//EOP-ECA-0.0/NONE/2012 NRC RO 09/055 EK1 .02/

Given the following plant conditions:
- A loss of Off-site power has occurred with the unit operating at 100%
- EDG 'A' fails to start
- EDG 'B' output breaker closes, but trips open
- EOP-ECA-0.0, Loss Of all AC Power, is being implemented
- ASI system is operating as expected

Which ONE of the following completes the statement below?

In accordance with EOP-ECA-0.0, the RCS cooldown during natural circulation is limited to a MAXIMUM rate of ___(1)___ AND the cooldown is required to ___(2)___ .

A. (1) 100°F / Hr

(2) minimize RCP seal leakage

B. (1) 100°F / Hr

(2) control Pressurizer level

C. (1) 50°F / Hr

(2) minimize RCP seal leakage

[✓ D.] (1) 50°F / Hr

(2) control Pressurizer level

Plausibility and Answer Analysis

Reason answer is correct: During a LOOSP the ASI system provides seal injection to the RCP seals. To prevent the RCS from going solid a cooldown must be initiated to offset the mass addition from the ASI system.
The maximum cooldown rate to control PRZ level is 50°F / hr.

A. Incorrect. The first part is plausible since a rate of 100°F / hr maximum cooldown rate under normal conditions, however this is incorrect because the cooldown rate during EOP-ECA-0.0 is limited to half the normal rate. The second part is plausible since this is the correct answer if the ASI system fails to operate, however this is incorrect because all other systems operate as designed.

.

B. Incorrect. The first part is plausible since a rate of 100°F / hr maximum cooldown rate under normal conditions, however this is incorrect because the cooldown rate during EOP-ECA-0.0 is limited to half the normal rate. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since this is the correct answer if the ASI system fails to operate, however this is incorrect because all other systems operate as designed.

D. Correct.

000055 Station Blackout / 6

055EK1 .02; Knowledge of the operational implications of the following concepts as they apply to the Station Blackout : Natural circulation cooling

(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	4.1	4.4
Technical Reference:	EOP-ECA-0.0, Step 33.a, Pg 56, Rev. 7	
References to be provided:	Steam Tables	
Learning Objective:	EOP-LP-3.7 Obj. 6	
Question Origin:	Bank	
Comments:	None	
Tier/Group:	T1/G1	

2018 NRC RO 010/BANK/C/A//AOP-020/NONE/2009A NRC RO 75/056 AG2.4.9/

Given the following plant conditions:
- The unit is in Mode 5
- The RCS is in solid plant operation
- Both Trains of RHR are aligned in the Shutdown Cooling Mode

Subsequently an earthquake occurs:
- Offsite power is lost and a large RCS leak has developed
- 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 is the action required by AOP-020 to mitigate the event?

A. Start the 'B' CSIP with flow through 1SI-3 and 1SI-4, BIT Outlet Valves

B. Start the 'B' CSIP with flow through 1SI-52, Alternate High Head SI to Cold Leg Valve

C. Align 'A' RHR Pump for Low Head SI with flow through 1SI-359, Low Head SI Trains to Hot Leg Valve

D.✓ Align 'A' RHR Pump for Low Head SI with flow through 1SI-340, Low Head SI Train A to Cold 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.

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 because 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 with two CSIPs, however this is incorrect because only one CSIP is Operable in this mode.

C. Incorrect. Plausible since alignment of one train of RHR for Low Head SI is directed in AOP-020, however this is incorrect because flow is through 1SI-340, Low Head SI Train A to Cold Leg Valve not 1SI-359, which is a possible alignment when implementing EOP-ES-1.4.

D. Correct.

000056 Loss of Offsite Power / 6

056AG2.4.9; Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.8 4.2
Technical Reference:	AOP-020, Section 3.6, pg 66, Rev 38
References to be provided:	None
Learning Objective:	AOP-LP-3.20, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 011/BANK/C/A//AOP-024-BD/NONE/2012 NRC RO 11/057 AK3.01/

Given the following plant conditions:
- The unit is operating at 100% power
- PRZ Level Control is selected to position 459/460

Subsequently:
- Instrument Bus S-II is lost
- The crew is implementing AOP-024, Loss of Uninterruptible Power Supply

Which ONE of the following completes the statements below?

Letdown flow will be isolated because __(1)__ went SHUT.

In accordance with AOP-024, the OATC will control Charging in manual with FK-122.1, Charging Flow, to __(2)__ with letdown isolated.

A. (1) 1CS-1, Letdown Isolation LCV-460

(2) prevent gas binding of the CSIP

B.✓ (1) 1CS-1, Letdown Isolation LCV-460

(2) minimize the PRZ level rise

C. (1) 1CS-2, Letdown Isolation LCV-459

(2) prevent gas binding of the CSIP

D. (1) 1CS-2, Letdown Isolation LCV-459

(2) minimize the PRZ level rise

Plausibility and Answer Analysis

Reason answer is correct: Instrument Bus SII supplies Pressurizer Level channel 460 and the affect of losing power is the channel will fail low, which causes its associated letdown isolation valve 1CS-1 to shut. With letdown isolated to reduce charging flow to the minimum amount the FK-122 must be placed in manual. This is required to minimize the rise in PRZ level while selecting a valid channel since seal injection will continue into the RCS with no letdown to lower the level.

A. Incorrect. The first part is correct. The second part is plausible since with letdown isolated the VCT level will lower as the CSIP continues to take suction from it which expands the VCT vapor space and increases the potential for gas intrusion at the pump impeller, however this is incorrect because the CSIP suction will automatically shift from the VCT to the RWST at 5% therefore placing charging flow to manaul and minimum is not required.

B. Correct.

C. Incorrect. Part 1 is plausible since 1CS-2 shutting will isolate letdown flowif it is determined that LT-459 has loss power.

Part 2 is plausible if the candidate believes the VCT Level transmitters are effected with the lost of power to S-II, since with letdown isolated the VCT level will lower as the CSIP continues to take suction from it which expands the VCT vapor space and increases the potential for gas intrusion at the pump impeller.

D. Incorrect. Part 1 is plausible since 1CS-2 shutting will isolate letdown flowif it is determined that LT-459 has loss power.

Part 2 is correct.

000057 Loss of Vital AC Instrument Bus / 6

057AK3.01; Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus

(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	4.1 4.4
Technical Reference:	AOP-024-BD, Section 1.0, Pg 5, Rev. 21 AOP-024-BD, Section 2.0, Step 3, Pg 7, Rev. 21
References to be provided:	None
Learning Objective:	AOP-LP-3.24, Obj. 4
Question Origin:	Bank
Comments:	Discuss use of actions contained in AOP verses EOP for loss of vital AC instrument bus with Dan Bacon for question use since HNP does not have any EOP references to the loss of an instrument bus. Phonecon 6/13: Dan agrees that using an AOP for loss of vital AC instrument bus is acceptable to meet this K/A. RO Q29 and ROQ11 have been looked at for double jeopardy and as an exam team we have determined that there is an adequate difference to NOT be a concern.
Tier/Group:	T1/G1

2018 NRC RO 012/BANK/C/A//6-B-401 SH 1922/NONE/2012 NRC RO 12/058 AA2.03/

Given the following plant conditions:
- The unit is in Mode 3
- MDAFW Pumps 'A' and 'B' are in service feeding all 3 SGs
- EDG 'B' is under clearance for maintenance

Subsequently:
- All power from 125 VDC Emergency Bus DP-1B-SB is lost
- One minute later Startup Transformer 1B lockout occurs

Which ONE of the following completes the statement below?

MDAFW Pump 'B' breaker indication on the MCB is ___(1)___ AND the pump motor breaker is ___(2)___ .

A. (1) available

(2) closed

B. (1) available

(2) open

C.✓ (1) not available

(2) closed

D. (1) not available

(2) open

Plausibility and Answer Analysis

Reason answer is correct:125 VDC DP-1B-SB provides control power to the breaker cubicle to provide remote indication of breaker status on the MCB and allows for the operation of the breaker from the MCB. With the loss of control power the indication on the MCB is not available and the ability to remotely operate the breaker is lost as well. The breaker remains shut even with a fault of 6.9KV bus 1B-SB, which would normally open the breaker on UV.

A. Incorrect. (1) Plausible because indication would remain available if the DC bus lost were not the specific supply for DC control power to the 'B' MDAFW. (2) is correct.

B. Incorrect. (1) Plausible because indication would remain available if the DC bus lost were not the specific supply for DC control power to the 'B' MDAFW. (2) Plausible because the breaker normally trips open on UV when its associated 6.9KV bus is de-energized.

C. Correct.

D. Incorrect. (1) is correct. (2) Plausible because the breaker normally trips open on UV when its associated 6.9KV bus is de-energized.

000058 Loss of DC Power / 6

058AA2.03; Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems

(CFR: 43.5 / 45.13)

Importance Rating:	3.5 3.9
Technical Reference:	Drawing 2166-B-401-1922, Rev 12
References to be provided:	None
Learning Objective:	DCP Lesson Plan, Obj. 9
Question Origin:	Bank
Comments:	Discuss if it was required to have a K/A match for both monitor and operate or if the K/A is met by meeting just one of the two topics. Phonecon 10/23: Dan stated that it was acceptable to meet this K/A by addressing either the monitoring or operating portion of this K/A topic based on the question specifics.
Tier/Group:	T1/G1

2018 NRC RO 013/BANK/FUNDAMENTAL//AOP-022-BD/NONE//062 AA2.06/

Which ONE of the following identifies (1) how long the EDG 1A-SA can operate fully loaded without ESW and (2) the reason why?

A.✓ (1) a maximum of one minute

(2) to protect against equipment damage due to overheating

B. (1) a maximum of one minute

(2) to allow adequate time to re-align the equipment to NSW

C. (1) a maximum of five minutes

(2) to protect against equipment damage due to overheating

D. (1) a maximum of five minutes

(2) to allow adequate time to re-align the equipment to NSW

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-022, Loss of Service Water Basis Document the EDG must be secured within 1 minute of operating at full load without cooling water flow. The EDG is considered an essential load and requires the component to be stopped to protect against equipment damage due to overheating.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since NSW is normally the cooling supply for the EDG 1A-SA during non-emergency conditions, however this is incorrect because essitial equipment such as EDG 1A-SA are stopped to protect *against equipment damage due to overheating under these conditions.*

C. Incorrect. The first part is plausible since five minutes is in reference to how long a CSIP has been stopped before a controlled heat is required when restarting the CSIP, however this is incorrect because the requirement for the stopping of a loaded EDG without ESW is 1 minute. Additionally five minutes is a reasonable amount of time to ensure actions can be taken in the procedure prior to reaching the point of checking EDG cooling.

D. Incorrect. The first part is plausible since five minutes is in reference to how long a CSIP has been stopped before a controlled heat is required when restarting the CSIP, however this is incorrect because the requirement for the stopping of a loaded EDG without ESW is 1 minute. Additionally five minutes is a reasonable amount of time to ensure actions can be taken in the procedure prior to reaching the point of checking EDG cooling. The second part is plausible since NSW is normally the cooling supply for the EDG 1A-SA during non-emergency conditions, however this is incorrect because essitial equipment such as EDG 1A-SA are stopped to protect *against equipment damage due to overheating under these conditions.*

000062 Loss of Nuclear Service Water / 4

062AA2.06; Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water:The length of time after the loss of SWS flow to a component before that component may be damaged

(CFR: 43.5 / 45.13)

Importance Rating:	2.8 3.1
Technical Reference:	AOP-022-BD, Section 2.0, Step 2, Pg 10, Rev. 14
References to be provided:	None
Learning Objective:	AOP-LP-022, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 014/BANK/C/A//AOP-017, OP-134.01/NONE/2012 NRC RO 14/065 AK3.04/

Given the following plant conditions:
- The unit is operating at 100% power
- IA header pressure lowers as indicated below:

Time	IA Header Pressure
0750	115 psig
0755	95 psig
0800	85 psig
0805	70 psig
0810	60 psig
0815	45 psig

Which ONE of the following completes the statements below in accordance with AOP-017, Loss Of Instrument Air?

At ___(1)___ an auto shut signal to the Main Feed Reg valves is FIRST generated.

The reason these valves have back up air accumulators is to provide ___(2)___ .

A. (1) 0800

(2) motive force to shut the valves

B. (1) 0800

(2) the ability to maintain the valves open for a short period of time following loss of air

C.✓ (1) 0810

(2) motive force to shut the valves

D. (1) 0810

(2) the ability to maintain the valves open for a short period of time following loss of air

Plausibility and Answer Analysis

Reason answer is correct: During a loss of IA when pressure is < 60# the valves receive an auto shut signal. The motive force to shut each valve is provided by air accumulators for each valve. The reason the air accumulators are neccessary is the valves would fail in the last position if there was not a motive force to shut them.

A. Incorrect. Plausible because at 85 psig it has been determined that spurious valve actuations will begin to occur on the Letdown system due to air pressure not being able to overcome spring pressure to maintain the valve open. The second part of the answer is correct.

B. Incorrect. Plausible because at 85 psig it has been determined that spurious valve actuations will begin to occur on the Letdown system due to air pressure not being able to overcome spring pressure to maintain the valve open. The second part of the answer is plausible because other plant valves have hydraulic accumulators (DEH system for GV's and Throttle valves) to allow operation during a loss of power

C. Correct.

D. Incorrect. Plausible since the pressure is correct and because other plant valves have hydraulic accumulators (DEH system for GV's and Throttle valves) to allow operation during a loss of power.

000065 Loss of Instrument Air / 8

065AK3.04; Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Cross-over to backup air supplies

(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	3.0 3.2
Technical Reference:	AOP-017, Note prior to Step 1, Pg 4, Rev 40 OP-134.01, P&L 17, 18, Pg 8, Rev 44
References to be provided:	None
Learning Objective:	CFW Lesson Plan, Obj. 8.j
Question Origin:	Bank
Comments:	HNP uses backup air supply accumulators on the Main Feed Reg Valves which provides motive force to shut the valve when pressure lowers to 60 psig. Discussed with Dan Bacon using this system for a question since HNP does not have a backup air supply system that has cross over capability. Phonecon 6/13: Dan agrees that using the backup air supply accumulators on the Main Feed Reg Valves is acceptable to meet this K/A.
Tier/Group:	T1/G1

2018 NRC RO 015/NEW/FUNDAMENTAL//ALB-022/NONE/2012 NRC RO 15/077 AA1.03/

Given the following plant conditions:

- The plant is operating at 100% power
- Annunciator ALB-022-5-3, Generator Voltage Balance Relay Operated, Alarms
- The voltage regulator remains in AUTO
- MW and MVAR indications are erratic

Which ONE of the following actions will be taken for these conditions in accordance with APP-ALB-022?

A. Place the Generator Voltage Regulator switch to OFF and use CS-1539, Voltage Setpoint Reference switch to restore Main Generator voltage

B✓ Place CS-1538, Operational Mode switch in Manual, then operate CS-1539, Voltage Setpoint Reference switch to stabilize Generator Stator voltage at 22KV.

C. Adjust the Generator Output Voltage to 230KV with CS-1539, Voltage Setpoint Reference switch. If the adjustment does not work, Trip the Reactor and enter EOP-E-0, Reactor Trip Or Safety Injection.

D. Transfer the Voltage Regulator control to Local by placing CS-1540, Local Control Enable switch in the Local, to stabilize Output Voltage and dispatch an operator to 286 TB Switchgear room. If this action does not work, Trip the Reactor and enter EOP-E-0, Reactor Trip Or Safety Injection.

Plausibility and Answer Analysis

Reason answer is correct: Per APP-ALB-022-5-3, if the Generator voltage regulator is in Auto then **PLACE** the CS-1538, Operation Mode switch, in MANUAL mode and **OPERATE** the CS-1539, Voltage Setpoint Reference switch, to stabilize the Generator Stator Voltage at 22KV.

A. Incorrect. Plausible because during a Generator shutdown the Generator Voltage Regulator is placed in MANUAL and the Isophase bus duct cooling fan control switches are placed in the OFF position and the candidate may have the misconception that the MANUAL position for the Voltage regulator corresponds to the OFF position.

B. Correct.

C. Incorrect. Plausible because the voltage regulator is normally adjusted using CS-1539, Voltage Setpoint Reference and 230 KV is the nominal voltage maintained on the Main Transformer high side. Additionally AOP-006 for turbine/generator trouble provides trip criteria when exceeding limits on the turbine.

D. Incorrect. Plausible because the AVR has the capability to be controlled locally from the switchgear, but event and alarm indications are available locally and AOP-006 for turbine/generator trouble provides trip criteria when exceeding limits on the turbine.

000077 Generator Voltage and Electric Grid Disturbances / 6

077AA1 .03; Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Voltage regulator controls

(CFR: 41.5 and 41.10 / 45.5, 45.7, and 45.8)

Importance Rating:	3.8 3.7
Technical Reference:	APP-ALB-022, Window 5-3, Page 38, Rev 85
References to be provided:	None
Learning Objective:	Main Generator Lesson Plan, Obj. 3.c
Question Origin:	New
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 016/NEW/C/A//E-1, ECA-1.2/NONE//WE04 EK1.3/

Given the following plant conditions:

- A Reactor Trip and Safety Injection has occured
- EOP-E-1, Loss of Reactor Or Secondary Coolant, is being implemented and SI has been reset
- RCS Pressure is 1425 psig and stable
- Containment pressure 2.1 psig
- PZR level is off scale low
- Subcooling is 8°F
- Rad monitor, RM-1RR-3598, RHR Pump 1A is in high alarm and rising
- Window 6-2, RAB Equip A/B Sump Alert Lvl, is lit on MLB-4A-SA and MLB-4B-SB

Which ONE of the following identifies the FIRST procedure to be entered from EOP-E-1 to mitigate the event in progress?

A. EOP-ES-1.2, Post LOCA Cooldown And Depressurization

B. EOP-ES-1.3, Transfer To Cold Leg Recirculation

C. EOP-ECA-1.1, Loss Of Emergency Coolant Recirculation

D✓ EOP-ECA-1.2, LOCA Outside Containment

Plausibility and Answer Analysis

Reason answer is correct: EOP-E-1 step 12 evaluates the status of the plant and directs entry into EOP-ECA-1.2 if high radiation levels are due to a loss of RCS inventory outside of containment.

A. Incorrect. The plausible since RCS pressure is above the Low Pressure ECCS system injection pressure and subcooling is less than 10˚F and PRZ level is less than 10% SI termination criteria is not satisfied, however this is incorrect because the leak is outside of containment and must be addressed first prior to depressurizing the RCS.

B. Incorrect. The plausible since this is the expected procedure transition when the ECCS system operates as designed to collect the coolant in the Emergency Recirculation sump to allow for long term cooling of the RCS, however this is incorrect because the coolant is bypassing the Emergency Recirculation sump and other actions must be taken to ensure the coolant is going to the desired location in the ECCS system.

C. Incorrect. The plausible since the coolant is bypassing the Emergency Recirculation sump and the ECCS system is not capable of returning the coolant collect outside of Containment to the RCS as designed the candidate will eventually transition to this procedure, however this is incorrect because the candidate with first attempt other actions to isolate the leakage outside of Containment to ensure the coolant is going to the desired location in the ECCS system.

D. Correct.

W/E04 LOCA Outside Containment / 3

WE04EK1.3; Knowledge of the operational implications of the following concepts as they apply to the (LOCA Outside Containment): Annunciators and conditions indicating signals, and remedial actions associated with the (LOCA Outside Containment).

(CFR: 41.8 / 41.10, 45.3)

Importance Rating:	3.5 3.9
Technical Reference:	EOP-E-1, Step 12.e, Pg 16, Rev 4 EOP-ECA-1.2, Pg 2, Rev 0
References to be provided:	None
Learning Objective:	EOP-LP-3.03, Obj. 2.d
Question Origin:	New
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 017/BANK/C/A//CSFST/NONE/2012 NRC RO 17/WE05 EK2.2/

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
- SG NR levels are 41% and lowering
- Containment pressure is 3.3 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)__.

A.✓ (1) 39%

(2) 195 KPPH

B. (1) 39%

(2) 205 KPPH

C. (1) 24%

(2) 195 KPPH

D. (1) 24%

(2) 205 KPPH

Plausibility and Answer Analysis

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 conditions).

A. Correct

B. Incorrect. The first part is correct. The second part is plausible since one of the parameters monitors for heat sink is below the required value the candidate may misapply this information and determine that both SG level and AFW flow need to be above the adverse Containment requirement to preclude entry into EOP-FR-H.1.

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

D. Incorrect. The first part is plausible see C (1). The second part is plausible see answer B (2).

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, Obj. 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 018/BANK/FUNDAMENTAL//FR-Z.1, BKGRD/NONE/2011 NRC RO 17/WE11 EK3.4/

Why does EOP-ECA-1.1, Loss of Emergency Coolant Recirculation take precedence over EOP-FR-Z.1, Response to High Containment Pressure for operation of the Containment Spray Pumps?

The reason is based on _______ .

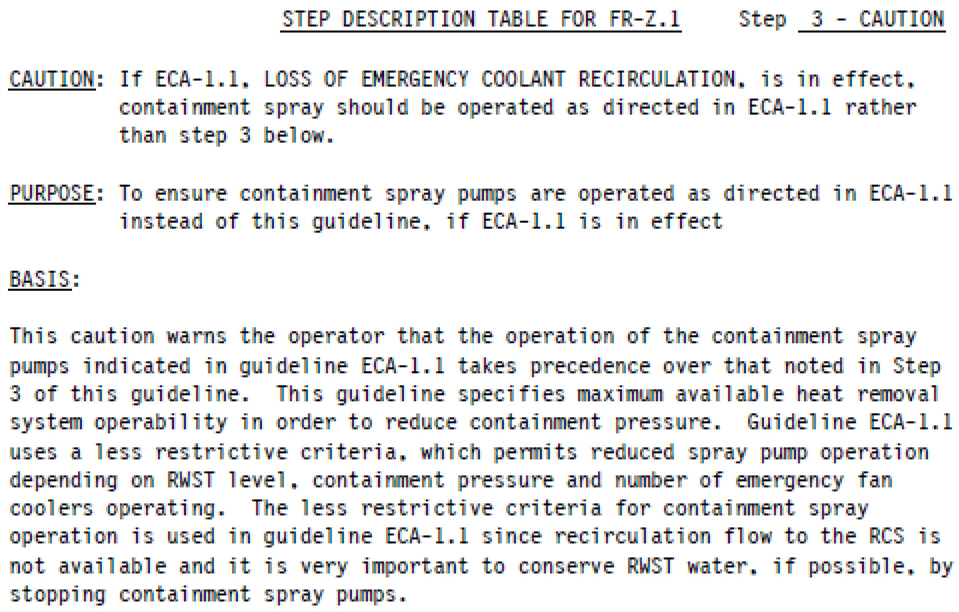
A. maintaining Containment iodine removal

B. maintaining Containment heat removal

C. limiting Containment pressure

D✓ conserving RWST inventory

Reason answer is correct:



STEP DESCRIPTION TABLE FOR FR-Z.1 Step 3 - CAUTION

CAUTION: If ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, containment spray should be operated as directed in ECA-1.1 rather than step 3 below.

PURPOSE: To ensure containment spray pumps are operated as directed in ECA-1.1 instead of this guideline, if ECA-1.1 is in effect

BASIS:

This caution warns the operator that the operation of the containment spray pumps indicated in guideline ECA-1.1 takes precedence over that noted in Step 3 of this guideline. This guideline specifies maximum available heat removal system operability in order to reduce containment pressure. Guideline ECA-1.1 uses a less restrictive criteria, which permits reduced spray pump operation depending on RWST level, containment pressure and number of emergency fan coolers operating. The less restrictive criteria for containment spray operation is used in guideline ECA-1.1 since recirculation flow to the RCS is not available and it is very important to conserve RWST water, if possible, by stopping containment spray pumps.

A. Incorrect. Plausible since containment iodine removal is a function of the containment spray pumps and is a factor in determination of securing containment spray pumps in E-0/E-1.

B. Incorrect. Plausible since EOP-ECA-1.1 uses the number of of cnmt fan coolers operating (heat removal capability) in evaluating the number of containment spray pumps to run.

C. Incorrect. Plausible since EOP-ECA-1.1 uses containment pressure in evaluating the number of containment spray pumps to run.

D. Correct.

W/E11 Loss of Emergency Coolant Recirculation / 4

WE11EK3.4; Knowledge of the reasons for the following responses as they apply to the (Loss of Emergency Coolant Recirculation): RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

(CFR: 41.5 / 41.10, 45.6, 45.13)

Importance Rating:	3.6 3.8
Technical Reference:	ERG-BKGRD-FR-Z.1, Step 3 Caution, Pg 8, Rev. 1c
References to be provided:	None
Learning Objective:	EOP-LP-2.3/3.3, Obj. 1.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G1

2018 NRC RO 019/NEW/C/A//GFES/NONE//001 AK1.22/

Given the following plant conditions:
- The unit is operating at 70% power
- NI Gain Adjustment is in progress
- N-43 has been completed and the BOP is performing adjustment on N-44

Subsequently the following conditions are observed:
- T_{avg} is rising
- T_{ref} remains constant
- Control Rods are in motion
- PRZ pressure and level are rising

Which ONE of the following completes the statements below?

A(An) __(1)__ event in progress.

As a result of the above, Axial Flux Distribution (AFD) will __(2)__ the original value.

(Assuming NO operator action)

A. (1) Inadvertant dilution

(2) LOWER then return to

B. (1) Inadvertant dilution

(2) be LESS negative than

C. (1) Continuous rod withdrawal

(2) LOWER then return to

D✓ (1) Continuous rod withdrawal

(2) be LESS negative than

Plausibility and Answer Analysis

Reason answer is correct: The procedure for performing NI Gain adjustments places the rod control system in manual to preclude a spurious rod motion event due to the adjustment. The combination of rising T_{avg}, PRZ level and PRZ pressure combined with rod motion are indications that the RCS is responding to a Continuous rod withdrawl event. Control rod motion out of the core exposes more fuel to nuclear reactions resulting in higher flux being genereated in the upper region of the core resulting in less negative AFD.

A. Incorrect. The first part is plausible since RCS Temperature/Pressurizer response are correct for an inadvertant dilution, however this is incorrect because the rods are placed in manual to support NI gain adjustments therefore an inadvertant dilution would not cause rod motion. The second part is plausible since rod motion should not occur for an inadverdant dilution event with rod contol in manual, however this is incorrect for the conditions of the question because outward rod motion is in progress which will result in AFD becoming less negative.

B. Incorrect. The first part is plausible since RCS Temperature/Pressurizer response are correct for an inadvertant dilution, however this is incorrect because the rods are placed in manual to support NI gain adjustments therefore an inadvertant dilution would not cause rod motion. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since rod motion should not occur with rod contol in manual therefore AFD should remain the same, however this is incorrect for the conditions of the question because outward rod motion is in progress which will result in AFD becoming less negative.

D. Correct.

000001 Continuous Rod Withdrawal / 1

001AK1.22; Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal: Delta flux (ΔI)

(CFR 41.8 / 41.10 / 45.3)

Importance Rating:	3.2 3.6
Technical Reference:	GFES, Reactor Theory
References to be provided:	None
Learning Objective:	TAA-LP-3.25, Obj. 1.a
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 020/NEW/FUNDAMENTAL//FHP-400/NONE/EARLY/036 AK3.02/

Given the following plant conditions:
- The unit is in Mode 6
- Fuel movement is in progress

Subsequently the Manipulator Crane upward motion suddenly stops and the Overload indicator is illuminated

Which ONE of the following completes the statement below?

The Manipulator Crane Overload Interlock stopped upward motion when the hoist detected ___(1)___ pounds above the weight of the mast and the ___(2)___ .

A.✓ (1) 150

(2) Fuel Assembly

B. (1) 150

(2) Rod Control Cluster Assembly

C. (1) 430

(2) Fuel Assembly

D. (1) 430

(2) Rod Control Cluster Assembly

Plausibility and Answer Analysis

Reason answer is correct: Overload Interlock - Prevents fuel damage by stopping the hoist up travel when hoist load is 150 lbs above weight of mast and fuel assembly. (variable set points depending on Load Select switch position).

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since the manipulator setpoint is varied based on the weight of the fuel assembly with or without an RCCA, however this is incorrect because the overload interlock is associated with the manipulator main hoist.

C. Incorrect. The first part is plausible since it is the weight of the Slack Cable interlock therefore the candidate may have the misconception that the overload interlock is set to 430 lbs vice 150 lbs. The second part is plausible since the manipulator setpoint is varied based on the weight of the fuel assembly with or without an RCCA, however this is incorrect because the overload interlock is associated with the manipulator main hoist.

D. Incorrect. The first part is plausible since it is the weight of the Slack Cable interlock therefore the candidate may have the misconception that the overload interlock is set to 430 lbs vice 150 lbs. The second part is correct.

000036 Fuel-Handling Incidents / 8

036AK3.02; Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents: Interlocks associated with fuel handling equipment

(CFR 41.5,41.10 / 45.6 / 45.13)

Importance Rating:	2.9 3.6
Technical Reference:	FHP-400, Attachment 3, Pg 69, Rev 3
References to be provided:	None
Learning Objective:	FHS Lesson Plan, Obj. 6
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 021/BANK/C/A//AOP-016/NONE/2006 NRC RO 33/037 AA2.01/EARLY

Given the following plant conditions:

- The unit is operating at 100% power

Subsequently

- REM-01TV-3534, Condenser Vacuum Pump Rad monitor, is indicating 1.32 x 10^{-06} µCi/cc and rising
- AOP-016, Excessive Primary Plant Leakage, is in progress
- The OATC has completed a leak rate estimate calcuation to quantify the leak rate

Which ONE of the following indications will serve to verify the value of actual primary to secondary leak rate?

A. Local surveys of Steam Generator Blowdown Lines

B. Trend on Turbine Building Vent Stack Effluent, RM-1TV-3536-1

C. Alarm status of Main Steam Line Radiation Monitors RM-01MS-3591 SB, 3592 SB, or 3593 SB

[✓ D.] Condenser Vacuum Pump Effluent Monitor indication and a conversion factor supplied by Chemistry after sampling

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-016 Primary-To-Secondary leak rate is estimated using ONE of the methods listed below
Condenser Vacuum Pump Rad Monitor, REM-01TV-3534 and Curve H-X-15
Condenser Vacuum Pump Rad Monitor, REM-01TV-3534 and Conversion factor provided by Chemistry after sampling has commenced

A. Incorrect. Plausible since local surveys of SGBD lines are directed to be performed in order to determine the which SG is leaking, however this is incorrect as this method does not determine the amount of leakage from Primary to Secondary.

B. Incorrect. Plausible since the Turbine Building Vent Stack is one of the radiation monitors used to determine if an Offsite Dose Calculation is required to be performed, however this is incorrect as this method does not determine the amount of leakage from Primary to Secondary.

C. Incorrect. Plausible since Main Steam line radiation monitor levels are directed to be monitored in order to determine the which SG is leaking, however this is incorrect as this method does not determine the amount of leakage from Primary to Secondary.

D. Correct.

000037 Steam Generator Tube Leak / 3

037AA2.01; Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Unusual readings of the monitors; steps needed to verify readings

(CFR: 43.5 / 45.13)

Importance Rating:	3.0 3.4
Technical Reference:	AOP-016, Attachment 1, Pg 14, Rev. 56
References to be provided:	None
Learning Objective:	AOP-LP-3.16, Obj. 3
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 022/BANK/C/A//ALB-018/NONE/2011 NRC RO 22/051 AA2.02/

Given the following plant conditions:
- The plant is operating at 45% power
- Condenser vacuum is degrading as indicated below:

Time	Zone 1 Pressure	Zone 2 Pressure
0900	4.4" Hg absolute	4.8" Hg absolute
0905	4.8" Hg absolute	5.3" Hg absolute
0910	5.4" Hg absolute	5.9" Hg absolute
0915	7.2" Hg absolute	7.8" Hg absolute
0920	7.6" Hg absolute	8.4" Hg absolute

What is the EARLIEST time that an automatic Turbine/Reactor Trip will be generated?

A. 0905

B.✓ 0910

C. 0915

D. 0920

Plausibility and Answer Analysis

Reason answer is correct: With the 1st stage pressure below 60% the Low vacuum trip of the main turbine occurs automatically when condenser vacuum is above 5" HgA on condenser zone 1. This condition is first satisfied at 0910

A. Incorrect. Plausible because the value is above the 5" HgA setpoint if detemined the low vacuum trip occurs automatically from zone 2.

B. Correct.

C. Incorrect. Plausible because the value is above the 7.5" HgA setpoint if detemined the low vacuum trip occurs automatically from zone 2, but this occurs when 1st stage pressure is above 60%.

D. Incorrect. Plausible because the value is above the 7.5" HgA setpoint if detemined the low vacuum trip occurs automatically from zone 1, but this occurs when 1st stage pressure is above 60%.

000051 Loss of Condenser Vacuum / 4

051AA2.02; Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip

(CFR: 43.5 / 45.13)

Importance Rating:	3.9 4.1
Technical Reference:	APP-ALB-018, Window 1-1, Pg 3, Rev 21
References to be provided:	None
Learning Objective:	AOP-LP-3.12, Obj. 4.a
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 023/NEW/C/A//AOP-004/NONE//068 AA1.28/

Given the following plant conditions:

- The unit was operating at 100% when a fire occurred in the MCR
- The crew has entered AOP-004, Remote Shutdown, and has relocated to the ACP
- The crew has secured the CSIP's and the ASI pump has started

Which ONE of the following actions by the operators at the ACP completes the statements below to control PZR level and pressure?

To support automatic control of 'A' and 'B' PRZ heaters, PK-444A2.A, Pressurizer Pressure controller, will be placed in automatic with the setpoint adjusted to ___(1)___ .

To prevent PRZ level from exceeding the high level band required by the procedure the operators will ___(2)___ .

A. (1) 67%

(2) secure the ASI pump

B.✓ (1) 67%

(2) perform a plant cooldown

C. (1) 75%

(2) secure the ASI pump

D. (1) 75%

(2) perform a plant cooldown

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-004 step 16.b (2) PK-444A2.A will be placed in automatic with a setpoint adjusted to 67% (2235 psig) to support automatic control of 'A' and 'B' PRZ heaters.

Master pressure controller at the ACP, PK- 444A2. A controller setpoint of 0% equates to 1700 psig setpoint and 100% to 2500 psig setpoint (1700-2500 psig range on PT-444). Therefore, 67% = 2235 psig nominal setpoint:
2500 - 1700 = 800 psig total meter span
2235 - 1700 = 535 psig
535/800 = 67%

A note prior to step 21 states that with RCS temperature stable, PRZ level will rise slowly with the ASI pump in service....Pressurizer level may not rise to 75% for several hours. Maintaining RCS Tcold between 555°F and 559°F is preferred. High Pressurizer level may require a plant cooldown below this band. Step 21 is a continuous action step of the procedure which is to maintain Pressurizer level between

25% and 75%. To reduce Pressurizer level the RCS is cooled down to shrink the RCS by controlling AFW flow and dumping steam using SG PORVs. The ASI pump is NOT secured until after a RCS cooldown is used to shrink the RCS and PRZ level is >75% and cannot be controlled with RCS cooldown. The question asks what the operators do to PREVENT PRZ level from exceeding the high level band not what to do AFTER the PRZ level high level band is exceeded.

A. Incorrect. The first part is correct. The second part is plausible since the ASI system will be running (due to fire in MCR the CSIP is secured IAW AOP-004 step 9). With the ASI pump operating and RCS temperature maintained stable the PRZ level will slowly rise. To maintain PRZ level in band the AOP directs the operators to cool the RCS. Attachment 11, Actions if ASI Pump Starts also has the operators maintain Pressurizer level between 25% and 75% by COOLDOWN to shrink the RCS. The ASI pump is secured ONLY if PRZ level is >75% and cannot be controlled with RCS cooldown.

B. Correct.

C. Incorrect. The first part is plausible since 75% is the upper control band limit for Pressurizer level (25% - 75% is the control band) but if the setpoint is adjusted to 75% PRZ pressure would be set to 2300 psig which would not support automatic control of 'A' and 'B' PRZ heaters. The second part is plausible since the ASI system will be running (due to fire in MCR the CSIP is secured IAW AOP-004 step 9). With the ASI pump in operating and RCS temperature maintained stable the PRZ level will slowly rise. To maintain PRZ level in band the AOP directs the operators to cool the RCS. Attachment 11, Actions if ASI Pump Starts also has the operators maintain Pressurizer level between 25% and 75% by COOLDOWN to shrink the RCS. The ASI pump is secured ONLY if PRZ level is >75% and cannot be controlled with RCS cooldown.

D. Incorrect. The first part is plausible since 75% is the upper control band limit for Pressurizer level (25% - 75% is the control band) but if the setpoint is adjusted to 75% PRZ pressure would be set to 2300 psig which would not support automatic control of 'A' and 'B' PRZ heaters. The second part is correct.

000068 Control Room Evacuation / 8

068 AA1.28; Ability to operate and / or monitor the following as they apply to the Control Room Evacuation: PZR level control and pressure control

(CFR 41.7 / 45.5 / 45.6)

Importance Rating:	3.8 4.0
Technical Reference:	AOP-004, Section 3.1, Step 12.b, Pg 23, Rev. 68 AOP-004, Section 3.1, Step 21.a, Pg 32, Rev. 68
References to be provided:	None
Learning Objective:	AOP-LP-3.4, Obj. 4
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 024/BANK/FUNDAMENTAL//AOP-032/NONE/2012 NRC RO 24/076 AK2.01/

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

Subsequently:
- ALB-026-2-1, Gross Failed Fuel Detector Trouble, is received
- Chemistry reports RCS activity is rising, and the latest sample is 1.4 microcuries per gram dose equivalent I-131

Which ONE of the following identifies (1) the source of radiation detected by the Gross Failed Fuel Detector AND (2) what action will be directed by AOP-032, High RCS Activity, to reduce RCS activity levels?

A✓ (1) neutron radiation from delayed neutrons

(2) place the Cation demineralizer in service

B. (1) neutron radiation from delayed neutrons

(2) place the second Mixed bed demineralizer in service

C. (1) gamma radiation from decay of fission products

(2) place Cation demineralizer in service

D. (1) gamma radiation from decay of fission products

(2) place the second Mixed bed demineralizer in service

Plausibility and Answer Analysis

Reason answer is correct: The GFFD is a BF3 proportional counter that measures neutron radiation from long lived delayed neutron emitters. IF alarm is due to high neutron activity then refer to AOP-032, High RCS Activity. IAW AOP-032 if a Cation demin is not in service then place one in service IAW OP-107.02. OP-107.02 directs throttling flow through the in service Mixed-bed demin such that some flow is diverted throught the cation deminerlizer (not to exceed 60 gpm).

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since Mixed bed demineralizers are normally utilized during operation, but they are utilized for removal of particles, not reduction of activity (they do not remove Cesium like the Cation bed demin does). If the Mixed bed demineralizer was effective in removing activity, maximizing flow would result in maximum purification and reduction of activity

C. Incorrect. The first part is plausible because increased decay gamma activity will be present in the fuel following a fuel failure. The second part of the answer is correct.

D. Incorrect. The first part is plausible see answer C (1). The second part is plausible see answer B (2).

000076 High Reactor Coolant Activity / 9

076AK2.01; Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors

(CFR 41.7 / 45.7)

Importance Rating:	2.6 3.0
Technical Reference:	GFFD Student Text, Pg 1, Rev. 2, AOP-032, Section 2.0, Pg 3 and 7, Rev. 20
References to be provided:	None
Learning Objective:	GFFD Lesson Plan, Obj. 3.a and 4.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 025/NEW/FUNDAMENTAL//EOP-ES-1.1/NONE//WE02 EG2.4.20/

Given the following plant conditions:

- The crew is performing EOP-ES-1.1, SI Termination, and have secured one CSIP and realigned the pump to the normal Charging line

Concerning SI reinitiation Criteria during these conditions, which ONE of the following completes the statement below?

EOP-ES-1.1 cautions that __(1)__ flow through the Charging and SI lines may cause __(2)__ as indicated by oscillating discharge pressure.

A. (1) isolating

(2) a relief valve to lift

B. (1) isolating

(2) cavitation

C✓ (1) simultaneous

(2) runout

D. (1) simultaneous

(2) cavitation

Plausibility and Answer Analysis

Reason answer is correct: The caution in EOP-ES-1.1 prior to Checking SI Reinitiation Criteria states simultaneous flow through the charging and SI lines may cause CSIP runout (as indicated by oscillating discharge pressure).

A. Incorrect. The first part is plausible since the SI Reinitiation Foldout isolates flow to the charging line prior to establishing SI flow through the BIT, however this is incorrect because the miniflow valve are open during this alignment and the CSIP will not be impacted. The second part is plausible for this action may result in discharge pressure oscillations, however it is not correct in accordance with EOP-ES-1.1. This distractor was chosen to support plausibility vice balance of the distractors due to implausibilty of runout occuring with the system isolated.

B. Incorrect. The first part is plausible since the SI Reinitiation Foldout isolates flow to the charging line prior to establishing SI flow through the BIT, however this is incorrect because the miniflow valve are open during this alignment and the CSIP will not be impacted. The second part is plausible for this action if the candidate misapplies the generic fundamental concept of pump cavitation which results in discharge pressure oscillations, however it is not correct in accordance with EOP-ES-1.1.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible for this action if the candidate misapplies the generic fundamental concept of pump cavitation which results in discharge pressure oscillations, however this is incorrect as cavitation is the result of low NPSH.

W/E02 SI Termination / 3

WE02 EG2.4.20; 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-ES-1.1, Step 14 Caution, Pg 18, Rev. 2
References to be provided:	None
Learning Objective:	EOP-LP-3.1, Obj. 5
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 026/BANK/C/A//USERS GUIDE, ES-1.2/NONE/2009A NRC RO 26/WE03 EK2.2/

Given the following plant conditions:

- A LOCA has occurred
- RCS pressure is 1350 psig and stable
- Containment pressure is 3.5 psig and stable
- The crew is performing actions contained in EOP-ES-1.2, Post LOCA Cooldown and Depressurization

Which ONE of the following describes the method that will be used to perform the cooldown of the RCS?

Perform the cooldown using _______ .

A✓ S/G PORVs at less than 100°F per hour

B. S/G PORVs at the maximum achievable rate

C. Condenser Steam Dumps at less than 100°F per hour

D. Condenser Steam Dumps at maximum achievable rate

Plausibility and Answer Analysis

Reason answer is correct: SG PORVs will be used for the cooldown because the condenser is not available due to an auto closure of the MSIV's when Containment pressure reached 3 psig and EOP-ES-1.2 limits cooldown to 100°F/hour.

A. Correct.

B. Incorrect. SG PORVs will be used for the cooldown because the condenser is not available. The Cooldown Rate is incorrect. Other EOPs perform a max rate cooldown such as in EOP-E-3 but EOP-ES-1.2 limits cooldown to 100°F/hour.

C. Incorrect. Condenser steam dumps are not available because at 3 psig in Containment a MSLI actuated to shut all MSIVs. Rate is correct.

D. Incorrect. Condenser steam dumps are not available because at 3 psig in Containment a MSLI actuated to shut all MSIVs. Plausible because it is the normal method of cooldown. The Cooldown Rate is incorrect. Other EOPs such as EOP-E-3 perform a max rate cooldown but EOP-ES-1.2 limits cooldown to 100°F/hour.

W/E03 LOCA Cooldown - Depressurization / 4

WE03EK2.2; Knowledge of the interrelations between the (LOCA Cooldown and Depressurization) 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:	3.7 4.0
Technical Reference:	EOP-Users Guide, Step 6.19, Pg 47, Rev. 49 EOP-ES-1.2, Step 10, Pg 12, Rev 2.
References to be provided:	None
Learning Objective:	LP-EOP-3.5, Obj. 5.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 027/BANK/FUNDAMENTAL//FR-P.1, BKGRD/NONE/2012 NRC RO 26/WE08 EK3.3/

Given the following plant conditions:

- The crew is implementing EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock
- RCS subcooling does not support SI termination but it does support starting an RCP

Which ONE of the following describes the basis for why it is desirable to re-start an RCP under these conditions?

A. Establish PRZ Spray control

B. Supply additional heat input into the RCS

C. Promote heat transfer from the SGs to the RCS

D✓ Promote mixing of the Safety Injection water and RCS

Plausibility and Answer Analysis

Reason answer is correct: Per the WOG Background document for EOP-FR-P.1 to reduce the likelihood of a PTS condition the start of an RCP should be attempted in order to mix cold SI water with warm RCS water.

A. Incorrect. Plausible since the preferred method of RCS pressure control is the use of PRZ Spray which requires an RCP running, however this is incorrect because this is not the desired reason for the actions under the conditions of the question stem.

B. Incorrect. Plausible since the start of a RCP will provide pump heat which reduces the low temperature concern of a PTS condition, however this is incorrect because this is not the desired reason for the actions under the conditions of the question stem.

C. Incorrect. Plausible since if SG where at a higher pressure and temperature than the RCS it will be a heat source not a heat sink and the more effective heat transfer method is to force circulation of the RCS vice natural circulation, however this is incorrect because this is not the desired reason for the actions under the conditions of the question stem.

D. Correct.

W/E08 RCS Overcooling - Pressurized Thermal Shock / 4

WE08EK3.3; Knowledge of the reasons for the following responses as they apply to the (Pressurized Thermal Shock): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

(CFR: 41.5 / 41.10, 45.6, 45.13)

Importance Rating:	3.7 3.8
Technical Reference:	ERG-BKGRD-FR-P.1, Step 6, Pg 28, Rev.2
References to be provided:	None
Learning Objective:	EOP-LP-3.14, Obj. 4.d
Question Origin:	Bank
Comments:	None
Tier/Group:	T1/G2

2018 NRC RO 028/BANK/C/A//AOP-018, ALB-008/AOP-018 ATTACHMENT 2//003 K6.02/

Given the following plant conditions:

- The unit is operating at 100% power

Subsequently:

- ALB-008-5-3, RCP-C Seal #1 Leakoff High Low Flow, is in alarm
- ALB-008-5-4, RCP-C Seal #2 Leakoff High Flow, is in alarm
- ALB-008-5-5A, RCP-C Standpipe High Level, is in alarm
- RCP ‘C’ #1 seal leakoff is indicating < 1 gpm
- RCP ‘C’ #1 seal ΔP is indicating > 400 psid
- VCT level is slowly lowering
- RCDT level is slowly rising

Which ONE of the following describes the condition of the 'C' RCP seal package?

(Reference provided)

A. Seal injection flow has been lost

B. #1 seal has failed

C✓ #2 seal has failed

D. #3 seal has failed

Plausibility and Answer Analysis

Reason answer is correct: AOP-018 Attachment 2 Specific Symptoms of Seal Malfunctions #2 Seal failed. #2 seal leakoff flow greater than 1.1 gpm (greater than 12.0 gpm if #1 seal is failed) with a corresponding reduction in #1 seal leakoff flow. #3 seal leakoff should remain fairly constant. In the absence of additional guidance, if the No.2 seal flow exceeds 1.1 gpm, follow the procedures for shutdown of the RCP. The alarm setpoint for ALB-008 window 5-4, RCP-C Seal #2 Leakoff High Flow, is 1.0 gpm. With this condition present and a #1 seal leakoff flow < 1.0 gpm the cause of window 5-4 is failure of # 2 seal.

A. Incorrect. Plausible since window 5-3, Seal #1 leakoff high/ low flow, is in alarm and # 1 seal leakoff is < 1 gpm, however this is incorrect because seal injection flow is provided by the CSIP and VCT level continues to lower indicating the CSIP is running taking a suction from the it's normal source, which is the VCT.

B. Incorrect. Plausible since window 5-3, Seal #1 leakoff high/ low flow, is in alarm , however this is incorrect because the # 1 seal leakoff is < 1 gpm and the #1 seal ΔP is > 400 psid, which is indication of seal blockage vice seal failure.

C. Correct.

D. Incorrect. Plausible since in the RCDT level is rising and 400 cc/hr from the #3 seal leak off is aligned to the RCDT, however this is incorrect because the RCP Standpipe high level is in alarm and therefore abnormally frequent filling of the standpipe is not required, which is an indication that the #3 seal is failed.

003 Reactor Coolant Pump / 4

003K6.02; Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: RCP seals and seal water supply

(CFR: 41.7 / 45.5)

Importance Rating:	2.7 3.1
Technical Reference:	AOP-018 Attachment 2, Pg 24 and 25, Rev. 49 APP-ALB-008, Window 5-4, Pg 32, Rev. 25
References to be provided:	AOP-018 Attachment 2, Pg 24 and 25, Rev. 49
Learning Objective:	LP-AOP-3.18, Obj. 3.b
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 029/NEW/C/A//AOP-024/NONE/EARLY/004 K2.06/

Given the following plant conditions:
- The unit is operating at 100%
- Pressurizer Level Control is selected to 459/460

Which ONE of the following identifies the power supply to the controlling PRZ Level channel?

A✓ SI

B. SII

C. SIII

D. SIV

Plausibility and Answer Analysis

Reason answer is correct: The normal position for PRZ level control is 459/460 or position 2. In this position Level Channel 459 is the controlling channel providing input to the PRZ level control system. The power supply to level channel 459 is Instrument Bus (IDP) 1A-SI.

A. Correct.

B. Incorrect. Plausible since level channel 460 is powered from SII and the control switch is in the 459/460 position, however this is incorrect because 460 will only provide input to the protection circuitry for PRZ Level.

C. Incorrect. Plausible since level channel 461 is powered from SIII and it is available to provide input to the PRZ level control circuitry, however this is incorrect because the control switch is in the 459/460 position, therefore channel 461 will not provide input to the control or protection circuitry for PRZ Level.

D. Incorrect. Plausible since the logic circuitry for certain plant systems (Safety Injection for example) only use channels supplied by 3 of the 4 instrument bus power supplies therefore the candidate may misapply this knowledge to the PRZ level control system and determine that SIV is a power supply to the controlling PRZ level channel, however this is incorrect because the PRZ level control system is only provided inputs powered from SI, SII or SIII.

004 Chemical and Volume Control / 2

004K2.06; Knowledge of bus power supplies to the following:Control instrumentation

(CFR: 41.7)

Importance Rating:	2.6 2.7
Technical Reference:	AOP-024, Attachment 1, Pg 28, Rev 57 AOP-024-BD, Section 1.0, Pg 5, Rev 21
References to be provided:	None
Learning Objective:	PRZLC Lesson Plan, Obj 3.a
Question Origin:	New
Comments:	RO Q11 and RO Q29 have been looked at for double jeopardy and as an exam team we have determined that there is an adequate difference to NOT be a concern.
Tier/Group:	T2/G1

2018 NRC RO 030/BANK/C/A//OP-185/NONE//004 K6.31/

Given the following plant conditions:
- At 0700 the unit is operating at 100% power
  - 'A' CSIP is running

- At 0701 the following occurs:
  - 'A' CSIP experiences a speed changer malfunction. As a result, discharge pressure and flow lowers.
  - Seal injection flow to all RCPs lowers to 3.4 gpm

- At 0705:
  - Seal injection flow to all RCPs remains at 3.4 gpm

Which ONE of the following identifies the operational status and impact on the Alternate Seal Injection (ASI) pump based on these conditions?

The ASI pump is _______ .

A. not running but is in standby for a subsequent auto start

B✓ not running and will not start automatically due to lockout

C. running and operators must take local readings of RCP Seal Injection flow

D. running and must be secured promptly to prevent overpressurizing the discharge piping

Plausibility and Answer Analysis

Reason answer is correct: The loss of flow to (2) out of three (3) flow switches will initiate three (3) timers; The first timer will automatically detonate the two (2) squib valves (1ASI-21 & 1ASI-22) after a 2.5 minutes time delay. The second timer will start the ASI pump after 2.75 minutes elapsed time. The third timer will stop the pump from running if the flow to two (2) out of three (3) flow switches hasn't been restored above 4.0 gpm, within 3.5 minutes elapsed time. The third timer will also prevent the ASI pump from being started after 3.5 minutes has elapsed and flow is not restored. If flow is restored above 4.0 gpm prior to 2.5 minutes, all three timers will reset and no initiation of the pump or squib valves will occur. However, if flow is restored above 4.0 gpm after the system timer has actuated local actions will be required to restore the system based on the time elapsed. For example if the 2.5 minute timer actuates and flow is restored before the 2.75 minute timer actuates, the squib valves will be fired by the 2.5 minute timer, but the ASI pump will not start because the 2.75 and 3.5 minute timers will have reset.

A. Incorrect. Plausible since the values for the ASI pump timers are 2.5 and 2.75 minutes, the candidate may misapply the values of the timers to the seal injection values and determine that the ASI system has not met the requirements to automatically start and therefore remain in standby, however this is incorrect because 4 gpm is the minimum seal injection value that must be restored to prevent the ASI system from automatically actuating during a low seal injection flow condition.

B. Correct.

C. Incorrect. Plausible since the 2.5 and 2.75 minute timers have elapsed, the squib valves and the ASI pump should have actuated to restore seal injection flow, however this is incorrect because 4 minutes have elapsed and the ASI system will automatically lockout the pump if seal injection flow is not restored before the 3.5 minute timer elapses.

D. Incorrect. Plausible since the 2.5 and 2.75 minute timers have elapsed, the squib valves and the ASI pump should have actuated and seal injection flow has not been restored the candidate may determine that because the ASI pump is a positive displacement pump damage to the system piping may occur if the ASI pump continues to run, however this is incorrect because 4 minutes have elapsed and the ASI system will automatically lockout the pump if seal injection flow is not restored before the 3.5 minute timer elapses.

004 Chemical and Volume Control / 2

004K6.31; Knowledge of the effect of a loss or malfunction on the following CVCS components: Seal injection system and limits on flow range

(CFR: 41.7 / 45.7)

Importance Rating:	3.1	3.5
Technical Reference:	OP-185, P&L #1, Section 5.2 Note, Pg 4 and 9, Rev. 12	
References to be provided:	None	
Learning Objective:	ASI Lesson Plan, Obj. 6	
Question Origin:	Bank	
Comments:	None	
Tier/Group:	T2/G1	

2018 NRC RO 031/BANK/FUNDAMENTAL//EOP-ES-1.3/NONE/2013 NRC RO 31/005 A1.05/

-----------------------Which ONE of the following identifies (1) the MINIMUM Containment Wide Range sump level required to place the RHR system in Cold Leg Recirculation in accordance with EOP-ES-1.3, Transfer To Cold Leg Recirculation AND (2) the basis for this level?

A. (1) 137.5 inches

(2) ensures the recirculation sump pH level is acceptable

B. (1) 137.5 inches

(2) ensures the recirculation sump strainers are completely submerged

C. (1) 142 inches

(2) ensures the recirculation sump pH level is acceptable

D✓ (1) 142 inches

(2) ensures the recirculation sump strainers are completely submerged

Plausibility and Answer Analysis

Reason answer is correct: Following a LOCA large enough to require transfer to the recirculation phase of the ECCS, containment water level is expected to be above the strainer modules ("top hats") and the vortex breakers located inside the recirculation sump (Design calculations for the limiting case predict a water level of 223 FEET, 8 INCHES, approximately 142 INCHES as read on the wide range containment sump level instruments or 87% as read on the recirculation sump level instruments.). Degraded sump performance could only occur if water level fell below this level or the strainer modules experienced excessive clogging.

A. Incorrect. The first part is plausible because both the RHR pumps and the Containment Spray pumps take a suction from the containment sump and the 137.5 inch level is the correct amount to allow the Containment Spray pumps to take a suction from the recirculation sump. The second part is plausible since both the Containment Spray and RHR systems will be in operation for a LOCA event in which the RWST is depleted to the point that shifting suctions the to the Containment Sump is required. With Containment Spray in operations the Spray Additive tank will be part of the suction supply to the pump. It is reasonable to believe that the sodium hydroxide in the tank requires a minimum amount of water level to ensure that the chemicals when mixed with the LOCA water in the Containment sumps combine to maintain the correct pH level.

B. Incorrect. The first part is plausible because both the RHR pumps and the Containment Spray pumps take a suction from the containment sump and the 137.5 inch level is the correct amount to allow the Containment Spray pumps to take a suction from the recirculation sump. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since both the Containment Spray and RHR systems will be in operation for a LOCA event in which the RWST is depleted to the point that shifting suctions the to the Containment Sump is required. With Containment Spray in operations the Spray Additive tank will be part of the suction supply to the pump. It is reasonable to believe that the sodium hydroxide in the tank requires a minimum amount of water level to ensure that the chemicals when mixed with the LOCA water in the Containment sumps combine to maintain the correct pH level.

D.Correct.

005 Residual Heat Removal / 4

005A1.05; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Detection of and response to presence of water in RHR emergency sump

(CFR: 41.5 / 45.5)

Importance Rating:	3.3 3.3
Technical Reference:	EOP-ES-1.3, Attachment 1, Pg 34, Rev. 2
References to be provided:	None
Learning Objective:	EOP-LP-2.3/3.3, Obj. 5.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 032/NEW/C/A//FR-C.2, BKGRD, CSFST/NONE/EARLY/006 K3.02/

Given the following plant conditions:
- A Small Break LOCA occurred
- 6.9 KV Bus 1B-SB is de-energized
- The 'A' CSIP has tripped
- Core Exit Thermocouple (CET) temperatures are 735°F
- RCS pressure is 985 psig
- Containment pressure is 15 psig
- RVLIS Full Range level is 46%

Which ONE of the following complete the statement below?

Based on the conditions above, the Reactor Vessel water level is __(1)___ the Top Of Active Fuel AND the Core Cooling Critical Safety Function Status Tree is __(2)___ .

A. (1) above

(2) Orange

B. (1) above

(2) Red

C✓ (1) below

(2) Orange

D. (1) below

(2) Red

Plausibility and Answer Analysis

Reason answer is correct: With no high head safety injection available, the cooling of the core becomes degraded. Degraded core cooling reduces RCS inventory below the top of active fuel when RVLIS Full Range indication is less than 63%. CSF-2 monitors for the symptoms of inadequate or degraded core cooling. With no RCPs in service the CSF-2 will be orange for these conditions because the CET temperature is greater than 730°F and RVLIS level is above 39%.

A. Incorrect. The first part is plausible since the EOP-FR-C.2 will monitor RCS void fraction vice the actual RCS water level with the RCPs in service. The RVLIS Dynamic Head Range level of 60% with 3 RCPs, 33% with 2 RCPs and 25% with 1 RCP in service requires no additional actions from the Functional Restoration procedure and the crew returns the procedure and step in effect and therefore the candidate may have a misconception that RCS water level is above the top of active fuel. This is incorrect however because all RCPs are secured when Containment pressure rises above 10 pounds (Phase B) and CCW to Containment is isolated. The second part is correct.

B. Incorrect. The first part is plausible since the EOP-FR-C.2 will monitor RCS void fraction vice the actual RCS water level with the RCPs in service. The RVLIS Dynamic Head Range level of 60% with 3 RCPs, 33% with 2 RCPs and 25% with 1 RCP in service requires no additional actions from the Functional Restoration procedure and the crew returns the procedure and step in effect and therefore the candidate may have a misconception that RCS water level is above the top of active fuel. This is incorrect however because all RCPs are secured when Containment pressure rises above 10 pounds (Phase B) and CCW to Containment is isolated. The second part is plausible since the CET temperature is above 730°F and RVLIS is below the dynamic range value of 60% for all RCPs, however this is incorrect because the RCPs were previously stopped and therefore RVLIS must be less than 39% in order to meet the Red conditions for CSF -2.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since the CET temperature is above 730°F and RVLIS is below the dynamic range value of 60% for all RCPs, however this is incorrect because the RCPs were previously stopped and therefore RVLIS must be less than 39% in order to meet the Red conditions for CSF -2.

006 Emergency Core Cooling 2/3

006K3.02; Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: Fuel

(CFR: 41.7 / 45.6)

Importance Rating:	4.3 4.4
Technical Reference:	ERG-BKGRD-FR-C.2, Introduction, Pg 1, Rev. 2 EOP-CSFST, CSF-2, Pg 2, Rev. 13
References to be provided:	None
Learning Objective:	EOP-LP-3.10, Obj. 1
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 033/MODIFIED/C/A//ALB-009/NONE//007 A1.03/

Given the following plant conditions:
- The unit is operating at 50% power
- The crew is responding to a leaking PRZ PORV

Time	TI-471.1	TI-463
0800	110°F	115°F
0805	115°F	155°F
0810	125°F	195°F
0815	145°F	235°F
0820	155°F	255°F

Which ONE of the following is the first time that annunciator ALB-009-8-1, PRT High-Low Level Press or Temp, will alarm?

Temperature Indicator Noun Name:
TI-471.1, PRZ Relief Tank Temperature
TI-463, PRZ PORV Line Temperature

A. 0805

B.✓ 0810

C. 0815

D. 0820

Reason answer is correct: Annunciator ALB-009-8-1 has multiple inputs that causes the annunciator to alarm. One of which is high temperature which has a setpoint of 120°F as sensed by TI-471.1, PRZ Relief Tank Temperature. At 0810 the PRT temperature is above the temperature at which the annunciator will go into alarm.

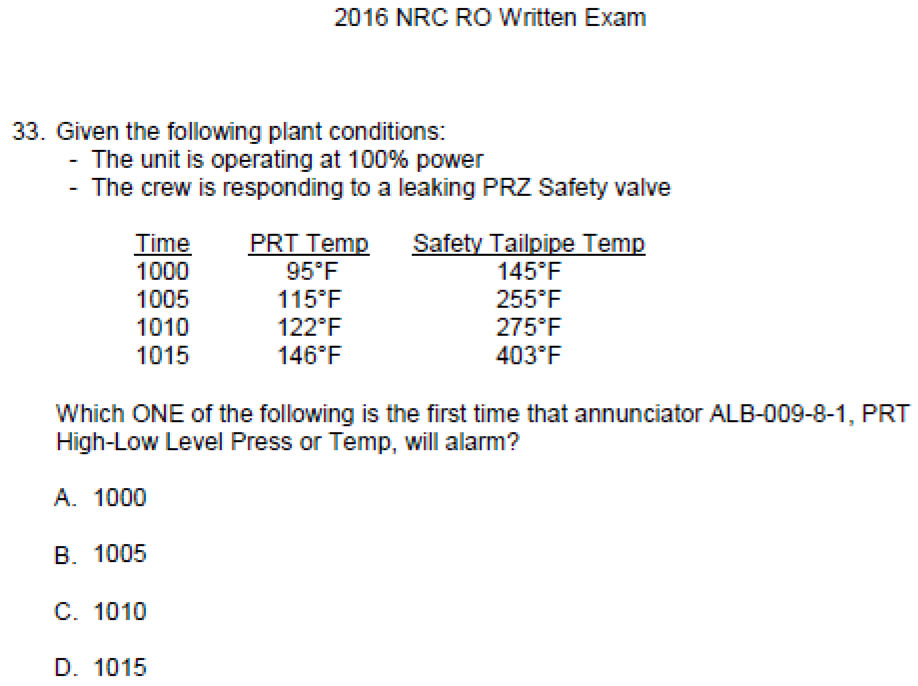
A. Incorrect. Plausible since at this PRT input temperature annunciator ALB-009-8-2, Pressurizer Relief Discharge High Temp goes into alarm. The alarm comes on when the PRZ PORV discharge line temperature exceeds 140°F as sensed by TI-463.

B. Correct.

C. Incorrect. Plausible since the candidate may mis-apply the setpoint of 140°F for the PRZ PORV discharge line high temperature annunciator ALB-009-8-2, Pressurizer Relief Discharge High Temp to TI-471.1, PRZ Relief Tank Temperature, however this is incorrect as PRZ PORV discharge line temperature is sensed by TI-463.

D. Incorrect. Plausible since at this temperature annunciator ALB-009-8-3, Pressurizer Safety Relief Discharge High Temp goes into alarm. The alarm comes on when the PRZ Safety valve discharge line temperature exceeds 250°F , however this is incorrect as it is sensed by TI-465, TI-467, or TI-469.

Original question:



2016 NRC RO Written Exam

33. Given the following plant conditions:
    - The unit is operating at 100% power
    - The crew is responding to a leaking PRZ Safety valve

Time	PRT Temp	Safety Tailpipe Temp
1000	95°F	145°F
1005	115°F	255°F
1010	122°F	275°F
1015	146°F	403°F

Which ONE of the following is the first time that annunciator ALB-009-8-1, PRT High-Low Level Press or Temp, will alarm?

A. 1000

B. 1005

C. 1010

D. 1015

007 Pressurizer Relief/Quench Tank / 5

007A1.03; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Monitoring quench tank temperature

(CFR: 41.5 / 45.5)

Importance Rating:	2.6 2.7
Technical Reference:	APP-ALB-009-8-1, Pg 29, Rev. 18
References to be provided:	None
Learning Objective:	PRZ Lesson Plan, Obj. 5.d
Question Origin:	Modified - 2016 NRC RO 33
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 034/NEW/FUNDAMENTAL//OP-100, 5-S-1301/NONE//007 A4.01/

Given the following plant conditions:

- The unit is operating at 100%
- ALB-009-8-1, Pressurizer Relief Tank High-Low Level Press or Temp, has just alarmed
- PRT level is at the low level alarm setpoint

Which ONE of the following completes the statements concerning PRT fill?

1RC-167, RMW to PRT shutoff valve ___(1)___ to restore normal PRT level.

1RC-167 ___(2)___ receive a signal to an automatic shut if a Phase A signal occurs.

A. (1) will open

(2) will

B. (1) will open

(2) will not

C. (1) must be manually opened

(2) will

D✓ (1) must be manually opened

(2) will not

Plausibility and Answer Analysis

Reason answer is correct: 1RC-167, RMW to PRT shutoff valve is an AOV that is operated by a switch on the Main Control Board. In accordance with OP-100, Reactor Coolant System Section 8.0 Pressuirzer Relief Tank Make Up, after verifying that 1RC-161, RMW TO PRT containment isolation valve is OPEN the operator then OPENS 1RC-167 until the desired level is reached then shuts 1RC-167.

1RC-167 does NOT receive a Phase A isolation signal. PRT make up Containment isolation valve 1RC-161 receives a shut signal on a Phase A.

A. Incorrect. The first part is plausible since other RCS systems auto-makeup when low level is reached (such as auto fill make up to the RCP stand pipes). The second part is plausible since this is the inside Containment valve to make up to the PRT but since there is a check valve between the PRT and 1RC-167 there is no Phase A isolation signal supplied to shut 1RC-167.

B. Incorrect. The first part is plausible since other RCS systems auto-makeup when low level is reached (such as auto fill make up to the RCP stand pipes). The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since this is the inside Containment valve to make up to the PRT but since there is a check valve between the PRT and 1RC-167 there is no Phase A isolation signal supplied to shut 1RC-167.

D. Correct.

007 Pressurizer Relief/Quench Tank / 5

007A4.01; Ability to manually operate and/or monitor in the control room: PRT spray supply valve

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	2.7 2.7
Technical Reference:	OP-100, Section 8.0, Pg 17, Rev. 44 Simplified Drawing 2165-S-1301 Sheet 2
References to be provided:	None
Learning Objective:	PRZ Lesson Plan, Obj. 5.a
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 035/BANK/FUNDAMENTAL//OP-145/NONE//008 K4.07/

Which ONE of the following completes the statements below?

The 'C' CCW pump can be powered from either ___(1)___ .

A ___(2)___ is a design feature that prevents two CCW pumps from being aligned to the same power supply.

A. (1) 6.9 KV Emergency Bus 1A-SA or 6.9 KV Emergency Bus 1B-SB

(2) manual transfer switch

B✓ (1) 6.9 KV Emergency Bus 1A-SA or 6.9 KV Emergency Bus 1B-SB

(2) key-operated interlock

C. (1) 480V Emergency Bus 1A2-SA or 480V Emergency Bus 1B2-SB

(2) manual transfer switch

D. (1) 480V Emergency Bus 1A2-SA or 480V Emergency Bus 1B2-SB

(2) key-operated interlock

Plausibility and Answer Analysis

Reason answer is correct: The 'C' CCW pump can be powered from either the 6.9kV Bus 1A-SA or 1B-SB and a key interlock prevents racking in if 'A' or 'B' CCW pump is racked in on the same bus.

A. Incorrect. Plausible since the power source is correct. The second part is plausible since the 'C' CSIP has a manual transfer switch which is allows for rapid pump swaps per OP-107 if required.

B. Correct.

C. Incorrect. Plausible since the RHR, Containment Spray Pump and Chiller P-4 which are all safety related equipment, are powered from these buses. The second part is plausible since the 'C' CSIP has a manual transfer switch which is allows for rapid pump swaps per OP-107 if required.

D. Incorrect. Plausible since the RHR, Containment Spray Pump and Chiller P-4 which are all safety related equipment, are powered from these buses. The key interlock prevents racking the breaker in if either 'A' or 'B' CCW pump is racked in on the bus.

008 Component Cooling Water / 8

008K4.07; Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Operation of the CCW swing-bus power supply and its associated breakers and controls

(CFR: 41.7)

Importance Rating:	2.6 2.7
Technical Reference:	OP-145, P&L #10, Pg 8, Rev. 62
References to be provided:	None
Learning Objective:	CCW Lesson Plan, Obj. 2.e
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 036/NEW/C/A//AOP-019/NONE//010 K5.02/

Given the following plant conditions:
- The unit is operating at 100% power
- The PRZ pressure master controller, PK-444A, is in AUTOMATIC
- The PRZ pressure master controller setpoint rapidly fails to 61%

Which ONE of the following completes the statements below?

Both spray valves will __(1)__ .

The fluid enthalpy across the spray valves __(2)__ .

A. (1) open

(2) lowers

B.✓ (1) open

(2) remains constant

C. (1) close

(2) lowers

D. (1) close

(2) remains constant

Plausibility and Answer Analysis

Reason answer is correct: The master PRZ controller is set to 66.88% during normal operation. With the controller in Automatic reducing the setpoint to 61% will raise the output signal from 25 % to 54.4 %. An output signal above 40.62% will send a signal for PRZ Spray valve to begin to throttle open until the output signal in 71.87% at which the valves are full open. The fluid in the PRZ spray line is subcooled water and the enthalpy remains constant as the fluid is throttled through the PRZ Spray valves.

A. Incorrect. The first part is correct. The second part is plausible since the PRZ system undergoes a phase change in order to providing the quenching of the PRZ Vapor space and reduce PRZ pressure, however this is incorrect because the PRZ Spray fluid is subcooled RCS fluid from the A and B loops and the constant enthalpy expansion through the valve results in no enthalpy change.

B. Correct.

C. Incorrect. The first part is plausible since the setpoint change is approximaitely 5% if the candidate misapplies the setpoint thumbrule as a 1 for 1 change the 5% change will raise the output to 30% which is less than 40.62% and therefore the PRZ spray valve would remain shut, however this is incorrect because the change should raise the output signal from 25% to 54.4% resulting in the PRZ Spray valves opening. The second part is plausible since the PRZ system undergoes a phase change in order to providing the quenching of the PRZ Vapor space and reduce PRZ pressure, however this is incorrect because the PRZ Spray fluid is subcooled RCS fluid from the A and B loops and the constant enthalpy expansion through the valve results in no enthalpy change.

D. Incorrect. The first part is plausible since the setpoint change is approximaitely 5% if the candidate misapplies the setpoint thumbrule as a 1 for 1 change the 5% change will raise the output to 30% which is less than 40.62% and therefore the PRZ spray valve would remain shut, however this is incorrect because the change should raise the output signal from 25% to 54.4% resulting in the PRZ Spray valves opening. The second part is correct.

010 Pressurizer Pressure Control / 3

010K5.02; Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: Constant enthalpy expansion through a valve

(CFR: 41.5 / 45.7)

Importance Rating:	2.6 3.0
Technical Reference:	AOP-019, Attachment 2, Pg 20, Rev. 25
References to be provided:	None
Learning Objective:	PZRPC Lesson Plan, Obj. 4.b
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 037/BANK/C/A//E-0/NONE//012 G2.4.1/

Given the following plant conditions:
- The unit is operating at 90% power
- Power Range NI-41 is under clearance and OWP-RP-23 is complete

Subsequently the following occur:
- Multiple system annuciators are received
- Power Range NI-43 fails high

Which ONE of the following completes the statement below?

The FIRST operator action required to be performed is to _______ immediate actions.

A. place Rod Control in manual in accordance with, AOP-001, Rod Control Malfunctions

B. place Rod Control in manual in accordance with, AOP-024, Loss Of Uninterruptible Power Supply

[✓ C.] verify that the Reactor is tripped in accordance with EOP-E-0, Reactor Trip Or Safety Injection

D. manually trip the Reactor using either MCB switches in accordance with EOP-FR-S.1, Response To Nuclear Power Generation/ATWS

Plausibility and Answer Analysis

Reason answer is correct: With OWP-RP-23 in place the bistable for NI-41 is in the tripped condition. NI-43 failing high will satisfy the 2/4 logic for the high flux Reactor trip and the Reactor should trip. EOP-E-0 immediate actions require the operator to verify a Reactor trip occurs.

A. Incorrect. Plausible since the NI-43 has failed high and Rod Control is auctioneered high the system will generate a continuous withdrawal signal and placing the rods in manual is immediate action number two of AOP-001, however this is incorrect because a reactor trip signal has been generated and entry into EOP-E-0 is required.

B. Incorrect. Plausible since multiple systems are in alarm and a failure of NI-43 are indicative of the failure of UPS power supply S-III and placing the rods in manual is immediate action number one of AOP-024, however this is incorrect because a reactor trip signal has been generated and entry into EOP-E-0 is required.

C. Correct.

D. Incorrect. Plausible since the first immdiate action of EOP-FR-S.1 is to verify the Reactor is tripped and manually insert negative reactivity if the reactor is not tripped automatically or manually, however this is incorrect because the actions directed by EOP-FR-S.1 are only required if the action to trip the Reactor in EOP-E-0 are not successful.

012 Reactor Protection / 7

012G2.4.1; Knowledge of EOP entry conditions and immediate action steps.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	4.6 4.8
Technical Reference:	EOP-E-0, Step 1 RNO, Pg 4, Rev. 7
References to be provided:	None
Learning Objective:	RPS Lesson Plan, Obj. 10
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 038/BANK/C/A//AOP-024-BD, ALB-015/NONE//012 K3.01/

Given the following plant conditions:
- The unit is operating at 100% power
- SSPS Train A, General Warning Light is in due to maintenance, but no other bistables have been affected

Subsequently a loss of Instrument Bus S-IV occurs

Which ONE of the following describes the resultant condition of the Control Rod Drive Stationary Gripper Coils?

A. De-energized based on the loss of one instrument bus ONLY.

B✓ De-energized based on both trains having a General Warning condition.

C. Energized due to the General Warning condition blocking the Reactor trip signal.

D. Enegerized due to redundant Instrument Bus power supplies.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-024 if all of the following conditions exist, a reactor trip or safeguards actuation may occur

- Power lost to any Instrument Bus
- Any ESFAS or RPS instrument loops out of service

Because the A SSPS train is in Test a General Warning condition is present 1 of 2 conditions for the General Warning RPS Trip logic to open the Reactor Trip breakers is satisfied. Once power is lost to Instrument Bus IV the B SSPS train will have lost AC power to the output relay cabinet resulting in a General Warning condition. Now the RPS logic for the 2 of 2 General Warning conditions are present and the logic to open the Reactor Trip breakers are satisfied which will remove power to the CRDM Stationary Gripper coils allowing control rods to be inserted into the Reactor.

A. Incorrect. Plausible since the loss of Instrument Bus IV results in the loss of the channel IV RPS inputs which will generate a signal to open the Reactor Trip breakers and de-energize the stationary coils , however this is incorrect because the loss of only an instrument bus does not satisfy the 2/4 logic to open the Reactor Trip breakers.

B. Correct.

C. Incorrect. Plausible since the RPS is designed to have the output signal of certain parameters blocked (i.e. SRNI and IRNI signals), however this is incorrect because the General Warning condition does not have the capability to be blocked.

D. Incorrect. Plausible since the single failure design criteria of RPS allow for the loss of one power supply from causing or preventing an actuation of the system, however this is incorrect because testing is also in progress and under these conditions will generate a signal to open the Reactor Trip breakers.

012 Reactor Protection / 7

012K3.01; Knowledge of the effect that a loss or malfunction of the RPS will have on the following:CRDS

(CFR: 41.7 / 45.6)

Importance Rating:	3.9 4.0
Technical Reference:	AOP-024-BD, Section 1.0, Step 15, Pg 4, Rev. 21 APP-ALB-015, Window 1-3, Pg 5, Rev. 30
References to be provided:	None
Learning Objective:	RPS Lesson Plan, Obj. 11.a
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 039/PREVIOUS/FUNDAMENTAL//AOP-024-BD/NONE/2016 NRC RO 39/013 K2.01/

Which ONE of the following completes the statement below?

Instrument Buses ___(1)___ AND ___(2)___ provide power to the ESFAS Slave Relays.

A. (1) SI

(2) SII

B. (1) SII

(2) SIII

[✓ C.] (1) SI

(2) SIV

D. (1) SIII

(2) SIV

Plausibility and Answer Analysis

Reason answer is correct: *Train ESFAS slave relays are powered from Instrument Bus SI (SIV). A loss of SI or SIV will result in a loss of ESFAS functions driven by slave relays for that train.*

A. Incorrect *Plausible since the ESFAS relays are powered from the safety instrument buses. If SI or SIII is lost, the MCB controller for MDAFW pump flow control valves will not be operable; flow control valves will not shut on an AFW isolation signal and will not open on an auto open signal. Train ESFAS slave relays are powered from Instrument Bus SI (SIV). A loss of SI or SIV will result in a loss of ESFAS functions driven by slave relays for that train. A loss of SI will cause a loss of 'A' Train ONLY the question is asking for BOTH 'A' and 'B' Train.*

B.Incorrect *Plausible since the ESFAS relays are powered from the safety instrument buses. If power is lost to Instrument Bus SII (B Train and TDAFW) or SIII (A Train) the associated AFW pump suction pressure instrument will read low. If the AFW pump is running, it will not trip on Lo-Lo suction pressure nor will it be prevented from being started. Additionally, if power is lost to Instrument Bus SII (B Train) or SIII (A Train), the associated CNMT Spray Additive Tank level indicators will read empty but their associated CNMT Spray Chemical Addition Valve will not automatically shut. If necessary, the valve(s) may be manually operated.*

C.Correct

D. Incorrect *Plausible since the ESFAS slave relays are powered from Instrumtment Bus SI (SIV). To answer this question it would take BOTH SI and SIV and only one of the two (SIV) are listed. If power is lost to Instrument Bus SII (B Train and TDAFW) or SIII (A Train) the associated AFW pump suction pressure instrument will read low. If the AFW pump is running, it will not trip on Lo-Lo suction pressure nor will it be prevented from being started. Train ESFAS slave relays are powered from Instrument Bus SI (SIV). A loss of SI or SIV will result in a loss of ESFAS functions driven by slave relays for that train.*

013 Engineered Safety Features Actuation / 2

013K2.01; Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

(CFR: 41.7)

Importance Rating:	3.6 3.8
Technical Reference:	AOP-024-BD, Section 1.0, Pg 2, Rev. 20
References to be provided:	None
Learning Objective:	ESFAS Lesson Plan, Obj. 2
Question Origin:	Previous 2016 NRC RO 39 radomly selected
Comments:	None
Tier/Group:	T2G1

2018 NRC RO 040/BANK/C/A//OWP-ESF/NONE//013 K5.02/

Given the following plant conditions:

- The unit is operating at 100% power
- Containment Pressure Channel I pressure indication was oscillating and has been removed from service in accordance with OWP-ESF, Engineered Safety Feature Actuation.

Which ONE of the following identifies the logic from the remaining channels for HI-1 and HI-3 Containment Pressure actuations AFTER Channel I is REMOVED from service?

	HI-1 SI Actuation	HI-3 CNMT Spray Actuation
A.	1 / 2	2 / 3
B.	1 / 2	1 / 3
C✓	2 / 3	2 / 3
D.	2 / 3	1 / 3

Plausibility and Answer Analysis

Reason answer is correct: HI-1 uses only CNMT pressure channels II, III & IV and is normally a 2/3 logic (at 3 psig in the Containment) so the logic remains unchanged when channel I is tripped. For HI-3, all four channel of CNMT pressure are used with a normal 2/4 logic at 10 psig in the Containment. Unlike all other RPS/ESFAS bistables, when a HI-3 bistable is removed from service, the channel is bypassed.

A. Incorrect. The first part is plausible if the candidate thinks that the HI-1 actuation uses all 4 channels of CNMT pressure. The second part is correct.

B. Incorrect. The first part is plausible if the candidate thinks the HI-1 actuation uses all 4 channels of CNMT pressure. The second part is plausible if the candidate thinks the HI-3 actuation is tripped when it is removed from service.

C. Correct.

D. Incorrect. First part is correct The second part is plausible if the candidate thinks the HI-3 actuation is tripped when it is removed from service

013 Engineered Safety Features Actuation / 2

013K5.02; Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Safety system logic and reliability

(CFR: 41.5 / 45.7)

Importance Rating:	2.9 3.3
Technical Reference:	OWP-ESF-01, Pg 4 and 8, Rev. 21
References to be provided:	None
Learning Objective:	ESFAS Lesson Plan, Obj. 10.b
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 041/BANK/C/A//USERS GUIDE, E-0/NONE/2009B NRC RO 41/022 A3.01/

Given the following plant conditions:

- The unit experiences a Reactor Trip and SI concurrent with a Loss Of Offsite Power
- During the performance of EOP-E-0, Reactor Trip Or Safety Injection, Attachment 3, the BOP notes the following alignment for Containment Fan Coolers:
  - 'A' Train - one fan per unit running in fast speed
  - 'B' Train - one fan per unit running in slow speed

Which ONE of the following identifes the actions required by the BOP?

A.✓ Shift the two 'A' Train fans from fast to slow speed.

B. Shift the two 'B' Train fans from slow to fast speed.

C. Start two additional 'A' Train fans in fast speed and secure the 'B' Train fans.

D. Start two additional 'B' Train fans in slow speed and secure the 'A' Train fans.

Plausibility and Answer Analysis

Reason answer is correct: During an SI all fans operating in high (i.e. fast) speed trip and one fan in each Fan Cooler will receive an automatic LOW speed start signal through the Sequencer at load block 2. If this signal fails in accordance with EOP-E-0, Attachment 3 step 16, the BOP should secure the fast speed fans and shift them to slow speed so that one fan per unit is running in slow speed.

A. Correct.

B. Incorrect. Plausible since this alignment would be used following a loss of offsite power, however the SI alignment requires that the fans operate in low speed.

C. Incorrect. Plausible because the normal fan alignment is to have all fans in one train running, however the SI alignment requires that the fans operate in low speed.

D. Incorrect. Plausible because the normal fan alignment is to have all fans in a train running, however the SI alignment requires that the fans operate in low speed.

022 Containment Cooling / 5

022A3.01; Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation

(CFR: 41.7 / 45.5)

Importance Rating:	4.1 4.3
Technical Reference:	EOP-User's Guide, Section 6.7, Pg 37, Rev. 49 EOP-E-0, Attachment 3, Step 16, Pg 61, Rev. 7
References to be provided:	None
Learning Objective:	CCS Lesson Plan, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 042/BANK/FUNDAMENTAL//OP-112/NONE//026 A4.05/

Given the following plant conditions:

- An automatic Containment Spray Actuation Signal (CSAS) has occurred.
- Containment Pressure has lowered to <3 psig.

Which ONE of the following is correct in order to reset the CSAS?

A. Containment Spray pump control switches A-SA AND B-SB must be positioned to STOP prior to resetting.

B. Either Containment Spray Train A OR B reset switch must be taken to RESET to completely reset the CSAS.

C.✓ Both Containment Spray Trains A AND B reset switches must be taken to RESET to completely reset the CSAS.

D. Containment Spray pump discharge valves 1CT-50 AND 1CT-88 must be positioned to SHUT prior to resetting.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with OP-112, Containment Spray System section 7.0, to place system for standby operation following Manual or Automatic initiation at the MCB the Containment Spray Train A and Train B reset switch is taken to RESET. Then the Spray pumps are secured and the valves realigned to complete the standy alignment.

A. Incorrect. Plausible since stopping both CNMT spray pumps is the step to be performed AFTER the reset switches are taken to reset.

B. Incorrect. Plausible since Phase A requires only one switch to be operated for the manaul actuation of the ESF signals. Additionally a student may have a misconception about OP-112 step 7.1.2 since there is only one step but the step is repeated to reset Train A and then Train B.

C. Correct.

D. Incorrect. Plausible since shutting the CNMT spray pump discharge valves are required to be performed in the next steps of the OP-112 to align the system for standby operation.

026 Containment Spray / 5

026A4.05; Ability to manually operate and/or monitor in the control room: Containment spray reset switches

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	3.5 3.5
Technical Reference:	OP-112, Section 7.0, Pg 11, Rev. 45
References to be provided:	None
Learning Objective:	CSS Lesson Plan, Obj. 4.b
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 043/PREVIOUS/FUNDAMENTAL//STUDENT TEXT/NONE/2016 NRC RO 44/039 A3.02/

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

Plausibility and Answer Analysis

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 to modulates open if header pressure is > 5 psig and therefore the valve would be shut on a turbine trip, however the ventilating valve 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 to modulates open if header pressure is > 5 psig and therefore the valve would be shut on a turbine trip, however the ventilating valve 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, however this is incorrect.

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:	MT Student text MSR Student text
References to be provided:	None
Learning Objective:	MT Lesson Plan, Obj. 9 MSR Lesson Plan, Obj. 4.e
Question Origin:	Previous 2016 NRC RO 44 radomly selected
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 044/NEW/C/A//AOP-042/NONE//039 K4.06/

Given the following plant conditions:

- The unit is operating at 97% power to perform OST-1080, Auxiliary Feedwater Pump 1X-SAB Full Flow Test Quarterly Interval Mode 1, 3
- The TDAFW Pump is running

Subsequently:

- The RAB AO reports there is a steam line leak in a steam supply line to the TDAFW pump
- Reactor power rises to 100.1% and stablilizes

Which ONE of the following completes the statements below?

The AOP that should be entered FIRST to address this event is __(1)__.

Reverse steam flow is prevented on the steam supply lines going to the TDAFW pump through the use of __(2)__ valve(s).

Procedure Title:
AOP-038, Rapid Downpower
AOP-042, Secondary Steam Leak / Efficiency Loss

A. (1) AOP-038

(2) check

B. (1) AOP-038

(2) a trip

[✓ C] (1) AOP-042

(2) check

D. (1) AOP-042

(2) a trip

Plausibility and Answer Analysis

Reason answer is correct: Entry conditions have been met for AOP-042 with a notification to the MCR of a secondary steam leak. Addtionally, AOP-042 will have the operators attempt to isolate the leak (could be done by shutting 1MS-70 or 1MS-72 or both). The Main Steam Lines supplying motive steam to the TDAFW pumps have check valves installed downstream of the individual steam line isolation valves (1MS-70 from "B" SG and 1MS-72 from "C" SG).

A. Incorrect. The first part is plausible since AOP-038 is used to rapidly lower Reactor power (one of the entry condtions is any condition requiring > 5 MW/min load reductions). With Reactor power just above 100% (currently at 100.01%) it is > 100% but would not require a reduction of load > 5 MW/min to reduce power to < 100%. Additionally, AOP-042 is written to address the steam leak where AOP-038 is not. The second part is correct.

B. Incorrect. The first part is plausible since AOP-038 is used to rapidly lower Reactor power (one of the entry condtions is any condition requiring > 5 MW/min load reductions). With Reactor power just above 100% (currently at 100.01%) it is > 100% but would not require a reduction of load > 5 MW/min to reduce power to < 100%. Additionally, AOP-042 is written to address the steam leak where AOP-038 is not. The second part is plausible since the Trip and Throttle valve automatically shuts to prevent the Turbine Driven AFW pump from overspeeding by isolating steam flow from the Main Steam supply lines.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since the Trip and Throttle valve automatically shuts to prevent the Turbine Driven AFW pump from overspeeding by isolating steam flow from the Main Steam supply lines.

039 Main and Reheat Steam / 4

039K4.06; Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Prevent reverse steam flow on steam line break

(CFR: 41.7)

Importance Rating:	3.3 3.6
Technical Reference:	AOP-042, Section 2.0, Pg 3, Rev. 6 Simplified Drawing 2165-S-0542, Rev 27 (Red circles show check valves installed in the MS lines, Blue circle shows where lines join and are not seperate lines to the TDAFW pump)
References to be provided:	None
Learning Objective:	MSSS Lesson Plan, Obj. 5 AOP-LP-3.42, Obj. 1
Question Origin:	New
Comments:	Discuss the use of Turbine Driven AFW Pump Main Steam supply lines as a possible topic to meet K/A as check valves in the TDAFW Pump steam supply lines prevent back flow during a steam break, but the MS system has no such design features. Phonecon 6/13: Dan agrees that using the Turbine Driven AFW Pump Main Steam supply lines is acceptable to meet this K/A.
Tier/Group:	T2/G1

2018 NRC RO 045/NEW/FUNDAMENTAL//AOP-010/NONE//059 A2.12/

Given the following plant conditions:

- The unit is operating at 7% power
- Main Feedwater Regulating Bypass valves are shut and all Main Feedwater Regulating valves are in automatic operation
- 'B' Main Feedwater Regulating Valve is oscillating causing 'B' SG level to fluctuate from 54% to 58%

Which ONE of the following completes the statements below concerning this failure?

In accordance with AOP-010, Feedwater Malfunctions, the operator should __(1)__ .

An Automatic Reactor trip would occur if 'B' SG level reaches a setpoint of __(2)__ .

A. (1) place 'B' Feed Reg valve in manual

(2) 30%

B.✓ (1) place 'B' Feed Reg valve in manual

(2) 25%

C. (1) initiate AFW flow to maintain SG level

(2) 30%

D. (1) initiate AFW flow to maintain SG level

(2) 25%

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-010 step 1 RNO actions the operator should place the affected Feedwater Reg valve in Manual. In accordance with EOP-E-0 an automatic Reactor Trip will occur from SG Low-Low Water Level of 25%.

A. Incorrect. The first part is correct. The second part is plausible since AOP-010 and OMM-001, Attachment 13 trip limits for low SG level is 30%

B. Correct.

C. Incorrect. The first part is plausible since this would be the action to take IF feed flow was not maintained to all three steam generators. (AOP-010, Step 8 RNO 8.b.1) however this is incorrect because only the B SG level is effected. The second part is plausible since AOP-010 and OMM-001, Attachment 13 trip limits for low SG level is 30%

D. Incorrect. The first part is plausible since this would be the action to take IF feed flow was not maintained to all three steam generators. (AOP-010, Step 8 RNO 8.b.1) however this is incorrect because only the B SG level is effected. The second part is correct.

059 Main Feedwater / 4

059A2.12; Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of feedwater regulating valves

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.1 3.4
Technical Reference:	EOP-E-0, Attachment 10, Pg 81, Rev. 7 AOP-010, Step 1 RNO, Pg 4, Rev. 39
References to be provided:	None
Learning Objective:	AOP-LP-3.10, Obj. 2
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 046/PREVIOUS/C/A//ALB-014, AOP-010/NONE/2014 NRC RO 48/061 K3.02/

Given the following plant conditions:

- A unit heat up is in progress in accordance with GP-002, Normal Plant Heatup From Cold Solid To Hot Subcritical Mode 5 To Mode 3
- 'A' MDAFW pump is feeding the SGs

Subsequently the following annunciator alarms:

- ALB-017-5-4, Aux Feedwater Pump 'A' Trip Or Close CKT Trouble

Which ONE of the following completes the statements below?

SG levels will lower to ___(1)___ where the 'B' MDAFW pump will automatically start, to restore SG levels.

Entry in to AOP-010, Feedwater Malfunctions, ___(2)___ required.

(Assume NO Operator actions)

A. (1) 20%

(2) is

B. (1) 20%

(2) is NOT

C. (1) 25%

(2) is

D✓ (1) 25%

(2) is NOT

Plausibility and Answer Analysis

Reason answer is correct:
With any SG level less than 25% (2 out of 3 NR detectors), the following occurs:
(1) Reactor trip
(2) Both MDAFW Pumps start
(3) All three MDAFW FCVs receive an auto open signal

AOP-010 Entry Conditions are any Main Feedwater or Condensate System malfunction causing a flow transient and may also be entered as directed by other approved procedures.

A. Incorrect. The first part is plausible since the AMSAC system generates a start signal to the AFW pumps when it is actuated at 20% SG level, however the AMSAC system is not is service until power is above 35%. The second part is plausible since AOP-010 is the abnormal procedure used to address SG level problems, however the procedure is designed to address issues caused by the loss of Main feedwater or Condensate.

B. Incorrect. The first part is plausible since the AMSAC system generates a start signal to the AFW pumps when it is actuated at 20% SG level, however the AMSAC system is not is service until power is above 35%. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since AOP-010 is the abnormal procedure used to to address SG level problems, however the procedure is designed to address issues caused by the loss of Main feedwater or Condensate.

D. Correct.

061 Auxiliary/Emergency Feedwater / 4

061K3.02; Knowledge of the effect that a loss or malfunction of the AFW will have on the following: S/G

(CFR: 41.7 / 45.6)

Importance Rating:	4.2 4.4
Technical Reference:	APP-ALB-014, Window 5-4B, Pg 32, Re.v 26 AOP-010, Section 2.0, Pg 3, Rev. 39
References to be provided:	None
Learning Objective:	AFW Lesson Plan, Obj. 7.a AOP-LP-3.10, Obj. 1
Question Origin:	Previous 2016 NRC RO 48 radomly selected
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 047/NEW/C/A//OP-155/OP-155, ATTACHMENT 9//062 A1.01/

Given the following plant conditions:
- OST-1073, 1B-SB Emergency Diesel Generator Operability Test Monthly Interval Modes 1-2-3-4-5-6, is in progress
- EDG 1B-SB indications are as follows:
  - Megawatts 6.4 MW
  - Megavars 3.4 MVar

Subsequently after an electrical transient, EDG 1B-SB indications are as follows:
- Megawatts 7.2 MW
- Megavars 3.8 MVar

Which ONE of the following competes the statement below?

The Emergency Diesel ___(1)___ Overload limit is exceeded and the operator must reduce load ___(2)___ .

(Reference Provided)

A✓ (1) Engine

(2) immediately

B. (1) Engine

(2) within 2 hours

C. (1) Generator

(2) immediately

D. (1) Generator

(2) within 2 hours

Plausibility and Answer Analysis

Reason answer is correct: In accordance with OP-155 P&L 28, the continuous and overload ratings are not to be exceeded. At 7.2 Megawatts and 3.8 Megavars the Engine 2 Hour Overload Limit of 7150 KW has been exceeded. Based on this condition the operator should immediately take action to reduce the diesel load to within the requirements of Attachment 9.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since operation at the Generator and Engine overload limit is allowed for 2 hours within a 24 hour period the candidate may misinterpet the capacity curve and determine these limits maybe exceeded for 2 hours in all cases, however this is incorrect because the Engine overload limit of 7150 KW is exceeded and operation above this limit is not allowed.

C. Incorrect. The first part is plausible since the Generator continuous load limit is being exceeded for these conditions, however this is incorrect because only the Engine overload limit is being exceeded for these conditions and operation above this limit is not allowed. The second part is correct.

D. Incorrect. The first part is plausible since the Generator continuous load limit is being exceeded for these conditions, however this is incorrect because only the Engine overload limit is being exceeded for these conditions and operation above this limit is not allowed. The second part is plausible since operation at the Generator and Engine overload limit is allowed for 2 hours within a 24 hour period the candidate may misinterpet the capacity curve and determine these limits maybe exceeded for 2 hours in all cases, however this is incorrect because the Engine overload limit of 7150 KW is exceeded and operation above this limit is not allowed.

062 AC Electrical Distribution / 6

062A1 .01; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significance of D/G load limits

(CFR: 41.5 / 45.5)

Importance Rating:	3.4 3.8
Technical Reference:	OP-155, Precaution and Limitation 28, Pg 10, Rev. 86 OP-155, Attachment 9, Pg 192, Rev. 86
References to be provided:	None
Learning Objective:	Diesel Lesson Plan, Obj. 6
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 048/NEW/C/A//AOP-025/NONE//062 G2.4.45/

Given the following plant conditions:

- The unit was operating at 100% power with 'A' Safety Train Equipment in service

Subsequently:

- A fault occurred on Unit Aux Transformer 'A'
- The bus transfer for 6.9KV Aux Bus 'A' and 'D' failed to occur
- The 'A' EDG failed to start
- The crew is performing EOP-ES-0.1, Reactor Trip Response, and AOP-025, Loss of One Emergency AC Bus (6.9KV) or One Emergency DC Bus (125V)

It has been two minutes since the unit has tripped and the following annunciators are locked in:

- ALB-005-8-2, CCW Pump B Disch Header Low Press
- ALB-023-1-18, Chiller WC2-A Trouble

Which ONE of the following identifies THE PRIORITY that the condition which caused these annunciators will be addressed in accordance with AOP-025 AND whether the annunciator is EXPECTED or NOT EXPECTED for the plant conditions?

A. 1. CCW Pump B Disch Header Low Press — Expected
   2. Chiller WC2-A Trouble — NOT Expected

B✓ 1. CCW Pump B Disch Header Low Press — NOT Expected
   2. Chiller WC2-A Trouble — Expected

C. 1. Chiller WC2-A Trouble — Expected
   2. CCW Pump B Disch Header Low Press — NOT Expected

D. 1. Chiller WC2-A Trouble — NOT Expected
   2. CCW Pump B Disch Header Low Press — Expected

Plausibility and Answer Analysis

Reason answer is correct: Priority: Priorities of AOP-025, Check for power, check for availability of cooling water to an EDG that has started on low bus voltage, then check CCW cooling to plant equipment, then check a Chiller running to provide cooling to ventilation in safety related areas. Expectation of annunciators: The CCW Pump 'B' Discharge Header Low Pressure is NOT expected because the 'B' CCW pump should have started on low pressure. The standby pump auto-starts on low discharge pressure of 52 psig sensed on its respective discharge header (PT 649 or 650). It is expected that the Chiller WC2-A Trouble annunciator will be locked in since the 1A-SA Emergency Bus is de-energized by the openng of breaker 105 and the failure of the EDG 1A-SA failure to start. Therefore the annuciator should be locked in. AOP-025 addresses the CCW Pump before addressing the Chiller.

A. Incorrect. 1. The priority of addressing the annunciators is correct.
2. Expectation of annunciators: The CCW Pump low pressure annunciator is plausible if there is a misconception that the 'B' CCW pump must be manually started (like a CSIP) since the 'A' CCW pump is without power system pressure will lower to < 52 psig at which time the 'B' CCW pump will auto start and the low pressure annunciator should clear when system pressure is restored. Annunciators: Chiller Trouble annunciator not expected - plausible since the 480 volt control power is provided by 2 phases of the AC from the power supply the breakers remain closed when the high side supply breaker is open. Because the breaker status is unchanged the the input for a breaker trip does not occur and the trouble alarm is not recieved. This is incorrect because the Chiller WC2-A is a 6.9KV breaker with DC control power which will allow the breaker to trip open on the loss of power resulting in a trouble alarm.

B. Correct.

C. Incorrect. 1. The priority of addressing the Chiller Trouble annunciator prior to the CCW low flow is plausible since this both issues are addressed in AOP-025 but they are listed in the opposite order of what is in the procedure.
2. Expectation of annunciators is correct.

D. Incorrect.
1. The priority of addressing the Chiller Trouble annunciator prior to the CCW low flow is plausible since this both issues are addressed in AOP-025 but they are listed in the opposite order of what is in the procedure.
2. Expectation of annunciators: Chiller Trouble annunciator not expected - plausible since the 480 volt control power is provided by 2 phases of the AC from the power supply the breakers remain closed when the high side supply breaker is open. Because the breaker status is unchanged the the input for a breaker trip does not occur and the trouble alarm is not recieved. This is incorrect because the Chiller WC2-A is a 6.9KV breaker with DC control power which will allow the breaker to trip open on the loss of power resulting in a trouble alarm. CCW annunciator expected - plausible since the 'A' CCW pump would have lost power and the annunciator would have initially have alarmed but should be clear after 2 minutes have elapsed from the event initiation. The 'B' CCW pump auto started on low system pressure (52 psig). After the 'B' CCW pump started it would have cleared the low pressure. The pump start and pressure restoration would have been completed prior to the 2 minutes that have elapsed.

062 AC Electrical Distribution / 6

062G2.4.45; 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: AOP-025, Section 3.1, Step 5 and 10, Pg 8 and 9, Rev. 42

References to be provided: None

Learning Objective: AOP-LP-3.25, Obj. 5.a

Question Origin: New

Comments: None

Tier/Group: T2/G1

2018 NRC RO 049/BANK/FUNDAMENTAL//6-G-0042 SH 01/NONE/2011 NRC RO 50/063 K1.02/

Which ONE of the following describes the normal power source for a safety-related 125-V DC bus?

480-V MCC through a ______ .

A.✓ battery charger to the DC bus

B. 7.5-KVA inverter to the DC bus

C. battery charger, through the DC battery, and then to the DC bus

D. 7.5-KVA inverter, through the DC battery, and then to the DC bus

Plausibility and Answer Analysis

Reason answer is correct: 125VDC Bus is supplied directly by battery chargers

A. Correct.

B. Incorrect. Plausible because the 7.5-KVA inverter has a DC power supply and is safety related equipment.

C. Incorrect. Plausible because the battery charger supplies the DC battery

D. Incorrect. Plausible because the 7.5-KVA inverter has a DC power supply and is safety related equipment.

063 DC Electrical Distribution / 6

063K1.02; Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: AC electrical system

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating:	2.7 3.2
Technical Reference:	Control Wire Drawing 2166-G-042, Sheet 1, 250V DC, 125V DC, & 120V UPS One Line Diagram
References to be provided:	None
Learning Objective:	DCP Lesson Plan, Obj. 2.b
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 050/BANK/FUNDAMENTAL//OP-155, AOP-025-BD/NONE//064 K1 .04/

Given the following plant conditions:
- The unit is operating at 100% power
- A loss of DC Bus 1A-SA occurs

Which ONE of the following completes the statements below regarding operation of the 'A' EDG?

The Governor and Generator Excitation circuits will be __(1)__ .

The EDG Output breaker __(2)__ be closed from the MCB.

A. (1) de-energized

(2) can

B.✓ (1) de-energized

(2) can NOT

C. (1) energized

(2) can

D. (1) energized

(2) can NOT

Plausibility and Answer Analysis

Reason answer is correct: Part 1: OP-155, Diesel Generator Emergency Power System, has a section to restore the Governor after DC power is lost. Additionally, notes are in the procedure for loss of DC power to the EDG. The unavailability of 125 VDC will prevent the generator to flash and electronic speed control is NOT available if there is a loss of 125 VDC control power. AOP-025 Basis document section 3.3 step 2 also provides indication that the governor will lose power with a loss of DC. Part 2: The EDG Output breaker is a 6.9 KV breaker that is remotely operated via 125 VDC power. The effects of losing 125 VDC to a 6.9 KV breaker are as follows:
No power to closing coil (can't shut breaker remotely)
No power to trip coil (can't open breaker remotely)
No power to charge the closing spring (depending on the status of the breaker before the loss of 125 VDC, may only get 1 close and 1 open cycle out of breaker before manually changing closing spring

A. Incorrect. Plausible since the Governor and Excitation circuits will de-energize, however to operate the EDG Output breaker from the MCB requires DC Control Power.

B. Correct

C. Incorrect. Plausible if the candidate has a misconception that the Governor, Excitation, and Control power circuits are supplied with power from the AC Electrical Distribution System similar to the Main Generator.

D. Incorrect. Plausible if the candidate has a misconception that the Governor and Excitation circuits are supplied by the AC Electrical Distribution System similar to the Main Generator.

064 Emergency Diesel Generator / 6

064K1 .04; Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: DC distribution system

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating: 3.6 3.9

Technical Reference: OP-155, Procedure Note, Pg 91, Rev. 86
AOP-025-BD, Section 2.0, Pg 53, Rev. 19

References to be provided: None

Learning Objective: AOP-LP-3.25, Obj. 5

Question Origin: Bank

Comments: None

Tier/Group: T2/G1

2018 NRC RO 051/BANK/FUNDAMENTAL//AOP-005-BD/NONE/2004 NRC RO 40/073 K4.01/

Given the following plant conditions:
- The unit is operating at 40% power
- REM-*1WC-3544, WPB CCW Hx Inlet Monitor, is in HIGH alarm

As a result of the high alarm, which ONE of the following will automatically close?

A. 3WC-321, Waste Evap B Condenser CCW FCV-329 (3WC-320) Outlet Isolation

B✓ 3WC-4, WPB CCW Surge Tank Overflow

C. 3WC-197, WG Compressor A CCW Outlet

D. 3WC-7, WPB CCW Surge Tank Drain

Plausibility and Answer Analysis

Reason answer is correct: A high rad alarm on this monitor will cause the surge tank overflow to isolate to prevent the release of radioactivity.

A. Incorrect. Plausible since this valve does automatically close if RCS leakage to the CCW thermal barrier HX is occurring, but closes on high flow.

B. Correct.

C. Incorrect. Plausible since the valve is normally throttled open and if shut would stop any flow from the system to a potential leak path but the valve does not have an auto close signal it must be manually shut.

D. Incorrect. Plausible since this valve does have an automatic action associated with the surge tank, but the valve closes on a low level.

073 Process Radiation Monitoring / 7

073K4.01; Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint

(CFR: 41.7)

Importance Rating:	4.0 4.3
Technical Reference:	AOP-005-BD, Section 1.0, Pg 3, Rev. 12
References to be provided:	None
Learning Objective:	AOP-LP-3.05, Obj. 4
Question Origin:	Bank
Comments:	HNP was not able to create a valid question for process radiation monitors for K/A 073K5.02, Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Radiation intensity changes with source distance. Requested new K/A. Phonecon 6/13: HNP previously suppressed this K/A dealing with HNP Process Radiation Monitors associated with a source distance relationship with liquid or gaseous monitors, so selected a new K/A, keeping 073 and randomly selecting from the remaining items for this K/A: New K/A 073K4.01: Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint
Tier/Group:	T2/G1

2018 NRC RO 052/BANK/C/A//AOP-022-BD/NONE//076 K2.04/

Given the following plant conditions:
- The unit is operating at 100% power
- NSW Pump 'A' is operating
- NSW Pump 'B' is in standby
- Breaker 104, AUX BUS D TO EMERGENCY BUS A-SA, trips

After one minute, which ONE of the following describes the expected Service Water system alignment?

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

No ESW pumps are running.

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

No ESW pumps are running.

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

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

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

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

Plausibility and Answer Analysis

Reason answer is correct: Power will lost to the 1A-SA Emergency AC Bus until supplied by the 'A' EDG. ESW Pump 'A' will start on Program A (Undervoltage - Load Block 3) and supply the 'A' ESW header. NSW Pump 'A' is powered from Auxiliary Bus 1D and will remain running.

A. Incorrect. Plausible since the candidate may not recognize that the sequencer started the ESW Pump 'A' in Load Block 3.

B. Incorrect. Plausible since the candidate may have a misconception that NSW Pump 'A' lost power and that NSW Pump 'B' auto started. Also, ESW Pump 'A' was started by the sequencer in Load Block 3.

C. Correct.

D. Incorrect. Plausible since the candidate may have a misconception that NSW Pump 'A' lost power and that NSW Pump 'B' auto started. Also, the second part of the distractor is correct in that ESW Pump 'A' will be supplying the 'A' ESW header.

076 Service Water / 4

076K2.04; Knowledge of bus power supplies to the following: Reactor building closed cooling water

(CFR: 41.7)

Importance Rating:	2.5 2.6
Technical Reference:	AOP-022-BD, Section 1.0, Pg 2, Rev. 14 HNP Electrical Load List, Pg 3, Rev. 0
References to be provided:	None
Learning Objective:	SWS Lesson Plan, Obj. 2.a
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 053/NEW/FUNDAMENTAL//AOP-017, AOP-017-BD/NONE//078 A4.01/

Given the following plant conditions:

- The unit is operating at 100% power
- Annunciator ALB-002-8-1, Instrument Air Low Press, alarms
- The OATC checks MCB pressure indication PI-9751.1, Instrument Air Header Pressure, and identifies that air pressure is slowly lowering

Which ONE of the following completes the statements below?

The MCB pressure indicator PI-9751.1 __(1)__ always indicative of pressure throughout the Instrument Air system.

At __(2)__ psig RCS letdown flowpath valves will begin to fail to mid-position.

A. (1) is

(2) 75

B. (1) is

(2) 85

C. (1) is not

(2) 75

D✓ (1) is not

(2) 85

Plausibility and Answer Analysis

Reason answer is correct: In the case of the Instrument Air system the system is piped to multiple locations in various buildings. The indication in the MCR is the pressure near and downstream of the IA compressors. This pressure may not be indicative of the pressure throughout the entire system. AOP-017 basis document informs the operator that in a loss of IA event the location of the pressure indicator may not indicate the pressure in all parts of the system and it may be lower than indicated on the MCB. AOP-017 Attachment 7 states at 85 psig RCS letdown flowpath valves begin to fail to mid-position.

A. Incorrect. The first part is plausible since MCB indications are what is used to accurately determine and monitor system parameters. The second part is plausible since 75 psig is less than the pressure at which the letdown orifice isolation valve will begin to shut, however this is incorrect because most air valves in the direct letdown path require 85 psig for full stroke, and may begin to fail to mid-position as pressure falls below that value.

B. Incorrect. The first part is plausible since MCB indications are what is used to accurately determine and monitor system parameters. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since 75 psig is less than the pressure at which the letdown orifice isolation valve will begin to shut, however this is incorrect because most air valves in the direct letdown path require 85 psig for full stroke, and may begin to fail to mid-position as pressure falls below that value.

D. Correct.

078 Instrument Air / 8

078A4.01; Ability to manually operate and/or monitor in the control room: Pressure gauges

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	3.1 3.1
Technical Reference:	AOP-017, Section 3.0, Step 1 Note, Pg 4, Rev. 40 AOP-017, Attachment 7, Pg 57, Rev. 40 AOP-017-BD, Section 2.0, Pg 7, Rev. 15 AOP-017-BD, Section 3.0, Pg 14, Rev. 15
References to be provided:	None
Learning Objective:	ISA Lesson Plan, Obj. 6 and 9
Question Origin:	New
Comments:	None
Tier/Group:	T2/G1

2018 NRC RO 054/NEW/C/A//EOP-E-1/NONE//103 A2.03/

Given the following plant conditions:
- The unit is operating at 100% power
- A LOCA in Containment results in peak pressure to rise to 7 psig

Which ONE of the following completes the statements below?

1SI-287, Accumulator and PRZ PORV N2 Supply Isolation Valve, will___(1)___ .

EOP-E-1, Loss of Reactor Or Secondary Coolant, requires operating the ___(2)___ .

A. (1) automatically shut

(2) BOTH Phase A and Phase B reset switches

B.✓ (1) automatically shut

(2) Phase A reset switch ONLY

C. (1) remain open

(2) BOTH Phase A and Phase B reset switches

D. (1) remain open

(2) Phase A reset switch ONLY

Plausibility and Answer Analysis

Reason answer is correct: A Phase 'A' (T) signal, Containment Phase A isolation, shuts the following valves associated with the SI Accumulators:
1SI-179, Accumulator fill line
1SI-287, Accumulator nitrogen supply line
1SI-263, Accumlator test line isolation
1SI-264, Accumlator test line to RWST
and 1SP-78, 1SP-81, 1SP-84 and 1SP-85, Accumulator sample valves
In accordance with EOP-E-1 prior to opening 1SI-287 Phase A and Phase B isolation signals if actuated will be reset (step 17). Containment pressure peaked above 10 psig which would have caused a Containment Isolation Phase A signal at 3 psig and a Phase B signal at 10 psig. ONLY the Phase A signal will need to be reset to restore control of 1SI-287.

A. Incorrect. The first part is correct. The second part is plausible since EOP-E-1 has the operator reset BOTH the Phase A and Phase B signals <u>if actuated</u> but ONLY the Phase A signal reset is required to restore control of Phase A valves.

B. Correct.

C. Incorrect. The first part is plausible since the Cold Leg Accumulator discharge valves all receive OPEN signals when a Safety Injection signal is generated. Since Containment pressure exceeded 3 psig a SI signal would have occurred. The second part is plausible since EOP-E-1 has the operator reset BOTH the Phase A and Phase B signals <u>if actuated</u> but ONLY the Phase A signal reset is required to restore control of Phase A valves.

D. Incorrect. The first part is plausible since the Cold Leg Accumulator discharge valves all receive OPEN signals when a Safety Injection signal is generated. Since Containment pressure exceeded 3 psig a SI signal would have occurred. The second part is correct.

103 Containment / 5

103A2.03; Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating: 3.5 3.8

Technical Reference: OMM-004, Attachmen4, Pg 48, Rev. 41
EOP-E-1, Step 17 and 18, Pg 18 and 20, Rev. 4

References to be provided: None

Learning Objective: SIS Lesson Plan, Obj. 7.b

Question Origin: New

Comments: None

Tier/Group: T2/G1

2018 NRC RO 055/NEW/C/A//TECH SPEC 3.6.1.3/NONE//103 A2.05/

The Containment Personnel Airlock inner door is stuck shut and can not be opened.

In accordance with OP-113, Penetration Pressurization System And Containment Air Locks, access to the Containment requires ___(1)___ in order to gain access to Containment through the EMERGENCY Air Lock doors.

The action required in accordance with Technical Specification 3.6.1.3, Containment Systems - Containment Air Locks, for the failure of the Containment PERSONNEL Airlock is to verify the outer door is closed ___(2)___ .

A. (1) operating a hydraulic control station

(2) immediately

B. (1) turning a manual handwheel

(2) immediately

C. (1) operating a hydraulic control station

(2) within 1 hour

D✓ (1) turning a manual handwheel

(2) within 1 hour

Plausibility and Answer Analysis

Reason answer is correct:

A. Incorrect. Part 1 is plausible since the Personnel Airlock is operated with a hydraulic control station that automatically opens the door. Part 2 is plausible since immediate action (to evaluate overall Containment leakage rate) is required if one or more Containment air locks are inoperable for reasons other than Tech Spec 3.6.1.4.a or 3.6.1.4.b

B. Incorrect. Part 1 is correct. Part 2 is plausible since immediate action (to evaluate overall Containment leakage rate) is required if one or more Containment air locks are inoperable for reasons other than Tech Spec 3.6.1.4.a or 3.6.1.4.b

C. Incorrect. Part 1 is plausible since the Personnel Airlock is operated with a hydraulic control station that automatically opens the door. Part 2 is correct.

D. Correct.

103 Containment / 5

103A2.05; Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency containment entry

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	2.9 3.9
Technical Reference:	OP-113, Section 8.3, Pg 23, Rev. 23 Technical Specification 3.6.1.3
References to be provided:	None
Learning Objective:	Containment Lesson Plan, Obj. 4.c and 9.d
Question Origin:	New
Comments:	K/A match by meeting the higher order part of the K/A
Tier/Group:	T2/G1

2018 NRC RO 056/NEW/C/A//AOP-018-BD/NONE//002 K6.07/

Given the following plant conditions:
- The unit is operating at 40% power
- PRZ Backup Heaters are energized

Subsequently:
- The 'B' RCP trips due to an electrical fault

Which ONE of the following completes the statements below?

As 'B' RCP coasts down, Loops 'A' and 'C' steam flow will rise by _______ .

(Assume that Reactor power and Tavg have remained constant)

A. 16.5%

B. 25%

C. 33.3%

D✓ 50%

Plausibility and Answer Analysis

Reason answer is correct: In accordance with the AOP-018, Reactor Coolant Pump Abnormal Conditions, if an RCP is stopped at power, Reactor power and T_{avg} should remain constant. The steam flow from the RCS loop with the stopped RCP will be insignificant, so the steam flow in the other two RCS loops will increase by 50%. These changes in steam flow and reactor coolant flow will cause the ΔT in the unaffected RCS loops to increase by 50% for a constant Reactor power.

A. Incorrect. Plausiblie since each RCP provides 1/3 of the total flow during normal operation and the flow is now split between the two remaining RCPs.

B. Incorrect. Plausible since the candidate may have the misconception that half of the 50% rise in flow is shared between the two RCPs.

C. Incorrect. Plausible since each RCP provides 1/3 of the total flow during normal operation, however this is incorrect because the remaining loops share half of the load from the 1/3 flow that was being provided by the loss of the RCP.

D. Correct.

002 Reactor Coolant / 2/4

002K6.07; Knowledge of the effect or a loss or malfunction on the following RCS components: Pumps

(CFR: 41.7 / 45.7)

Importance Rating:	2.5 2.8
Technical Reference:	AOP-018-BD, Section 1.0, Step 7, Pg 2, Rev 25
References to be provided:	None
Learning Objective:	AOP-LP-3.18, Obj. 4
Question Origin:	New
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 057/NEW/FUNDAMENTAL//AOP-024/NONE//015 K2.01/

Which ONE of the following identifies the power supply to NI-36, Intermediate Range Nuclear Instrument?

A. SI

B.✓ SII

C. SIII

D. SIV

Plausibility and Answer Analysis

Reason answer is correct: The power supply to NI-36, Intermediate Range Nuclear Instrument, is Instrument Bus (IDP) 1B-SII.

A. Incorrect. Plausible since Instrument buses provide power to all nuclear instruments and are typically labelled based on the channel they are supplied from i.e. Channel I is supplied from IDP 1A-SI, however this convention cannot be applied to NI-36 as HNP only has 4 Instrument buses.

B. Correct.

C. Incorrect. Plausible since Instrument buses provide power to all nuclear instruments and are typically labelled based on the channel they are supplied from i.e. Channel III is supplied from IDP 1A-SIII, however this convention cannot be applied to NI-36 as HNP only has 4 Instrument buses.

D. Incorrect. Plausible since Instrument buses provide power to all nuclear instruments and are typically labelled based on the channel they are supplied from i.e. Channel IV is supplied from IDP 1B-SIV, however this convention cannot be applied to NI-36 as HNP only has 4 Instrument buses.

015 Nuclear Instrumentation / 7

015K2.01; Knowledge of bus power supplies to the following: NIS channels, components, and interconnections

(CFR: 41.7)

Importance Rating:	3.3 3.7
Technical Reference:	AOP-024, Attachment 2, Sheet 1, Pg 29, Rev. 57
References to be provided:	None
Learning Objective:	NIS Lesson Plan, Obj 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 058/BANK/FUNDAMENTAL//DBD-301/NONE//016 K5.01/

Which ONE of the following describes the process instrumentation interface with Solid-State Protection System (SSPS) for separation of indication/control circuits from protection circuits?

A. Separate detectors are used to generate indication/control and protection signals.

B✓ Isolation amplifiers are used to separate indication/control from protection signals.

C. Separate instrument channels are used for indication/control and protection signals.

D. Redundant lag filters are used to separate indication/control from protection signals.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with the Reactor Control and Protection system design basis document where redundant protective channels are combined to provide non-protective functions, the required signals are derived through isolation amplifiers. These devices are designed so that open or short circuit conditions, as well as application of 120 vac or 140 vdc, to the isolated side of the circuit will have no effect on the input or protection side of the circuit.

A. Incorrect. Plausible since systems such as the Safeguards Sequencer system use separate relays to detect the loss of power to each safety bus in order to determine which program the sequencer should run for the accident conditions, however this is incorrect because isolation amplifiers are used in the SSPS system to separate the control and protectino signals.

B. Correct.

C. Incorrect. Plausible since systems such as the PRZ Pressure control system have separate channels for control and protection, however this is incorrect because isolation amplifiers are used in the SSPS system to separate the control and protectino signals.

D. Incorrect. Plausible since systems such as the Rod control system use redundant half wave rectifiers to filter the command signals from the Slave Cycler to individual rod coils in order to move a control rod in the Reactor, however this is incorrect because isolation amplifiers are used in the SSPS system to separate the control and protectino signals.

016 Non-Nuclear Instrumentation / 7

016K5.01; Knowledge of the operational implication of the following concepts as they apply to the NNIS: Separation of control and protection circuits

(CFR: 41.5 / 45.7)

Importance Rating:	2.7 2.8
Technical Reference:	DBD-301, Step 4.6, Pg 35, Rev. 7
References to be provided:	None
Learning Objective:	PSPR-LP-3.5, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 059/BANK/C/A//FSAR OP-170 AOP-005/NONE/2012 NRC RO 62/033 A1.02/

Given the following conditions:

- The unit is operating at 100% power
- The suction pipe from Spent Fuel Pool 'B' to the Spent Fuel Pool Cooling Pump completely severs
- Spent Fuel Pool area radiation monitor RM-*1FR-3566A-SA is in HIGH alarm and monitor RM-*1FR-3567B-SB is in ALERT

Which ONE of the following completes the statements below?

Level in the Spent Fuel Pool will stabilize at a MAXIMUM of __(1)__ above the fuel assemblies.

These Radiation monitor conditions will cause __(2)__ train(s) of Fuel Handling Ventilation Emergency Exhaust to automatically start.

[✓ A.] (1) 18'

(2) 'A' ONLY

B. (1) 18'

(2) both 'A' and 'B'

C. (1) 23'

(2) 'A' ONLY

D. (1) 23'

(2) both 'A' and 'B'

Plausibility and Answer Analysis

Reason answer is correct: The location and design of the suction pipe prevents the Spent Fuel Pool from draining less than 18 feet above the top of the spent fuel assemblies which is 5 feet below the normal level of 23' above the fuel. Any one of a train's ARMs in HIGH alarm in the FHB will start 1 train of FHB emergency exhaust. Since only the 'A' Train is in HIGH alarm the 'A' train will auto start.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible because in some actuations it requires a 2 / 2 logic but for Fuel Handling Ventilation it only requires a 1 / 2 (one Rad monitor) in HIGH alarm from either train to stop both Normal Supply fan A and B

C. Incorrect. The first part is plausible because 23 feet is the correct level the Fuel Pool is required to be maintained above the Fuel by Tech Specs and the A train Exhaust fan will auto start . The second part is correct.

D. Incorrect. The first part is plausible because 23 feet is the correct level the Fuel Pool is required to be maintained above the Fuel by Tech Specs. The second part is plausible because in some actuations it requires a 2 / 2 logic but for Fuel Handling Ventilation it only requires a 1 / 2 (one Rad monitor) in HIGH alarm from either train to stop both Normal Supply fan A and B

033 Spent Fuel Pool Cooling / 8

033A1.02; Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including: Radiation monitoring systems

(CFR: 41.5 / 45.5)

Importance Rating:	2.8 3.3
Technical Reference:	FSAR Chapter 9, Section 9.1.3.2, Pg 9.1.3-3, Amend. 56 OP-170, Section 8.1, Pg 23, Rev. 37 AOP-005-BD, Section1.0, Pg 3, Rev. 8
References to be provided:	None
Learning Objective:	FPC Lesson Plan, Obj. 7.a
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 060/BANK/FUNDAMENTAL//AOP-013/NONE//034 A4.01/

Given the following plant conditions:

- The unit is in Mode 6 with fuel movements in progress
- A fuel assembly has been slightly damaged during removal from the core
- Radiation levels are rising steadily and are currently as follows:

REM-01LT-3502A-SA, CNMT RCS Leak Detection Monitor, is in HIGH ALARM

Which ONE of the following completes the statements below?

Normal Containment Purge ___(1)___ automatically isolate.

Containment Pre-Entry Purge ___(2)___ automatically isolate.

A. (1) will

(2) will

B.✓ (1) will

(2) will NOT

C. (1) will NOT

(2) will

D. (1) will NOT

(2) will NOT

Plausibility and Answer Analysis

Reason answer is correct: AOP-013, Fuel Handling Accident section 3.2, Fuel Handling Accident In Containment step 7 has the operator check REM-01LT-3502A-SA, Cnmt RCS Leak Detection Monitor, to verifiy that the HIGH alarm clear. If the high alarm is NOT clear the operator verifies that Normal Containment Purge is isolated.

A. Incorrect. Plausible since the misconception could be that when REM-01LT-3502A-SA, Cnmt RCS Leak Detection Monitor, is in HIGH ALARM it effects both Normal and Pre-Entry Purge.

B. Correct.

C. Incorrect. Plausible if the candidate has a misconception about the auto actions associated with the given rad monitor in HIGH alarm will cause. REM-01LT-3502A-SA, Cnmt RCS Leak Detection Monitor, is in HIGH ALARM will ONLY cause the Normal Containment Purge to isolate. Since there isn't a name "Normal or Pre-Entry Purge" associated with this rad monitors name this misconception could occur. There is a separate Rad monitor associated with Pre-Entry Purge, REM-01LT-3502B, Cnmt Pre-Entry Purge Monitor. If this monitor goes into HIGH ALARM then the Pre-Entry Purge would isolate but NOT the Normal Purge.

D. Incorrect. Plausible if there is a misconception that it would require a 2 of 2 conicidence (both REM-01LT-3502A and 3502B) to cause the Normal or Containment Pre-Entry Purge to isolate and at this time ONLY one Rad monitor is in high alarm.

034 Fuel-Handling Equipment / 8

034A4.01; Ability to manually operate and/or monitor in the control room: Radiation levels

(CFR: 41.7 / 45.5 to 45.8)

Importance Rating:	3.3 3.7
Technical Reference:	AOP-013, Section 3.2, Step 7, Pg 12 and 13, Rev. 16
References to be provided:	None
Learning Objective:	AOP-LP-3.13, Obj. 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 061/BANK/C/A//STEAM TABLES/STEAM TABLES//035 A3.02/

What should RCS temperature stabilize at if a Reactor Trip occurred with a Main Steam Line Isolation and all three SG PORV in automatic with their controller setpoints set at 89%?

A. 557°F

B. 558°F

C. 561°F

D✓ 564°F

Plausibility and Answer Analysis

Reason answer is correct: The SG PORV's controller has a range of 0-1300 psig. With the setting of 89% the PSAT (RCS Temperature) is calculated to be 564°F

0.89 x 1300 = 1157 + 15 = 1172 psia PSAT for 1172 is approximately 564°F

A. Incorrect. Plausible since this is the normal temp after Rx trip with Stm Dumps available but a MS Line Isolation has closed the MSIV's so Steam Dumps are not available.

B. Incorrect. Plausible since this is the normal RCS temp with the SG PORV setpoint at 85% which is what the normal at power PORV setpoint is adjusted to.

C. Incorrect. Plausible since this is the temperature you would calculate if you subracted 15 psi instead of added it.

D. Correct.

035 Steam Generator / 4

035A3.02; Ability to monitor automatic operation of the S/G including: MAD valves

(CFR: 41.7 / 45.5)

Importance Rating:	3.7 3.6
Technical Reference:	GP-005, Section 6.0, Step 5.e, Pg 20 and 56, Rev. 102 Steam Tables
References to be provided:	None
Learning Objective:	MSSS Lesson Plan, Obj. 5.a
Question Origin:	Bank
Comments:	Phonecon 6/9: HNP was not familiar with the term 'MAD' Valve and requested clarification of the term. Per discussion with Dan Bacon 'MAD' was determined to be an acronym for Manual/Automatic Depressurization and a question should be written to address the automatic operation of SG PORVs:
Tier/Group:	T2/G2

2018 NRC RO 062/NEW/C/A//ALB-511/NONE//071 G2.1.30/

Given the following plant conditions:

- The Gaseous Waste Processing System is receiving purge flow from the VCT gas space
- ALB-511-2-2, OARC-1119 Product Gas HI-HI Oxygen O_2 Shutdown, is received
- 3WG-400, Recombiner B TCV-1114B, is shut
- 3WG-401, O_2 Addition Valve HACV-1118B, is shut
- 1CS-137, Volume Control Tank Purge Valve, is shut

Which ONE of the following completes the statement below?

The WG Catalytic H_2 Recombiner B has stopped due to a __(1)__ Trip.

To restore purge flow to the VCT gas space 3WG-401, O_2 Addition Valve HACV-1118B, must be reset from the __(2)__ .

A. (1) Fast

(2) Radwaste Control Room

B.✓ (1) Fast

(2) Recombiner B Control Panel

C. (1) Slow

(2) Radwaste Control Room

D. (1) Slow

(2) Recombiner B Control Panel

Plausibility and Answer Analysis

Reason answer is correct: Nitrogen is aligned to the VCT to purge oxygen from the VCT until the oxygen level in the VCT is below 4%. Once the oxygen level is below 4% hydrogen is aligned to replace the nitrogen in the VCT gas space to support normal online operation.

A. Incorrect. The first part is correct. The second part is plausible since the Radwaste Control Room has switch indications for 1CS-137, along with keylock switches to shut effluent relief isolation valves, however this is incorrect because 3WG-401, O_2 *Addition Valve HACV-1118B is controlled locally at the Recombiner Control Panel.*

B. Correct.

C. Incorrect. The first part is plausible since 3WG-401, O_2 *Addition Valve HACV-1118B, will shut when a Slow Trip of the recombiner occurs, however this is incorrect because 1CS-137, VCT Purge Isolation valve will remain open. The second part is plausible since the Radwaste Control Room has switch indications for 1CS-137, along with keylock switches to shut effluent relief isolation valves, however this is incorrect because 3WG-401,* O_2 *Addition Valve HACV-1118B is controlled locally at the Recombiner Control Panel.*

D. Incorrect. The first part is plausible since 3WG-401, O_2 *Addition Valve HACV-1118B, will shut when a Slow Trip of the recombiner occurs, however this is incorrect because 1CS-137, VCT Purge Isolation valve will remain open. The second part is correct.*

071 Waste Gas Disposal / 9

071G2.1.30; Ability to locate and operate components, including local controls.

(CFR: 41.7 / 45.7)

Importance Rating:	4.4 4.0
Technical Reference:	APP-ALB-511, Window 2-2, Pg 19 and 21, Rev. 21
References to be provided:	None
Learning Objective:	GWPS Lesson Plan, Obj 9
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 063/BANK/C/A//AOP-005-BD/NONE//072 K4.01/

Given the following plant conditions:

- The unit is operating at 100% power
- Pressurizer level and RCS pressure are lowering
- Containment temperature and pressure are rising

Containment Radiation Monitors read as follows:

- RM-01CR-3561ASA - high (RED) alarm
- RM-01CR-3561BSB - under clearance
- RM-01CR-3561CSA - does not respond to changing conditions
- RM-01CR-3561DSB - high (RED) alarm

Which ONE of the following completes the statements below?

Containment Ventilation Isolation Train A __(1)__ automatically actuate.

Containment Ventilation Isolation Train B __(1)__ automatically actuate.

A✓ (1) will

(2) will

B. (1) will

(2) will NOT

C. (1) will NOT

(2) will

D. (1) will NOT

(2) will NOT

Plausibility and Answer Analysis

Reason answer is correct: CVI occurs on 2/4 rad monitors in high alarm and generates both a Train A and a Train B CVI signal.

A. Correct.

B. Incorrect. Plausible since a Train B rad monitor is in high alarm, but both trains will receive a CVI signal.

C. Incorrect. Plausible since a Train A rad monitor is in high alarm, but both trains will receive a CVI signal.

D. Incorrect. Plausible since only a single rad monitor on each train is in alarm, but both trains receive a CVI signal when any 2/4 rad monitors alarm.

072 Area Radiation Monitoring / 7

072K4.01; Knowledge of ARM system design feature(s) and/or interlock(s) which provide for the following: Containment ventilation isolation

(CFR: 41.7)

Importance Rating:	3.3 3.6
Technical Reference:	AOP-005-BD, Section 1.0, Pg 2, Rev. 12
References to be provided:	None
Learning Objective:	AOP-LP-3.5, Obj. 1
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 064/BANK/FUNDAMENTAL//AOP-017, ALB-002/NONE/2009A NRC RO 64/079 K1.01/

Given the following plant conditions:
- A rupture in the Instrument Air system has occurred
- Instrument Air header pressure is 85 psig and lowering slowly

Which ONE of the following describes the status of 1SA-506, Instrument Air from Service Air Isolation Valve, and the status of ALB-002-8-1, Instrument Air Low Pressure Alarm?

	1SA-506	Alarm status
A.	OPEN	Lit
B.	OPEN	NOT Lit
C.	CLOSED	Lit
D✓	CLOSED	NOT Lit

Plausibility and Answer Analysis

Reason answer is correct: *1SA-506 shuts at 90 psig. The alarm however will not come in until 75 psig*

A Incorrect. 1SA-506 closes at 90 psig. Plausible if candidate confuses the setpoint for autoclosure with one of the other IA setpoints (101, 96, 95, 90, 85, 75, 60, and 35 are all significant Instrument Air Pressure Setpoints. See Attachment 7 of AOP-017). Alarm will NOT be lit. Plausible if candidate believes an alarm will alert operators to the condition prior to the automatic action occurring.

B Incorrect. 1SA-506 will shut automatically at 90 psig. Plausible if candidate confuses the setpoint for autoclosure with one of the other IA setpoints (101, 96, 95, 90, 85, 75, 60, and 35 are all significant Instrument Air Pressure Setpoints. See Attachment 7 of AOP-017)

C Incorrect. 1SA-506 will close automatically to isolate the Service Air System from the Instrument Air system, however the alarm is not lit until 75 psig. Plausible if candidate believes an alarm will alert operators to the condition prior to the automatic action occurring.

D. Correct.

079 Station Air / 8

079K1.01; Knowledge of the physical connections and/or cause-effect relationships between the SAS and the following systems: IAS

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

Importance Rating:	3.0 3.1
Technical Reference:	AOP-017, Attachment 7, Pg 57, Rev. 40 APP-ALB-002, Window 8-4, Pg 43, Rev. 53
References to be provided:	None
Learning Objective:	AOP-LP-3.17, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 065/BANK/FUNDAMENTAL//AOP-036.02/NONE//086 A2.04/

Given the following plant conditions:

- A fire occurs in the Electrical Penetration Area A on RAB 261' elevation
- Automatic fire suppression failed to actuate resulting in damage to the following:
  - The Electrical Penetration Area A Ionization detector
  - MCB level indicators LI-9010A1 SA, CST Level

Subsequently AOP-036, Safe Shutdown Following a Fire, is being implemented

Which ONE of the following completes the statements below?

In accordance with FPP-012-02-RAB261, RAB Elevation 261 Fire Pre-Plan, the alternate method of detecting a fire in Electrical Penetration Area A is a ___(1)___ detector.

In accordance with AOP-036.02, Fire Area: 1-A-BAL-A, 1-A-BAL-G, 1-A-BAL-H, CST level is determined by monitoring the AFW Pump ___(2)___ pressure.

A. (1) thermal

(2) discharge

B✓ (1) thermal

(2) suction

C. (1) ultraviolet

(2) discharge

D. (1) ultraviolet

(2) suction

Plausibility and Answer Analysis

Reason answer is correct: The RAB 261 elevation is monitored by Ionization and Thermal detector in accordance FPP-012-02-RAB261. Identifying the Thermal detectors as the alternate to the Ionization detection system predicts the impact the fire and the resultant damage it has caused. In accordance with AOP-036.02, the CST level is checked greater than 10% using LI-CE-9010A1SA, LI-CE-9010B1SB or Attachment 3, AFW Pump Suction Pressure vs. CST Level.

A. Incorrect. The first part is correct. The second part is plausible since pump discharge pressure oscillations are an indication of pump cavitation which can occur as a result of inadequate NPSH and NPSH is an indication of the level of the suction source of a pump, however this is incorrect because Attachment 3 compares the suction pressure of the AFW pump to determine the CST Level.

B. Correct.

C. Incorrect. The first part is plausible since the fire protection system consists of Thermal, Ionization, and Ultraviolet detectors, however this is incorrect because the Ultraviolet detectors are used primarily around Fuel Oil Storage Systems. The second part is plausible since pump discharge pressure oscillations are an indication of pump cavitation which can occur as a result of inadequate NPSH and NPSH is an indication of the level of the suction source of a pump, however this is incorrect because Attachment 3 compares the suction pressure of the AFW pump to determine the CST Level.

D. Incorrect. The first part is plausible since the fire protection system consists of Thermal, Ionization, and Ultraviolet detectors, however this is incorrect because the Ultraviolet detectors are used primarily around Fuel Oil Storage Systems. The second part is correct.

086 Fire Protection / 8

086A2.04; Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure to actuate the FPS when required, resulting in fire damage

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.3 3.9
Technical Reference:	FPP-012-02, Fire Pre-Plan A27, Pg 62, Rev. 13 AOP-036.02, Section 3.1, Step 9, Pg 15, Rev. 19
References to be provided:	None
Learning Objective:	AOP-LP-3.36, Obj. 3
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2

2018 NRC RO 066/BANK/FUNDAMENTAL//AD-OP-ALL-1000/NONE//G2.1.2/

An RO assigned as the OATC is responsible for which ONE of the following actions in accordance with AD-OP-ALL-1000, Conduct of Operations?

A✓ Uses diverse and redundant indications for verification of plant or equipment status.

B. Ensures engineered safety feature equipment operates without operator action.

C. Prioritizes focus and support for activities that ensure reactor core protection and accident mitigation strategies.

D. Documents all plant deficiencies discovered during the shift and reports them to the CRS.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AD-OP-ALL-1000 the RO uses diverse and redundant indications for verification of plant or equipment status.

A. Correct.

B. Incorrect. Plausible since the RO position is responsible for using diverse and redundant indications for verification of plant or equipment status, however this is incorrect because he/she is allowed to operate ESF equipment that does not properly respond to changes in plant conditions.

C. Incorrect. Plausible since the RO position is responsible for operating the Reactor safely and efficiently, however this is incorrect because the CRS is the position repsonsible for proritizing the activities required to support safe operation of the Reactor.

D. Incorrect. Plausible since the RO position is responsible to initiate prompt corrective action upon the reciept of abnormal conditions, however this is incorrect because the AO position is required to document all deficencies discovered during the shift and report them to shift supervision.

2.1 Conduct of Operations

G2.1.2; Knowledge of operator responsibilities during all modes of plant operation.

(CFR: 41.10 / 45.13)

Importance Rating: 4.1 4.4

Technical Reference: AD-OP-ALL-1000, Section 4.9, Step 3, Pg 13, Rev. 8

References to be provided: None

Learning Objective: PP-LP-3.0, Obj. 2

Question Origin: Bank

Comments: None

Tier/Group: T3

2018 NRC RO 067/BANK/FUNDAMENTAL//TS 3.9/NONE/2012 NRC RO 66/G2.1.36/

Which ONE of the following would require suspension of Core Alterations?

A. The Containment equipment hatch can be closed with 8 bolts.

B✓ Source Range audible indication inside Containment becomes unavailable.

C. Reactor Cavity water level is 23 feet 3 inches above the vessel flange with only one RHR loop operable and in operation.

D. Train 'A' Fuel Handling Building Emergency Exhaust unit is OPERABLE and in operation; Train 'B' has just been declared INOPERABLE.

Plausibility and Answer Analysis

Reason answer is correct: TS 3.9.2 states that 1 SR channel must have audible indication in the MCR and CNMT during Mode 6. If audible indication is not available in CNMT, TS 3.9.2 states: immediately suspend all operations involving core alterations or positive reactivity additions.

Plausible because these are all Tech Spec immediate actions for a certain event. They cause LCO actions to be completed immediately but not all apply to core alterations or are already in an alignment that is allowed.

A. Incorrect. TS 3.9.4.this meets the requirement and will not require suspension because the equipment door is cabable of being closed and only needs to be held in place by a minimum of four bolts.

B. Correct.

C. Incorrect. TS 3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation in MODE 6 with irradiated fuel in the vessel when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet

D. Incorrect. Alterations involving movement of fuel within the storage pool or crane operation with loads over the storage pool are not allowed but Core Alterations are not affected. (TS 3.9.12)

2.1 Conduct of Operations

G2.1.36; Knowledge of procedures and limitations involved in core alterations.

(CFR: 41.10 / 43.6 / 45.7)

Importance Rating:	3.0 4.1
Technical Reference:	Technical Specification 3.9.2
References to be provided:	None
Learning Objective:	LP-TS-2.0, Obj. 2
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 068/NEW/C/A//CURVES/NONE//G2.1.45/

The operating crew is raising power to 100% in accordance with GP-005, Power Operation (Mode 2 to Mode 1).

PR NIs were indicating 15% Reactor power when the Generator was synched to the grid.

Current indications are:
- Average PR NIs are indicating 20% Reactor power
- T_{avg} is 563°F

In accordance with GP-005, which ONE of the following is a diverse indication that could be used by the crew as an indication of true Reactor power?

(Reference Provided)

A. Pressurizer Level is 31.5%

B✓ Average Percent RCS Loop ΔT is 22%

C. Average Axial Flux (% Delta-I) is -1.0

D. First Stage Pressure is 110 psig

Plausibility and Answer Analysis

Reason answer is correct: Until a calorimetric is performed at 30% power, true reactor power shall be assumed equal to the highest of the following indicators: average Power Range NI value, average percent RCS Loop ΔT, or Main Turbine load. Average Percent RCS Loop ΔT is 22% which is slightly higher is an expected plant parameter and until the 30% power calorimetric is performed should be used as a diverse indication of true Reactor power.

A Incorrect. Plausible since Pressurizer level would read approximately 31.5% using Curve H-X-14 Rev. 2 as a reference but it is NOT part of the diverse indications for true Reactor power in accordance with GP-005.

B Correct.

C. Incorrect. Plausible since the Average Axial Flux for 20% power using curve F-20-2 is approximately -1.0 but but it is NOT part of the diverse indications for true Reactor power in accordance with GP-005.

D. Incorrect. Plausible since the First Stage Pressure could be used as a indication of true reactor power below 30% prior to the calorimetric being performed but this reading is NOT the highest of the indicators for true Reactor power.

2.1 Conduct of Operations

G2.1.45; Ability to identify and interpret diverse indications to validate the response of another indication.

(CFR: 41.7 / 43.5 / 45.4)

Importance Rating:	4.3 4.3
Technical Reference:	GP-005, Precaution and Limitation 1, Pg 4, Rev. 102
References to be provided:	Curves F-20-2, G-4, H-12, and H-X-14
Learning Objective:	LP-IE-17.3, Obj. 1
Question Origin:	NEW
Comments:	None
Tier/Group:	T3

2018 NRC RO 069/BANK/FUNDAMENTAL//OMP-003/NONE/2012 NRC RO 68/G2.2.18/

Given the following plant conditions:
- Mode 6, Refueling is in progress

Which ONE of the following lists the Key Safety Functions per OMP-003, Outage Shutdown Risk Management?

A. Subcriticality, Core Cooling, Integrity, Inventory, Containment

B. Decay Heat Removal, Reactivity Control, Integrity, Containment, Residual Heat Removal

C. Residual Heat Removal, Inventory, Cavity Level, Electrical Power, Auxiliary Feed Water

D.✓ Containment, Inventory, Decay Heat Removal, Electrical Power, Reactivity Control

Plausibility and Answer Analysis

Reason answer is correct: Per OMP-003 the key safety functions when shutdown are Containment, Inventory, Decay Heat Removal, Electrical Power and Reactivity Control (CIDER)

A. Incorrect. Plausible because the catagories listed are those for the CSFST 's

B. Incorrect. Plausible because the catagories listed are the key safety functions while determining on-line risk

C. Incorrect. Plausible because the categories listed are subcategories of the key safety functions while shutdown.

D. Correct. CIDER

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:	OMP-003, Step 5.2.1.12, Pg 15, Rev. 41
References to be provided:	None
Learning Objective:	None
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 070/NEW/C/A//TECH SPEC 3.1.2.5/NONE//G2.2.35/

Given the following plant conditions:
- Reactor Vessel Head Closure bolts are fully tensioned
- RCS T_{avg} is 135°F
- RWST level is 20%

Which ONE of the following identifies (1) the current plant OPERATIONAL MODE and (2) the status of the RWST Limiting Condtion For Operation in accordance with Technical Specification 3.1.2.5, Reactivity Control Systems, Borated Water Source - Shutdown?

A. (1) Mode 5

(2) NOT met

B✓ (1) Mode 5

(2) met

C. (1) Mode 6

(2) NOT met

D. (1) Mode 6

(2) met

Plausibility and Answer Analysis

Reason answer is correct: Mode 5 is defined in Tech Spec Table 1.2 as a plant condition where the average coolant temperature is = 200°F T_{avg}. With an RCS T_{avg} of 135°F Tech Spec Mode 5 applies. With the unit in Mode 5 per Tech Spec 3.1.2.5 in order for the RWST to be operable it must have a minimum volume of 106,000 gallons which is 12%, with boron concentration between 2400 and 2600 ppm and a minimum solution temperature of 40°F.

A. Incorrect. The first part is correct. The second part is plausible since RWST level is below 23% which is the minimum level for the BAT and below the 23.4% value for the ECCS automatic swap to the Containment Sump. However this is incorrect as the minimum RWST level in mode 5 is 12%.

B. Correct.

C. Incorrect. The first part is plausible since Refueling is Mode 6, which is = 140°F and the temperature of 135°F is < 140°F. However this is incorrect as the reactor vessel head bolts must be less than fully tensioned or the head removed in order to be in Mode 6. The second part is plausible since RWST level is below 23% which is the minimum level for the BAT and below the 23.4% value for the ECCS automatic swap to the Containment Sump. However this is incorrect as the minimum RWST level in mode 5 is 12%.

D. Incorrect. The first part is plausible since Refueling is Mode 6, which is = 140°F and the temperature of 135°F is < 140°F. However this is incorrect as the reactor vessel head bolts must be less than fully tensioned or the head removed in order to be in Mode 6. The second part is correct.

2.2 Equipment Control

G2.2.35; Ability to determine Technical Specification Mode of Operation.

(CFR: 41.7 / 41.10 / 43.2 / 45.13)

Importance Rating: 3.6 4.5

Technical Reference: Technical Specification Table 1.2, Operational Modes, Technical Specification 3.1.2.5

References to be provided: None

Learning Objective: TS-LP-2.0/3.0/5.0/8.0, Obj. 3.a and 4.a

Question Origin: New

Comments: None

Tier/Group: T3

2018 NRC RO 071/BANK/FUNDAMENTAL//FSAR 15.0/NONE//G2.2.38/

Which ONE of the following identifies WHY it is essential to operate the plant in accordance with the limiting conditions required by Technical Specifications?

A. To ensure minimum operator actions are required to mitigate the consequences of a Reactor accident.

B✓ To ensure the assumptions made in the accident analysis remain valid in the event of a Reactor accident.

C. To ensure no radiation is released to the environment as a result of a Reactor accident.

D. To ensure that DNBR will remain greater than the design limit for all Reactor accidents.

Plausibility and Answer Analysis

Reason answer is correct: AREVA Analyses - The accidents presented in Chapter 15 are analyzed at limiting conditions consistent with the Technical Specifications.

A. Incorrect. *Plausible since accident analysis is combined with the control system setpoint study, which is designed to automatically maintain prescribed conditions in the plant, to show the plant can meet both safety and operability requirements, however this is incorrect because this is a method used to comply with the limiting conditions of operations and not the reason why the plant is operated within these limited conditions.*

B. Correct.

C. Incorrect. *Plausible since the basic principle of plant design is to limit the radiological risk to the public, however this is incorrect because the design requirement for radiological risk is the extreme situation having the greatest risk to the public that is least likely to occur and not the reason why the plant is operated within these limited conditions..*

D. Incorrect. *Plausible since DNBR is one of the criteria designed to be limited by the Reactor Protection and Emergency Core Cooling systems, however this is incorrect because this is a result of compliance with the limiting conditions of operations and not the reason why the plant is operated within these limited conditions.*

2.2 Equipment Control

G2.2.38; 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:	FSAR Chapter 15.0
References to be provided:	None
Learning Objective:	TAA-LP-2.10, Obj. 6
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 072/BANK/FUNDAMENTAL//10CFR20/NONE//G2.3.4/

Which ONE of the following is the NRC annual dose limit for a declared pregnant female for the duration of the pregnancy?

A. 200 mrem

B✓ 500 mrem

C. 2 rem

D. 5 rem

Plausibility and Answer Analysis

Reason answer is correct: In accordance with PD-RP-ALL-0001, the NRC annual dose limit for a declared pregnant female is 500 mrem for the pregnancy duration.

A. Incorrect. Plausible since this 10% of an adult limit for Duke Energy imposed on minors (<18 years of age).

B. Correct.

C. Incorrect. Plausible since this is theTotal Effective Doese Equivalent (TEDE) for Duke Energy.

D. Incorrect. Plausible since this is theTotal Effective Doese Equivalent (TEDE) for the NRC Annual Dose.

2.3 Radiation Control

G2.3.4; Knowledge of radiation exposure limits under normal or emergency conditions.

(CFR: 41.12 / 43.4 / 45.10)

Importance Rating:	3.2 3.7
Technical Reference:	PD-RP-ALL-0001, Section 5.2.2, Pg 18, Rev. 7
References to be provided:	None
Learning Objective:	PP-LP-3.7, Obj. 1.c
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 073/BANK/FUNDAMENTAL//AOP-038/NONE//G2.4.11/

Which ONE of the following ramp rates is the LOWEST that would require using AOP-038, Rapid Down Power, to reduce power?

A. 4 MW/min

B✓ 6 MW/min

C. 10 MW/min

D. 15 MW/min

Plausibility and Answer Analysis

Reason answer is correct: Plant Conditions that require a rapid reduction in power level to preclude a plant trip (or in lieu of a plant trip) may warrant entry into this procedure. Any condition requiring greater than 5 MW/min load reductions.

A. Incorrect. Plausible since this is the power level that GP-006 uses as the maximum power level and there are many AOP's that have the operator determine which shutdown procedure will be implented either GP-006 or AOP-038.

B. Correct.

C. Incorrect. Plausible since AOP-038 identifies a target load reduction rate of 10 Mw/Min when the ASI system is supplying RCP seal injection at the beginning of life, however this is incorrect since entry in to AOP-038 is allowed for any condition requiring greater than 5 MW/min load reduction.

D. Incorrect. Plausible since a notification to chemistry is required if power is changed greater than 15% in any 1 hour period, however this is incorrect since entry in to AOP-038 is allowed for any condition requiring greater than 5 MW/min load reduction.

2.4 Emergency Procedures / Plan

G2.4.11; Knowledge of abnormal condition procedures.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	4.0 4.2
Technical Reference:	AOP-038, Section 2.0, Pg 3, Rev. 44
References to be provided:	None
Learning Objective:	AOP-LP-3.38, Obj. 1
Question Origin:	Bank
Comments:	None
Tier/Group:	T3

2018 NRC RO 074/BANK/FUNDAMENTAL//FPP-002/NONE//G2.4.25/

In accordance with FPP-002, Fire Emergency, when shall the Control Room Operator sound the plant fire alarm?

A. Upon the receipt of any fire alarm in the Control Room

B✓ After a second fire alarm is received in an adjacent zone

C. Upon the receipt of an Incipient Fire Detection system alert

D. After receipt of a single fire alarm for areas inside Containment

Plausibility and Answer Analysis

Reason answer is correct: Per FPP-002 When a second alarm is received in the Control room, then sound the plant fire alarm.

A. Incorrect. Plausible since a single fire alarm inside containment will require action by the control room to monitor for diverse indications to confirm a fire exists, however this is incorrect the plant fire alarm is not sounded until confirmation a fire exists.

B. Correct.

C. Incorrect. Plausible since if an Incipient Fire Detection system alarm is received before an Operator or on-shift I&C tech has responded to an Incipient Fire Detection system alert, requires the plant fire alarm to be sounded, however this is incorrect the plant fire alarm is not sounded until confirmation a fire exists if the Incipient Fire Detection system is only in alert status.

D. Incorrect. Plausible since a single fire alarm inside containment has specific actions for the control room to monitor for diverse indications to determine a fire exists, however this is incorrect the plant fire alarm is not sounded until confirmation a fire exists.

2.4 Emergency Procedures / Plan

G2.4.25; Knowledge of fire protection procedures.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating: 3.3 3.7

Technical Reference: FPP-002, Section 6.1.6, Pg 12, Rev. 42

References to be provided: None

Learning Objective: PP-LP-3.15, Obj. 1

Question Origin: Bank

Comments: None

Tier/Group: T3

2018 NRC RO 075/BANK/FUNDAMENTAL//ALB-011, 018/NONE//G2.4.31/

Given the following plant conditions:

- The unit is operating at 100% power
- A plant transient results in the crew manually tripping the Turbine

What is the expected color of the following flashing MCB Annunciators?

	ALB-018-2-5 Turbine Trip Manual	ALB-011-3-3 Reactor Trip Turbine Trip P7
A✓	Red	Red
B.	Red	White
C.	White	Red
D.	White	White

Plausibility and Answer Analysis

Reason answer is correct: Three ALBs, two for reactor trip (ALB-011 and ALB-012), one for turbine trip (ALB-018), are provided to function as First Out ALBs. The first alarm received on these ALBs will be red to identify the cause of the trip (i.e., one red for the reactor trip and another red for the turbine trip). Subsequent alarms will be white.

A. Correct.

B. Incorrect. Plausible since the Turbine Trip was generated manually the candidate may determine that only the first annunciator generated for the Turbine Trip System will be Red and subsequent alarms will be white; this however is incorrect because the MCB has a First Out annuciator system for the Reactor system that is separate from the Turbine System and will generate a Red First Out alarm for both systems.

C. Incorrect. Plausible since the Turbine Trip was generated manually the candidate may determine that only the first annunciator generated for the Reactor Trip system will be Red and remaining alarms will be white; this however is incorrect because the MCB has a First Out annuciator system for the Reactor system that is separate from the Turbine System and will generate a Red First Out alarm for both systems.

D. Incorrect. Plausible since the Turbine Trip was generated manually the candidate may determine that only automatic trip signals for both the Reactor and Turbine Trip systems will be Red and subsequent alarms will be white; this however is incorrect because the MCB has a First Out annuciator system for the Reactor system that is separate from the Turbine System and will generate a Red First Out alarm for both systems based on automatic or manual trip signals.

2.4 Emergency Procedures / Plan

G2.4.31; Knowledge of annunciator alarms, indications, or response procedures.

(CFR: 41.10 / 45.3)

Importance Rating: 4.2 4.1

Technical Reference: APP-ALB-011, Window 3-3, Pg 8, Rev. 8
APP-ALB-018, Window 2-5, Pg 10, Rev. 21

References to be provided: None

Learning Objective: MCB Lesson Plan, Obj. 4

Question Origin: Bank

Comments: None

Tier/Group: T3

2018 NRC SRO 001/NEW/C/A//EOP-ES-1.2, BKGRD/NONE//008 AG2.4.18/

Given the following plant conditions:
- The unit is operating at 100% power
- 'A' MFW Pump has just tripped

Subsequently:
- The crew has performed a manual Reactor Trip
- PCV-445B, PRZ PORV has lifted and has not closed, the associated block valve breaker tripped when the valve was taken to close

Current plant conditions are:
- PRZ level is 100%
- RCS pressure is 1350 psig and slowly lowering
- All RCP's have been secured
- The crew is implementing EOP-E-1, Loss of Reactor or Secondary Coolant

Which ONE of the following completes the statement below concerning the event in progress?

The crew will next transition to __(1)__ , which directs the 'B' RCP to be started in order to __(2)__ .

A. (1) EOP-ES-1.1, SI Termination

(2) provide normal PRZ spray

B. (1) EOP-ES-1.1, SI Termination

(2) mix the RCS coolant to nearly uniform temperature

C. (1) EOP-ES-1.2, Post LOCA Cooldown and Depressurization

(2) provide normal PRZ spray

D.✓ (1) EOP-ES-1.2, Post LOCA Cooldown and Depressurization

(2) mix the RCS coolant to nearly uniform temperature

Plausibility and Answer Analysis

Reason answer is correct: With one of the PRZ PORV's open and RCS pressure decreasing. When RCS status is checked for pressure < 230 psig or RHR HX header flow > 1000 GPM both answers will be NO and the RNO will be used to transition to EOP-ES-1.2, Post LOCA Cooldown and Depressurization. The basis document for ES-1.2 states that starting the RCP would not be needed for normal spray capability but to serve to mix the RCS coolant to nearly uniform temperature. (PRZ level is 100% therefore spray will not be effective)

A. Incorrect. The first part is plausible since this would be the procedure to transition to from EOP-E-1 if the following conditions were met: RCS subcooling > 10°F (subcooling would not have been met), secondary heat sink was adequate (no information provided but if AFW functioned as designed it will be adequate) RCS pressure stable or rising (NO slowly lowering) and PRZ level > 10% (Yes). Since a given parameter of RCS pressure is not stable or rising the transition to ES-1.1 is not appropriate.
The second part is plausible since there are notes in both EOP-ES-1.1 and EOP-ES-1.2 stating that the RCPs should be run in order of priority (B only, A AND C, A only, C only) to provide normal PRZ spray. This is NOT the basis for starting the B RCP since the PRZ level is 100% spray will not be effective.

B. Incorrect. The first part is plausible since this would be the procedure to transition to from EOP-E-1 if the following conditions were met: RCS subcooling > 10°F (subcooling would not have been met), secondary heat sink was adequate (no information provided but if AFW functioned as designed it will be adequate) RCS pressure stable or rising (NO slowly lowering) and PRZ level > 10% (Yes). Since a given parameter of RCS pressure is not stable or rising the transition to ES-1.1 is not appropriate.
The second part is correct.

C. Incorrect. The first part is correct.
The second part is plausible since there are notes in both EOP-ES-1.1 and EOP-ES-1.2 stating that the RCPs should be run in order of priority (B only, A AND C, A only, C only) to provide normal PRZ spray. This is NOT the basis for starting the B RCP since the PRZ level is 100% spray will not be effective.

D. Correct.

000008 Pressurizer Vapor Space Accident / 3

008AG2.4.18; Knowledge and specific bases for EOPs.

(CFR: 41.10 / 43.1 / 45.13)

Importance Rating:	3.3 4.0
Technical Reference:	EOP-ES-1.2, Step 13, Pg 16, Rev. 4 ERG-BKGRD-ES-1.2, Pg 44, Rev. 2
References to be provided:	None
Learning Objective:	EOP-LP-3.05, Obj. 3.b
Question Origin:	New
Comments:	HNP was not able to create a valid SRO question for K/A 008AG2.4.2; Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. Requested new K/A. Phonecon 10/5: HNP discussed being unable to create an SRO level question based on the guidance of ES-401 Attachment 2. This K/A topic dealing with APE 008, PZR Vapor Space Accident was tied with the Generic K/A 2.4.2, knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions, so selected a new K/A, keeping 008 and randomly selecting from the remaining SRO Level items for the Generic 2.4 K/As: New K/A 008AG2.4.18: Knowledge and specific bases for EOPs .
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by correctly selecting the transition procedure based on plant conditions and have detailed knowledge of the content of the procedure versus general knowledge of the procedures overall mitigative strategy or purpose.

2018 NRC SRO 002/NEW/FUNDAMENTAL//EOP-FR-H.1, BKGRD/NONE//054 AG2.4.35/

Given the following plant conditions:

- The Reactor has tripped from 50% power due to a loss of all Feedwater
- The crew is performing EOP-FR-H.1, Response To Loss Of Secondary Heat Sink, and have started RCS Bleed and Feed when the 'B' Main FW pump was returned to service
- Core Exit TCs are 540°F and stable

Which ONE of the following completes the statements below?

Heat sink will be recovered with Main Feed flow to one intact SG locally utilizing an Aux Operator in accordance with __(1)__ .

In accordance with the EOP-FR-H.1 ERG Background Document, this method of feed is used to __(2)__ .

A. (1) Attachment 1, Guidance on Restoration of Feed Flow

(2) improve feed control while refilling a hot dry SG

B✓ (1) Attachment 1, Guidance on Restoration of Feed Flow

(2) minimize the potential impact of excessive thermal stresses

C. (1) Attachment 2, Establishing Main FW Flow to SGs

(2) improve feed control while refilling a hot dry SG

D. (1) Attachment 2, Establishing Main FW Flow to SGs

(2) minimize the potential impact of excessive thermal stresses

Plausibility and Answer Analysis

Reason answer is correct: In accordance with EOP-FR-H.1, after Bleed and Feed is performed Attachment 1, step 4 will be used to restore Feed Flow. With Core Exit TC's steady or lowering if MFW flow is used to recover heat sink then feed flow will be established to ONE intact SG at the lowest controllable rate (flow noise just becomes audible at the associated Feed Reg Bypass valve). The WOG background document for establishing Feed flow with stable or lowering RCS temperatures states that Feed flow is limited to minimize the potential impact of excessive thermal stresses since a direct measure of the SG temperature is not available.

A. Incorrect. The first part is correct. The second part is plausible since using only a single controller will allow the operator to focus his/her attention to one parameter which would potentially improve control of the evolution.

B. Correct.

C. Incorrect. The first part is plausible because a Main Feed water pump has been returned to service and this attachment would be used if Bleed and Feed had not been started. The second part is plausible since using only a single controller will allow the operator to focus his/her attention to one parameter which would potentially improve control of the evolution.

D. Incorrect. The first part is plausible because a Main Feed water pump has been returned to service and this attachment would be used if Bleed and Feed had not been started. The second part is correct.

000054 Loss of Main Feedwater /4

054AG2.4.35; Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.8 4.0
Technical Reference:	EOP-FR-H.1, Attachment 1, Step 4.a.2, Pg 63, Rev. 3 ERG-BKGRD-FR-H.1, Pg 52, Rev. 2
References to be provided:	None
Learning Objective:	EOP-LP-3.11, Obj. 5.c
Question origin:	New
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate or recover, or with which to proceed. The candidate's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by having the knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps

2018 NRC SRO 003/BANK/C/A//EOP-ECA-0.0/NONE/2013 NRC SRO 05/055 EA2.04/

Given the following plant conditions:
- The unit was operating at 100% power when a loss of Offsite power occurred
- 6.9 KV Emergency Bus 1B-SB 86 lockout actuates
- EDG 'A' fails to start
- The ASI system is supplying RCP seal injection
- The crew is implementing ECA-0.0, Loss of All AC Power
- EOP-ECA-0.0, step 33 to initiate a cooldown to control PZR level using the SG PORVs is in progress

Which ONE of the following completes the statements below in accordance with EOP-ECA-0.0?

___(1)___ SG PORV(s) can be operated from the MCB.

The CRS will direct the OATC to stop the RCS cooldown when ___(2)___.

A. (1) All three

(2) the RCS pressure is < 700 psig

B. (1) All three

(2) all cold leg temperature are < 400°F

C. (1) ONLY the 'C'

(2) the RCS pressure is < 700 psig

D✓ (1) ONLY the 'C'

(2) all cold leg temperature are < 400°F

Plausibility and Answer Analysis

Reason answer is correct: In accordance with ECA-0.0 the operator is directed to dump steam using SG C PORV from the MCB. 'C' SG PORV servos are powered from a UPS power supply IDP-1A-SIII (DC power when loss of all AC occurrs) which allows ONLY the 'C' SG PORV to operate during a loss of AC power. RCS cooldown is secured when RCS cold leg temperatures are all less than 400°F. When the RCS is below 400°F minimal RCP seal leakage will occur after the ASI pump is secured in subsequent steps of the procedure.

A. Incorrect. The first part is plausible because normally all three steam generator PORVs are operated from the Main Control Board. Second part of answer is correct.

B. Incorrect. The first part is plausible because normally all three steam generator PORVs are operated from the Main Control Board. The second part is plausible becuase 700 psig is one of the termination criteria for RCS depressurization in a subsequent step in ECA 0.0.

C. Incorrect. The first part of answer is correct. The second part is plausible becuase 700 psig is one of the termination criteria for RCS depressurization in a subsequent step in ECA 0.0.

D. Correct.

000055 Station Blackout / 6

055EA2.04; Ability to determine or interpret the following as they apply to a Station Blackout: Instruments and controls operable with only DC battery power available

(CFR 43.5 / 45.13)

Importance Rating:	3.7 4.1
Technical Reference:	ECA-0.0, Step 33-34, Pg 56 and 58, Rev. 7
References to be provided:	None
Learning Objective:	EOP-LP-3.7, Obj. 6
Question Origin:	Bank
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must have detailed knowledge of the content of the procedure versus general knowledge of the procedures overall mitigative strategy or purpose. The candidate's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item.

2018 NRC SRO 004/NEW/C/A//ALB-002/NONE//065 AA2.04/

Given the following plant conditions:
- The plant is operating at 100% power
- Instrument Air is aligned in SEQUENCE 2 with all Air Compressors available

Over the next minute the following indications are observed:
- ALB-002-8-3B, Air Comp A/B Trouble, alarms
- Instrument Air pressure is 105 psig and slowly lowering

In accordance with APP-ALB-002, which ONE of the following identifies (1) the status of the 1B Air Compressor AND (2) the procedure the CRS will implement to initate corrective actions for this condition?

A. (1) Running

(2) AOP-017, Loss of Instrument Air

B. (1) Running

(2) OP-151.01, Compressed Air

C. (1) Tripped

(2) AOP-017, Loss of Instrument Air

D.✓ (1) Tripped

(2) OP-151.01, Compressed Air

Plausibility and Answer Analysis

Reason answer is correct: In accordance with APP-ALB-002 window 8-3B, the cause of the alarm are from conditions that will trip Air Compressors 1A-NNS or 1B-NNS. If alarm is due to Air Compressor 1B trip, then ensure Air Compressor 1A AND 1C are operating properly. Dispatch an Operator to determine the cause of Air Compressor trip, and initiate corrective actions per OP-151.01, Compressed Air.

A. Incorrect. The first part is plausible since air compressor 1B is connected to the CAS Panel in sequence 2, the lead compressor will load at 98 psig and unload at 114 psig, however this is incorrect because ALB-002 window 8-3B has been received therefore the compressor will be tripped. The second part is plausible since AOP-017 is entered to address lowering instrument air pressure due to an abnormal partial loss of Instrument Air pressure, however this is incorrect because APP-ALB-002 directs the crew to initiate corrective actions per OP-151.01.

B. Incorrect. The first part is plausible since air compressor 1B is connected to the CAS Panel in sequence 2, the lead compressor will load at 98 psig and unload at 114 psig, however this is incorrect because ALB-002 window 8-3B has been received therefore the compressor will be tripped. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since AOP-017 is entered to address lowering instrument air pressure due to an abnormal partial loss of Instrument Air pressure, however this is incorrect because APP-ALB-002 directs the crew to initiate corrective actions per OP-151.01.

D. Correct.

000065 (APE 65) Loss of Instrument Air / 8

065AA2.04; Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Typical conditions which could cause a compressor trip (e.g., high temperature)
(CFR: 43.5 / 45.13)

Importance Rating:	2.2 2.7
Technical Reference:	APP-ALB-002, Window 8-3B, Pg 42, Rev. 53
References to be provided:	None
Learning Objective:	ISA Lesson Plan, Obj. 4.b
Question Origin:	New
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must have detailed knowledge of the content of the procedure versus general knowledge of the procedures overall mitigative strategy or purpose. The candidate’s knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure’s content is required to correctly answer the written test item.

2018 NRC SRO 005/NEW/C/A//EOP-ECA-1.2 AP-617/AP-617 ATT.1//WE04 EG2.4.30/

Given the following plant conditions:

- At 1117 A LOCA has caused an automatic SI
- At 1119 it has been determined that the LOCA is in the RAB
- At 1124 the Shift Manager has declared the approriate EAL classification for this event

Which ONE of the following completes the statements below?

The LATEST time that the SM had to declare the event was by ___(1)___ .

The LATEST time that the NRC should be notified about this event is by ___(2)___ .

(Reference provided)

A✓ (1) 1132

(2) 1224

B. (1) 1132

(2) 1517

C. (1) 1134

(2) 1224

D. (1) 1134

(2) 1517

Plausibility and Answer Analysis

Reason answer is correct: In accordance with PEP-110 Section 3.1, General Guidelines for Use of the EAL Matrix, All emergency classifications shall be made within 15 minutes from the time that the conditions have reached an EAL threshold. At 1117 a LOCA has caused an automatic SI which would be classified as an ALERT FA1.1 At 1119 it was determined that a LOCA outside Containment is occurring which would be a loss of Containment Barrier. The combination of the loss of 2 barriers would now make the EAL classification a Site Area Emergency FS1.1 The first notification time started at 1117 when the SI actuated. The SM had 15 minutes from 1117 to make the notification which would have been no later than 1132. Notification to the NRC for this event is required to be performed within 1 hour from when the Shift Manager has made a declaration, 1124 + 1 hour would be no later than 1224.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since a valid ECCS discharge into the RCS occurred at 1117 (4 hours earlier) additionally a RPS actuation also occurred with the Reactor critical (at that same time). Both of these conditions would warrant a notification to the NRC within 4 hours.

C. Incorrect. The first part is plausible since a Site Area Emergency occurred at 1119 and 15 minutes from then would be 1134 but this is incorrect since the first EAL classification start point would have been at 1117.
The second part is correct.

D. Incorrect. The first part is plausible since a Site Area Emergency occurred at 1119 and 15 minutes from then would be 1134 but this is incorrect since the first EAL classification start point would have been at 1117. The second part is plausible since a valid ECCS discharge into the RCS occurred at 1117 (4 hours earlier) additionally a RPS actuation also occurred with the Reactor critical (at that same time). Both of these conditions would warrant a notification to the NRC within 4 hours.

W/E04 LOCA Outside Containment / 3

WE04EG2.4.30; Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

(CFR: 41.10 / 43.5 / 45.11)

Importance Rating:	2.7 4.1
Technical Reference:	EOP-ECA-1.2, Step 6, Pg 5, Rev. 0 AP-617, Attachment 1, Pg 16, Rev. 39
References to be provided:	AP-617 Attachment 1
Learning Objective:	EOP-LP-3.03, Obj. 2.d PP-LP-2.17, Obj. 5
Question Origin:	New
Comments:	Discuss with Dan Bacon to confirm K/A is 2.4.3 or 2.4.30, Verbiage for K/A topic is the wording for 2.4.30. Phonecon 6/13: Dan confirms the intented K/A to be evaluated is WE04 EG2.4.30.
Tier/Group:	T1/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(1): Condition and limitations in the facility license. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, the candidate's knowledge can be evaluated at the level of 10 CFR 55.43(b)(1) by ensuring that the additional knowledge of the reporting requirements are required to correctly answer the written test item. In this instance the candidate must know the difference in reporting requirements between 2 different situations contained in the AP-617 for actions when a declaration of an emergency classification is made.

2018 NRC SRO 006/NEW/C/A//FR-H.1, BKGRD/NONE/EARLY/WE05 EA2.2/

Given the following plant conditions:

- The unit was operating at 100% power when a LOCA occurred
- While implementing EOP-E-0, Reactor Trip and Safety Injection, Containment pressure peaked at 11.4 psig
- The crew has transitioned to EOP-E-1, Loss of Reactor or Secondary Coolant
- During implentation of EOP-E-1 a loss of heat sink occurred

Current conditions are:

- The crew is implementing EOP-FR-H.1, Response to Loss of Secondary Heat Sink
- RCS pressure is 830 psig and stable
- SG pressures are 875 psig and slowly lowering
- Containment pressure is 2.2 psig and slowly lowering
- SG Levels stable as follows:

	A	B	C
NR	0%	0%	0%
WR	28%	29%	49%

Which ONE of the following completes the statements below?

RCS Feed and Bleed ___(1)___ required.

The CRS should ___(2)___ .

A. (1) is

(2) transition back to EOP-E-1

B. (1) is

(2) transition to EOP-E-2, Faulted SG Isolation

C✓ (1) is not

(2) transition back to EOP-E-1

D. (1) is not

(2) transition to EOP-E-2, Faulted SG Isolation

Plausibility and Answer Analysis

Reason answer is correct: Because two of three Wide Range S/G Water Level indicators are NOT below 15% (30% for adverse containment) Bleed and Feed criteria is not met. EOP-FR-H.1 loops the candidate from step 15 back to step 1 until the criteria for Bleed and Feed is met or the conditions to restore a heat sink are no longer required. Step 2 checks for Secondary Heat Sink requirement. Since the RCS pressure is less than the current SG pressures Secondary Heat Sink is not required for the current conditions. The crew should transition back to the procedure and step in effect which would have been the last step in EOP-E-1.

A. Incorrect. The first part is plausible since two of three Wide Range S/G Water Level indicators are below 30% which meet the requirements to transition to Bleed and Feed for adverse containment, however this is incorrect because the pressure in Containment has lowered below the reset value (3 psig) for adverse Containment and the Bleed and Feed transition requirement is now two of three Wide Range S/G Water Level indicators are below 15%. The second part is correct.

B. Incorrect. The first part is plausible since two of three Wide Range S/G Water Level indicators are below 30% which meet the requirements to transition to Bleed and Feed for adverse containment, however this is incorrect because the pressure in Containment has lowered below the reset value (3 psig) for adverse Containment and the Bleed and Feed transition requirement is now two of three Wide Range S/G Water Level indicators are below 15%. The second part is plausible since the SG pressures are low and lowering which could indicate a steam break is occurring but the transition from EOP-FR-H.1 would be back to the procedure and step in effect not to a different procedure. The transition to EOP-E-2 would only occur after the crew was implementing EOP-E-1.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since the SG pressures are low and lowering which could indicate a steam break is occurring but the transition from EOP-FR-H.1 would be back to the procedure and step in effect not to a different procedure. The transition to EOP-E-2 would only occur after the crew was implementing EOP-E-1.

QUESTIONS REPORT

for 2018 NRC RO SRO WRITTEN EXAM REV 75 Submittal

W/E05 Loss of Secondary Heat Sink / 4

WE05EA2.2; Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

(CFR: 43.5 / 45.13)

Importance Rating:	3.7 4.3
Technical Reference:	EOP-FR-H.1, Step 2, Pg 3, Rev. 3 ERG-BKGRD-FR-H.1, Pg 65, Rev. 2
References to be provided:	None
Learning Objective:	EOP-LP-3.11, Obj. 5
Question Origin:	New
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must have detailed knowledge of the content of the procedure versus general knowledge of the procedures overall mitigative strategy or purpose. The candidate's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item.

2018 NRC SRO 007/NEW/C/A//AOP-001/NONE/EARLY/001 AG2.2.44/

Given the following plant conditions:

Time = 0931
- Reactor Power is 50% and stable
- TI-408A, T_{avg} - T_{ref} Mismatch meter deviation is 0°F and stable
- PZR level is 42.5% and stable
- Control Bank D step counters are at 154 steps

Time = 0932
- Reactor Power is 51% and rising (no load change is in progress)
- TI-408A, T_{avg} - T_{ref} Mismatch meter deviation is +2.1°F and rising
- PZR level 44% and rising
- Control Bank D step counters are at 163 steps and stepping out

Which ONE of the following completes the statements below?

Selected first stage pressure channel high failure ___(1)___ occurred.

In accordance with AOP-001, Malfunction of Rod Control, the CRS will direct the OATC to restore equilibrium power and temperature conditions by ___(2)___ .

A. (1) has

(2) performing a boration

B. (1) has

(2) manually inserting control bank D

C. (1) has not

(2) performing a boration

D✓ (1) has not

(2) manually inserting control bank D

Plausibility and Answer Analysis

Reason answer is correct: Since the Tavg - Tref meter is reading a + 2°F if T_{ref} (selected first stage pressure channel) had failed high the indication would be negative not positive therefore the selected first stage pressure channel failure high has NOT occurred.
In accordance with AOP-001 section 3.2, Continuous Spurious Control Bank Motion step 2 the directions provided by the CRS to the OATC is to manually operate affected control bank to restore equilibrium power and temperature conditions.

A. Incorrect. The first part is plausible since a failure of the selected first stage pressure channel high will cause the rod control system to withdraw the control rods if in automatic rod control. This is incorrect since the Tavg - Tref meter is reading a + 2°F if T_{ref} (selected first stage pressure channel) had failed high the indication would be negative not positive. The second part is plausible since a boration would reduce T_{avg} to restore an elevated T_{avg} condition and will also cause the rod control system to produce a rods in signal to move the affected control bank back to the original position but this is not directed by AOP-001 it is directed in AOP-002 to reduce an increase of RCS temperature. Additionally, AOP-002 entry conditions are not met.

B. Incorrect. The first part is plausible since a failure of the selected first stage pressure channel high will cause the rod control system to withdraw the control rods if in automatic rod control. This is incorrect since the Tavg - Tref meter is reading a + 2°F if T_{ref} (selected first stage pressure channel) had failed high the indication would be negative not positive. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since a boration would reduce T_{avg} to restore an elevated T_{avg} condition and will also cause the rod control system to produce a rods in signal to move the affected control bank back to the original position but this is not directed by AOP-001 it is directed in AOP-002 to reduce an increase of RCS temperature. Additionally, AOP-002 entry conditions are not met.

D. Correct.

000001 Continuous Rod Withdrawal / 1

001AG2.2.44; Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

(CFR: 41.5 / 43.5 / 45.12)

Importance Rating:	4.2 4.4
Technical Reference:	AOP-001, Section 3.2, Step 2, Pg 14, Rev. 48
References to be provided:	None
Learning Objective:	AOP-LP-3.01, Obj. 4
Question Origin:	New
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must have detailed knowledge of the content of the procedure versus general knowledge of the procedures overall mitigative strategy or purpose. The candidate’s knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure’s content is required to correctly answer the written test item. In this instance the candidate must know the information contained in the AOP for continuous rod motion on how to recover from the reactivity addition caused by a failure of the control circuitry for rod control.

2018 NRC SRO 008/NEW/C/A//TS 3.3.1/TS 3.3.1/EARLY/033 AG2.4.46/

Given the following plant conditions:

- A plant startup is in progress in accordance with GP-004, Mode 3 to Mode 2
- The Shutdown Banks have been withdrawn, Control banks are fully inserted
- NI-35 reads 6 x 10^{-11} amps
- NI-36 reads 6 x 10^{-11} amps

Subsequently:

- Annunciator ALB-013-3-2, Intermediate Range Loss of Comp Voltage, alarms due to Intermediate Range NI-35

Which ONE of the following completes the statements below?

Intermediate Range NI-35 is now indicating ___(1)___ than Intermediate Range NI-36.

In accordance with the actions of Technical Specification 3.3.1, Instrumentation - Reactor Trip System Instrumentation, the failed instrument must be restored prior to raising Thermal Power to above ___(2)___ .

(Reference provided)

A. (1) lower

(2) the P-6 setpoint

B. (1) lower

(2) 10% of Rated Thermal Power

C✓ (1) higher

(2) the P-6 setpoint

D. (1) higher

(2) 10% of Rated Thermal Power

Plausibility and Answer Analysis

Reason answer is correct: The compensating voltage when properly adjusted cancels the gamma radiation that is detected and the detector output only displays the neutron flux. If the compensating voltage on an Intermediate Detector is lost the detector will indicate an additive count of both the neutron flux and gamma radiation and show a higher than actual power reading.
You must determine that the P-6 setpoint has NOT been reached then Tech Spec 3.3.1 action 3 while in Mode 2 and below the P-6 setpoint is applicable. (one of two Intermediate Range detectors indicating > $1x10^{-10}$ amps). For a failed IR channel below this condition Tech Specs require that the channel be restored to operable status prior to exceeding P-6.

A. Incorrect. The first part is plausible since the candidate may misapply the detector response to a loss of compensating voltage and determine the indication will be lower which is the typical response of an indicator in the plant upon a loss of voltage. The second part is correct.

B. Incorrect. The first part is plausible since the candidate may misapply the detector response to a loss of compensating voltage and determine the indication will be lower which is the typical response of an indicator in the plant upon a loss of voltage. The second part is plausible since this would be correct if the Reactor power level was above P-6, however this is incorrect because only the shutdown banks are withdrawn, therefore the reactor is not critical and reactor power is not above P-6.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since this would be correct if the Reactor power level was above P-6, however this is incorrect because only the shutdown banks are withdrawn, therefore the reactor is not critical and reactor power is not above P-6.

000033 Loss of Intermediate Range Nuclear Instrumentation / 7

033AG2.4.46; 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:	Technical Specification 3.3.1
References to be provided:	Technical Specification 3.3.1
Learning Objective:	NIS Lesson Plan, Obj. 4 and 12
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2
SRO Justification:	10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know the application of required Technical Specification actions that are greater than one hour. Requires knowledge of Technical Specification rules of application requirements that are not system knowledge. In this instance the candidate must know the application of Limiting Condition for Operation (LCO) requirements (LCO 3.3.1)

2018 NRC SRO 009/NEW/C/A//OP-120.07, ODCM/NONE//060 AA2.05/

A release of WGDT 'F' is in progress in accordance with OP-120.07, Section 8.39, Venting a Gas Decay Tank with Waste Gas System in Service.

Subsequently:

- REM-1WV-3546, WPB Stack 5 PIG Monitor, has gone into HIGH ALARM

Which ONE of the following completes the statements below?

In accordance with AOP-005, Radiation Monitoring System, the operator will verify ___(1)__ automatically shuts.

In accordance with ODCM 3.11.2.1, Gaseous Effluents - Dose Rates, the basis of the methodology for determining the stack monitor setpoint ensures that dose rates at the site boundary from noble gases do not exceed ___(2)___ mrem/year to the whole body.

A.✓ (1) 3WG-229, WG Decay Tanks E & F To Plant Vent Valve

(2) 500

B. (1) 3WG-229, WG Decay Tanks E & F To Plant Vent Valve

(2) 3000

C. (1) 3WG-221, Tank F Gas Decay Tank Outlet Isolation Valve

(2) 500

D. (1) 3WG-221, Tank F Gas Decay Tank Outlet Isolation Valve

(2) 3000

Plausibility and Answer Analysis

Reason answer is correct: In accordance with AOP-005, Radiation Monitoring System Attachment 3 if a Waste Gas Decay Tank the release is in progress and REM-*1WV-3546, WPB Stack 5 PIG Monitor is in HIGH ALARM, then perform the following: Verify 3WG-229, WG Decay Tanks E & F To Plant Vent Vlv, is Shut.

In accordance with the ODCM opertional requirement 3.11.2.1, The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Technical Specification Figure 5.1-1) shall be limited to the following for noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin. Section 3.1 provides the methodology for stack effluent monitor setpoints to ensure that the dose rates from noble gases at the site boundary do not exceed the limits of 500 mrem/year to the whole body or 3000 mrem/year to the skin as specified in ODCM Operational Requirement 3.11.2.1.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since this is the correct value for the ODCM requirement for dose to the skin, however this is incorrect because the whole body limit is 500 mrem/year.

C. Incorrect. The first part is plausible since WGDT 'F' is in service during the release and shutting the outlet isolation for the tank will terminate the release in progress, however this is incorrect because 3WG-229 will automatically shut to terminate the release while 3WG-221 will require an AO to shut the valve manually. The second part is correct

D. Incorrect. The first part is plausible since WGDT 'F' is in service during the release and shutting the outlet isolation for the tank will terminate the release in progress, however this is incorrect because 3WG-229 will automatically shut to terminate the release while 3WG-221 will require an AO to shut the valve manually. The second part is plausible since this is the correct value for the ODCM requirement for dose to the skin, however this is incorrect because the whole body limit is 500 mrem/year.

000060 Accidental Gaseous Radwaste Release / 9

060AA2.05; Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: That the automatic safety actions have occurred as a result of a high ARM system signal

(CFR: 43.5 / 45.13)

Importance Rating: 3.7 4.2

	ODCM 3.1 Gaseous Effluent - Monitor Alarm Setpoint Determination ODCM 3/4.3.11.2.1.Gaseous Effluent - Dose Rate
References to be provided:	None
Learning Objective:	AOP-LP-3.5, Obj. 4 GWPS Lesson Plan, Obj. 9
Question Origin:	2014 NRC SRO 18
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must have detailed knowledge of the content of the procedure versus general knowledge of the procedures overall mitigative strategy or purpose. The candidate's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item. In this instance the candidate must know the information contained in the AOP for radiation monitoring during a Waste Gas Decay tank release.

2018 NRC SRO 010/NEW/C/A//CSFST, AD-OP-0101/NONE//074 EA2.07/

Given the following plant conditions:

- The crew is implementing EOP-E-1, Loss Of Reactor Or Secondary Coolant
- Plant conditions are as follows:
  - CNMT pressure - 12.6 psig
  - RCS Hot leg temperature - 650°F
  - The five hottest core exit thermocouples on each train are:

Train A	Train B
A08 - 751°F	C08 - 743°F
B05 - 718°F	H05 - 718°F
G02 - 757°F	G08 - 767°F
H11 - 713°F	E12 - 713°F
N06 - 724°F	L14 - 734°F

  - RCS pressure - 200 psig
  - RVLIS Full Range level - 34%
  - ERFIS is NOT available

Which ONE of the following completes the statements below?

EOP-FR-C.1, Response to Inadequate Core Cooling ___(1)___ required to be implemented.

The Site Duty Manager ___(2)___ will be notified by the Shift Manager of the current plant status in accordance with AD-OP-ALL-0101, Event Response And Notifications.

A. (1) is

(2) and Corporate Duty Manager

B.✓ (1) is

(2) ONLY

C. (1) is NOT

(2) and Corporate Duty Manager

D. (1) is NOT

(2) ONLY

Plausibility and Answer Analysis

Reason answer is correct: Since ERFIS is not available neither the SPTOP display or CSFSTs are available and the status of the Core Cooling tree must be determined manually. EOP-USERS-GUIDE states, "The TC value should be considered 'greater than' if five functioning TCs are greater than the setpoint." Therefore, with the conjunction of 5 valid TC reading being greater than 730 (732.6)°F and RVLIS full range level being less than 39 (34)%, CSFST Core Cooling-RED entry conditions are met therefore FR-C.1 is required to be implemented at this time. In accordance with AD-OP-ALL-0101 the Shift Manager is responsible to ensure prompt notification to the Site Duty Manager of events listed in Attachment 3, Event Notification Requirements.

A. Incorrect. The first part is correct. The second part is plausible since Shift Manager is responsible for making notifications the NRC Operations Duty Officer in accordance with AP-617, Reportability Determination And Notification the candidate may misapply this responsibility to the notifying the Corporate Duty Manager, however this is incorrect because the Site Duty manager is responsible to make the notification of plant status to the Corporate Duty Manager.

B. Correct.

C. Incorrect. The first part is plausible since neither train has all five of the hottest CET are above the temperature setpoint to enter EOP-FR-C.1 the candidate may misapply this information and determine implementation of EOP-FR-C.1 is not required, however this is incorrect because under these conditions the TC value should be considered 'greater than' if five functioning TCs are greater than the setpoint per the EOP-User's Guide. The second part is plausible since Shift Manager is responsible for making notifications the NRC Operations Duty Officer in accordance with AP-617, Reportability Determination And Notification the candidate may misapply this responsibility to the notifying the Corporate Duty Manager, however this is incorrect because the Site Duty manager is responsible to make the notification of plant status to the Corporate Duty Manager.

D. Incorrect: The first part is plausible since neither train has all five of the hottest CET are above the temperature setpoint to enter EOP-FR-C.1 the candidate may misapply this information and determine implementation of EOP-FR-C.1 is not required, however this is incorrect because for these conditions the TC value should be considered 'greater than' if five functioning TCs are greater than the setpoint per the EOP-User's Guide. The second part is correct.

000074 Inadequate Core Cooling / 4

074EA2.07; Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: The difference between a LOCA and inadequate core cooling, from trends and indicators
(CFR 43.5 / 45.13)

Importance Rating:	4.1 4.7
Technical Reference:	EOP User Guide, Section 6.0, Step 6.21, Pg 48, Rev. 49 EOP-CSFST, CSF-2, Pg 3, Rev.13 AD-OP-ALL-0101, Section 4.0, Step 4.3, Pg 5, Rev. 7
References to be provided:	None
Learning Objective:	PP-LP-2.17, Obj. 1
Question Origin:	New
Comments:	None
Tier/Group:	T1/G2
SRO Justification:	10 CFR Part 55 Content - 43(b)(1): Condition and limitations in the facility license. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know procedural knowledge of the required actions for reporting requirements when an undesirable event has occurred. The candidate's knowledge can be evaluated at the level of 10 CFR 55.43(b)(1) by ensuring that the additional knowledge of the reporting requirements are required to correctly answer the written test item. In this instance the candidate must know the information contained in the AD-OP-ALL-0101 for the Shift Manger reporting requirement within the corporation.

2018 NRC SRO 011/NEW/FUNDAMENTAL//EOP-E-0, ECA-1.2/NONE//004 G2.4.35/

Given the following plant conditions:
- The crew is implementing EOP-E-0, Reactor Trip or Safety Injection due to a LOCA
- An Auxiliary Operator is dispatched to locally unlock and turn on the breakers for the CSIP suction and discharge cross-connect valves

Which ONE of the following completes the statements below concerning this operation?

Turning on these breakers is directed in __(1)__ .

These local actions in EOP-E-0 are performed earlier rather than later in order to __(2)__ .

A. (1) Attachment 1, SI Emergency Alignment

(2) minimize operator dose

B. (1) Attachment 1, SI Emergency Alignment

(2) prevent ASI pump operation

C✓ (1) Attachment 2, Local Actions To Prepare For Cold Leg Recirculation

(2) minimize operator dose

D. (1) Attachment 2, Local Actions To Prepare For Cold Leg Recirculation

(2) prevent ASI pump operation

Plausibility and Answer Analysis

Reason answer is correct: In accordance with EOP-E-0 step 11 a local operator will be dispatched to unlock and turn on the breakers for the CSIP suction and discharge cross-connect valves using Attachment 2. The reason for performing these actions is to prepare for cold leg recirculation (title of the attachment). A note in this attachment states that it is perferred that these actions be performed prior to the switchover from the injection phase of SI to the recirculation phase to <u>minimize operator dose.</u>

A. Incorrect. The first part is plausible since a Safety Injection is in progress. There are multiple CSIP valves that are positioned using this Attachment but power to the valves are assumed to be ON. The second part is correct.

B. Incorrect. The first part is plausible since a Safety Injection is in progress. There are multiple CSIP valves that are positioned using this Attachment but power to the valves are assumed to be ON.
The second part is plausible if the candidate has a misconception that during the alignment to cold leg recirc when the CSIPs are stopped to realign the suction path to the CSIPs from the RWST to the RHR pump discharge the ASI pump will / may operate based on low RCP seal injection flow. By having the power applied the valves early would prevent unneccasary wait time and prevent the starting of the ASI pump while the CSIP is secured.

C. Correct.

D. Incorrect. The first part is correct. The purpose of turning on power to the valves is that when transfer to Cold Leg Recirc is being performed the suction and discharge valves must be shut to align suction to the CSIPs from the RHR pump discharge. The second part is plausible if the candidate has a misconception that during the alignment to cold leg recirc when the CSIPs are stopped to realign the suction path to the CSIPs from the RWST to the RHR pump discharge the ASI pump will / may operate based on low RCP seal injection flow. By having the power applied the valves early would prevent unneccasary wait time and prevent the starting of the ASI pump while the CSIP is secured

004 Chemical and Volume Control / 1/2

004G2.4.35; Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.8 4.0
Technical Reference:	EOP-E-0, Step 11, Pg 18, Rev. 7 EOP-E-0, Attachment 2, Pg 58, Rev. 7 EOP-ECA-1.2, Step 2, Pg 37, Rev. 2
References to be provided:	None
Learning Objective:	EOP-LP-3.03, Obj. 3.a
Question Origin:	New
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must have detailed knowledge of the content of the procedure versus general knowledge of the procedures overall mitigative strategy or purpose. The candidate's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item. In this instance the candidate must know the information contained in the EOP for the basis of performing local actions to energize CSIP suction and discharge cross connect valves.

2018 NRC SRO 012/NEW/C/A//TS 3.5.2/TS 3.5.2//006 G2.2.36/

Given the following plant conditions:

- March 14, 2018 the unit is operating at 100%

Subsequently the following sequence occurrs:

- At 0530 CSIP 'A' is INOPERABLE due to failed motor bearings
- At 0730 while aligning CSIP 'C' to replace CSIP 'A', the breaker racking mechanism for 1A-SA-7, CSIP 'C', is found to be broken, preventing the breaker from being racked in and closed

Which ONE of the following completes the statement below?

Demonstrate the operability of CSIP __(1)__ prior to __(2)__ .

(Reference Provided)

A. (1) 'C'

(2) March 14 at 0830 OR be in MODE 3 prior to 1430.

B. (1) 'A'

(2) March 14 at 1330 OR be in MODE 4 prior to 1930.

C✓ (1) 'C'

(2) March 17 at 0530 OR be in MODE 4 prior to 1730.

D. (1) 'A'

(2) March 17 at 0730 OR be in MODE 3 prior to 1330.

Plausibility and Answer Analysis

Reason answer is correct: Technical Specification 3.5.2 requires 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.

With one ECCS subsystem inoperable (the loss of CSIP A), 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. CSIP C is used to replace the CSIP on either train. Because the action clock to restore a second CSIP to operable status is from March 14 at 0530 when the CSIP A was initially declared inoperable the action must be completed by March 17 at 0530 or be placed in HOT SHUTDOWN by 1730.

A. Incorrect. The first part is correct. The scond part is plausible since the candidate may misapply the actions of Technical Specification 3.0.3 for the CSIPs, however this is incorrect because CSIP 'B' is still operable, therefore the actions of Technical Specification 3.5.2 are applicable.

B. Incorrect. The firs part is plausible since two CSIPs are not Operable at 0730 and Technical Specifications do not have an action for the CSIPs with both trains being inoperable. The second part is plausible since the candidate may misapply the actions of Technical Specification 3.5.2 and determine that shutdown potion of action a is now applicable. This is incorrect because CSIP 'B' is still operable, therefore the 72 Hour action of Technical Specification 3.5.2 are applicable.

C. Correct.

D. Incorrect. The firs part is plausible since two CSIPs are not Operable at 0730 and Technical Specifications do not have an action for the CSIPs with both trains being inoperable. The second part is plausible since the actions are correct for the time of discovery (0730) for the failure of the CSIP C breaker cubicle. This is incorrect because the action clock to restore a second CSIP to operable status is from 0530 when the CSIP A was initially declared inoperable.

006 Emergency Core Cooling / 2/3

006G2.2.36; Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.
(CFR: 41.10 / 43.2 / 45.13)

Importance Rating:	3.1 4.2
Technical Reference:	Technical Specification 3.5.2, Action a
References to be provided:	Technical Specification 3.5.2
Learning Objective:	SIS Lesson Plan, Obj 11.c
Question Origin:	New
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know the application of required Technical Specification actions that are greater than one hour. Requires knowledge of Technical Specification rules of application requirements that are not system knowledge. In this instance the candidate must know the application of Limiting Condition for Operation (LCO) requirements (LCO 3.5.2)

2018 NRC SRO 013/NEW/C/A//ALB-011, TS 3.8.3.1/NONE//013 A2.04/

Given the following plant conditions:

- The unit is operating at 100% power
- PT-952, Containment Pressure Channel III has failed high
- The actions of OWP-ESF-01 have been completed

Subsequently power has been lost to Instrument Bus SI and then re-energized from it's bypass source.

Which ONE of the following completes the statement below?

The combination of these conditions ___(1)___ have caused SI to actuate.

Technical Specifications 3.8.3.1 LCO __(2)__ applies.

(Reference provided)

A. (1) would

(2) action b

B. (1) would

(2) action c

C. (1) would not

(2) action b

D.✓ (1) would not

(2) action c

Plausibility and Answer Analysis

Reason answer is correct: There are 4 Containment pressure channels. PT-950, PT-951, PT-952 and PT-953. Each channel is powered from an Instrument Bus. SI, SII, SIII or SIV. All 4 channels are used for Containment Spray actuation signal development but ONLY 3 of them are used to develop SI logic. The 3 pressure channels for High Containment SI logic are PT-951, PT-952 and PT-953. These three channels powered from SII, SIII, and SIV.
A failure of Instrument Bus SI will NOT cause a SI bistable to actuate since there isn't a SI high pressure bistable on this bus. Therefore a 2 of 3 logic would not be made up and SI would NOT actuate.

AOP-024, Loss of Uniterruptible Power Supply, Section 3.1, Note prior to Step 14, Rev 57. NOTE: When aligned to the Manual Bypass, Tech Spec 3.8.3.1 ACTION c applies. (Modes 1, 2, 3, and 4)

A. Incorrect. The first part is plausible since the logic for SI activation is a 2 / 3 logic and a misconception could be that the logic is comprised of the first three channels and not channels 2, 3, and 4. The second part is plausible because this is the action required for aligning to alternate power supply and would yield action statement b, but the manual bypass supply was aligned which is action statement c.

B. Incorrect. The first part is plausible since the logic for SI activation is a 2 / 3 logic and a misconception could be that the logic is comprised of the first three channels and not channels 2, 3, and 4. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible because this is the action required for aligning to alternate power supply and would yield action statement b, but the manual bypass supply was aligned which is action statement c.

D. Correct.

013 Engineered Safety Features Actuation / 2

013A2.04; Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument bus

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.6 4.2
Technical Reference:	APP-ALB-011-5-1, Page 11, Rev. 8 Technical Specficiation 3.8.3.1.d, Action c AOP-024, Section 3.1, Note prior to step 14, Rev. 57
References to be provided:	Technical Specficiation 3.8.3.1
Learning Objective:	ESFAS Lesson Plan, Obj. 8.a 120V UPS Lesson Plan, Obj. 12
Question Origin:	New
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know the application of required Technical Specification actions that are greater than one hour. Requires knowledge of Technical Specification rules of application requirements that are not system knowledge. In this instance the candidate must know the application of Limiting Condition for Operation (LCO) requirements (LCO 3.8.3.1)

2018 NRC SRO 014/NEW/FUNDAMENTAL//ALB-016 AND TS BASES/NONE//059 A2.01/

Given the following plant conditions:
- The unit is at 50% power
- 'B' MFW pump is under clearance

Subsequently:
- 'A' MFW pump trips
- All SG levels are rapidly lowering

Which ONE of the following completes the statements below?

The MD AFW Pumps auto started due to the __(1)__ signal.

In accordance with Technical Specification 3.7.1.2, Plant Systems - Auxiliary Feedwater System, Bases the AFW system is required to mitigate a __(2)__ line break accident analyzed in FSAR Chapter 15.

A. (1) SG Low-Low Level

(2) Feedwater

B. (1) SG Low-Low Level

(2) Main Steam

C.✓ (1) loss of both MFW Pumps

(2) Feedwater

D. (1) loss of both MFW Pumps

(2) Main Steam

Plausibility and Answer Analysis

Reason answer is correct: A loss of both MFW pumps will cause an autostart signal to both of the MD AFW Pumps. ALB-016-2-2
Technical Specification bases section 3/4.7.1.2 states that The AFW System provides decay heat removal immediately following a station blackout event, and is required to mitigate the Loss of Normal Feedwater and Feedwater Line break accidents analyzed in FSAR Chapter 15.

A. Incorrect. The first part is plausible since on a loss of ALL Main Feedwater the SG levels will rapidly lower and when Low-Low Level (25%) on 1 / 3 SG's is reached an autostart signal to the MD AFW Pumps will be generated and both MD AFW Pumps will start. This is not correct because the starting signal would have occurred immediately following the trip of the last running MFW Pump. The second part is correct.

B. Incorrect. The first part is plausible since on a loss of ALL Main Feedwater the SG levels will rapidly lower and when Low-Low Level (25%) on 1 / 3 SG's is reached an autostart signal to the MD AFW Pumps will be generated and both MD AFW Pumps will start. This is not correct because the starting signal would have occurred immediately following the trip of the last running MFW Pump. The second part is plausible since a misconception could be made from the fact that the MD AFW Pumps receive a start signal from a SI which would be generated when SG pressure lowers to < 601 psig during a Main Steam line break accident therefore the system could have been designed to mitigate the MSL break event.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since a misconception could be made from the fact that the MD AFW Pumps receive a start signal from a SI which would be generated when SG pressure lowers to < 601 psig during a Main Steam line break accident therefore the system could have been designed to mitigate the MSL break event.

059 Main Feedwater / 4

059A2.01; Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feedwater actuation of AFW system

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.4 3.6
Technical Reference:	ALB-016-2-2, Rev. 22 Tech Spec Basis 3/4.7.1.2 AFW System, page B 3/4 7-1a
References to be provided:	None
Learning Objective:	EOP-LP-3.11, Obj. 4.c
Question Origin:	Bank
Comments:	HNP does not have steam driven Main Feedwater Pumps therefore cannot create a valid question for Loss of steam flow to MFW system for K/A 059A2.06, Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of steam flow to MFW system. Requested new K/A. Phonecon 6/14: HNP does not have steam driven Main Feedwater Pumps therefore cannot create a valid question for Loss of steam flow to MFW system, so selected a new K/A, keeping 059 and randomly selecting from the remaining items for this K/A: New K/A 059A2.04: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feeding a dry S/G Phonecon 10/23: The replacement K/A (059A2.04) provided will cause overlap with SRO Q#2 therefore HNP has requested another K/A replacement. So selected a new K/A, keeping 059 and randomly selecting from the remaining items for this K/A: New K/A 059A2.01: Ability to (a) predict the impacts of

the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feedwater actuation of AFW system

Tier/Group:	T2/G1
SRO Justification:	10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know the application of required Technical Specification actions that are greater than one hour. Requires knowledge of Technical Specification rules of application requirements that are not system knowledge. In this instance the candidate must know the bases for Limiting Condition for Operation (LCO) requirements (LCO 3.7.1.2)

2018 NRC SRO 015/NEW/C/A//OP-155, TS 3.8.1 BAS/NONE/EARLY/064 A2.16/

Given the following plant conditions:

- The unit is operating at 100% power with OST-1073, 1B-SB Emergency Diesel Generator Operability Test, in progress
- The 6.9kV Aux Buses are being supplied by the Start-up Transformers
- Emergency Diesel Generator 1B-SB, is loaded to 6300 KW while operating in parallel with the grid

Subsequently a loss of offsite power occurs.

Which ONE of the following completes the statements below?

Given these conditions, the EDG B-SB Output Breaker (126) ___(1)___ .

In accordance with Technical Specification 3.8.1, A.C. Sources - Operating, Bases action statements for inoperable A.C. sources or EDG's, states that the term "verify" used in "Verify required feature(s) are OPERABLE" means to ___(2)___ the OPERABILITY of required feature(s).

A. (1) must be manually opened to allow the sequencer to load

(2) administratively check by examining logs or other information to determine

B. (1) must be manually opened to allow the sequencer to load

(2) perform the Surveillance Requirement needed to demonstrate

C✓ (1) will automatically open and then reclose to allow the sequencer to load

(2) administratively check by examining logs or other information to determine

D. (1) will automatically open and then reclose to allow the sequencer to load

(2) perform the Surveillance Requirement needed to demonstrate

Plausibility and Answer Analysis

Reason answer is correct: OP-155 precaution and limitation 24 states, If a loss of offsite power (LOOP) occurs while an EDG is paralleled to the grid, breakers 125 and 126 should automatically trip open, which will leave the diesel running unloaded. Breaker 126 should then automatically reclose and the sequencer start to load. If breaker 126 fails to open, operator action is required to manually open the breaker.

The term "verify", as used in these ACTION statements means to administratively check by examining logs or other information to determine the OPERABILITY of required feature(s). It does not mean to perform the Surveillance Requirement needed to demonstrate the OPERABILITY of the required feature(s).

A. Incorrect. The first part is plausible since the diesel is in parallel with the grid the safety bus will remain engerized, therefore the 86UV relay would not actuate. OP-155, P&L 24 directs the candidate to manual open the Diesel Generator Output Breaker (Breaker 126) if it fails to automatically open. This direction may be misapplied by the candidate and they may determine that manual operation of Breaker 126 is required under these conditions. The second part is correct.

B. Incorrect. The first part is plausible since the diesel is in parallel with the grid the safety bus will remain engerized, therefore the 86UV relay would not actuate. OP-155, P&L 24 directs the candidate to manual open the Diesel Generator Output Breaker (Breaker 126) if it fails to automatically open. This direction may be misapplied by the candidate and they may determine that manual operation of Breaker 126 is required under these conditions. The second part is plausible since a candidate could have a misconception of the meaning based on what the basis section has written to define what "verify" does NOT mean.

C. Correct.

D. Incorrect. The first part is correct. The second part is plausible since a candidate could have a misconception of the meaning based on what the basis section has written to define what "verify" does NOT mean.

064 Emergency Diesel Generator / 6

064A2.16; Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of offsite power during full-load testing of ED/G

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.3 3.7
Technical Reference:	OP-155, Precaution and Limitation 24, Pg 9, Rev. 86 Technical Specifiaction Bases 3.8.1
References to be provided:	None
Learning Objective:	EDG -Lesson Plan, Obj. 13.a
Question Origin:	New
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know the application of required Technical Specification actions that are greater than one hour. Requires knowledge of Technical Specification rules of application requirements that are not system knowledge. In this instance the candidate must know the application of Limiting Condition for Operation (LCO) requirements (LCO 3.8.1)

2018 NRC SRO 016/BANK/C/A//FR-H.1/NONE//002 A2.04/

Given the following plant conditions:
- A Reactor trip from 100% power occurred due to loss of both MFW pumps
- All AFW pumps unavailable
- The crew is implementing EOP-FR-H.1, Response to Loss of Secondary Heat Sink
- RCPs are secured

The following conditions exist:
- PRZ pressure is 2170 psig and rising
- Core exit temperature is 620°F and rising
- Cold leg temperatures are 560°F and rising
- Containment pressure is 0.4 psig and rising
- Safety injection is NOT actuated
- SG WIDE RANGE levels:
  - 'A' = 26%
  - 'B' = 32%
  - 'C' = 24%

Which ONE of the following identifies the action the operating crew should do NEXT?

A.✓ Depressurize the RCS to 1900 to 1950 psig and block SI.

B. Depressurize 'B' Steam Generator to less than 500 psig.

C. Energize buses 1A1 and 1B1.

D. Initiate RCS Bleed and Feed.

Plausibility and Answer Analysis

Reason answer is correct: FR-H.1, Step 12 RNO With MFW pumps unavailable, step 12 is applied. Since SI has not actuated, direction is given to lower RCS pressure and block SI signals.

A. Correct.

B. Incorrect. Plausible because for the given conditions, this is a required action per step 13, however this is incorrect since it comes after the RCS depressurization and SI blocks in step 12.

C. Incorrect. Plausible because this action is performed in step 9, however this is incorrect since SI has not been actuated so step 9 is not applicable.

D. Incorrect. Plausible because Bleed and Feed foldout criteria for adverse values is met for the given information, however this is incorrect since the containment is not adverse.

002 Reactor Coolant / 2/4

002A2.04; Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of heat sinks

(CFR: 41.5 / 43.5 / 45.3 / 45.5)

Importance Rating:	4.3 4.6
Technical Reference:	EOP-FR-H.1, Step 12.e, Pg 24, Rev. 3
References to be provided:	None
Learning Objective:	EOP-LP-3.11, Obj 1a
Question Origin:	Bank
Comments:	Question based on 2nd part of the K/A since there is no discriminatory value to ask for for the prediction piece of the K/A.
Tier/Group:	T2/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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know knowledge of the content of the procedure versus knowledge of the procedure’s overall mitigative strategy or purpose. The applicant’s knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure’s content is required to correctly answer the written test item. In this instance the applicant must know the information contained in the EOP for actions to perform when the RCS heat sink is lost.

2018 NRC SRO 017/NEW/C/A//ALB-009, TS 3.3.1/TS 3.3.1/EARLY/011 G2.4.50/

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

At 0930 the following occurs:
- ALB-009-2-1, Pressurizer High Level Deviation And Heaters On, alarms
- ALB-009-4-2, Pressurizer High Level Alert, alarms
- PRZ Level indications are as follows:
  - LI-459 indicates 88% and rising
  - LI-460 indicates 56% and lowering
  - LI-461 indicates 55% and lowering

Which ONE of the following completes the statements below?

In accordance with APP-ALB-009-4-2 alarm response, the CRS will direct the OATC to ___(1)___ .

At 1020 the inoperable channel is placed into bypass for testing. 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)___ .

(Reference provided)

A. (1) lower Charging flow as necessary to return PRZ level to normal

(2) 1330

B. (1) lower Charging flow as necessary to return PRZ level to normal

(2) 1420

C. (1) reposition the PRZ level controller selector to unaffected PRZ level channels

(2) 1330

D.✓ (1) reposition the PRZ level controller selector to unaffected PRZ level channels

(2) 1420

Reason answer is correct: In accordance with the APP directions a changing Pressurizer level should be evaluated and if necessary, then manually adjust charging or letdown flow to return PRZ level to normal. Actual PRZ level is lowering so the operator directions should be to raise flow to return level to the normal level. If the alarm is due to a failed level instrument, then perform the following: Using the Pressurizer Level Controller Selector, select a position which places the two operable channels into service (for example: if 459 fails, place the selector switch to 460/461).

Technical Specifiaction, 3.3.1 for PRZ Water Level -High (Above P-7, 10% power) 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.

1020 + 4 hours = 1420 hours

A. Incorrect. The first part is plausible since LI-459 is at a level higher than the PRZ program level and is rising therefore the CRS could have directed the OATC to lower Charging flow to return PRZ level to normal. The second part is plausible since this is the time LT-459 became inoperable the candidate may have the misconception that the channel in bypass must be restored to operable four hours from the time the initial channel is declared inoperable, however this is incorrect because action 6.b allows an action clock for the bypassed channel which is independent of the action clock for the inoperable channel.

B. Incorrect. The first part is plausible since LI-459 is at a level higher than the PRZ program level and is rising therefore the CRS could have directed the OATC to lower Charging flow to return PRZ level to normal. The second part is correct.

C. Incorrect. The first part is correct. The second part is plausible since this is the time LT-459 became inoperable the candidate may have the misconception that the channel in bypass must be restored to operable four hours from the time the initial channel is declared inoperable, however this is incorrect because action 6.b allows an action clock for the bypassed channel which is independent of the action clock for the inoperable channel

D. Correct.

011 Pressurizer Level Control / 2

011G2.4.50; Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

(CFR: 41.10 / 43.5 / 45.3)

Importance Rating:	4.2 4.0
Technical Reference:	APP-ALB-009, Window 4-2, Pg 16, Rev. 18 Technical Specifiaction 3.3.1, PZR Water Level RPS requirements, Action 6
References to be provided:	Technical Specifiaction 3.3.1
Learning Objective:	PRZLC Lesson Plan, Obj. 7.f
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2
SRO Justification:	10 CFR Part 55 Content - 43(b)(5) and (2): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Facility oeprating limitations in the technical specifications and their bases. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose. The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item. In this instance the candidate must know the information contained in the APP for actions to restore or maintain PRZ level control to automatic. This question also meets the SRO level of knowledge by ensuring that the candidate must know the application of required Technical Specification actions that are greater than one hour. Requires knowledge of Technical Specification rules of application requirements that are not system knowledge. In this instance the candidate must know the application of Limiting Condition for Operation (LCO) requirements (LCO 3.3.1)

2018 NRC SRO 018/NEW/FUNDAMENTAL//AOP-042/NONE//041 A2.02/

The unit is operating at 99% power when 1MS-101, Atmospheric Steam Dump PCV, indicates that it is cycling open on Status Light Box 1

Which ONE of the following completes the statement below?

The cycling open steam dump is capable of passing approximately ___(1)___ percent of total steam flow.

In accordance with AOP-042, Secondary Steam Leak/Efficiency Loss, the CRS will ___(2)___ .

Procedure Title:
GP-006, Normal Plant Shutdown From Power Operation To Hot Standby (Mode 1 To Mode 3)

A. (1) five

(2) GO TO GP-006 and perform a plant shutdown then shut the MSIV's

B.✓ (1) five

(2) remain in AOP-042 and attempt to isolate 1MS-101

C. (1) seven

(2) GO TO GP-006 and perform a plant shutdown then shut the MSIV's

D. (1) seven

(2) remain in AOP-042 and attempt to isolate 1MS-101

Plausibility and Answer Analysis

Reason answer is correct: The total steam flow capacity of one Steam dump is 5%. AOP-042 will initially direct the crew to reduce steam demand to return Reactor power below 100% if the plant remains within the limits of the Reactor Protection system and the RAB and Turbine buildings remain safe for entry due to the size of the steam leak. Once the plant is stabilized AOP-042 will direct the candidate to dispatch personnel to identify the leak location and attempt to isolate the leak source.

A. Incorrect. The first part is correct. The second part is plausible since AOP-042 directs the use of GP-006 to remove the unit from service if the size of the steam leak is less than 10% of rated stem flow, however this is incorrect because the atmospheric steam dump are capable of being isolated while online and therefore the candidate will remain in AOP-042 to isolate the failed steam dump prior to entering GP-006.

B. Correct.

C. Incorrect. The first part is plausible since Main Steam Safeties are part of the Main Steam relief flowpath similar to the Atmospheric Steam dumps and the candidate may misapply this information to determine the Atmospheric Steam Dumps flow rate is the same as the Main Steam Safeties, however this is incorrect because the Main Steam Safeties are designed to relieve 100% steam flow while the Steam Dumps only relieve 70% steam flow and therefore the total steam flow capacity of one Steam dump is 5% vice 7% for a Main Steam Safety. The second part is plausible since AOP-042 directs the use of GP-006 to remove the unit from service if the size of the steam leak is less than 10% of rated stem flow, however this is incorrect because the atmospheric steam dump are capable of being isolated while online and therefore the candidate will remain in AOP-042 to isolate the failed steam dump prior to entering GP-006.

D. Incorrect The first part is plausible since Main Steam Safeties are part of the Main Steam relief flowpath similar to the Atmospheric Steam dumps and the candidate may misapply this information to determine the Atmospheric Steam Dumps flow rate is the same as the Main Steam Safeties, however this is incorrect because the Main Steam Safeties are designed to relieve 100% steam flow while the Steam Dumps only relieve 70% steam flow and therefore the total steam flow capacity of one Steam dump is 5% vice 7% for a Main Steam Safety The second part is correct.

041 Steam Dump/Turbine Bypass Control / 4

041A2.02; Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam valve stuck open

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Importance Rating:	3.6 3.9
Technical Reference:	AOP-042, Section 3.0, Step 18, Pg 9, Rev 6
References to be provided:	None
Learning Objective:	AOP-LP-3.42, Obj. 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T2/G2
SRO Justification:	10 CFR Part 55 Content - 43(b)(5) and (2): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Facility oeprating limitations in the technical specifications and their bases. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know knowledge of the content of the procedure versus knowledge of the procedure’s overall mitigative strategy or purpose. The applicant’s knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure’s content is required to correctly answer the written test item. In this instance the candidate must know the information contained in the AOP for actions to isolate the source of the steam leak.

2018 NRC SRO 019/BANK/FUNDAMENTAL//FHP-400/NONE/2011 NRC SRO 19/G2.1.35/

Given the following plant conditions:
- Refueling activites are in progress.
- Fuel movement has stopped due to a problem with the Manipulator crane.
- The Manipulator crane operator desires to operate TS-5, TROLLEY INTERLOCK BYPASS switch.

In accordance with FHP-400, Manipulator Crane Checkouts, which ONE of the following describes the permission AND concurrence required to bypass the trolley interlock?

A. The CRS must approve with the concurrence of the SM.

B.✓ The SRO-Fuel Handling must approve with the concurrence of the SM.

C. The SRO-Fuel Handling must approve with the concurrence of the CRS.

D. The SM must approve with the concurrence of the Maintenance Outage Manager of Refueling.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with FHP-400, Manipulator Crane checkouts in order to use the Manipulator Interlock Bypass Switches the crane operator must obtain the SRO Fuel Handling and Shift Manager concurrence for operation to be performed.

A. Incorrect. Plausible since concurrence is required from SM per FHP-400 but the SRO-Fuel Handling must approve the operation of the bypass switch not the CRS.

B. Correct.

C. Incorrect. Plausible since SRO-Fuel Handling can give permission, but for bypassing interlocks the SM must concur.

D. Incorrect. Plausible since the SM needs to give concurrence for bypassing the interlock per FHP-400 and the Maintenance Outage Manager is informed if troubleshooting or problems are occurring during refueling activities.

2.1 Conduct of Operations

G2.1.35; Knowledge of the fuel-handling responsibilities of SROs.

(CFR: 41.10 / 43.7)

Importance Rating:	2.2 3.9
Technical Reference:	FHP-400, Attachment 2, Step 3.0.1, Pg 67, Rev. 3
References to be provided:	None
Learning Objective:	PP-LP-2.8, Obj. 1
Question Origin:	Bank
Comments:	None
Tier/Group:	T3
SRO Justification:	10 CFR Part 55 Content - 43(b)(7): Fuel-Handling Facilities and Procedures. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know knowledge of Fuel-Handling Facilities and Procedures. The candidate’s knowledge can be evaluated at the level of 10 CFR 55.43(b)(7) by ensuring that the additional knowledge of the Fuel Handling Facilities and Procedure’s content is required to correctly answer the written test item. In this instance the candidate must know the information contained in the FHP-400 for the approval requirements to bypass a fuel handling bridge interlock.

2018 NRC SRO 020/BANK/C/A//GP-004/NONE//G2.1.43/

Given the following plant conditions:
- A Reactor startup is in progress per GP-004, Reactor Startup (Mode 3 To Mode 2)
- Current core age is 200 EFPD
- The RCS boron concentration used in the Estimated Critical Condition (ECC) calculation was 1100 ppm
- Actual RCS boron concentration is 1000 ppm
- Criticality was NOT achieved within 500 pcm of the Estimated Critical Condition due to the current RCS boron concentration

Which ONE of the following describes (1) the effect of this error on the ECC and (2) the actions the CRS will direct the OATC to take in accordance with GP-004?

A. (1) The actual critical rod position will be lower than estimated.

(2) Trip the Reactor, enter EOP-E-0, Reactor Trip And Safety Injection, and initiate emergency boration until required boron concentration is reached.

B✓ (1) The actual critical rod position will be lower than estimated.

(2) Insert Control and Shutdown Banks, open the Reactor Trip and Bypass Breakers, and perform a new GP-004 prior to the next startup.

C. (1) The actual critical rod position will be higher than estimated.

(2) Trip the Reactor, enter EOP-E-0, Reactor Trip And Safety Injection, and initiate emergency boration until required boron concentration is reached.

D. (1) The actual critical rod position will be higher than estimated.

(2) Insert Control and Shutdown Banks, open the Reactor Trip and Bypass Breakers, and perform a new GP-004 prior to the next startup.

Plausibility and Answer Analysis

Reason answer is correct: With actual boron less than estimated, the change in reactivity required to achieve criticality is lower, therefore the magnitude of change by the control rods is less and the actual control rod position will be lower than the position calculated during the Estimated Critical Condition (ECC). Because criticality was not achieved within the +/- 500 pcm band for the ECC GP-004 section 6.0 step 26 directs the operator to perform the following:

Fully insert all Control and Shutdown Banks per OP-104
Open Reactor Trip AND Bypass Breakers
Perform a new GP-004 for any subsequent startup

A. Incorrect. The first part is correct. The second part is plausible since GP-004 directs the Reactor Trip breakers to be opened for these conditions therefore candidate may misinterpret the direction and enter EOP-E-0 which is the normal post trip procedure, however this is incorrect because the entry conditions of E-0 are symptoms that require a reactor trip based on applicable RPS signals.

B. Correct

C. Incorrect. The first part is plausible since the core age is middle of cycle the candidate may misapply the concept of the end of cycle boron concentration vs. actual boron concentration and determine that actual boron is higher than the amount of boron required at the end of cycle and therefore the actual rod position will be higher, however this is incorrect because regardless of the core age with the ECC boron concentration higher than the actual boron concentration the reactor will become critical at a rod position that is lower that the estimated position. The second part is plausible since GP-004 directs the Reactor Trip breakers to be opened for these conditions therefore candidate may misinterpret the direction and enter EOP-E-0 which is the normal post trip procedure, however this is incorrect because the entry conditions of E-0 are symptoms that require a reactor trip based on applicable RPS signals.

D. Incorrect. The first part is plausible since the core age is middle of cycle the candidate may misapply the concept of the end of cycle boron concentration vs. actual boron concentration and determine that actual boron is higher than the amount of boron required at the end of cycle and therefore the actual rod position will be higher, however this is incorrect because regardless of the core age with the ECC boron concentration higher than the actual boron concentration the reactor will become critical at a rod position that is lower that the estimated position. The second part is correct.

2.1 Conduct of Operations

G2.1.43; Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

(CFR: 41.10 / 43.6 / 45.6)

Importance Rating:	4.1 4.3
Technical Reference:	GP-004, Step 6.0.26, Pg 20, Rev. 64
References to be provided:	None
Learning Objective:	GP-LP-3.4, Obj. 3
Question Origin:	Bank
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose. The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item. In this instance the candidate must know the information contained in the GP-004 for actions to achieving criticality outside of the +/- 500 pcm limits.

2018 NRC SRO 021/BANK/FUNDAMENTAL//AD-OP-ALL-0105/NONE//G2.2.19/

Which ONE of the following completes the statements below concerning the Operability Determination Process in accordance with AD-OP-ALL-0105, Operability Determinations And Functionality Assessments?

The SM will ensure an Immediate Functionality Assessment (IFA) is performed ___(1)___.

The IFA should be completed within ___(2)___ of identification.

A✓ (1) during the screening of Work Requests and CRs

(2) 12 hours

B. (1) during the screening of Work Requests and CRs

(2) 3 days

C. (1) by groups other than Operations (e.g. Engineering)

(2) 12 hours

D. (1) by groups other than Operations (e.g. Engineering)

(2) 3 days

Plausibility and Answer Analysis

Reason answer is correct: AD-OP-ALL-0105 Section 5.6.2 Expectations:

4. Immediate Functionality Assessments (IFAs) will be performed by the SM or designee during the screening of Work Requests (WRs) and CRs.
a. The IFA shall adhere to the template in Attachment 7, Immediate Functionality Assessment Template.
b. The IFA should be completed within 12 hours of identification.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since Prompt Functionality Assessments (PFAs) are expected to be completed within three days

C. Incorrect. The first part is plausible plausible since a PFAs are performed by groups other than Operations (typically Engineering) when requested by the SM. The second part is correct.

D. Incorrect. The first part is plausible since a PFAs are performed by groups other than Operations (typically Engineering) when requested by the SM. The second part is plausible since PFAs are expected to be completed within three days

2.2 Equipment Control

G2.2.19; Knowledge of maintenance work order requirements.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	2.3 3.4
Technical Reference:	AD-OP-ALL-0105, Section 5.6.2, Pg 28, Rev. 3
References to be provided:	None
Learning Objective:	PP-LP-2.19, Obj. 4
Question Origin:	Bank
Comments:	None
Tier/Group:	T3
SRO Justification:	10 CFR Part 55 Content - 43(b)(1): Condition and limitations in the facility license. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know procedural knowledge of the required actions to process a maintenance work order request. The candidate's knowledge can be evaluated at the level of 10 CFR 55.43(b)(1) by ensuring that the additional knowledge of Operational Determination process is required to correctly answer the written test item. In this instance the candidate must know the information contained in the AD-OP-ALL-0105 for the Shift Manger expectations for completing the screening of a work request.

2018 NRC SRO 022/BANK/C/A//TS3.7.8 PLP-106 ATT4/TS3.7.8 PLP-106 ATT4//G2.2.22/

Given the following plant conditions:

- The unit is operating at 100% power
- Snubber AF-H-0213, attached to the 'A' MDAFW pump discharge piping, failed a visual inspection
- Maintenance has removed the snubber and initiated repairs

Which ONE of the following describes the operational impact, if any?

(Reference Provided)

A. Operability of the AFW system is unaffected as long as the affected piping is isolated and depressurized.

B✓ The snubber must be repaired and re-installed within 72 hours or declare the affected portion of the AFW system inoperable.

C. The affected portion of the AFW system should immediately be declared inoperable and the snubber must be repaired and re-installed within 72 hours.

D. The snubber must be repaired and re-installed within 72 hours or the affected AFW system piping must be isolated, removed from service, and depressurized within the next 72 hours.

Plausibility and Answer Analysis

Reason answer is correct: Per PLP-106 Attachment 4 Section 4.1.2.b: If a snubber is declared inoperable based on visual inspection, the 72 hour clock may be stopped with either: (1) a successful functional test and reinstallation of the snubber, or (2) installation of a repaired or replacement snubber and verification that the "as left" condition of the pipe is acceptable and meets the design criteria of the system (PCR 4839); this is documented on the Work Order (WO). If the snubber fails the functional test, the clock continues to run, unless option (2) is completed.

A. Incorrect. Plausible since PLP-106 Section 4.1.1 allows these actions for non-essential portions of an affected system; however, this is incorrect since the AFW discharge piping is an essential component.

B. Correct.

C. Incorrect. Plausble since the snubber must be repaired or replaced with 72 hours; however, this is incorrect since the affected system does not have to be declared inoperable until the end of this time period.

D. Incorrect. Plausible since these actions apply to non-Tech Spec "Safety-Significant" snubbers only; however, this is incorrect since AFW is a Tech Spec covered system.

2.2 Equipment Control

G2.2.22; Knowledge of limiting conditions for operations and safety limits.

(CFR: 41.5 / 43.2 / 45.2)

Importance Rating:	4.0 4.7
Technical Reference:	Technical Specification 3.7.8 PLP-106, Attachment 4, Pg 26, Rev. 64
References to be provided:	Tech Spec 3.7.8 PLP-106 Attachment 4, Rev. 64
Learning Objective:	TS-LP-3.00, Obj. 7
Question Origin:	Bank
Comments:	None
SRO Justification:	10 CFR Part 55 Content - 43(b)(2): Facility operating limitations in the Technical Specifications and their bases. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must know the application of required Technical Specification actions that are greater than one hour. Requires knowledge of Technical Specification rules of application requirements that are not system knowledge. In this instance the candidate must know the application of Limiting Condition for Operation (LCO) requirements (LCO 3.7.8)

2018 NRC SRO 023/NEW/C/A//EP-EAL, EAL MATRIX/EAL MATRIX//G2.3.15/

Given the following plant conditions:

- The unit was operating at 100% power with an RCS leak was in progress
- 'A' CSIP in sevice, 'B' CSIP in Standby
- The Reactor was tripped and SI manually initiated due to the size of the leak exceeding VCT makeup capability

The following alarms and indications were observed at 0810:

- GFFD is in alarm and has risen by 85,000 CPM over the last 20 minutes
- RM-21AV-3509-1SA, Plant Vent Stack 1 WRGM Effluent, is alarming at 1.24E+6 uCi/sec and rising
- RM-1RR-3599C, Charging Pump 1C Room Area Radiation monitor, is alarming at 1100 times normal and rising

The time is now 0826.

Which ONE of the following identifies the EAL identifier and HIGHEST classification for this event?

(Reference Provided)

A. SU4.2 (Unusual Event)

B. RA3.2 (Alert)

C✓ RA1.1 (Alert)

D. FS1.1 (Site Area Emergency)

Plausibility and Answer Analysis

Reason answer is correct: Plant Vent Stack 1 WRGM Effluent is alarming at 1.24E+6 uCi/sec and rising. An Alert will be declared based on EAL identifer RA1.1, reading on any Table R-1 effluent radiation monitor > column "ALERT" (1.05E+6) for ≥ 15 minutes.

A. Incorrect. Plausible since the condition for this EAL has been met (valid GFFD high alarm); however, this is incorrect as it is not the highest classification.

B. Incorrect. Plausible since the condition for this EAL has been met (unplanned event results in radiation levels that prohitbit or impede access to any Table R-3/H-2 rooms or area); however, this is incorrect with the exception of the 1C Charging pump is not normally aligned and therefore EAL does not apply. Note 5 applies and no classification is required if the equipment is not in service before the event occurred.

C. Correct.

D. Incorrect. Plausible since the note for escalation of EAL idenitifer SU4.2 refers the candidate to the fission product barrier matrix due to fuel clad failures which would result in the loss or potentially loss of any two barriers; however this is incorrect because the RCS barrier is the only barrier loss or potentially loss based on the information given to the candidate.

2.3 Radiation Control

G2.3.15; Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

(CFR: 41.12 / 43.4 / 45.9)

Importance Rating:	2.9 3.1
Technical Reference:	EAL Matrix, Rev. 20 EP-EAL, Attachment 1, Pg 36, Rev. 17
References to be provided:	EAL Matrix
Learning Objective:	EP-LP-2.00, Obj. 10
Question Origin:	New
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must analysis and interpret radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures. Requires knowledge that is not RO knowledge of radiological safety principles. In this instance the candidate must know the application of various radiation monitors and how they apply to the EAL Matrix.

2018 NRC SRO 024/NEW/FUNDAMENTAL//OMM-027/NONE/EARLY/G2.4.11/

Which ONE of the following completes the statement below dealing with the protocol for variance from procedure during AOP Implementation in accordance with OMM-027, AOP User's Guide?

The ___(1)___ authorizes any variance from the AOPs. Prior to authorizing a procedure variance, the SRO should obtain concurrence from ___(2)___ , if time permits.

A.✓ (1) CRS

(2) a second SRO

B. (1) CRS

(2) other key personnel (license not needed), e.g. PGM

C. (1) SM (Only)

(2) a second SRO

D. (1) SM (Only)

(2) other key personnel (license not needed), e.g. PGM

Plausibility and Answer Analysis

Reason answer is correct: In accordance with OMM-027, section 5.1 step 10 for protocol for variance from procedure during AOP implementation the CRS (SRO) authorizes any variance from the AOPs. Prior to authorizing a procedure variance, the SRO should obtain concurrence from a second SRO, if time permits and if a second SRO is immediately available.

A 'variance' is defined as taking action during an event that is not consistent with the instructions or step sequence of an approved AOP in order to minimize personnel injury or damage to the facility or to protect public health and safety.

A 'departure' is a special and extreme case of 'variance' which also changes the intent of the procedure or departs from a license condition or Technical Specification. Guidance for 'departure' from an AOP, including departure from a license condition or Technical Specification is provided in AD-HU-ALL-0004.

A. Correct.

B. Incorrect. The first part is correct. The second part is plausible since based on event complexity and time available the SRO in command should obtain input / review from other key personnel such as the PGM (Plant General Manager) to determine if the decision was either a variance in the procedure or a departure from the procedure.

C. Incorrect. The first part is plausible since the responsibility of the Shift Manager is to ensure that plant operations are conducted in accordance with the requirements of the plant operating license, Technical Specifications, plant procedures, and company requirements. The second part is correct.

D. Incorrect. The first part is plausible since the responsibility of the Shift Manager is to ensure that plant operations are conducted in accordance with the requirements of the plant operating license, Technical Specifications, plant procedures, and company requirements. The second part is plausible since based on event complexity and time available the SRO in command should obtain input / review from other key personnel such as the PGM (Plant General Manager) to determine if the decision was either a variance in the procedure or a departure from the procedure.

2.4 Emergency Procedures / Plan

G2.4.11; Knowledge of abnormal condition procedures.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	4.0 4.2
Technical Reference:	OMM-027, Section 5.1.10, Pg 12, Rev. 6
References to be provided:	None
Learning Objective:	AOP-LP-3.0, Obj. 8
Question Origin:	New
Comments:	None
Tier/Group:	T3
SRO Justification:	10 CFR Part 55 Content - 43(b)(3): Facility Licensee Procedures Required To Obtain Authority for Design and Operating Changes in the Facility. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must have detailed knowledge of processes for changing the plant or plant procedures. The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(2) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item. In this instance the candidate must know the process for a a procedure variance in accordance with OMM-027.

2018 NRC SRO 025/BANK/C/A//USERS GUIDE, ECA-1.1/NONE/2008 NRC SRO 10/G2.4.23/

Given the following plant conditions:
- A LOCA has occurred
- The crew is performing actions of EOP-ECA-1.1, Loss of Emergency Coolant Recirculation, based on plant conditions upon transitioning from EOP-E-1, Loss of Reactor or Secondary Coolant

Current Conditions:
- RWST level is 2.9%
- RCS Integrity CSF Status Tree (CSFST) indicates steady ORANGE

Which ONE of the following describes (1) the action required and (2) the basis for prioritizing the use of the required procedure?

A✓ (1) Stop all pumps taking suction from the RWST;

(2) Go to FR-P.1, Response to Imminent Pressurized Thermal Shock, because Optimal Recovery procedures in progress are suspended if any CSFST is ORANGE.

B. (1) Stop all pumps taking suction from the RWST;

(2) Remain in EOP-ECA-1.1 because actions in EOP-ECA-1.1 are NOT limited by cooldown rate and an Orange condition on Integrity is expected.

C. (1) Align the RHR pump suction to Containment Recirc Sump;

(2) Go to FR-P.1, Response to Imminent Pressurized Thermal Shock, because Optimal Recovery procedures in progress are suspended if any CSFST is ORANGE.

D. (1) Align the RHR pump suction to Containment Recirc Sump;

(2) Remain in EOP-ECA-1.1 because actions in EOP-ECA-1.1 are NOT limited by cooldown rate and an Orange condition on Integrity is expected.

Plausibility and Answer Analysis

Reason answer is correct: In accordance with EOP-ECA-1.1 Foldout criteria for loss of suction. If RWST level decreases to 3% (Empty alarm/ALB-004-2-5), THEN secure all pumps taking suction only from the RWST. An ORANGE condition for RCS Integrity requires implementation of EOP-FR-P.1 based on the EOP rules of usage as described in the EOP Users Guide*.*

A. Correct.

B. Incorrect. Action is correct based on RWST level, but procedure usage is incorrect. Procedure usage is plausible since EOP Network does often direct user not to implement an FR if their actions are expected to cause a Red or an Orange (for instance caution in EOP-FR-H.1 provides this direction) but no such caution exists for this condition in EOP-ECA-1.1.

C. Incorrect. Action is plausible but would have been performed at 23.4% not at RWST empty. At 3% in the RWST the required action is to stop all pumps taking a suction only from the RWST. Procedure usage is correct.

D. Incorrect. Incorrect action and incorrect procedure usage. Action is plausible but would have been performed at 23.4% not at RWST empty. At 3% in the RWST the required action is to stop all pumps taking a suction only from the RWST. Procedure usage is plausible since EOP Network does often direct user not to implement an FR if their actions are expected to cause a Red or an Orange (for instance caution in EOP-FR-H.1 provides this direction) but no such caution exists for this condition in EOP-ECA-1.1.

2.4 Emergency Procedures / Plan

G2.4.23; Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

(CFR: 41.10 / 43.5 / 45.13)

Importance Rating:	3.4 4.4
Technical Reference:	EOP-ECA-1.1, Foldout criteria, Pg 3, Rev. 27 EOP Users Guide, Section 5.2.3, Pg 23, Rev. 49
References to be provided:	None
Learning Objective:	LP-EOP-2.3/3.3, Obj. 5.a
Question Origin:	Bank
Comments:	None
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. Per ES-401 Attachment 2, Clarification Guidance for SRO-only Questions, this question meets the SRO level of knowledge by ensuring that the candidate must have detailed knowledge of the content of the procedure versus general knowledge of the procedure's overall mitigative strategy or purpose. The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item.

HARRIS 2018 NRC SCENARIO 1

Facility:	Harris Nuclear Plant	Scenario No.: 1	Op Test No.: 05000400/2018301
Examiners:		Operators:	SRO:
			RO:
			BOP:
Initial Conditions:	IC-28, MOL, 42% power		

- ‘B’ MD AFW Pump is under clearance for pump packing replacement
- ‘B’ DEH Oil Pump is under clearance for motor repairs
- 1CS-9 is under clearance for solenoid replacement

Turnover:	A plant shutdown is required due to problems encountered during the repairs on the ‘B’ MDAFW Pump. Repairs have not been able to be completed and the LCO is expiring. The plant is operating at ~42% power in MOL. ‘A’ Heater Drain Pump was secured just prior to shift turnover. When turnover is complete a power reduction at 4 MW/min must be started to support being in Mode 3 within the next 6 hours.
Critical Task:	• Open 1MS-70 or 1MS-72 to establish a minimum of 200 KPPH AFW flow to the Steam Generators prior to 2 of 3 SG’s Wide Range Levels < 15% • Start the “B” Emergency Diesel Generator to restore AC Emergency bus power prior to initiation of ELAP mitigating actions in accordance with EOP-ECA-0.0

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R – RO/SRO N – BOP/SRO	Plant Shutdown (GP-006) Secure ‘B’ Heater Drain Pump
2 #	RCS10	C – RO/SRO TS - SRO	Reactor Vessel Flange Leak (ALB-010 and/or AOP-016)
3 #	gen15, 17, 18	C – BOP/SRO	Main Generator Automatic Voltage Regulator Failure (ALB-022) (Voltage Regulator to Manual - Operate plant controls in Manual)
4	cws01a	C – BOP/SRO	Trip of the ‘A’ Circ Water Pump and discharge valve, 1CW-10 fails to automatically shut (AOP-012)
5	cvc05A	C - RO/SRO TS-SRO	CSIP Trip (AOP-018) (FK-122.1 to Manual - Operate plant controls in Manual)
6 #	eps01	M – ALL	Loss of Offsite Power, Reactor Trip (EOP-E-0)
7 #	dsg38 zdg2032b	C – BOP/SRO C – RO/SRO	Loss of ALL AC power (EOP-ECA-0.0)
8 #	z1974tdi z1975tdi	C – BOP/SRO	1MS-70 and 1MS-72 fail to auto open (Loss of all AFW until operator opens 1MS-70 or 72)

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

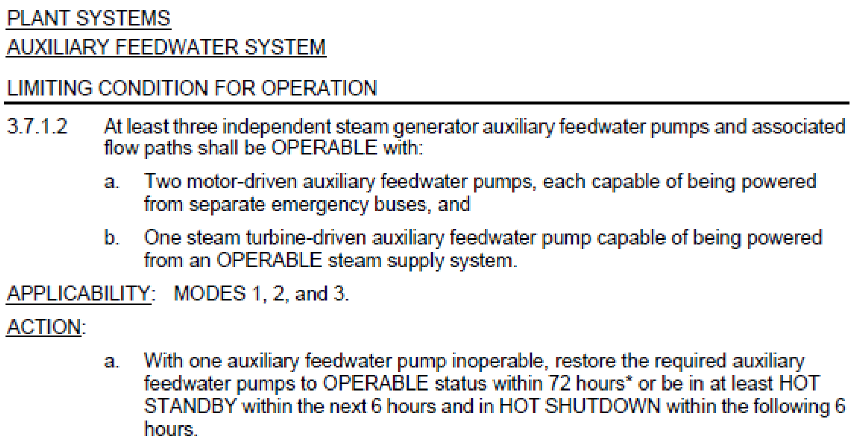
\# Event or Major Transient NOT used on the previous 2 NRC initial licensing operating tests

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 1

A plant shutdown is required due to problems encountered during the repairs on the ‘B’ MDAFW Pump. Repairs will not be able to be completed prior to the LCO expiring. The plant is operating at ~42% power in MOL. The ‘A’ Heater Drain pump has been secured prior to turnover. When turnover is complete the crew will secure the ‘B’ Heater Drain pump then continue a power reduction at 4 MW/min must be started to support being in Mode 3 within the next 6 hours. All required notifications have been made to individuals concerning the reason for the shutdown.

The following equipment is under clearance:

- ‘B’ MDAFW Pump is under clearance for pump packing repairs. The pump has been inoperable for 62 hours and will NOT be restored to operable status within the next 10 hours. Tech Spec 3.7.1.2 LCO Action **a** and Tech Spec 3.3.3.5.b Action **c** applies. 72 hour LCO or HSB within the next 60 hours, HSD following 6 hours.



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:

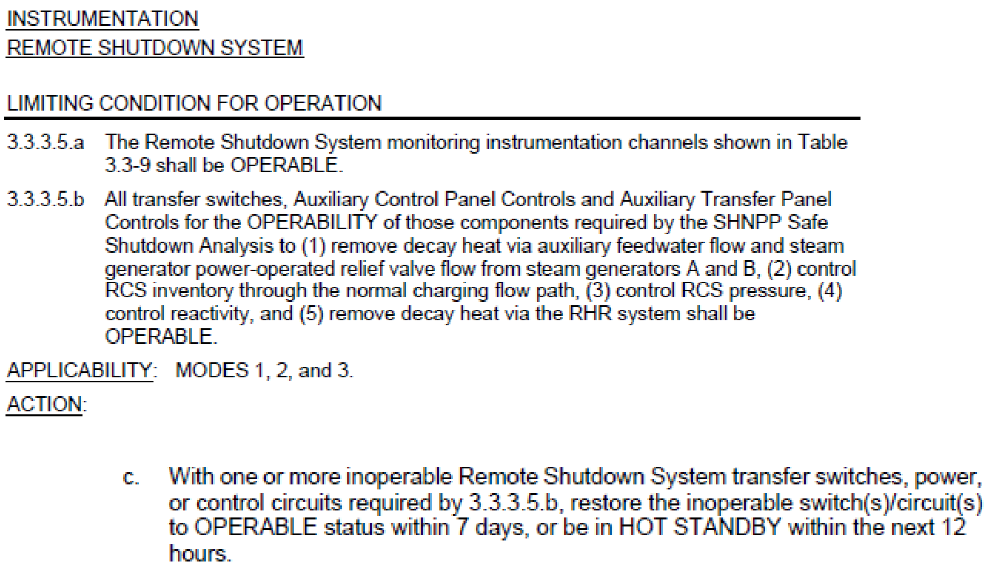
a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency buses, and

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

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

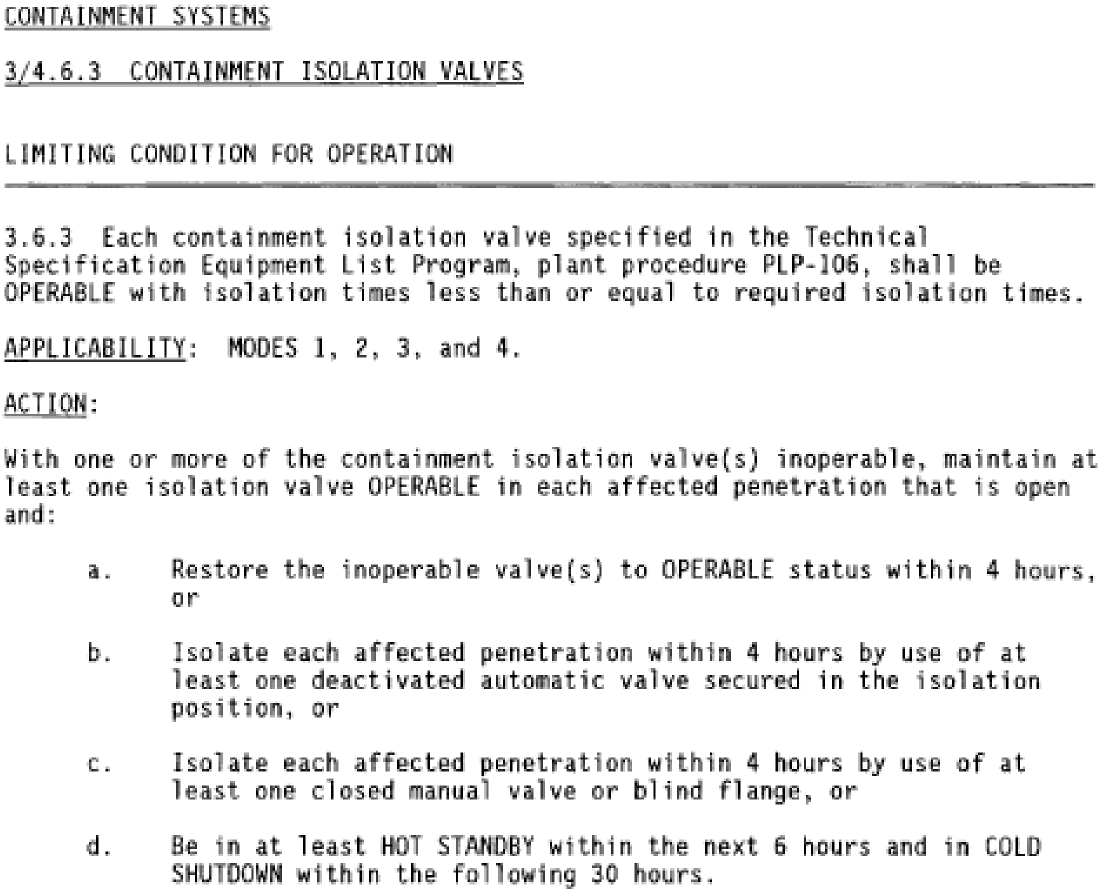
c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

SCENARIO SUMMARY: 2018 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.

- Letdown Orifice Isolation Valve 1CS-9 is under clearance for solenoid replacement. Tech Spec 3.6.3 LCO Action **b** applies. OWP-CS-09 has been completed.



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: 2018 NRC EXAM SCENARIO 1 continued

Event 1: Plant Shutdown (GP-006). Turnover takes place with the unit at 42% Reactor power. The BOP will stop 'B' Heater Drain Pumps in accordance with OP-136, Section 7.1. After securing the Heater Drain Pump the crew will continue the power reduction in accordance with GP-006, Normal Plant Shutdown From Power Operation To Hot Standby (Mode 1 To Mode 3). The crew will be given credit for a reactivity manipulation during the down power. 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 the crew has demonstrated that they have control of the plant during a shutdown (at Evaluator discretion) event 2 can be inserted.

Event 2: Reactor Vessel Flange leak of ~ 15 gpm. Annunciator ALB-10-5-5, Reactor vessel flange leakoff high temp will alarm when MCB temperature indicator TI-401 reaches 140°F. The crew should notice Pressurizer level slowly decreasing and an increase in Charging flow. The SRO should direct the BOP to place Turbine controls in HOLD to stop the downpower and allow the crew to focus on the event in progress. Annunciator response 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 could 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 the following Tech Spec for the time the leak exceeded 10 gpm (briefly):

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.

Event 3: Main Generator Automatic Voltage Regulator Failure (APP-ALB-022). A progressive failure of the Auto Voltage Regulator will occur. ALB-22 Window 5-4, GEN EXCITATION AUTO FAULT IN CH 1 OR CH 2 will alarm. Channel 1 Auto has failed this will have no impact on AVR because the unit switches to Channel 2. After a 4 minute delay, oscillations will be observed in the Main Generator MVAR output and SWYD voltage. The crew should use AD-OP-ALL-1000 guidance to ***take manual control of the AVR***. ***(Competency - Operate Plant Component Controls In Manual Control)***

If the crew takes no action the AVR will fail to manual 6 minutes after the oscillations begin. Once in Manual on the AVR, the crew should restore MVAR output and notify the dispatcher. AVR manual MVAR limits are 75 to 175 MVARs, (normal operation limits). This failure will also require the crew to notify the Load Dispatcher that the voltage regulator is in Manual control within 30 minutes. The SRO will complete OMM-001 Attachment 5.

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 1 continued

Event 4: Trip of the ‘A’ Circ Water Pump and discharge valve 1CW-10 failure to automatically shut. The crew should identify the trip of the ‘A’ Circ Water pump from annunciator ALB-021-4-4 and respond using the APP. The trip of the Circ Water pump is entry conditions for AOP-012, Partial Loss Of Condenser Vacuum. The BOP should identify that the discharge valve for the ‘A’ Circ Water pump 1CW-10 did not automatically shut (interlock with pump) and should attempt to shut the valve by taking the Circ Water pump control switch to stop. The crew should monitor condenser vacuum for Reactor trip criteria (5” Hg absolute AND Turbine first stage pressure is < 60% Turbine Load) while continuing with the GP-006 shutdown.

Event 5: Trip of the running ‘A’ Charging Pump (CSIP) breaker. The crew will enter AOP-018, RCP Abnormal Conditions and carry out the immediate action of isolating letdown. They will continue in AOP-018 and place the ‘B’ CSIP in service. During this evaluation the RO will take ***FK-122.1 (Charging Flow Control) M/A station from manual to automatic***. ***(Competency - Operate Plant Component Controls In Manual Control).***

A low RCP seal injection will cause the ASI system to function. After placing the ‘B’ CSIP in service the crew will have to secure the ASI pump. They will also have to evaluate the boration effects caused by the ASI pump running. The SRO should evaluate the loss of the CSIP in Tech Specs 3.1.2.2, 3.1.2.4 and 3.5.2

TS 3.1.2.2 - At least two of the following three boron injection flow paths shall be OPERABLE:

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

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

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 1 continued

Event 5: Tech Spec Evaluation of loss of Charging Pump (continued)

TS 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.

ACTION: a. With one ECCS subsystem inoperable restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. The SRO should also prepare OMM-001, Attachment 5 Equipment Problem Checklist for the failure.

Event 6 (**Major**): Loss of Offsite Power, Reactor Trip – Once the crew has stabilized the unit and have started the standby CSIP a Loss of Offsite Power will occur. The crew will enter EOP-E-O, Reactor Trip or Safety Injection. While implementing the actions of E-0 for the loss of Offsite power/Reactor Trip Event 7, Loss of ALL AC Power will occur. This will require a transition from EOP-E-0 to EOP-ECA-0.0. The crew may assign an individual to refer to AOP-025 actions based on the de-energized Emergency Busses.

Event 7: Loss of ALL AC power: During the loss of offsite power the ‘A’ EDG will start but the output breaker to the Emergency Bus, Breaker 106, will trip prior to the sequencer reaching Load Block 9 (< 60 seconds from breaker 106 closing). ‘B’ EDG will fail to start and the ‘B’ Emergency Bus will remain de-energized. At this point there will be a complete loss of AC power to the station. The crew should transition from EOP-E-0 step 3 to EOP-ECA-0.0 or directly enter EOP-ECA-0.0 depending on timing. During implementation of EOP-ECA-0.0 the crew checks for each EDG availability. Since Breaker 106 is NOT available for the ‘A’ EDG the crew will be required to place the EDG Emergency Stopped to EMERG STOP.
After determining that the ‘B’ EDG is not running and not tripped, did not have a start failure alarm and Breaker 126 is available the crew should perform the RNO actions for no EDG running and manually start the ‘B’ EDG. When started the EDG will slow start which is a 30 second start as opposed to an emergency start in 10 seconds. When at normal speed and voltage and the output breaker will close and the sequencer will load equipment onto the emergency buss. (The failure was a problem with the EDG receiving a start signal on the UV signal from the Emergency bus). The crew will check AC Emergency bus voltages, initiate monitoring Critical Safety Function Status Trees and then return to EOP-E-0. The crew will then transition to EOP-ES-0.1, Reactor Trip Response and stabilize the plant.

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 1 continued

Event 8: 1MS-70 and 1MS-72 fail to auto open (Loss of all AFW until operator opens 1MS-70 or 1MS-72). ‘B’ MD AFW Pump is under clearance and ‘A’ MD AFW Pump will lose power and not have emergency power since the ‘A’ EDG output breaker has tripped. The Turbine Driven AFW pump will start on a loss of power to the Emergency Bus. There is also another start signal to the TDAFW pump on 2/3 SG’s <25% level. Both of these start signals have failed preventing the auto start of the TDAFW pump by the opening of both 1MS-70 and 1MS-72. If the crew does not respond by opening either 1MS70 or 1MS-72 then a loss of all FW to the Steam Generators will create a RED path on Heat Sink (FR-H.1). Since a loss of ALL AC Power will occur the crew will be implementing EOP-ECA-0.0. A caution prior to step 1 of EOP-ECA-0.0 states: Critical Safety Function Status Trees should be monitored for information only. Function Restoration Procedures should NOT be implemented unless directed by this procedure. The crew should remain in EOP-ECA-0.0 and NOT transition to FR-H.1 if there is a time when a RED path exists. The crew should identify that there is no Feedwater flow to the SG’s and open either 1MS-70 or 1MS-72 (or both) to establish a motive force to run the Turbine Driven AFW pump. Additionally, after opening either 1MS-70 or 1MS-72 to establish flow to the SG the TDAFW pump speed controller should be manually increased and monitored to continue supplying the SG’s a minimum of 200 KPPH AFW flow until at least one intact SG is > 25%.

Scenario termination is met after the crew restores power to the ‘B’ Emergency bus in accordance with EOP-ECA-0.0, has transitioned from EOP-ECA-0.0 to EOP-E-0 then to EOP-ES-0.1 and reached step 5.c, which checks adequate heat sink by having established a minimum feed flow of > 200 KPPH to the SGs.

CRITICAL TASK JUSTIFICATION:

1. Open 1MS-70 or 1MS-72 to establish a minimum of 200 KPPH AFW flow to the Steam Generators prior to 2 of 3 SG's WR Levels < 15%

Wide Range level indications: SG 'A' LI-477.1-SA, SG 'B' 487.1-SB, SG 'C' 497.1-SA

Failure to establish minimum AFW flow is a violation of the basic objective of ECA-0.0 and of the assumptions of the analyses upon which ECA-0.0 is based. Both intend to mitigate deterioration of RCS conditions while ac emergency power is not available. Without AFW flow, the SGs could not support any significant plant cooldown. Thus, the crew would lose the ability to delay the adverse consequences of core uncovery. Also without AFW flow, decay heat would still open the SG safety valves and would rapidly deplete the SG inventory, leading to a loss of secondary heat sink. Decay heat would then increase RCS temperature and pressure until the pressurizer PORVs open, imposing a larger LOCA than RCP seal leakage. Both of these examples violate the basic assumptions of the analyses on which ECA-0.0 is based, complicating the mitigation actions.

2. Start the "B" Emergency Diesel Generator and restore AC Emergency bus power prior to declaring an Extended Loss of AC Power has occurred

Failure to energize an AC emergency bus constitutes mis-operation or incorrect crew performance with a loss and degradation of emergency power capability. Initiating Extended Loss of AC Power (ELAP) mitigating actions will needlessly prolong the return of power to emergency equipment.

Not starting the Emergency Diesel will require the crew to use the Emergency Stop switch on the MCB. Placing this switch to Emergency Stop will prevent any auto start attempts of the Diesel. This will then require the crew and possible offsite assistance, to continue with the Loss of All AC Power procedure and rely on either getting power restored from an offsite source or using the standby equipment designed for extended power loss. During this extended loss of power the potential for other events to occur is increased including the loss of heat sink. If the extended power loss equipment is determined to be needed then this equipment will or may need to be staged and locally operated.

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

For the 2018 NRC Exam Simulator Scenario #1

Reset to IC-161 password “NRC2018”

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

(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 18 and step 19.b circled needing completion

Press START on Counter Scaler
Post conditions for status board from IC-28
Reactor Power 42%
Control Bank D at 146 steps
RCS boron 1145 ppm

Note: The ‘A’ Heater Drain Pump has been secured and the plant was stabilized prior to snapping this IC.

Turnover: A plant shutdown is required due to problems encountered during the repairs on the ‘B’ MDAFW Pump. Repairs will not be able to be completed prior to the LCO expiring. The plant is operating at ~42% power in MOL. The ‘A’ Heater Drain pump has been secured prior to turnover. When turnover is complete the crew will secure the ‘B’ Heater Drain pump then continue a power reduction at 4 MW/min must be started to support being in Mode 3 within the next 6 hours.

Equipment Under Clearance:

- ‘B’ MDAFW Pump is under clearance for pump packing repairs. The pump has been inoperable for 62 hours and will NOT be restored to operable status within the next 10 hours. Tech Spec 3.7.1.2 LCO Action **a** and Tech Spec 3.3.3.5.b Action **c** applies. 72 hour LCO or HSB within the next 60 hours, HSD following 6 hours.

Equipment 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.
- Letdown Orifice Isolation Valve 1CS-9 is under clearance for solenoid replacement. Tech Spec 3.6.3 LCO Action **b** applies. OWP-CS-09 has been completed.

Align equipment for repairs:

Hang CIT on “B” MDAFW Pump MCB switch then place protected train placards per OMM-001 Attachment 16 on "A" MDAFW Pump, MS-70 and 72, "B" ESW Pump, "B" RHR Pump and "B" CCW Pump – note IAW OMM-001, “A” Train PICs: 1, 3, 9, 13, and 17 are also protected.

Place the “B” DEH Pump in PTL and then hang a CIT on MCB switch

Place a CIT on the switch for 1CS-9.

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-09 clearance

Hang restricted access signs on MCR entry swing gates

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	11	of	47

Event Description: **Plant Shutdown (GP-006) – Secure "B" Heater Drain Pump**

Time	Position	Applicant's Actions or Behavior

Lead Evaluator:	**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 has been directed to continue with the plant shutdown using GP-006, Normal Plant Shutdown, due to "B" MDAFW pump LCO expiring. They have been directed to secure the "B" Heater Drain Pump prior to commencing the down power.**

OP-136		OP-136, Feedwater Heaters, Vents, and Drains
Section 7.0 Shutdown		7.1 Shutdown of Heater Drain Pumps
		7.1.1 Initial Conditions
Procedure Note:		Normally the Heater Drain Pumps are stopped when Reactor power is 40 to 45% per GP-006.
	BOP	1. IF only one Heater Drain Pump is to be stopped, THEN the following conditions should be met: a. Reactor power is less than 99% to accommodate for the loss of secondary efficiency. (YES) b. The MW feedback loop is removed from service (YES) 2. IF both Heater Drain pumps are to be stopped, THEN Maintenance has verified that PS-01MS-110 is reset to prevent a turbine runback (YES)

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	12	of	47

Event Description: **Plant Shutdown (GP-006) – Secure "B" Heater Drain Pump**

Time	Position	Applicant's Actions or Behavior
OP-136		Section 7.1.2 Procedure Steps
Procedure Note:		• The intent of this section is to establish 4A (B) Feedwater Heater level control on the Condenser Dump valve before stopping the Heater Drain Pump. This minimizes the level transient when the pump is secured. • As the Condenser Dump valves starts to control level, the HDP discharge level control valve will start to shut and discharge flow will decrease. • The Main Control Room operator must monitor HDP flow and provide trending information to the operator at the pneumatic alternate level controller.
Procedure Caution:		Stopping Heater Drain Pumps at power levels greater than 50% can result in oscillations in heater levels. Heater 4A (4B) Condenser Dump Controller may need adjustment to stabilize levels.
Evaluator Note:		**ERFIS group display or either one of these quick plots "HDPB" or "B_HDP" has been previously created and is a plot available to the Operators**
	BOP	1. CREATE a plot on ERFIS to monitor Heater Drain Pump discharge flow, discharge pressure and heater level. (FHD-1255B, PHD1255B and LHD1250B)
	BOP	2. ESTABLISH communications between the Main Control Room and the operator at 4B pneumatic alternate level controller
Simulator Communicator:		**Acknowledge directions to establish communications with the BOP.**

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Event Description: **Plant Shutdown (GP-006) – Secure "B" Heater Drain Pump**

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**Monitor the FW Heater 4b using simulator drawing FWH05 *(LC-01HD-1251B is HD-323 on the drawing)*** **NOTE: the as-found LC-01HD-1251B (*HD-323*) pneumatic controller setting is also on this display and will be asked for in step 4. Provide the settings value to the Communicator.**
	BOP	**3. IF** desired, **THEN PLACE** the 4B Feedwater Heater Sight Glass in service by slowly opening the applicable isolation valves listed below: • 1HD-299-LI1-2, LG-01HD-1250B Instrument Valve. • 1HD-299-HI1-2, LG-01HD-1250B Instrument Valve. **N/A – Not desired**
	BOP	4. PERFORM the following at LC-01HD-1251B: a. RECORD as-found LC-01HD-1251B pneumatic controller setting in the control room log.
Simulator Communicator:		**Report the as-found LC-01HD-1251B (*HD-323 setpoint*) pneumatic controller setting to the BOP.**
Procedure Note:		Actions in Step 7.1.2.4.b cause response being monitored in Step 7.1.2.4.c. Step 7.1.2.4.b will cause HDP discharge flow to lower.
	BOP	b. While monitoring Heater Drain Pump discharge flow, DIRECT the local operator to slowly lower the set point on 4B pneumatic alternate level controller.
Simulator Operator:		**Run Trigger 12 - *(HD-323 is put to manual and the setpoint is adjusted down to 2")* to open the 4B FWH alternate level control valve to lower HDP "B" discharge flow**
	BOP	c. WHEN Heater Drain Pump discharge flow is less than or equal to 500 kpph, THEN STOP Heater Drain Pump B.
Simulator Operator:		**DO NOT run** Trigger 13**:** conditionally activates when "B" HDP control switch is taken to STOP.

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	14	of 47

Event Description: **Plant Shutdown (GP-006) – Secure "B" Heater Drain Pump**

Time	Position	Applicant's Actions or Behavior
	BOP	d. DIRECT the operator at LC-01HD-1251B to slowly adjust 4B Feedwater Heater level to approximately 2 inches.
	BOP	e. RECORD as-left LC-01HD-1251B pneumatic controller setting in the control room log.
Simulator Communicator:		**(As left setting of LC-01HD-1251B can be found on drawing FWH05)** **Report the as-left LC-01HD-1251B pneumatic controller setting to the BOP.** ***(HD-323 should read 2")***
	BOP	5. IF necessary, THEN REPEAT Steps 7.1.2.1 through 7.1.2.4 for the remaining pump. **(N/A)**
	BOP	6. VERIFY the 4A and 4B Feedwater Heater Sight Glasses are isolated by shutting isolation valves listed below: • 1HD-293-HI1-2, LG-01HD-1250A Instrument Valve • 1HD-293-LI1-2, LG-01HD-1250A Instrument Valve • 1HD-299-HI1-2, LG-01HD-1250B Instrument Valve • 1HD-299-LI1-2, LG-01HD-1250B Instrument Valve **N/A sight glasses were NOT cut in.**
	BOP	Reports to CRS that the "B" Heater Drain Pump is secured
Evaluator Note:		**The following steps will re-initiate turbine load reduction IAW GP-006. These steps have been previously completed and should be validated prior to recommencing the power reduction.**
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 prepare for a power reduction at 4 DEH Units/Min. and directs the RO to initiate a boration before the power reduction begins.

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	15	of	47	

Event Description: **Plant Shutdown (GP-006) – Secure "B" Heater Drain Pump**

Time	Position	Applicant's Actions or Behavior
GP-006		Normal Plant Shutdown From Power Operation to Hot Standby
	SRO	Direct RO to perform boration IAW the Reactivity Plan
OP-107.01		Section 5.2 Blender Boration Operation
	RO	• DETERMINE the reactor coolant boron concentration from chemistry OR the Main Control Room status board. • DETERMINE the magnitude of boron concentration change required. • DETERMINE the volume of boric acid to be added using the reactivity plan associated with the IC.
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.
		• SET controller 1CS-283, FK-113 BORIC ACID FLOW, for the desired flow rate. • VERIFY the RMW CONTROL switch has been placed in the STOP position. • VERIFY the RMW CONTROL switch green light is lit.

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	16	of 47
Event Description:	**Plant Shutdown (GP-006) – Secure “B” Heater Drain Pump**							

Time	Position	Applicant’s Actions or Behavior
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: • 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. 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: ○ TURN control switch RMW CONTROL to START momentarily. ○ VERIFY the RED indicator light is LIT. ○ IF expected system response is not obtained, THEN TURN control switch RMW CONTROL to STOP. IF controller 1CS-283, FK-113 BORIC ACID FLOW, was changed in Step 5.2.2.5, THEN: ○ REPOSITION controller 1CS-283, FK-113 BORIC ACID FLOW, to the position recorded in Step 5.2.2.5.a.

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	17	of 47

Event Description: **Plant Shutdown (GP-006) – Secure "B" Heater Drain Pump**

Time	Position	Applicant's Actions or Behavior
	BOP	• 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.
	RO	TURN control switch RMW MODE SELECTOR to AUTO. START the makeup system as follows: o TURN control switch RMW CONTROL to START momentarily. o VERIFY the RED indicator light is LIT.
Procedure Note:		Systems and components operated from the Main Control Board on a daily basis to support normal plant operations do not require Independent Verification. If this evolution is performed daily or more frequently, then performance of Section 5.2.3 is not required.

GP-006		Normal Plant Shutdown From Power Operation to Hot Standby
	BOP	Requests PEER check prior to manipulations of DEH Control • DEPRESS the LOAD RATE MW/MIN push-button. • ENTER the desired rate, **NOT** to exceed 5 MW/MIN, in the DEMAND display. (4 DEH Units/minute) • DEPRESS the ENTER push-button. • DEPRESS the REF push-button. • ENTER the desired load (120 MW) in the DEMAND display. • DEPRESS the ENTER push-button. The HOLD push-button should illuminate.

Op Test No.:	NRC	Scenario #	1	Event #	1	Page	18 of	47

Event Description: **Plant Shutdown (GP-006) – Secure "B" Heater Drain Pump**

Time	Position	Applicant's Actions or Behavior
Procedure Note:		The unloading of the unit can be stopped at any time by depressing the HOLD push-button. The HOLD lamp will illuminate and the GO lamp will extinguish. The load reduction can be resumed by depressing the GO push-button. The HOLD lamp will extinguish and the GO lamp will illuminate.
	BOP	DEPRESS the GO push-button to start the load reduction and inform crew through 'Crew Update' Turbine in 'GO'. • VERIFY the number in the REFERENCE display decreases. • VERIFY Generator load is decreasing. • WHEN Turbine 1st Stage pressure is less than 260 psig, THEN PLACE the SG LVL ATWS PANEL BYPASS Switch to BYPASS.
Lead Evaluator:		**AFTER the crew has reduced power to the satisfaction of the evaluation crew, cue the Simulator Operator to insert Trigger 2** **Event 2 - "Reactor Vessel Flange Leak"**

Op Test No.:	NRC	Scenario #	1	Event #	2	Page	19	of	47

Event Description: **Reactor Vessel Flange Leak**

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 2:** **Reactor Vessel Flange Leak**
Indications Available:		• **ALB-10-5-5, Reactor Vessel Flange Leakoff High Temp** • **TI-401, Reactor Vessel Flange Leakoff Temp increasing**
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.**
APP-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. Decreasing • 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
	RO	Informs SRO Reactor Vessel Flange leakage is isolated
	SRO	• Reviews/prepares OMM-001, Attachment 5 Equipment Problem Checklist • Contacts WCC to coordinate Containment entry per AP-545

Op Test No.:	NRC	Scenario #	1	Event #	2	Page	20	of 47

Event Description: **Reactor Vessel Flange Leak**

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	Enters 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 met for AOP-016, Excessive Primary Plant Leakage NOTE- AOP-016 contains NO Immediate Actions
AOP-016	SRO	Enters AOP-016 Makes a plant PA announcement for AOP entry
	SRO	CHECK RHR in operation (NO) CHECK RCS leakage within VCT makeup capability (YES) MAINTAIN VCT level GREATER THAN 5% (YES) CHECK Containment Ventilation monitors clear (YES) Radiation monitors normal (YES) Evacuate personnel from area (NO)

Op Test No.:	NRC	Scenario #	1	Event #	2	Page	21	of	47

Event Description: **Reactor Vessel Flange Leak**

Time	Position	Applicant's Actions or Behavior
	BOP	NOTIFY Chemistry to stop any primary sampling activities (Calls Chemistry)
Communicator:		**Acknowledge request to stop primary sampling activities.**
	RO	PERFORM a qualitative RCS flow balance ESTIMATE leak rate considering the following parameters: • PRZ level rate of change (~55 gal/% at 683°F) • Charging flow • Total seal injection flow • Letdown flow • Total seal return flow (Estimate = 15 gpm flow to RCDT) Operate Letdown as necessary to maintain Charging on scale (NO changes required)
	SRO	Determines it is not necessary to more accurately quantify leakage using either OST-1026 or OST-1226
Evaluator Note:		**Any Tech Spec evaluation can be conducted with a follow up question after the scenario.**
	SRO	Enters 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.

Op Test No.:	NRC	Scenario #	1	Event #	2	Page	22	of 47

Event Description: **Reactor Vessel Flange Leak**

Time	Position	Applicant's Actions or Behavior
	SRO	Leak location identified from MCB indications • From RV Flange
	BOP	NOTIFY HP of Reactor Vessel Flange leakage
Simulator Communicator:		**Acknowledge RCs leakage is coming from Reactor Vessel Flange.**
	SRO	Transitions to Attachment 6: Leakage From RV Flange • Consult with Operation Management to determine leak isolation and recovery actions Exit AOP-016 Contacts WCC to coordinate Containment entry per AP-545 Reviews/prepares OMM-001, Attachment 5 Equipment Problem Checklist
Evaluator Cue:		**After Tech Spec is identified and a request for support has been completed, cue Simulator Operator to insert Trigger 3** **Event 3 – Main Generator Automatic Voltage Regulator Failure**

Op Test No.:	NRC	Scenario #	1	Event #	3	Page	23 of	47

Event Description: **Main Generator Automatic Voltage Regulator Failure**

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 3:** **Main Generator Automatic Voltage Regulator Failure**
Indications Available:		**• ALB-022, Window 5-4, GEN EXCITATION AUTO FAULT IN CH 1 OR CH 2** **• ALB-022, Window 8-3, Gen Voltage Reg Common Alarm** **• MVARS and Switchyard voltage oscillating**
	BOP	RESPONDS to alarms ALB-022-5-4 and 8-3
ALB-022	BOP	Reviews APP-ALB-022-5-4 and 8-3 Dispatches an operator to 286 TB switchgear room to check the Excitation Control Terminal (ECT)
Simulator Communicator Note:		**When dispatched as TB AO to check the Excitation Control Terminal (ECT), report channel 1 is in alarm and the AVR is operating in channel 2.**
	CRS	Contacts WCC for maintenance support and notifications.
Simulator Communicator Note:		**If contacted as WCC, acknowledge any requests for support.**
	CREW	Identifies oscillations in Main Generator MVAR Output and SWYD Voltage - Annunciator ALB-022-9-4, Computer Alarm Gen/Exciter Systems
	BOP	Contacts the dispatcher about oscillations
Simulator **Communicator Note:**		**If contacted as the Dispatcher, report conditions on the Grid are normal and the problem appears to be localized to Harris**

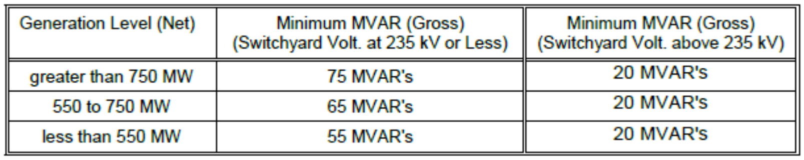
Op Test No.:	NRC	Scenario #	1	Event #	3	Page	24 of	47

Event Description: **Main Generator Automatic Voltage Regulator Failure**

Time	Position	Applicant's Actions or Behavior
Evaluator Note:		**After a 4 minute delay, oscillations will be observed in the Main Generator MVAR output and SWYD voltage. The crew should use AD-OP-ALL-1000 guidance to take manual control of the AVR. Once in Manual on the AVR, the crew should restore MVAR output and notify the dispatcher.**
	BOP	Determines the AVR is malfunctioning in Auto
Evaluator Note:		**If the crew takes no action the AVR will fail to manual 6 minutes after the oscillations begin.** **ALB-022, Window 4-2, Gen Excitation Auto Fault In CH 1 And CH 2, will annunciate when the AVR has automatically tripped to Manual.**
	BOP	Transfers the AVR to Manual Control in accordance with AD-OP-ALL-1000 for a failed controller. (OP-153.01 Section 8.8 Shifting AVR from Auto to Manual) Places CS-1538, OPERATION MODE, in MANUAL. ***(Competency - Operate Plant Component Controls In Manual Control).*** Checks that the WHITE light is lit. Verifies EI-543, GENERATOR STATOR VOLTAGE, indicates 22kV (21.5 - 22.5kV) and is stable. • Restores MVAR output to band per the Generator Capability Curve.
	CRS/BOP	Informs the dispatcher that the AVR is in Manual Control
Simulator Communicator Note:		**Acknowledge report that the Harris Main Gen Automatic Voltage Regulator is in Manual.**

Op Test No.: NRC Scenario # 1 Event # 3 Page 25 of 47

Event Description: **Main Generator Automatic Voltage Regulator Failure**

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>SRO</td><td>Contacts WCC for support
Provides control band to BOP for MVAR control based on OP-153.01 normal limits

<table>
<tr><th>Generation Level (Net)</th><th>Minimum MVAR (Gross) (Switchyard Volt. at 235 kV or Less)</th><th>Minimum MVAR (Gross) (Switchyard Volt. above 235 kV)</th></tr>
<tr><td>greater than 750 MW</td><td>75 MVAR's</td><td>20 MVAR's</td></tr>
<tr><td>550 to 750 MW</td><td>65 MVAR's</td><td>20 MVAR's</td></tr>
<tr><td>less than 550 MW</td><td>55 MVAR's</td><td>20 MVAR's</td></tr>
</table></td></tr>
<tr><td colspan="2">Simulator Communicator:</td><td>Acknowledge request and reports from SRO.</td></tr>
<tr><td colspan="2"></td><td></td></tr>
<tr><td colspan="2">Evaluators Note:</td><td>After the Main Gen AVR is stabilized insert Event 4 “Trip of the ‘A’ Circ Water Pump and disch valve 1CW-10 failure”</td></tr>
</table>

Op Test No.:	NRC	Scenario #	1	Event #	4	Page	26	of	47

Event Description: **‘A’ Circ Water Pump Trip and Failure of Discharge Valve 1CW-10 to shut**

Time	Position	Applicant’s Actions or Behavior
Simulator Operator:		**When directed by Lead Evaluator: Actuate Trigger 4 “Trip of the ‘A’ Circ Water Pump and Failure of Discharge Valve 1CW-10 to shut”**
Indications Available:		- ALB-021-4-4, CIRC WTR PMP A O/C - GND – TRIP
	BOP	RESPONDS to alarm ALB-021-4-4.
Evaluator Note:		**The crew may enter AOP-012, Partial Loss of Condenser Vacuum, without doing the alarm response procedure. The SRO may elect to reduce power to control vacuum.**
APP-ALB-021	SRO	ENTERS APP-ALB-021-4-4.
Evaluator Note:		**In accordance with AD-OP-ALL-1000, the operator may take MANUAL actions when automatic actions do not occur and place the CWP ‘A’ control switch in the stop position to shut the pump discharge valve before being directed by a procedure.**
	BOP	CONFIRM alarm using: - Circ Wtr Pump A status lights - Circ Wtr Pump A discharge valve position
	BOP	VERIFY Automatic Functions: CWP A trips (YES)

Op Test No.: NRC Scenario # 1 Event # 4 Page 27 of 47

Event Description: **'A' Circ Water Pump Trip and Failure of Discharge Valve 1CW-10 to shut**

Time	Position	Applicant's Actions or Behavior
	BOP	PERFORM Corrective Actions: • IF Circulating Water Pump trips OR Condenser vacuum is degrading, THEN GO TO AOP-012, Partial Loss of Condenser Vacuum. (YES) • IF necessary, THEN START the standby CWP. (N/A)
	SRO	DISPATCHES AO to investigate.
Simulator Communicator:		**Wait 3 minutes and report the breaker tripped on overcurrent.**
	BOP	IF STOP signal is not given to CWP 'A' control switch, DISPATCHES AO to manually CLOSE discharge valve. NOTE: they may direct opening the discharge valve breaker prior to manually stoking the valve
Simulator Operator		**NOTE:** The crew may direct opening the discharge valve breaker prior to having the valve manually stoked. IF they do wait 2 minutes, and instead of running Trigger 20 go to the Summary page and **modify** ilo xb2o069b (1CW-10 light status) to **OFF**.
Simulator Communicator:		**Report back that the power has been removed.** **5 minutes later report back that the discharge valve has been shut.**
Simulator Operator:		**IF power has been left on the CW pump discharge valve THEN after approximately 5 minutes from when the AO was dispatched actuate Trigger 20. This will time out the discharge valve MCB light indications and provide the BOP indication that the discharge valve is stoking closed.**

Op Test No.:	NRC	Scenario #	1	Event #	4	Page	28 of	47

Event Description: **‘A’ Circ Water Pump Trip and Failure of Discharge Valve 1CW-10 to shut**

Time	Position	Applicant’s Actions or Behavior
Simulator Communicator:		**IF the discharge valve has been manually stoked using Trigger 20, report back as the AO assigned the valve is closed when the discharge valve lights indicate the valve is closed.**
	SRO	ENTERS and directs actions of AOP-012, PARTIAL LOSS OF CONDENSER VACUUM Makes PA announcement for AOP entry Holds a crew focus brief
AOP-012		Partial Loss Of Condenser Vacuum
Procedure Note:		**This procedure contains no immediate actions.**
	SRO	CHECK Turbine – IN OPERATION (YES)
	BOP	CHECK Condenser pressure in both Zones less than: 7.5 inches Hg absolute AND Turbine first stage pressure is greater than 60% TURBINE LOAD (YES) OR 5 inches Hg absolute AND Turbine first stage pressure is less than 60% TURBINE LOAD (YES)
	SRO	REDUCE Turbine load as necessary to maintain Condenser vacuum using ONE of the following: • GP-006, Normal Plant Shutdown from Power Operation to Hot Standby • AOP-038, Rapid Downpower

Op Test No.:	NRC	Scenario #	1	Event #	4	Page	29 of	47

Event Description: **‘A’ Circ Water Pump Trip and Failure of Discharge Valve 1CW-10 to shut**

Time	Position	Applicant’s Actions or Behavior
	SRO	CONTINUE Turbine load reduction until directed otherwise by CRS based on the following: • Cause of vacuum loss identified and corrected • Vacuum stable or increasing • Plant condition require Reactor or Turbine trip (NO action required.)
	BOP	CHECK Condenser Vacuum Pump – OPERATING. (YES)
	BOP	Start standby Vacuum Pump
Simulator Communicator:		**When contacted report 1AE-16, B Vacuum Pump Suction Isolation Valve, is open.**
	CREW	DISPATCH Operator(s) to locally perform actions of Attachment 1, Local Actions for a Loss of Condenser Vacuum. (N/A)
	BOP	VERIFY the following valves – SHUT: • 1CE-447, Condenser Vac Breaker(YES) • 1CE-475, Condenser Vac Breaker(YES)
	BOP	CONTACT Radwaste Control Room to determine if recent equipment operations using Auxiliary Steam or Condensate may have caused loss of vacuum.
Simulator Communicator:		**Report no Auxiliary Steam or Condensate equipment has been recently operated.**
	BOP	CHECK Circulating Water Pumps – ANY TRIPPED (YES)

Op Test No.:	NRC	Scenario #	1	Event #	4	Page	30 of	47
Event Description:	**‘A’ Circ Water Pump Trip and Failure of Discharge Valve 1CW-10 to shut**							

Time	Position	Applicant’s Actions or Behavior
	BOP	VERIFY associated pump discharge valve – SHUT. IF STOP signal is not given to CWP ‘A’ control switch, DISPATCHES AO to CLOSE valve. (NO) (If not already done)
Procedure Note:		If a Circulating Water Pump has tripped, it is not considered available until the cause of the trip has been identified and corrected.
	SRO	CHECK ALL available Circulating Water Pumps – RUNNING. (YES)
Evaluator Note:		**AOP-012 does not have to be completed to continue the scenario after the discharge valve is being closed.** **Event 5 (CSIP Trip) can be cued after the CW pump discharge valve has been closed.**

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	31	of	47
Event Description:			**CSIP Trip**						

Time	Position	Applicant's Actions or Behavior

Simulator Operator:	**On cue from the Lead Evaluator actuate Trigger 5 'A' CSIP trip**

Indications Available:		ALB-06-1-1 Charging Pump Discharge Header High-Low Flow ALB-06-1-2 Chrg Pump A Trouble ALB-06-1-3 Chrg Pump A Trip Or Close Ckt Trouble ALB-08-2-1 RCP Seal Water Injection Low Flow ALB-08-2-2 ASI Pump Auto Start Timer Initiated
	RO	• RESPONDS to multiple alarms on ALB-06 (1-1, 1-2, 1-3) and ALB-08 (2-1 & 2-2). • REPORTS CSIP 'A' tripped.
	CREW	Identifies Entry Conditions met for AOP-018, Reactor Coolant Pump Abnormal Conditions
AOP-018		Reactor Coolant Pump Abnormal Conditions
Immediate Action	RO	PERFORMS immediate actions. • CHECK ANY CSIP RUNNING. (NO) • ISOLATE letdown by verifying the following valves SHUT: ○ 1CS-7, 45 GPM Letdown Orifice A ○ 1CS-8, 60 GPM Letdown Orifice B ○ 1CS-9, 60 GPM Letdown Orifice C

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	32 of	47
Event Description:			**CSIP Trip**					

Time	Position	Applicant's Actions or Behavior
	SRO	ENTERS AOP-018, RCP Abnormal Conditions Makes PA announcement for AOP entry Conducts a focus brief
	BOP	Dispatch operators to investigate cause of trip
Simulator Communicator:		**If dispatched to investigate, wait 2-3 minutes then report a breaker overcurrent trip flag on Phase A. Report as second AO that there are no obvious problems at the pump.**
	SRO	Informs SM to 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.**
	SRO	EVALUATE plant conditions AND GO TO the appropriate section:

MALFUNCTION	SECTION	PAGE
Loss of CCW and/or Seal Injection to RCPs	3.1	5

	RO	CHECK ALB-5-1-2A, RCP Thermal Bar HDR High Flow, alarm CLEAR. (**YES**)
	SRO	CHECK ALL RCPs operating within the limits of Attachment 1. (**YES**)

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	33 of 47
Event Description:			**CSIP Trip**				

Time	Position	Applicant's Actions or Behavior	
	RO	• CHECK ALL RCPs RUNNING. (**YES**) • CHECK the following NORMAL for ALL RCPs: ○ CCW flow (**YES**) ○ Seal Injection flow (NO)	
	SRO	RESTORE using the applicable attachment:	
		MALFUNCTION	ATTACHMENT
		Loss of Seal Injection flow only	Attachment 4 (Page 28)
Procedure Note:		**The ASI System will actuate in 2 minutes and 45 seconds from timer initiation. ALB-8-2-4 ASI pump start will alarm**	
Evaluator Note:		**The ASI system when actuated will provide RCP seal injection of highly borated water. During the time the ASI pump is running a negative reactivity addition is being conducted in the form of boration. The sooner the system is shut down the less effect it will have on reactivity.**	
	RO	• CHECK at least one CSIP RUNNING. (NO) • Dispatch an operator to monitor operation of the ASI System	
Simulator Communicator:		**Acknowledge request.**	
Simulator Operator:		**Be prepared to STOP the ASI pump when requested to.**	

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	34 of	47
Event Description:			**CSIP Trip**					

Time	Position	Applicant's Actions or Behavior
	RO	• ***PLACE controller FK-122.1, Charging Flow in MANUAL*** AND SHUT. ***(Competency - Operate Plant Component Controls In Manual Control).*** • CHECK RCS pressure GREATER THAN 1400 PSIG. (**YES**) • SET FK-122.1 DEMAND position to 30%. • SHUT HC-186.1, RCP Seal WTR INJ Flow. • VERIFY a suction path for the standby CSIP by performing the following: o VERIFY CSIP suction flowpath from VCT as follows: ▪ VERIFY > 5% level is established in VCT. (**YES**) o VERIFY the following valves are OPEN: • LCV-115C, VCT Outlet (1CS-165) (**YES**) • LCV-115E, VCT Outlet (1CS-166) (**YES**)
	SRO	Before exiting AOP-018, provide Pressurizer level control bands and trip limits per OMM-001 Att. 13 – Control band - Maintain level within 5% of Reference level – trip limits of 10% low and 90% high
Procedure Caution:		**Low VCT level is a precursor to gas binding the CSIPs**
	RO	CHECK VCT level is greater than 5%, AND STABLE OR RISING (**YES**)
	RO	MAINTAIN CCW HX outlet temperature less than 105°F.
	RO	START the standby CSIP. (**Starts 'B' CSIP**)
	RO	CHECK seal injection flow being supplied by the ASI System. (**YES**)

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	35 of	47
Event Description:				**CSIP Trip**				

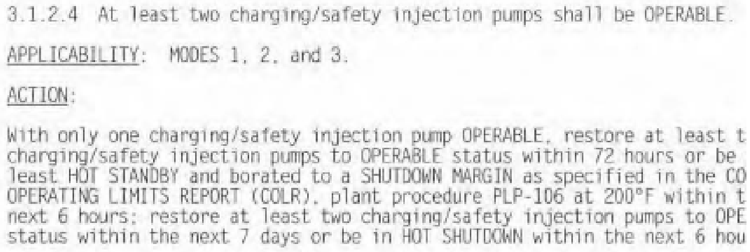
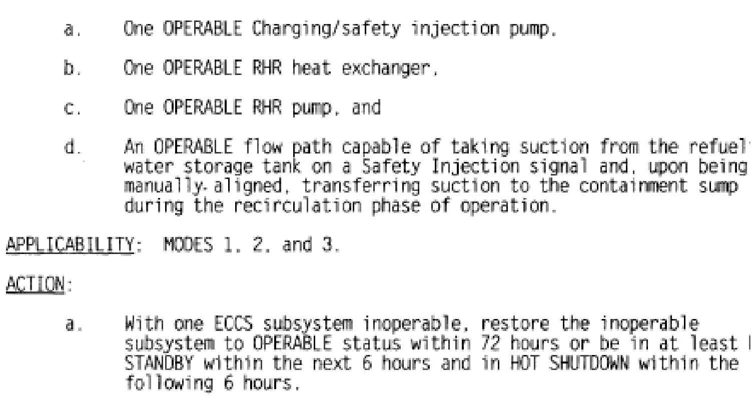
Time	Position	Applicant's Actions or Behavior
	RO	OPEN HC-186.1, RCP Seal WTR INJ Flow. DIRECT the operator monitoring the ASI System to STOP the ASI Pump by placing CS-210.1, ASI PUMP MOTOR CONTROL SWITCH, in STOP. (At the ASI System Control Panel)
Simulator Communicator:		**Acknowledge request to secure the ASI pump**
Simulator Operator:		**Secure the ASI pump when communications are complete** **CVC 195 STOP**
Evaluator Note:		**ALB-8-2-3 ASI system Trouble will alarm when ASI pump is stopped**
Simulator Communicator:		**Report back that the ASI pump is secured**
	RO	ADJUST HC-186.1, RCP Seal WTR INJ Flow, to establish seal injection flow as necessary to maintain the following: • Less than 31 gpm total flow to all RCPs. • Between 8 and 13 gpm to all RCPs.
	RO	DIRECT the operator monitoring the ASI System to PLACE CS-210.1, ASI PUMP MOTOR CONTROL SWITCH, in AUTO. (At the ASI System Control Panel)
Simulator Communicator:		**Acknowledge request**
Simulator Operator:		**Place ASI control back to AUTO - CVC 195 AUTO**

Op Test No.:	NRC	Scenario #	1	Event #	5	Page	36 of	47
Event Description:		**CSIP Trip**						

Time	Position	Applicant's Actions or Behavior
Simulator Communicator:		**After ASI controls are in AUTO:** **Report:CS-210.1, ASI PUMP MOTOR CONTROL SWITCH is in AUTO**
	BOP	START CSIP room ventilation per OP-172, Reactor Auxiliary Building HVAC System. IF B Train is being started, THEN PLACE the following Air Handling Units control switches to START AND VERIFY proper damper and valve operation (if they start): • CSIP SB AREA FAN COOLER AH-9 B SB
	RO	RESTORE Charging and Letdown flow per OP-107, Chemical and Volume Control System.
Evaluator Note:		**There is no need to wait for letdown to be restored – Continue with scenario.**
	BOP	Start 'B' Chiller per OP-148, section 5.2. Contact AO for Chiller pre-start checks (NOTE: At this time the crew may only start Pump P-4 B)
Evaluator Note:		**It is NOT intended to wait for the crew to place the standby Chiller in service – Continue with scenario.**
	RO	MONITOR Tavg to Tref. (ASI injection has added negative reactivity)
	SRO	INITIATE action to determine and correct the cause of the loss of the CSIP.
	RO	CHECK seal injection flow between 8 and 13 gpm has been established to all RCPs.

Op Test No.: NRC Scenario # 1 Event # 5 Page 37 of 47

Event Description: **CSIP Trip**

Time	Position	Applicant's Actions or Behavior
	RO	WHEN seal injection flow has been established between 8 and 13 gpm, THEN PERFORM OST-1126, Reactor Coolant Pump Seals Controlled Leakage Evaluation Monthly Interval Modes 1-4.
	RO	CHECK CCW flow is established to the RCPs.
	SRO	EXIT AOP-018 and contacts support personnel for repairs.
Simulator Communicator:		**Acknowledge requests**
	SRO	Addresses Technical Specifications: - 3.1.2.4 - CSIP's  3.1.2.4 At least two charging/safety injection pumps shall be OPERABLE. APPLICABILITY: MODES 1, 2, and 3. ACTION: With only one charging/safety injection pump OPERABLE, restore at least t charging/safety injection pumps to OPERABLE status within 72 hours or be least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CO OPERATING LIMITS REPORT (COLR), plant procedure PLP-106 at 200°F within t next 6 hours; restore at least two charging/safety injection pumps to OPE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hou - 3.5.2 Action a.- ECCS Subsystems  3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shal 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 refuel 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 STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Both are 72 hours to restore action statements.

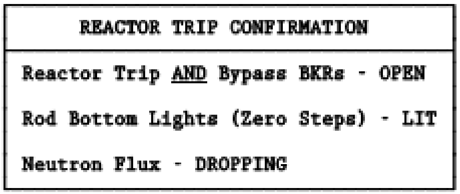
Op Test No.:	NRC	Scenario #	1	Event #	5	Page	38 of 47

Event Description: **CSIP Trip**

Time	Position	Applicant's Actions or Behavior
	SRO	May perform Plant Status brief.
Evaluator Note:		**Tech Specs can be discussed as follow up questions after the scenario is completed. When appropriate the Lead Evaluator can cue Event 6 (Loss of Offsite Power, Reactor Trip – Major Event) and continue in this scenario.**

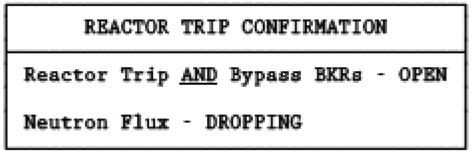
Op Test No.:	NRC	Scenario #	1	Event #	6 & 7	Page	39 of	47

Event Description: **Loss of Offsite Power, Loss of ALL AC Power**

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**When directed by Lead Evaluator: Actuate Trigger 6** **Loss of Offsite Power, Reactor Trip** **NOTE: The Loss of Offsite Power will also cause a loss of ALL AC. The 'A' EDG output breaker will trip during sequencer operation and the 'B' EDG will not start.**
	Crew	Identifies that the Reactor has tripped and a loss of offsite power has occurred. (The control room normal lighting goes off and the emergency lighting comes on.)
EOP-E-0		Reactor Trip Or Safety Injection
	SRO	Steps through immediate actions with crew Makes plant PA announcement
Immediate Action	RO	Verifies Reactor is Tripped (YES)  REACTOR TRIP CONFIRMATION Reactor Trip AND Bypass BKRs - OPEN Rod Bottom Lights (Zero Steps) - LIT Neutron Flux - DROPPING
Immediate Action	BOP	Verifies Turbine is Tripped – All throttle valves shut (YES)  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

Op Test No.: NRC Scenario # 1 Event # 6 & 7 Page 40 of 47

Event Description: **Loss of Offsite Power, Loss of ALL AC Power**

Time	Position	Applicant's Actions or Behavior
Immediate Action	BOP	Verify Power To AC Emergency Buses - NO AC emergency buses – AT LEAST ONE ENERGIZED ('A' Emergency Bus – No, 'B' Emergency Bus –NO) Identifies that the 'A' EDG output breaker 106 has tripped prior to the sequencer reaching Load Block 9 (with time they will also identify that the 'B' EDG did not attempt to start)
	SRO	GO TO ECA-0.0, "LOSS OF ALL AC POWER", Step 1.
EOP- ECA-0.0		Loss of All AC Power
Procedure Note:		Steps 1 AND 2 are immediate action steps. • Critical Safety Function Status Trees should be monitored for information only. • Function Restoration procedures should NOT be implemented unless directed by this procedure.
	SRO	Enter ECA-0.0 Makes PA announcement for EOP entry Crew performs immediate actions (Steps 1 and 2)
Immediate Action	RO	Verify Reactor Trip:  REACTOR TRIP CONFIRMATION Reactor Trip <u>AND</u> Bypass BKRs - OPEN Neutron Flux - DROPPING o Trip breakers RTA AND BYA – OPEN (YES) o Trip breakers RTB AND BYB – OPEN (YES) o Neutron flux – DECREASING (YES)

Op Test No.: NRC Scenario # 1 Event # 6 & 7 Page 41 of 47

Event Description: **Loss of Offsite Power, Loss of ALL AC Power**

Time	Position	Applicant's Actions or Behavior
Evaluator Note:		**The BOP is required to check Turbine throttle valve positions using status light indications. With a loss of power all other MCB indications for the Turbine throttle and governor valves have no indication.**
Immediate Action	BOP	Verify 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 • All turbine throttle valves – SHUT (YES)
	RO	**Check If RCS Isolated** Check letdown isolation valves - SHUT: • 1CS-1 (LCV-460) (YES) • 1CS-2 (LCV-459) (YES) Check PRZ PORVs – SHUT (YES) Verify excess letdown valves - SHUT: • 1CS-460 (YES) • 1CS-461 (YES) Check RCPs – SECURED (YES)
	BOP	**Verify AFW Flow AND Control SG Levels**: • Verify AFW Flow – GREATER THAN 200 KPPH (NO) • Reports to SRO (or identifies 1MS-70 and 1MS-72 are not open and OPENS one or both valves)
	SRO	Directs BOP to verify the TDAFW pump is running (NO) Directs BOP to open either 1MS-70 or 1MS-72

Op Test No.:	NRC	Scenario #	1	Event #	8	Page	42 of 47

Event Description: **1MS-70 and 1MS-72 fail to auto open**

Time	Position	Applicant's Actions or Behavior
Event 8 Critical Task #1	BOP	Opens 1MS-70 or 1MS-72 and establishes a minimum of 200 KPPH to the Steam Generators by adjusting TD AFW pump speed. Any level - GREATER THAN 25% [40%] (NO) Control AFW flow to maintain all intact levels between 25% and 50% [40% and 50%] Critical to Open 1MS-70 or 1MS-72 to establish a minimum of 200 KPPH AFW flow to the Steam Generators prior to 2 of 3 SG's Wide Range Levels < 15% WR level indication: SG 'A' LI-477.1-SA, SG 'B' 487.1-SB, SG 'C' 497.1-SA
	SRO	Evaluate EAL Matrix
	BOP	Verify AC Emergency Bus Cross-Ties to Non-Emergency AC Buses - OPEN
	BOP	Verify any cross tie to Bus 1A-SA - OPEN o Breaker 104 o Breaker 105 Verify Any cross tie to Bus 1B-SB - OPEN o Breaker 124 o Breaker 125

Op Test No.:	NRC	Scenario #	1	Event #	7 cont.	Page	43 of	47
Event Description:			**Loss of ALL AC power**					

Time	Position	Applicant's Actions or Behavior

Procedure Caution:		• Emergency stopping an EDG will deenergize the field flashing circuit and prevent a fire in the GCP control section. • Do NOT start any EDG that is emergency stopped OR close any tripped EDG output breaker until problem corrected.
Procedure Note:		If an EDG was paralleled to the grid prior to LOOP, and the LOOP signal did not open Breaker 106 (126), the breaker is still considered available as long as it was manually opened successfully.
	BOP	ECA-0.0 Step 7 Check EDGs 1A-SA AND 1B-SB - AVAILABLE (FOR START FROM MCB) Check all of the following for EDG 1A-SA: • DIESEL GENERATOR A TRIP annunciator [ALB-024-3-1] - CLEAR – YES • DIESEL GENERATOR A START FAILURE annunciator [ALB-024-3-3] - CLEAR - YES • Breaker 106 – AVAILABLE (**NOT** TRIPPED DUE TO ELECTRICAL FAULT) - Contacts AO to investigate Breaker 106 locally
Simulator Communicator:		**Breaker 106 has tripped on overcurrent on the 'C' Phase**
	BOP	RNO for A EDG: **Place the EDG 1A-SA emergency stop switch to EMERG STOP.** Check any EDG – AVAILABLE (NOT Emergency Stopped) (YES, EDG 1B-SB is available)

Op Test No.:	NRC	Scenario #	1	Event #	7 cont.	Page	44	of	47
Event Description:				**Loss of ALL AC power**					

Time	Position	Applicant's Actions or Behavior
Critical Task #2	BOP	Check any EDG – RUNNING (NO) – RNO • Perform the following as necessary to start EDGs (listed in order of preference) ○ Manually start EDGs Critical to Start the "B" Emergency Diesel Generator and restore AC Emergency bus power prior to declaring an Extended Loss of AC Power has occurred NOTE: The EDG will SLOW start and after the EDG is up to speed the sequencer will run and ESF equipment will be sequenced on the bus. **Takes EDG 1B-SB start switch to START and EDG starts – slow start**
	BOP	Check any EDG – RUNNING (YES – EDG 1B-SB is running)
	BOP	Energize AC Emergency Buses using EDGs: Check any AC emergency bus - ENERGIZED: • 1A-SA bus voltage (NO) • 1B-SB bus voltage (YES)
	SRO	Initiate monitoring of Critical Safety Function Status Trees.
	SRO	Return to procedure and step in effect. Transitions to EOP-E-0 step 3.b

Op Test No.:	NRC	Scenario #	1	Event #	7 cont.	Page	45 of	47
Event Description:				Loss of ALL AC power				

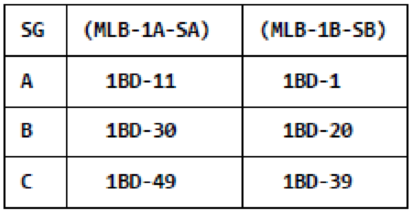
Time	Position	Applicant's Actions or Behavior

EOP-E-0		Step 3.b
	SRO	AC emergency buses – BOTH ENERGIZED (NO only 'B') Perform the following while continuing with EOP implementation: As time allows restore power to de-energized emergency bus. (Refer to AOP-025, "LOSS OF ONE EMERGENCY AC BUS (6.9KV) OR ONE EMERGENCY DC BUS 125V)".)
	SRO	Safety Injection – ACTUATED (BOTH TRAINS) – NO, and not required Transition to EOP-ES-0.1, "Reactor Trip Response", step 1
EOP-ES-0.1		Reactor Trip Response
	SRO	Foldout applies. Initiate Monitoring Of Critical Safety Function Status Trees.
	SRO	Informs SM to Evaluate EAL Matrix.
	BOP	Check RCS Temperature: Check RCPs - ANY RUNNING (NO) Perform the following: • Place steam dump pressure controller in manual **AND** lower output to 0%. • Place steam dump mode select switch in STEAM PRESS.

Op Test No.: NRC Scenario # 1 Event # 7 cont. Page 46 of 47

Event Description: **Loss of ALL AC power**

Time	Position	Applicant's Actions or Behavior
	BOP	Check SG blowdown isolation valves – SHUT (YES)



SG	(MLB-1A-SA)	(MLB-1B-SB)
A	1BD-11	1BD-1
B	1BD-30	1BD-20
C	1BD-49	1BD-39

Time	Position	Applicant's Actions or Behavior
	BOP	Stabilize **AND** Maintain Temperature Between 555°F **AND** 559°F using Table 1.

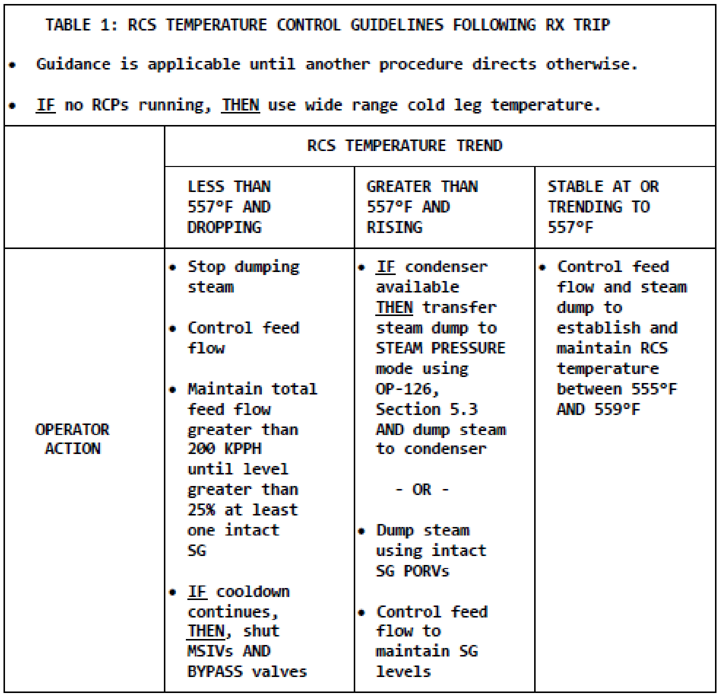


TABLE 1: RCS TEMPERATURE CONTROL GUIDELINES FOLLOWING RX TRIP

- Guidance is applicable until another procedure directs otherwise.
- <u>IF</u> no RCPs running, <u>THEN</u> use wide range cold leg temperature.

	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% at least one 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 #	1	Event #	7 cont.	Page	47 of	47
Event Description:			**Loss of ALL AC power**					

Time	Position	Applicant's Actions or Behavior

Lead Evaluator:	**Terminate the scenario upon determination of RCS Temperature Control.** **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 the Lead Examiner place the Simulator in FREEZE.**

HARRIS 2018 NRC Scenario 2

Facility:	Harris Nuclear Plant	Scenario No.:	2	Op Test No.:	05000400/2018301
Examiners:				Operators:	SRO:
					RO:
					BOP:

Initial Conditions: IC-8, MOL, 3-4% power Turbine at 1700 RPM with Throttle Valve control

- 'A' Gland Steam Condenser Exhaust Fan is under clearance for motor replacement
- PORV Block valve 1RC-113 is SHUT due to PZR PORV 444B Seat Leakage
- 1CS-9 is under clearance for solenoid replacement

Turnover:	The plant is at 3-4% power, MOL, plant startup in progress. Criticality was achieved 2 hours ago, 72 hours after a trip from 100% power. GP-005, Power Operation (Mode 2 to Mode 1) is being implemented.
Critical Task:	• Control PRZ Spray Valve, PCV-444C, prior to RCS pressure reaching the Safety Injection setpoint of 1850 psig • Isolate AFW flow to 'B' SG prior to any RCS Cold Leg Temp lowering to < 240°F after a RCS temp drop in any cold leg of >100° within the last hour • Shut BIT Outlet valves 1SI-3 and 1SI-4 prior to PZR SRV's Discharge Line High Temperature occurring (250°F)

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R – RO/SRO N – BOP/SRO	Start power escalation to 4 – 8% to raise turbine speed from 1700 rpm to 1800 rpm (BOP - DEH Main Turbine Controls - Competency - Operate plant controls in Manual).
2 #	RMS007 ZCR744	I - BOP/SRO TS - SRO	Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically (AOP-005)
3 #	hva04	C – BOP/SRO TS – SRO	"A" Emergency Services Chilled Water Pump Trip (AOP-026)
4 #	ccw01a ccw047	C – RO/SRO TS – SRO	Trip of 'A' CCW Pump on O/C with standby CCW pump 'B' failure to auto start (AOP-014)
5 #	JFB7579 Z2715TIC	C-BOP/SRO TS - SRO	AH-39 Containment Fan Coil Unit fan trip with back up auto start failure ('C' RCP cooling fan)
6 #	PRS14A	I - RO/SRO	Pressurizer Spray Valve, PCV-444C, fails Open (AOP-019 - Manual Control available) (RO – With PRZ Spray in Manual, Competency - Operate plant controls in Manual).
7 #	CFW20B	M - All	Feedline Break on 'B' SG inside Containment (EOP-E-0)
8	ZRPK616A ZRPK616B	I – BOP/SRO	Failure of Auto AFW Isolation on 'B' SG
9	NIS06A	I – RO/SRO	SR Nuclear Instruments fail to energize post trip due to IR NI-35 undercompensated

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

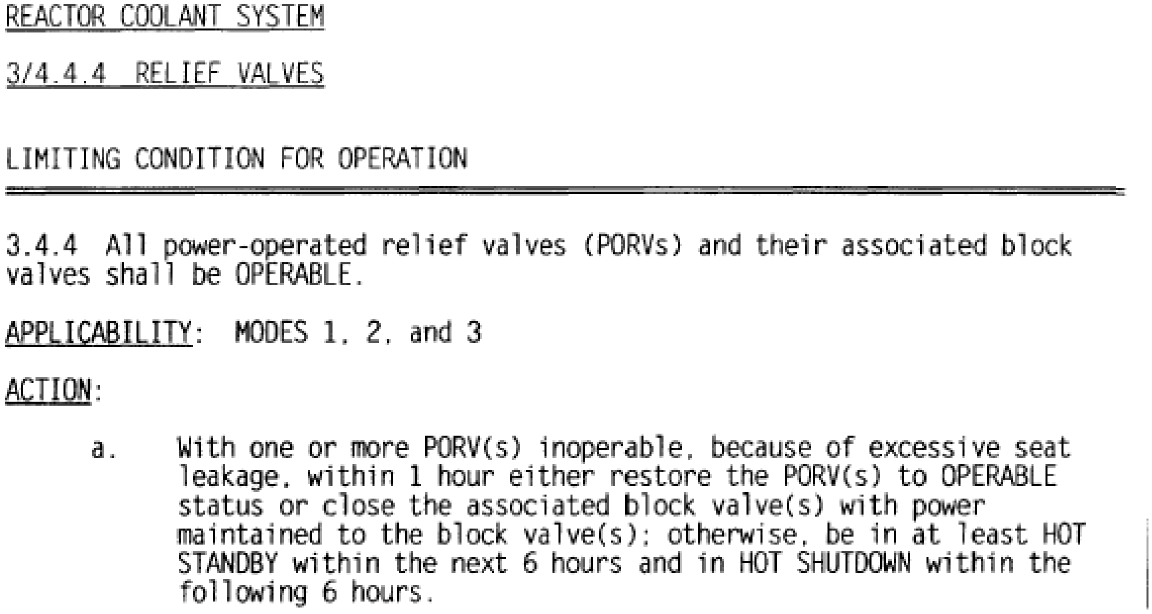
\# Event or Major Transient NOT used on the previous 2 NRC initial licensing operating tests

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 2

The plant is at 3%-4% power, MOL, with a plant startup in progress. GP-005, Power Operation (Mode 2 to Mode 1) is being implemented. Criticality was achieved 2 hours ago, 72 hours after a trip from 100% power. During the previous shift secondary chemistry parameters degraded and Reactor power was reduced to <5%. Chemistry reports that all secondary chemistry is now adequate to continue the power increase. The Turbine is at 1700 rpm with Turbine Throttle Valves controlling Turbine speed. After taking turnover the crew will raise Reactor power to 7% - 9% power then transfer control from Throttle Valves to Turbine Governor valves then ramp the Turbine speed up to 1800 rpm.

The following equipment is under clearance:

- The ‘A’ Gland Steam Condenser Exhaust Fan is under clearance for motor replacement. Repairs are expected to be completed within 24 hours.
- PORV Block valve 1RC-113 is SHUT due to PZR PORV 444B Seat Leakage. The actions of Tech Spec 3.4.4 are met (block valve is shut). OWP-RC-02 has been completed.



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

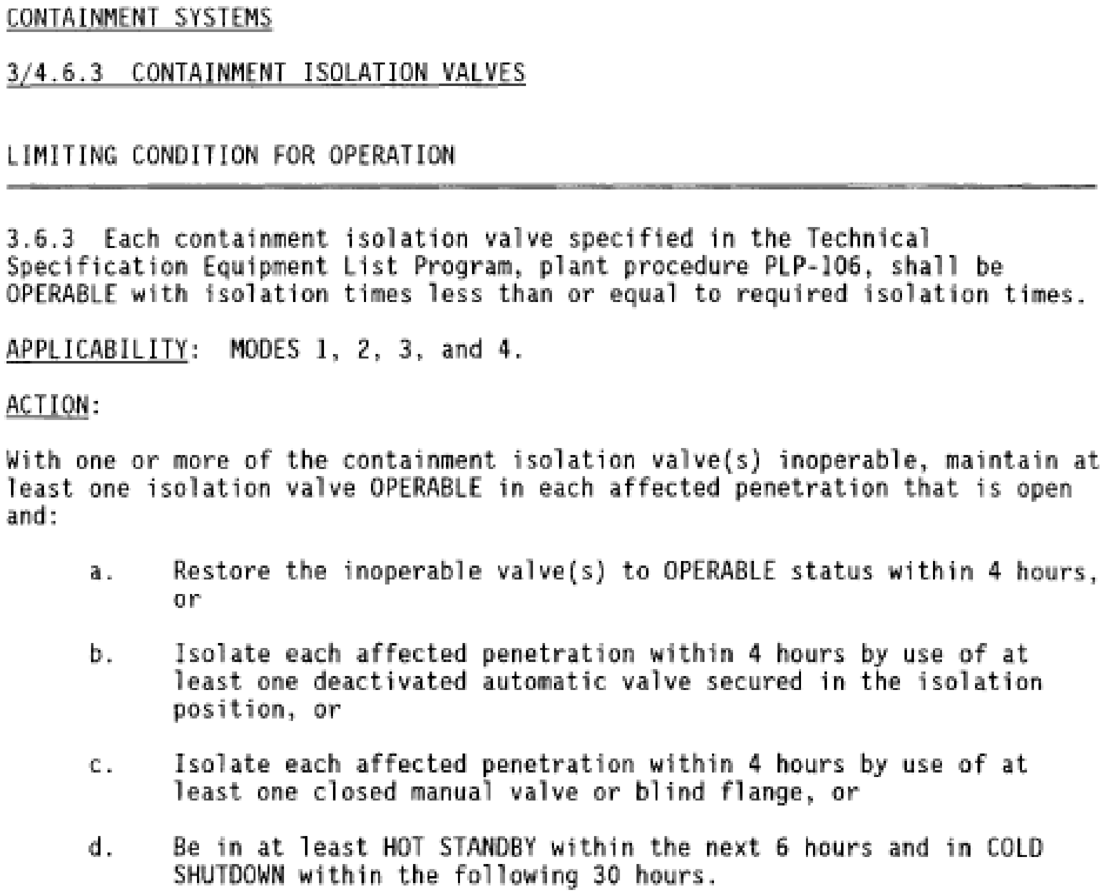
ACTION:

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.

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 2 (continued)

The following equipment is under clearance (continued):

- Letdown Orifice Isolation Valve 1CS-9 is under clearance for solenoid replacement. Tech Spec 3.6.3 LCO Action **b** applies. OWP-CS-09 has been completed.



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: 2018 NRC EXAM SCENARIO 2

Event 1: Start power escalation to 7% – 9% then transfer Turbine valve control from Throttle valves to Governor valves. Once Governor valves are controlling the Turbine the speed will be raised from 1700 rpm to 1800 rpm. (BOP – With DEH Main Turbine Controls in Manual, Competency - Operate plant controls in Manual). After Main Turbine speed is at 1800 rpm the crew will continue with GP-005. After completion of placing one DEH pump in Auto the Evaluator can continue with event 2.

Event 2: Failure of REM-01LT-3502ASA, CNMT RCS Leak Detection Radiation Monitor. This failure will cause the output to immediately fail high and RM-11 to go into high alarm. The automatic response to isolate Normal Containment Purge fails to occur due to a failed relay. The crew should respond to the alarms and enter AOP-005, Radiation Monitoring. AOP-005 Attachment 1 will direct verifying that the automatic response for this alarm has occurred (other procedure options are available and detailed in exercise guide). This will also require the SRO to evaluate Tech Spec 3.3.3.1 for the failed Containment Isolation and Tech Spec 3.4.6.1, Leakage Detection Systems.

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

- ACTION 26 - Must satisfy the ACTION requirement for Specification 3.4.6.1.
- ACTION 27 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge makeup and exhaust isolation valves are maintained closed.

Tech Spec 3.4.6.1 action

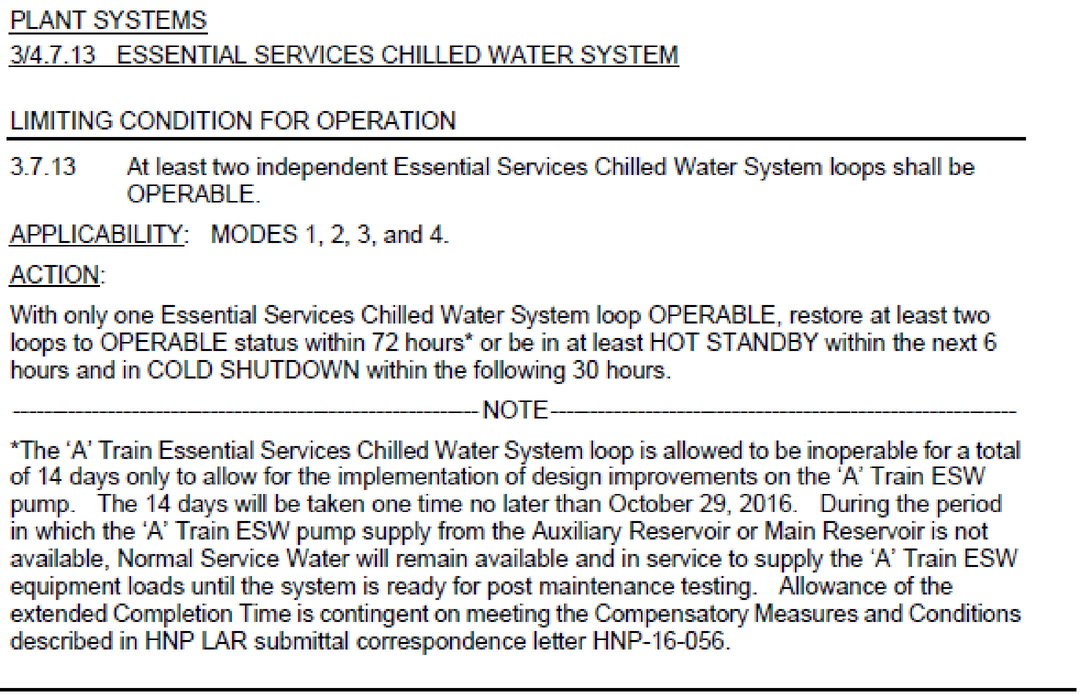
- With the Leakage Detection Systems INOPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for airborne gaseous and particulate radioactivity at least once per 24 hours when the required Airborne Gaseous or Particulate Radioactivity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

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

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 2 (continued)

Event 3: : "A" Emergency Services Chilled Water Pump Trip (AOP-026). The crew will respond to various alarms on ALB-023, diagnose the event, and enter AOP-026, Loss of Essential Chill Water System (no immediate actions). The SRO will direct the BOP to start the 'B' Train ESCWS Chiller in accordance with OP-148, Essential Service Chilled Water System. The crew should implement OWP-ECW-01 for the ESCW Chiller 1A-SA failure. The SRO should evaluate Tech Spec 3.7.13, Essential Services Chilled Water System and PLP-114, Relocated Technical Specifications and Design Basis Requirements – Attachment 4 for Area Temperature Monitoring. Note that the 'A' Chiller will be inoperable for the remainder of the scenario and this will impact plant response during the Major Event in that this failure will prevent Load Block 9 on sequencer Train 'A" from energizing. The SRO will commence OMM-001 Attachment 5 for the failure.

TS 3.7.13



PLANT SYSTEMS

3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

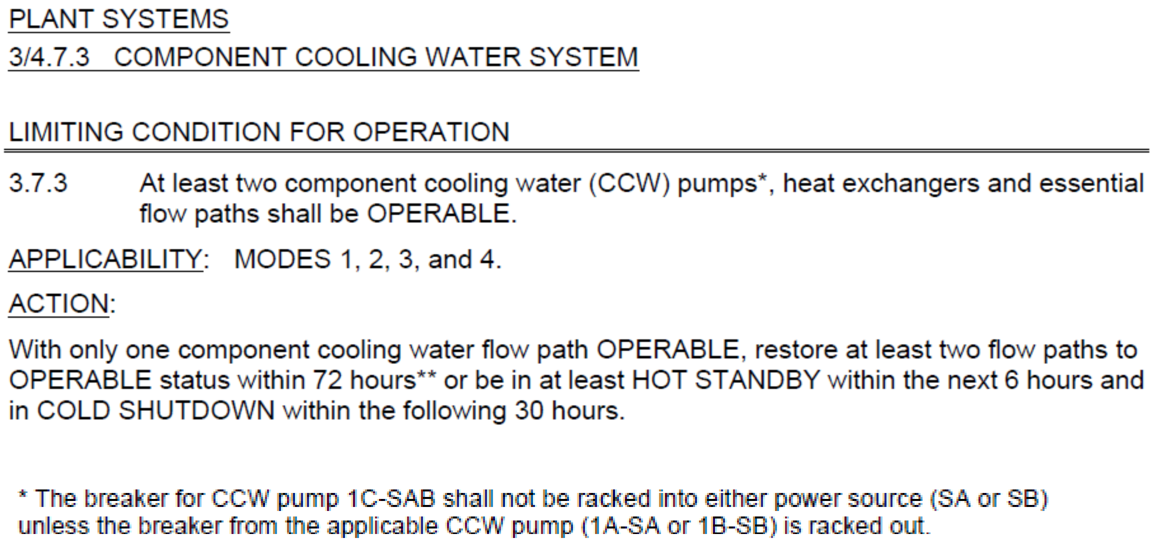
With only one Essential Services Chilled Water System loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

--NOTE--

*The 'A' Train Essential Services Chilled Water System loop 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 supply 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 loads 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 letter HNP-16-056.

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 2 (continued)

Event 4: : Trip of 'A' CCW Pump on O/C with standby CCW pump failure to auto start (AOP-014). The standby 'B' CCW Pump fails to Auto Start due to a pressure transmitter failure (instrument is isolated therefore pressure decrease is not sensed). The crew should recognize the loss and enter AOP-014, Loss of Component Cooling Water. AOP-014 will direct the restoration of the CCW system. The RO will be directed by the SRO to manually start the 'B' CCW (or will have started it in accordance with AD-OP-ALL-1000 when it did not auto start). The SRO should also commence OMM-001, Attachment 5 and evaluate Tech Spec 3.7.3, Component Cooling Water System.



PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

Event 5: Trip of AH-39 Containment Fan Coil Unit fan with back up auto start failure. The failure will cause annunciator ALB-029 4-5 "Containment Fan Coolers AH-39 Low Flow-O/L to alarm. The crew should identify that the standby fan did not auto start and start the standby fan.

Event 6: Pressurizer Spray Valve, PCV-444C, fails Open (***Manual Control available***). This failure will cause one of the Pressurizer spray valves to fail to 100% open while the other valve closes to 0% open. The crew should respond to multiple alarms and enter AOP-019, Malfunction of RCS Pressure Control. The RO should complete the immediate actions by gaining control of the Pressurizer Spray Valves. ***(RO – With PRZ Spray in Manual, Competency - Operate plant controls in Manual)***. A critical task is associated with this malfunction in that the crew must control PRZ Spray Valve, PCV-444C, prior to RCS pressure reaching the Safety Injection setpoint of 1850 psig. The justification for the critical task is based on the crew/individual causing an unnecessary plant trip or ESF actuation.

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 2 (continued)

Event 6: (continued) This malfunction may require the SRO to evaluate Technical Specification 3.2.5 (If RCS pressure decreases to < 2202 psig during the event)
3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System $T_{avg} \leq 594.8°F$ after addition for instrument uncertainty and
b. Pressurizer Pressure $\geq$ 2185 psig after subtraction for instrument uncertainty and
c. RCS total flow rate $\geq$ 293.540 gpm after subtraction for instrument uncertainty

- 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.

Event 7: Once RCS pressure control has been established, a Feed Line Break inside Containment on the ‘B’ SG will occur. The crew should enter and carry out the immediate actions of EOP-E-0. The crew should diagnose that there is not a LOCA in progress and transition to EOP-E-2, Faulted Steam Generator Isolation. RCS pressure will continue to reduce as the RCS cools down requiring securing RCPs in accordance with Foldout A.

Event 8: AFW Auto Isolation for the ‘B’ SG fails requiring the crew to manually isolate AFW flow. With ‘B’ SG pressure 100 psig below the other 2 SG’s pressure an automatic FW isolation should have occurred. When the BOP is asked to verify that isolation has occurred he/she will identify that the MDAFW AND TDAFW pump isolation valves to the ‘B’ SG (faulted SG) is NOT shut and will then shut both valves.

Event 9: Source Range channels will fail to energize due to under compensation of Intermediate Range channel N-35. The crew will need to identify the failure and then manually energize the SR channels to establish an audio count rate.

The scenario termination is met in EOP-E-2, when Safety Injection has been terminated. The scenario ends when the crew transitions to EOP-ES-1.1, SI Termination.

CRITICAL TASK JUSTIFICATION:

1. Control PRZ Spray Valve, PCV-444C, prior to RCS pressure reaching the Safety Injection setpoint of 1850 psig

Justification is based on the crew/individual causing an unnecessary plant trip or ESF actuation. See note below.

2. Isolate AFW flow to ‘B’ Steam Generator prior to any RCS Cold Leg Temperature lowering to less than 240°F after a RCS temperature drop in any cold leg of greater than 100° within the last hour.

Justification is based on WOG ERG-Based Critical task CT-17, Isolate the faulted SG before transition out of EOP-E-2 and based on identification of critical task from NuReg 1021 Rev. 11 Appendix D. Failure to isolate a faulted SG that can be isolated causes challenges to CSFs beyond those irreparably introduced by the postulated conditions (possible vessel brittle fracture concern). Also, depending upon the plant conditions, it could constitute a failure by the crew to “demonstrate the ability to recognize a failure or an incorrect automatic actuation of an ESF system or component. If the faulted SG is not isolated, the cooldown transient for reactor vessel inlet temperature could result in an ORANGE path challenge to the integrity CSF, especially if RCPs are not running.

3. Shut BIT Outlet valves 1SI-3 and 1SI-4 prior to PZR SRV’s Discharge Line High Temperature occurring (250°F on any Safety valve discharge line temperature indicator either TI-465, or TI-467 or TI-469)

Justification is based on NuReg 1021 Rev. 11 Appendix D – Take one or more actions that would prevent a challenge to plant safety. Shutting BIT outlet valves 1SI-3 and 1SI-4 prior to water relief through the PZR Safety Relief Valves (SRV’s). FSAR Section 15.1.5.2 (page 15.1.5-7) states the operator will secure one of the two CSIPs to facilitate PZR level indication remaining on scale and controllable. At low fluid temperature (like those present in the PZR at this time). SRV’s may not reset after fluid operation and if they will not shut RCS mass will be lost out a release path to the PRT which in turn will rupture the PRT and be released into Containment.

Note: Causing an unnecessary plant trip or ESF actuation may constitute a Critical Task failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

SIMULATOR SETUP

For the 2018 NRC Exam Simulator Scenario # 2

Reset to IC-162 password “NRC2018”

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

(The examiner has provided to the candidate with initial conditions and the initiating cues prior to placing the simulator in RUN.)

SPECIAL INSTRUCTIONS

Post conditions for status board from IC-8
Reactor Power 3% steady state
Control Bank D at 94 steps
RCS boron 1615 ppm
GP-005 step 84

- Update the status board: "A" MDAFW Pump is OOS for motor overhaul
  Pump has been OOS for 12 total hours and is expected back within the next 24 hours
  Tech Spec 3.7.1.2, 72 hour LCO or HSB within the next 6 hours, HSD following 6 hours

- ‘A’ Gland Seal Exhauster Fan is under clearance for motor repairs. The fan has been under clearance for 8 hours. Repairs are expected to be completed within 24 hours. Place a CIT on the MCB switch.

- PORV Block valve 1RC-113 is SHUT due to PZR PORV 444B Seat Leakage. Place an OFF NORMAL placard on the MCB switch. Place a completed copy of OWP-RC-02 in the OWP book.

- Letdown Orifice Isolation Valve 1CS-9 is under clearance for solenoid replacement. Tech Spec 3.6.3 LCO Action b applies. Place a completed copy of OWP-CS-09 in the OWP book. Place a CIT on the MCB switch.

- Hang restricted access signs on MCR entry swing gates

Op Test No.:	NRC	Scenario #	2	Event #	1	Page	10	of	70
Event Description:		**Start Power Escalation – Place Gov valves in Turbine Control**							

Time	Position	Applicant's Actions or Behavior

Lead Evaluator:	**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:		**May manually withdraw Control rods or perform RCS dilution:**
GP-005	CREW	Raise Reactor Power to ~ 7%-9% to support Main Turbine Roll
	BOP	Adjusts steam dump demand signal as necessary.
	RO	Withdraws Control Rods as necessary then initiates dilution per the reactivity plan with SRO concurrence
OP-104	RO	Withdraw Control Rods per OP-104, Section 5.4
	RO	Verifies Initial Conditions: • All shutdown rods have been withdrawn, per Section 5.3, by observing the Group Step Counters and Digital Rod Position Indication System. All Shutdown Rod Group Step Counters must read greater than or equal to 225 steps. • Reactivity evolution signs have been posted to limit MCR access.

Op Test No.: NRC	Scenario #	2	Event #	1	Page	11	of	70
Event Description:	Start Power Escalation – Place Gov valves in Turbine Control							

Time	Position	Applicant's Actions or Behavior

Procedure Note:		**Reactivity Evolution category to be determined by the CRS.**
	RO	Verifies At the MCB, the ROD BANK SELECTOR Switch in MAN.
	RO	VERIFY Rod Speed of 48 steps per minute on SI-408.
Procedure Note:		**During a Reactor Startup or testing, Steps 5.4.2.3 through 5.4.2.7 may be repeated multiple times, with rod motion stopped to observe reactivity affects, record 1/M data, or for other reasons. The intent is to initial for these Steps at the completion of the entire evolution, not for each time it is performed.**
	RO	At the MCB, POSITION ROD MOTION Switch to WITHDRAW. OBSERVE that the RODS OUT Direction Lamp lights.
	RO	OBSERVE Bank Step Counters for proper rod motion, overlap and sequencing.
	RO	VERIFY the rods are moving out by OBSERVING the Digital Rod Position Indication System Display.
	RO	At the MCB, STOP rod motion by RELEASING the ROD MOTION Switch allowing it to return to the neutral position. VERIFY the RODS OUT Direction Lamp extinguishes.

Op Test No.:	NRC	Scenario #	2	Event #	1	Page	12	of	70
Event Description:		**Start Power Escalation – Place Gov valves in Turbine Control**							

Time	Position	Applicant's Actions or Behavior
	RO	IF necessary, THEN REPEAT Steps 5.4.2.3 through 5.4.2.7.
OP-107.01	RO	Dilution per OP-107.01, Section 5.4
	RO	• DETERMINE the volume of makeup water to be added using the reactivity plan associated with the IC.
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.**
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 pre-set value.**
	RO	SET FIS-114, TOTAL MAKEUP WTR BATCH COUNTER, to obtain the desired quantity.
	RO	• VERIFY the RMW CONTROL switch has been placed in the STOP position. • VERIFY the RMW CONTROL switch green light is lit. • PLACE control switch RMW MODE SELECTOR to the ALT DIL position.

Op Test No.:	NRC	Scenario #	2	Event #	1	Page	13	of	70
Event Description:		**Start Power Escalation – Place Gov valves in Turbine Control**							

Time	Position	Applicant's Actions or Behavior
Procedure Note:		**Alternate Dilution may be manually stopped at any time by turning the control switch RMW CONTROL to STOP.**
	RO	START the makeup system as follows: • TURN control switch RMW CONTROL to START momentarily. • VERIFY the red indicator light is lit. • IF expected system response is not obtained, THEN TURN control switch RMW CONTROL to STOP.
	RO	**VERIFY** dilution automatically terminates when the desired quantity has been added. **MONITOR** Tavg and rod control for proper operation. **TURN** control switch RMW MODE SELECTOR to AUTO.
	RO	• START the makeup system as follows: o TURN control switch RMW CONTROL to START momentarily. o VERIFY the RED indicator light is LIT. • Reports to CRS that dilution is complete and Makeup is back in AUTO

Op Test No.: NRC	Scenario # 2	Event # 1	Page 14 of 70
Event Description:	**Start Power Escalation – Place Gov valves in Turbine Control**		

Time	Position	Applicant's Actions or Behavior
GP-005	CREW	As power is raised above 5% identifies entry into Mode 1
	SRO	Completes step 57 in GP-005 Directs BOP to perform Step 84, TRANSFER control from the Throttle Valves to the Governor Valves
	BOP	Verifies Main Turbine speed on DEH control panel indicates the Turbine is at 1700 RPM then transfers control from the Throttle Valves to the Governor Valves by depressing the TRANSFER TV-GV pushbutton.
	BOP	CHECK that the transfer from the Throttle Valves to the Governor Valves is complete by checking the following indications: • Valve position indicators • TRANSFER TV light extinguished • GV light illuminated • Local observation (Throttle Valves smoothly transition to full open)
Simulator Communicator:		**For local observation of the Throttle Valves operation as Turbine Building AO report smooth operation to the full open position.**
	BOP	ENTER 1800 RPM into the DEMAND display AND VERIFY the HOLD pushbutton is illuminated. ***(Operate the DEH Main Turbine Controls in Manual – Competency – Operate plant controls in Manual)***

Op Test No.:	NRC	Scenario #	2	Event #	1	Page	15	of 70
Event Description:	**Start Power Escalation – Place Gov valves in Turbine Control**							

Time	Position	Applicant's Actions or Behavior
Procedure Note:		**The REFERENCE display will count up to 1800 RPM at the previously selected acceleration rate, and then the GO pushbutton will extinguish.**
	BOP	Depresses the GO pushbutton.
	BOP	Ensures the Main Turbine speed stops increasing at 1800 rpm AND the GO pushbutton extinguishes.
	BOP	At 1800 RPM, LOWER the Valve Position Limiter, as indicated in the REFERENCE display, until it indicates the percent (%) value read in the DEMAND display plus an additional 2%.
	BOP	At 1800 RPM locates the controls for the BRG OIL & SEAL OIL BU Pump from Main RSVR and STOPS the BRG OIL & SEAL OIL BU FROM MAIN RSVR Pumps, then place the control switch in AUTO.
	BOP	PLACE one DEH Pump in AUTO (Standby) operation.
Lead Evaluator:		**Cue Event 2 – Radiation Monitor 3502A high alarm and Containment Purge fails to isolate automatically, when satisfied with power escalation performance.**

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	16	of	70

Event Description: **Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically**

Time	Position	Applicant's Actions or Behavior

Simulator Operator:	**On cue from Lead Evaluator insert Trigger 2** **Radiation monitor 3502A failure**

Indications Available		**ALB-10-4-5, RAD MONITOR SYSTEM TROUBLE**
	RO	Responds to ALB-10-4-5, RAD MONITOR SYSTEM TROUBLE. (APP response below)
Simulator Communicator:		**If HP contacted to validate alarm wait one minute and then report that the monitor has failed.** **If someone other than HP is dispatched to investigate, wait three minutes and then report REM-3502 Gas Channel failed – no power, no indication.**
Evaluator Note:		**There are automatic actions associated with the failed channel that have been blocked by malfunction. The BOP may take the actions to place equipment in the required position from directions in AOP-005 or do so IAW OWP-RM-03.**
The first section of this guide is written to the response of the APP and then AOP-005, Radiation Monitoring System. The second part is written as if it will be done in the OWP which provides minor additional actions not contained in the AOP.		
	CREW	APP-ALB-010-4-5 response: CONFIRM alarm using: • RM-23, Radiation Monitoring Panel

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	17	of	70

Event Description: **Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically**

Time	Position	Applicant's Actions or Behavior
	BOP	VERIFY Automatic Functions: • Automatic Actions are dependent upon which RM-23 Radiation Monitor is in ALARM
	CREW	PERFORM Corrective Actions: • IF the alarm is a Fuel Handling Building High Radiation alarm, THEN MANUALLY START the Spent Fuel Pool Purification System, using OP-116.01, Fuel Pool Cooling Purification System. (NO)
	SRO	• IF the alarm is RM-21AV-3509-1SA or an Area Monitor in the vicinity of the VCT Valve Gallery and air is being purge from the VCT to the plant vent per OP-120.07, THEN MANUALLY SECURE the air purge from the VCT to the plant vent per OP-120.07. (NO) • IF any radiation monitor is in alarm condition, THEN GO TO AOP-005, Radiation Monitoring System. (YES) • IF maintenance is to be performed, THEN REFER TO OWP-RM, Radiation Monitoring. (maintenance will be required) • Diagnoses as a failure of Channel 3502A (GAS) (May diagnose early)

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	18 of	70
Event Description:		**Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically**						

Time	Position	Applicant's Actions or Behavior
	SRO	Enters AOP-005, Radiation Monitoring System Makes PA announcement (No Immediate Actions)
	SRO	CHECK radiation levels NOT in HIGH ALARM: • Area Radiation Monitors (YES - Not in high Alarm) • In-Plant Airborne Radiation Monitors (YES - Not in high Alarm) • NOTIFY Health Physics to perform the following: • a. EVALUATE ANY alarm received using HPP-780, Radiation Monitoring Systems Operator's Manual. • b. IF necessary, THEN SURVEY the affected area.
Simulator Communicator:		**When notified acknowledge request to investigate alarm using HPP-780.**
	SRO	• CHECK ALL Stack Monitor radiation levels NOT in ALARM. (YES – Not in Alarm) • CHECK ALL Process Monitors NOT in ALARM. (YES – Not in Alarm) • REFER TO the following: • Tech Spec Section 3.3.3.1 – Action 27 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge makeup and exhaust isolation valves are maintained closed. • Tech Spec Section 3.3.2 Functional Unit 3.4.c • Tech Spec section 3.4.6.1.a – 30 day

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	19	of	70

Event Description: **Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically**

Time	Position	Applicant's Actions or Behavior
	SRO	• REFER TO the applicable attachment based on the affected area or system monitors: • Containment Monitors – **Attachment 1 p. 8**
	SRO	AOP-005, Attachment 1 • IF the plant is in Mode 5 or 6, THEN PERFORM the following: • (N/A plant in Mode 1) • IF Containment Ventilation Isolation has actuated, THEN VERIFY proper equipment alignment using OMM-004, Post-Trip/Safeguards Actuation Review. (NO) • IF REM-01LT-3502ASA, Cnmt RCS Leak Detection Monitor, is in HIGH ALARM, THEN VERIFY Normal Containment Purge is ISOLATED, as follows: • VERIFY BOTH Cnmt Normal Purge Supply Fans are STOPPED: ▪ AH-82 A ▪ AH-82 B • VERIFY ALL Cnmt Normal Purge Inlet/Discharge Dampers are SHUT: ▪ 1CP-5 SA ▪ 1CP-9 SA ▪ 1CP-3 SB ▪ 1CP-6 SB

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	20	of 70

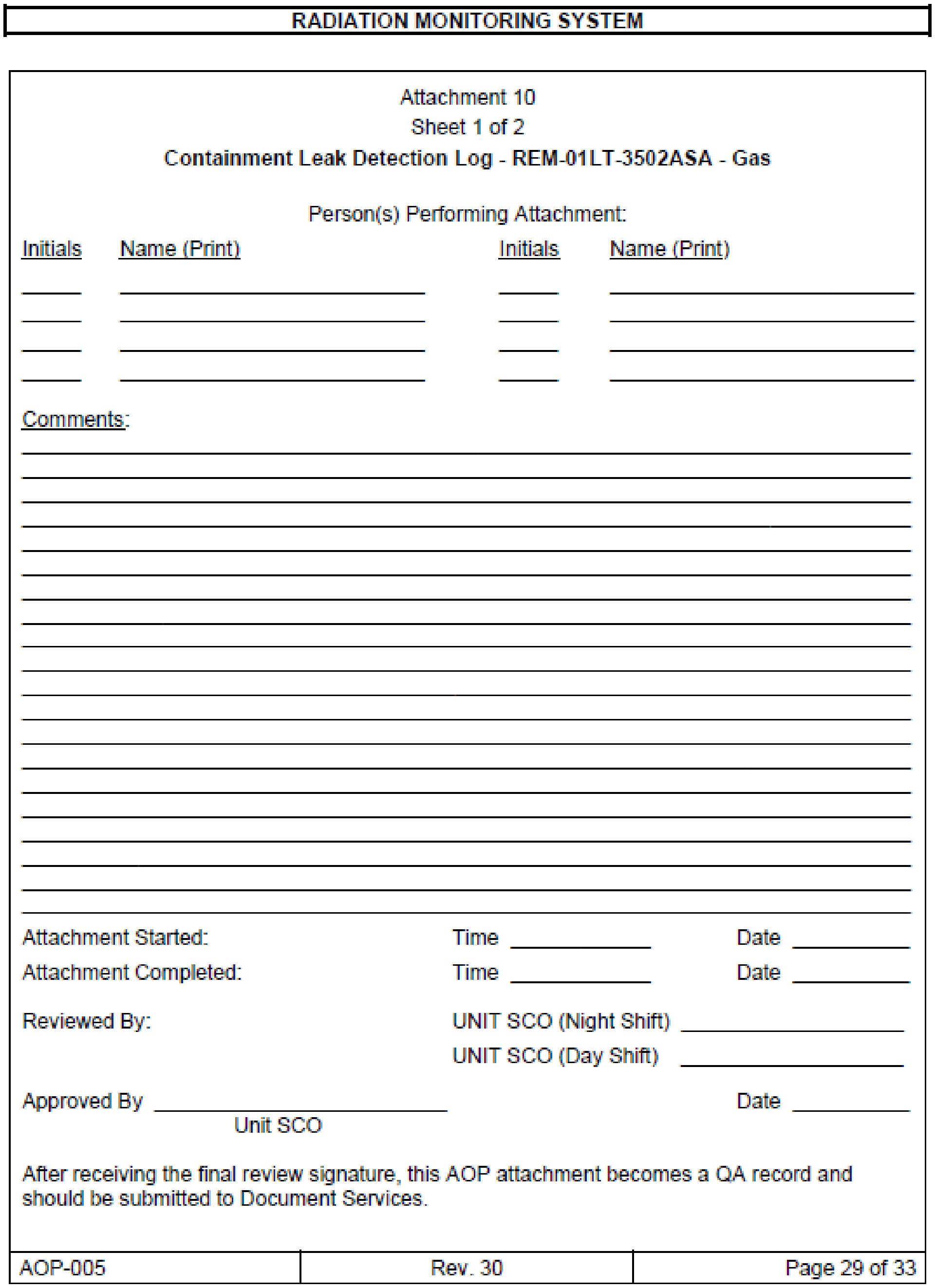
Event Description: **Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically**

Time	Position	Applicant's Actions or Behavior
	BOP	• Places AH-82A, Normal Containment Supply Fan, in STOP and releases • Places AH-82B, Normal Containment Supply Fan, in STOP and releases • Verifies 1CP-5, Normal Purge Inlet – CLOSED • Verifies 1CP-9, Normal Purge Inlet – CLOSED • Verifies 1CP-3, Normal Purge Discharge – CLOSED • Verifies 1CP-6, Normal Purge Discharge – CLOSED
	SRO	• Notes that no further actions in AOP-005 Att. 1 are applicable. Reviews the remainder of the section and reaches step to EXIT procedure • Direct BOP to perform Attachment 10, Containment Leak Detection Log for REM-01LT-3502ASA – Gas (included in following pages)

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	21	of	70

Event Description: **Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically**

Time	Position	Applicant's Actions or Behavior



RADIATION MONITORING SYSTEM

Attachment 10
Sheet 1 of 2
Containment Leak Detection Log - REM-01LT-3502ASA - Gas

Person(s) Performing Attachment:

Initials	Name (Print)	Initials	Name (Print)

Comments:

Attachment Started: Time ____________ Date __________

Attachment Completed: Time ____________ Date __________

Reviewed By: UNIT SCO (Night Shift) ___________________

UNIT SCO (Day Shift) ___________________

Approved By _________________________ Date __________
Unit SCO

After receiving the final review signature, this AOP attachment becomes a QA record and should be submitted to Document Services.

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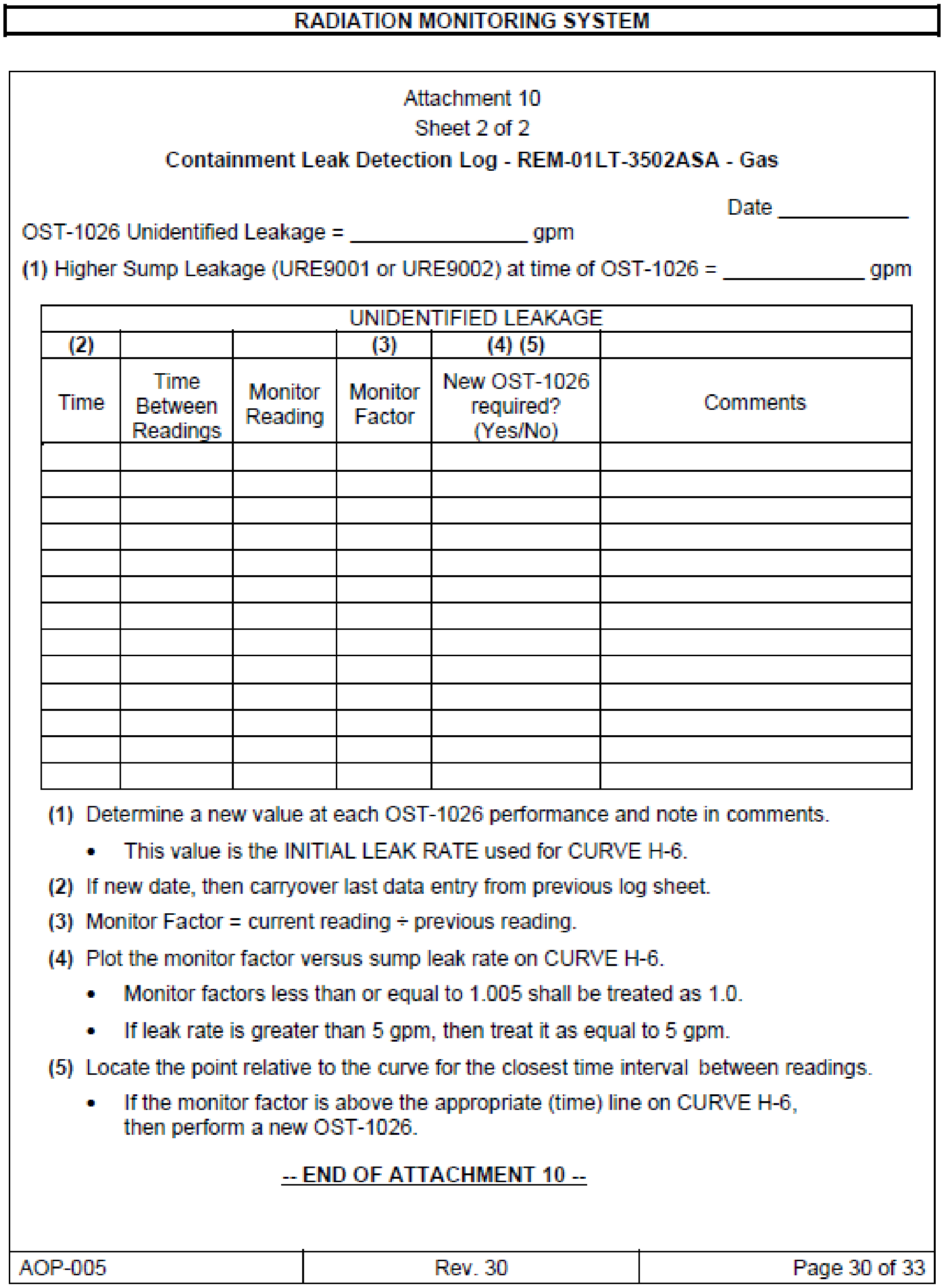
Op Test No.: NRC	Scenario #	2	Event #	2	Page	22 of	70
Event Description:	**Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically**						

Time	Position	Applicant's Actions or Behavior



RADIATION MONITORING SYSTEM

Attachment 10
Sheet 2 of 2

Containment Leak Detection Log - REM-01LT-3502ASA - Gas

Date ___________

OST-1026 Unidentified Leakage = _______________ gpm

(1) Higher Sump Leakage (URE9001 or URE9002) at time of OST-1026 = ____________ gpm

UNIDENTIFIED LEAKAGE					
(2)			**(3)**	**(4) (5)**	
Time	Time Between Readings	Monitor Reading	Monitor Factor	New OST-1026 required? (Yes/No)	Comments

(1) Determine a new value at each OST-1026 performance and note in comments.

- This value is the INITIAL LEAK RATE used for CURVE H-6.

(2) If new date, then carryover last data entry from previous log sheet.

(3) Monitor Factor = current reading ÷ previous reading.

(4) Plot the monitor factor versus sump leak rate on CURVE H-6.

- Monitor factors less than or equal to 1.005 shall be treated as 1.0.
- If leak rate is greater than 5 gpm, then treat it as equal to 5 gpm.

(5) Locate the point relative to the curve for the closest time interval between readings.

- If the monitor factor is above the appropriate (time) line on CURVE H-6, then perform a new OST-1026.

-- END OF ATTACHMENT 10 --

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	23	of	70

Event Description: **Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically**

Time	Position	Applicant's Actions or Behavior

Evaluator Note:		**The following section is utilized if AOP-005 actions are not utilized:**
	SRO	Implement OWP-RM-03, CONTAINMENT LEAK DETECTION RADIATION MONITORS.
	BOP	Performs OWP-RM-03 component lineup:
Procedure Caution:		**The control switches for AH-82A and AH-82B must be taken to STOP momentarily to ensure they will not AUTO start.**
	BOP	• Places AH-82A, Normal Containment Supply Fan, in STOP and releases. • Places AH-82B, Normal Containment Supply Fan, in STOP and releases. • Verifies 1CP-6, Normal Purge Inlet – CLOSED. • Verifies 1CP-9, Normal Purge Inlet – CLOSED. • Verifies 1CP-3, Normal Purge Discharge – CLOSED • Verifies 1CP-5, Normal Purge Discharge – CLOSED • Contact AO to place 1D21-2B, AH-82 (1A-NNS) Normal Containment Purge Makeup Air Handler breaker in OFF • Contact AO to place 1E21-2F, AH-82 (1B-NNS) Normal Containment Purge Makeup Air Handler breaker in OFF
Simulator Communicator:		If contacted acknowledge request to place breakers to OFF
Simulator Operator:		RF HVA052 BRK_OFF, RF HVA053 BRK_OFF

Op Test No.:	NRC	Scenario #	2	Event #	2	Page	24 of	70
Event Description:		**Radiation Monitor 3502A high alarm, Containment Purge fails to isolate automatically**						

Time	Position	Applicant's Actions or Behavior

	SRO	Review/prepare OWP-RM-03 LCO Action Log. Contacts support personnel for repairs.
	SRO	• Enters TS 3.3.3.1, Action b • Table 3.3-6: o Action 26 - Must satisfy the ACTION requirement for Specification 3.4.6.1 and; o Action 27 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge makeup and exhaust isolation valves are maintained closed). • Enters TS 3.4.6.1, Action a - With a. or c. of the above required Leakage Detection Systems inoperable: o Can operate up to 30 days o Obtain and analyze a grab sample of the containment atmosphere for gaseous and particulate radioactivity at least once per 24 hours o Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
	SRO	Reviews/prepares OMM-001, Attachment 5 Equipment Problem Checklist for the failure of the radiation monitor. Contacts support personnel for repairs.
Evaluator Note:		**After the Radiation Monitor failure is addressed insert Event 3 "Trip of the running ESCWS Chiller WC-2 A-SA"**

Op Test No.: NRC	Scenario # 2	Event # 3	Page 25 of 70
Event Description:	**'A' Essential Services Chilled Water Pump Trip**		

Time	Position	Applicant's Actions or Behavior

Simulator Operator:		**When directed by Lead Evaluator: Actuate Trigger 3 "Trip of the running ESCWS Chiller WC-2 A-SA"**
Indications Available:		• **ALB-23-1-18 CHILLER WC2-A TROUBLE**
	BOP	• RESPONDS to alarm on ALB-23 (1-18). • REPORTS WC-2A-SA tripped.
AOP-026		LOSS OF ESSENTIAL SERVICE CHILLED WATER SYSTEM
	SRO	ENTERS AOP-026, LOSS OF ESSENTIAL SERVICE CHILLED WATER SYSTEM Makes PA announcement for AOP entry
Procedure Note:		**This procedure contains no immediate actions.**
	BOP	CHECK the in-service chiller RUNNING. (NO)
	CREW	DISPATCH an AO to determine the cause of the chiller trip.
Simulator Communicator:		**When contacted, wait 2 minutes and then the TB AO report that the breaker for the chiller has tripped on overcurrent and as the RAB AO report that there are no visible problems locally at the chiller.**
	BOP	PERFORM the following using OP-148, Essential Service Chilled Water System: START the Standby chiller (Start P-4B and 'B' Chiller) section 5.1 or 5.2 of OP-148. NOTE: IF ESW Header Temps are > 92°F then OP-148 will require a start of an ESW pump to support the Chiller start.

Op Test No.:	NRC	Scenario #	2	Event #	3	Page	26 of	70
Event Description:	**'A' Essential Services Chilled Water Pump Trip**							

Time	Position	Applicant's Actions or Behavior

Evaluator Note:		**OP-148 sections 5.1 and 5.2 can be found at the end of the guide in Attachment 1.** **Section 5.2 of OP-148 may be used if crew determines that loss will be short term.**
Simulator Communicator:		**If contacted, report "Pre-start checks on P-4B and 'B' Chiller are complete." No simulator booth operations are required.**
OP-148, Section 5.1 or Section 5.2		NOTE: Due to crew preference the OP-148 sections are located at the end of this guide in Attachment 1. The BOP will perform the actions of the OP procedure.
Simulator Communicator:		**If contacted, report "Pre-start checks on 'B' Emergency Service Water Pump are complete." No simulator booth operations are required.**
Simulator Communicator:		**IF contacted by the BOP to RESET the Low Chilled Water Flow alarm, wait 15 seconds and then report "The Low Chilled Water No Flow Alarm has been reset, and there are no other alarms." There are NO simulator operations required.**
	CREW	Makes a PA announcement prior to starting chiller. Starts Chiller
AOP-026 (step 5)	CREW	CONTACT Maintenance as necessary for troubleshooting and appropriate corrective actions.
Evaluator NOTE:		**Chiller start is delayed for 30 seconds after switch is placed in start.**

Op Test No.:	NRC	Scenario #	2	Event #	3	Page	27	of	70
Event Description:		**'A' Essential Services Chilled Water Pump Trip**							

Time	Position	Applicant's Actions or Behavior

AOP-026 (step 6)	BOP	CHECK EITHER chiller STARTED. (YES)
AOP-026 (step 7)	SRO	Step 7 GO to step 16
AOP-026 (steps 16 and 17)	BOP	VERIFY the following AH units for the operating train chiller are RUNNING: • AH-15, Control Room Normal Supply • AH-17, Fuel Vent FP Pump Room Fan Cooler • AH-16, Elec Equip Prot Rm Supply VERIFY the following alarm is CLEAR for the running chiller • ALB-23-1-20, Expansion TK A LO-LO Level • ALB-23-2-20, Expansion TK B LO-LO Level
AOP-026 (step 18)	SRO	REFER TO Tech Spec 3.7.13. At least two independent Essential Services Chilled Water System loops shall be OPERABLE. • ACTION: With only one ESCW System loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HSB within the next 6 hours and in CSD within the following 30 hours.
	SRO	• Contacts WCC for Work Request and LCO Tracking Record. Contacts Maintenance to investigate and fills out an Equipment Problem Checklist. • Obtains OWP-ECW • Direct BOP to perform Train Swap • EXIT this procedure.
Evaluators Note:		**After the ESCWS Chiller is running and the Evaluators have seen enough of the event - Initiate Event 4 "A" CCW Pump Trips**

Op Test No.:	NRC	Scenario #	2	Event #	4	Page	28 of	70
Event Description:	Trip of Running CCW Pump, ‘A’							

Time	Position	Applicant’s Actions or Behavior

Simulator Operator:	**On cue from the Lead Evaluator insert Trigger 4** **Trip of the “A” CCW Pump**

Evaluator Note:	**This event is a trip of the running ‘A’ CCW Pump. The standby ‘B’ CCW Pump fails to Auto Start due to a pressure transmitter failure. The crew should recognize the loss and enter AOP-014, Loss of Component Cooling Water and/or Manually start ‘B’ CCW pump IAW AD-OP-ALL-1000 guidance which allows the operator to take MANUAL actions when automatic actions do not occur**

Available Indications		**Multiple CCW alarms on ALB-005**
AOP-014	SRO	ENTER AOP-014, Loss of Component Cooling Water No Immediate Actions
Procedure Note:		This procedure contains no immediate actions. Loss of CCW may require implementation of the SHNPP Emergency Plan.
	SRO	Directs SM to REFER TO PEP-110, Emergency Classification And Protective Action Recommendations, AND ENTER the EAL Matrix.
	SRO	EVALUATE plant conditions AND GO TO the appropriate section. (Section 3.3, Loss of a CCW Pump)
Procedure Note:		The standby CCW pump starts at 52 psig discharge pressure.

Op Test No.: NRC	Scenario # 2	Event # 4	Page 29 of 70
Event Description:	Trip of Running CCW Pump, 'A'		

Time	Position	Applicant's Actions or Behavior
	RO	CHECK the standby CCW pump has STARTED. (NO) Dispatch an operator to investigate
Simulator Communicator:		**If dispatched to the field to investigate report back after 2-3 minutes that 'A' CCW Pump breaker is tripped on overcurrent on "C" Phase.**
	RO	START the standby CCW pump.
	RO	CHECK ALL RCPs operating within the limits of Attachment 1. (YES)
	RO	CHECK CCW header pressure greater than 52 psig. (YES)
	RO	VERIFY adequate ESW cooling water flow to the associated CCW heat exchanger. (YES)
	RO	CHECK RHR operating. (NO)
	SRO	REFER TO Technical Specification 3.7.3 • With only one component cooling water flow path OPERABLE. restore at least two flow paths to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
	SRO	CONTACT Maintenance to determine the cause of the CCW pump failure, AND INITIATE corrective action.

Op Test No.:	NRC	Scenario #	2	Event #	4	Page	30 of	70
Event Description:	Trip of Running CCW Pump, 'A'							

Time	Position	Applicant's Actions or Behavior
	SRO	CHECK with Operations Staff to determine the desirability of using the swing CCW pump.
	SRO	CHECK CCW flow RESTORED to the affected train.
	Crew	May dispatch Aux Operator to Open the control power knife switch for the 'A' CCW pump.
Simulator Communicator / Operator		**Acknowledge request.** **Open control power knife switch on 'A' CCW pump then contact MCR that control power has been removed.**
Evaluator Note:		**Crew may implement OWP-CC at this point. This OWP will have the crew verify the ESF Status Light Boxes.** **The implementation of the OWP is not required to continue with the scenario.**
	SRO	EXIT this procedure.
Evaluator Cue:		**After the completion of AOP-014, cue Simulator Operator to insert Trigger 5** **Event 5 – AH-39 Containment Fan trip**

Op Test No.:	NRC	Scenario #	2	Event #	5	Page	31	of 70
Event Description:			**Trip of AH-39**					

Time	Position	Applicant's Actions or Behavior

Simulator Operator:	**On cue from the Lead Evaluator actuate Trigger 5:** **AH-39 Containment Fan Coil Unit Fan trip**

Indications Available		• **ALB-029-4-5 CONTAINMENT FAN COOLERS AH-39 LOW FLOW - O/L** • **Increasing 'C' RCP stator winding temperatures**
	BOP	RESPONDS to alarms and ENTERS APP-ALB-029-4-5
	BOP	• CONFIRM alarm using: o AH-39 fans running indication (NO) o Damper position indication (YES) • VERIFY Automatic Functions: o Running fan trips (YES) o Backup fan starts (NO) (BOP starts the standby fan, may utilize OP-169 section 5.2) • PERFORM Corrective Actions: o CHECK standby fan STARTS AND lead fan STOPS. o DISPATCH an operator to check status of the following breakers: ▪ 1D1-1A, AH-39 (1A-NNS) CNMT Fan Coil ▪ 1E1-7C, AH-39 (1B-NNS) CNMT Fan Coil
Simulator Communicator:		**Three minutes after being dispatched to check the breaker for 1D1-1A, AH-39 (1A-NNS) CNMT Fan cooler breaker, report that:** ***"The indications on the Static Trip Unit show that an Overload Condition occurred for AH-39 A fan. There are no abnormalities on the AH-39B breaker."***

Op Test No.:	NRC	Scenario #	2	Event #	5	Page	32 of	70
Event Description:				**Trip of AH-39**				

Time	Position	Applicant's Actions or Behavior
	BOP	○ IF any breaker has tripped on OVERLOAD or SHORT CIRCUIT as indicated on the Static Trip Unit, THEN PERFORM the following: **(Directs AO to perform based on report from communicator)** ▪ DEPRESS the breaker Alarm Reset. ▪ RACK OUT the breaker using OP-156.02, AC Electrical Distribution. ▪ VERIFY cause of the over current trip is determined prior to returning the breaker to service.
Simulator Communicator:		**Acknowledge request to perform directed actions** **at 1D1-1A**
Simulator Operator:		**Rack out breaker 1D1-1A for AH-39** **Run AMS file "AH39ARackedOut"** **This will override the switch to STOP and turn off the RED and GREEN MCB switch lights.** **Have communicator report back after running file.**
	RO	Monitors RCP "C" parameters on ERFIS and or OSI PI
	SRO	Reviews/prepares OMM-001, Attachment 5 Equipment Problem Checklist for the failure of AH-39. Contacts support personnel for repairs.
Evaluator Cue:		**When breaker racking is completed, cue Simulator Operator to insert Trigger 6** **Event 6 - Pressurizer Spray Valve PCV-444C fails open**

Op Test No.:	NRC	Scenario #	2	Event #	6	Page	33	of	70
Event Description:		**Pressurizer Spray Valve, PCV-444C, fails OPEN (w/manual control available)**							

Time	Position	Applicant’s Actions or Behavior

Simulator Operator:	**On cue from Lead Evaluator actuate Trigger 6** **Pressurizer Spray Valve, PCV-444C, fails Open**

Evaluator Note:		**When Pressurizer Spray Valve PCV-444C fails open, PZR pressure will decrease and all PZR heaters will energize. Annunicators for PZR low pressure will alarm. The crew should respond by entering AOP-019, Malfunction of RCS Pressure Control, and placing the malfunctioning spray valve in manual per the immediate actions. RCS pressure may drop below the DNB limit depending on how fast the operator responds to the failure. If so, the SRO should evaluate Tech Spec 3.2.5, DNB Parameters.**
Indications Available		• **ALB-09-3-3 PRZ CONT LOW PRESS AND HEATERS ON** • **ALB-09-5-1 PRESSURIZER HIGH-LOW PRESS** • **Pressurizer Pressure decreasing**
	RO	• Responds to ALB-09 alarms. • Reports malfunction in the RCS Pressure Control system.

Op Test No.:	NRC	Scenario #	2	Event #	6	Page	34	of	70
Event Description:	**Pressurizer Spray Valve, PCV-444C, fails OPEN (w/manual control available)**								

Time	Position	Applicant’s Actions or Behavior

	SRO	Enters AOP-019, MALFUNCTION OF RCS PRESSURE CONTROL. Makes PA announcement
Immediate Actions	RO	Perform AOP-019 Immediate Actions. • CHECK that a bubble exists in the PRZ. (YES) • VERIFY ALL PRZ PORVs AND associated block valves properly positioned for current PRZ pressure and plant conditions. (YES) • CHECK Both PRZ spray valves properly positioned for current PRZ pressure and plant conditions. (NO)
Evaluator Note:		**The malfunction only affects PCV-444C. It is expected that the operator will recognize that only one spray valve is malfunctioning and operate that controller in MANUAL.**
Immediate Actions **Critical Task #1**	RO	• CONTROL PRZ spray valves in MANUAL using ONE of the following (listed in order of preference): • AFFECTED Spray Valve controller in MANUAL (if only one is obviously malfunctioning) OR • PK-444A, Master Pressure Controller OR • Both individual spray valve controllers Reports IAs complete *(Critical Task - Control PRZ Spray Valve, PCV-444C, prior to RCS pressure reaching the SI setpoint of 1850 psig)*

Op Test No.:	NRC	Scenario #	2	Event #	6	Page	35	of	70
Event Description:	**Pressurizer Spray Valve, PCV-444C, fails OPEN (w/manual control available)**								

Time	Position	Applicant’s Actions or Behavior
	SRO	• GO TO Section 3.1, Pressure Control Malfunctions While Operating With a Pressurizer Bubble. • Inform SM to REFER to PEP-110, Emergency Classification and Protective Action Recommendations, AND ENTER the EAL Matrix.
	RO	MONITOR PRZ pressure by observing other reliable indications.
	SRO	CHECK plant in MODE 1 OR 2. (YES)

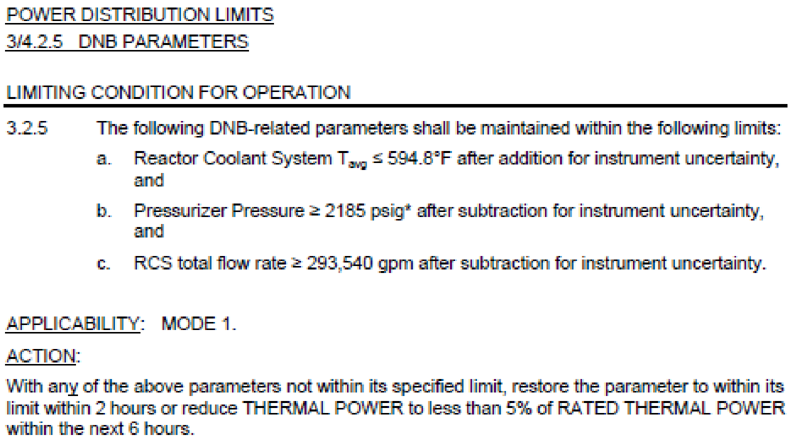
Op Test No.:	NRC	Scenario #	2	Event #	6	Page	36 of	70

Event Description: **Pressurizer Spray Valve, PCV-444C, fails OPEN (w/manual control available)**

Time	Position	Applicant's Actions or Behavior
	RO	• CHECK PRZ pressure CONTROLLED. (YES) • CHECK PRZ pressure 2335 PSIG OR LESS. (YES) • CHECK ALL of the following PRZ PORV block valves OPEN: o 1RC-117 (for PCV-445A SA) (YES) o 1RC-115 (for PCV-445B) (YES) o 1RC-113 (for PCV-444B SB) (YES) • CHECK that a malfunction of one or more of the following has occurred: o PT-444 (NO) o PK-444A (NO) o PRZ heater(s) (NO) o PRZ spray valve(s) or controller(s) (YES 1RC-107 failed while in AUTO) • CHECK PK-444A controlling properly in AUTO. (YES) • CONTROL PRZ pressure as follows: o CHECK BOTH PRZ spray valve controllers in AUTO AND BOTH spray valves operating as desired. (NO) o VERIFY PRZ Spray Valve controllers in ONE of the following alignments: ▪ AFFECTED Spray Valve controller in MANUAL (if only one is obviously malfunctioning) (YES) o OPERATE Spray Valves as necessary to control PZR pressure. ***(RO – PRZ spray valve control in Manual - Competency - Operate plant controls in Manual).*** o CHECK ALL PRZ heaters operating as desired. (YES) • CHECK at least one of the following conditions present: o PRZ pressure is UNCONTROLLED (NO) o Status of a normal spray valve or a PRZ heater bank is UNCONTROLLED (NO)

Op Test No.:	NRC	Scenario #	2	Event #	6	Page	37 of 70

Event Description: **Pressurizer Spray Valve, PCV-444C, fails OPEN (w/manual control available)**

Time	Position	Applicant’s Actions or Behavior
	SRO	• REFER TO Tech Spec 3.2.5 (DNB Parameters) AND IMPLEMENT action where appropriate.  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} ≤ 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. (Limit is 2185 psig – restore within 2 hours) • Commences an Equipment Problem Checklist and contacts WCC for assistance. (WR, LCO Tracking Record and Maintenance support)
Evaluator Note:		**The Lead Evaluator can cue Event 7 (Feedline break on ‘B’ SG inside Containment) once the plant has stabilized back in its normal pressure band.**

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	38	of	70
Event Description:			**‘B’ SG Feedline Break Inside Containment**						

Time	Position	Applicant’s Actions or Behavior

Evaluator Note:	**A Feedline Break inside Containment from the ‘B’ SG will occur requiring tripping the Reactor and entry into EOP-E-0. The crew will initiate a MSL Isolation. The crew should diagnose that a LOCA is NOT in progress and transition to EOP-E-2, Faulted Steam Generator Isolation.** **AFW isolation will not occur for the ‘B’ SG, requiring manual action to isolate the AFW flow to the ‘B’ SG.** **Source Range channel NI-31 will fail to energize due to IR NI-35 compensating voltage failure.**
Simulator Operator:	**On cue from the Lead Evaluator, insert Trigger 7 (Feedline break inside Containment)**

Indications Available		• **Multiple alarms on ALB-014 associated with the B SG** • **Lowering level in the ‘B’ SG** • **“B” SG FF/STM Flow mismatch** • **Containment press/temp and humidity increasing** • **Containment Sump level increasing**
	CREW	• Identify secondary transient • (Identify AOP-010 entry) • Identify feedline rupture
AOP-010	BOP	• AOP-010 Immediate actions when Feedwater Regulator valves are NOT operating properly • Place ‘B’ FW Reg valve in manual • Maintain SG level 52%-62% • ‘B’ SG level cannot be maintained in band

Op Test No.: NRC	Scenario # 2	Event # 7	Page 39 of 70
Event Description:	**‘B’ SG Feedline Break Inside Containment**		

Time	Position	Applicant’s Actions or Behavior
	SRO	- Direct tripping the Reactor - Direct Main Steam Line Isolation - Direct initiation of Safety Injection
	RO	Manually trips the Reactor
EOP-E-0	SRO	Makes PA announcement for Reactor Trip
Immediate Action	RO	VERIFY Reactor Trip: - AUTO or MANUAL Reactor Trip successful: - CHECK for any of the following: ○ Trip breakers RTA and BYA OPEN (YES) ○ Trip breakers RTB and BYB OPEN (YES) ○ ROD Bottom lights LIT (YES) ○ NEUTRON flux decreasing (YES)
Immediate Action	BOP	VERIFY Turbine Trip: - CHECK for any of the following: ○ ALL turbine throttle valves – SHUT (YES) ○ ALL turbine governor valves – SHUT (YES) - VERIFY power to AC Emergency Buses ○ 1A-SA AND 1B-SB Buses energized by off-site power or EDG’s. (YES)

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	40	of	70
Event Description:	**‘B’ SG Feedline Break Inside Containment**								

Time	Position	Applicant’s Actions or Behavior
Immediate Action	RO	CHECK SI Actuation: • CHECK for any of the following – LIT ○ SI Actuated bypass permissive light (NO) ○ ALB-11-2-2 (NO) ○ ALB-11-5-1 (NO) ○ ALB-11-5-3 (NO) ○ ALB-12-1-4 (NO) CHECK SI Actuation criteria: • CNMT pressure - GREATER THAN 3.0 PSIG (NO) • PRZ pressure – LESS THAN 1850 PSIG (NO) • Steam pressure – LESS THAN 601 PSIG (NO) • SI Actuation – REQUIRED (YES/NO – time dependent) Verifies SI actuation
	SRO	Perform the following: • Review Foldout page and assign foldout • Evaluate EAL Matrix
	RO	When conditions met, trip all RCP’s based on Foldout Page A. • Secures ALL RCP’s and reports to SRO when complete
	RO	Verify All CSIPs AND RHR pumps – RUNNING (YES) Check SI Flow: • SI flow - GREATER THAN 200 GPM (YES) • RCS pressure - LESS THAN 230 PSIG (NO)

Op Test No.: NRC	Scenario # 2	Event # 7	Page 41 of 70
Event Description:	**‘B’ SG Feedline Break Inside Containment**		

Time	Position	Applicant’s Actions or Behavior
	BOP	Check Main Steam Isolation: • Main Steam Isolation – ACTUATED (YES) • Verify all MSIVs and bypass valves – SHUT (YES)
Evaluator Note:		**The BOP or the Crew may identify that “B” SG AFW isolation should have occurred but did not and isolate AFW to the “B” SG at any time prior to guidance from the procedure.**
	BOP	Any SG pressure - 100 PSIG LOWER THAN PRESSURE IN TWO OTHER SGs (time dependent YES/NO) **If YES then next step applies** If NO then skips verification of AFW Isolation valves for now
	BOP	Verify MDAFW **AND** TDAFW Isolation Valves **AND** Flow Control Valves To Affected SG – SHUT (NO – “B” SG is NOT isolated)
	RO	Check CNMT Pressure – HAS REMAINED LESS THAN 10 PSIG (YES)
	BOP	Check AFW Status: AFW flow - AT LEAST 200 KPPH ESTABLISHED (YES)
Evaluator Note:		**Sequencer Load Block 9 will NOT be actuated for the “A” Train since the “A” Chiller did not start. The crew will have to perform a manual actuation of load block 9 after waiting for the 150 timer to expire.**

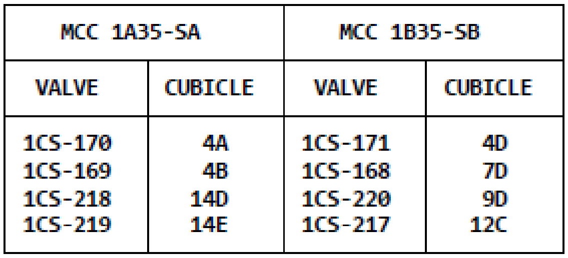
Op Test No.:	NRC	Scenario #	2	Event #	7	Page	42	of	70
Event Description:		**‘B’ SG Feedline Break Inside Containment**							

Time	Position	Applicant’s Actions or Behavior

RNO for next step - Procedure Note:		**Manual actuation of load block 9 cannot occur for 150 SECONDS after sequencer operation.**
	BOP	Sequencer Load Block 9 (Manual Loading Permissive) – ACTUATED (BOTH TRAINS) (NO only “B” Train) Performs RNO action: a. Place sequencer MANUAL LOADING switch to PERM. b. Check the MAN PERM light is LIT (**YES**)
	BOP	Energize AC buses 1A1 **AND** 1B1. Locates MCB switches for AC buses 1A1 and 1B1 and closes breakers to energize each bus
	BOP	Verify Alignment Of Components From Actuation Of ESFAS Signals Using Attachment 3, "Safeguards Actuation Verification", While Continuing With This Procedure.
Evaluator Note:		**EOP-E-0 Attachment 3 is included in the back of this scenario.** **The RO will perform all board actions until the BOP completes Attachment 3. The BOP is permitted to properly align plant equipment IAW 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.**

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	43	of	70
Event Description:	**‘B’ SG Feedline Break Inside Containment**								

Time	Position	Applicant’s Actions or Behavior

<table>
<tr><td></td><td>BOP</td><td>Directs AO to place 1A and 1B Air Compressor in the local control mode (Refer to Attachment 7)</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Simulator Communicator:</td><td>Acknowledge the request to place 1A and 1B Air Compressor in the local control mode</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Simulator Operator:</td><td>When directed to place the 1A and 1B Air Compressor in the local control mode:
Run APP\air\acs_to_local</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Simulator Communicator:</td><td>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.</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td>Direct RAB AO to locally unlock AND turn ON the breakers for the CSIP suction AND discharge cross-connect valves:
(Refer to Attachment 2)

<table>
<tr><th colspan="2">MCC 1A35-SA</th><th colspan="2">MCC 1B35-SB</th></tr>
<tr><th>VALVE</th><th>CUBICLE</th><th>VALVE</th><th>CUBICLE</th></tr>
<tr><td>1CS-170</td><td>4A</td><td>1CS-171</td><td>4D</td></tr>
<tr><td>1CS-169</td><td>4B</td><td>1CS-168</td><td>7D</td></tr>
<tr><td>1CS-218</td><td>14D</td><td>1CS-220</td><td>9D</td></tr>
<tr><td>1CS-219</td><td>14E</td><td>1CS-217</td><td>12C</td></tr>
</table></td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Simulator Communicator:</td><td>Acknowledge request to unlock and turn on the breakers for the CSIP suction and discharge cross-connect valves</td></tr>
</table>

Op Test No.:	NRC	Scenario #	2	Event #	7	Page	44 of 70
Event Description:		**‘B’ SG Feedline Break Inside Containment**					

Time	Position	Applicant’s Actions or Behavior
Simulator Operator:		**Run APP\cvc\E-0 Att 3 CSIP suct & disch valve power.txt – when the APP has completed running inform MCR that the valves have power.**
	RO/BOP	Control RCS Temperature: Stabilize AND maintain temperature between 555°F AND 559°F using Table 1.

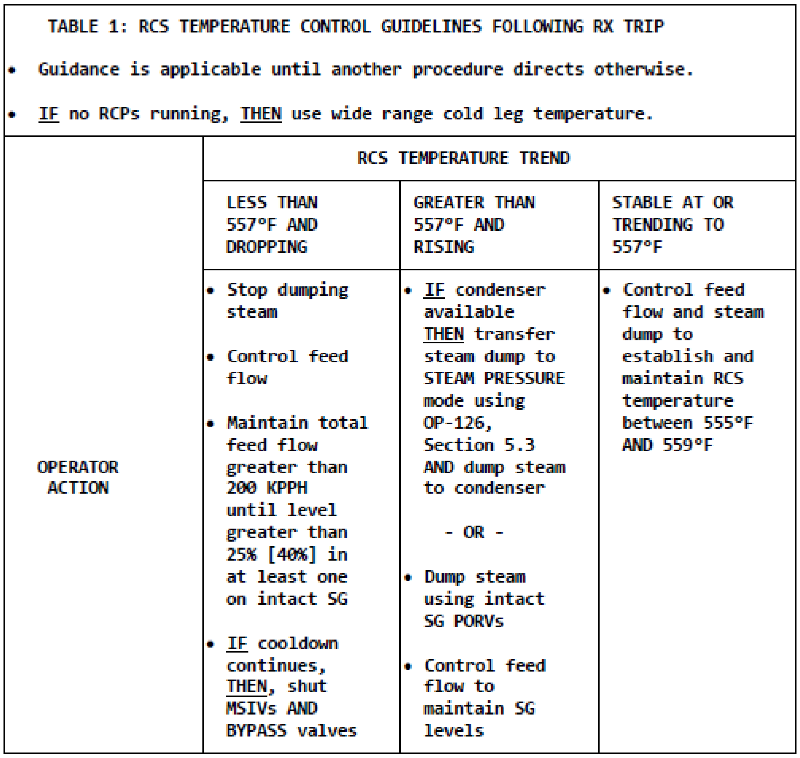


TABLE 1: RCS TEMPERATURE CONTROL GUIDELINES FOLLOWING RX TRIP

- 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 AND 559°F

Op Test No.:	NRC	Scenario #	2	Event #	8 & 9	Page	45 of	70
Event Description:		**Failure of Auto AFW Isolation and SR NI’s fail to Energize ‘B’ SG Feedline Break Inside Containment (continued)**						

Time	Position	Applicant’s Actions or Behavior
	RO/BOP	Check PRZ PORVs AND Spray Valves: • Check PRZ PORVs – SHUT (YES) • PRZ spray valves – SHUT (YES) • Check block valves – AT LEAST ONE OPEN (YES) Identify Any Faulted SG: Check for any of the following: • Any SG pressures - DROPPING IN AN UNCONTROLLED MANNER (YES ‘B’ SG) OR Any SG – COMPLETELY DEPRESSURIZED (NO)
Evaluator Note:		***EVENT 9 – Time dependent - The SR nuclear instrumentation will fail to energize due to under compensation on NI-35. When recognized, the crew should take action to manually energize the SR NIS.**
Event 9	CREW	SR failure: IR NI-35B MCB Amps 10^{-9} amps, IR NI-36B MCB Amps 10^{-11} amps When SR instrument failure to energize is recognized, take the following switches to RESET • SOURCE RANGE TRAIN A TRIP BLOCK • SOURCE RANGE TRAIN B TRIP BLOCK CHECK that Source Range detector high voltage is energized
	SRO	GO TO EOP-E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1

Op Test No.:	NRC	Scenario #	2	Event #	8 & 9	Page	46	of	70
Event Description:		**Failure of Auto AFW Isolation and SR NI’s fail to Energize** **‘B’ SG Feedline Break Inside Containment (continued)**							

Time	Position	Applicant’s Actions or Behavior

EOP-E-2	SRO	EOP-E-2, FAULTED STEAM GENERATOR ISOLATION
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/RO	• Check MSIVs AND Bypass Valves:
		  • Verify all MSIVs – SHUT (YES)
		  • Verify all MSIV bypass valves – SHUT (YES)
		• Check Any SG NOT Faulted:
		  • Any SG pressure - STABLE OR RISING (YES)
		• Identify Any Faulted SG:
		  • Check for any of the following:
		    o Any SG pressure - DROPPING IN AN UNCONTROLLED MANNER (YES)
		    OR
		    o Any SG - COMPLETELY DEPRESSURIZED (NO)

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.

Op Test No.:	NRC	Scenario #	2	Event #	8 & 9	Page	47	of	70

Event Description: **Failure of Auto AFW Isolation and SR NI’s fail to Energize ‘B’ SG Feedline Break Inside Containment (continued)**

Time	Position	Applicant’s Actions or Behavior
Event 8 **Critical Task #2**	BOP/RO	Isolate Faulted SG(s): • Verify faulted SG(s) PORV – SHUT (YES) • Verify main FW isolation valves – SHUT (YES) • Verify MDAFW AND TDAFW pump isolation valves to faulted SG(s) – SHUT **(NO – Event 8)** ○ IF NO, close isolation valves **(Shuts isolation valves)** *(Critical task - Isolate AFW flow to ‘B’ Steam Generator prior to any RCS Cold Leg Temperature lowering to less than 240°F after a RCS temperature drop in any cold leg of greater than 100° within the last hour)*
	BOP/RO	• Shut faulted SG(s) steam supply valve to TDAFW pump – SHUT • SG B: 1MS-70 • SG C: 1MS-72 ○ IF Open, close 1MS-70 (SHUTS) • Verify main steam drain isolation(s) before MSIVs - SHUT: • SG A: 1MS-231 (YES) • SG B: 1MS-266 (YES) • SG C: 1MS-301(YES) • Verify SG blowdown isolation valves – SHUT (YES) • Verify main steam analyzer isolation valves – SHUT (YES) Check CST Level - GREATER THAN 10% (YES)
Procedure Note:		**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.**

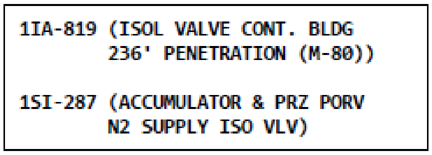
Op Test No.:	NRC	Scenario #	2	Event #	8 & 9	Page	48 of	70

Event Description: **Failure of Auto AFW Isolation and SR NI’s fail to Energize ‘B’ SG Feedline Break Inside Containment (continued)**

Time	Position	Applicant’s Actions or Behavior
	BOP/RO	Any SG – Abnormal Radiation or Uncontrolled Level Rise Check Secondary Radiation: • Check for all of the following (All NORMAL): Secondary Radiation Monitors And Indications RM-01MS-3591 SB, Main Steam Line A RM-01MS-3592 SB, Main Steam Line B RM-01MS-3593 SB, Main Steam Line C REM-01TV-3534, Condenser Vacuum Pump Effluent (RM-11: Grid 2 or Group 16) REM-1BD-3527, Steam Generator Blowdown (RM-11: Grid 2 or Group 16) RM-1TV-3536-1, Turbine Building Vent Stack Effluent (RM-11: Grid 2 or Group 16) SG Activity Sample • Check SG Levels: • Any level – RISING Uncontrolled (NO)
	RO/BOP	Check If SI Has Been Terminated: (NO) • Check for all of the following: ○ Check BIT outlet valves – SHUT OR ISOLATED 1SI-3 1SI-4 • Check cold leg AND hot leg injection valves - SHUT 1SI-52 1SI-86 1SI-107 (SI flow - GREATER THAN 200 GPM)

Op Test No.:	NRC	Scenario #	2	Event #	8 & 9	Page	49	of	70

Event Description: **Failure of Auto AFW Isolation and SR NI’s fail to Energize ‘B’ SG Feedline Break Inside Containment (continued)**

Time	Position	Applicant’s Actions or Behavior
	RO/BOP	• Check SI Termination Criteria: ○ Check Subcooling - GREATER THAN 10°F [40°F] - C 20°F [50°F] – M (YES) (Note the ‘C’ and ‘M’ above refers to how subcooling is calculated. ‘C’ is by the Computer, ‘M’ is Manual)
	BOP/RO	• Check secondary heat sink by observing any of the following: ○ Level in at least one intact SG – GREATER THAN 25% [40%] (YES) ○ Total feed flow to SGs - GREATER THAN 200 KPPH (YES or Available) ○ RCS pressure - STABLE OR RISING (YES) ○ PRZ level - GREATER THAN 10% [30%] (YES)
	RO/BOP	Reset SI
	SRO	(to crew) Manually Realign Safeguards Equipment Following A Loss Of Offsite Power. (Refer to E-0, Attachment 6.)
	RO/BOP	• Reset Phase A AND Phase B Isolation Signals. (Resets Phase A – Phase B • Open Instrument Air AND Nitrogen To CNMT: ○ Opens the following valves:  1IA-819 (ISOL VALVE CONT. BLDG 236' PENETRATION (M-80)) 1SI-287 (ACCUMULATOR & PRZ PORV N2 SUPPLY ISO VLV)

<table>
<tr><td>Op Test No.: NRC</td><td>Scenario # 2</td><td>Event # 8 & 9</td><td>Page 50 of 70</td></tr>
<tr><td>Event Description:</td><td colspan="3">Failure of Auto AFW Isolation and SR NI's fail to Energize
'B' SG Feedline Break Inside Containment (continued)</td></tr>
</table>

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>RO/BOP</td><td>
Stop All But One CSIP
RCS Pressure - STABLE OR RISING (YES)
Isolate High Head SI Flow:
Check CSIP suction - ALIGNED TO RWST (YES)
<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:
<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>Critical Task #3</td><td>RO</td><td>
Shut BIT outlet valves:
1SI-3
1SI-4
<p><i>(Critical task - Shut BIT Outlet valves 1SI-3 and 1SI-4 prior to PZR SRV's Discharge Line High Temperature occurring (250°F on any Safety valve discharge line temperature indicator either TI-465, or TI-467 or TI-469)</i></p>
Verify cold leg AND hot leg injection valves - SHUT
1SI-52
1SI-86
1SI-107
</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Procedure Caution:</td><td>High head SI flow should be isolated before continuing.</td></tr>
</table>

Op Test No.:	NRC	Scenario #	2	Event #	8 & 9	Page	51 of	70
Event Description:	**Failure of Auto AFW Isolation and SR NI’s fail to Energize ‘B’ SG Feedline Break Inside Containment (continued)**							

Time	Position	Applicant’s Actions or Behavior
	RO/BOP	• Establish Charging Lineup: o Shut charging flow control valve: FK-122.1 • Open charging line isolation valves: o 1CS-235 (MUST OPEN) o 1CS-238 (MUST OPEN)
Procedure Note:		**RCS temperature must be stabilized to allow evaluation of PRZ level trend.**
	RO/BOP	Monitor RCS Hot Leg Temperature: • Check RCS hot leg temperature – STABLE (YES) • 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.**
	RO/BOP	Control Charging Flow To Maintain PRZ Level: • Control charging using charging flow control valve: • FK-122.1 • Maintain charging flow less than 150 GPM. • PRZ Level - CAN BE MAINTAINED STABLE OR RISING (YES)
	SRO	GO TO EOP-ES-1.1, SI TERMINATION, Step 1.

Op Test No.:	NRC	Scenario #	2	Event #	8 & 9	Page	52	of	70
Event Description:		**Failure of Auto AFW Isolation and SR NI’s fail to Energize** **‘B’ SG Feedline Break Inside Containment (continued)**							

Time	Position	Applicant’s Actions or Behavior

Lead Evaluator:	**On Transition to EOP-ES-1.1 Ensure all Evaluators have collected the information needed to perform the evaluation then TERMINATE THE SCENARIO** **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**

5.0 STARTUP

5.1. Startup Train A-SA (B-SB) from Main Control Room or Local Panel

5.1.1. Initial Conditions

> **NOTE:** CM-I0014 Section 7.1 covers Control Module Replacement and Section 7.6 covers the current limiter circuit. If maintenance is performed on any portion of the current limiter circuit, additional Post Maintenance Testing per OP-148 section 8.16 is required. This includes calibration, adjustment, or other intrusive maintenance on any of the following components:
> - Temperature Current Module (current limiter portion only)
> - Current limiter resistor
> - B-phase current transformer
> - Current limiter circuit wiring/connections.

1. **IF** any maintenance activities were performed on any portion of the current limiter circuit of WC-2 Chiller 1A-SA (1B-SB), **THEN PERFORM** Post-maintenance Testing per OP-148 Section 8.16 for the applicable WC-2 Chiller Unit. ______

> **NOTE:** Section 5.2, Placing Standby Train in Operation, should be used when swapping Trains of ESCWS.

2. No Chiller Train is in service. ______
3. System filled and vented per Section 8.1. ______
4. System lineup Attachments 1 and 2 are complete. ______
5. For non-emergency starts the prestart checks of Attachment 5 have been performed and an operator should be present to observe start of chiller. ______
6. Section 8.12 Manual Chiller Reset has been performed, if necessary due to chiller trip. ______
7. The L.O. heaters have been in service for twelve hours. (See Precaution and Limitation 4.0.3 for applicability of this Initial Condition) ______

5.1.1 Initial Conditions (continued)

NOTE: The requirement to run the ESW Pump for 30 minutes does not apply if WC-2 Chiller start is due to AOP/EOP direction.

NOTE: If service water header temperature is greater than 92°F and the ESW pump is available startup of ESW is **required**. The pump should run for approximately 30 minutes before chiller start. ESW provides additional flow at typically lower temperatures when used for service water supply. Starting ESW prior to a chiller start minimizes condenser pressure. Historically, High Condenser Pressure alarms have been received during summer months due to high service water temperatures and high chilled water loads.

8. **IF** desired due to Service Water temperatures being high, **THEN VERIFY** a same train ESW Pump is running. Pump should run for approximately 30 minutes before chiller start. ______

5.1.2. Procedural Steps

NOTE: Whenever an "A" Train component is referred to in the body of this procedure it's "B" Train counterpart will immediately follow, enclosed by parentheses.

NOTE: ESR 99-00142 has evaluated and determined that long-term closure of the supply and return valves to the NNS AH units will not affect operability of the Essential Services Chiller system. The next two steps will align the NNS AH units however, if it is desired to maintain the NNS isolation valves shut, then steps 5.1.2.1 and 5.1.2.2 may be skipped.

1. **ISOLATE** the supply and return valves to the NNS AH units from the train that will not be placed in service by shutting the following valves:

1CH-125 SB (1CH-196 SB)	CHILLED WATER FROM NESSR FAN CLRS ISOL.	______
1CH-126 SA (1CH-197 SA)	CHILLED WATER FROM NESSR FAN CLRS ISOL.	______
1CH-115 SA (1CH-148 SB)	CHILLED WATER TO NESSR FANS CLR ISOL	______
1CH-116 SB (1CH-149 SA)	CHILLED WATER TO NESSR FAN CLRS ISOL	______

2. **ALIGN** the supply and return valves to the NNS AH units associated with the train that will be placed in service by opening the following valves:

1CH-125 SB (1CH-196 SB)	CHILLED WATER FROM NESSR FAN CLRS ISOL.	______
1CH-126 SA (1CH-197 SA)	CHILLED WATER FROM NESSR FAN CLRS ISOL.	______
1CH-115 SA (1CH-148 SB)	CHILLED WATER TO NESSR FANS CLR ISOL	______
1CH-116 SB (1CH-149 SA)	CHILLED WATER TO NESSR FAN CLRS ISOL	______

5.1.2 Procedural Steps (continued)

NOTE: The local alarm indication for low chilled water flow and low chilled water temperature will lock in until manually reset at the WC-2 control panel.

3. **START** WC-2 Chiller 1A-SA (1B-SB) Chilled water pump P-4 to establish chilled water flow. ______

4. At the Local Control panel, **RESET** the Low Chilled Water Flow alarm using the CHILLED WATER NO FLOW TRIP INDICATION RESET push-button. ______

5. **IF** starting the chiller for the first time following maintenance where the chiller lube oil heater circuit was under clearance.
   THEN PERFORM the following:
   a. Locally **START** the oil pump on the 1A-SA (1B-SB) compressor by taking the control switch on the local panel to the MAN position. ______
   b. **RUN** pump for 5 minutes. ______
   c. **STOP** the oil pump on the 1A-SA (1B-SB) chiller compressor by taking the control switch on the local panel to the AUTO position. ______

6. At the Local Control Panel, **CHECK** that all alarm lights are **NOT** lit. ______

7. **IF** any alarm light(s) is lit,
   THEN PERFORM the following:
   a. **IF** the Local Select switch is in the LOCAL position,
      THEN locally **DEPRESS** the STOP push-button. ______
   b. **IF** the Local Select switch is in the MCB HVAC position,
      THEN place the 1A-SA (1B-SB) compressor control switch on AEP-1 to STOP. ______
   c. **IF** any alarm light is still lit,
      THEN PERFORM the following:
      (1) **DECLARE** the chiller inoperable. ______
      (2) **INITIATE** corrective actions. ______

5.1.2 Procedural Steps (continued)

NOTE: If the unit cycles off due to low chilled water flow or low chilled water temperature, the unit will automatically restart if all start permissive conditions exist.

NOTE: An anti-recycle feature prevents more than one normal start within a 30 minute period. This anti-recycle feature is bypassed upon any automatic start signal from the ESF sequencer.

NOTE: After going to START on the Chiller Control Switch, the oil pump will start and bring oil pressure up to normal operating pressure prior to chiller start.

NOTE: OPT-1512 rotates the Temperature Control Point potentiometer to clean the surfaces. While OPT-1512 restores the potentiometer to its original position, it is possible that due to the surface cleaning the characteristics of the potentiometer have changed sufficiently to require a manual temperature adjustment per Section 8.14 of this procedure. This will be determined by monitoring temperature after chiller start in the following Step.

NOTE: ALB-023/1-14 (2-14), WC-2 CH 1A (1B) CNDSR REFRIG HI PRESS, may alarm during startup of the Chillers. High chiller condenser pressure is caused by inadequate cooling of the refrigerant. Causal factors for high condenser pressure include high chiller service water inlet temperature, condenser tube fouling, condenser shell air binding, or reduction of service water flow.

8. **START** the chiller by performing one of the following:

   a. At AEP-1, **PLACE** Water Chiller Compressor WC-2 A-SA (WC-2 B-SB) control switch to the START position and release. ______

   OR

   b. **DEPRESS** the START push-button at the local control panel with the Local Select switch in the LOCAL position. ______

NOTE: Engineering recommends running ESW for about 5-10 minutes after the chiller starts to ensure it reaches steady state operation. Operator judgment should be used to determine if continuing to run the ESW pump to prevent the High Condenser Pressure alarm is warranted. There is no operability impact, but a nuisance alarm can be prevented.

9. **IF** desired,
   THEN STOP the ESW Pump started in Step 5.1.1.8. ______

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5.2. Placing Standby Train In Operation

NOTE: It is necessary to shift associated trains of HVAC units when shifting trains of Essential Services Chilled Water.

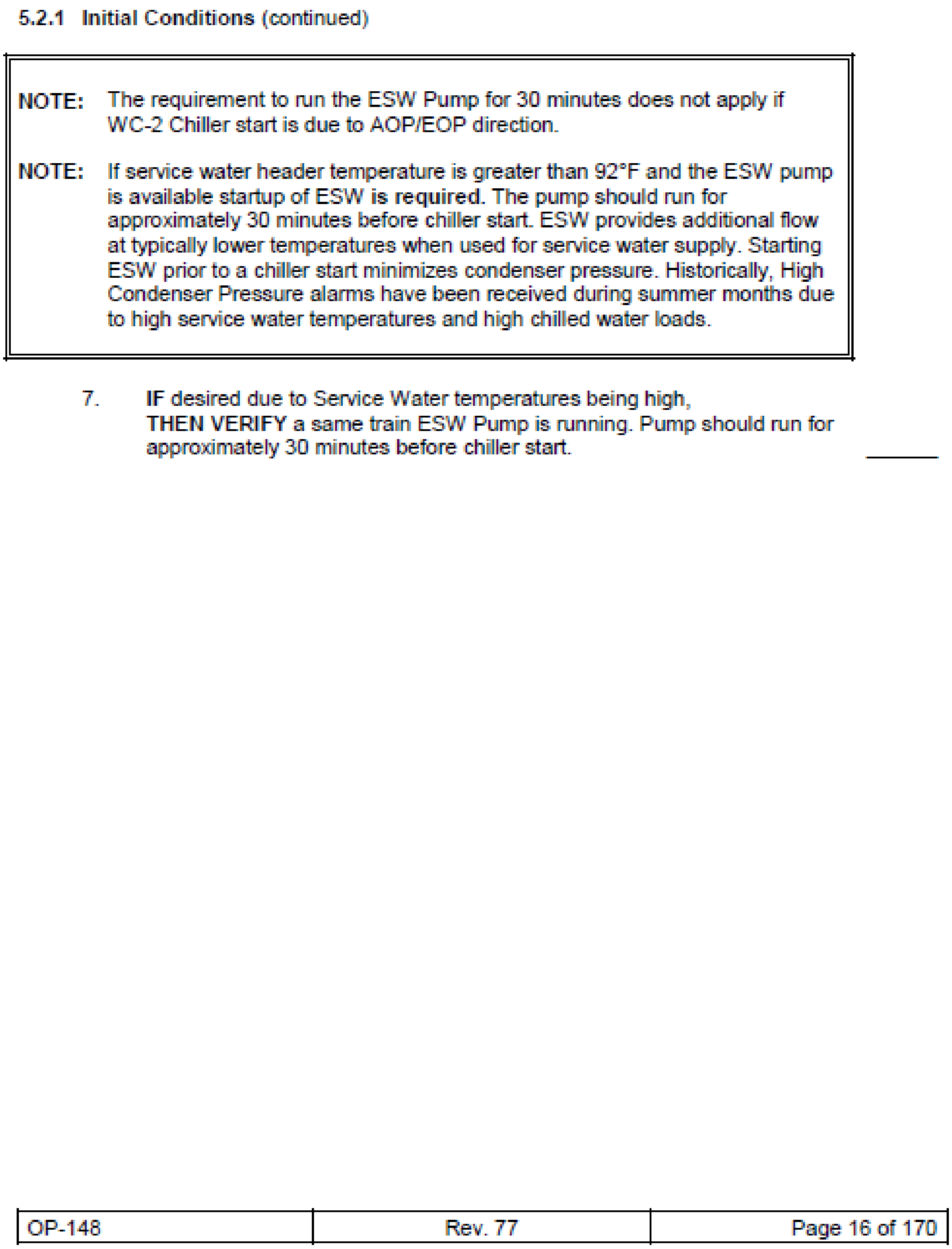
NOTE: This Section is written for swapping from Train B ESCW to Train A ESCW, with components for swapping from Train A ESCW to Train B ESCW in parentheses.

5.2.1. Initial Conditions

NOTE: CM-I0014 Section 7.1 covers Control Module Replacement and Section 7.6 covers the current limiter circuit. If maintenance is performed on any portion of the current limiter circuit, additional Post Maintenance Testing per OP-148 section 8.16 is required. This includes calibration, adjustment, or other intrusive maintenance on any of the following components:

- Temperature Current Module (current limiter portion only)
- Current limiter resistor
- B-phase current transformer
- Current limiter circuit wiring/connections.

1. **IF** any maintenance activities were performed on any portion of the current limiter circuit of WC-2 Chiller 1A-SA (1B-SB), **THEN PERFORM** Post-maintenance Testing per OP-148 Section 8.16 for the applicable WC-2 Chiller Unit. ______
2. Service water is being supplied to the non-operating chiller WC-2 1A-SA (WC-2 1B-SB). ______
3. One train of ESCW is already in operation. ______
4. For non-emergency starts the prestart checks of Attachment 5 have been performed and an operator should be present to observe start of chiller. ______
5. Section 8.12, Manual Chiller Reset performed if necessary for non-operating chiller. ______
6. The L.O. heaters have been in service for twelve hours. (See Precaution and Limitation 4.0.3 for applicability of this Initial Condition) ______



5.2.1 Initial Conditions (continued)

NOTE: The requirement to run the ESW Pump for 30 minutes does not apply if WC-2 Chiller start is due to AOP/EOP direction.

NOTE: If service water header temperature is greater than 92°F and the ESW pump is available startup of ESW **is required.** The pump should run for approximately 30 minutes before chiller start. ESW provides additional flow at typically lower temperatures when used for service water supply. Starting ESW prior to a chiller start minimizes condenser pressure. Historically, High Condenser Pressure alarms have been received during summer months due to high service water temperatures and high chilled water loads.

7. **IF** desired due to Service Water temperatures being high, **THEN VERIFY** a same train ESW Pump is running. Pump should run for approximately 30 minutes before chiller start. ______

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5.2.2. Procedural Steps

NOTE: The local alarm indication for low chilled water flow and low chilled water temperature will lock in until manually reset at the WC-2 control panel.

NOTE: If starting the chiller compressor is delayed following the start of the P-4 Pump in the next Step, the compressor oil could cool down to the point that the compressor will trip on low oil pressure.

NOTE: In Winter months (December - February) Step 5.2.2.7 can be performed anytime after Step 5.2.2.1. It is preferable to start the fans before the chiller. This allows the chill water to heat up and prevents the chiller cycling on and off on low temperature.

1. At AEP-1, **START** the non-operating Chiller WC-2 A-SA (B-SB) Chilled Water Pump P-4 A-SA (B-SB) to establish chilled water flow in the non-operating train. ______
2. At the Local Control panel, **RESET** the Low Chilled Water Flow alarm using the CHILLED WATER NO FLOW TRIP INDICATION RESET push-button. ______
3. **IF** starting the chiller for the first time following maintenance where the chiller lube oil heater circuit was under clearance, **THEN PERFORM** the following:
   a. Locally **START** the oil pump on the standby chiller compressor by taking the control switch on the local panel to the MAN position. ______
   b. **RUN** pump for 5 minutes. ______
   c. **STOP** the standby chiller compressor oil pump by taking the control switch on the local panel to the AUTO position. ______
4. At the Local Control Panel, **CHECK** that all alarm lights are **NOT** lit. ______
5. **IF** any alarm light(s) is lit, **THEN PERFORM** the following:
   a. **IF** the Local Select switch is in the LOCAL position, **THEN** locally **DEPRESS** the STOP push-button. ______
   b. **IF** the Local Select switch is in the MCB HVAC position, **THEN** place the standby chiller compressor control switch on AEP-1 to STOP. ______

5.2.2 Procedural Steps (continued)

c. **IF** any alarm light is still lit,
THEN PERFORM the following:

(1) **DECLARE** the chiller inoperable. ______

(2) **INITIATE** corrective actions. ______

NOTE: OPT-1512 rotates the Temperature Control Point potentiometer to clean the surfaces. While OPT-1512 restores the potentiometer to its original position, it is possible that due to the surface cleaning the characteristics of the potentiometer have changed sufficiently to require a manual temperature adjustment per Section 8.14 of this procedure. This will be determined by monitoring temperature after chiller start in the following Step.

NOTE: ALB-023/1-14 (2-14), WC-2 CH 1A (1B) CNDSR REFRIG HI PRESS, may alarm during startup of the Chillers. High chiller condenser pressure is caused by inadequate cooling of the refrigerant. Causal factors for high condenser pressure include high chiller service water inlet temperature, condenser tube fouling, condenser shell air binding, or reduction of service water flow.

6. **START** the chiller by performing **ONE** of the following:

a. At AEP-1, **PLACE** Water Chiller Compressor WC-2 A-SA (WC-2 B-SB) control switch to the START position **AND RELEASE**. ______

OR

b. **DEPRESS** the START push-button at the local control panel with the local select switch in the LOCAL position. ______

7. **START** Train A (B) ESF Equipment Cooling System per OP-172, Section 5.6. ______

5.2.2 Procedural Steps (continued)

NOTE: ESR 99-00142 has evaluated and determined that long-term closure of the supply and return valves to the NNS AH units will not affect operability of the Essential Services Chiller system. The next two Steps will align the NNS AH units however, if it is desired to maintain the NNS isolation valves shut, then steps 5.2.2.8 and 5.2.2.9 may be skipped.

8. **ISOLATE** the supply and return valves to the NNS AH units from the train that was already operating by shutting the following valves:

1CH-196 SB (1CH-125 SB)	CHILLED WATER FROM NESSR FAN CLRS ISOL	______
1CH-197 SA (1CH-126 SA)	CHILLED WATER FROM NESSR FAN CLRS ISOL	______
1CH-148 SB (1CH-115 SA)	CHILLED WATER TO NESSR FANS CLR ISOL	______
1CH-149 SA (1CH-116 SB)	CHILLED WATER TO NESSR FAN CLRS ISOL	______

9. **ALIGN** NNS AH units to the train that will remain operating by opening the following valves:

1CH-125 SB (1CH-196 SB)	CHILLED WATER FROM NESSR FAN CLRS ISOL.	______
1CH-126 SA (1CH-197 SA)	CHILLED WATER FROM NESSR FAN CLRS ISOL.	______
1CH-115 SA (1CH-148 SB)	CHILLED WATER TO NESSR FANS CLR ISOL	______
1CH-116 SB (1CH-149 SA)	CHILLED WATER TO NESSR FAN CLRS ISOL	______

10. **IF** shifting chillers to support placing the standby safety equipment train in service,
THEN PERFORM Attachment 8. ______

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 1 of 8
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. Verify Two CSIPs - RUNNING

☐ 2. Verify Two RHR Pumps - RUNNING

☐ 3. Verify Two CCW Pumps - RUNNING

☐ 4. Verify All ESW **AND** ESW Booster Pumps - RUNNING

☐ 5. Verify SI Valves - PROPERLY ALIGNED

(Refer to Attachment 1.)

☐ 6. Verify 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 8
SAFEGUARDS ACTUATION VERIFICATION

☐ 7. Verify 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** Verify 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** Verify 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 8
SAFEGUARDS ACTUATION VERIFICATION

☐ 10. Verify Both Main FW Pumps - TRIPPED

☐ 11. Verify FW Isolation Valves - SHUT

(Refer to OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 6.)

☐ 12. Verify both MDAFW pumps - RUNNING

13. **IF** any of the following conditions exist, **THEN** verify 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. Verify AFW Valves - PROPERLY ALIGNED

- ☐ **IF** no AFW Isolation Signal, **THEN** verify 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** verify MDAFW and TDAFW isolation and flow control valves to affected SG - SHUT

☐ 15. Verify Both EDGs - RUNNING

☐ 16. Verify CNMT Fan Coolers - ONE FAN PER UNIT RUNNING IN SLOW SPEED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 4 of 8
SAFEGUARDS ACTUATION VERIFICATION

☐ 17. Verify CNMT Ventilation Isolation Valves - SHUT

(Refer to OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 7.)

☐ 18. Verify 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. Verify Essential Service Chilled Water System Operation:

☐ • Verify both WC-2 chillers - RUNNING

☐ • Verify both P-4 pumps - RUNNING

☐ (Refer to AOP-026, "LOSS OF ESSENTIAL SERVICE CHILLED WATER SYSTEM" for loss of any WC-2 chiller.)

20. Verify 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. Verify 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 8
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** Turn ON 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:

☐ • <u>IF</u> two charging pumps can <u>NOT</u> be verified to be running, **AND** C CSIP is available, <u>THEN</u> 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 8
SAFEGUARDS ACTUATION VERIFICATION

25. Start The Spent Fuel Pump Room Ventilation System:

    a. At AEP-1, verify 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 8
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 and 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)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 8 of 8
SAFEGUARDS ACTUATION VERIFICATION

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 Co-ordinator - Control Room

☐ • Site Emergency Co-ordinator - Technical Support Center

(Refer to PEP-230, "CONTROL ROOM OPERATIONS".)

- END -

HARRIS 2018 NRC SCENARIO 3

Facility: Harris Nuclear Plant Scenario No.: 3 Op Test No.: 05000400/2018301

Examiners: ______ Operators: SRO: ______

______ RO: ______

______ BOP: ______

Initial Conditions: IC-26, MOL, 86% power

- 'B' RHR Pump is under clearance for high motor vibrations
- 'B' DEH Oil Pump is under clearance for motor repairs
- 1CS-9 is under clearance for solenoid replacement

Turnover:	The plant is at 86% power, middle of core life. Due to the 'B' RHR 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.
Critical Task:	• Maintain control of SG 'C' level above 25% to prevent an automatic Reactor trip after the controlling level transmitter LT-496 fails high. • Manually trip all RCPs within 10 minutes of reaching RCP Trip Criteria of > 200 gpm SI Flow with < 1400 psig RCS Pressure

Event No.	Malf. No.	Event Type*	Event Description
1	pt:308c	R – RO/SRO N – BOP/SRO	Reduce power (GP-006)
2	crf08	I – RO/SRO TS – SRO	T_{ref} Failure (high) (AOP-001) (RO – Rod Control shifted from Auto to Manual - Competency - Operate plant controls in Manual)
3 #	lt:496	I – BOP/SRO TS – SRO	SG 'C' Controlling Level Transmitter, LT-496 Channel III (selected for 1C SG) fails high – (AOP-010) (BOP - Main Feedwater Controls - Competency - Operate plant controls in Manual)
4 #	rcs14b	TS – SRO	RCP 'B' #1 Seal degrades (AOP-018)
5	pt:308b	C – BOP/SRO TS – SRO	SG 'B' PORV Pressure Instrument fails high - PORV Opens. (AD-OP-ALL-1000)
6 #	rcs14b	C – RO/SRO	RCP 'B' #1 Seal fails (AOP-018)
7 #	prs04b	M – ALL	Steam Space LOCA inside containment (EOP-E-0)
8 #	rhr01a	C – RO/SRO	'A' RHR Pump trips on overcurrent on start
9 #	zrpk502b zrpk622a	C – BOP/SRO	Phase 'A' fails on the 'B' train and partially isolates on the 'A' train

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

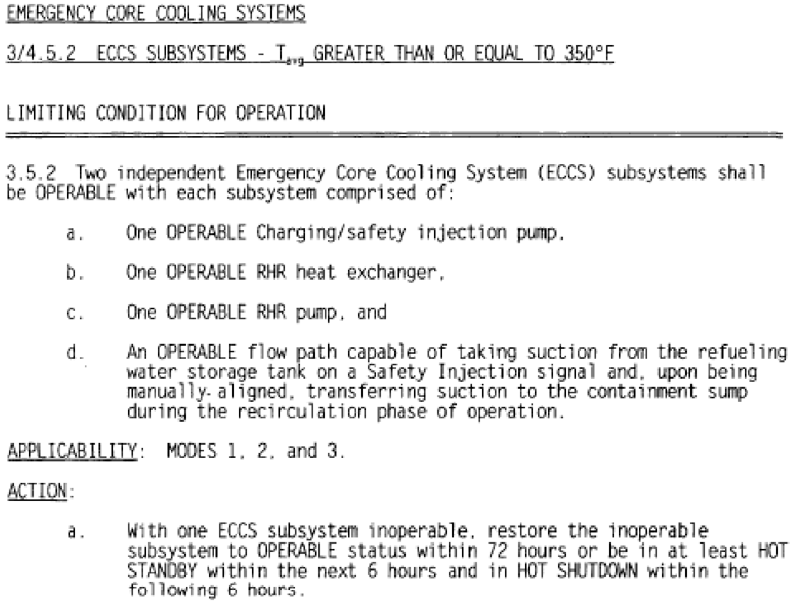
\# Event or Major Transient was not used on the previous 2 NRC initial licensing operating tests

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 3

The plant is at 86% power, middle of core life. Due to the ‘B’ RHR 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:

- RHR Pump B-SB is under clearance for high motor vibrations. The pump has been inoperable for 66 hours and cannot be restored to operable status. Tech Spec 3.5.2 LCO Action **a** and Tech Spec 3.3.3.5.b Action **c** applies. OWP-RH-02 has been completed.



EMERGENCY CORE COOLING SYSTEMS

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

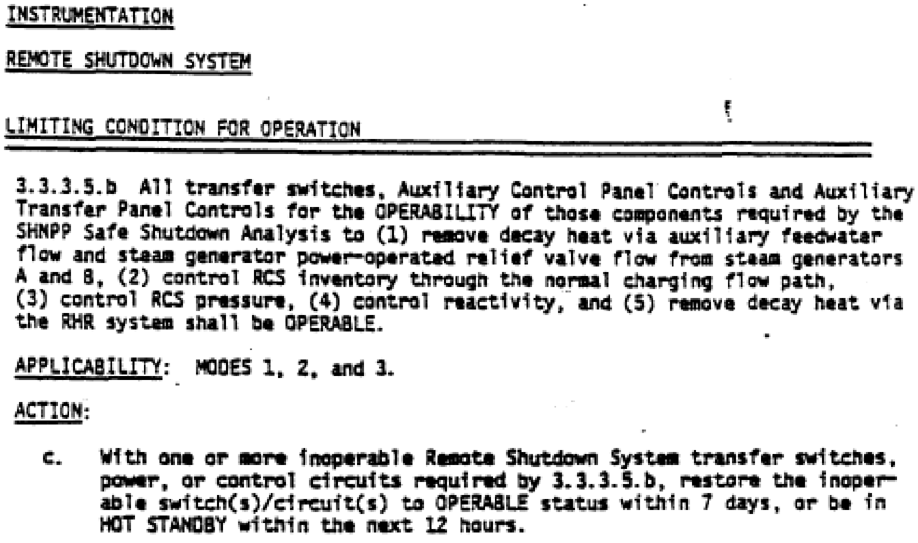
c. One OPERABLE RHR pump, and

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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



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:

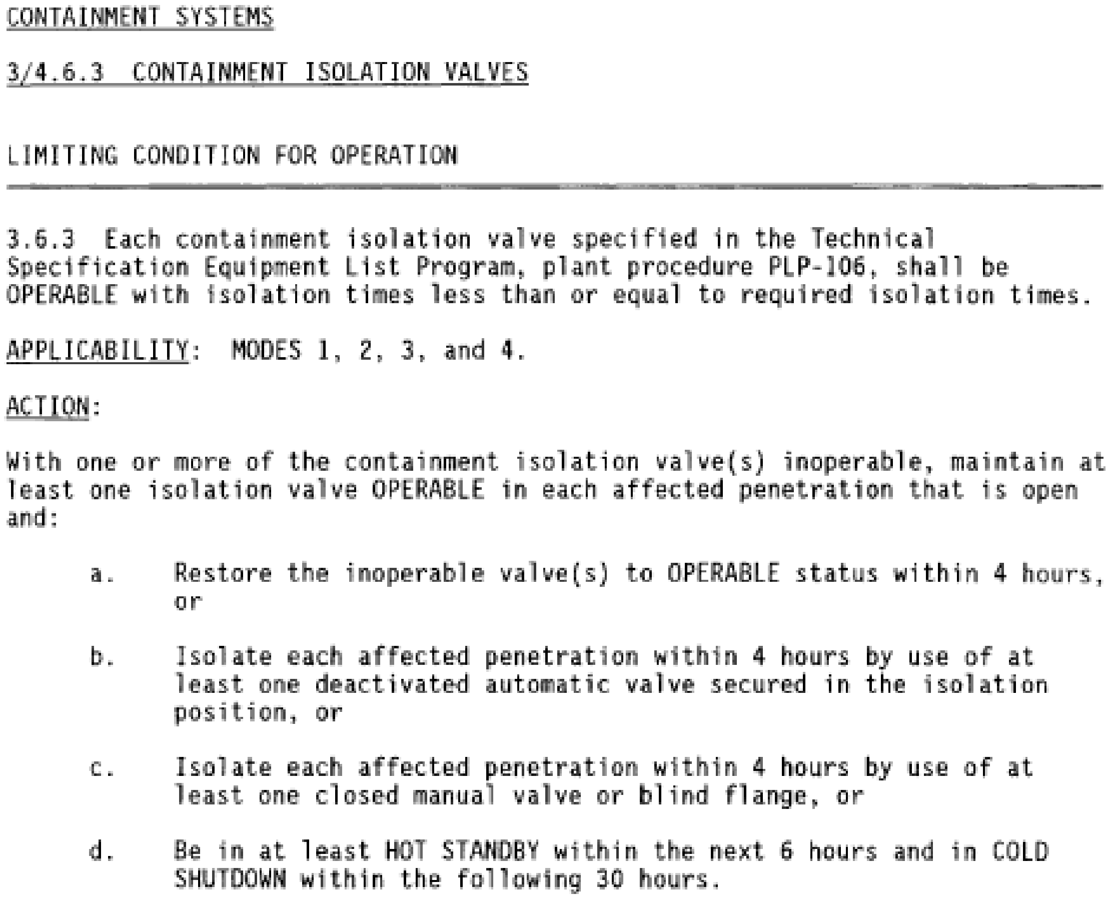
c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 3 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.

- Letdown Orifice Isolation Valve 1CS-9 is under clearance for solenoid replacement inspection. Tech Spec 3.6.3 LCO Action **b** applies. OWP-CS-09 has been completed.



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: 2018 NRC EXAM SCENARIO 3 (Continued)

Event 1: Plant Shutdown (GP-006). Turnover takes place with the unit at 86% Reactor power. The crew will be given credit for a reactivity manipulation during the down power. 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 (at Evaluator discretion) event 2 can be inserted.

Event 2: Failure of the T_{ref} Processor (fails high). The crew should 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. (RO – Rod Control shifted from Auto to Manual - Competency - Operate plant controls in Manual). With concurrence from the SRO the OATC will restore T_{ave} to pre-failure conditions by inserting the rods in manual.
The SRO should set control and trip limits in accordance with OMM-001 for rod control in manual.

Event 3: SG 'C' Controlling Level Transmitter LT-496 fails. The crew will respond by entering AOP-010, Feedwater Malfunction and taking manual control of 'C' Main Feedwater Regulating Valve to increase Feedwater flow and stabilize level. With the controller in manual and the plant stabilized the crew will implement OWP-RP-07 to remove the failed channel from service. (BOP - Main Feedwater Controls - Competency - Operate plant controls in Manual).
The SRO should set control and trip limits and evaluate the following Tech Specs for failure of LT-496:

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

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
13. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. each operating stm. gen.	1, 2	6(1)

(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: 2018 NRC EXAM SCENARIO 3 (Continued)

Event 3: Tech Spec evaluation continued

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

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

ACTION STATEMENTS (Continued)

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

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

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

T.S. 3.3.3.6: The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 3 (Continued)

Event 3: Tech Spec evaluation continued

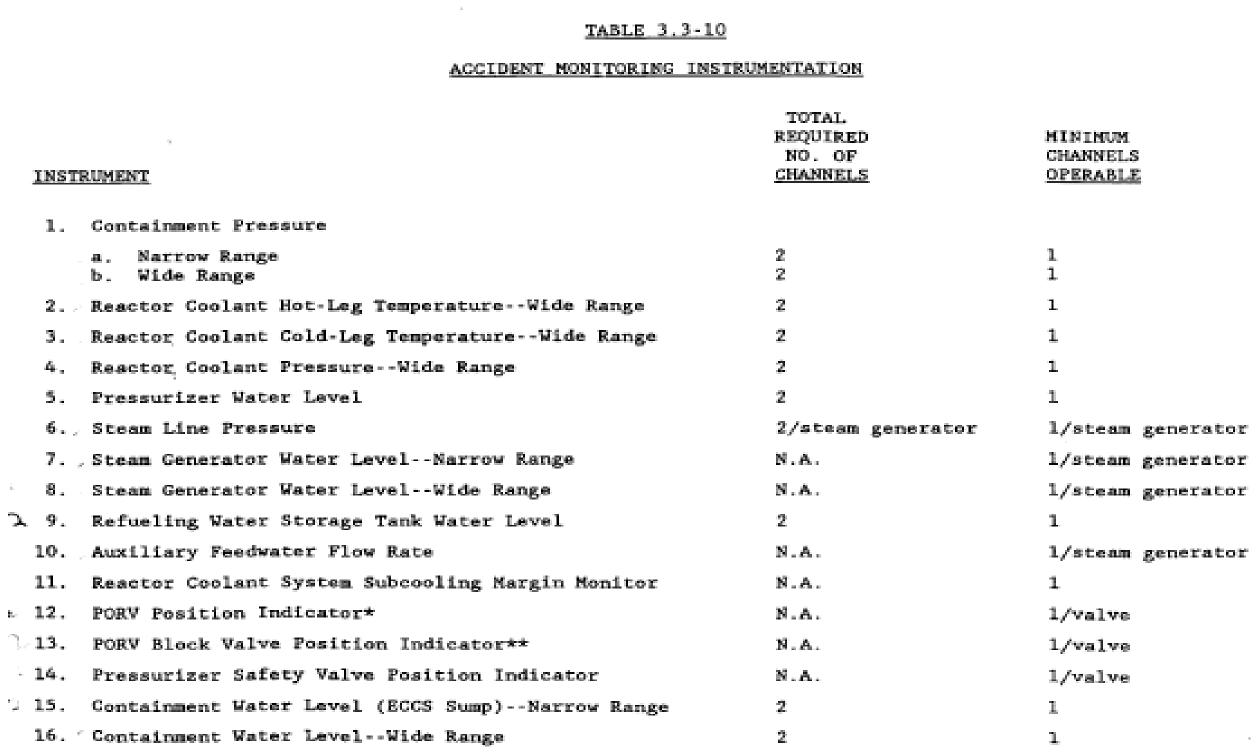


TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure		
a. Narrow Range	2	1
b. Wide Range	2	1
2. Reactor Coolant Hot-Leg Temperature--Wide Range	2	1
3. Reactor Coolant Cold-Leg Temperature--Wide Range	2	1
4. Reactor Coolant Pressure--Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level--Narrow Range	N.A.	1/steam generator
8. Steam Generator Water Level--Wide Range	N.A.	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Auxiliary Feedwater Flow Rate	N.A.	1/steam generator
11. Reactor Coolant System Subcooling Margin Monitor	N.A.	1
12. PORV Position Indicator*	N.A.	1/valve
13. PORV Block Valve Position Indicator**	N.A.	1/valve
14. Pressurizer Safety Valve Position Indicator	N.A.	1/valve
15. Containment Water Level (ECCS Sump)--Narrow Range	2	1
16. Containment Water Level--Wide Range	2	1

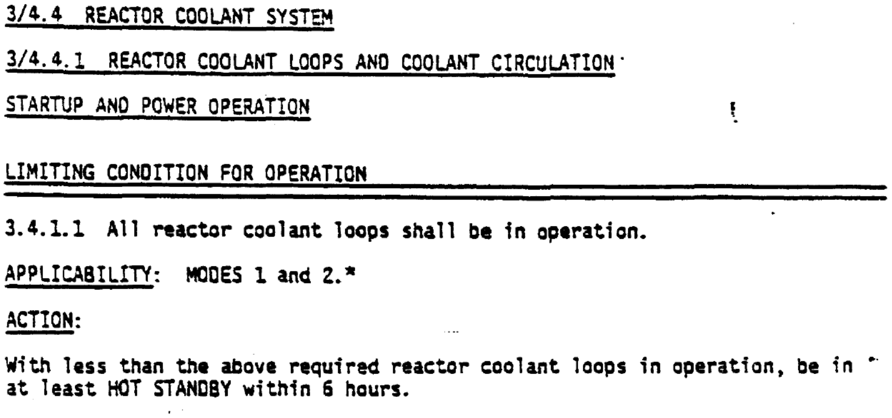
***NOTE:** The OWP is not required to be implemented in order to continue with the scenario. If the crew allows SG levels to lower to < 30% they will be required to perform a manual Reactor Trip. IF the crew does not respond to the low water level in the SG a Low level (< 25%) Automatic Reactor trip will occur.

An automatic Reactor Trip for this event will create critical task. (See **Note** after critical task justification statements for details on unanticipated critical tasks.)

Event 4: RCP ‘B’ #1 Seal degrades – The crew should identify an RCP seal malfunction then enter AOP-018, Reactor Coolant Pump Abnormal Conditions to evaluate the seal malfunction. The crew should identify the ‘B’ RCP #1 seal as “degraded”. The crew should continue with the plant shutdown using GP-006. They should determine that they are required to stop the ‘B’ RCP within eight hours of the seal leakoff flow exceeding 6 gpm and additionally must shut 1CS-396, ‘B’ RCP #1 Seal Water Return valve, between three and five minutes after securing the RCP. Tthe SRO should evaluate Tech Spec 3.4.1.1, Reactor Coolant Loops and Coolant Circulation.

SCENARIO SUMMARY: 2018 NRC EXAM SCENARIO 3 (Continued)

Event 4: Tech Spec evaluation continued



3/4.4 REACTOR COOLANT SYSTEM

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.

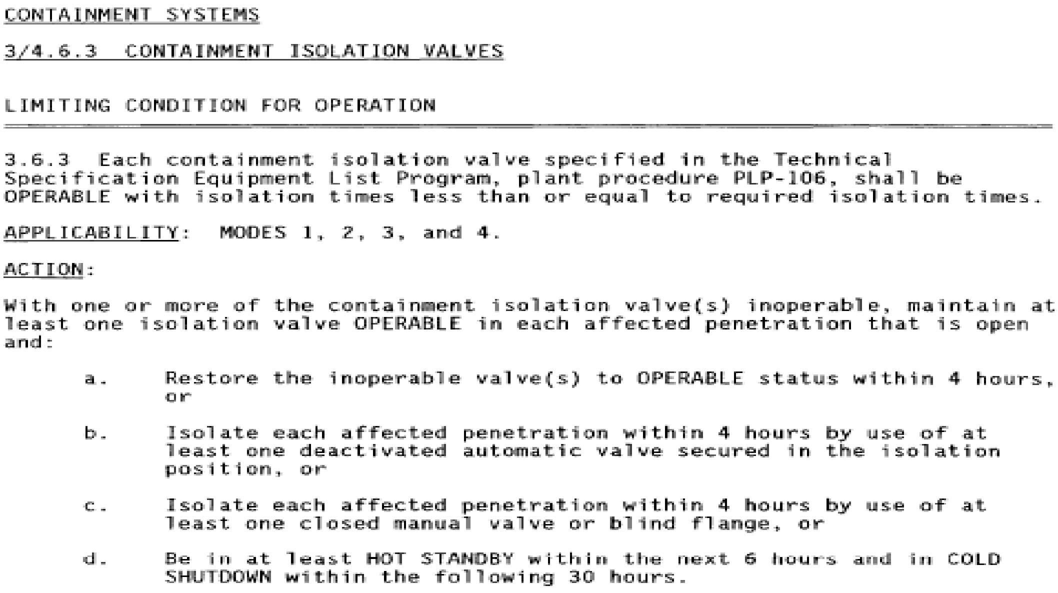
APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

Event 5: SG ‘B’ PORV Pressure Instrument fails high. A transmitter failure will cause the ‘B’ SG PORV to fail 100% open. The crew should identify this failure by annunciator ALB-014-8-5, Computer Alarm Steam Generators alarming and status light indications for the ‘B’ SG PORV. Note: The PT-308b does not have MCB indications. In accordance with AD-OP-ALL-1000, the BOP should identify a system malfunction and notify the CRS prior to taking manual control of the PORV. He/she will then place the controller in manual and shut the PORV. The SRO should evaluate Tech Specs 3.6.3, Containment Isolation Valves and PLP-106 Technical Specification Equipment List Program and Core Operating Limits Report.

TS 3.6.3 – Action **c**, isolate the affected penetration within 4 hours. The redundant manual isolation valve per PLP-106 is Cont Isolation valve 1MS-63. (The 4 hour action is met by shutting the PORV Isolation valve.)



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: 2018 NRC EXAM SCENARIO 3 (Continued)

Event 6: RCP ‘B’ #1 Seal fails – The RCP ‘B’ #1 seal will fail requiring the crew to implement the continuous action for AOP-018 and trip the Reactor, perform the immediate actions of EOP-E-0 then complete the remaining required actions of AOP-018 which is securing the RCP ‘B’ and shutting the associated spray valve. In accordance with AOP-018 the actions with the RCP will be performed as time permits.

Event 7: Steam Space LOCA inside Containment – The major event is a Steam Space LOCA from the PZR. During the Reactor trip actions called for in AOP-018, the crew will determine a Safety Injection is currently NOT required and transition to EOP-ES-0.1, Reactor Trip Response. Shortly after entering EOP-ES-0.1, the crew will identify unexpected PRT and PRZ parameter changes (pressure, level, temperature). Based on the foldout criteria in EOP-ES-0.1, when PRZ level cannot be maintained > 5% the crew should actuate Safety Injection. After returning to EOP-E-0 and with SI actuated the crew will identify the Foldout Criteria for securing RCPs has been met and secure the RCPs. 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 and go to EOP-E-1, Loss of Reactor or Secondary Coolant.

Event 8: ‘A’ RHR Pump trips on overcurrent on start – The ‘A’ RHR Pump will start automatically by the sequencer with the actuation of the SI. When the pump is started it will immediately trip on overcurrent. The RO should identify this failure and due to the annunciator indicating a trip/overcurrent condition occurred the operator should not make an attempt to manually restart the pump. The loss of RHR will complicate recovery actions and result in the crew transitioning from EOP-E-1 to EOP-ECA-1.1, Loss of Emergency Coolant Recirculation which will address the loss of RHR capability. Since the ‘B’ RHR Pump is under clearance for just routine maintenance the crew should be making efforts to restore the ‘B’ RHR pump. After the crew has transitioned into EOP-ECA-1.1 the Work Control Center will inform the crew that the ‘B’ RHR Pump is ready to be returned to service. The crew should then ensure proper pump alignment and start the ‘B’ RHR pump.

Event 9: The ‘B’ train of CNMT Phase A isolation valve will fail to automatically realign and the ‘A’ train CNMT Phase A isolation valve for the CNMT Fan Coiling Units 1SW-240 will fail to isolate. The crew should identify this failure and manually shut both Service Water return line Containment Isolation valves, 1SW-240 and 1SW-242.

The scenario termination is met in EOP-ECA-1.1 after the crew determines a RCS cooldown of 100°F has occurred within the last hour and that an additional RCS cooldown must wait until at least one hour has past. They can carry out actions of other procedures that do NOT cause an RCS cooldown OR raise RCS pressure.

CRITICAL TASK JUSTIFICATION:

1. Maintain control of SG ‘C’ level above 25% to prevent an automatic Reactor trip after the controlling level transmitter LT-496 fails high.

An unnecessary automatic Reactor Trip for this event will create critical task. See note below.

2. Manually trip all RCPs within 10 minutes of reaching RCP Trip Criteria of > 200 gpm SI Flow with < 1400 psig RCS Pressure

Securing RCPs during a SB LOCA event will prevent depleting the RCS to a critical inventory by pumping more mass through the break than would occur if the RCP operation were ceased. (Critical inventory is defined as the amount of inventory remaining in the RCS when the break completely uncovers and the break flow changes from a mixture of liquid and steam to all steam.) The LOCA event in this scenario is a SB LOCA that requires the RCPs to be secured when E-0 foldout conditions are met. IF the crew continues to allow the RCPs to operate due to lack of establishment of SI flow of > 200 gpm then RCS inventory will continue to deplete. Manually tripping the RCPs before depletion below the critical inventory conservatively ensures that Peak Clad Temperature remains below 2200°F. This action should be accomplished within 10 minutes of RCP Trip Criteria of > 200 gpm SI Flow with < 1400 psig RCS 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.

For the 2018 NRC Exam Simulator Scenario #3

Reset to IC-163 password “NRC2018”

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 3

Press START on Counter Scaler
Post conditions for status board from IC-26
Reactor Power 86%
Control Bank D at 200 steps
RCS boron 1094 ppm

Turnover: The plant is at 86% power, middle of core life. Due to the ‘B’ RHR 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:

- RHR 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 3.5.2 LCO Action **a** and Tech Spec 3.3.3.5.b Action **c** applies. OWP-RH-02 has been completed.

- ‘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.

- Letdown Orifice Isolation Valve 1CS-9 is under clearance for solenoid replacement. Tech Spec 3.6.3 LCO Action **b** applies. OWP-CS-09 has been completed.

Align equipment for repairs:

Place CIT on 'B-SB' RHR pump MCB Switch
Place protected train placards IAW OMM-001 Attachment 7
Protected Train placards on 'A-SA' RHR pump,'B-SB' MDAFW Pump

Place the “B” DEH Pump in PTL and then hang a CIT on MCB switch

Place a CIT on the switch for 1CS-9.

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-09 clearance

Hang restricted access signs on MCR entry swing gates

Op Test No.:	NRC	Scenario #	3	Event #	1	Page	12	of 69
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 86% 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.**
	OATC	OP-107.01, Section 5.2
	OATC	• DETERMINE the reactor coolant boron concentration from chemistry OR the Main Control Room status board. • DETERMINE the magnitude of boron concentration increase required. • DETERMINE the volume of boric acid to be added using the reactivity plan associated with the IC.
Procedure Note:		**FIS-113, BORIC ACID BATCH COUNTER, has a tenths position.**

Op Test No.:	NRC	Scenario #	3	Event #	1	Page	13	of	69	
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.**
	OATC	SET FIS-113, BORIC ACID BATCH COUNTER, to obtain the desired quantity.
	SRO	Directs boration
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.**
	OATC	• VERIFY the RMW CONTROL switch has been placed in the STOP position. • VERIFY the RMW CONTROL switch green light is lit. • SET controller 1CS-283, FK-113 BORIC ACID FLOW, for the desired flow rate. • 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.**

Op Test No.:	NRC	Scenario #	3	Event #	1	Page	14	of 69
Event Description:				Power Reduction				

Time	Position	Applicant's Actions or Behavior
	OATC	- START the makeup system as follows: o TURN control switch RMW CONTROL to START momentarily. o VERIFY the RED indicator light is LIT. o IF expected system response is not obtained, THEN TURN control switch RMW CONTROL to STOP. - VERIFY boration automatically terminates when the desired quantity of boron has been added. - 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 VERIFY the RED indicator light is LIT.
Evaluator Note:		**The following steps have been completed to achieve the current power level. The crew should validate status of the turbine load reduction IAW GP-006 section 6.2 step 5 before re-initiating the turbine load reduction.**
Procedure Note:		**Routine load changes must be coordinated with the Load Dispatcher to meet system load demands.**

Op Test No.: NRC Scenario # 3 Event # 1		Page 15 of 69
Event Description:		**Power Reduction**
Time	Position	Applicant's Actions or Behavior

Procedure Caution:		A failure of the VIDAR in the DEH computer has resulted in a plant trip in the past. This failure would affect operation in Operator Auto, and can be detected as follows: • If OSI-PI is available, then VIDAR is functioning properly if the 'DEH_MEGAWATTS' point is updating. • 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. • If the DEH graphics computer is out of service, then VIDAR can be checked as updating on the operator panel as follows: o Depress 'Turbine Program' display button. o Check 'Turbine Program' display button is illuminated. o Check 'Reference' and 'Demand' displays indicate '0000'. o Depress '1577'. o Depress 'Enter'. o If the 'Demand' display indicates '0000', then VIDAR is updating. o If the 'Demand' display indicates '0001', then VIDAR is not updating
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 DEH Control

Op Test No.: NRC	Scenario # 3	Event # 1	Page 16 of 69
Event Description:	**Power Reduction**		

Time	Position	Applicant's Actions or Behavior
	BOP	• DEPRESS the LOAD RATE MW/MIN push-button. • ENTER the desired rate, **NOT** to exceed 5 MW/MIN, in the DEMAND display. (4 DEH Units/minute) • DEPRESS the ENTER push-button. • DEPRESS the REF push-button. • ENTER the desired load (per CRS) in the DEMAND display. • DEPRESS the ENTER push-button. The HOLD push-button should illuminate.
Procedure Note:		**The unloading of the unit can be stopped at any time by depressing the HOLD push-button. The HOLD lamp will illuminate and the GO lamp will extinguish. The load reduction can be resumed by depressing the GO push-button. The HOLD lamp will extinguish and the GO lamp will illuminate.**
	BOP	• DEPRESS the GO push-button to start the load reduction. • VERIFY the number in the REFERENCE display decreases. • VERIFY Generator load is decreasing. • Communicate to the SRO that the Turbine is in GO.
Evaluator Note:		**The crew has demonstrated a load reduction (≥5%) at the satisfaction of the evaluators. Inform Simulator Operator to insert Trigger 2, Tref Failure (fails to 589°F)**

Op Test No.:	NRC	Scenario #	3	Event #	2	Page	17	of	69
Event Description:				**Tref Fails to 589°F**					

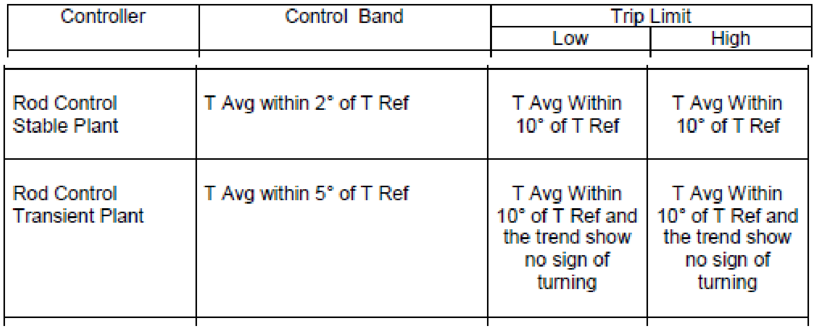
Time	Position	Applicant's Actions or Behavior

Simulator Operator:		On cue from Lead Evaluator insert Trigger 2 T-ref processor failure to 589°F
Indications Available		• **Uncontrolled rod motion** • **Tave – Tref MCB digital indication reads Tref at 589°F**
	OATC	RESPONDS to uncontrolled rod motion.
AOP-001		Malfunction of Rod Control and Indication System
	SRO	ENTERS and directs actions of AOP-001. 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	Conduct a Crew Update on entry into AOP-001.
	SRO	READS immediate actions and proceeds to Section 3.2. Directs BOP to place Turbine to HOLD if in GO.
	BOP	Places Turbine to HOLD if in GO.

Op Test No.:	NRC	Scenario #	3	Event #	2	Page	18	of	69

Event Description: **Tref Fails to 589°F**

Time	Position	Applicant's Actions or Behavior
	OATC	CHECK that instrument channel failure has NOT OCCURRED by observing the following: • RCS Tavg (NO) • RCS Tref (YES)
	OATC	PERFORM the following: • IF a power supply is lost, THEN GO TO AOP-024, Loss of Uninterruptible Power Supply. (NO) • 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. (N/A)
	SRO	Provides trip limits and operating bands IAW OMM-001 Attachment 13 for Rod Control in Manual



Controller	Control Band	Trip Limit	
		Low	High
Rod Control Stable Plant	T Avg within 2° of T Ref	T Avg Within 10° of T Ref	T Avg Within 10° of T Ref
Rod Control Transient Plant	T Avg within 5° of T Ref	T Avg Within 10° of T Ref and the trend show no sign of turning	T Avg Within 10° of T Ref and the trend show no sign of turning

Time	Position	Applicant's Actions or Behavior
	OATC	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. *(RO – Rod Control shifted from Auto to Manual - Competency - Operate plant controls in Manual)*
	OATC	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.

Op Test No.: NRC	Scenario # 3	Event # 2	Page 19 of 69
Event Description:		Tref Fails to 589°F	

Time	Position	Applicant's Actions or Behavior
Evaluator Note:		The following will be done when Tave is restored.
	OATC	VERIFY proper operation of the following: • CVCS demineralizers (YES) • BTRS (N/A) • REACTOR Makeup Control System (YES)
	SRO	CHECK that this section was entered due to control banks MOVING OUT. (YES)
	OATC	ENSURE CSIP suction aligned to the VCT: CHECK VCT level greater than 5%. (YES) ENSURE that the following valves are OPEN: • LCV-115C, VCT OUTLET (YES) • LCV-115E, VCT OUTLET (YES) ENSURE that the following valves are SHUT: • LCV-115B, SUCTION FROM RWST (YES) • LCV-115D, SUCTION FROM RWST (YES)
	OATC	CHECK that NEITHER of the following OCCURRED: • Unexplained RCS boration (YES) • Unplanned RCS dilution (YES)
	SRO	CHECK that spurious rod motion is due to malfunction of the Automatic Rod Control System (NO)
	SRO/ OATC	Reviews/prepares OMM-001, Attachment 5 Equipment Problem Checklist. Contacts support personnel for repairs.

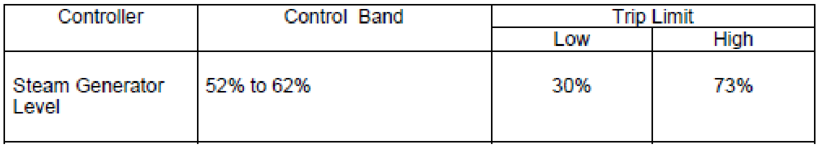
Op Test No.: <u>NRC</u>	Scenario # 3	Event # 2	Page <u>20</u> of <u>69</u>
Event Description:	**Tref Fails to 589°F**		

Time	Position	Applicant's Actions or Behavior
	SRO	Exits AOP-001
	SRO/ OATC	May discuss Equipment Problem Checklist with WCC and ask for support for the failure.
Evaluator's Note:		**When Tavg is restored and AOP-001 exited, cue initiation of Event 3 SG "C" LT-496 fails high**

Op Test No.:	NRC	Scenario #	3	Event #	3	Page	21	of	69
Event Description:		SG 'C' Controlling Level Transmitter (LT-496) fails high							

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 3 "SG 'C' Controlling Level Transmitter (LT-496) fails high"**
Indications Available:		• ALB-014-3-1B SG C NR LVL/SP HI/LO DEV • ALB-014-6-3B STEAM GEN C HIGH-HIGH LVL
	BOP	RESPONDS to alarms and ENTERS AOP-010
AOP-010		Feedwater Malfunctions
	SRO	ENTERS and directs actions of AOP-010. Makes PA announcement for AOP entry
Procedure Note:		**Steps 1 through 4 are immediate actions.**
Critical Task # 1	BOP	CHECK Feedwater Regulator valves operating properly. (NO) PERFORM the following: • PLACE affected Feedwater Regulator valve(s) in MANUAL. Places SG 'C' 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% (BOP - Main Feedwater Controls - Competency - Operate plant controls in Manual) ***Critical Task: Maintain control of SG 'C' level above 25% to prevent an automatic Reactor trip after the controlling level transmitter LT-496 fails high.*** IF Steam Generator level(s) cannot be controlled, THEN TRIP the Reactor AND GO TO EOP-E-0. (Should be controlled)
	BOP	CHECK ANY Main Feedwater Pump TRIPPED. (NO)

Op Test No.:	NRC	Scenario #	3	Event #	3	Page	22	of 69
Event Description:	**SG 'C' Controlling Level Transmitter (LT-496) fails high**							

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>CHECK DEH controlling Turbine Valves PROPERLY. (YES)</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td>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%</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td>CHECK Feedwater Regulator Valves operating properly in AUTO: (NO not 'C')
Response to SG levels
Valve position indication
Response to feed flow/steam flow mismatch
CONTROL Steam Generator levels using Feedwater Regulator Valve(s) in MANUAL.
MAINTAIN Steam Generator levels between 52-62%.
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.
(BOP - Main Feedwater Controls - Competency - Operate plant controls in Manual)</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>SRO</td><td>Provides control bands and trip limits IAW OMM-001 Attachment 13
<table><tr><th rowspan="2">Controller</th><th rowspan="2">Control Band</th><th colspan="2">Trip 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></td></tr>
<tr><td colspan="2">Procedure Note:</td><td>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.</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>BOP</td><td>CHECK turbine runs back less than 25% turbine load. (YES)</td></tr>
</table>

Op Test No.:	NRC	Scenario #	3	Event #	3	Page	23	of	69
Event Description:	**SG 'C' Controlling Level Transmitter (LT-496) fails high**								

Time	Position	Applicant's Actions or Behavior
	SRO	CHECK the following Pump status: • ALL Feedwater Train Pumps RUNNING (YES) • BOTH Heater Drain Pumps RUNNING (YES) GO TO the applicable section : 3.1 page 12, All Condensate/Feedwater flow malfunctions
	BOP	CHECK the following Recirc and Dump Valves operating properly in MODU: (YES) • Main Feedwater Pumps • Condensate Booster Pumps • Condensate Pumps • 1CE-293, Condensate Recirc • 1CE-142, Condensate Dump To CST Isolation Valve (SLB-4/7-1)
	BOP	CHECK the Condensate and Feedwater System INTACT. (YES)
	BOP	CHECK pumps for NORMAL OPERATION. (YES)
	SRO	NOTIFY Load Dispatcher of ANY load limitations. (NONE)
	SRO	CHECK Reactor thermal power changed by less than 15% in any one hour period. (YES) EXIT this procedure.
OWP-RP-07	SRO	Refer to OWP-RP-07 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.**

Op Test No.: NRC	Scenario # 3	Event # 3	Page 24 of 69
Event Description:	SG 'C' Controlling Level Transmitter (LT-496) fails high		

Time	Position	Applicant's Actions or Behavior
	SRO	Enters Instrumentation TS 3.3.1 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 3.3.2 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.** **Cue Event 4 (RCP 'B' #1 Seal degrades) after SG level is under control and the TS has been identified.**

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	25	of 69
Event Description:				**RCP ‘B’ #1 Seal degrades**				

Time	Position	Applicant’s Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 4** **“RCP ‘B’ #1 Seal degrades”**
Indications Available:		• ALB-008-4-3 RCP-B SEAL #1 LEAKOFF HIGH LOW FLOW
	RO	RESPONDS to alarm on ALB-008-4-3
Evaluator Note:		**ALB-008-4-3, response directs entry into AOP-018 IF No.1 or No. 2 seal appears to be damaged. Because seal injection remains higher than seal return the RCP temperature may remain stable. If crew does not enter AOP-018 based on high seal return flow Cue the booth to confirm the alarm with the following report.**
Simulator Communicator:		**Call as the RWCR operator and report the Seal Water Return Filter ∆P was ~5# over the previous 2 shifts, now it has risen to ~8# over the last 5 minutes.**
	CREW	CONFIRM alarm using ERFIS GD AOP-018 or FR-154A Identifies AOP-018 entry conditions
Evaluator Note:		**ALB-008-4-3, annunciator response directs entry into AOP-018 IF No.1 or No. 2 seal appears to be damaged.**
AOP-018		RCP Abnormal Conditions
	SRO	ENTERS and directs actions of AOP-018, RCP ABNORMAL CONDITIONS. Makes PA announcement for AOP entry
Procedure Note:		Step 1 is an immediate action. RCP abnormal conditions may require implementation of the SHNPP Emergency Plan.

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	26	of 69
Event Description:				**RCP ‘B’ #1 Seal degrades**				

<table>
<tr><th>Time</th><th>Position</th><th>Applicant’s Actions or Behavior</th></tr>
<tr><td>Immediate Action</td><td>RO</td><td>CHECK ANY CSIP RUNNING. (YES)</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>SRO</td><td>REFER TO PEP-110, Emergency Classification And Protective Action Recommendations, AND ENTER the EAL Matrix.</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Procedure Note:</td><td>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.</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>SRO</td><td>EVALUATE plant conditions AND GO TO the appropriate section:
<table>
<tr><th>MALFUNCTION</th><th>SECTION</th><th>Page</th></tr>
<tr><td>Loss of CCW and/or Normal Seal Injection to RCPs</td><td>3.1</td><td>5</td></tr>
<tr><td>High Reactor Coolant Pump Vibration</td><td>3.2</td><td>8</td></tr>
<tr><td>Reactor Coolant Pump Seal Malfunction</td><td>3.3</td><td>10</td></tr>
<tr><td>Reactor Coolant Pump Motor Trouble</td><td>3.4</td><td>18</td></tr>
</table>
Reactor Coolant Pump Seal Malfunction, Section 3.3 (Page 10)</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>CREW</td><td>CHECK ANY of the following conditions exist:
ANY RCP #1 Seal FAILED as defined in
• Attachment 2 (Page 25) (NO)
ANY RCP #2 Seal FAILED as defined in
• Attachment 2 (Page 25) (NO)
ANY RCPs operating outside the limits of
• Attachment 1 (Page 23) (NO)</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>SRO</td><td>RNO: GO TO Step 10.</td></tr>
</table>

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	27	of 69
Event Description:				**RCP 'B' #1 Seal degrades**				

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>RO</td><td>MONITOR RCP Radial Bearing and RCP Seal Water Inlet temperatures (reference OSI PI for AOP-018):
<table>
<tr><th></th><th colspan="3">ERFIS Points</th></tr>
<tr><th></th><th>RCP A</th><th>RCP B</th><th>RCP C</th></tr>
<tr><td>Pump Radial Brg Temp</td><td>TRC0131</td><td>TRC0128</td><td>TRC0125</td></tr>
<tr><td>Seal Water Inlet Temp</td><td>TRC0132</td><td>TRC0129</td><td>TRC0126</td></tr>
</table></td></tr>
<tr><td></td><td>RO</td><td>DETERMINE RCP Seal status using the following:
• Attachment 2
• Available plant indications</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>RO</td><td>CHECK RCPs free of #1 Seal degradation. (NO)</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>SRO</td><td>RNO: PERFORM the following for #1 Seal DEGRADED:</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>CREW</td><td>• DISPATCH an operator to CNMT to read the RCP #2 seal leakoff flow.
• VERIFY seal injection flow for the affected RCP is greater than or equal to 9 gpm.</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Procedure Note:</td><td>Total #1 seal flow is defined as the sum of #1 and #2 seal leakoff flows. When calculating total #1 seal flow with #1 seal leakoff flow greater than 6.0 gpm, #2 seal leakoff flow should be considered negligible until it can be read locally.</td></tr>
</table>

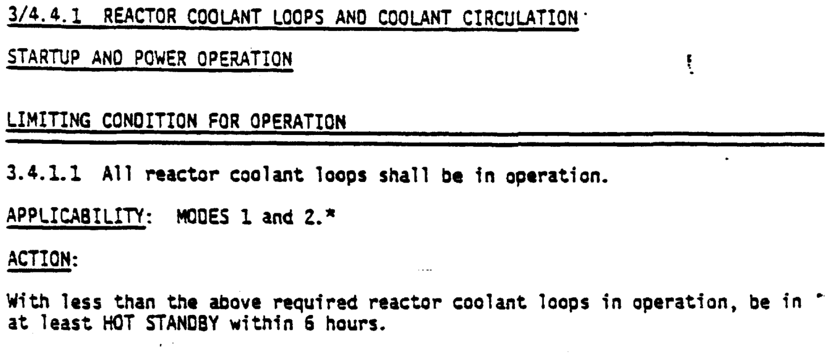
Op Test No.:	NRC	Scenario #	3	Event #	4	Page	28	of	69
Event Description:		**RCP 'B' #1 Seal degrades**							

Time	Position	Applicant's Actions or Behavior
	RO	• CALCULATE total #1 seal flow for the affected RCP as follows: (_______gpm + _______gpm) = _______gpm #1 Seal Leakoff Flow #2 Seal Leakoff Flow Total #1 Seal Flow • MAINTAIN seal injection flow greater than calculated total #1 seal flow. • INITIATE a plant shutdown using ONE of the following: o GP-006, Normal Plant Shutdown from Power Operation to Hot Standby o AOP-038, Rapid Downpower • STOP the affected RCP within 8 hours of exceeding 6.0 gpm #1 seal leakoff flow. • MONITOR ALB-005/1-2B, RCP THERM BAR HDR LOW FLOW, is CLEAR. (ensure adequate flow to RCP thermal barrier heat exchangers)
	RO	CHECK RCPs free of #1 Seal blockage. (YES)
	SRO	GO TO Step 19.
	RO	CONTINUE to monitor #1 seal parameters for indications of failure.
	RO	MONITOR RCP Seal injection flow to the unaffected RCPs.
	RO	ADJUST unaffected RCP Seal injection flow as necessary to maintain the following: • 8 to 13 gpm to the unaffected RCPs • Less than 31 gpm total flow to all RCPs
	RO	MAINTAIN CCW flow to all RCP Thermal Barrier HXs.

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	29	of	69
Event Description:				RCP ‘B’ #1 Seal degrades					

Time	Position	Applicant’s Actions or Behavior
Procedure Note:		High RCP #1 seal leakoff may cause local boiling in the thermal barrier of the affected pump. This may cause 1CC-252 to auto shut on high flow. Locally opening this valve may result in a slow heatup of the CCW System resulting in a small rise in CCW Surge Tank level.
Procedure Caution:		DO NOT restore CCW to an RCP that has lost all seal cooling for 4 minutes.
	RO	CHECK 1CC-252, RCP THERMAL BARRIERS FLOW CONTROL, OPEN. (YES)
	RO	CHECK RCPs free of #2 or #3 seal failure per Attachment 2 (Page 24). (YES)
	SRO	CONSULT with Responsible Engineer for recommended follow-up actions.
Procedure Caution:		Following a complete loss of Seal Cooling, the affected RCP(s) should NOT be started prior to a status evaluation.
	SRO	INITIATE appropriate corrective action for repair of the damaged seal(s).
	SRO	Contacts Maintenance to coordinate Containment entry to obtain local reading of #2 seal leakoff for RCP ‘B’ and plan repair actions.
Simulator Communicator:		**Respond to crew requests.**

Op Test No.:	NRC	Scenario #	3	Event #	4	Page	30	of	69
Event Description:		**RCP ‘B’ #1 Seal degrades**							

Time	Position	Applicant’s Actions or Behavior
	SRO	REFER TO the following Tech Specs: • 3.2.3 • 3.3.1 • 3.4.1.1 • 3.4.1.2 • 3.4.1.3 • 3.4.6.2 Evaluates Reactor Coolant System T.S. 3.4.1.1  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. **Entry into the action statement is not required until the RCP ‘B’ is secured. Tech Specs may be asked as a follow up question.**
Evaluator Note:		**Actions to reduce power to remove the RCP ‘B” from service do NOT have to be restarted to continue the scenario. Cue Event 5 SG’B’ PORV opens**

Op Test No.: NRC	Scenario # 3	Event # 5	Page 31 of 69
Event Description:	**SG 'B' PORV Opens**		

Time	Position	Applicant's Actions or Behavior

Simulator Operator:	**On cue from the Lead Evaluator insert Trigger 5** **SG 'B' PORV Press Inst fails high w/ PORV staying OPEN – can be manually shut**

Evaluator Note:	**This event is a Steam Generator PORV Pressure Instrument failing high. This will require the BOP to take manual control of the PORV to shut it. The SRO should evaluate Tech Specs 3.3.3.5, Remote Shutdown System, and 3.6.3, Containment Isolation Valves.**

Available Indications:		• **ALB-014-8-5, Computer Alarm Steam Generators**
ALB-014	SRO	ENTERS APP-ALB-014-8-5
	BOP	IDENTIFIES 'B' SG PORV is OPEN
	BOP	In accordance with AD-OP-ALL-1000, the operator may take MANUAL actions when automatic actions do not occur and DEPRESSES the Manual Pushbutton for PK-308C1 to take manual control of 'B' SG PORV
	SRO	Provide pressure band for PORV manual control (Maintain < 1170 psig).
	BOP	LOWER output for PK-308 to SHUT 'C' SG PORV 1MS-82 (PORV will shut) Informs CRS that 'B' SG PORV is SHUT

Op Test No.:	NRC	Scenario #	3	Event #	5	Page	32	of	69
Event Description:				**SG ‘B’ PORV Opens**					

Time	Position	Applicant’s Actions or Behavior
	SRO	The SRO will complete OMM-001 Attachment 5 Equipment Problem Checklist for the failure of SG B PORV The SRO should evaluate TS 3.3.3.5 and TS 3.6.3. TS 3.6.3 Action: Isolate the affected penetration within 4 hours. The redundant manual isolation valve per PLP-106, Attachment 5 is 1MS-61. Contacts WCC and support personnel for repairs.
Simulator Operator:		**If requested to Shut 1MS-61 open Sim Drawing MSS mss01 and shut 1MS-61 (rf mss025 to zero)** **After shutting 1MS-61 wait 1 minute and report that 1MS-61 has been manually shut .**
Lead Evaluator:		**Once the plant has stabilized, cue Event 6, RCP “B” #1 seal fails**

Op Test No.: NRC	Scenario # 3	Event # 6	Page	33	of	69	
Event Description:	RCP 'B' #1 Seal fails						

Time	Position	Applicant's Actions or Behavior

Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 6 "RCP 'B' #1 Seal failure"**
Indications Available:		ALB-008-4-3 RCP-B SEAL #1 LEAKOFF HIGH LOW FLOW ALB-008-4-4 RCP-B SEAL #2 LEAKOFF HIGH FLOW
	CREW	CONFIRM alarm using ERFIS GD AOP-018 or FR-154A

	CREW	CHECK ANY of the following conditions exist: ANY RCP #1 Seal FAILED as defined in Attachment 2 (YES) ANY RCP #2 Seal FAILED as defined in Attachment 2 (NO) ANY RCPs operating outside the limits of Attachment 1 (NO)
	RO	CHECK the Reactor is TRIPPED. (NO)
	SRO	RNO: TRIP the Reactor AND GO TO EOP-E-0. (Perform Steps 3 through 9 as time permits.)
Evaluator Note:		**The SRO should conduct a Crew Update and review AOP-018 Section 3.3 steps 3 through 9 and direct these actions to be performed after the E-0 immediate actions are verified complete.**
	SRO	Directs RO to perform a MANUAL Reactor trip IAW AOP-018. Transitions to EOP-E-0.
	RO	Initiates a MANUAL Reactor trip.

EOP-E-0		REACTOR TRIP OR SAFETY INJECTION		
	SRO	ENTERS and directs actions of EOP-E-0 Makes PA announcement for EOP entry		
	RO/BOP	Performs EOP-E-0 immediate actions.		
Procedure Note:		**Steps 1 through 4 are immediate action steps. (All high level steps and confirmatory steps are performed and broadcast by the OATC and BOP from memory.** **Informational NOTES, including this one, the phrase "Perform the following," and information presented in table format need not be verbalized.** **Foldout applies. (Immediate actions should be completed prior implementing Foldout Page items.)**		
Immediate Actions	RO	VERIFY Reactor Trip:		
		REACTOR TRIP CONFIRMATION		
		Reactor Trip AND Bypass BKRs – OPEN		(YES)
		Rod Bottom Lights (Zero Steps) – LIT		(YES)
		Neutron Flux – DROPPING		(YES)
Immediate Actions	BOP	Check Turbine Trip – ALL THROTTLE VALVES SHUT		
		TURB STOP VLV 1	TSLB-2-11-1	(YES)
		TURB STOP VLV 2	TSLB-2-11-2	(YES)
		TURB STOP VLV 3	TSLB-2-11-3	(YES)
		TURB STOP VLV 4	TSLB-2-11-4	(YES)

Op Test No.:	NRC	Scenario #	3	Event #	6	Page	35 of 69
Event Description:			RCP 'B' #1 Seal fails				

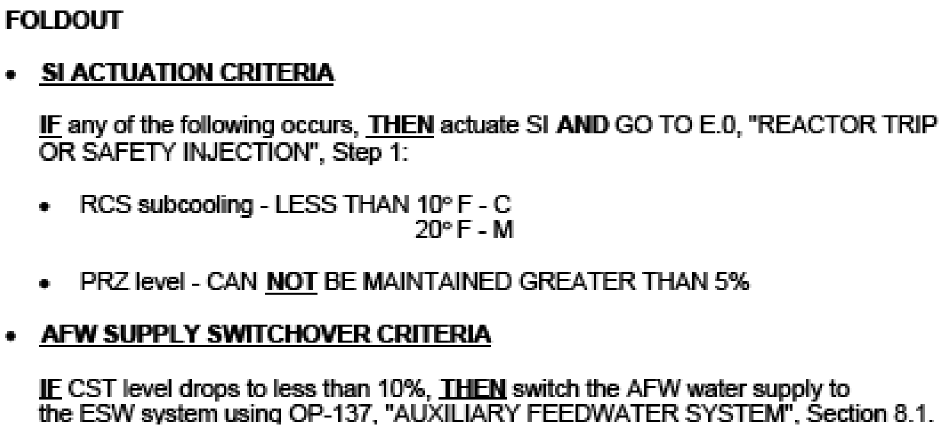
Time	Position	Applicant's Actions or Behavior	
Immediate Actions	BOP	Perform The Following: • AC Emergency Buses – AT LEAST ONE ENERGIZED • AC Emergency Buses – BOTH ENERGIZED	(YES) (YES)
Immediate Actions	RO	Safety Injection – ACTUCATED (BOTH TRAINS) BPLP 4-1,"SI ACTUATED" - LIT (CONTINUOUSLY)	(NO)
Immediate Actions	RO	RNO Perform the following: ○ Check Safety Injection – REQUIRED SI ACTUATION CRITERIA PRZ Pressure - LESS THAN OR EQUAL TO 1850 PSIG CNMT Pressure - GREATER THAN OR EQUAL TO 3.0 PSIG Any SG Pressure - LESS THAN OR EQUAL TO 601 PSIG Manual - DEGRADATION TOWARDS AUTOMATIC ACTUATION Abnormal Operating Procedure - DIRECTS MANUAL ACTUATION One SI Train - FAILED (BPLP 4-1 FLASHING) ○ **IF** Safety Injection actuation is **NOT** required, **THEN** GO TO ES-0.1, "REACTOR TRIP RESPONSE", Step 1.	(NO)
Evaluator Note:		**AOP-018 Section 3.3 steps 3 through 9 will be completed as time permits during the transition to EOP-ES-0.1.**	
	RO	STOP the affected RCP(s).	
	RO	CHECK that ANY RCP was SECURED due to a #1 Seal FAILED.	(YES)

Op Test No.: NRC	Scenario # 3	Event # 6	Page	36 of 69
Event Description:		RCP 'B' #1 Seal fails		

Time	Position	Applicant's Actions or Behavior	
	RO	SHUT the affected RCP Seal Water Return Valve(s) between three (3) and five (5) minutes after securing the RCP: • 1CS-355, RCP A #1 Seal Water Return • **1CS-396, RCP B #1 Seal Water Return** • 1CS-437, RCP C #1 Seal Water Return (1CS-396, RCP B #1 Seal Water Return required to be shut)	
	RO	MONITOR ALB-005/1-2B, RCP THERM BAR HDR LOW FLOW, is CLEAR. (ensure adequate flow to RCP thermal barrier heat exchangers)	
	RO	CHECK RCP A RUNNING.	(YES)
	RO	CHECK RCP B RUNNING.	(NO)
	RO	RNO: PLACE PK-444D.1 in MANUAL and SHUT with demand at 0% (1RC-103, PRZ Spray Loop B).	
	SRO	EXIT this procedure.	
	SRO	Returns to procedure in effect (E-0) and transitions to ES-0.1	
Evaluator's Note:		**Once AOP-018, steps 3 through 9 are complete and the transition to EOP-ES-0.1 has occurred the crew will stabilize RCS temperature per step 4.**	

Op Test No.:	NRC	Scenario #	3	Event #	6	Page	37 of 69
Event Description:				**RCP 'B' #1 Seal fails**			

Time	Position	Applicant's Actions or Behavior
	SRO	GO TO EOP-ES-0.1, Step 1. Holds a crew update
EOP-ES-0.1		REACTOR TRIP RESPONSE
Procedure Note:		**Foldout applies.**



FOLDOUT

- **SI ACTUATION CRITERIA**

  IF any of the following occurs, **THEN** actuate SI **AND** GO TO E.0, "REACTOR TRIP OR SAFETY INJECTION", Step 1:

  - RCS subcooling - LESS THAN 10° F - C
    20° F - M
  - PRZ level - CAN **NOT** BE MAINTAINED GREATER THAN 5%

- **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.

Time	Position	Applicant's Actions or Behavior	
	SRO	Initiates Monitoring Of Critical Safety Function Status Trees Informs Shift Manager to evaluate EAL Matrix. (Refer to PEP-110)	
	RO	Check RCS Temperature: • Check RCPs – ANY RUNNING • Check SG blowdown isolation valves – SHUT	(YES) (NO) SG levels are >25%

SG	(MLB-1A-SA)	(MLB-1B-SB)
A	1BD-11	1BD-1
B	1BD-30	1BD-20
C	1BD-49	1BD-39

- Shuts SG Blowdown FCVs

<table>
<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 210 KPPH until level greater than 25% at least one 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>

Controls feed flow and steam dumps to stabilize temperature between 555˚F AND 559˚F</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td colspan="2">Evaluator's Note:</td><td>Cue Event 7 - Steam Space LOCA inside Containment after the crew has stabilized RCS temperature per above table to satisfaction of Evaluators (Step 4 of ES-0.1).</td></tr>
</table>

Op Test No.:	NRC	Scenario #	3	Event #	7	Page	39	of	69
Event Description:				**Steam Space LOCA inside Containment**					

Time	Position	Applicant's Actions or Behavior
Simulator Operator:		**On cue from the Lead Evaluator actuate Trigger 7** **Steam Space LOCA inside containment**
Indications Available:		• ALB-009-3-3, PRZ CONT Low Press And Heaters ON • ALB-009-5-1, Pressurizer High-Low Press • ALB-009-5-3, Pressurizer Low Press Alert • ALB-009-8-1, Pressurizer Relief Tank High-Low Level Press Or Temp • ALB-009-8-2, Pressurizer Relief Discharge High Temp • ALB-009-8-3, Pressurizer Safety Relief Discharge High Temp
	CREW	CONFIRM alarm using: • Safety valve discharge line temperatures TI-465, TI-467, TI-469 • Pressurizer relief tank level, pressure, and temperature LI-470.1, PI-472.1, and TI-471.1 • PRZ pressure indication PI-444, PI-445.1, PI-455.1, PI-456, and PI-457
Evaluator Note:		**The foldout for SI Actuation on Pressurizer level cannot be maintained > 5% may not trigger the crew to manually SI before the automatic low pressurizer press Safety Injection value of 1850 psig will be reached.**
	RO	Review Foldout SI Actuation Criteria: IF any of the following occurs, THEN actuate SI AND GO TO E.0, "REACTOR TRIP OR SAFETY INJECTION", Step 1: • RCS subcooling - LESS THAN 10°F – C / 20°F – M • PRZ level - CAN NOT BE MAINTAINED > 5% (RCS subcooling less than 10°F – C) **Actuates SI – takes the SI manual actuate switch to actuate and verifies BOTH Trains actuated**

Op Test No.:	NRC	Scenario #	3	Event #	7	Page	40	of	69
Event Description:			**Steam Space LOCA inside Containment**						

Time	Position	Applicant's Actions or Behavior
EOP-E-0		Reactor Trip Or Safety Injection
	SRO	Enters EOP-E-0 Holds crew update
	RO/BOP	Performs E-0 Immediate Actions.
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 2200 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. **Crew should verify Alternate Miniflow Isolation Valves or Miniflow Block Valves CLOSE when RCS Pressure lowers to less than 1800 PSIG** **AND** **Stop all RCPs when RCS pressure is less than 1400 PSIG with SI flow is greater than 200 GPM.**

Op Test No.:	NRC	Scenario #	3	Event #	8	Page	41	of	69
Event Description:				Steam Space LOCA / 'A' RHR Pump Trip					

Time	Position	Applicant's Actions or Behavior	
Critical Task # 2	RO	Any of the following - RNO action met or Foldout - RCP trip criteria is met or PHASE B Actuation RCP Trip Criteria: IF both of the following occurs, THEN stop all RCPS: • SI flow - GREATER THAN 200 GPM • RCS pressure - LESS THAN 1400 PSIG ***Critical Task: Manually trip all RCPs during a Small Break LOCA event within 10 minutes after RCS pressure is < 1400 psig and SI flow is > 200 gpm***	(YES) (YES)
	SRO	Evaluate EAL Matrix (Refer to PEP-110)	
	RO	Verify CSIPs – ALL RUNNING	(YES)
Event 8	RO	Verify RHR pumps – ALL RUNNING • REPORTS 'A' RHR has tripped and 'B' RHR is under clearance	(NO)
	CREW	DISPATCH operators to investigate trip of 'A' RHR	
Simulator Communicator:		**After being contacted -** **TB AO: Report back in ~ 2 minutes: 'A' RHR Pump Breaker has overcurrent flags dropped. The breaker cubical appears to be damaged. You contacted Electrical maintenance who are now looking at the breaker damage. They are in contact with WCC and are assessing what would need to be done to repair the problem.** **RAB AO: Nothing is visibly wrong locally at the pump.**	
	SRO	CONTACTS WCC to have 'B' RHR restored	
Simulator Communicator:		**WCC will contact maintenance and work toward lifting the clearance on the 'B' RHR Pump.**	

Op Test No.: NRC Scenario # 3 Event # 8		Page 42 of 69
Event Description:		**Steam Space LOCA / ‘A’ RHR Pump Trip**
Time	Position	Applicant’s Actions or Behavior

Evaluator Note:		**‘B’ RHR Pump will NOT be returned to service during this scenario.**	
Simulator Communicator:		**IF pressed by the crew to restore ‘B’ RHR pump inform them that Maintenance said estimated time for recover is no sooner than 2 hours from now.**	
	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:	
	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 MANUAL - DEGRADATION TOWARDS AUTOMATIC ACTUATION *This will depend on Containment pressure when the crew gets to this step	(NO*)
	RO	CHECK CNMT Pressure – HAS REMAINED LESS THAN 10 PSIG	(YES)
	BOP	Verify AFW flow – AT LEAST 200 KPPH ESTABLISHED	(YES)

Op Test No.:	NRC	Scenario #	3	Event #	7	Page	43	of	69
Event Description:				**Steam Space LOCA (continued)**					

Time	Position	Applicant's Actions or Behavior

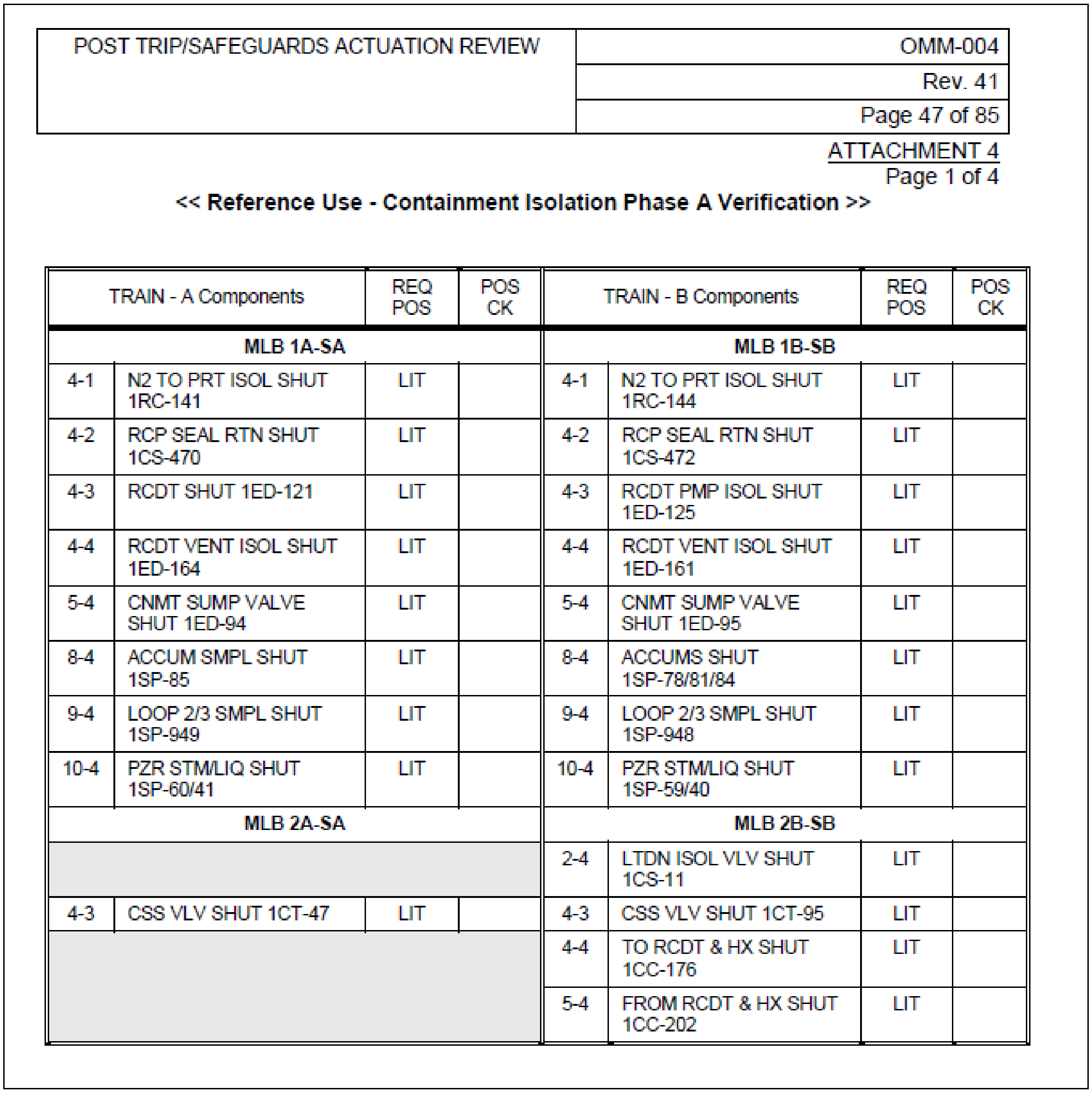
Evaluator Note:		**RNO may apply if AFW has been previously reduced.**	
	BOP	Sequencer Load Block 9 (Manual Loading Permissive) - ACTUATED (BOTH TRAINS)	(YES)
	BOP	Energize AC buses 1A1 AND 1B1	
Evaluator Note:		**The RO will perform all board actions until the BOP completes Attachment 3. The BOP is 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.**	
Simulator Operator:		**When contacted to place A/B air compressors in Local Control mode, run CAEP :\air\ACs_to_local.txt.**	

Op Test No.: NRC	Scenario # 3	Event # 7	Page 44 of 69
Event Description:	Steam Space LOCA (continued)		

Time	Position	Applicant's Actions or Behavior
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 & disch valve power.txt.**
Simulator Communicator:		**When the CAEP is complete, report task to the MCR.**

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	45	of 69
Event Description:	**Phase ‘A’ fails on the ‘B’ train and partially isolates on the ‘A’ train**							

Time	Position	Applicant’s Actions or Behavior
Event 9	BOP	**Attachment 3, Step 6** Verify CNMT Phase A Isolation Valves – SHUT (Refer to OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 4.) See below and the following pages for Attachment 4 valve verifications.



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<< Reference Use - Containment Isolation Phase A Verification >>

TRAIN - A Components		REQ POS	POS CK	TRAIN - B Components		REQ POS	POS CK
MLB 1A-SA				**MLB 1B-SB**			
4-1	N2 TO PRT ISOL SHUT 1RC-141	LIT		4-1	N2 TO PRT ISOL SHUT 1RC-144	LIT	
4-2	RCP SEAL RTN SHUT 1CS-470	LIT		4-2	RCP SEAL RTN SHUT 1CS-472	LIT	
4-3	RCDT SHUT 1ED-121	LIT		4-3	RCDT PMP ISOL SHUT 1ED-125	LIT	
4-4	RCDT VENT ISOL SHUT 1ED-164	LIT		4-4	RCDT VENT ISOL SHUT 1ED-161	LIT	
5-4	CNMT SUMP VALVE SHUT 1ED-94	LIT		5-4	CNMT SUMP VALVE SHUT 1ED-95	LIT	
8-4	ACCUM SMPL SHUT 1SP-85	LIT		8-4	ACCUMS SHUT 1SP-78/81/84	LIT	
9-4	LOOP 2/3 SMPL SHUT 1SP-949	LIT		9-4	LOOP 2/3 SMPL SHUT 1SP-948	LIT	
10-4	PZR STM/LIQ SHUT 1SP-60/41	LIT		10-4	PZR STM/LIQ SHUT 1SP-59/40	LIT	
MLB 2A-SA				**MLB 2B-SB**			
				2-4	LTDN ISOL VLV SHUT 1CS-11	LIT	
4-3	CSS VLV SHUT 1CT-47	LIT		4-3	CSS VLV SHUT 1CT-95	LIT	
				4-4	TO RCDT & HX SHUT 1CC-176	LIT	
				5-4	FROM RCDT & HX SHUT 1CC-202	LIT	

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	46 of 69

Event Description: **Phase 'A' fails on the 'B' train and partially isolates on the 'A' train**

Time	Position	Applicant's Actions or Behavior

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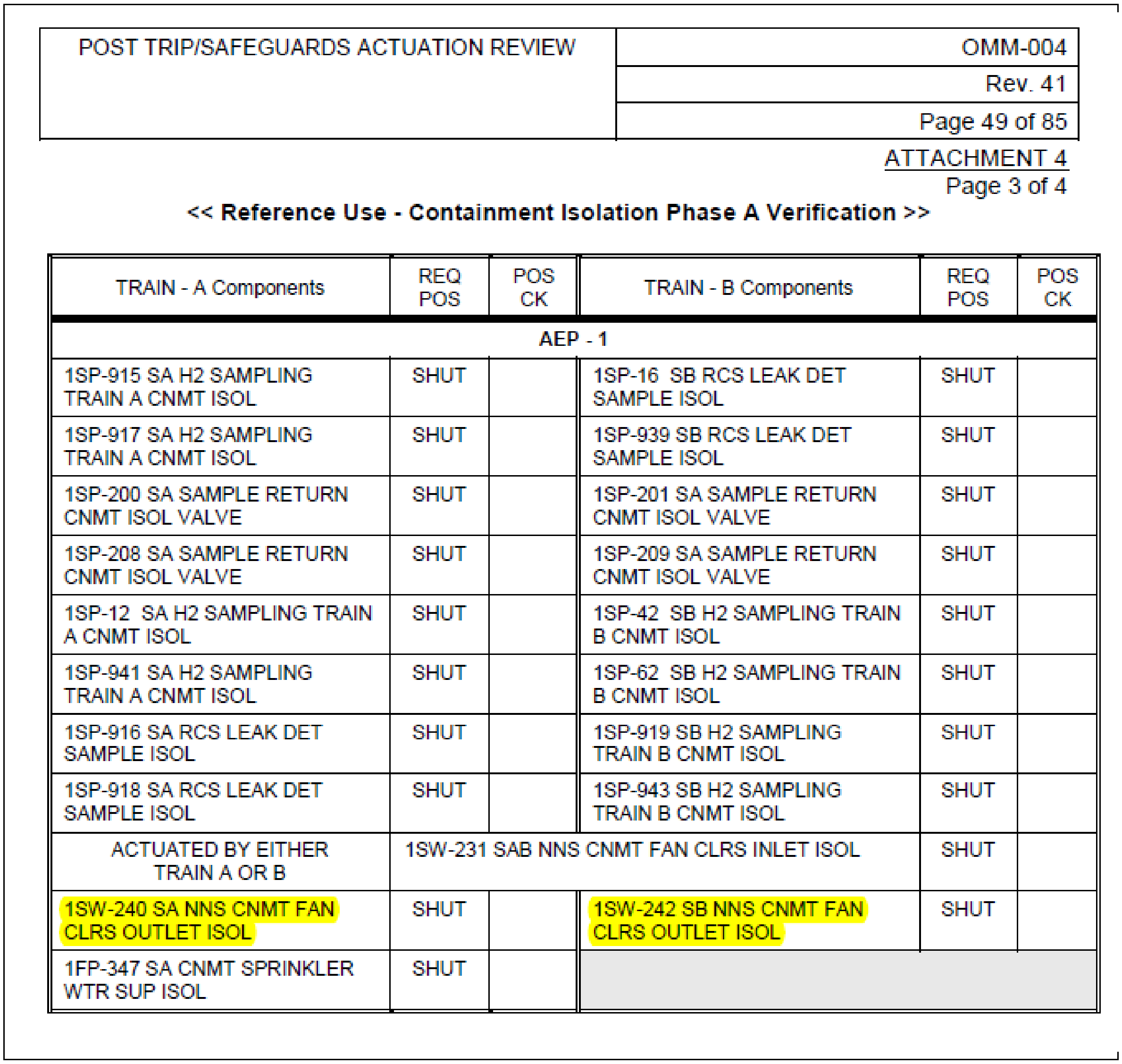
<< Reference Use - Containment Isolation Phase A Verification >>

TRAIN - A Components	REQ POS	POS CK	TRAIN - B Components	REQ POS	POS CK
MAIN CONTROL BOARD					
A-SA CONTAINMENT SPRAY PUMP	STOP		B-SB CONTAINMENT SPRAY PUMP	STOP	
1CT-24 SA CONTAINMENT SPRAY EDUCTOR TEST	SHUT		1CT-25 SB CONTAINMENT SPRAY EDUCTOR TEST	SHUT	
1IA-819 CONTAINMENT INSTRUMENT AIR	SHUT		1SI-179 ACCUMULATOR FILL FROM RWST	SHUT	
1SI-263 ACCUMULATOR CHECK VALVE TEST RETURN	SHUT		1SI-264 ACCUMULATOR CHECK VALVE TEST RETURN	SHUT	
1CS-7 45 GPM LETDOWN ORIFICE A	SHUT		1SI-287 ACCUMULATORS & PZR PORV N2 SUPPLY	SHUT	
1CS-8 60 GPM LETDOWN ORIFICE B	SHUT		1RC-161 RMW TO CNMT	SHUT	
1CS-9 60 GPM LETDOWN ORIFICE C	SHUT				
ACTUATED BY EITHER TRAIN A OR B	AH-37 A NNS FAN COOLER			STOP	
	AH-37 B NNS FAN COOLER			STOP	
	AH-38 A NNS FAN COOLER			STOP	
	AH-38 B NNS FAN COOLER			STOP	
	AH-39 A NNS FAN COOLER			STOP	
	AH-39 B NNS FAN COOLER			STOP	
AEP - 1					
ACTUATED BY EITHER TRAIN A OR B	1MS-25 SAB STM GEN A STM ANALYZER ISOL			SHUT	
	1MS-27 SAB STM GEN B STM ANALYZER ISOL			SHUT	
	1MS-29 SAB STM GEN C STM ANALYZER ISOL			SHUT	

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	47	of	69

Event Description: **Phase ‘A’ fails on the ‘B’ train and partially isolates on the ‘A’ train**

Time	Position	Applicant’s Actions or Behavior



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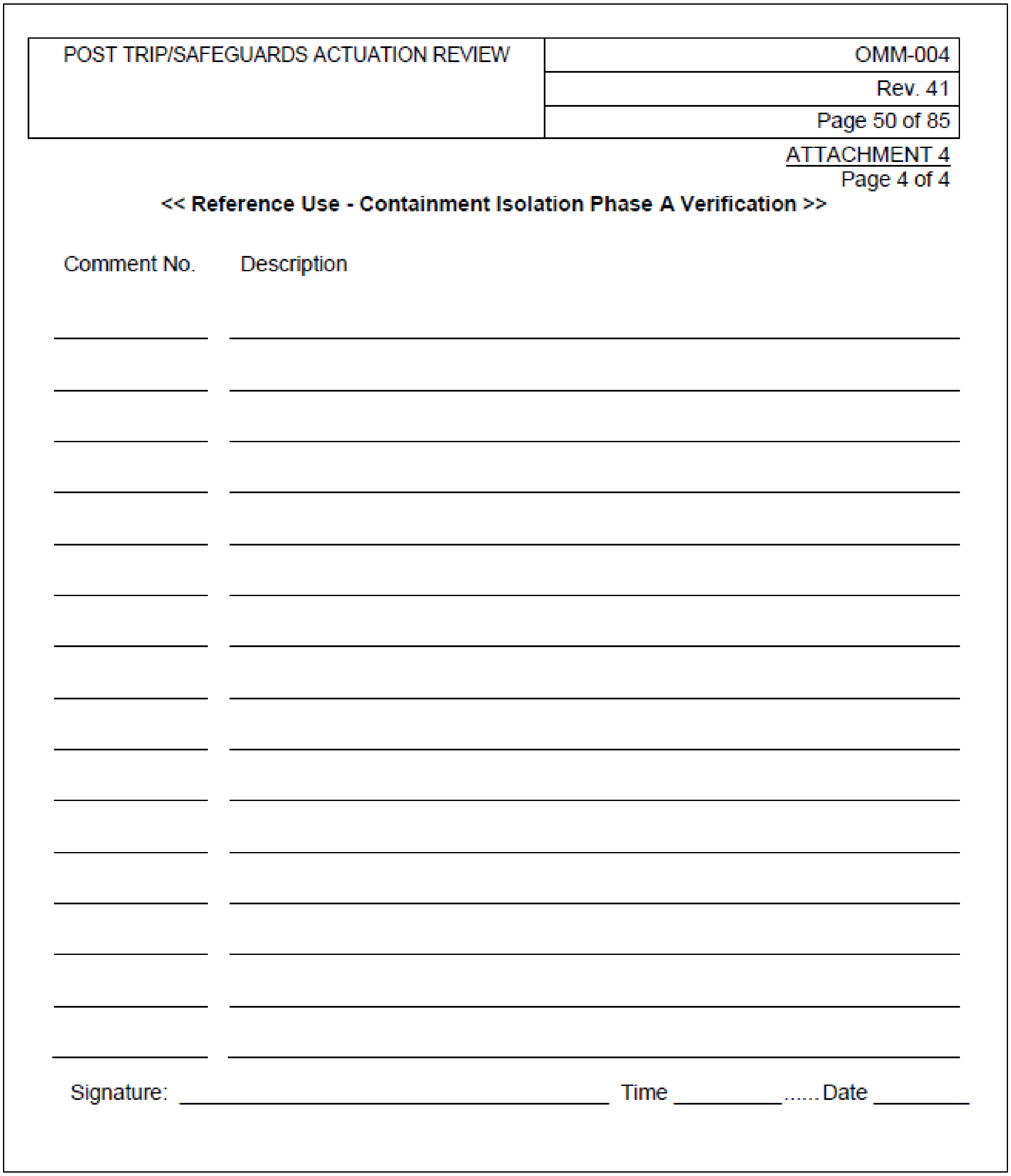
<< Reference Use - Containment Isolation Phase A Verification >>

TRAIN - A Components	REQ POS	POS CK	TRAIN - B Components	REQ POS	POS CK
AEP - 1					
1SP-915 SA H2 SAMPLING TRAIN A CNMT ISOL	SHUT		1SP-16 SB RCS LEAK DET SAMPLE ISOL	SHUT	
1SP-917 SA H2 SAMPLING TRAIN A CNMT ISOL	SHUT		1SP-939 SB RCS LEAK DET SAMPLE ISOL	SHUT	
1SP-200 SA SAMPLE RETURN CNMT ISOL VALVE	SHUT		1SP-201 SA SAMPLE RETURN CNMT ISOL VALVE	SHUT	
1SP-208 SA SAMPLE RETURN CNMT ISOL VALVE	SHUT		1SP-209 SA SAMPLE RETURN CNMT ISOL VALVE	SHUT	
1SP-12 SA H2 SAMPLING TRAIN A CNMT ISOL	SHUT		1SP-42 SB H2 SAMPLING TRAIN B CNMT ISOL	SHUT	
1SP-941 SA H2 SAMPLING TRAIN A CNMT ISOL	SHUT		1SP-62 SB H2 SAMPLING TRAIN B CNMT ISOL	SHUT	
1SP-916 SA RCS LEAK DET SAMPLE ISOL	SHUT		1SP-919 SB H2 SAMPLING TRAIN B CNMT ISOL	SHUT	
1SP-918 SA RCS LEAK DET SAMPLE ISOL	SHUT		1SP-943 SB H2 SAMPLING TRAIN B CNMT ISOL	SHUT	
ACTUATED BY EITHER TRAIN A OR B	1SW-231 SAB NNS CNMT FAN CLRS INLET ISOL			SHUT	
1SW-240 SA NNS CNMT FAN CLRS OUTLET ISOL	SHUT		1SW-242 SB NNS CNMT FAN CLRS OUTLET ISOL	SHUT	
1FP-347 SA CNMT SPRINKLER WTR SUP ISOL	SHUT				

Op Test No.:	NRC	Scenario #	3	Event #	9	Page	48	of	69

Event Description: **Phase ‘A’ fails on the ‘B’ train and partially isolates on the ‘A’ train**

Time	Position	Applicant’s Actions or Behavior



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<< Reference Use - Containment Isolation Phase A Verification >>

Comment No. Description

Signature: ______________________ Time __________ ...... Date _________

Op Test No.: NRC	Scenario # 3	Event # 7	Page 49 of 69
Event Description:	**Steam Space LOCA inside containment (continued)**		

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>RO / BOP</td><td colspan="2">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.
• IF no RCPs running, THEN 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 210 KPPH until level greater than 25% [40%] in at least one on intact SG
• IF cooldown continues, THEN, shut MSIVs AND BYPASS valves</td><td>• 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</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>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td>PRZ PORVs – SHUT</td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>RO</td><td>PRZ spray valves – SHUT</td><td>(YES)</td></tr>
<tr><td></td><td></td><td></td><td></td></tr>
<tr><td></td><td>RO</td><td>PRZ PORV block valves – AT LEAST ONE OPEN (All OPEN)</td><td>(YES)</td></tr>
</table>

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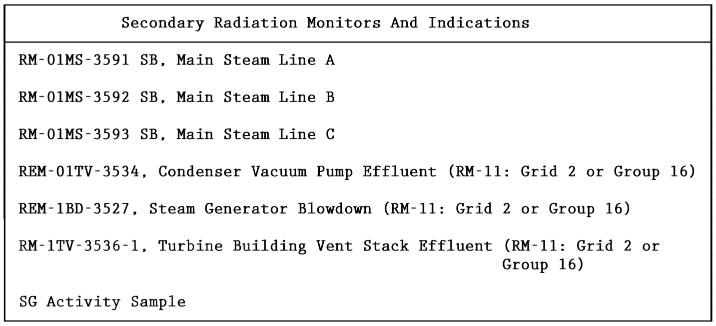
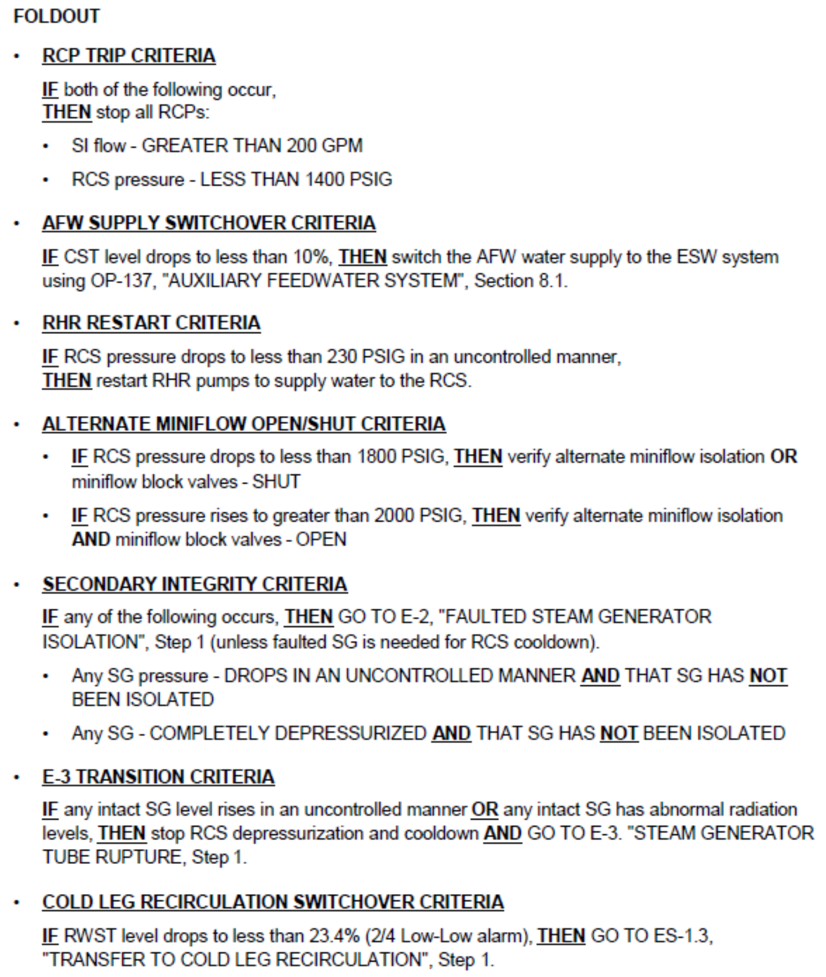
Event Description: **Steam Space LOCA inside containment (continued)**

Time	Position	Applicant's Actions or Behavior	
	RO / BOP	ANY SG pressures – DROPPING IN AN UNCONTROLLED MANNER OR COMPLETELY DEPRESSURIZED	(NO) (NO)
	SRO	RNO: GO TO Step 27.	
	BOP	ANY SG pressures – ABNORMAL RADIATION OR UNCONTROLLED LEVEL RISE  Secondary Radiation Monitors And Indications RM-01MS-3591 SB, Main Steam Line A RM-01MS-3592 SB, Main Steam Line B RM-01MS-3593 SB, Main Steam Line C REM-01TV-3534, Condenser Vacuum Pump Effluent (RM-11: Grid 2 or Group 16) REM-1BD-3527, Steam Generator Blowdown (RM-11: Grid 2 or Group 16) RM-1TV-3536-1, Turbine Building Vent Stack Effluent (RM-11: Grid 2 or Group 16) SG Activity Sample	(NO) (NO)
	SRO	RNO: GO TO Step 30.	
Evaluator Note:		**Containment pressure will not begin to rise until the PRT rupture disk pressure of 100 psig is exceeded. Due to this delay and the quench volume of the PRT the pressure in containment will slowly rise, but the crew should determine the containment pressure is NOT normal.**	
	RO	CNMT Pressure - NORMAL	(NO)
	SRO	RNO: GO TO EOP-E-1, "LOSS OF REACTOR OR SECONDARY COOLANT", Step 1.	

Op Test No.:	NRC	Scenario #	3	Event #	7	Page	51	of	69
Event Description:	**Steam Space LOCA inside containment (continued)**								

Time	Position	Applicant's Actions or Behavior

EOP-E-1		LOSS OF REACTOR OR SECONDARY COOLANT
	SRO	Prior to giving crew directions from EOP-E-1, performs crew update. Implements procedure
Procedure Note:		**Foldout applies.**
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.

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Event Description: **Steam Space LOCA inside containment (continued)**

Time	Position	Applicant's Actions or Behavior	
	SRO	Reviews foldout items Then: Initiates Monitoring Of Critical Safety Function Status Trees	
	RO	Maintain RCP Seal Injection Flow Between 8 GPM And 13 GPM.	
	BOP	Check Intact SG Levels:	
	BOP	• Any level - GREATER THAN 25% [40%] • 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.	
	SRO	Check PRZ PORV AND Block Valves:	
	BOP	• Verify AC buses 1A1 AND 1B1 – ENERGIZED	(YES)
	RO	• Check PRZ PORVs – SHUT	(YES)
	SRO	• GO TO Step 4.f	
	RO	• Check block valves – AT LEAST ONE OPEN (All OPEN)	(YES)
	SRO	Check SI Termination Criteria:	
	RO	• RCS subcooling - GREATER THAN 10°F [40°F] – C 20°F [50°F] - M	(NO)

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Event Description: **Steam Space LOCA inside containment (continued)**

Time	Position	Applicant's Actions or Behavior	
	SRO	• RNO: GO TO Step 6.	
	SRO	Check CNMT Spray Status:	
	RO	• Check any CNMT spray pump - RUNNING	(NO)
	SRO	• RNO: GO TO Step 7.	
	SRO	Check Source Range Detector Status:	
	RO	• Intermediate range flux - LESS THAN 5x10-11 AMPS	(YES)
	RO	• Verify source range detectors - ENERGIZED	
	RO	• Transfer nuclear recorder to source range scale.	
	SRO	Check RHR Pump Status:	
	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	RCS pressure – STABLE OR RISING RNO – GO TO Step 9	(NO)

Op Test No.:	NRC	Scenario #	3	Event #	7	Page	54	of	69

Event Description: **Steam Space LOCA inside containment (continued)**

Time	Position	Applicant's Actions or Behavior	
	SRO	Check RCS And SG Pressures:	
	BOP	• All SG Pressures - STABLE OR RISING • RCS pressure - STABLE OR DROPPING	(YES) (YES)
	SRO	Establish CCW Flow To The RHR Heat Exchangers:	
	RO	• Verify both CCW pumps - RUNNING	(YES)
	RO	• Open the following valves: o 1CC-147 o 1CC-167	
	RO	• Verify CCW flow to the RHR heat exchanger(s).	
	RO	• Perform one of the following to establish two independent CCW systems: o Shut train A CCW non-essential supply AND return valves: • 1CC-99 • 1CC-128 o Shut train B CCW non-essential supply AND return valves: • 1CC-113 • 1CC-127	
	SRO	Check EDG Status:	
	BOP	• Check AC emergency buses 1A-SA AND 1B-SB - ENERGIZED BY OFFSITE POWER: o Check bus voltages o Check breakers 105 AND 125 - CLOSED	(YES) (YES)

Op Test No.: NRC	Scenario # 3	Event # 7	Page 55 of 69
Event Description:	**Steam Space LOCA inside containment (continued)**		

Time	Position	Applicant's Actions or Behavior	
	SRO	• GO TO Step 11e.	
	SRO	• Check any EDG - RUNNING UNLOADED (YES)	
	RO	• Reset SI.	
	SRO	• Manually Realign Safeguards Equipment Following A Loss Of Offsite Power. • (Refer to EOP-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:	
	RO	• RHR system - CAPABLE OF COLD LEG RECIRCULATION	(NO)
	SRO	• RNO: GO TO EOP-ECA-1.1, "LOSS OF COLD LEG RECIRCULATION EMERGENCY COOLANT RECIRCULATION", Step 1.	
Evaluator Note:		**The Steam Space LOCA will cause a rapid RCS pressure and temperature decrease. Entry into EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock will be required based on these conditions. When the crew identifies that a RED path condition is occurring they will immediately transition from the procedure they are implementing to EOP-FR-P.1. The following pages are steps from EOP-FR-P.1. Steps for EOP-ECA-1.1 follow the EOP-FR-P.1 steps.**	

Op Test No.: NRC	Scenario # 3	Event # 7	Page 56 of 69	
Event Description:	**Steam Space LOCA inside containment (continued)**			
Time	Position	Applicant's Actions or Behavior		

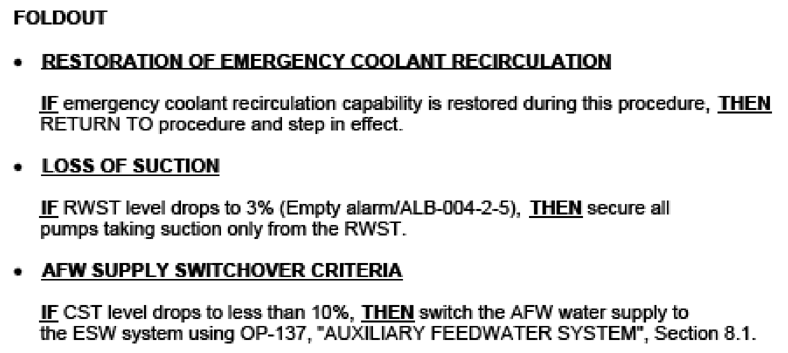
EOP-FR-P.1		Response to Imminent Pressurized Thermal Shock	
	SRO	Prior to giving crew directions from EOP-FR-P.1, performs crew update. Foldout applies	
	RO	RCS pressure - LESS THAN 230 PSIG Any RHR HX header flow > 1000 GPM RNO – GO TO STEP 2	(NO) (NO)
	RO	Check RCS Cold Leg Temperature Trend: Check RCS Cold Leg Temperatures - STABLE **OR** RISING RNO – GO TO STEP 3	(NO)
Procedure Note:		**A faulted SG is any SG that is depressurizing in an uncontrolled manner or is completely depressurized.**	
	RO	Stop RCS Cooldown: Verify SG PORVs – SHUT	(YES)
	BOP	Verify condenser steam dump valves – SHUT	(YES)
	RO	Check RHR system – IN SHUTDOWN COOLING MODE RNO – GO TO STEP 3.e	(NO)
	BOP	Any non-faulted SG level - GREATER THAN 25% [40%]	(YES)
	BOP	Control feed flow to non-faulted SG(s) to stop RCS cooldown. Reduces feed flow if necessary	

Op Test No.:	NRC	Scenario #	3	Event #	7	Page	57 of 69
Event Description:	**Steam Space LOCA inside containment (continued)**						

Time	Position	Applicant's Actions or Behavior

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**.	
	SRO	Minimize RCS Cooldown From Faulted SG(s): Check any SG – FAULTED RNO – GO TO STEP 5	(NO)
	RO	Check PRZ PORV Block Valves: Verify 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) RNO – GO TO STEP 6.d	(NO)
	RO	Check PRZ pressure - LESS THAN 2335 PSIG Verify PRZ PORVs – SHUT	(YES) (YES)
	RO	Check SI Flow - GREATER THAN 200 GPM	(YES)
	SRO	Check SI Termination Criteria: (NOT MET) RNO – GO TO STEP 9	
	SRO	Check if a RCP should be started: (Criteria not met) RNO – 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.**	

Op Test No.:	NRC	Scenario #	3	Event #	7	Page	58	of 69
Event Description:	**Steam Space LOCA inside containment (continued)**							

Time	Position	Applicant's Actions or Behavior
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 – GREATER THAN 100°F IN ANY SIXTY MINUTE PERIOD (YES) 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.
EOP-ECA-1.1		LOSS OF EMERGENCY COOLANT RECIRCULATION
	SRO	Prior to giving crew directions from EOP-ECA-1.1, performs crew update.
	SRO	Foldout applies
Evaluator Note:		 **FOLDOUT** • **RESTORATION OF EMERGENCY COOLANT RECIRCULATION** **IF** emergency coolant recirculation capability is restored during this procedure, **THEN** RETURN TO procedure and step in effect. • **LOSS OF SUCTION** **IF** RWST level drops to 3% (Empty alarm/ALB-004-2-5), **THEN** secure all pumps taking suction only from the RWST. • **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.
	RO	Reset SI - Locates 2 MCB SI reset switches and turns to "reset" then verifies correct status light changes indicating SI is reset Reports SI is reset to the SRO

Op Test No.:	NRC	Scenario #	3	Event #	7		Page	59	of	69
Event Description:		**Steam Space LOCA inside containment (continued)**								

Time	Position	Applicant's Actions or Behavior	
	SRO	Manually Realign Safeguards Equipment Following A Loss Of Offsite Power. (Refer to EOP-E-0, "REACTOR TRIP OR SAFETY INJECTION", Attachment 6.)	
Procedure Note:		**Resetting the SI suction auto switchover signal also defeats the automatic open and shut signals to the CSIP alternate miniflow isolation valves.**	
	RO	Reset SI Suction Auto Switchover.	
	BOP	Verify CNMT Fan Coolers - ONE FAN PER UNIT RUNNING IN SLOW SPEED	
	RO	Check RWST Level - GREATER THAN 3% (Empty alarm)	(YES)
	SRO	Determine CNMT Spray Requirements:	
	RO	• Spray pump suction - ALIGNED TO RWST	(YES)

<table>
<tr><td colspan="3">Op Test No.: NRC Scenario # 3 Event # 7 Page 60 of 69</td></tr>
<tr><td colspan="3">Event Description: Steam Space LOCA inside containment (continued)</td></tr>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
</table>

<table>
<tr><td></td><td>SRO</td><td colspan="2">• Determine required number of CNMT spray pumps from Table:
<table>
<tr><th colspan="4">CONTAINMENT SPRAY REQUIREMENTS</th></tr>
<tr><th>RWST LEVEL</th><th>CONTAINMENT PRESSURE</th><th>TOTAL # OF FAN COOLER UNITS RUNNING</th><th>REQUIRED # OF CNMT SPRAY PUMPS</th></tr>
<tr><td rowspan="5">GREATER THAN 23.4%</td><td>GREATER THAN 45 PSIG</td><td>N/A</td><td>2</td></tr>
<tr><td rowspan="3">BETWEEN 10 PSIG AND 45 PSIG</td><td>0 OR 1</td><td>2</td></tr>
<tr><td>2 OR 3</td><td>1</td></tr>
<tr><td>4</td><td>0</td></tr>
<tr><td>LESS THAN 10 PSIG</td><td>N/A</td><td>0</td></tr>
<tr><td rowspan="4">BETWEEN 3% AND 23.4%</td><td>GREATER THAN 45 PSIG</td><td>N/A</td><td>2</td></tr>
<tr><td rowspan="2">BETWEEN 10 PSIG AND 45 PSIG</td><td>0, 1 OR 2</td><td>1</td></tr>
<tr><td>3 OR 4</td><td>0</td></tr>
<tr><td>LESS THAN 10 PSIG</td><td>N/A</td><td>0</td></tr>
<tr><td>LESS THAN 3%</td><td>N/A</td><td>N/A</td><td>0</td></tr>
</table></td></tr>
<tr><td></td><td>RO</td><td>• Verify spray pumps – REQUIRED NUMBER RUNNING
(No CT Pump running at this point CNMT Pressure has remained below 10 psig)</td><td>(YES)</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">Align CNMT Spray For Recirculation:</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>RO</td><td>• Any CNMT spray pump - RUNNING</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: GO TO Step 9.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">Add Makeup To RWST Using OP-107.01, "CVCS BORATION, DILUTION, AND CHEMISTRY CONTROL", Section 8.4.</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>SRO</td><td colspan="2">Check Intact SG Levels:</td></tr>
<tr><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>BOP</td><td>• Any level - GREATER THAN 25% [40%]
• Control feed flow to maintain all intact levels between 25% and 50% [40% and 50%].</td><td>(YES)</td></tr>
</table>

Op Test No.: NRC	Scenario # 3	Event # 7	Page 61 of 69
Event Description:	**Steam Space LOCA inside containment (continued)**		

Time	Position	Applicant's Actions or Behavior	
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.**	
	SRO	Check PRZ Pressure:	
	RO	• Pressure - LESS THAN 2000 PSIG • Block low steam pressure SI.	(YES)
Procedure Caution:		**The RCS cooldown should be performed as quickly as possible to minimize potential offsite releases.**	
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:	
	SRO	Maintain RCS cooldown rate less than 100°F/HR.	
Lead Evaluator:		**Terminate the scenario when the crew determines the RCS cooldown rate has exceeded 100°F/HR and the crew must wait before recommencing the RCS cooldown.** **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 8
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. Verify Two CSIPs - RUNNING

☐ 2. Verify Two RHR Pumps - RUNNING

☐ 3. Verify Two CCW Pumps - RUNNING

☐ 4. Verify All ESW **AND** ESW Booster Pumps - RUNNING

☐ 5. Verify SI Valves - PROPERLY ALIGNED

(Refer to Attachment 1.)

☐ 6. Verify 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 8
SAFEGUARDS ACTUATION VERIFICATION

☐ 7. Verify 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** Verify 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** Verify 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 8
SAFEGUARDS ACTUATION VERIFICATION

☐ 10. Verify Both Main FW Pumps - TRIPPED

☐ 11. Verify FW Isolation Valves - SHUT

(Refer to OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 6.)

☐ 12. Verify both MDAFW pumps - RUNNING

13. **IF** any of the following conditions exist, **THEN** verify 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. Verify AFW Valves - PROPERLY ALIGNED

- ☐ **IF** no AFW Isolation Signal, **THEN** verify 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** verify MDAFW and TDAFW isolation and flow control valves to affected SG - SHUT

☐ 15. Verify Both EDGs - RUNNING

☐ 16. Verify CNMT Fan Coolers - ONE FAN PER UNIT RUNNING IN SLOW SPEED

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 4 of 8
SAFEGUARDS ACTUATION VERIFICATION

☐ 17. Verify CNMT Ventilation Isolation Valves - SHUT

(Refer to OMM-004, "POST TRIP/SAFEGUARDS ACTUATION REVIEW", Attachment 7.)

☐ 18. Verify 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. Verify Essential Service Chilled Water System Operation:

☐ • Verify both WC-2 chillers - RUNNING

☐ • Verify both P-4 pumps - RUNNING

☐ (Refer to AOP-026, "LOSS OF ESSENTIAL SERVICE CHILLED WATER SYSTEM" for loss of any WC-2 chiller.)

20. Verify 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. Verify 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 8
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** Turn ON 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 8
SAFEGUARDS ACTUATION VERIFICATION

25. Start The Spent Fuel Pump Room Ventilation System:
    a. At AEP-1, verify 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 8
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 and 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)

REACTOR TRIP OR SAFETY INJECTION

Attachment 3
Sheet 8 of 8
SAFEGUARDS ACTUATION VERIFICATION

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 Co-ordinator - Control Room

☐ • Site Emergency Co-ordinator - Technical Support Center

(Refer to PEP-230, "CONTROL ROOM OPERATIONS".)

- END -

Facility:	Harris Nuclear Plant	Task No.:	301009H401
Task Title:	Initiate Emergency Boration Following a Reactor Trip (AOP-002)	JPM No.:	2018 HNP NRC Exam Simulator JPM CR a
K/A Reference:	APE024 AA1.17 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 ________

Initial Conditions:	• The unit was at 100% power when the 'A' MFW pump tripped • The crew performed a manual Reactor Trip in accordance with AOP-010, Feedwater Malfunctions • The crew completed the immediate actions of EOP-E-0, Reactor Trip or Safety Injection and have transitioned to EOP-ES-0.1, Reactor Trip Response • RCS temperature has been stabilized in accordance with EOP-ES-0.1 step 4

Initiating Cue:	• Your position is the OATC • You have the responsibility for the Foldout items in EOP-ES-0.1 • Continue EOP-ES-0.1 starting with step 5

DO NOT READ TO THE EXAMINEE:	**Allow the candidate to use the procedures from the Simulator for this JPM. You will need to have pre-made copies of EOP-ES-0.1 and AOP-002 ready for replacements after the JPM is complete.**

Task Standard:	Emergency Boration flow established from the RWST maintaining at least 90 gpm charging flow from the RWST to the RCS.
Required Materials:	BOP required to perform EOP-ES-0.1, Step 4
General References:	EOP-ES-0.1, REACTOR TRIP RESPONSE Rev. 3 AOP-002, EMERGENCY BORATION Rev. 24
Handout:	None – use simulator references
Time Critical Task:	No
Validation Time:	8 minutes

Critical Step Justification	
Step 5	The operator must establish the suction flow path to the CSIP from the RWST by opening one of the RWST suction valves (one or the other is critical since they are in a parallel path configuration)
Step 6	The operator must secure at least one of the suction valves from the VCT to the CSIP's to prevent gas intrusion and binding of the CSIPs during the required Emergency Boration of the RCS. (one or the other is critical since these valves are in a series configuration)
Step 7	A flow rate of > 90 gpm ensures that the boron concentration and required flow of the action statements of LCOs 3.1.1.1 and 3.1.1.2 are being met.

2018 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-165
- Password “NRC2018”
- Go To Run
- Adjust volume / range for Source Range audio counts to an in plant expected level to reduce distraction from source range audio counts

- (**IF NEEDED**) The 86 relays <u>should</u> roll when the simulator is placed in run. If not then run the APP file “Roll 86 Gen” or they can be manually overridden with override LO’s
    - XGAO018A GEN LOCKOUT G1A-TRIP COIL ON
    - XGBO017A GEN LOCKOUT G1B-TRIP RELAY ON

- **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.)

The following setup information is how this exam IC was developed.

- Reset to IC-19
- Go to run
- Insert a malfunction to prevent two control rods from inserting on the trip
    - CRF16a 220.0 4 (shutdown bank A Rod N-7)
    - CRF16b 220.0 27 (control bank A Rod F-14)

- Place the ‘B’ BAT pump under clearance for motor replacement
    - idi xa2i175 (n 00:00:00 00:00:00)STOP,AUTO
    - ilo xa2o175g (n 00:00:00 00:00:00)OFF
    - Hang CIT on CB Switch for ‘B’ BA Pump
    - Place MCB switch to STOP
    - Place protected equipment tag on ‘A’ BA pump

- Insert a failure of ‘A’ BAT pump to start
    - ior xa2i174 (n 0 0) ASIS

- Shut 1CS-8 (60 gpm letdown orifice)
- Reduce flow on FK-122.1 to < 30 gpm (somewhere close to 20 gpm)
- Place a trip of the ‘A’ MFW Pump on Trigger 1 (IMF CFW16A)
- Go to run, insert Trigger 1 then manually trip the Reactor
- Verify immediate action conditions are met
- Secure the TDAFW pump and adjust AFW flows to obtain SG levels > 25%
- Stabilize RCS temperature within the required range of ES-0.1, Step 4
- Acknowledge and reset annunciator alarms
- Freeze and snap to exam IC

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ____________

ES-0.1, Step 5

Performance Step: 1 Check Feed System Status:

- RCS temperature – less than 564˚F
- Verify Feed Reg valves – SHUT
- Check feed flow to SG's – GREATER THAN 200 KPH

Standard:

Verifies RCS temperature indication less than 564˚F **YES**

Verifies each Feed Reg Valve indicating SHUT **YES**

Verifies feed flow to SG's greater than 200 KPH **YES**

Comment:

ES-0.1, Step 6

Performance Step: 2 Check control rod status:

a. Check DRPI – available
b. Verify all control rods – fully inserted

Standard:

Determines DRPI available by indicating lights on AEP-1

Determines two rods stuck fully out

6.b RNO - IF two **OR** more control rods **NOT** fully inserted, **THEN** emergency borate. Refer to AOP-002, EMERGENCY BORATION.

Locates a copy of AOP-002

Evaluator Note:	**Applicant may go to AEP-1 to determine which rods are stuck.**

Comment:

Evaluator Note:	**Provide Candidate blue copy of AOP-002 once MCR copy is located at either the CRS desk or front of the SM desk**

AOP-002, Note prior to Step 1

Performance Step: 3 This procedure contains no immediate actions.

Standard: Reads and placekeeps note

Comment:

AOP-002, Note prior to Step 1

Performance Step: 4 VERIFY a Boric Acid (BA) Pump RUNNING.

Standard: Reads step 1 and starts one BA Pump

- ‘A’ BA pump fails to start – green light lit
- Informs CRS that no Boric Acid pumps are available
- (May dispatch Aux Operators to investigate breaker and locally at the pump)

Takes RNO path for step 1. GO TO Step 6

Evaluator Cue: Acknowledge any reports to CRS

Simulator Communicator: Acknowledge request to investigate ‘A’ BA pump problems

Comment:

AOP-002, Step 6.a **Alternate Path Begins**

√ **Performance Step: 5** ESTABLISH boration flow from RWST as follows:

a. **OPEN** the following valves:

- 1CS-291, Suction From RWST LCV-115B
- 1CS-292, Suction From RWST LCV-115D

Standard: Locates the MCB control switches for the following valves and takes switch to OPEN (Red light lit)

- 1CS-291, Suction From RWST LCV-115B ☐
- 1CS-292, Suction From RWST LCV-115D ☐

Comment: **Critical to open either 1CS-291 or 1CS-292 since the valves are in parallel.**

AOP-002, Step 6.b

√ **Performance Step: 6** **SHUT** the following valves:

- 1CS-165, VCT Outlet LCV-115C
- 1CS-166, VCT Outlet LCV-115E

Standard: Locates MCB switches then turns switch to SHUT (Green light lit)

- 1CS-165, VCT Outlet LCV-115C ☐
- 1CS-166, VCT Outlet LCV-115E ☐

Comment: **Critical to shut either 1CS-165 or 1CS-166 since the valves are in a series alignment.**

AOP-002, Step 6.c

√ **Performance Step: 7** VERIFY and MAINTAIN at least 90 gpm charging flow from the RWST to the RCS (FI-122A.1) until required boration is completed.

Standard: Verifies flow indicated on FI-122A.1 as < 90 gpm.

With FK-122.1 in manual candidate increases demand to increase flow to ≥ 90 gpm flow to CSIP suction on FI-121A.1

Comment:

Evaluator Cue:	**After the candidate has established and verified at least 90 gpm charging flow from the RWST to the RCS on FI-122A.1** **Announce: I have the shift, 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.: 2018 HNP NRC Exam Simulator JPM CR a

Initiate Emergency Boration Following a Reactor Trip
In Accordance With EOP-ES-0.1 and AOP-002

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 the ‘A’ MFW pump tripped • The crew performed a manual Reactor Trip in accordance with AOP-010, Feedwater Malfunctions • The crew completed the immediate actions of EOP-E-0, Reactor Trip or Safety Injection and have transitioned to EOP-ES-0.1, Reactor Trip Response • RCS temperature has been stabilized in accordance with EOP-ES-0.1 step 4

Initiating Cue:	• Your position is the OATC • You have the responsibility for the Foldout items in EOP-ES-0.1 • Continue EOP-ES-0.1 starting with step 5

Facility:	Harris Nuclear Plant	Task No.:	301142H601
Task Title:	Manually Load Safeguards Equipment On AC Emergency Buses After A LOSP	JPM No.:	2018 NRC Exam Simulator JPM b
K/A Reference:	006 A4.04 RO 3.7 SRO 3.6	**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:	• With the Unit operating at 100% power a Loss of Coolant Accident occurred • A Reactor Trip and Safety Injection have been actuated • Both EDG's have failed to start and a loss of offsite power occurred • The crew implemented EOP-ECA-0.0, Loss of All AC Power Subsequently: • Offsite power has been restored to both Emergency Buses • The CRS has transitioned to EOP-ECA-0.2, Loss of All AC Power Recovery With SI Required • EOP-ECA-0.2 steps 1-3 have been completed

Initiating Cue:	• You are the OATC • The CRS has directed you to continue with EOP-ECA-0.2 starting at step 4.

Task Standard:	Manually Load Safeguards Equipment On AC Emergency Buses After A LOSP.
Required Materials:	None
General References:	EOP-ECA-0.2 Rev. 1
Time Critical Task:	No
Validation Time:	15 minutes

Critical Step Justification	
Step 4	1CC-251, RCP Thermal Barrier CCW return outside CMNT isolation, fails to open requiring the candidate to shut 1CC-249, RCP Thermal Barrier CCW return inside CMNT isolation, from the MCB or 1CC-251 locally to prevent damage to the RCP that due to water hammer once CCW flow is restored.
Step 7	The operator must direct the AO to locally close the control power knife switch to restore the ability to control the CCW pump from the MCB.
Step 9	The operator must direct the AO to locally close the control power knife switch to restore the ability to control the CCW pump from the MCB.
Step 10	The operator must start the standby CCW pump to place the system in the ECCS alignment required for Safety Injection actuation to restore the ability to provide adequate core cooling.
Step 11	The operator must start the both RHR pumps to place the system in the ECCS alignment required for Safety Injection actuation to restore the ability to provide adequate core cooling.

ALTERNATE PATH JUSTIFICATION	
Step 4	1CC-251, RCP Thermal Barrier CCW return outside CMNT isolation, fails to open requiring the candidate to shut 1CC-249, RCP Thermal Barrier CCW return inside CMNT isolation, from the MCB or 1CC-251 locally to prevent damage to the RCP that due to water hammer once CCW flow is restored.

SIMULATOR SETUP

For the 2018 NRC Exam Simulator JPM 'b'

Simulator Operator - Exam Setup

- Reset to IC-166 password "NRC2018"
- Go to RUN
- Silence and Acknowledge annunciators
- (**IF NEEDED**) The 86 relays should roll when the simulator is placed in run. If not then run the APP file "Roll 86 Gen" or they can be manually overridden with override LO's
    - XGAO018A GEN LOCKOUT G1A-TRIP COIL ON
    - XGBO017A GEN LOCKOUT G1B-TRIP RELAY ON
- **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.)

The following setup information is how this exam IC was developed.

- Initialized to IC-19 (100% power condition)
- Insert a failure of the "A" EDG and "B" EDG to start <IMF DSG01 BOTH>. Then insert a loss of all AC <IMF EPS01 W/O DELAY>.
- Fail the ASI Pump to start <irf cvc195 (n 0 0) STOP>.
- Once the plant is stable, initiate an SI and wait 60 seconds. RESET the SI signal. Using Simulator MSC Drawing for ECA-0.0 Open the breakers for the sequencers, remove control power for both CCW pumps, both Containment Spray pumps and 1A and 1B AFW pump.
- Then insert RCS leakage to the extent that a safety injection is needed. <IMF RCS18A 6 0> Allow the Pressurizer to empty and pressure to drop to ≈ 1250 psig.
- Then delete the loss of all AC malfunction <DMF EPS01> and restore power to all buses. Use the EOP-ECA-0.0 Attachment 1 to restore power to 1A-SA and 1B-SB from off-site. Perform actions in EOP-ECA-0.0 after power restoration which includes EOP-ECA-0.0 step 36.c including setting SG PORV set points to 84%, transition to EOP-ECA-0.2, perform steps 1, 2 & 3.
- Insert a malfunction to prevent 1CC-251, RCP thermal barrier CCW return outside CNMT isolation valve from shutting from the MCB <irf ccw051 (n 0 0) engaged> <irf ccw052 (n 0 0) 100 0 100 <ior xa1073 (n 0 0) ASIS>
- Acknowledge and reset all alarms and place the simulator in FREEZE. Snap this to an IC

Simulator Operator: **When directed by the Lead Examiner go to Run.**

START TIME: ___________

EOP-ECA-0.2, Step 4.a

Performance Step: 1 Check RCP Thermal Barrier Status:
a. Check ASI pump - RUNNING

Standard: Identifies ALB-008-2-3, ASI System Trouble annunciator is in alarm and RCP Seal water injection flow indications are 0 gpm. Determines the ASI Pump is NOT running. Takes RNO action for step 4.a - GO TO Step 4.c

(May dispatch AO to investigate)

Comment:

Evaluator Cue:

IF an AO is dispatched to investigate the ASI pump – acknowledge request. (IF they continue with EOP-ECA-0.2 do not make a follow up report)

IF candidate stops and waits for report then wait 10-20 seconds and cue: The ASI pump has not started and the 24VDC Control Power Available light is NOT lit. You will contact I&C to investigate problem.

EOP-ECA-0.2, Step 4.c

Performance Step: 2 Check RCP Thermal Barrier Status:
c. Check CCW pumps – ALL STOPPED

Standard: Locates the control switches for CCW Pumps and determines all are DE-ENERGIZED (GREEN/RED lights NOT LIT)

.

Locates CCW Flow and Pressure Meters and determines all CCW Pumps are STOPPED (0 flow and 0 pressure)

Comment:

EOP-ECA-0.2, Step 4.d

Performance Step: 3

Check RCP Thermal Barrier Status:

d. Check RCP thermal barrier CCW return outside CNMT isolation – SHUT

- 1CC-251: A-252-FV34-W7-S10

Standard: Locates control switch for 1CC-251 and determines the valve is in the OPEN position

Comment:

EOP-ECA-0.2, Step 4.d RNO - ALTERNATE PATH

✓ **Performance Step: 4**

Check RCP Thermal Barrier Status:

d. Prior to starting a CCW pump, verify RCP thermal barrier CCW return isolated by **any** of the following:

- Shut outside CNMT isolation valve from the MCB OR locally:
  1CC-251

- Shut inside CNMT isolation valve from the MCB:
  1CC-249

Standard:

Locates control switch for 1CC-251 and takes switch to SHUT

Identifies 1CC-251 does NOT shut.

(May) contact Aux Operator to locally shut 1CC-251.

Locates control switch for 1CC-249 and takes switch to SHUT

Simulator Operator Cue:	**If contacted to shut 1CC-251 wait 20-30 seconds then report: 1CC-251 is mechanically stuck and the handwheel will not turn.**

Comment:

EOP-ECA-0.2, Caution prior to Step 5

Performance Step: 5

CAUTION:
The loads placed on the energized AC emergency bus should NOT exceed the capacity of the power source.

Restoration of DC control power to the first CCW pump will cause it to auto start on low pressure.

Standard:

Operator reads and placekeeps at any procedure note or caution

Comment:

EOP-ECA-0.2, Step 5.a

Performance Step: 6

Manually Load Safeguards Equipment On AC Emergency Buses
    a. Check CCW pumps – ALL STOPPED

Standard:

Locates the control switches for CCW Pumps and determines all are DE-ENERGIZED (GREEN/RED lights NOT LIT).

Locates CCW Flow and Pressure Meters and determines all CCW Pumps are STOPPED (0 flow and 0 pressure)

Comment:

Evaluator Note:	• **Candidate may choose to shut control power knife switch for CCW pump "A" or "B" in performance step 7 and remaining pump in performance step 9** • **Candidate may ask CRS which CCW pump to perform actions for first. IF so, then direct the actions for CCW Pump "A" first, then CCW Pump "B".**

EOP-ECA-0.2, Step 5.b

✓ **Performance Step: 7**

Manually Load Safeguards Equipment On AC Emergency Buses

b. Locally shut control power knife switch for breaker of CCW pump to be started:
  - **1A-SA CUB 8 (CCW pump A)**
  - 1A-SA CUB 3 (CCW pump C)
  - 1B-SB CUB 8 (CCW pump B)
  - 1B-SB CUB 2 (CCW pump C)

Standard:

Contacts Turbine Building AO and directs them to shut the desired control power knife switch
  - **1A-SA CUB 8 (CCW pump A)**
  - 1A-SA CUB 3 (CCW pump C)
  - 1B-SB CUB 8 (CCW pump B)
  - 1B-SB CUB 2 (CCW pump C)

Evaluator Note:	**When the control power knife switch is closed on the first CCW pump the pump will auto start due to low pressure in the CCW system.**

Simulator Communicator:	Acknowledge request to shut the desired control power knife switch

Simulator Operator:	Shut the desired control power knife switch and report to communicator that it is shut

Simulator Communicator:	Report that the desired control power knife switch is shut

Comment:

EOP-ECA-0.2, Step 5.c

Performance Step: 8

Manually Load Safeguards Equipment On AC Emergency Buses
c. Verify CCW pump – RUNNING

Standard:

Locates the control switches for CCW Pumps and determines the desired CCW Pump is RUNNING (RED light LIT) – other redundant checks should also be performed such as pressure increasing in the CCW system, pump flow, ect.

Comment:

EOP-ECA-0.2, Step 5.d

✓ **Performance Step: 9**

Manually Load Safeguards Equipment On AC Emergency Buses
d. Locally shut control power knife switch for breaker of standby CCW pump.

- 1A-SA CUB 8 (CCW pump A)
- 1A-SA CUB 3 (CCW pump C)
- **1B-SB CUB 8 (CCW pump B)**
- 1B-SB CUB 2 (CCW pump C)

Standard:

Contacts Turbine Building AO and directs them to shut the desired control power knife switch

- 1A-SA CUB 8 (CCW pump A)
- 1A-SA CUB 3 (CCW pump C)
- **1B-SB CUB 8 (CCW pump B)**
- 1B-SB CUB 2 (CCW pump C)

Simulator Communicator:	Acknowledge request to shut the desired control power knife switch

Simulator Operator:	Shut the desired control power knife switch and report to communicator that it is shut

Simulator Communicator:	Report that the desired control power knife switch is shut

Comment:

EOP-ECA-0.2, Step 5.e

✓ **Performance Step: 10** Start standby CCW pump.

Standard: Locates control switches for the standby CCW Pump and takes switch to START

Comment:

EOP-ECA-0.2, Step 5.f

✓ **Performance Step: 11** Start both RHR pumps.

Standard: Locates control switch for RHR Pump 1A-SA and takes switch to START

Locates control switch for RHR Pump 1B-SB and takes switch to START

Evaluator Cue: **After candidate has started two CCW Pumps and BOTH RHR pumps are in service. Evaluation on this JPM is complete.**

Announce: I have the shift, the remainder of Step 5 will be completed by the BOP. End of JPM

Inform Simulator Operator to place the Simulator in Freeze.

Comment:

STOP TIME: ______________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Simulator JPM b

Manually Load Safeguards Equipment On AC Emergency Buses After A LOSP

In Accordance With EOP-ECA-0.2, Loss of All AC Power Recovery With SI Required

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:	• With the Unit operating at 100% power a Loss of Coolant Accident occurred • A Reactor Trip and Safety Injection have been actuated • Both EDG's have failed to start and a loss of offsite power occurred • The crew implemented EOP-ECA-0.0, Loss of All AC Power Subsequently: • Offsite power has been restored to both Emergency Buses • The CRS has transitioned to EOP-ECA-0.2, Loss of All AC Power Recovery With SI Required • EOP-ECA-0.2 steps 1-3 have been completed

Initiating Cue:	• You are the OATC • The CRS has directed you to continue with EOP-ECA-0.2 starting at step 4.

Facility:	Harris Nuclear Plant	Task No.:	301135H601
Task Title:	Take Corrective Action For Failure of CSIP Mini-Flow Valves to Re-Position	JPM No.:	2018 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 error resulted in a Reactor Trip / 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.

Task Standard:	Obtain adequate flow through a running CSIP.
Required Materials:	E-0, Reactor Trip or Safety Injection, Rev. 7
General References:	E-0, Reactor Trip or Safety Injection, Rev. 7
Time Critical Task:	No
Validation Time:	10 minutes

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.
Step 14	Establishing a flow rate of >60 gpm is required by procedure.

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.

2018 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-167
- Password "NRC2018"
- 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:
Circle / Slashes note to indicate it is read and understood.

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 Verify 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).** **Greater than 60 GPM flow can be expected with FK-122.1 set to approximately 30% demand**

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/verifies 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.: 2018 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 error resulted in a Reactor Trip / 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.:	301170H601
Task Title:	Initiate RCS Bleed and Feed (EOP-FR-H.1)	JPM No.:	2018 NRC Exam Simulator JPM CR d
K/A Reference:	EPE E05 EA1.1 RO 4.1 SRO 4.0	**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 operating at 100% power • Motor Driven AFW Pump ‘A’ is under clearance and partially disassembled for maintenance Subsequently: • The Reactor tripped due to a loss of off-site power • ‘B’ EDG tripped when it started, the cause is being investigated • The Turbine-Driven AFW Pump failed while starting • A SBLOCA occurred following the Reactor Trip • Adverse Containment values are in effect • The crew has transitioned from EOP-E-0 to EOP-FR-H.1, Response To Loss Of Secondary Heat Sink • No source of Feedwater is available • The Foldout criteria for initiation of RCS Bleed and Feed have just been met

Initiating Cue:	You are the OATC. The CRS directs you to observe the procedure CAUTION prior to EOP-FR-H.1, Step 16, then initiate RCS bleed and feed.

Task Standard: RCS feed established with maximum available bleed path.

Required Materials: Attach EOP-E-0, Attachment 1 to this JPM for use by the evaluator.

General References: EOP-FR-H.1, Response To Loss Of Secondary Heat Sink, Rev. 3

Handout: Use Simulator copy of EOP-FR-H.1 and ensure it is replaced after each use or provide a paper copy.

Time Critical Task: No

Validation Time: 10 minutes

Critical Step Justification	
Step 7	Must reset Phase A to gain control of components allowing operation of valves that were automatically repositioned from Phase A actuation.
Step 16	Critical to establish the maximum available bleed path to ensure adequate core heat removal during accident conditions.

2018 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-168
- Password “NRC2018”
- Hang CIT on the ‘A’ MDAFW Pump and place Star Placards in accordance with OMM-001 on required equipment
- It may be necessary to roll the 86 relays between or at the start of the set up. Run AMS file “Roll Gen 86 Relays” to get the 86 relays to the trip condition.
- Go to Run
- Turn off horn on RM-11 – turn back on at conclusion of JPMs
- 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, stay in freeze
- Remove power from the breaker on ‘A’ MDAFW Pump
  - irf CFW113 cp_off
- Insert a malfunction to prevent the ‘B’ EDG from starting
  - imf DSG01 B
- Insert a malfunction to trip TDAFW Pump during AUTO start
  - imf CFW01 C
- Insert overrides to block MANUAL OPEN on PCV-445A, PCV-445B and PCV-444B
  - idi xa2i136 (n o o) ASIS
  - idi xa2i137 (n o o) ASIS
  - idi xa2i138 (n o o) ASIS
- Insert a loss of Off-site Power
  - imf eps01 w/o delay
- Go to run
- Perform EOP-E-0 with a transition to EOP-FR-H.1, go to freeze
- Insert a SBLOCA after entering EOP-FR-H.1 to get to adverse containment values.
  - imf RCS18A 1.5
- Go to run
  - Modify SB LOCA RCS18A from the initial value of 1.5 to 10 until Containment pressure reaches 3.0 psig then reduce the leak size back to 1.5
- Perform EOP-FR-H.1 without establishing any source of feed flow
- Run AMS file AIR\AC’S to local
- Allow SG levels to reach feed and bleed Foldout (ADVERSE) criteria
  ALL SG WR Levels should be >15% but <30%
- 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: ____________

EOP-FR-H.1, CAUTION prior to Step 16

Performance Step: 1 Perform Steps 16 through 26 without delay to establish RCS heat removal by RCS bleed and feed.

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

EOP-FR-H.1, Step 16

Performance Step: 2 Actuate Safety Injection.

Standard: (Already actuated) Verifies SI actuated or may actuate manual MCB switch.

Comment:

EOP-FR-H.1, Step 17

Performance Step: 3 Verify RCS Feed Path:

a. SI flow - > 200 GPM
b. Observe NOTE prior to Step 19 AND GO TO Step 19

Standard: Verifies SI flow indication > 200 GPM (YES)

Comment:

EOP-FR-H.1, Note Prior to Step 19

Performance Step: 4 SI reset can **NOT** occur until sixty seconds after SI signal actuation.

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

EOP-FR-H.1, Step 19

Performance Step: 5 Reset SI

Standard: Places both SI Train RESET Switches in RESET and releases.
Verifies RESET on Bypass Permissive Panel.

Comment:

- **Bypass Permissive Panel light 4-1 SI Activated – OFF**
- **Bypass Permissive Panel light 5-1 SI Reset Auto SI Blocked - ON**

EOP-FR-H.1, Step 20

Performance Step: 6 Manually Realign Safe-Guards Equipment Following A Loss Of Off-Site Power.
(Refer To EOP-E-0, Reactor Trip Or Safety Injection, Attachment 6)

Standard: Reads step but at this time there is no need to follow through with actions.

Comment:

EOP-FR-H.1, Step 21

√ **Performance Step: 7** Reset Phase A Isolation Signals.

Standard:

- Places Train "A" and Train "B" Phase "A" RESET Switches in RESET and releases.

Comment:

EOP-FR-H.1, Step 21

Performance Step: 8 Reset Phase B Isolation Signals.

Standard:

- Phase "B" has not actuated (may reset since procedure directs)
- Places Train "A" and Train "B" Phase "B" RESET Switches in RESET and releases.

Comment:

EOP-FR-H.1, Step 22

Performance Step: 9 Check Sequencers - RESET (BOTH TRAINS)

Standard:

- Identifies that Train A Sequencer is NOT reset. (NO)

- (No actions required for Train B as it cannot be energized)

Comment:

EOP-FR-H.1, Note Prior to Step 22.a RNO

Performance Step: 10 Manual actuation of Load Block 9 cannot occur for 150 SECONDS after sequencer operation.

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

EOP-FR-H.1, Step 22.a RNO

Performance Step: 11 For any Sequencer that is **NOT** reset, Perform the following:

a. Check Sequencer Load Block 9 (Manual Loading Permissive) – ACTUATED

Standard:

- Identifies that Train A Sequencer has reached Load Block 9. (No actions required for Train B as it cannot be energized)

Comment:

Evaluator Note:	**Bus 1B1 cannot be energized**

EOP-FR-H.1, Step 23

Performance Step: 12 Energize AC Buses 1A1 AND 1B1

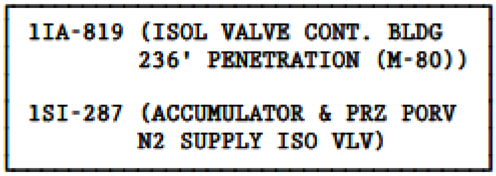
Standard:

- Energizes Bus 1A1 by closing the cross-tie "Emergency Bus A-SA to XFMR A1-SA Breaker A1 A-SA".

Comment:

EOP-FR-H.1, Step 24

Performance Step: 13 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)

Standard:

- Locates MCB switch for 1IA-819 and takes switch to OPEN
- Locates MCB switch for 1SI-287 and takes switch to OPEN

Comment:

Evaluator Note:	**All PORVs are blocked from opening (failed closed)**

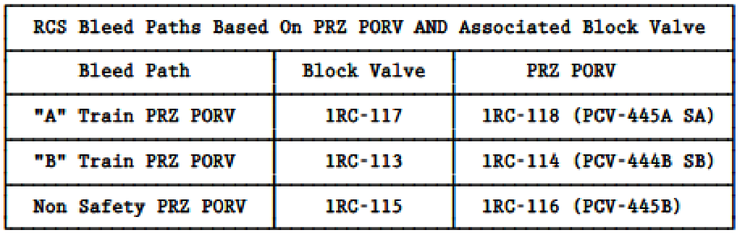
EOP-FR-H.1, Step 25

Performance Step: 14 Establish RCS Bleed Path:

a. Establish all RCS bleed paths listed in table by performing the following:

1) Verify PRZ PORV Block valves – ALL OPEN

2) Open all PRZ PORVs (safety and non-safety regardless of operability status).



RCS Bleed Paths Based On PRZ PORV AND Associated Block Valve		
Bleed Path	Block Valve	PRZ PORV
"A" Train PRZ PORV	1RC-117	1RC-118 (PCV-445A SA)
"B" Train PRZ PORV	1RC-113	1RC-114 (PCV-444B SB)
Non Safety PRZ PORV	1RC-115	1RC-116 (PCV-445B)

Standard:

- Verifies block valves RC-117, and RC-115 indicate OPEN (RED light) (RC-113 does not have power but was open)
- Attempts to open PCV-445A, PCV-445B, and PCV-444B
  - NONE WILL OPEN (ALL Green Lights)
- Informs CRS that none of the PZR PORV's will open and no RCS bleed paths through PORV's can be established.

Evaluator Cue:	**CRS acknowledges PORV's cannot be opened, continue with EOP-FR-H.1.**

Comment:

EOP-FR-H.1, Step 26 – ALTERNATE PATH

Performance Step: 15

Verify adequate RCS bleed path –

a. Check PRZ PORVs AND associated block valves – AT LEAST ONE BLEED PATH OPEN **(NO)**

Standard:

- Determines no PRZ PORVs are OPEN.
  - Proceeds to 26.a RNO
  - GO TO Step 26.c

Evaluator Note:	**There is no power available to 1RC-901, 1RC-903, 1RC-905.**

EOP-FR-H.1, Step 26.c

√ **Performance Step: 16**

Open all RCS vent valves to commence venting:

- 1RC-900
- 1RC-901
- 1RC-902
- 1RC-903
- 1RC-904
- 1RC-905

Standard:

(To operate each valve – must take control switch out of Pull To Lock then go to OPEN on switch)

Opens:

- 1RC-900 ___
- 1RC-902 ___
- 1RC-904 ___

Comment:

Lead Evaluator Cue:	**After RCS Vent Valves with power available are OPENED:** **Evaluation on this JPM is complete.** **Announce: I have the shift. END OF JPM** **Inform the Simulator Operator to place the Simulator in Freeze.**

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

STOP TIME: ________________

√ - Denotes Critical Steps

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Simulator JPM CR d

Initiate RCS Bleed and Feed
EOP-FR-H.1, Response To Loss Of Secondary Heat Sink

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 was operating at 100% power • Motor Driven AFW Pump 'A' is under clearance and partially disassembled for maintenance Subsequently: • The Reactor tripped due to a loss of off-site power • 'B' EDG tripped when it started, the cause is being investigated • The Turbine-Driven AFW Pump failed while starting • A SBLOCA occurred following the Reactor Trip • Adverse Containment values are in effect • The crew has transitioned from EOP-E-0 to EOP-FR-H.1, Response To Loss Of Secondary Heat Sink • No source of Feedwater is available • The Foldout criteria for initiation of RCS Bleed and Feed have just been met

Initiating Cue:	• You are the OATC. • The CRS directs you to observe the procedure CAUTION prior to EOP-FR-H.1, Step 16, then initiate RCS bleed and feed.

Facility:	Harris Nuclear Plant	Task No.:	088017H101
Task Title:	Perform Containment Ventilation Isolation Valve ISI Test (OST-1056)	JPM No.:	2018 NRC Exam Simulator JPM CR e
K/A Reference:	103 A4.01 RO 4.0 SRO 4.0	**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 plant is operating at 100 percent power • OST-1056 is being performed to test the operability of the Containment ventilation isolation valves per the ISI program • Airborne Radioactive Removal & Normal Purge Systems were shutdown in accordance with OP-168, Containment Ventilation And Vacuum Relief

Initiating Cue:	• The CRS has directed you to perform Section 7.2 of OST-1056. • **During the performance of this OST, the additional component timing will be performed by the Evaluator.** • **All IV's will only confirm observation of your actions.**

Evaluator NOTE:	The candidates should be briefed outside of the Simulator prior to performing this JPM. Provide them with a copy of OST-1056 and inform them that prerequisites are met. This will allow them to review the Precautions and Limitations associated with OST-1056 and have time for a task preview of the steps. Expect that the candidates will take about 10 - 15 minutes to complete this review.

Task Standard: Critical tasks of OST-1056, Containment Ventilation Isolation Valve ISI Test Quarterly Interval Modes 1 – 6, Section 7.2 completed.

Required Materials: **Calibrated Stopwatch**

General References: OST-1056, Containment Ventilation Isolation Valve ISI Test Quarterly Interval Modes 1 – 6, Revision 14

Time Critical Task: No

Handout: OST-1056 marked up with Prerequisites completed and CRS permission signature. Initials for ensuring the Airborne Radioactive Removal and Normal Purge Systems are shutdown per OP-168 is also initialed.

Validation Time: 25 minutes

Critical Step Justification	
Step 8	For testing single switch valves 1CP-5 and 1CP-9, 1CP-3 and 1CP-6 in the SHUT direction and for testing of 1CB-2 and CB-D51SA and 1CB-6 and CB52SB in both Open and Shut directions. Required to document the valve stroke time is within the time required by Technical Specifications for operability. **Testing of 1CB-2 and 1CB-6 OPEN** When 1CB-2 and 1CB-6 are stroked open, if these valves are allowed to shut prior to obtaining required data, then reopening could result in pre-conditioning and invalidation of the results.
Step 9	For testing single switch valves 1CP-5 and 1CP-9, 1CP-3 and 1CP-6 in the SHUT direction and for testing of 1CB-2 and CB-D51SA and 1CB-6 and CB52SB in both Open and Shut directions. Required to document the valve stroke time is within the time required by Technical Specifications for operability.
Step 10	Required for all tested valves to document the valve stroke time is within the time required by Technical Specifications for operability.
Step 11	Required for all tested valves to document the valve stroke time is within the time required by Technical Specifications for operability.

2018 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-169
- Password "NRC2018"
- 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

- Perform OP-168 section 7.1 to secure both ARR Fans S-1A and B

- Perform OP-168 section 7.1 to secure both Normal Purge fans AH-82A and B

- Momentarily open vacuum relief valves 1CB-2 & 1CB-6 to get Containment Pressure equal to 0" water on PDI-7680A & B

- Silence Acknowledge and Reset Annunciators

- Freeze and Snap these conditions to your exam IC

Evaluator:	***When the student is ready to assume the shift direct the Simulator Operator to place the Simulator in Run.***

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ___________

OST-1056 Section 7.2 step 1

Performance Step: 1 **Ensure** the Airborne Radioactive Removal and Normal Purge Systems are shutdown per OP-168.

Standard: Initials step 1 (part of initial conditions)

Comment: **IF the candidate goes to look for OP-168 remind them of the initial conditions stating that it has been completed.**

OST-1056 Section 7.2 step 2

Performance Step: 2 Referring to Attachment 2, test all valves listed per the following instructions:

Standard: Reviews Attachment 2 and identifies the valves to be tested in accordance with step 7.2.2

Comment:

OST-1056 Attachment 2

Performance Step: 3 All spaces next to valve number shall be filled in with initials, data or N/A as applicable.

Standard: Operator reads and placekeeps information

Comment:

Evaluator Note:	The following JPM steps identify the actions required to stroke time each valve which is performed in pairs, and based on the Attachment 2 stroke timing of the valve pairs may not be required in both the OPEN and the SHUT directions. The evaluator times provided as cues will be the same time that the operator gets while stroking his/her valve for the following valves, 1CP-9, 1CP-6, CB-051-SA-1, and CB-D52-SB-1.

OST-1056 NOTE prior to step 7.2.2.a

Performance Step: 4

- When multiple components are controlled by a single control switch, the timing of all the components should be performed concurrently using a different individual to time each component. This eliminates the need to perform multiple stroke time tests which could result in pre-conditioning of the components and invalidation of the stroke times.
- If an Air-Operated Valve must be operated to place it in its pre-test position, allow at least one minute for pressure in the actuator to stabilize before performing stroke timing.
- The following components are operated from the same switch:
    - 1CP-5 and 1CP-9
    - 1CP-3 and 1CP-6
    - 1CB-2 and CB-D51SA
    - 1CB-6 and CB-D52SB

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

OST-1056 Step 7.2.2.a

Performance Step: 5 Ensure the valve to be tested is aligned to the pretest position.

1CP-5: 1CP-3: 1CB-2: 1CB-6:
1CP-9: 1CP-6: CB-D51SA-1: CB-D52SB-1:

Standard: Locates the following valves

1CP-5: 1CP-3: 1CB-2: 1CB-6:
1CP-9: 1CP-6: CB-D51SA-1: CB-D52SB-1:

and ensures all valves indicate they are in the correct Pretest Position – SHUT

Comment: **NOTE: this check is completed for each group of valves prior to testing the single switch paired valves**

OST-1056 Step 7.2.2.b

Performance Step: 6 INITIAL for Pretest Position on Attachment 2.

Standard: Refers to Attachment 2 and initials the pretest position for

1CP-5: 1CP-3: 1CB-2: 1CB-6:
1CP-9: 1CP-6: CB-D51SA-1: CB-D52SB-1:
as SHUT

Comment:

OST-1056 Notes prior to step 7.2.2.c

Performance Step: 7

- 1CB-2 and 1CB-6 must be timed in both directions.
- When 1CB-2 and 1CB-6 are stroked open, they will automatically stroke closed when switch is released if Containment D/P is less negative than -0.25 INWC. The control switch may be held in OPEN while recording data until personnel are ready to perform stroke CLOSE testing. If these valves are allowed to SHUT prior to obtaining required data, reopening could result in pre-conditioning and invalidation of the results.

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

Evaluator Note:	Since the valves can be stroke timed in any order the procedure is written to have all of the valves listed in each step. The person performing the test chooses which valves to test if they test per the order listed in Attachment 2 they should test 1CP-5 and 9, then 1CP-3 and 6, then 1CB-2 and CB-D51SA-1 and then 1CB-6 and CB-D52SB-1. This JPM is written to this order.

OST-1056 Step 7.2.2.c

√ **Performance Step: 8**

SIMULTANEOUSLY PERFORM the following:

- PLACE the control switch for the valve being tested to the position opposite the pretest position.

1CP-5: 1CP-3: 1CB-2: 1CB-6:
1CP-9: 1CP-6: CB-D51SA-1: CB-D52SB-1:

- IF timing the valve in this direction, THEN START the stopwatch.

Standard: Locates the control switch 1CP-5 and 1CP-9 and places the switch in the OPEN position (not timed).

Comment:

Evaluator Note:	**When testing 1CB-2 OR 1CB-6 in the OPEN position, Operator should wait 1 minute prior to shutting 1 CB-2 OR 1CB-6** **√ Critical when testing 1 CB-2 or 1CB-6**

OST-1056 Step 7.2.2.d

√ **Performance Step: 9**

IF timing the valve in this direction, THEN:

- WHEN the valve has completed its travel as indicated by a singular position indicating light for the demanded position (no dual indication), THEN stop the stopwatch

Standard:

N/A - not timing the valve in the OPEN direction for

1CP-5 &1CP-9 OR 1CP-3 & 1CP-6

Comment:

OST-1056 Step 7.2.2.e

√ **Performance Step: 10**

SIMULTANEOUSLY PERFORM the following:

- PLACE the control switch for the valve being tested to the pretest position shown on Attachment 2.
- START the stopwatch.

Standard:

Locates the control switch for 1CP-5 and 1CP-9 switch in the SHUT position while operating the stopwatch.

Comment:

OST-1056 Step 7.2.2.f

√ **Performance Step: 11** WHEN the valve has completed its travel as indicated by a singular position indicating light for the demanded position (no dual indication), THEN stop the stopwatch.

Standard: When 1CP-5 and 1CP-9 have completed travel as indicated by a singular position indicating light for the demanded position (no dual indication).

Stops operation of the stopwatch.

Comment:

OST-1056 Step 7.2.2.g

Performance Step: 12 RECORD valve stroke time on Attachment 2.

Standard: Documents the appropriate “stroke time (sec)” space in the Full Stroke Test column of Attachment 2.

Comment:

Evaluator Cue:	**When asked for the stoke time of 1CP-9 report that :** **I have the same stroke time you have.**

OST-1056 Step 7.2.2.h

Performance Step: 13 INITIAL the fail-safe column on Attachment 2.

Standard: Initials the “Position Verified” space of the Fail Safe Test column on Attachment 2.

Comment:

OST-1056 Step 7.2.2.i

Performance Step: 14 INITIAL for the full stroke test on Attachment 2 as verification of satisfactory valve operation (as previously performed per Step 2.a through 2.f above).

Standard: Initials the “Verification of Travel by” space in the Full Stroke Test column of Attachment 2.

Comment:

OST-1056 Step 7.2.2.j

Performance Step: 15 Ensure the valve is in the post-test position per Attachment 2.

Standard: Refers to Attachment 2 and determines the pair of valves to be tested are in the required SHUT position for posttest position

Comment:

OST-1056 Step 7.2.2.k

Performance Step: 16 INITIAL for Posttest Position on Attachment 2.

Standard: Refers to Attachment 2 and initials the pair of valves to be tested are in the required SHUT position for posttest position

Comment:

OST-1056 Step 7.2.2.l

Performance Step: 17 **REPEAT** Steps 2.a through 2.k above for all remaining valves to be tested per Attachment 2.

Standard: Refers to Attachment 2 and repeats steps 2.a through 2.k for the remaining valves

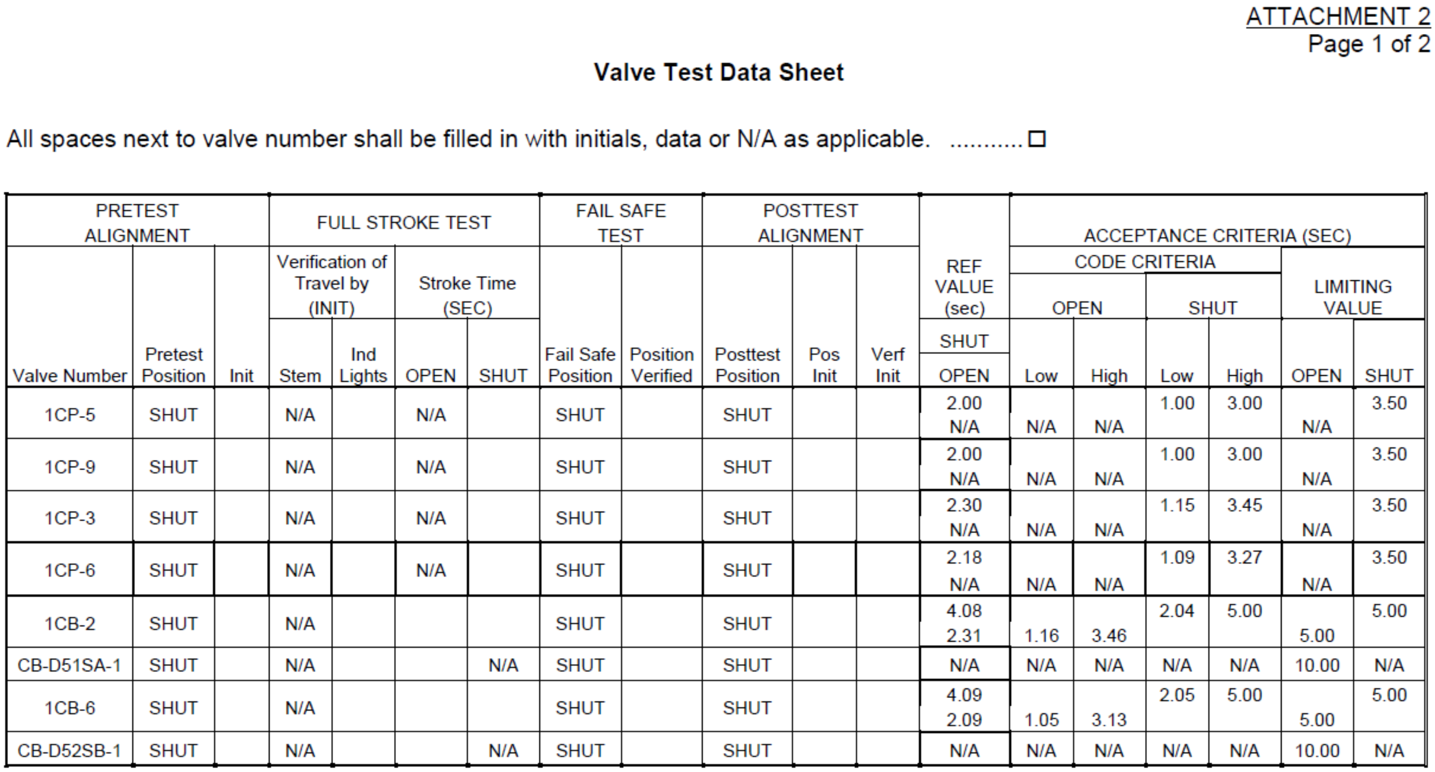
- 1CP-3 and 1CP-6
- 1CB-2 and CB-D51SA
- 1CB-6 and CB-D52SB

Comment:

Evaluator Cue:	**Candidate may request an IV. Per initiating cue: During the performance of this OST, an IV will only confirm observation of your actions.**

Evaluator Note:	**The students' progress can be followed using Attachment 2 Valve Test Data Sheet (next page).** **After each pair of valves are stroke tested when asked for the timing of the second valve provide the CUE:** **I have the same stroke time you do for the valve that I am timing.** **When testing 1CB-2 and 1CB-6 these valves will be stroke timed OPEN and SHUT. If Containment pressure is less negative than -0.25 INWC the student will NOT have to hold the control switch in OPEN while recording data.**

Evaluator Note:	**The JPM is written to test the valves from top to bottom of Attachment 2. This order is not required.** **Use the table to follow the students' progress in the JPM.**



ATTACHMENT 2
Page 1 of 2

Valve Test Data Sheet

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			REF VALUE (sec)	ACCEPTANCE CRITERIA (SEC)					
			Verification of Travel by (INIT)		Stroke Time (SEC)								CODE CRITERIA				LIMITING VALUE	
												SHUT	OPEN		SHUT			
Valve Number	Pretest Position	Init	Stem	Ind Lights	OPEN	SHUT	Fail Safe Position	Position Verified	Posttest Position	Pos Init	Verf Init	OPEN	Low	High	Low	High	OPEN	SHUT
1CP-5	SHUT		N/A		N/A		SHUT		SHUT			2.00 N/A	N/A	N/A	1.00	3.00	N/A	3.50
1CP-9	SHUT		N/A		N/A		SHUT		SHUT			2.00 N/A	N/A	N/A	1.00	3.00	N/A	3.50
1CP-3	SHUT		N/A		N/A		SHUT		SHUT			2.30 N/A	N/A	N/A	1.15	3.45	N/A	3.50
1CP-6	SHUT		N/A		N/A		SHUT		SHUT			2.18 N/A	N/A	N/A	1.09	3.27	N/A	3.50
1CB-2	SHUT		N/A				SHUT		SHUT			4.08 2.31	1.16	3.46	2.04	5.00	5.00	5.00
CB-D51SA-1	SHUT		N/A			N/A	SHUT		SHUT			N/A	N/A	N/A	N/A	N/A	10.00	N/A
1CB-6	SHUT		N/A				SHUT		SHUT			4.09 2.09	1.05	3.13	2.05	5.00	5.00	5.00
CB-D52SB-1	SHUT		N/A			N/A	SHUT		SHUT			N/A	N/A	N/A	N/A	N/A	10.00	N/A

OST-1056 Step 7.2.3

Performance Step: 18 REVIEW all data taken on Attachment 2, ensuring all stroke times are within the stated acceptance criteria.

Standard: Ensures all data is entered on Attachment 2.

Comment:

Evaluator Cue:	**After data entry is completed:** **This JPM is complete.** **END OF JPM 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.: 2018 NRC Exam Simulator JPM CR e

Containment Ventilation Isolation Valve ISI Test Quarterly Interval Modes 1 - 6

OST-1056, Containment Ventilation Isolation Valve ISI Test Quarterly Interval Modes 1 – 6

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 percent power • OST-1056 is being performed to test the operability of the Containment ventilation isolation valves per the ISI program • Airborne Radioactive Removal & Normal Purge Systems were shutdown in accordance with OP-168, Containment Ventilation And Vacuum Relief

Initiating Cue:	• The CRS has directed you to the perform Section 7.2 of OST-1056 • **During the performance of this OST, the additional component timing will be performed by the Evaluator.** • **All IV's will only confirm observation of your actions.**

Facility:	Harris Nuclear Plant	Task No.:	301194H601
Task Title:	Restoration of Offsite Power to Emergency Buses (EOP ECA-0.0)	JPM No.:	2018 HNP NRC Exam Simulator JPM CR f
K/A Reference:	055 EA1.07 RO 4.3 SRO 4.5	**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 operating at 100% power • 'A' EDG is under clearance due to a failure that caused the Generator field to not flash during OST-1013 Subsequently: • A failure of a transmission line on the Duke grid resulted in the cascading trip of several units which resulted in low grid frequency • The HNP unit has experienced a loss of offsite power • 'B' EDG failed to start. The problem is being investigated • The crew entered ECA-0.0, Loss Of All AC Power • The load dispatcher has contacted HNP and informed the MCR that the grid is now stable

Initiating Cue:	• Your position is the BOP • The CRS has directed you to restore offsite power to a (one) AC emergency bus using ECA-0.0 Attachment 1. • The Load Dispatcher has given permission to restore offsite power to 6.9 KV buses and to reset any tripped Start Up XFMR lockout relays.

Evaluator Note:	**Prior to starting this JPM position a second board operator at the OATC desk. The second board operator will be silencing annunciators not related to the initiating event. Provide the candidate a copy of ECA-0.0 Attachment 1.**

Task Standard: Bus 1B-SB energized from the SUT

Required Materials: None

General References: EOP-ECA-0.0, Attachment 1, Rev. 7

Time Critical Task: NO

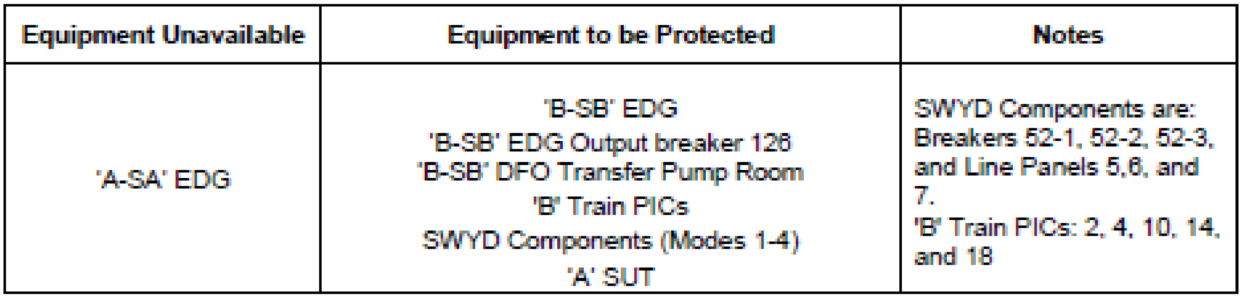
Validation Time: 15 Minutes

CRITICAL STEP JUSTIFICATION	
Step 15	Critical to place synchronizer control switch to proper position to allow closing breaker in next step.
Step 16	Critical to close Start Up XFMR B To Aux Bus E Breaker 121, without the breaker being closed power cannot be restored to Emergency Bus B-SB.
Step 18	Critical to close breaker 124 for Aux Bus E To Emergency Bus B‑SB, without the breaker being closed power cannot be restored to Emergency Bus B-SB.
Step 20	Critical to place synchronizer control switch to proper position to allow closing breaker in next step.
Step 21	Critical to close tie breaker 125 for Emergency Bus B‑SB To Aux Bus E, without the breaker being closed power cannot be restored to Emergency Bus B-SB.
Step 23	Critical to close Emergency Bus B‑SB To XFMR B1‑SB Breaker B1 A‑SB and Emergency Bus B‑SB To XFMR B3‑SB Breaker B3 A‑SB to supply power to safeguards emergency equipment.
Step 24	Critical to close 6.9 KV Emergency Bus B‑SB To XFMR B2‑SB Breaker B2 A‑SB to supply power to safeguards emergency equipment.

2018 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-170
- Password “NRC2018”
- Hang clearance tags on 1A-EDG
- Protect Equipment IAW OMM-001
    - Protected Train Equipment Tags on:
        - B-SB EDG Start Switch
        - B-SB Fuel Oil Transfer Pump Switch
        - Breaker 52-1, Breaker 52-2 and Breaker 52-3



Equipment Unavailable	Equipment to be Protected	Notes
'A-SA' EDG	'B-SB' EDG 'B-SB' EDG Output breaker 126 'B-SB' DFO Transfer Pump Room 'B' Train PICs SWYD Components (Modes 1-4) 'A' SUT	SWYD Components are: Breakers 52-1, 52-2, 52-3, and Line Panels 5,6, and 7. 'B' Train PICs: 2, 4, 10, 14, and 18

- (**IF NEEDED**) The 86 relays should roll when the simulator is placed in run. If not then run the APP file “Roll 86 Gen” or they can be manually overridden with override LO’s
    - XGAO018A GEN LOCKOUT G1A-TRIP COIL ON
    - XGBO017A GEN LOCKOUT G1B-TRIP RELAY ON
- 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.)

The following setup information is how this exam IC was developed

- Reset to IC-19
- Place 1A-EDG under clearance
    - IRF DSG005 (n 0 0) LOCAL
    - IRF DSG006 (n 0 0) MAINTAIN
- Fail Emergency Bus A-SA to Aux Bus D Tie Breaker 105 SA **ASIS** (this will not allow the breaker to be manually closed from the MCB switch)
    - IOR XD1I066 (n 0 0) ASIS
- Fail Emergency Bus B-SB to Aux Bus E Tie Breaker 125 SB **ASIS** (this will not allow the breaker to be manually closed from the MCB switch)
    - IOR XD1I075 (n 0 0) ASIS
- Fail 1B-SB EDG to start
    - IMF DSG01 (n 0 0) B
- Loss of Offsite Power (trigger 1)
    - IMF EPS01 (1 0 0) W/O_DELAY

JPM IC development – continued

- Since Attachment 1 allows the operator to choose energizing either bus ‘A’ or ‘B’, malfunctions were developed to fail breakers 105 and 125 ASIS. The JPM is written to have ONLY one of the buses energize due to an problem with the opposite train breaker (alternate path development). When the candidate first attempts to close either breaker 105 or breaker 125 the breaker they initially choose will NOT close. They will then have to restore power to the other bus. The conditional triggers will clear the other breakers failure when the first breaker switch is taken to the CLOSE position.
- Create 2 trigger files (note these files will NOT need to be recreated I have saved them to the Simulator trigger file this is just how I did it)
    - Breaker104toclose
        - @xbbi073|JIS|DI.value==3
    - Breaker124toclose
        - @xbbi077|JIS|DI.value==3

- Open ET (Event Trigger Summary)

- On trigger 2 – click assign file then type in the following
    - Breaker104toclose
- Click – link command – then type in the following
    - dor xd1i075 (n 0 0) ASIS

- On trigger 3 – click assign file then type in the following
    - Breaker124toclose
- Click – link command – then type in the following
    - dor xd1i066 (n 0 0) ASIS

- Place the Simulator in Run – insert Trigger 1
    - Isolate Letdown
    - Adjust TDAFW flow to maintain AFW flow > 200 KPPH and NR levels rising to restore levels to between 25% to 50% (this may require adjusting TDAFW pump speed as necessary to raise flow)
    - Place the EDG 1B-SB emergency stop switch to EMERG STOP

- Delete the Loss of Offsite Power malfunction
    - DMF EPS01

- FREEZE and SNAP these conditions to your exam IC

Simulator Operator:	***When directed by the Lead Examiner go to Run.***

START TIME: ___________

Evaluator Note:	EOP ECA-0.0 Step 9 Directs energizing AC Emergency Buses from Offsite Power using Attachment 1 The attachment allows flexibility of energizing Emergency Bus 'A' with steps 2-8 or 'B' with steps 9-15. There isn't a fault indicated on either bus so a candidate should NOT be suspecting that either bus has a fault. Since the JPM is going to be ran as an ALTERNATE PATH the candidate has the choice of attempting to re-energize either bus first. Either choice will yield a failure of energizing the first bus but will have a success path for energizing the second bus. Since there could be a decision made by the candidate on which bus to restore first the JPM has a Part A (steps 2-8) and Part B steps 9-15). IF the candidate starts with trying to energize the 'A' bus (more than likely) use Part A of the JPM. IF the candidate starts with trying to energize the 'B' bus (least likely – maybe suspects a fault due to failure of EDG 'B' to start) use Part B.

Common step for Part A and Part B

EOP-ECA-0.0 Attachment 1 - RESTORATION OF OFFSITE POWER TO EMERGENCY BUSES

Caution prior to step 1

Performance Step: 1 **CAUTION**

Tripping of a Start Up XFMR lockout relay indicates a major fault on the XFMR. Re-energizing the XFMR may cause additional damage and should **NOT** be done without dispatcher's permission.

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

Common step for Part A and Part B

EOP ECA-0.0, Attachment 1, Step 1.a, b

Performance Step: 2 Obtain Load Dispatcher's permission prior to performing the following:

a. Restoring offsite power to 6.9 KV buses
b. Resetting any tripped Start Up XFMR lockout relays

Standard: Information provided by CRS stated that the Load Dispatcher has provided permissions to restore offsite power to the 6.9 KV buses and reset any tripped Startup XFMR lockout relays

Comment:

EOP ECA-0.0, Attachment 1, Caution / Note prior to Step 2

Performance Step: 3 **CAUTION**

An AC Bus should **NOT** be re‑energized if it is suspected the bus may be faulted.

NOTE

Steps 2 through 8 restore power to Bus A‑SA and Steps 9 through 15 restore power to Bus B‑SB.

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

Part A, Energizing the ‘A’ Emergency Bus first starts on the next page

Part B, Energizing the ‘B’ Emergency Bus first starts on page 14

PART A – Attempting restoration of power to the ‘A’ Emergency Bus first

PART B – Attempting restoration of power to the ‘B’ Emergency Bus first (go to page 14)

EOP ECA-0.0, Attachment 1, Step 2.a

Performance Step: 4.a On Start Up XFMR Protective Relay Panel 1A, verify off‑site power to Start Up XFMR A:
a. Verify the Start Up XFMR 1A Lockout SU 1A Relay is reset.

Standard: Locates Startup XFMR 1A Lockout SU 1A Relay and verifies that the relay is reset. **(Relay is reset)**

Comment:

EOP ECA-0.0, Attachment 1, Step 2.b

Performance Step: 5.a b. Verify closed any of the following switch yard tie breakers to energize Start Up XFMR A:

- Breaker 52‑2
- Breaker 52‑3

Standard: Locates tie breaker switches for Startup XFMR A

- Breaker 52‑2 **(Verifies already closed)**
- Breaker 52‑3 **(Not required to be closed but maybe closed w/o consequences)**

Comment:

EOP ECA-0.0, Attachment 1, Step 3.a

Performance Step: 6.a Restore offsite power to 6.9 KV Aux Bus D:

a. Place Start Up XFMR To Aux Buses A & D Synchronizer control switch to BREAKER 101 position.

Standard: Locates Synchronizer control switch for Start Up XFMR To Aux Buses A & D and places switch to Breaker 101 position

Comment:

EOP ECA-0.0, Attachment 1, Step 3.b

Performance Step: 7.a b. Close Start Up XFMR A To Aux Bus D Breaker 101.

Standard: Locates switch for Start Up XFMR A To Aux Bus D Breaker 101 and places switch to CLOSE. **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 3.c

Performance Step: 8.a c. Place Start Up XFMR To Aux Buses A & D Synchronizer control switch to OFF.

Standard: Locates Synchronizer control switch for Start Up XFMR To Aux Buses A & D and places switch to OFF

Comment:

EOP ECA-0.0, Attachment 1, Step 4

Performance Step: 9.a Verify Aux Bus D To Emergency Bus A‑SA Breaker 104 ‑ CLOSED

Standard: Locates Aux Bus D to Emergency Bus A-SA Breaker 104 switch and takes switch to CLOSE **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 5

Performance Step: 10.a Verify Diesel Generator A‑SA Breaker 106 A SA ‑ OPEN

Standard: Locates Diesel Generator A‑SA Breaker 106 A SA switch and verifies breaker is Open **(GREEN LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 6.a

Performance Step: 11.a Energize 6.9 KV Bus A‑SA:

a. Place Emergency Bus A‑SA To Aux Bus D Synchronizer control switch to SYNC.

Standard: Locates Synchronizer control switch for Emergency Bus A‑SA To Aux Bus D and places control to SYNC

Comment:

EOP ECA-0.0, Attachment 1, Step 6.b

Performance Step: 12.a b. Close Emergency Bus A‑SA To Aux Bus D Tie Breaker 105.

Standard: Locates switch for Emergency Bus A‑SA To Aux Bus D Tie Breaker 105 and takes switch to CLOSE.

(GREEN LIGHT STAYS LIT) – Reports to CRS that Emergency Bus A‑SA To Aux Bus D Tie Breaker 105 will not close

(may dispatch AO to investigate)

Evaluator Cue:	**Acknowledge report that Emergency Bus A‑SA To Aux Bus D Tie Breaker 105 will not close.**

Simulator Communicator:	**IF AO is dispatched: Acknowledge and repeat back communications to investigate breaker**

Evaluator NOTE:	**IF needed to get the candidate back on task: Ask for an estimation on when power will be restored to an Emergency Bus.**

Comment:

EOP ECA-0.0, Attachment 1, Step 9.a – Alternate Path Begins

Restoration of power from the Start Up XFMR 1B to the B-SB Emergency Bus

Performance Step: 13.a On Start Up XFMR Protective Relay Panel 1B, verify off‑site power to Start Up XFMR B:
a. Verify the Start Up XFMR 1B Lockout SU 1B Relay is reset.

Standard: Locates Startup XFMR 1B Lockout SU 1B Relay and verifies that the relay is reset. **(Relay is reset)**

Comment:

EOP ECA-0.0, Attachment 1, Step 9.b

Performance Step: 14.a b. Verify closed any of the following switch yard tie breakers to energize Start Up XFMR B:

- Breaker 52‑13
- Breaker 52‑14

Standard: Locates tie breaker switches for Startup XFMR B

- Breaker 52‑13 **(Not required to be closed but maybe closed w/o consequences)**
- Breaker 52‑14 **(Verifies already closed)**

Comment:

EOP ECA-0.0, Attachment 1, Step 10.a

✓ **Performance Step: 15.a** Restore offsite power to 6.9 KV Aux Bus E:

a. Place Start Up XFMR To Aux Buses B & E Synchronizer control switch to BREAKER 121 position.

Standard: Locates Synchronizer control switch for Start Up XFMR To Aux Buses B & E and places switch to Breaker 121 position

Comment:

EOP ECA-0.0, Attachment 1, Step 10.b

✓ **Performance Step: 16.a** b. Close Start Up XFMR B To Aux Bus E Breaker 121.

Standard: Locates switch for Start Up XFMR B To Aux Bus E Breaker 121 and places switch to CLOSE. **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 10.c

Performance Step: 17.a c. Place Start Up XFMR To Aux Buses B & E Synchronizer control switch to OFF.

Standard: Locates Synchronizer control switch for Start Up XFMR To Aux Buses B & E and places switch to OFF

Comment:

EOP ECA-0.0, Attachment 1, Step 11

✓ **Performance Step: 18.a** Verify Aux Bus E To Emergency Bus B‑SB Breaker 124 ‑ CLOSED

Standard: Locates Aux Bus D to Emergency Bus B-SB Breaker 124 switch and takes switch to CLOSE **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 12

Performance Step: 19.a Verify Diesel Generator B‑SB Breaker 126 B SB ‑ OPEN

Standard: Locates Diesel Generator B‑SB Breaker 126 B SB switch and verifies breaker is Open **(GREEN LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 13.a

✓ **Performance Step: 20.a** Energize 6.9 KV Bus B‑SB:

a. Place Emergency Bus B‑SB To Aux Bus E Synchronizer control switch to SYNC.

Standard: Locates Synchronizer control switch for Emergency Bus B‑SB To Aux Bus E and places control to SYNC

Comment:

EOP ECA-0.0, Attachment 1, Step 13.b

✓ **Performance Step: 21.a** b. Close Emergency Bus B‑SB To Aux Bus E Tie Breaker 125.

Standard: Locates switch for Emergency Bus B‑SB To Aux Bus E Tie Breaker 125 and takes switch to CLOSE. **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 13.c

Performance Step: 22.a c. Place Emergency Bus B‑SB To Aux Bus E Synchronizer control switch to OFF.

Standard: Locates Synchronizer control switch for Emergency Bus B‑SB To Aux Bus E and places control to OFF

Comment:

EOP ECA-0.0, Attachment 1, Step 14

✓ **Performance Step: 23.a**

Close the following 6.9 KV breakers:

- Emergency Bus B‑SB To XFMR B1‑SB Breaker B1 A‑SB
- Emergency Bus B‑SB To XFMR B3‑SB Breaker B3 A‑SB

Standard:

- Locates control switch for Emergency Bus B‑SB To XFMR B1‑SB Breaker B1 A‑SB and places control to CLOSE **(RED LIGHT LIT)**
- Locates control switch for Emergency Bus B‑SB To XFMR B3‑SB Breaker B3 A‑SB and places control to CLOSE **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 15

✓ **Performance Step: 24.a**

Verify 6.9 KV Emergency Bus B‑SB To XFMR B2‑SB Breaker B2 A‑SB ‑ CLOSED

Standard:

- Locates control switch for 6.9 KV Emergency Bus B‑SB To XFMR B2‑SB Breaker B2 A‑SB and places control to CLOSE **(RED LIGHT LIT)**

Informs CRS that power is restored to Emergency Bus B-SB

Evaluator Cue:	**Acknowledge any reports:** **After the 6.9 KV Emergency Bus B‑SB power is restored: Evaluation on this JPM is complete.** **I have the shift, END OF JPM** **Inform Simulator Operator to place the Simulator in Freeze.**

Comment:

STOP TIME: ______________

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

PART B – Attempting restoration of power to the 'B' Emergency Bus first

Restoration of power from the Start Up XFMR 1B to the B-SB Emergency Bus

Performance Step: 4.b On Start Up XFMR Protective Relay Panel 1B, verify off‑site power to Start Up XFMR B:
a. Verify the Start Up XFMR 1B Lockout SU 1B Relay is reset.

Standard: Locates Startup XFMR 1B Lockout SU 1B Relay and verifies that the relay is reset. **(Relay is reset)**

Comment:

EOP ECA-0.0, Attachment 1, Step 9.b

Performance Step: 5.b b. Verify closed any of the following switch yard tie breakers to energize Start Up XFMR B:

- Breaker 52‑13
- Breaker 52‑14

Standard: Locates tie breaker switches for Startup XFMR B

- Breaker 52‑13 **(Not required to be closed but maybe closed w/o consequences)**
- Breaker 52‑14 **(Verifies already closed)**

Comment:

EOP ECA-0.0, Attachment 1, Step 10.a

Performance Step: 6.b Restore offsite power to 6.9 KV Aux Bus E:

d. Place Start Up XFMR To Aux Buses B & E Synchronizer control switch to BREAKER 121 position.

Standard: Locates Synchronizer control switch for Start Up XFMR To Aux Buses B & E and places switch to Breaker 121 position

Comment:

EOP ECA-0.0, Attachment 1, Step 10.b

Performance Step: 7.b e. Close Start Up XFMR B To Aux Bus E Breaker 121.

Standard: Locates switch for Start Up XFMR B To Aux Bus E Breaker 121 and places switch to CLOSE. **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 10.c

Performance Step: 8.b f. Place Start Up XFMR To Aux Buses B & E Synchronizer control switch to OFF.

Standard: Locates Synchronizer control switch for Start Up XFMR To Aux Buses B & E and places switch to OFF

Comment:

EOP ECA-0.0, Attachment 1, Step 11

Performance Step: 9.b Verify Aux Bus E To Emergency Bus B‑SB Breaker 124 ‑ CLOSED

Standard: Locates Aux Bus D to Emergency Bus B-SB Breaker 124 switch and takes switch to CLOSE **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 12

Performance Step: 10.b Verify Diesel Generator B‑SB Breaker 126 B SB ‑ OPEN

Standard: Locates Diesel Generator B‑SB Breaker 126 B SB switch and verifies breaker is Open **(GREEN LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 13.a

Performance Step: 11.b Energize 6.9 KV Bus B-SB:

d. Place Emergency Bus B-SB To Aux Bus E Synchronizer control switch to SYNC.

Standard: Locates Synchronizer control switch for Emergency Bus B-SB To Aux Bus E and places control to SYNC

Comment:

EOP ECA-0.0, Attachment 1, Step 13.b

Performance Step: 12.b

e. Close Emergency Bus B-SB To Aux Bus E Tie Breaker 125.

Standard: Locates switch for Emergency Bus B-SB To Aux Bus E Tie Breaker 125 and takes switch to CLOSE.

(GREEN LIGHT STAYS LIT) – Reports to CRS that Emergency Bus B-SB To Aux Bus E Tie Breaker 125 will not close

(may dispatch AO to investigate)

Evaluator Cue:	**Acknowledge report that Emergency Bus B-SB To Aux Bus E Tie Breaker 125 will not close.**

Simulator Communicator:	**IF AO is dispatched: Acknowledge and repeat back communications to investigate breaker**

Evaluator NOTE:	**IF needed to get the candidate back on task: Ask for an estimation on when power will be restored to an Emergency Bus.**

Comment:

EOP ECA-0.0, Attachment 1, Step 2.a – Alternate Path Begins

Restoration of power from the Start Up XFMR 1A to the A-SA Emergency Bus

EOP ECA-0.0, Attachment 1, Step 2.a

Performance Step: 13.b On Start Up XFMR Protective Relay Panel 1A, verify off‑site power to Start Up XFMR A:
a. Verify the Start Up XFMR 1A Lockout SU 1A Relay is reset.

Standard: Locates Startup XFMR 1A Lockout SU 1A Relay and verifies that the relay is reset. **(Relay is reset)**

Comment:

EOP ECA-0.0, Attachment 1, Step 2.b

Performance Step: 14.b b. Verify closed any of the following switch yard tie breakers to energize Start Up XFMR A:

- Breaker 52‑2
- Breaker 52‑3

Standard: Locates tie breaker switches for Startup XFMR A

- Breaker 52‑2 **(Verifies already closed)**
- Breaker 52‑3 **(Not required to be closed but maybe closed w/o consequences)**

Comment:

EOP ECA-0.0, Attachment 1, Step 3.a

✓ **Performance Step: 15.b** Restore offsite power to 6.9 KV Aux Bus D:

d. Place Start Up XFMR To Aux Buses A & D Synchronizer control switch to BREAKER 101 position.

Standard: Locates Synchronizer control switch for Start Up XFMR To Aux Buses A & D and places switch to Breaker 101 position

Comment:

EOP ECA-0.0, Attachment 1, Step 3.b

✓ **Performance Step: 16.b** e. Close Start Up XFMR A To Aux Bus D Breaker 101.

Standard: Locates switch for Start Up XFMR A To Aux Bus D Breaker 101 and places switch to CLOSE. **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 3.c

Performance Step: 17.b f. Place Start Up XFMR To Aux Buses A & D Synchronizer control switch to OFF.

Standard: Locates Synchronizer control switch for Start Up XFMR To Aux Buses A & D and places switch to OFF

Comment:

EOP ECA-0.0, Attachment 1, Step 4

✓ **Performance Step: 18.b** Verify Aux Bus D To Emergency Bus A‑SA Breaker 104 ‑ CLOSED

Standard: Locates Aux Bus D to Emergency Bus A-SA Breaker 104 switch and takes switch to CLOSE **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 5

Performance Step: 19.b Verify Diesel Generator A‑SA Breaker 106 A SA ‑ OPEN

Standard: Locates Diesel Generator A‑SA Breaker 106 A SA switch and verifies breaker is Open **(GREEN LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 6.a

✓ **Performance Step: 20.b** Energize 6.9 KV Bus A‑SA:

b. Place Emergency Bus A‑SA To Aux Bus D Synchronizer control switch to SYNC.

Standard: Locates Synchronizer control switch for Emergency Bus A‑SA To Aux Bus D and places control to SYNC

Comment:

EOP ECA-0.0, Attachment 1, Step 6.b

✓ **Performance Step: 21.b**

c. Close Emergency Bus A‑SA To Aux Bus D Tie Breaker 105.

Standard:

Locates switch for Emergency Bus A‑SA To Aux Bus D Tie Breaker 105 and takes switch to CLOSE.
(RED LIGHT LIT)

Comment:

EOP ECA-0.0, Attachment 1, Step 6.c

Performance Step: 22.b

a. Place Emergency Bus A‑SA To Aux Bus D Synchronizer control switch to OFF.

Standard:

Locates Synchronizer control switch for Emergency Bus A‑SA To Aux Bus D and places control to OFF

Comment:

EOP ECA-0.0, Attachment 1, Step 7

✓ **Performance Step: 23.b**

Close the following 6.9 KV breakers:

- Emergency Bus A‑SA To XFMR A1‑SA Breaker A1 A‑SA
- Emergency Bus A‑SA To XFMR A3‑SA Breaker A3 A‑SA

Standard:

- Locates control switch for Emergency Bus A‑SA To XFMR A1‑SA Breaker A1 A‑SA and places control to CLOSE **(RED LIGHT LIT)**
- Locates control switch for Emergency Bus A‑SA To XFMR A3‑SA Breaker A3 A‑SA and places control to CLOSE **(RED LIGHT LIT)**

Comment:

EOP ECA-0.0, Attachment 1, Step 8

✓ **Performance Step: 24.b** Verify 6.9 KV Emergency Bus A‑SA To XFMR A2‑SA Breaker A2 A‑SA ‑ CLOSED

Standard:

- Locates control switch for 6.9 KV Emergency Bus A‑SA To XFMR A2‑SA Breaker A2 A‑SA and places control to CLOSE **(RED LIGHT LIT)**

Informs CRS that power is restored to Emergency Bus A-SA

Evaluator Cue:	**Acknowledge any reports:** **After the 6.9 KV Emergency Bus A‑SA power is restored: Evaluation on this JPM is complete.** **I have the shift, END OF JPM** **Inform Simulator Operator to place the Simulator in Freeze.**

Comment:

STOP TIME: ______________

Simulator Operator:	**When directed by the Lead Examiner then go to Freeze.**

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 HNP NRC Exam Simulator JPM f

Restoration of Offsite Power to Emergency Buses
In accordance with EOP ECA-0.0, Loss Of All AC Power, Attachment 1

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 • 'A' EDG is under clearance due to a failure that caused the Generator field to not flash during OST-1013 Subsequently: • A failure of a transmission line on the Duke grid resulted in the cascading trip of several units which resulted in low grid frequency • The HNP unit has experienced a loss of offsite power • 'B' EDG failed to start. The problem is being investigated • The crew entered ECA-0.0, Loss Of All AC Power • The load dispatcher has contacted HNP and informed the MCR that the grid is now stabile

Initiating Cue:	• Your position is the BOP • The CRS has directed you to restore offsite power to a (one) AC emergency bus using ECA-0.0 Attachment 1. • The Load Dispatcher has given permission to restore offsite power to 6.9 KV buses and to reset any tripped Start Up XFMR lockout relays.

Facility:	Harris Nuclear Plant	Task No.:	015005H401
Task Title:	Take an Excore NI Channel Out Of Service at Power (OWP-RP-26)	JPM No.:	2018 NRC Exam Simulator JPM CR g
K/A Reference:	015 A4.03 RO 3.8 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 operating at 100 percent power • NI-44 has failed low

Initiating Cue:	The CRS has directed you to remove NI-44 from service per OWP-RP-26.

Evaluator Note:	**The candidates should be briefed outside of the Simulator prior to performing this JPM. Provide them with a copy of OWP-RP-26, Reactor Protection .** **This will allow them to review the procedure and perform a task preview of the steps to remove the NI channel from service. Expect that the candidates will take about 5-10 minutes to complete this review.**

Task Standard: NI-44 removed from service in accordance with OWP-RP-26

Required Materials: OWP-RP-26, Rev. 17

General References: OWP-RP-26, Rev. 17

Time Critical Task: No

Validation Time: 15 minutes

Critical Step Justification	
Step 4	Placing rod bank selector switch to manual prevents inadvertent reactivity event with unnecessary auto rod movement.
Step 6	Must select correct switch and correct switch position for channel to defeat upper detector comparator
Step 7	Must select correct switch and correct switch position for channel to restore rod control system to allow auto rod control
Step 8	Must select correct switch and correct switch position for channel to defeat comparison of NI channels

2018 NRC Exam - SIMULATOR SETUP

Simulator Operator

- Reset to IC-171
- Password "NRC2018"
- 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

- Insert IMF NIS08D 0.0, PRNIS Channel 44 failed low

- Create Trigger 1

  Irf nis032 (1 0 0) disconnect

  NOTE: Running Trigger 1 will simulate disconnecting P312 from J312 at the rear of N44 Drawer A.

- Silence Acknowledge and Reset Annunciators

- Freeze and Snap these conditions to your exam IC

START TIME: ____________

Performance Step: 1 Obtain procedure (Provided by Examiner)

Standard: Obtains OWP-RP-Section 26.

Comment:

OWP-RP-26 Sheet 1

Performance Step: 2 Sheet 1 contains information on which component the OWP is written for, the scope, applicable requirements, precautions, the component lineup, testing requirements, testing action, component lineups restore, remarks and reviewed by.

Standard: Reads sheet 1 to ensure the correct component and scope is for N-44, reviews precautions, testing required on redundant equipment while NI-44 is inoperable and actions to restore to operability.

Comment:

Evaluator Cue:	**If needed, “LCO actions have already been addressed.”**

OWP-RP-26 Sheet 2

Performance Step: 3 **Reviews NOTE: This OWP must be performed in order to prevent possible spurious rod motion or level control swings.**

Standard: Reads and place keeps Notes or Cautions prior to performing step.

Comment:

OWP-RP-26 Sheet 2 continued

✓ **Performance Step: 4** On Main Control Board Check position:

- Rod Bank Selector switch – MANUAL ***(critical portion)***

Standard: Places Rod Bank Selector Switch in MANUAL.

-Voices OMM-001, Att. 13, Normal Bands and Trip Limits

-Control Band: Tavg within +/- 2°F of Tref

-Trip Limits: Tavg exceed +/- 10°F of Tref

Comment:

OWP-RP-26 Sheet 2 continued

Performance Step: 5 On Main Control Board check position:

FW Reg BYP Valve Controllers:

- FK-479.1 – MANUAL
- FK-489.1 – MANUAL
- FK-499.1 - MANUAL

Standard: Verifies all three SG FW Reg Bypass controllers are in MANUAL:

FK-479.1 ☐
FK-489.1 ☐
FK-499.1. ☐

Comment:

OWP-RP-26 Sheet 2 continued

✓ **Performance Step: 6** On Detector Current Comparator Drawer position:

- Upper Section Switch – to PR N44
- Lower Section Switch – to PR N44

Standard: Selects PR N44 on UPPER SECTION SWITCH ☐
Selects PR N44 on LOWER SECTION SWITCH ☐

Evaluator Note:	**Channel Defeat lights on drawer will illuminate.**

Comment:

OWP-RP-26 Sheet 2 continued

✓ **Performance Step: 7** On Miscellaneous Control and Indication Panel position:

- Rod Stop Bypass Switch – to Bypass PR N44
- Power Mismatch Bypass Switch – to PR44

Standard: Selects BYPASS PR N44 on ROD STOP BYPASS SWITCH ☐

Selects BYPASS PR N44 on POWER MISMATCH BYPASS SWITCH ☐

Comment:

OWP-RP-26 Sheet 2 continued

✓ **Performance Step: 8** On Comparator and Rate Drawer position:

Comparator Channel Defeat Switch – to N44.

Standard: Selects N44 on the COMPARATOR CHANNEL DEFEAT switch ☐

Evaluator Note:	Defeat light on drawer illuminates. PR CH DEV annunciator alarm clears.

Comment:

OWP-RP-26 Sheet 2 continued

Performance Step: 9 NOTE: The purpose of the sign installed below is to alert personnel of tripped bistables that may not be obvious at the NI drawer. The wording in quotations is the recommended wording, but similar words may also be used.

Standard: Reads and place keeps note prior to performing step

Comment:

OWP-RP-26 Sheet 2 continued

Performance Step: 10 Sign stating “Bistables Tripped - OWP-RP in Affect”

Standard: Locates sign and places it in an obvious position on the NI drawer for NI-44.

Comment:

OWP-RP-26 Sheet 3

Performance Step: 11 NOTE: Concurrent verification is preferred in the following step.

At the rear of N44 Drawer A, disconnect P312 from J312

Contact Maintenance to disconnect leads

Standard: Calls Maintenance to disconnect cable at rear of N44 Drawer A and requests that second I&C person accompanies person lifting leads to perform Concurrent verification

Evaluator / Simulator Operator Cue:	**If candidate calls for Maintenance - acknowledge request with proper communications.** *** Inform the applicant that time compression is being used for I&C to report to MCR and that I&C is ready to disconnect leads.** **(Contact Simulator Operator to run Trigger 1 to simulate lifting leads)** **NOTE: The applicant may request that the step to disconnect the cable is initialed prior to continuing.** **IF SO then cue them to assume that the step is initialed.**

Simulator Operator:	**Run Trigger 1 (remote function nis032)** **This file simulates disconnecting P312 from J312.** **After the file is completed wait 10 seconds then report back that the disconnect P312 from J312 has been completed.**

Comment:

OWP-RP-26 Sheet 3 continued

Performance Step: 12 On completion of the above lineup, check the following.
On TSLB-4
PR P-8 NC44N (Window 3-4) ENERGIZED

Standard: Locates window 3-4 on TSLB-4 and initials “ENERGIZED” line

Comment:

OWP-RP-26 Sheet 3 continued

Performance Step: 13 On TSLB-4
PR P-7/P-10 NC44M (Window 4-4) ENERGIZED

Standard: Locates window 4-4 on TSLB-4 and initials “ENERGIZED” line

Comment:

OWP-RP-26 Sheet 3 continued

Performance Step: 14 On TSLB-4
PR LO PWR LO FLUX NC 44P (Window 5-4) ENERGIZED

Standard: Locates window 5-4 on TSLB-4 and initials “ENERGIZED” line

Comment:

OWP-RP-26 Sheet 3 continued

Performance Step: 15 On TSLB-4

PR LO PWR HI FLUX NC 44R (Window 6-4) ENERGIZED

Standard: Locates window 6-4 on TSLB-4 and initials “ENERGIZED” line

Comment:

OWP-RP-26 Sheet 3 continued

Performance Step: 16 On TSLB-4

PR HI FLUX RATE NC 44U/K (Window 7-4) ENERGIZED

Standard: Locates window 7-4 on TSLB-4 and initials “ENERGIZED” line

Comment:

OWP-RP-26 Sheet 3 continued

Performance Step: 17 On BYPASS PERMISSIVE LIGHTS Panel.

PR OVERPWR ROD WTHDRWL BLK BYPASS CHAN IV

(Window 3-8) ENERGIZED

Standard: Locates window 3-8 on BYPASS PERMISSIVE LIGHTS Panel and initials “ENERGIZED” line

Comment:

OWP-RP-26 Sheet 4

Performance Step: 18 On ERFIS Computer - After status lights have been checked, perform the following using the DR function.

ANM0123M - PWR RNG CHANNEL N44 Q3 1-MIN AVG

DELETED FROM PROCESSING

Standard: Uses the DR function on the ERFIS Computer and removes ANM0123M - PWR RNG CHANNEL N44 Q3 1-MIN AVG from processing

Comment:

OWP-RP-26 Sheet 4 continued

Performance Step: 18 On MAIN CONTROL BOARD: Circle appropriate position as determined by plant conditions.

ROD BANK SELECTOR Switch MAN/AUTO+

Standard: Checks Rod Bank Selector Switch position

(can circle MAN after cue provided)

Evaluator Cue:	**The applicant may determine that AUTO rod control can be accomplished - CUE them prior to obtaining the procedure:** **The CRS directs that the Rod Bank Selector Switch be left in MANUAL to support other plant activities.**

Comment:

OWP-RP-26 Sheet 4 continued

Performance Step: 18 FW Reg Byp Valve Controllers:

Circle appropriate position as determined by plant conditions:

FK-479.1 MAN/AUTO+
FK-489.1 MAN/AUTO+
FK-499.1 MAN/AUTO+

Standard: (Per current plant conditions)

Circles MAN for:

FK-479.1 [✓ MAN]/AUTO+
FK-489.1 [✓ MAN]/AUTO+
FK-499.1 [✓ MAN]/AUTO+

Comment:

OWP-RP-26 Sheet 4 continued

Performance Step: 19 Reports to CRS

Standard: Reports to CRS that N44 has been removed from service in accordance with OWP-RP-26

Evaluator Cue:	**The CRS acknowledges that N44 has been removed from service in accordance with OWP-RP-26**

Comment:

Evaluator Cue:	**After lineup has been completed and the report provided to CRS this JPM is complete.** **Announce: I have the shift. END OF JPM** **Contact the Simulator Operator to place the Simulator in FREEZE.**

STOP TIME: _______________

Simulator Operator:	**When directed by the Evaluator place the Simulator in FREEZE.** **Note: When the Simulator is reset the ERFIS computer will reset and there is NO NEED to restore the point the candidate took out of processing.**

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Simulator JPM CR g

Place An Excore NI Channel Out Of Service in accordance with OWP-RP-26

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 percent power • NI-44 has failed low

Initiating Cue:	• The CRS has directed you to remove NI-44 from service per OWP-RP-26.

Facility:	Harris Nuclear Plant	Task No.:	301064H401
Task Title:	Respond to an Instrument Air Header Rupture at 50% power (AOP-017)	JPM No.:	2018 NRC Exam Simulator JPM CR h
K/A Reference:	APE 065 AA2.06 RO 3.6 SRO 4.2	**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 50% power during a startup - Startup is on hold due to chemistry concerns

Initiating Cue:	You are the OATC. Your directions are to maintain current plant conditions.

Evaluator Note:	***Prior to starting this JPM position a second board operator at the Shift Managers desk. The second board operator will take no actions during the initiating event prior to the Reactor Trip and completion of EOP-E-0 immediate actions. AFTER the immediate actions are completed by the candidate the second board operator will be introduced by the Evaluator and will be stabilizing the unit by controlling AFW and silencing annunciators not related to the initiating event.***

Task Standard: Trips the reactor, carries out immediate actions of EOP-E-0

Required Materials: AOP-017, Rev 40

General References: AOP-017, Rev 40

Handout:

Time Critical Task: No

Validation Time: 15 min

CRITICAL STEP JUSTIFICATION	
Step 4	Identification of the need to trip the Reactor and carrying out the immediate actions of E-0 will place the plant in a known stable condition.
Step 16	The controllers listed in this attachment are positioned as specified by the operator at a point directed by the procedure main body, in order to ensure that the controlled devices will remain in an appropriate condition after restoring air pressure. At that point in the event, the operator can recover the systems in a controlled manner.
Step 17	Maintaining PRZ Pressure in a stable band allows the operator minimize the impact of the loss of air to the RCS in order to recover the systems in a controlled manner.
Step 18	The controllers listed in this attachment are positioned as specified by the operator at a point directed by the procedure main body, in order to ensure that the controlled devices will remain in an appropriate condition after restoring air pressure. At that point in the event, the operator can recover the systems in a controlled manner.
Step 19	The controllers listed in this attachment are positioned as specified by the operator at a point directed by the procedure main body, in order to ensure that the controlled devices will remain in an appropriate condition after restoring air pressure. At that point in the event, the operator can recover the systems in a controlled manner.

SIMULATOR SETUP

Simulator Operator

- Reset to IC-172
- Password “NRC2018”
- Plant status board updated per IC-5 data
- Initial conditions Reactor ~50% power
- 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.)

NOTE: Since the candidate will be using the Simulator copy of AOP-017 ensure that replacement copies are made prior to starting the JPM. REPLACE THE ENTIRE PROCEDURE AFTER EACH CANDIDATE COMPLETES THIS JPM.

To recreate the IC setup for this JPM:

- Initial Simulator IC was IC-5
- Go to run
- Disable ‘A’ and ‘B’ Air Compressors by shutting compressor discharge valves
- On Trigger 1 place a trip of the ‘C’ Air Compressor and an Instrument Air Header Rupture (severity of 100%)

    - ifr air002 (1 0 0) 0 0 0 (Air Comp 1A Disc Valve shut)
    - ifr air003 (1 0 0) 0 0 0 (Air Comp 1B Disc Valve shut)
    - imf air02 (1 0 0) 100 00:05:00 0 (Air header leak 100% 5 min ramp)

- On Trigger 2 place commands to turn off All Air Compressors

    - irf air012 (2 0 0) LOCKED_OFF (Air Comp 1A Locked Off)
    - irf air013 (2 0 0) LOCKED_OFF (Air Comp 1B Locked Off)
    - irf air020 (2 0 0)STOP (Air Compressor C Stop)

- On Trigger 3 place commands to Vent IA header per request by candidate (AFTER Instrument Air pressure is < 35 psig)

    - irf air024 (3 0 0) 100 0 0 (Opens IA-814 to 100%)

Simulator Operator:	**When directed by Lead Examiner go to Run** **10-15 seconds after the candidate assumes the watch, insert Trigger 1**

START TIME: ____________

Performance Step: 1 Responds to Instrument Air Header alarms

ALB-02-8-5, Computer Alarm Air Systems

- IF the alarm screen is checked the alarm is due to too many Air Compressors running

Standard: Diagnoses loss of Instrument air, enters AOP-017

Comment:

AOP-017, Note prior to Step 1

Performance Step: 2

- This procedure contains no immediate actions.
- FW regulating valves receive a shut signal when pressure falls to 60 psig on the Control Air header.
- PI-9751.1, Instrument Air Header Pressure, may not be indicative of pressure throughout the Instrument Air System. The plant should be monitored closely for possible spurious valve operations due to low system pressure.

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

AOP-017, Section 3.0 Step 1

Performance Step: 3 MAINTAIN BOTH of the following:

- ALL Steam Generator levels greater than 30% (YES)
- Main Feedwater flow to ALL Steam Generators (YES/NO)

NOTE: Depending on how long it takes the operator to get to this step (evaluating Air Compressors, dispatching AO's ect.), Main Feedwater could be lost and/or SG levels could be < 30% Narrow Range.

Standard: Determines all SG levels can/cannot be maintained greater 30% and Feedwater flow continues via the Bypasses

Comment:

AOP-017, Section 3.0 Step 1 RNO

✓ **Performance Step: 4** TRIP the Reactor AND PERFORM EOP-E-0 while continuing with this AOP.

Standard: Trips the Reactor and begins to carry out the Immediate Actions of EOP-E-0 prior to an Automatic Reactor Trip occurring

- Verify the Reactor tripped (YES)
- Verify the Turbine tripped (YES)
- Emergency Buses energized from Offsite or the Diesels (YES)
- Safety Injection actuated or required (NO)

Evaluator Cue:	**Once the immediate actions of EOP-E-0 have been completed then inform the candidate that "Additional operators will perform actions of EOP-E-0, the CRS directs you to continue on with actions of AOP-017."**

Comment:

AOP-017, Section 3.0 Step 2

Performance Step: 5 CHECK Instrument Air pressure MAINTAINED ABOVE 35 PSIG.

Standard: Determines Instrument Air pressure is <35 psig

Evaluator / Simulator Operator Note:	**Candidates may direct AO's to check Instrument Air compressors and look for air leaks.** **Acknowledge any of the requests.**

Comment:

AOP-017, Section 3.0 Step 2.a RNO

Performance Step: 6 **PERFORM** the following:

a. PERFORM Attachment 8, Loss Of Instrument Air Pressure.

Standard: Transitions and implements AOP-017, Attachment 8, Loss Of Instrument Air Pressure

Comment:

AOP-017, Attachment 8 NOTE Prior to Step 1

Performance Step: 7 Depressurizing Instrument Air precludes spurious valve actuations.

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

AOP-017, Attachment 8, Step 1

Performance Step: 8 CHECK Instrument Air pressure LESS THAN 35 PSIG at any time during the event.

Standard: Reviews MCB indications or ERFIS trends and determines IA pressure is below 35 psig

Comment:

AOP-017, Attachment 8, Step 2

Performance Step: 9 STOP ALL air compressors.

Standard: Directs field operator to stop all air compressors.

Simulator Operator:	**When contacted, acknowledge direction to secure all Air Compressors – RUN TRG-2**

Comment:

AOP-017, Attachment 8, Step 3.a

Performance Step: 10 VENT Instrument Air System until depressurized.

a. TRACK valve status using OPS-NGGC-1308.

Standard: Reads step that local actions for valve manipulations must be documented and tracked using the appropriate tracking procedures.

Comment: **Note: The candidate may comment that they know these local actions performed to vent IA system should be document on AD-OP-ALL-0204, Attachment 2, Configuration Control Card which has replaced OPS-NGGC-1308.**

AOP-017 has not been updated to the new AD-OP-ALL procedure yet.

AOP-017, Attachment 8 NOTE Prior to Step 3.b

Performance Step: 11

Suggested vent points:

- Instrument Air Receiver or Breathing Air Receiver drains
- Any Instrument Air drain or vent
- Drawing 2165-S-0801 may be used to identify additional vent and drain points

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

AOP-017, Attachment 8, Step 3.b

Performance Step: 12

VENT Instrument Air System until depressurized.

b. VENT the system using multiple vent points.

Standard: Contacts TB AO to perform local actions to vent IA system. Verifies Instrument air system is completely depressurized by the rupture.

Comment:

Simulator Operator:	**IF contacted to vent the IA system, acknowledge direction to do this task and then – RUN TRG-3** **Monitor IA pressure and report back when pressure has lowered to 0 psig.** **NOTE: IA pressure will continue to lower to 0 psig if the system is vented or not**

AOP-017, Attachment 8, Step 4

Performance Step: 13 VERIFY SHUT ALL MSIVs and MSIV bypasses

Standard: Checks all three MSIVs and bypasses SHUT

SG ‘A’ MSIV 1MS-80 ☐ Bypass 1MS-81 ☐
SG ’B’ MSIV 1MS-82 ☐ Bypass 1MS-83 ☐
SG ‘C’ MSIV 1MS-84 ☐ Bypass 1MS-85 ☐

Comment:

AOP-017, Attachment 8 NOTE Prior to Step 5

Performance Step: 14 The fail positions of critical valves controlled by Instrument Air can be determined from:

- Drawing 2165-S-0801
- Attachment 1, Fail Positions for Major Valves Controlled by Instrument Air

Standard: Operator reads and placekeeps at any procedure note or caution

Comment:

AOP-017, Attachment 8, Step 5

Performance Step: 15 REFER TO Attachment 2, Positioning MCB Controllers, AND PLACE listed controllers in the status indicated.

Standard: Transitions and implements AOP-017, Attachment 2, Positioning MCB Controllers

Comment:

AOP-017, Attachment 2, Step 1

✓ **Performance Step: 16** PLACE the following MCB controllers in MANUAL with ZERO demand:

- **FK-122.1, CHARGING FLOW**
- PK-464.1, STEAM DUMP HEADER PRESSURE CONTROLLER
- FK-605A1, RHR HEAT XCHG A BYPASS FLOW CONT
- FK-605B1, RHR HEAT XCHG B BYPASS FLOW CONT
- **PK-444C.1, LOOP A (PRZ Normal Spray)**
- **PK-444D.1, LOOP B (PRZ Normal Spray)**

Standard: Places each controller to MANUAL and lowers the demand to zero

- **FK-122.1, CHARGING FLOW** ☐
- PK-464.1, STEAM DUMP HEADER PRESSURE CONTROLLER
- FK-605A1, RHR HEAT XCHG A BYPASS FLOW CONT
- FK-605B1, RHR HEAT XCHG B BYPASS FLOW CONT
- **PK-444C.1, LOOP A (PRZ Normal Spray)** ☐
- **PK-444D.1, LOOP B (PRZ Normal Spray)** ☐

Comment: **Performance Step 16 items that are BOLDED and have a check box (3) are the CRITICAL STEPS.**

AOP-017, Attachment 2, Step 2

✓ **Performance Step: 17** MAINTAIN PZR pressure:

a. PLACE MCB controller PK444A, PRZ PRESS CONTROL, in MANUAL.
b. CONTROL PK444A OUTPUT between 15 and 35% (adjusted for C PZR HTR CONTROL).
c. MAINTAIN PZR pressure stable (within applicable control band).

Standard: Places controller to MANUAL and adjusts the demand output between 15 and 35%

(IAW OMM-001 Attachment 13 Control Bands and Trip Limits)

Control Band is 2210-2260 PSIG

Trip Limits: Low – 2050 PSIG, High – 2350 PSIG

Comment:

AOP-017, Attachment 2, Step 3

✓ **Performance Step: 18** **MAINTAIN** PZR pressure:

- **FK-478, MAIN FW A REGULATOR**
- **FK-488, MAIN FW B REGULATOR**
- **FK-498, MAIN FW C REGULATOR**
- FK-479.1, MN FW A REG BYP
- FK-489.1, MN FW B REG BYP
- FK-499.1, MN FW C REG BYP

Standard: Places each controller to MANUAL and lowers the demand to zero

- **FK-478, MAIN FW A REGULATOR** ☐
- **FK-488, MAIN FW B REGULATOR** ☐
- **FK-498, MAIN FW C REGULATOR** ☐
- FK-479.1, MN FW A REG BYP
- FK-489.1, MN FW B REG BYP
- FK-499.1, MN FW C REG BYP

Comment: **Performance Step 18 items that are BOLDED and have a check box (3) are the CRITICAL STEPS.**

AOP-017, Attachment 2, Step 4

✓ **Performance Step: 19** PLACE the following MCB controllers in MANUAL with 100% demand:

- HC-186.1, RCP SEAL WTR INJ FLOW
- HC-603A1, RHR HEAT XCHG A OUT FLOW CONT
- HC-603B1, RHR HEAT XCHG B OUT FLOW CONT

Standard: Places each controller to MANUAL and raises the demand to 100%

- HC-186.1, RCP SEAL WTR INJ FLOW ☐
- HC-603A1, RHR HEAT XCHG A OUT FLOW CONT ☐
- HC-603B1, RHR HEAT XCHG B OUT FLOW CONT ☐

Examiner Cue:	**When candidate exits Attachment 2** **Announce: I have the shift, END of JPM** **Contact the Simulator Operator and place the Simulator in FREEZE.**

Comment:

STOP TIME: ______________

Simulator Operator:	**When directed by Lead Examiner go to Freeze**

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Simulator JPM CR h

Respond to a rupture in the Instrument Air Header at 50% power

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 50% power during a startup • Startup is on hold due to chemistry concerns

Initiating Cue:	• You are the OATC. • Your directions are to maintain current plant conditions.

Facility:	Harris Nuclear Station	Task No.:	344058H504, 344059H504
Task Title:	Manually isolate the SG "C" PORV and SHUT the SG "C" TDAFW Pump steam supply MOV	JPM No.:	2018 NRC Exam Inplant JPM i
K/A Reference:	APE037 G2.1.30 RO 4.4 SRO 4.0	**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 Unit was initially at 100% power when tube leakage developed in 'C' SG • The Reactor is now shutdown and the crew is currently performing AOP-016, Excessive Primary Leakage, Attachment 11, Plant Shutdown Actions for Primary-To-Secondary Leakage Action Level 2 and 3 • While attempting to isolate SG 'C', the SG 'C' PORV failed to fully SHUT from the MCB • SG 'C' pressure is 1015 psig

Initiating Cue:	• You have been directed to locally shut 1MS-63, the SG "C" PORV block valve, per AOP-016, Attachment 11, Step 12.b RNO and report completion to the MCR

Task Standard:	1MS-63 (MS Line C PORV Isol Vlv) and 1MS-72 (MS "C" to Aux FW Turbine) manually shut
Required Materials:	PPE is optional for AOP performance.
General References:	AOP-016, EXCESSIVE PRIMARY LEAKAGE, Attachment 11, Rev 56
Handout:	AOP-016, Attachment 11 step 12
Time Critical Task:	No
Validation Time:	8 minutes

CRITICAL STEP JUSTIFICATION	
Steps 2 & 3	Terminates an uncontrolled release of radioactivity to the environment.

SIMULATOR SETUP

N/A – Inplant JPM

BEFORE YOU START THIS JPM

INPLANT 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 Core 4 checks.**

Evaluator:	**Provide candidate a copy of AOP-016, Attachment 11, Pages 47 – 49 with Step 12.a marked as complete and 12.b and 12.b RNO circled.**

Note: Log JPM start time when the operator arrives at 1MS-63

Log Start Time: _______________

Evaluator Note:	

AOP-016, Attachment 11, Step 12.b RNO

Performance Step: 1 WHEN leaking SG Pressure is less than 1145 PSIG, then verify the associated SG PORV SHUT.

Standard:

- Notes SG “C” Pressure < 1145 PSIG in Initial Conditions.

Comment:

AOP-016, Attachment 11, Step 12.b RNO

√ **Performance Step: 2** IF leaking SG(s) PORV(s) can NOT be SHUT, THEN LOCALLY SHUT the leaking SG(s) PORV Block valve(s).

Standard:

- Locates SG "C" PORV manual isolation valve 1MS-63.
- Simulates/discusses unlocking and rotating the valve handwheel in the clockwise direction until it is shut.

Simulates contacting the MCR to report that 1MS-63 is shut.

Evaluator Cue:	**1MS-63 has stopped rotating in the clockwise direction and flow through the valve has ceased.** **Acknowledge report that 1MS-63 is shut** **MCR informs you that 1MS-72, MS Line C to Steam Driven AFW Turbine MOV, failed to shut from the MCB. You are being directed to perform AOP-016, Attachment 11, Step 12.d.(2) RNO - locally shut 1MS-72.** **Another Aux Operator has been dispatched to open the breaker for 1MS-72. The breaker is now open and you may proceed to manually shutting 1MS-72, 'C' SG TDAFW Steam Supply valve.**

Comment:

AOP-016, Attachment 11 – Step 12.d

√ **Performance Step: 3** SHUT leaking SG(s) steam supply valve to TDAFW pump:

- SG C: 1MS-72 SB

Standard:

- Locates 1MS-72
- Simulates depressing the clutch lever
- Simulates rotating the handwheel in the clockwise direction until it stops rotating.

Evaluator Cue:	• **The clutch lever is depressed.** • **The handwheel has stopped turning in the clockwise direction.**

Comment:

Performance Step: 4 **Report task completion to Control Room.**

Standard: Simulates contacting the MCR to report that 1MS-63 and 1MS-72 are closed.

Evaluator Cue:	**Acknowledge report. Evaluation on this JPM is complete.**

Comment:

STOP TIME: ______________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Inplant JPM i

Manually isolate the SG “C” PORV and SHUT the SG “C” TDAFW Pump steam supply MOV

In accordance with AOP-016 Attachment 11

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

INPLANT 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 Unit was initially at 100% power when tube leakage developed in ‘C’ SG • The Reactor is now shutdown and the crew is currently performing AOP-016, Excessive Primary Leakage, Attachment 11, Plant Shutdown Actions for Primary-To-Secondary Leakage Action Level 2 and 3 • While attempting to isolate SG ‘C’, the SG ‘C’ PORV failed to fully SHUT from the MCB • SG ‘C’ pressure is 1015 psig

Initiating Cue:	• You have been directed to locally shut 1MS-63, the SG “C” PORV block valve, per AOP-016, Attachment 11, Step 12.b RNO and report completion to the MCR

Facility:	Harris Nuclear Plant	Task No.:	061012H104
Task Title:	Reset the Turbine-Driven AFW Pump Mechanical Overspeed	JPM No.:	2018 NRC Exam In-Plant JPM j
K/A Reference:	061 A2.04 RO 3.4 SRO 3.8		**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 plant was manually tripped from 100% power due to a loss of the 'A' MFW pump. • The Turbine-driven AFW pump is needed for plant cooldown but the pump tripped on overspeed. • The cause of the overspeed trip has been identified and corrected. • Main Steam isolation valves 1MS-70 and 1MS-72 are shut.

Initiating Cue:	• The CRS has directed you to reset the Turbine-driven AFW pump mechanical overspeed trip linkage in accordance with OP-137, Auxiliary Feedwater System, Section 8.4. • **Assume that the Mechanical Overspeed Trip Linkage is currently in the tripped position.** • The Trip and Throttle Valve will be reopened from the Control Room. • All Initial Conditions are met.

Evaluator:	**At this time provide the student with a copy of OP-137, Section 8.4**

NOTE: Expect that the entry and exit from the RCA will add time to complete this JPM.

Task Standard:	The Turbine-driven AFW pump turbine trip and throttle valve is latched.
Required Materials:	Standard PPE
General References:	OP-137, Auxiliary Feedwater System, Rev. 45
Handout:	OP-137, Section 8.4, and Attachment 6, Rev. 45 Note: OP-137, Attachment 6 is also locally mounted on wall as an operator aid
Time Critical Task:	No
Validation Time:	10 minutes

SIMULATOR SETUP

N/A

This is an In-Plant JPM

CRITICAL STEP JUSTIFICATION	
Step 9	If the connecting rod is not properly positioned and locked in place the over speed reset cannot be accomplished.
Step 10	If the tappet nut is not held down properly and in the correct sequence the over speed trip cannot be reset.

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 Core 4 checks.**

Start time begins when the candidate is at the TDAFW Pump after the Take a Minute concludes

START TIME: ____________

Reviews OP-137, Section 8.4 prior to task performance

OP-137, Section 8.4.1 Initial Conditions #1

Performance Step: 1 Mechanical Over speed Trip Linkage in the tripped position.

Standard: Reviews OP-137, Section 8.4 prior to task performance
Inspects Mechanical over speed trip linkage and determines that the linkage is in the tripped position.

Evaluator Cue:	**The trip hook and latch lever are not engaged.**

Comment:

OP-137, Section 8.4.1 Initial Conditions
Note prior to initial conditions #2

Performance Step: 2 **NOTE:** Loss of B-SB DC Power is not considered "normal operation" in the following initial condition. If B-SB DC Power has been lost and cannot be restored then the following initial condition does not apply.

Standard: Reads and placekeeps the note

Comment:

OP-137, Section 8.4.1 Initial Conditions #2

Performance Step: 3 During normal operations, the cause of any over speed trip of the turbine-driven AFW pump has been investigated and corrected prior to resuming the operation of the pump.

Standard: Reads and initials initial condition #2

(cause of the over speed trip was identified and corrected which was part of initial conditions provided in the JPM)

Comment:

OP-137 Section 8.4.2 NOTES prior to step 1

Performance Step: 4 NOTE: Attachment 6 diagram may be used as a reference for nomenclature.

NOTE: If any of the following information is changed, Attachment 6 and local pump information should also be changed.

Standard: Operator reads and placekeeps any note or caution

Comment:

OP-137 Section 8.4.2 step 1

Performance Step: 5 Verify the following valves are shut:

- 1MS-70 SA, MAIN STEAM B TO AUX FW TURBINE
- 1MS-72 SB, MAIN STEAM C TO AUX FW TURBINE

Standard: Status provided in Initial Conditions.

Evaluator's Cue:	**If asked: 1MS-70 and 1MS-72 are shut** **(as provided on cue sheet)**

Comment:

OP-137 Section 8.4.2 step 2.a

Performance Step: 6 **IF** DP-1B-SB 125V DC Power is available,

THEN PERFORM the following steps:

CHECK the local red indicating lamp for TURBINE OVERSPEED TRIP is ON

Standard: Verifies that the red lamp is lit for the TURBINE OVERSPEED TRIP on the local control panel.

Evaluator's Cue:	(Lamp is located on Aux Feedwater Control Panel 1X-SAB) **The red TURBINE OVERSPEED TRIP lamp is lit.**

Comment:

OP-137 Section 8.4.2 step 2.b

Performance Step: 7 VERIFY the flat side of the tappet nut is aligned toward the tappet lever.

Standard: Verifies flat side of the tappet nut aligned toward the tappet lever.

Evaluator's Cue:	**The flat side of tappet nut is aligned toward the tappet lever.**

Comment:

OP-137 Section 8.4.2 NOTES prior to step 2.c

Performance Step: 8 NOTE: The next two Steps must be coordinated to ensure proper reset of the Trip and Throttle valve.

NOTE: If the local red indicating lamp for TURBINE OVERSPEED TRIP does not extinguish, it is an indication that one of the limit switches did not reset, and further investigation may be warranted.

Standard: Reads and placekeeps the note

Comment:

OP-137 Section 8.4.2 step 2.c

√ **Performance Step: 9** PULL the connecting rod toward the Trip and Throttle valve until the rod locks in place AND the local red indicating lamp for TURBINE OVERSPEED TRIP is OFF.

Standard: Locates connecting rod and pulls it toward the trip/throttle valve.

Verifies rod locked in place AND the local red indicating lamp for TURBINE OVERSPEED TRIP is OFF.

Evaluator's Cue:	**The connecting rod is locked in place and the red indicating lamp for TURBINE OVERSPEED TRIP is OFF.** (Light is located on Aux Feedwater Control Panel 1AF-E002)

Comment:

OP-137 Section 8.4.2 step 2.d

√ **Performance Step: 10** PRESS DOWN AND HOLD the tappet nut in the fully seated position while releasing the connecting rod.

Standard: Presses down and holds the tappet nut in the fully seated position until the connecting rod is released.

Evaluator's Cue:	**The tappet remains fully seated and the connecting rod is locked in place.**

Comment:

OP-137 Section 8.4.2 step 2.e

Performance Step: 11 VERIFY the Trip and Throttle valve operator in the shut position by observing the T & T VALVE OPERATOR CLOSED light on the Aux Feedwater Control Panel 1AF-E002.

Standard: Verifies trip/throttle valve operator is shut by observing indicating lights on local panel 1AF-E002.

Evaluator's Cue:	**The green shut light is ON and the red open light is OFF.** **(If necessary: Valve stem indication is at the shut position.)**

Comment:

OP-137 Section 8.4.2 step 2.f

Performance Step: 12 VERIFY the flat side of the tappet nut is against the tappet lever and fully seated.

Standard: Verifies flat side of the tappet nut against the tappet lever and fully seated.

Evaluator's Cue:	**The flat side of tappet nut is against the tappet lever and fully seated.**

Comment:

OP-137 Section 8.4.2 step 2.g

Performance Step: 13 VERIFY the latch lever is being held up by the trip hook.

Standard: Verifies latch lever is being held up by the trip hook.

Evaluator's Cue:	**The latch is being held up by the trip hook.**

Comment:

OP-137 Section 8.4.2 step 2.h

Performance Step: 14 VERIFY the TURBINE OVERSPEED TRIP light is extinguished on the AFW Control Panel 1AF-E002

Standard: Verifies TURBINE OVERSPEED TRIP light status on Panel 1AF-E002.

Evaluator's Cue:	**The TURBINE OVERSPEED TRIP light is extinguished.**

Comment:

OP-137 Section 8.4.2 step 2.i

Performance Step: 15 Notify the Control Room that the mechanical over speed linkage is reset and inform them they can now open the Trip and Throttle valve.

Standard: Simulates notifying the Control Room.

Evaluator's Cue:	**Acknowledge report.** **END OF JPM**

Comment:

STOP TIME: _______________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam In-Plant JPM j

Reset the Turbine-Driven AFW Pump Mechanical Overspeed
In accordance with OP-137, Auxiliary Feedwater 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: ______________

JPM CUE SHEET

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 Unit was manually tripped from 100% power due to a loss of the ‘A’ MFW pump. • The Turbine-driven AFW pump is needed for plant cooldown but the pump tripped on overspeed. • The cause of the overspeed trip has been identified and corrected. • Main Steam isolation valves 1MS-70 and 1MS-72 are shut.

Initiating Cue:	• The CRS has directed you to reset the Turbine-driven AFW pump mechanical overspeed trip linkage in accordance with OP-137, Auxiliary Feedwater System, Section 8.4. • **Assume that the Mechanical Overspeed Trip Linkage is currently in the tripped position.** • The Trip and Throttle Valve will be reopened from the Control Room. • All Initial Conditions are met..

8.4. Resetting the Turbine-Driven AFW Pump Mechanical Over Speed Trip Linkage

8.4.1. Initial Conditions

1. Mechanical Over speed Trip Linkage in the tripped position. ______

> **NOTE:** Loss of B-SB DC Power is not considered "normal operation" in the following initial condition. If B-SB DC Power has been lost and cannot be restored then the following initial condition does not apply.

2. During normal operations, the cause of any over speed trip of the turbine-driven AFW pump has been investigated and corrected prior to resuming the operation of the pump. ______

8.4.2. Procedural Steps

> **NOTE:** Attachment 6 diagram may be used as a reference for nomenclature.
>
> **NOTE:** If any of the following information is changed, Attachment 6 and local pump information should also be changed.

1. **VERIFY** the following valves are shut:
   - 1MS-70 SA, MAIN STEAM B TO AUX FW TURBINE ______
   - 1MS-72 SB, MAIN STEAM C TO AUX FW TURBINE ______
2. **IF** DP-1B-SB 125V DC Power is available,
   THEN PERFORM the following steps:
   a. **CHECK** the local red indicating lamp for AFW TURBINE MECH O/S TRIP is ON. ______
   b. **VERIFY** the flat side of the tappet nut is aligned toward the tappet lever. ______

8.4.2 Procedural Steps (continued)

NOTE: The next two Steps must be coordinated to ensure proper reset of the Trip and Throttle valve.

NOTE: If the local red indicating lamp for AFW TURBINE MECH O/S TRIP does not extinguish, it is an indication that one of the limit switches did not reset, and further investigation may be warranted.

c. **PULL** the connecting rod toward the Trip and Throttle valve until the rod locks in place **AND** the local red indicating lamp for AFW TURBINE MECH O/S TRIP is OFF. ______

d. **PRESS DOWN AND HOLD** the tappet nut in the fully seated position while releasing the connecting rod. ______

e. **VERIFY** the Trip and Throttle valve operator in the shut position by observing the T & T VALVE OPER CLOSED light on the Turbine Driven Auxiliary Feedwater Control Panel 1AF-E002. ______

f. **VERIFY** the flat side of the tappet nut is against the tappet lever and fully seated. ______

g. **VERIFY** the latch lever is being held up by the trip hook. ______

h. **VERIFY** the AFW TURBINE MECH O/S TRIP light is extinguished on the Turbine Driven Auxiliary Feedwater Control Panel 1AF-E002. ______

i. **OPEN** the Trip and Throttle valve from the MCB. ______

3. **IF** DP-1B-SB 125V DC Power is NOT available,
THEN PERFORM the following steps:

a. **ENGAGE** the TDAFW Trip and Throttle Valve manual operator. ______

b. **ROTATE** the hand-wheel in the SHUT direction until the Latch Lever is in the normal position (angled up). ______

c. **VERIFY** the flat side of the tappet nut is aligned toward the tappet lever. ______

8.4.2 Procedural Steps (continued)

NOTE: The next two Steps must be coordinated to ensure proper reset of the Trip and Throttle valve.

d. **PULL** the connecting rod toward the Trip and Throttle valve until the rod locks in place. ______

e. **PRESS DOWN AND HOLD** the tappet nut in the fully seated position while releasing the connecting rod. ______

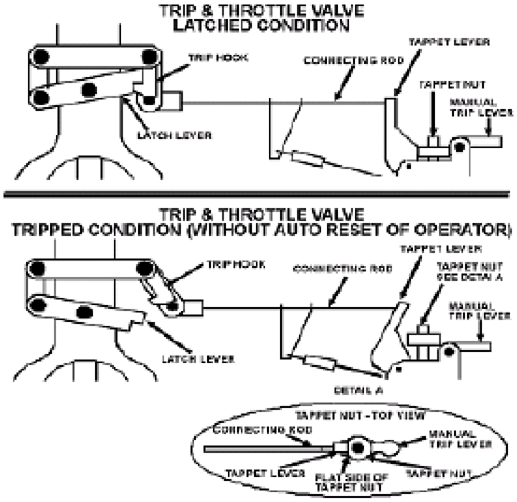
f. **VERIFY** the flat side of the tappet nut is against the tappet lever and fully seated. ______

g. **VERIFY** the latch lever is being held up by the trip hook. ______

4. **IF** TDAFW pump operation is desired,
THEN GO TO Section 5.5 or 8.7. ______

Attachment 6 - Resetting the TDAFW Pump Mechanical Overspeed Trip Linkage

Sheet 1 of 2



<u>DP-1B-SB 125V DC Power Available</u>

1. **Verify** shut 1MS-70 and 1MS-72. ______
2. **Check** the local red indicating lamp for AFW TURBINE MECH O/S TRIP is ON. ______
3. **Verify** the flat side of the tappet nut is aligned towards the tappet lever. ______

> **NOTE:** The next two Steps must be coordinated to ensure proper reset of the Trip and Throttle Valve.
>
> **NOTE:** If the local red indicating lamp for AFW TURBINE MECH O/S TRIP does not extinguish, it is an indication that one of the limit switches did not reset, and further investigation may be warranted.

4. **Pull** the connecting rod toward the Trip and Throttle valve until the rod locks in place and the local red indicating lamp for AFW TURBINE MECH O/S TRIP is OFF. ______
5. **Press down and hold** the tappet nut in the fully seated position while releasing the connecting rod. ______
6. **Verify** the Trip and Throttle valve operator in the shut position by observing the T & T VALVE OPER CLOSED light on the Turbine Driven Auxiliary Feedwater Control Panel 1AF-E002. ______
7. **Verify** the flat side of the tappet nut is against the tappet lever and fully seated. ______
8. **Verify** the latch lever is being held up by the trip hook. ______
9. **Verify** the AFW TURBINE MECH O/S TRIP light is extinguished on the Turbine Driven Auxiliary Feedwater Control Panel 1AF-E002. ______
10. **Open** the Trip and Throttle Valve from the MCB. ______

Facility: Harris Nuclear Plant

Task No.: 012010H101

Task Title: Perform Local Actions For Placing an OTΔT Channel In TEST

JPM No.: 2018 NRC Exam In-plant JPM k

K/A Reference: 012 A4.04 RO 3.3 SRO 3.6

ALTERNATE PATH - NO

Examinee: ______________________

NRC Examiner: ________________

Facility Evaluator: ______________________

Date: ________

Method of testing:

Simulated Performance: ______

Actual Performance: X

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 unit is operating at 100% power when Loop 1 Hot Leg temperature input to Tavg and OTΔT failed low.

Initiating Cue:	To meet Technical Specifications, the CRS is directing you to perform the local actions of OWP-RP-01 for troubleshooting and tripping bi-stables for Loop 1 Tavg and OTΔT. Inform the Control Room when all switches have been positioned to allow the Control Room to complete the actions required in the Control Room. The CRS informs you that all Master Test Switches are to be placed in test for troubleshooting. The Control Room has placed Rod Control in MANUAL.

Evaluator:	**Provide candidate with a copy of the procedure now to review prior to getting to location of actions performed in this JPM**.

Task Standard:	Place the PIC Cabinet Master Test switches and bistables in the Test position.
Required Materials:	None
General References:	OWP-RP-01, Reactor Protection Rev 17
Time Critical Task:	No
Validation Time:	15 minutes

SIMULATOR SETUP

- N/A Evaluation will be performed by Simulating in plant activities.
- Cues will be provided to the candidate by the examiner.

Critical Step Justification	
Step 6	Must locate then place Master Test switch SW1 to proper position to perform testing and troubleshooting
Step 7	Must locate then place Master Test switch SW2 to proper position to perform testing and troubleshooting
Step 8	Must locate then place Master Test switch SW4 to proper position to perform testing and troubleshooting
Step 9	Must locate then place Master Test switch SW5 to proper position to perform testing and troubleshooting
Step 10	Must locate then place switch for BS1 to proper position to perform testing and troubleshooting
Step 11	Must locate then place switch for BS2 to proper position to perform testing and troubleshooting
Step 12	Must locate then place switch for BS3 to proper position to perform testing and troubleshooting
Step 13	Must locate then place switch for BS1 to proper position to perform testing and troubleshooting
Step 14	Must locate then place switch for BS2 to proper position to perform testing and troubleshooting
Step 15	Must locate then place switch for BS3 to proper position to perform testing and troubleshooting
Step 16	Must locate then place switch for BS4 to proper position to perform testing and troubleshooting

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 Core 4 checks.**

Start time begins when the candidate is at the PIC cabinet 1 and after the Take a Minute concludes

START TIME: ____________

Performance Step: 1 **OBTAIN PROCEDURE**

Standard: Reviews provided OWP-RP-01 and refers to Section for Tavg/ΔT Channel I

Comment:

OWP-RP-01, Step 6

Performance Step: 2 PRECAUTION: To prevent a Reactor Trip, prior to removing a channel from service, verify the corresponding Trip Status lights for the other channels are de-energized.

Standard: Reviews precaution

Evaluator Cue:	**The corresponding Trip Status lights for the other channels are de-energized.**

Comment:

OWP-RP-01 TAVG/ΔT Protection Channel I - On MCB

Performance Step: 3 NOTE: The Rod Bank Selector should be restored last.

** For the purposes of this OWP, MAN can be any position on the Rod Bank Selector Switch except AUTO.

Standard: Reviews note

Comment:

OWP-RP-01

Performance Step: 4 On Main Control Board Place the Rod Bank Selector to MAN

Standard: Per initial conditions provided in this JPM the Rod Bank Selector is in MAN.

Evaluator Cue:	**If asked or if the candidate is heading for the Control room ask their intentions then state that the Control room reports rod bank selector is in manual.**

Comment:

Evaluator Note:	**There is a PIC room layout drawing in the PIC room that indicates where each cabinet is located. Additionally, the cabinet doors are hinged to open to the right. Since it is very difficult to see what the candidate is performing photos of the cabinet layout and close ups of the individual cards are included. If unable to view the candidates activities use the photos and ask them to show you which switches and how the switches will be manipulated.**

OWP-RP-01

Performance Step: 5 NOTE: Master Test switches may be positioned to TEST for troubleshooting. They are not required to be in TEST to meet Tech Specs. Operating these switches before operating the bistable switches aids in troubleshooting by maintaining system conditions the same as they were when the trouble occurred.

Concurrent verification is preferred while tripping bistable.

Standard: Reviews note and Initiating Cue to determine that Master Test Switches are to be placed in TEST for troubleshooting

Evaluator Cue:	**For this JPM only, assume that concurrent verification is being performed and that verifier agrees with all actions you have taken.**

Comment:

OWP-RP-01

✓ **Performance Step: 6** In PIC 1 on Card C1-861: SW1 (TS/412F) Master Test Switch for TS/412D in TEST

Standard: Locates Card C1-861 and places SW1 in TEST position (UP)

Evaluator Cue: **SW1 IS IN THE UP - TEST POSITION.**
(Toggle switch 1 of 7 on Card C1-861 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 7** In PIC 1 on Card C1-861: SW2 (TS/412G) Master Test Switch for TS/412B1 in TEST

Standard: Locates Card C1-861 and places SW2 in TEST position (UP)

Evaluator Cue: **SW2 IS IN THE UP - TEST POSITION.**
(Toggle switch 2 of 7 on Card C1-861 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 8** In PIC 1 on Card C1-863: SW4 (TS/412R) Master Test Switch for TS/412B2 in TEST

Standard: Locates Card C1-863 and places SW4 in TEST position (UP)

Evaluator Cue:	**SW4 IS IN THE UP - TEST POSITION.** (Toggle switch 4 of 7 on Card C1-863 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 9** In PIC 1 on Card C1-863: SW5 (TS/412S) Master Test Switch for TS/412B3

Standard: Locates Card C1-863 and places SW5 in TEST position (UP)

Evaluator Cue:	**SW5 IS IN THE UP - TEST POSITION.** (Toggle switch 5 of 7 on Card C1-863 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 10** In PIC 1 on Card C1-821: BS1 (TB/412D1 Low Tavg) in TEST

Standard: Locates Card C1-821 and places BS1 in TEST position (UP)

Evaluator Cue: **BS1 IS IN THE UP - TEST POSITION and the red test light is lit.**
(Toggle switch 1 of 3 on Card C1-821 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 11** In PIC 1 on Card C1-821: BS2 (TB/412D2 High Tavg) in TEST

Standard: Locates Card C1-821 and places BS2 in TEST position (UP)

Evaluator Cue: **BS2 IS IN THE UP - TEST POSITION and the red test light is lit.**
(Toggle switch 2 of 3 on Card C1-821 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 12** In PIC 1 on Card C1-821: BS3 (TB/412E Low Low Tavg) in TEST

Standard: Locates Card C1-821 and places BS3 in TEST position (UP)

Evaluator Cue:	**BS3 IS IN THE UP - TEST POSITION and the red test light is lit.** (Toggle switch 3 of 3 on Card C1-821 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 13** In PIC 1 on Card C1-822: BS1 (TB/412B1 OP∆T) in TEST

Standard: Locates Card C1-822 and places BS1 in TEST position (UP)

Evaluator Cue:	**BS1 IS IN THE UP - TEST POSITION and the red test light is lit.** (Toggle switch 1 of 4 on Card C1- 822 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 14** In PIC 1 on Card C1-822: BS2 (TB/412B2 OPΔT C-4) in TEST

Standard: Locates Card C1-822 and places BS2 in TEST position (UP)

Evaluator Cue:	**BS2 IS IN THE UP - TEST POSITION and the red test light is lit.** (Toggle switch 2 of 4 on Card C1-822 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 15** In PIC 1 on Card C1-822: BS3 (TB/412C1 OTΔT) in TEST

Standard: Locates Card C1-822 and places BS3 in TEST position (UP)

Evaluator Cue:	**BS3 IS IN THE UP - TEST POSITION and the red test light is lit.** (Toggle switch 3 of 4 on Card C1-822 image from top to bottom)

Comment:

OWP-RP-01

✓ **Performance Step: 16** In PIC 1 on Card C1-822: BS4 (TB/412C2 OTΔT C-3) in TEST

Standard: Locates Card C1-822 and places BS4 in TEST position (UP)

Evaluator Cue:	**BS4 IS IN THE UP - TEST POSITION and the red test light is lit.** (Toggle switch 4 of 4 on Card C1-822 image from top to bottom)

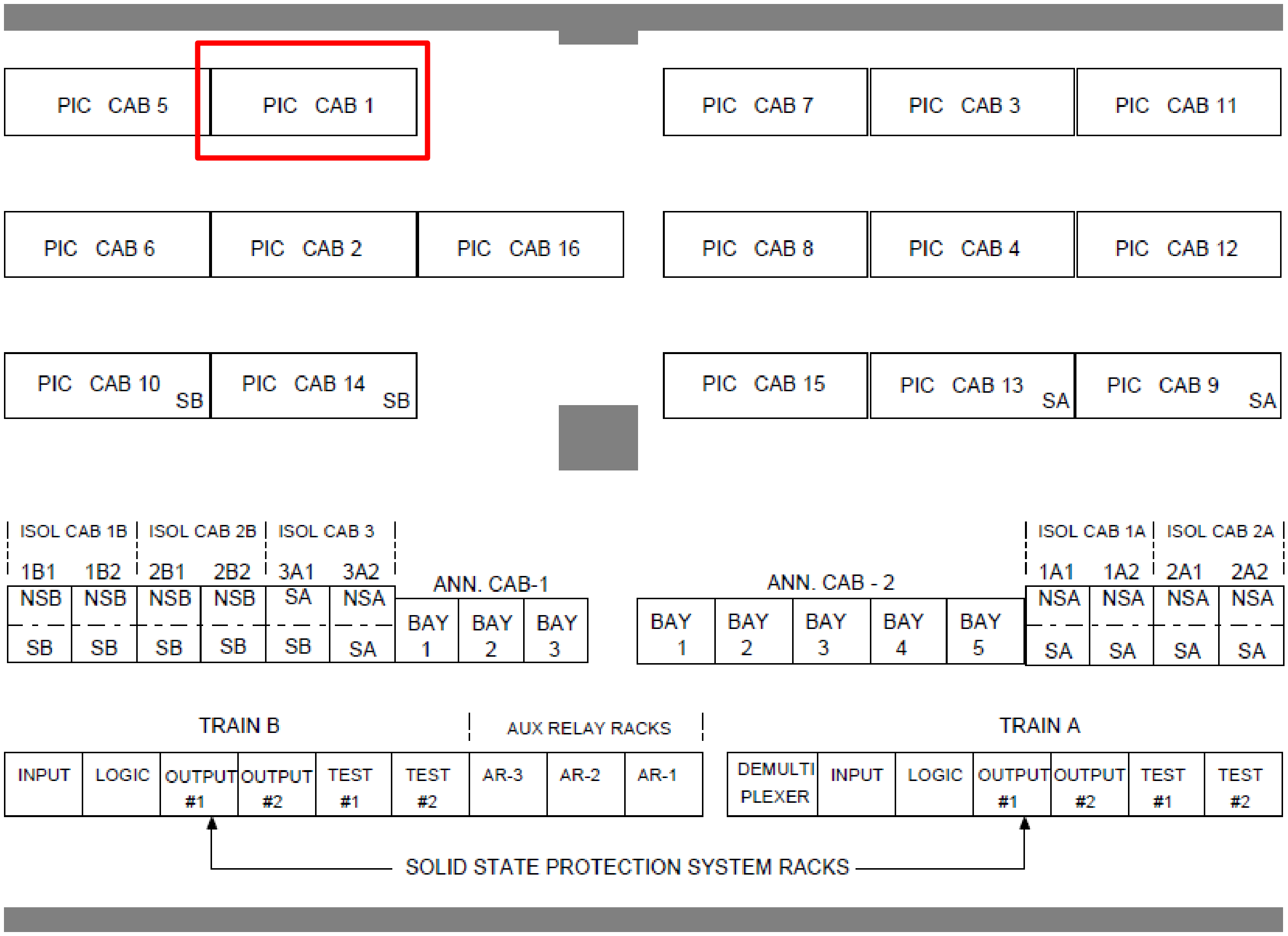
Standard: Reports to or contacts MCR to inform them that the test switches have been positioned IAW OWP-RP-01 for Tavg/ΔT Protection Channel I

Evaluator Cue:	**MCR acknowledges completion of the OWP section. MCR will verify correct bi-stables and complete OWP.** **After communications are complete announce: END OF JPM** **NOTE: Prior to leaving the area ensure any cabinets opened during the performance of this JPM are properly secured.**

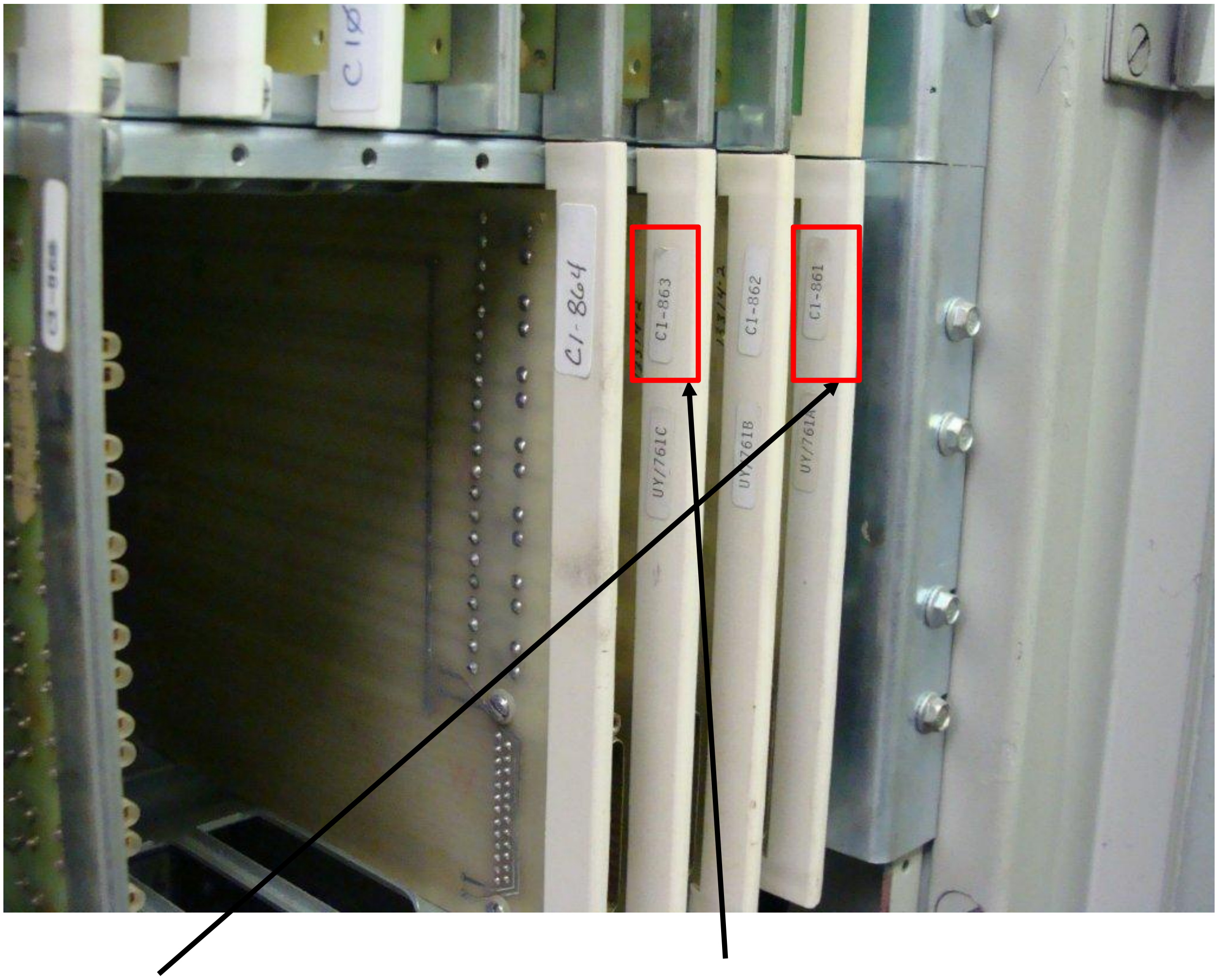
Comment:

STOP TIME: ______________

KEY


PIC CAB 5
PIC CAB 1
PIC CAB 7
PIC CAB 3
PIC CAB 11
PIC CAB 6
PIC CAB 2
PIC CAB 16
PIC CAB 8
PIC CAB 4
PIC CAB 12
PIC CAB 10 SB
PIC CAB 14 SB
PIC CAB 15
PIC CAB 13 SA
PIC CAB 9 SA
ISOL CAB 1B
ISOL CAB 2B
ISOL CAB 3
1B1
1B2
2B1
2B2
3A1
3A2
NSB
SA
NSA
SB
ANN. CAB-1
BAY 1
BAY 2
BAY 3
ANN. CAB - 2
BAY 1
BAY 2
BAY 3
BAY 4
BAY 5
ISOL CAB 1A
ISOL CAB 2A
1A1
1A2
2A1
2A2
NSA
SA
TRAIN B
AUX RELAY RACKS
TRAIN A
INPUT
LOGIC
OUTPUT #1
OUTPUT #2
TEST #1
TEST #2
AR-3
AR-2
AR-1
DEMULTI PLEXER
INPUT
LOGIC
OUTPUT #1
OUTPUT #2
TEST #1
TEST #2
SOLID STATE PROTECTION SYSTEM RACKS

KEY



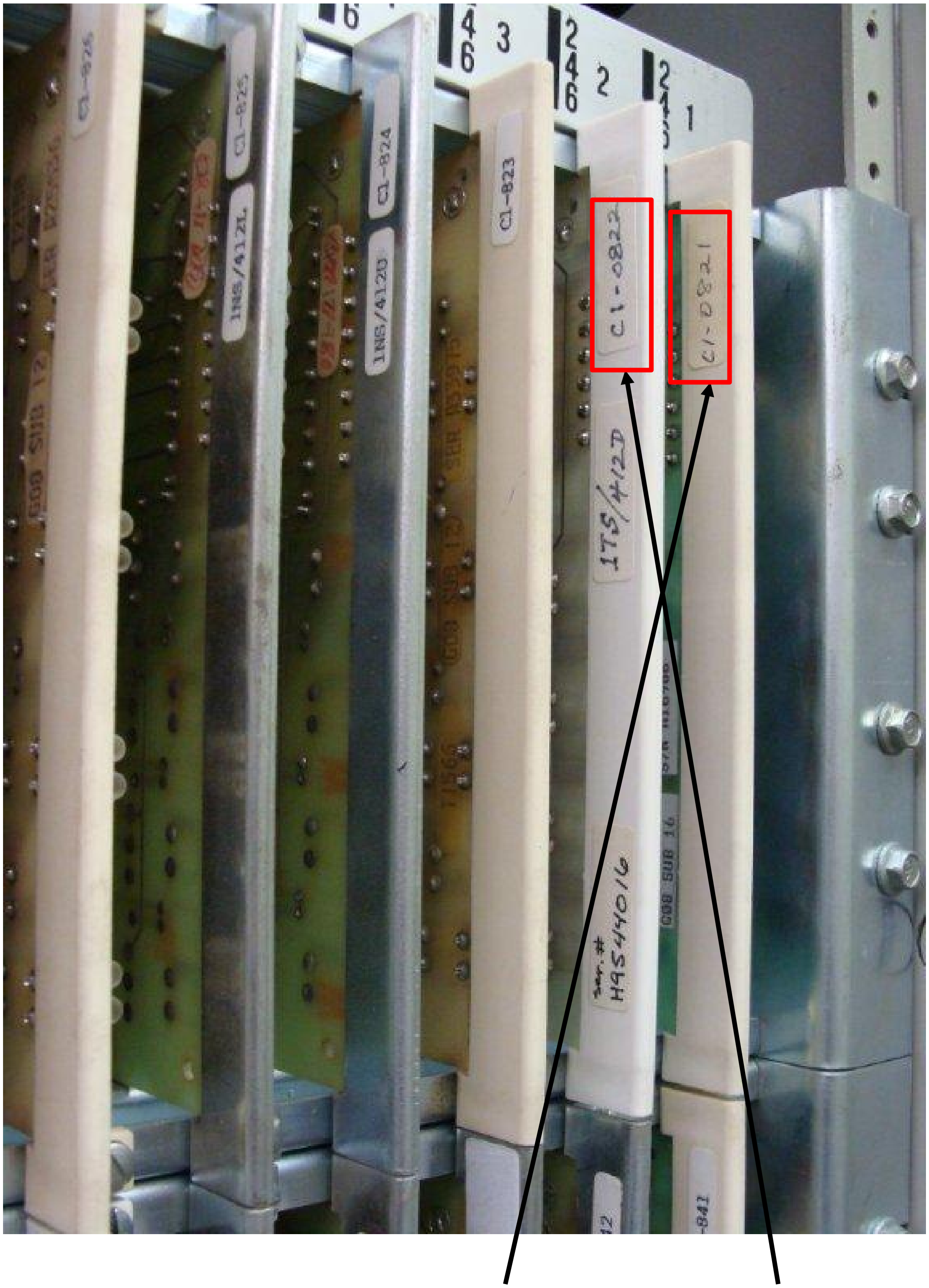
Card C1-861 (far right card) Card C1-863 (third from right)

In PIC 1, this where the label locations are at for the cards and these are the 2 cards they should be manipulating switches on

KEY

On the other side of the cards are rows of switches.
The switches are numbered SW1 – SW7 from top to bottom.
UP is TEST, DOWN is NORMAL

KEY



The above cards are Bi-stable cards C1-0821 (far right) and C1-0822

KEY

Opposite side of cards C1-0821 and C1-0822 are the Bi-stable switches. The switches are labeled 1-4 from top to bottom UP is TEST, DOWN is NORMAL
When in TEST the RED light above the associated switch will light.

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam In-plant JPM k

Perform Local Actions For Placing an OT∆T Channel In TEST
(In accordance with OWP-RP-01, Reactor Protection)

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:	The unit is operating at 100% power when Loop 1 Hot Leg temperature input to Tavg and OTΔT failed low.

Initiating Cue:	To meet Technical Specifications, the CRS is directing you to perform the local actions of OWP-RP-01 for troubleshooting and tripping bi-stables for Loop 1 Tavg and OTΔT. Inform the Control Room when all switches have been positioned to allow the Control Room to complete the actions required in the Control Room. The CRS informs you that all Master Test Switches are to be placed in test for troubleshooting. The Control Room has placed Rod Control in MANUAL.


C1-864
C1-863
UY/761C
C1-862
UY/761B
C1-861
UY/761A


TEST
BS 4
NORMAL
2840A14G01
INMT 2
INMT 3
TEST
SW 1
NORMAL
TEST
SW 2
NORMAL
TEST
SW 3
NORMAL
TEST
SW 4
NORMAL
TEST
SW 5
NORMAL
TEST
SW 6
NORMAL
TEST
SW 7
NORMAL


C1-825
C1-825
INS/412L
C1-824
INS/412D
C1-823
C1-0822
ITS/412D
ser.#
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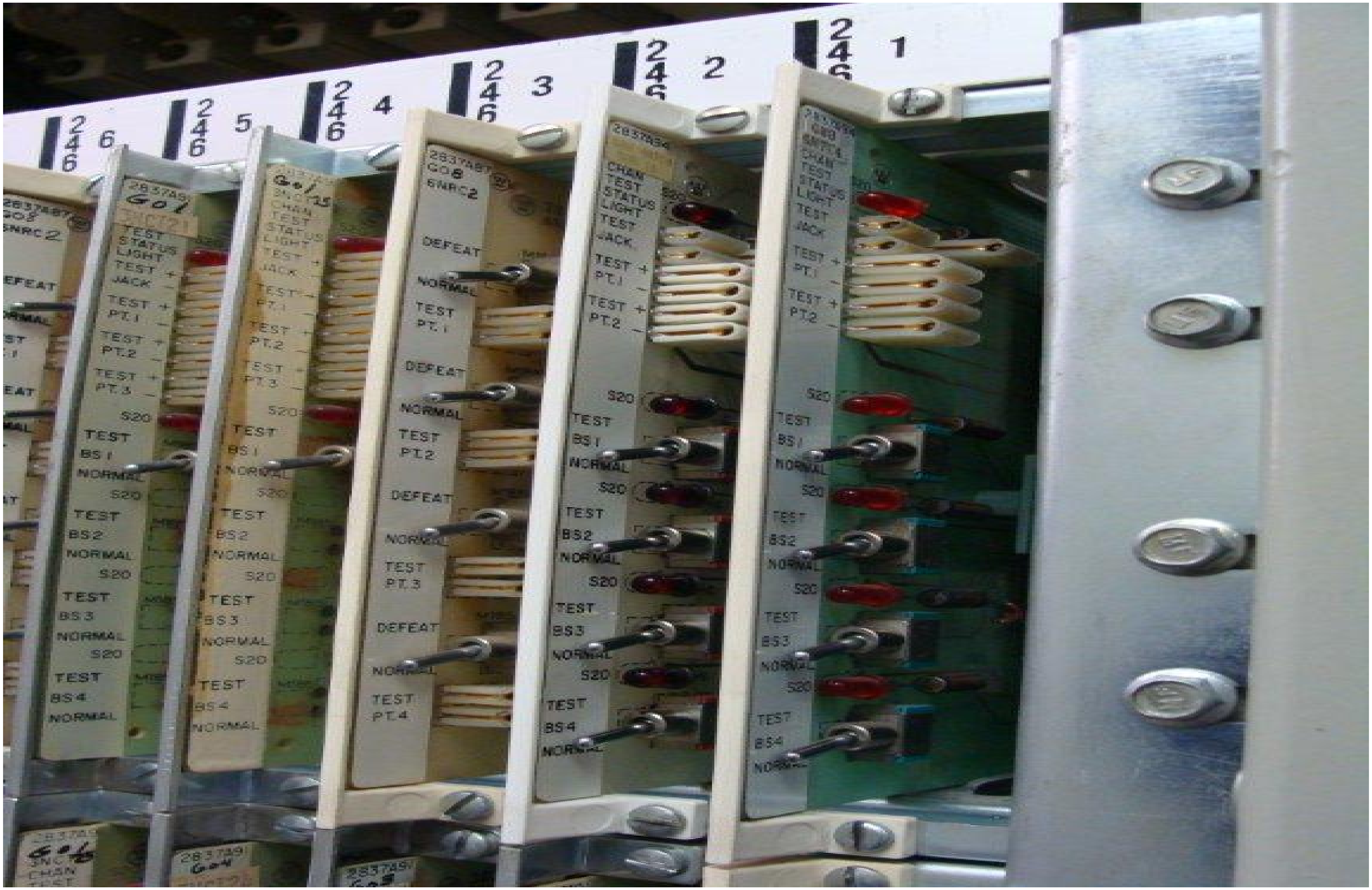
4
6
1
G08 SUB 16
-841



Facility: Harris Nuclear Plant
Task No.: 001004H101

Task Title: Perform A Manual Shutdown Margin Calculation
JPM No.: 2018 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 has been operating at 92% power for 2 weeks. OST-1005, Control Rod and Rod Position Indicator Exercise Quarterly Interval Modes 1 – 3, was just performed. One rod in Control Bank 'B' was determined to be immovable/stuck. • Core burnup is 350 EFPD • RCS boron concentration is 600 ppm • POWERTRAX is **NOT** available

Initiating Cue:	The CRS has entered Tech Spec 3.1.1.1 and has directed you to complete OST-1036, Shutdown Margin Calculation Modes 1-5, Section 7.3, Manual SDM Calculation (Modes 1 and 2) for current plant conditions. **NOTE: For this JPM notify evaluator when independent verification is required to be performed.**

Task Standard:	OST-1036, Attachment 3, Manual SDM Calculation (Modes 1 and 2), completed with SDM of 2410 ± 100 pcm **(tolerance based on total of curves used and their division readability)**
Required Materials:	Calculator, ruler
General References:	OST-1036, Shutdown Margin Calculation Modes 1-5 (Rev. 53) Curve Book (Cycle 21)
Handouts:	OST-1036, Shutdown Margin Calculation Modes 1-5 (Rev. 53), pg 21, 26 and 27
Time Critical Task:	No
Validation Time:	20 minutes

Critical Step Justification	
Step 4	Must determine correct rod insertion limit based on curve value. The number of rod steps will be an input to the calculation.
Step 7	Must determine correct power defect based on curve value. The power defect will provide one of the inputs to the calculation.
Step 8	Must determine the correct rod worth based on curve value. The rod worth will provide one of the inputs to the calculation.
Step 10	The total shutdown margin was the task that the CRS directed applicant to perform.

START TIME: ____________

OST-1036

Performance Step: 1 OBTAIN PROCEDURE

Standard: Reviews Procedure

Evaluator Cue:	**Provide OST-1036 Section 7.3 and Attachment 3.**

Comment:

Evaluator Note:	**NOTE: The curve numbers provided in this JPM are numbers from the 2018 NRC Exam Frozen Procedures Curve Book folder.**

OST-1036 Section 7.3.1

Performance Step: 2 Enter the absolute value for each parameter on Attachment 3.

Standard: Reviews Attachment 3 and determines value for each parameter.

Comment:

OST-1036 Attachment 3 Step 1

Performance Step: 3 Enters Reactor Power Level

Standard: Refers to given conditions and enters 92%

Comment:

OST-1036 Attachment 3 Step 2

✓ **Performance Step: 4** Determine Rod Insertion Limit for power level

Standard: Refers to Curve F-21-1 and determines TS limit for RIL to be 171 steps (166 – 176 steps, tolerance based on curve division readability)

Comment:

OST-1036 Attachment 3 Step 3

Performance Step: 5 Enters core Burn Up

Standard: Refers to given conditions and enters 350 EFPD

Comment:

OST-1036 Attachment 3 Step 4

Performance Step: 6 Enters RCS Boron Concentration

Standard: Refers to initial conditions and enters 600 ppm

Comment:

Evaluator Note:	**ATT 3, STEP 5 NOT PERFORMED SINCE VALUE IS INCLUDED AS PART OF ATTACHMENT.**

OST-1036 Attachment 3 Step 6

✓ **Performance Step: 7** Determines Power Defect for current power level

Standard: Refers to Curve C-21-3 and determines power defect to be 2560 ± 50 pcm

(tolerance based on curve division readability)

Comment:

OST-1036 Attachment 3 Step 7

✓ **Performance Step: 8** Determines Rod Worth for RIL position determined above

Standard: Refers to Curve A-21-11 and determines rod worth to be 400 ± 50 pcm

(tolerance based on rod position tolerance from performance step 4 curve division readability)

Comment:

OST-1036 Attachment 3 Step 8

Performance Step: 9 Enters worth of any additional immovable or untrippable rods

Standard: Refers to given conditions and enters 1724 pcm

Comment:

OST-1036 Section 7.3.2 Attachment 3 Step 9

✓ **Performance Step: 10** Determines Total Shutdown Margin:

Perform the calculation listed on Attachment 3 Item 9 for the required SDM boron concentration for the projected conditions.

Standard: Refers to Attachment 3 Item 9 to document SDM

Determines Total Shutdown Margin to be **2410 ± 75** pcm (tolerance based on total of all curves used and their division readability)

Comment:

OST-1036 Section 7.3.3

Performance Step: 11 Perform an independent verification of Attachment 3..

Standard: Contacts evaluator to perform independent verification per initial conditions

Comment:

Evaluator Cue:	**When independent verification of OST-1036, Attachment 3, Manual SDM Calculation is requested.** **END OF JPM**

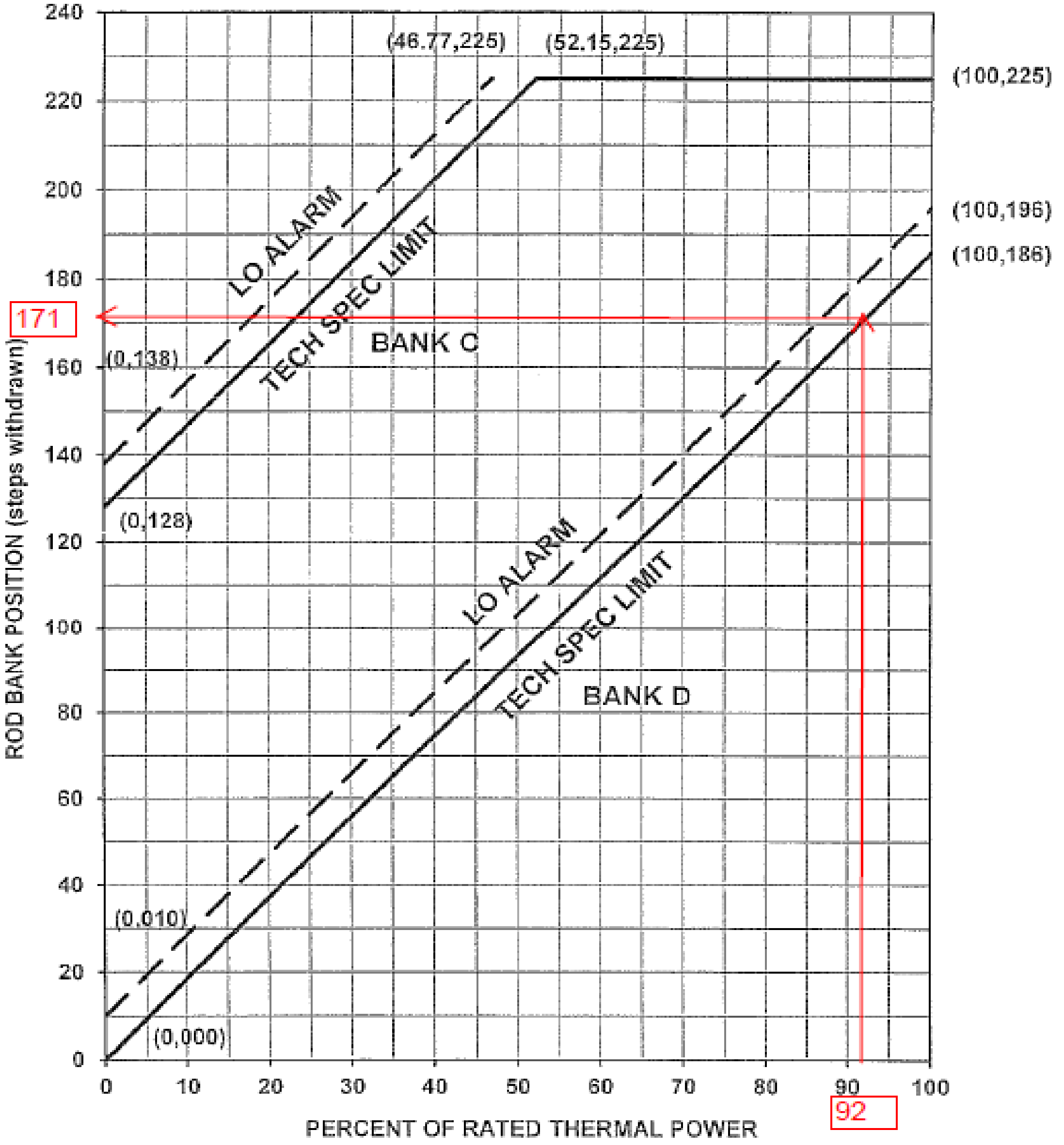
STOP TIME: ____________

KEY

HNP Cycle 21
Curvebook
Operator Curves

HNP-F/NFSA-0269
Appendix A, Rev 0
Page A25 of A25

HARRIS UNIT 1 CYCLE 21
ROD INSERTION LIMITS



CURVE NO.	F-21-1	REV NO.	0
ORIGINATOR	JOSHUA CHEN	DATE	10/05/2016
SUPERVISOR	NOAH WHITE	DATE	10/11/2016
SHIFT MANAGER	TIM ENGLISH	DATE	10/11/16

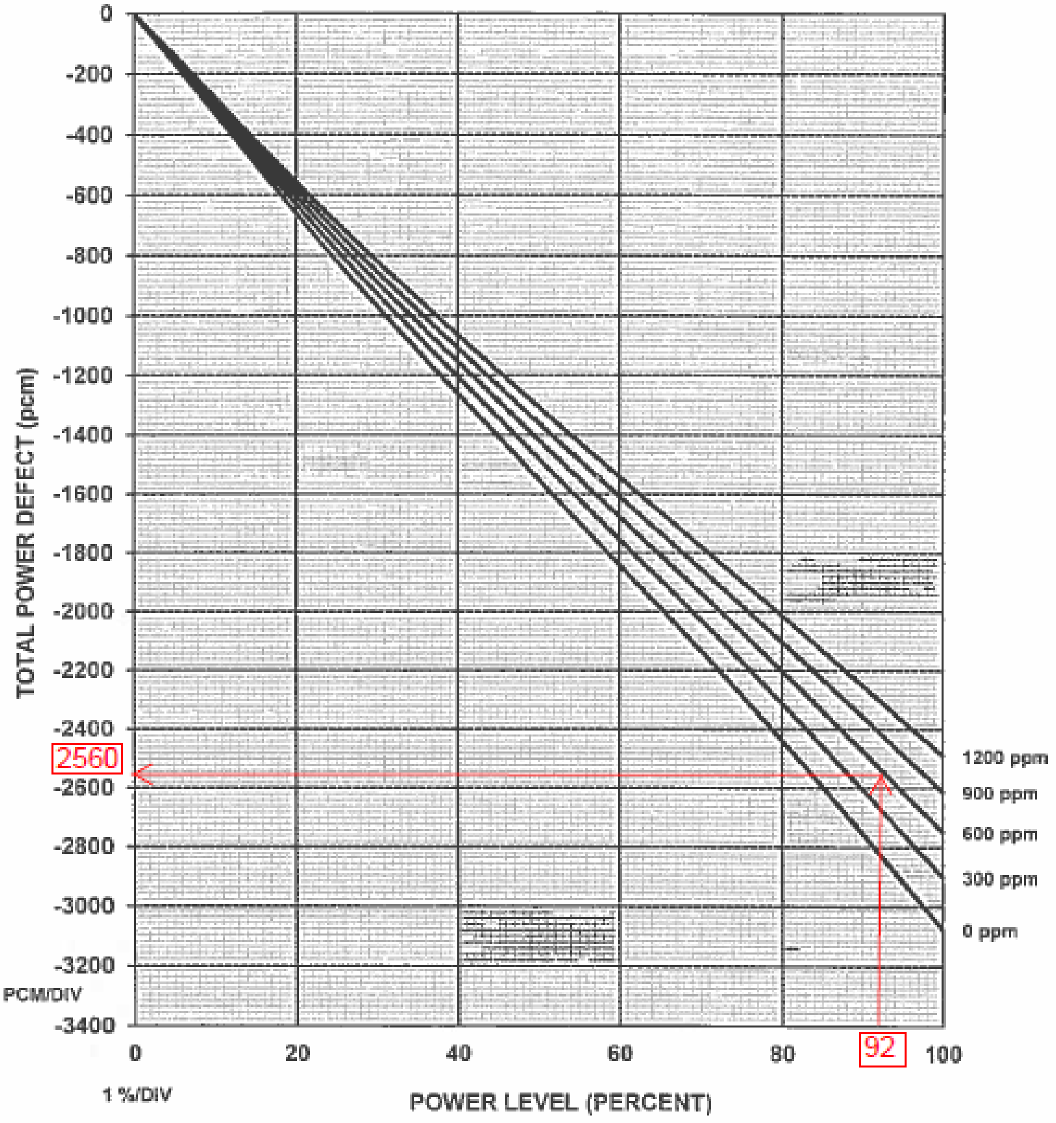
KEY

HNP Cycle 21
Curvebook
Operator Curves

HNP-F/NFSA-0269
Appendix A, Rev 0
Page A24 of A25

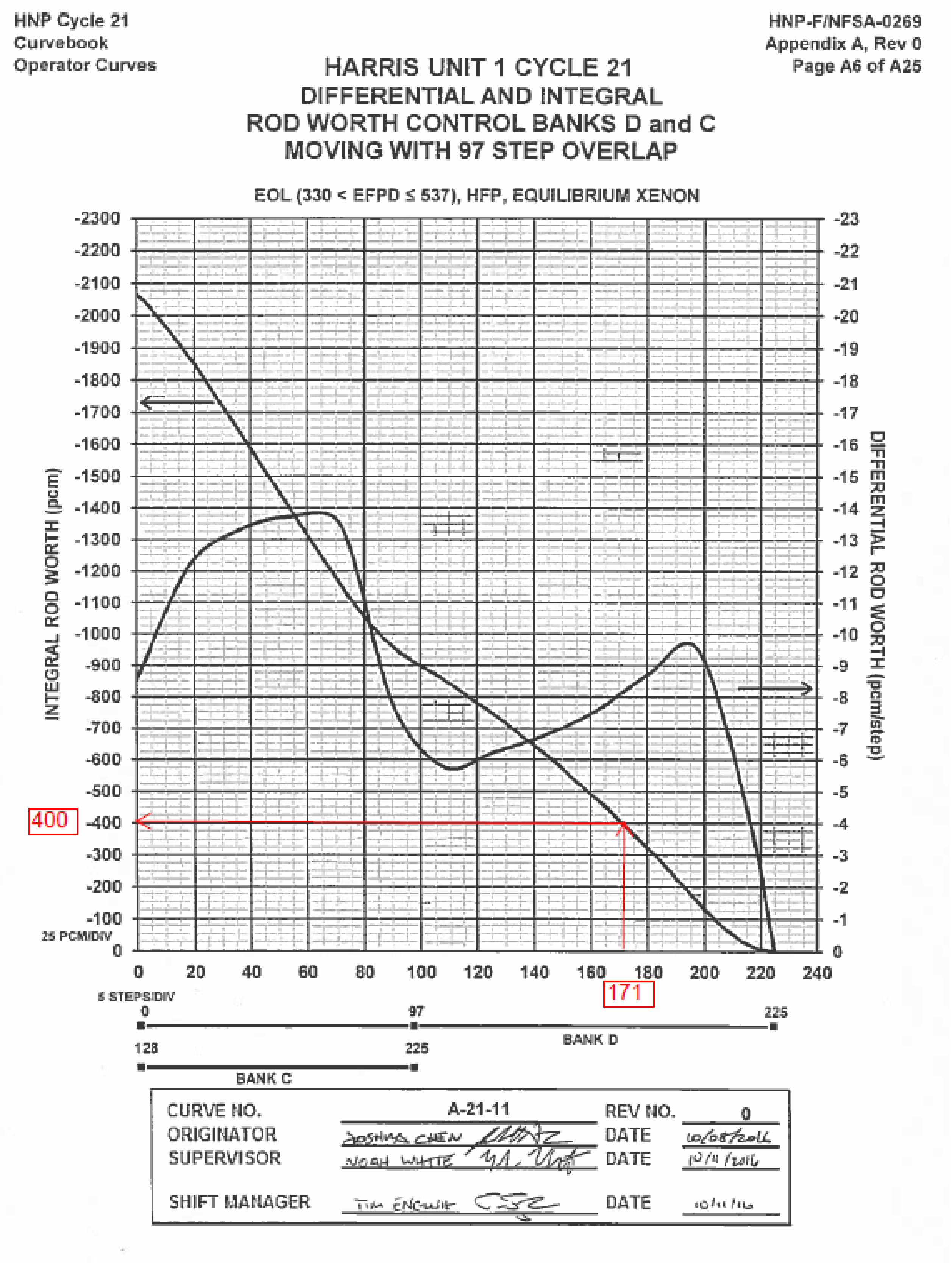
HARRIS UNIT 1 CYCLE 21 POWER DEFECT vs. POWER LEVEL for VARIOUS BORON CONCENTRATIONS

EOL (330 < EFPD ≤ 537)



CURVE NO.	C-21-3	REV NO.	0
ORIGINATOR	JOSHUA CHEN	DATE	10/08/2016
SUPERVISOR	NOAH WHITE	DATE	10/11/2016
SHIFT MANAGER	TIM ENGLISH	DATE	10/11/16

KEY



HNP Cycle 21
Curvebook
Operator Curves

HNP-F/NFSA-0269
Appendix A, Rev 0
Page A6 of A25

HARRIS UNIT 1 CYCLE 21
DIFFERENTIAL AND INTEGRAL ROD WORTH CONTROL BANKS D and C MOVING WITH 97 STEP OVERLAP

EOL (330 < EFPD ≤ 537), HFP, EQUILIBRIUM XENON

CURVE NO.	A-21-11	REV NO.	0
ORIGINATOR	JOSHUA CHEN	DATE	10/08/2016
SUPERVISOR	NOAH WHITE	DATE	10/4/2016
SHIFT MANAGER	TIM ENGWIR	DATE	10/11/16

EXAMINER CALCULATION KEY

(SHADED AREA BELOW INDICATES DATA ALREADY PROVIDED)

Manual SDM Calculation (Modes 1 and 2)

1. Reactor power level. **92** %

2. Rod insertion limit for the above power level

   171 (166-176) steps on bank **D**

3. Burn up (POWERTRAX/MCR Status Board). **350** EFPD

4. Present RCS Boron Concentration **600** ppm

NOTE: Use absolute values of numbers obtained from curves.

5. Total worth of all control and shutdown banks, minus the worth of the most reactive rod for Fuel Cycle 21.

   7054 pcm
   (a)

6. Cycle 19 Power defect for the power level recorded in Step 1. (Refer to Curves C-X-1 to C-X-3).

   Curve used **C-21-3** **2560 ± 50** pcm
   (b)

NOTE: HFP curves are used for power levels of 10% or greater.

7. Inserted control rod worth at the rod insertion limit recorded in Step 2. (Refer to Curves A-X-6 to A-X-11)

   Curve used **A-21-11** **400 ± 50** pcm
   (c)

8. Worth of any additional immovable or untrippable rods (for each stuck rod, use the most reactive single rod worth (1724 pcm).

   1724 pcm
   (d)

9. Determine the Total Shutdown Margin using the following formula:

Total SDM C_B =	7054	-	2520 ± 50	-	400 ± 50	-	1724
(e)	(a)		(b)		(c)		(d)

2410 ± 100 pcm
(p)

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Admin JPM RO A1-1
Perform A Manual Shutdown Margin Calculation
OST-1036

Examinee's Name:

Date Performed:

Facility Evaluator:

Number of Attempts:

Time to Complete:

Question Documentation:

Question:

Response:

Result: SAT ______ UNSAT _______

Examiner's Signature: ________________________________ Date: ____________________

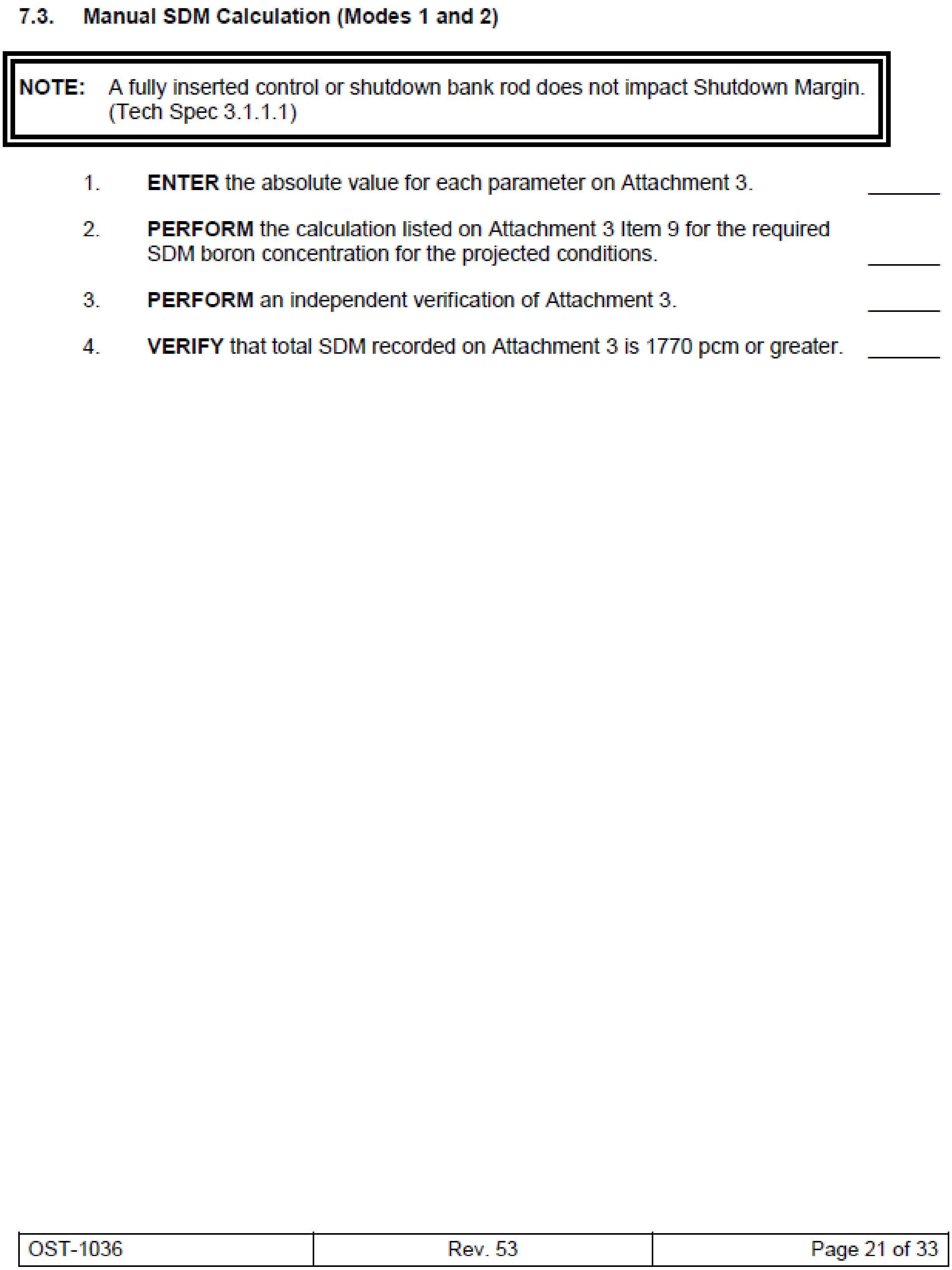
Name: ____

Date: __________________

Initial Conditions:	The plant has been operating at 92% power for 2 weeks. OST-1005, Control Rod and Rod Position Indicator Exercise Quarterly Interval Modes 1 – 3, was just performed. One rod in Control Bank 'B' was determined to be immovable/stuck. • Core burnup is 350 EFPD • RCS boron concentration is 600 ppm • POWERTRAX is **NOT** available

Initiating Cue:	The CRS has entered Tech Spec 3.1.1.1 and has directed you to complete OST-1036, Shutdown Margin Calculation Modes 1-5, Section 7.3, Manual SDM Calculation (Modes 1 and 2) for current plant conditions. Write the Total Shutdown Margin in the blank provided below. **NOTE: For this JPM notify evaluator when independent verification is required to be performed.**

Total Shutdown Margin is ________________ pcm.



7.3. Manual SDM Calculation (Modes 1 and 2)

> **NOTE:** A fully inserted control or shutdown bank rod does not impact Shutdown Margin. (Tech Spec 3.1.1.1)

1. **ENTER** the absolute value for each parameter on Attachment 3. ______
2. **PERFORM** the calculation listed on Attachment 3 Item 9 for the required SDM boron concentration for the projected conditions. ______
3. **PERFORM** an independent verification of Attachment 3. ______
4. **VERIFY** that total SDM recorded on Attachment 3 is 1770 pcm or greater. ______

OST-1036	Rev. 53	Page 21 of 33

Attachment 3 - Manual SDM Calculation (Modes 1 and 2)
Sheet 1 of 2

NOTE: A fully inserted control or shutdown bank rod does not impact Shutdown Margin. (Tech Spec 3.1.1.1)

1. **RECORD** Reactor power level. __________ % ______

2. **RECORD** Rod insertion limit for the above power level

   __________ steps on bank __________ ______

3. **RECORD** Burn up (POWERTRAX/MCR Status Board). __________ EFPD ______

4. **RECORD** Present RCS Boron Concentration. __________ ppm ______

NOTE: Use absolute values of numbers obtained from curves.

5. **OBSERVE** that the total worth of all control and shutdown banks, minus the worth of the most reactive rod for Fuel Cycle 21 is:

   _7054_ pcm
   (a) ______

6. **DETERMINE** Cycle 21 Power defect for the power level recorded in Step 1, from Curves C-X-1 through C-X-3.

   Curve Used __________

   Power defect = __________ pcm
   (b) ______

NOTE: HFP curves are used for power levels of 10% or greater.

7. **DETERMINE** inserted control rod worth at the rod insertion limit recorded in Step 2, using Curves A-X-6 to A-X-11.

   Curve Used __________

   Inserted Rod Worth = __________ pcm
   (c) ______

Attachment 3 - Manual SDM Calculation (Modes 1 and 2)

Sheet 2 of 2

8. **IF** any rod is known to be stuck or untrippable
AND is **NOT** completely inserted in the core,
THEN PERFORM the following:

    a. **IF** more than 5 rods are stuck,
    THEN:

        (1) **STOP** the calculation. _______

        (2) **NOTIFY** Reactor Engineering. _______

NOTE: Each rod (up to five total) is assigned a Stuck Rod Pair worth (SRP) value of 1724 pcm as determined by AREVA calculations with added conservatism.

The single Most Reactive Rod worth (MRR) is 1085 pcm. However, to account for an unknown stuck rod, the first stuck rod is also assigned the same SRP value of 1724 pcm as subsequent stuck rods.

Example: 1 stuck rod = 1724, 3 stuck rods = 1724 + 1724 + 1724

    b. **DETERMINE** the worth of any known stuck or untrippable rods from Table:

# Stuck Rods	1		2		3		4		5
Reactivity	1724		1724		1724		1724		1724

_______ + _______ + _______ + _______ + _______ = _______ _______
(d)

Verify

9. DETERMINE the Total Shutdown Margin (p) using the following formula:

7054 - _______ - _______ - _______ = _______ _______
(a) (b) (c) (d) Total SDM (p)

Facility:	Harris Nuclear Plant	Task No.:	301005H401
Task Title:	Determine Rod Misalignment Using Thermocouples	JPM No.:	2018 NRC Exam Admin JPM RO A1-2
K/A Reference:	G 2.1.7 RO 4.4 SRO 4.7	**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 95% power, with a load reduction in progress. During the down power DRPI indication for rod F08 shows a difference of 24 steps higher than the group demand. - The load reduction has been stopped and AOP-001 was entered. - ALB-013-8-5, Computer Alarm Rod Dev/Seq NIS Power Range Tilts alarm is the only MCB alarm received. - I&C investigated and found no obvious electrical problems.

Initiating Cue:	The CRS has directed you to calculate the temperature difference between thermocouple(s) adjacent to the misaligned rod and the average of symmetric thermocouple(s), using Attachment 2 of AOP-001 and the provided T/C Core Maps. After performing the calculation evaluate the results and circle the response below then return your results to the evaluator.

Task Standard: All calculations within ± 2° of actual.

Required Materials: Calculator

General References: AOP-001, Attachment 1, Attachment 2, Rev. 48

Handouts: JPM Cue Sheets Pages 11,12
AOP-001, Attachments 1 and 2, Rev. 48

Time Critical Task: No

Validation Time: 10 minutes

Critical Task Justification	
Step 2	If the wrong thermocouples are used then none of the results will be correct
Step 4	If the wrong values are selected then none of the results will be correct
Step 5	If the calculation for the averages were incorrect the results will be incorrect
Step 6	If the differences are calculated incorrectly then the candidate may come to the wrong conclusion for Tech Specs

Start Time: __________.

	AOP-001
Performance Step: 1	OBTAIN PROCEDURE (provided with handout)
Standard:	Obtains AOP-001 and refers to Attachments 1 and 2.
Comment:	
	AOP-001 Attachment 2 Step 1
✓ **Performance Step: 2**	DETERMINE THERMOCOUPLE LOCATION(S) ADJACENT TO THE MISALIGNED ROD USING THE CORE GRID MAP (SHEET 1).
Standard:	Using the core grid map (Attachment 2, page 1 of 3), Determines affected thermocouples to be E07, E08, F09, and G08.
Comment:	
	AOP-001 Attachment 2 Step 2
Performance Step: 3	CIRCLE LOCATION(S) IN TABLE ABOVE.
Standard:	Circles E07, E08, F09, and G08.on the table (Attachment 2, page 2 of 3).
Comment:	

AOP-001 Attachment 2 Step 3

✓ **Performance Step: 4**

RECORD the following in the table below:

- Adjacent TC number
- Adjacent TC value using the RVLIS Console, ERFIS, or OSI-PI
- Symmetric TC numbers (not including adjacent TCs)
- Symmetric TC values for all OPERABLE TCs using the RVLIS Console, ERFIS, or OSI-PI

EXAMINERS NOTE:	**If the candidate request OSI-PI or ERFIS Data inform the candidate to continue the determination using the that the RVLIS data provided.**

Standard:

Locates RVLIS Console and accesses T/C CORE MAP for Train A and Train B. **(Printout of RVLIS core map provided in handout)**

Records value for Affected TC E07(640°F) and Notes it does not have any Symmetric TC's.

Records value for Affected TC E08 (648°F) and Symmetric TC H05 (644°F), H11 (652°F), and L08 (642°F).

Records value for Affected TC F09 (644°F) and Symmetric TC G06 (640°F), and J10 (650°F).

Records value for Affected TC G08 (646°F) and Symmetric TC H09 (642°F).

Comment:

AOP-001 Attachment 2 Step 4

✓ **Performance Step: 5** DETERMINE THE AVERAGE OF SYMMETRIC THERMOCOUPLES, FOR EACH ADJACENT THERMOCOUPLE.

Standard: Determines (646°F ± **2°F**) for E08's Symmetric TCs

Determines (645°F ± **2°F**) for F09's Symmetric TCs

Determines (642°F) for G08's Symmetric TC

Comment:

EXAMINERS NOTE:	**If the candidate includes the adjacent TCs with the Symmetric TC numbers the averages will be wrong and the end result will be that a wrong final difference will be given:** Determines (646.5°F) for E08's Symmetric TCs Determines (644.7°F) for F09's Symmetric TCs Determines (642°F) for G08's Symmetric TCs.

AOP-001 Attachment 2 Step 5

✓ **Performance Step: 6** COMPARE EACH ADJACENT THERMOCOUPLE VALUE LISTED TO ITS SYMMETRIC THERMOCOUPLE AVERAGE FOR INDICATION OF A MISALIGNED ROD. (REFER TO ATTACHMENT 1.)

Standard: Calculates difference of **4°F** (± 2°F between all affected TCs and their symmetric TCs.)

2°F for TC E08

1°F for TC F09

4°F for TC G08

Identifies that the temperature difference is less than 10°F and

Circles - 1. A malfunction of Rod Position Indication (DRPI) is occurring.

Evaluator Cue:	**CRS acknowledges calculations and report.**

Terminating Cue:	Difference between each affected thermocouple and it's symmetric thermocouples has been calculated evaluation of this JPM is complete.

Stop Time: _________

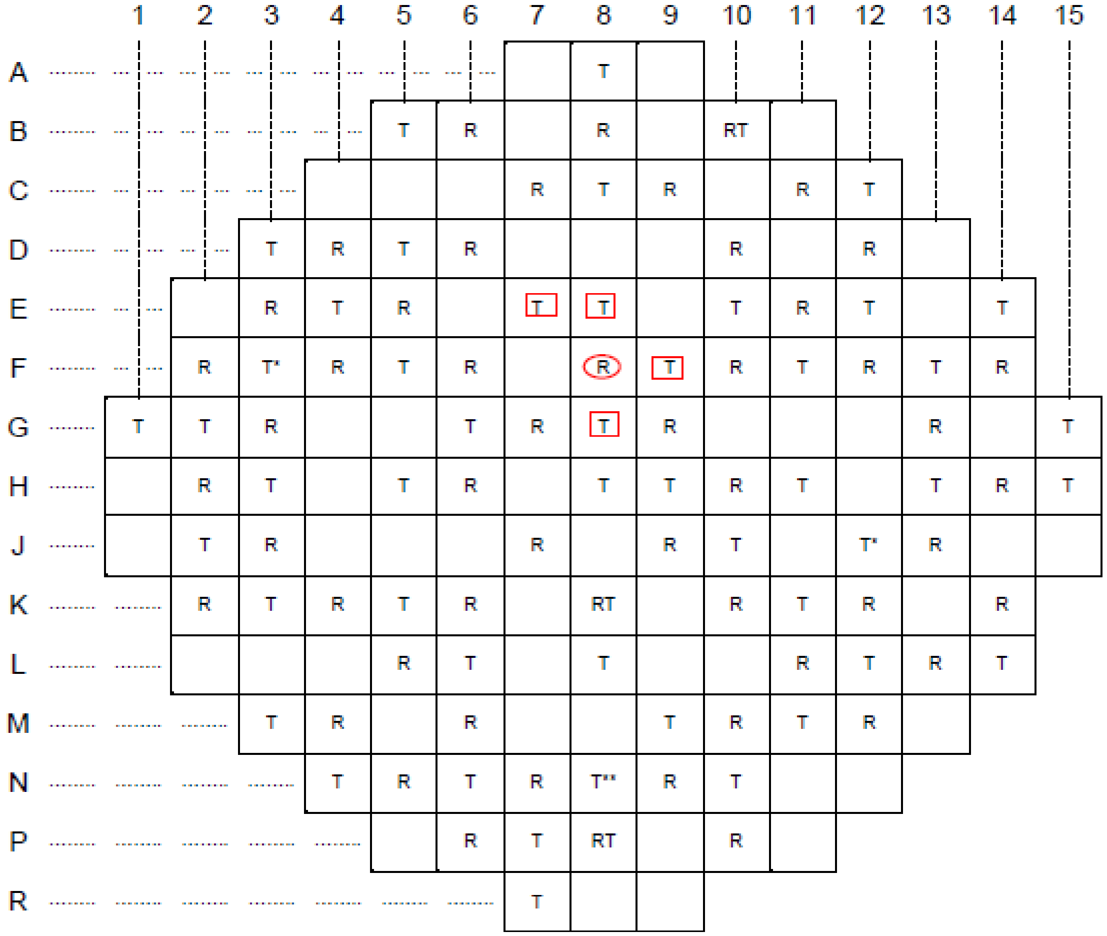
KEY

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 1 of 3

THERMOCOUPLE LOCATIONS



	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A								T							
B					T	R		R		RT					
C							R	T	R		R	T			
D			T	R	T	R				R		R			
E			R	T	R		T	T		T	R	T		T	
F		R	T*	R	T	R		R	T	R	T	R	T	R	
G	T	T	R			T	R	T	R				R		T
H		R	T		T	R		T	T	R	T		T	R	T
J		T	R				R		R	T		T*	R		
K		R	T	R	T	R		RT		R	T	R		R	
L					R	T		T			R	T	R	T	
M			T	R		R			T	R	T	R			
N				T	R	T	R	T**	R	T					
P						R	T	RT		R					
R							T								

R - Control Rod

T - Thermocouple

T* - Thermocouple(s) abandoned by EC 47997 (core location[s] F03, J12)

T** - Thermocouple(s) abandoned by EC 76393 (core location[s] N08)

KEY

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 2 of 3

NOTE

- B10, E07, H08, K08, and P08 have no symmetric locations.
- Symmetric thermocouples are those in the same row.

SYMMETRIC LOCATIONS

GRID	I		II		III		IV	
TRAIN	A	B	A	B	A	B	A	B
	A08				H15			
		G01		G15			R07	
S L	B05			E14		L14		
Y O		C08	H13				N08**	H03
M C		D03	C12				N04	M03
M A	E04	D05		E12	M11	L12		
E T			H11	E08		L08		H05
T I		F05	F11	E10	K11		K05	L06
R O		F03*	F13			N10	N06	K03
I N	G06		F09			J10		
C S		G08			H09			
	G02						J02	P07
					M09	J12*		

* - Thermocouple(s) abandoned by EC 47997 (core location[s] F03, J12)

** - Thermocouple(s) abandoned by EC 76393 (core location[s] N08)

☐ 1. **DETERMINE** thermocouple location(s) adjacent to the misaligned rod using core grid map (Sheet 1).

☐ 2. **CIRCLE** location(s) in Table above.

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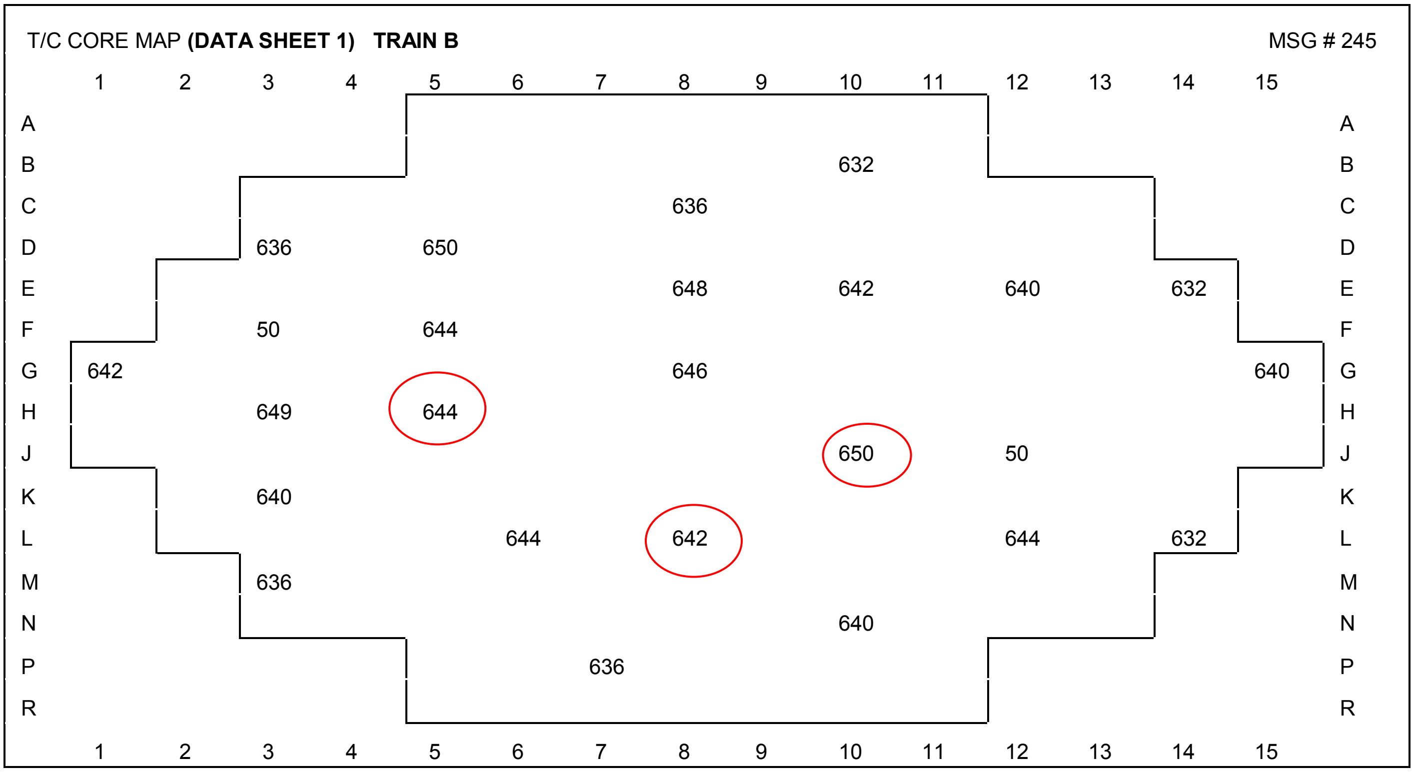
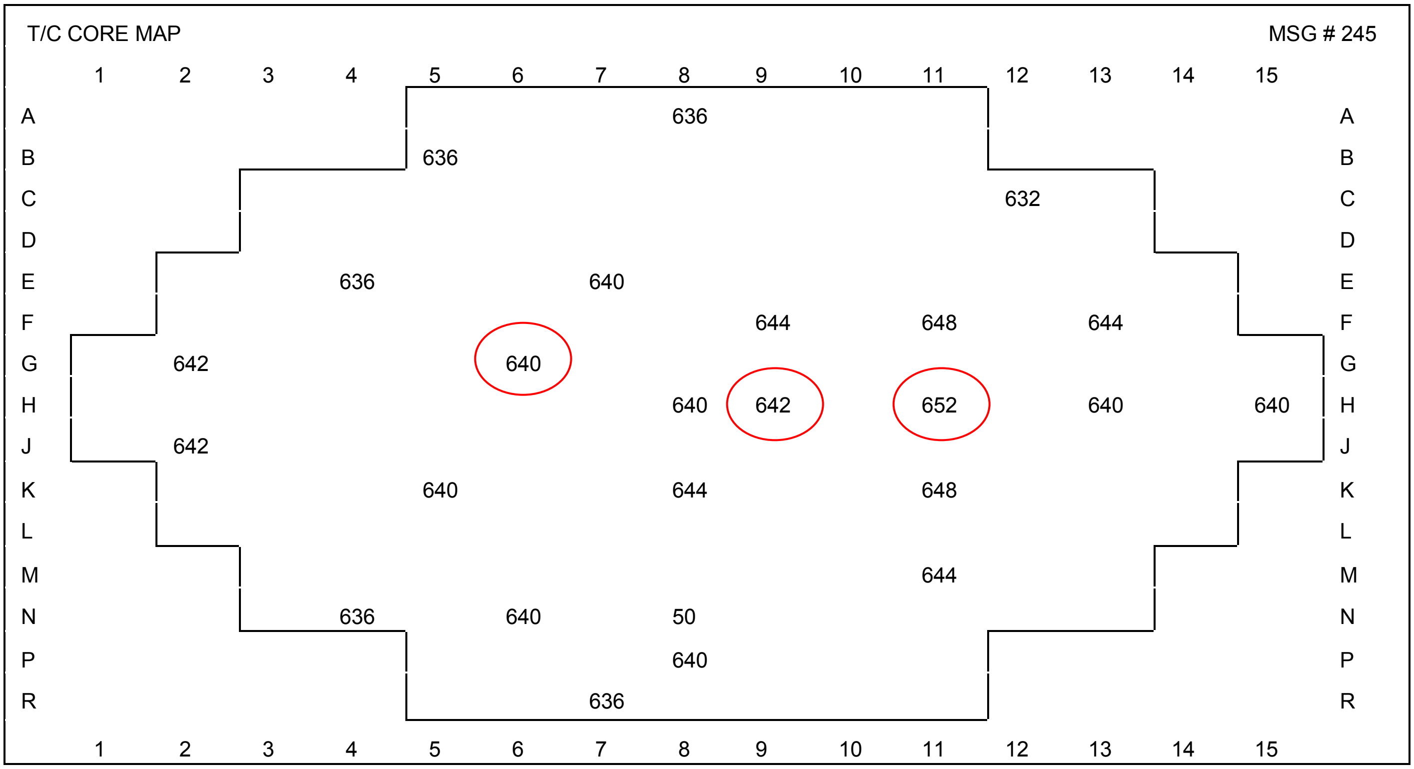
KEY



T/C CORE MAP **(DATA SHEET 1)** **TRAIN B** MSG # 245

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A																A
B										632						B
C								636								C
D			636		650											D
E								648		642		640		632		E
F			50		644											F
G	642							646							640	G
H			649		644											H
J										650		50				J
K			640													K
L						644		642				644		632		L
M			636													M
N										640						N
P							636									P
R																R
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	

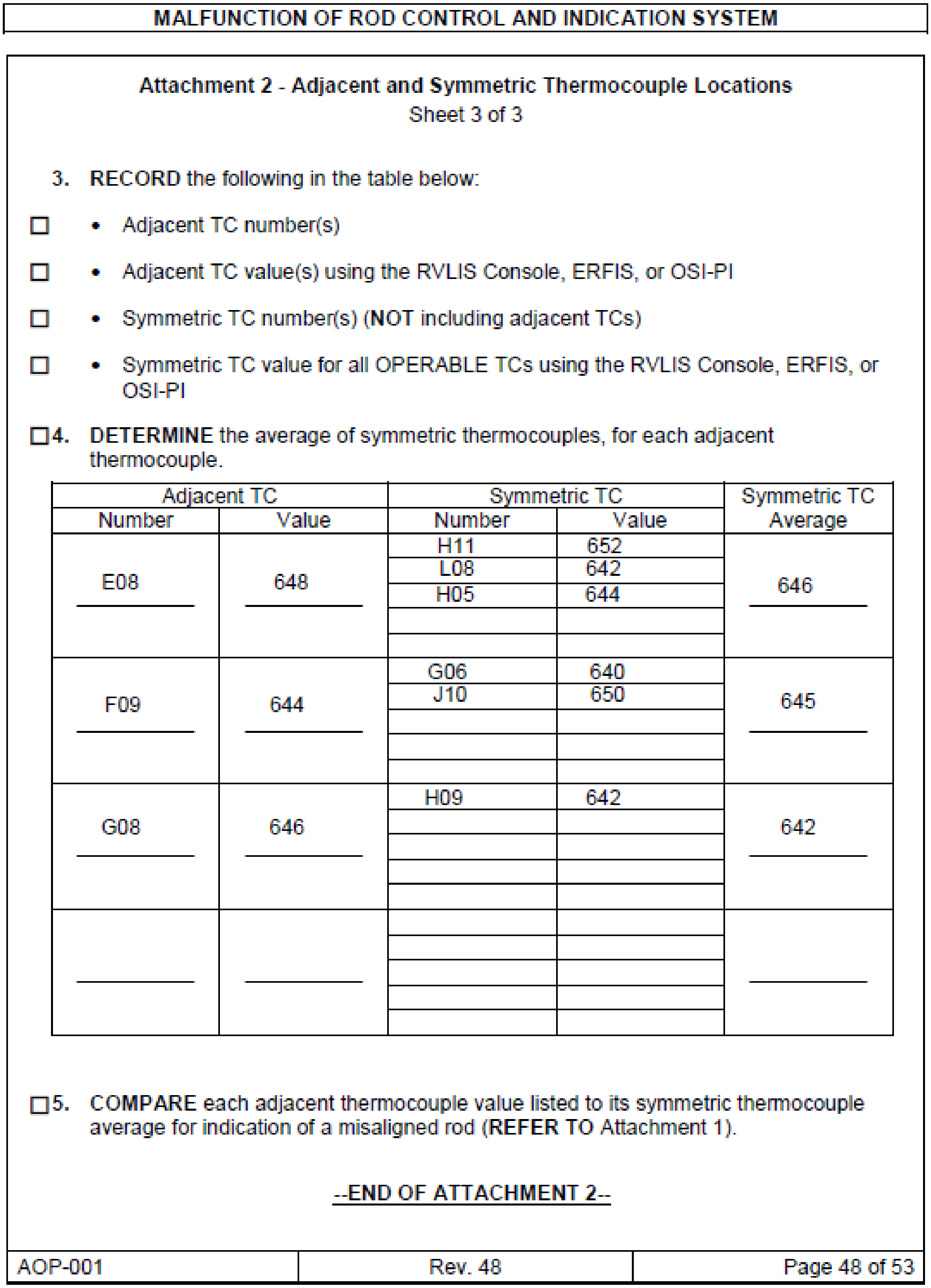
TRAIN A



T/C CORE MAP MSG # 245

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A								636								A
B					636											B
C												632				C
D																D
E				636			640									E
F									644		648		644			F
G		642				640										G
H								640	642		652		640		640	H
J		642														J
K					640			644			648					K
L																L
M											644					M
N				636		640		50								N
P								640								P
R							636									R
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	

KEY



MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 3 of 3

3. **RECORD** the following in the table below:

☐ • Adjacent TC number(s)

☐ • Adjacent TC value(s) using the RVLIS Console, ERFIS, or OSI-PI

☐ • Symmetric TC number(s) (**NOT** including adjacent TCs)

☐ • Symmetric TC value for all OPERABLE TCs using the RVLIS Console, ERFIS, or OSI-PI

☐4. **DETERMINE** the average of symmetric thermocouples, for each adjacent thermocouple.

Adjacent TC		Symmetric TC		Symmetric TC Average
Number	Value	Number	Value	
E08	648	H11	652	646
		L08	642	
		H05	644	
F09	644	G06	640	645
		J10	650	
G08	646	H09	642	642

☐5. **COMPARE** each adjacent thermocouple value listed to its symmetric thermocouple average for indication of a misaligned rod (**REFER TO** Attachment 1).

--END OF ATTACHMENT 2--

AOP-001 Rev. 48 Page 48 of 53

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Admin Exam RO A1-2
Determine Rod Misalignment Using Thermocouples
AOP-001

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 95% power, with a load reduction in progress. During the down power DRPI indication for rod F08 shows a difference of 24 steps higher than the group demand. • The load reduction has been stopped and AOP-001 was entered. • ALB-013-8-5, Computer Alarm Rod Dev/Seq NIS Power Range Tilts alarm is the only MCB alarm received. • I&C investigated and found no obvious electrical problems.

Initiating Cue:	The CRS has directed you to calculate the temperature difference between thermocouple(s) adjacent to the misaligned rod and the average of symmetric thermocouple(s), using Attachment 2 of AOP-001 and the provided T/C Core Maps. After performing the calculation evaluate the results and circle the response below then return your results to the evaluator.

Name: ____

Date: ____________________

Circle the correct response that applies:

1. A malfunction of Rod Position Indication (DRPI) is occurring
2. A Rod Misalignment is occurring

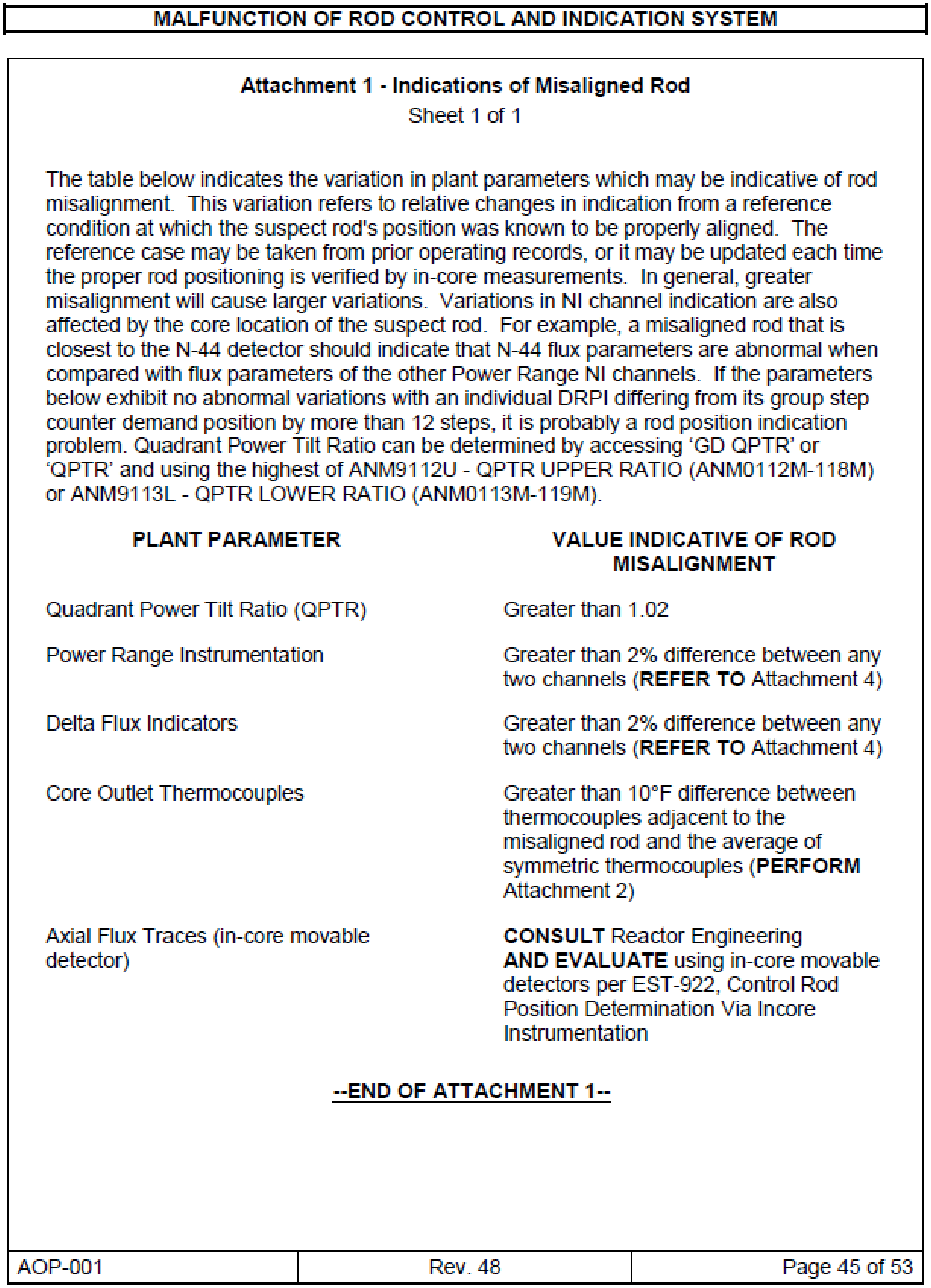
T/C CORE MAP **(DATA SHEET 1)** **TRAIN B** MSG # 245

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A																A
B										632						B
C								636								C
D			636		650											D
E								648		642		640		632		E
F			50		644											F
G	642							646							640	G
H			649		644											H
J										650		50				J
K			640													K
L						644		642				644		632		L
M			636													M
N										640						N
P							636									P
R																R
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	

TRAIN A

T/C CORE MAP MSG # 245

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A								636								A
B					636											B
C												632				C
D																D
E				636			640									E
F									644		648		644			F
G		642				640										G
H								640	642		652		640		640	H
J		642														J
K					640			644			648					K
L																L
M											644					M
N				636		640		50								N
P								640								P
R							636									R
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	



Attachment 1 - Indications of Misaligned Rod

Sheet 1 of 1

The table below indicates the variation in plant parameters which may be indicative of rod misalignment. This variation refers to relative changes in indication from a reference condition at which the suspect rod's position was known to be properly aligned. The reference case may be taken from prior operating records, or it may be updated each time the proper rod positioning is verified by in-core measurements. In general, greater misalignment will cause larger variations. Variations in NI channel indication are also affected by the core location of the suspect rod. For example, a misaligned rod that is closest to the N-44 detector should indicate that N-44 flux parameters are abnormal when compared with flux parameters of the other Power Range NI channels. If the parameters below exhibit no abnormal variations with an individual DRPI differing from its group step counter demand position by more than 12 steps, it is probably a rod position indication problem. Quadrant Power Tilt Ratio can be determined by accessing 'GD QPTR' or 'QPTR' and using the highest of ANM9112U - QPTR UPPER RATIO (ANM0112M-118M) or ANM9113L - QPTR LOWER RATIO (ANM0113M-119M).

PLANT PARAMETER	VALUE INDICATIVE OF ROD MISALIGNMENT
Quadrant Power Tilt Ratio (QPTR)	Greater than 1.02
Power Range Instrumentation	Greater than 2% difference between any two channels (**REFER TO** Attachment 4)
Delta Flux Indicators	Greater than 2% difference between any two channels (**REFER TO** Attachment 4)
Core Outlet Thermocouples	Greater than 10°F difference between thermocouples adjacent to the misaligned rod and the average of symmetric thermocouples (**PERFORM** Attachment 2)
Axial Flux Traces (in-core movable detector)	**CONSULT** Reactor Engineering **AND EVALUATE** using in-core movable detectors per EST-922, Control Rod Position Determination Via Incore Instrumentation

--END OF ATTACHMENT 1--

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 1 of 3

THERMOCOUPLE LOCATIONS

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A								T							
B					T	R		R		RT					
C							R	T	R		R	T			
D			T	R	T	R				R		R			
E			R	T	R		T	T		T	R	T		T	
F		R	T*	R	T	R		R	T	R	T	R	T	R	
G	T	T	R			T	R	T	R				R		T
H		R	T		T	R		T	T	R	T		T	R	T
J		T	R				R		R	T		T*	R		
K		R	T	R	T	R		RT		R	T	R		R	
L					R	T		T			R	T	R	T	
M			T	R		R			T	R	T	R			
N				T	R	T	R	T**	R	T					
P						R	T	RT		R					
R							T								

R - Control Rod

T - Thermocouple

T* - Thermocouple(s) abandoned by EC 47997 (core location[s] F03, J12)

T** - Thermocouple(s) abandoned by EC 76393 (core location[s] N08)

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 2 of 3

NOTE

- B10, E07, H08, K08, and P08 have no symmetric locations.
- Symmetric thermocouples are those in the same row.

SYMMETRIC LOCATIONS

GRID	I		II		III		IV	
TRAIN	A	B	A	B	A	B	A	B
	A08				H15			
		G01		G15			R07	
S L	B05			E14		L14		
Y O		C08	H13				N08**	H03
M C		D03	C12				N04	M03
M A	E04	D05		E12	M11	L12		
E T			H11	E08		L08		H05
T I		F05	F11	E10	K11		K05	L06
R O		F03*	F13			N10	N06	K03
I N	G06		F09			J10		
C S		G08			H09			
	G02						J02	P07
					M09	J12*		

* - Thermocouple(s) abandoned by EC 47997 (core location[s] F03, J12)

** - Thermocouple(s) abandoned by EC 76393 (core location[s] N08)

☐ **1.** **DETERMINE** thermocouple location(s) adjacent to the misaligned rod using core grid map (Sheet 1).

☐ **2.** **CIRCLE** location(s) in Table above.

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 3 of 3

3. **RECORD** the following in the table below:

☐ • Adjacent TC number(s)

☐ • Adjacent TC value(s) using the RVLIS Console, ERFIS, or OSI-PI

☐ • Symmetric TC number(s) (**NOT** including adjacent TCs)

☐ • Symmetric TC value for all OPERABLE TCs using the RVLIS Console, ERFIS, or OSI-PI

☐ 4. **DETERMINE** the average of symmetric thermocouples, for each adjacent thermocouple.

Adjacent TC		Symmetric TC		Symmetric TC Average
Number	Value	Number	Value	

☐ 5. **COMPARE** each adjacent thermocouple value listed to its symmetric thermocouple average for indication of a misaligned rod (**REFER TO** Attachment 1).

--END OF ATTACHMENT 2--

Facility:	Harris Nuclear Plant	Task No.:	015004H201
Task Title:	Perform the Quadrant Power Tilt Ratio Surveillance	JPM No.:	2018 NRC Exam Admin JPM RO A2
K/A Reference:	G2.2.12 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:	• The plant is operating at 90% power when a rod in Control Bank 'A' (P-10) dropped. • The crew is performing AOP-001, MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM. • There are NO deficiency tags on PR NIs. • ERFIS points ANM9112U and ANM9113L have a BAD quality code. HNP IT has been notified and they are evaluating the ERFIS points.

Initiating Cue:	The CRS has directed you to perform a manual QPTR IAW OST-1039, CALCULATION OF QPTR. The Power Range NIS indications are provided. **For the purposes of the examination, there will be no independent verification. Show values of your work.**

Task Standard:	Calculations within required band. Correct Tech Spec actions are identified.
Required Materials:	Calculator
General References:	OST-1039, CALCULATION OF QPTR, Revision 17 Technical Specifications
Handouts:	• OST-1039 • Power Range NI – Current and Voltage Set point Table • Technical Specifications
Time Critical Task:	No
Validation Time:	10 minutes

Critical Step Justification	
Step 9	Must accurately determine the correct calculation based on collecting and inputting either provided data or visual inspection data. The calculation will yield an unsatisfactory QPTR.
Step 10	Must accurately determine the correct calculation based on collecting and inputting either provided data or visual inspection data. The calculation will yield an unsatisfactory QPTR.
Step 11	Must accurately determine the correct calculation based on collecting and inputting either provided data or visual inspection data. The calculation will yield an unsatisfactory QPTR.
Step 14	Must identify that the QPTR upper is outside the band which will make this overall results unsatisfactory.
Step 15	Must determine that QPTR is greater than 1.02 (which is a Tech Spec limit) and that the QPTR is unsatisfactory.

Start Time: __________.

Performance Step: 1 Obtain procedure.

Standard: Reviews procedure.

Evaluator Cue:	**Provide OST-1039.**

Evaluator Note: **A KEY is provided for your use on JPM prior to candidate pages.**

Comment:

Evaluator Note:	**NOTE: The NI curve numbers provided in this JPM are numbers from the 2018 NRC Exam Frozen Procedures Curve Book folder.**

Procedure Note: Precaution and Limitation 3.1.1 has guidance if performing this OST with one Power Range Channel inoperable.

Performance Step: 2 Completes Prerequisites section:

- Ensure instrumentation needed for the performance of this test is free of deficiencies that affect instrument indication.
- Ensure the most recent Curve F-x-8 is used in the performance of this procedure. (Reference 9.5.7 and 9.5.1)
- Obtain CRS permission to perform this OST.
- Obtain necessary tools and equipment from the following list
  - IBM PC or compatible

Standard:

- Logs F-20-8 revision number : 4
- Initials/signs all blocks

Comment:

Performance Step: 3

- IF Quadrant Power Tilt Ratio Calculation Computer Program is used, THEN PERFORM the following:
  - MARK Step 7.1. Step 2 N/A.
  - MARK Section 7.3 N/A.
  - PERFORM Section 7.2.
- IF manual calculation of the Quadrant Power Tilt Ratio is used, THEN PERFORM the following:
  - MARK Section 7.2 N/A.
  - PERFORM Section 7.3.

Standard:

- Marks Section 7.2 N/A
- Proceeds to Section 7.3

Comment:

OST-1039 Section 7.3 Note prior to step 1

Performance Step: 4

NOTE: The detector current meters on each power range channel drawer are designated as left-upper, right-lower.

Standard:

Reads and circle slashes note

Comment:

OST-1039 Section 7.3, Step 1

Performance Step: 5

Prior to reading the value of detector current, VERIFY the meter range/rate switch is in the 400 µA/SLOW position.

Standard:

Prior to reading the value of detector current, VERIFIES the Meter Range/Rate switch is in the 400 µA/SLOW position.

Evaluator Note:	**This information is on the JPM Cue Sheet**

Comment:

OST-1039 Section 7.3, Step 2

Performance Step: 6 RECORD on Attachment 2, in column A, the upper and lower detector currents from all operable power range channels as read on the Nuclear Instrumentation Cabinet.

Standard: Transposes readings from PRNIS Readings Table onto Attachment 2.

Comment:

OST-1039 Section 7.3, Step 3

Performance Step: 7 RECORD on Attachment 2, in column B, the 100% power normalized current for each channel from Curve F-x-8.

Standard: Transposes TOP and BOTTOM 100% current values from the Curve Book provided.

Comment:

OST-1039 Section 7.3, Note prior to Step 4

Performance Step: 8 NOTE: When recording all fractions and ratios, record to four decimal places, dropping the fifth and subsequent decimal places.

Standard: Reads and annotates note

Comment:

OST-1039 Section 7.3, Step 4

✓ **Performance Step: 9** Divide values in Column A by the respective normalized current in Column B and record the result in Column C as the Normalized Fraction.

Standard: Divides each Upper and Lower reading by the respective 100% normalized current value and records in Column C.

Comment:

OST-1039 Section 7.3, Step 5

✓ **Performance Step: 10** CALCULATE the average value for the upper and the lower Normalized Fractions as follows:

- ADD the Normalized Fraction in each section of column C, recording the sum in the space provided.
- DIVIDE the sum obtained in Step 7.3.5.a by the number of operable NI channels, recording the result in column D of Attachment 2.

Standard: Adds all Normalized Fractions for the same plane and records the sum in the space provided.

Divides by the sum by four and records result in Column D.

Comment:

OST-1039 Section 7.3, Step 6

✓ **Performance Step: 11** Using the formula and values from Attachment 2, CALCULATE the Upper and Lower Ratios.

Standard:

- Divides the Maximum Normalized Fraction by the Average Normalized Fraction on each plane.
- Determines the UPPER ratio is ≥ 1.02

Evaluator Note:	**The applicant may inform the CRS as soon as any calculation is > 1.02. If so, acknowledge and direct applicant to complete Attachment 2.**

Comment:

OST-1039 Section 7.3, Step 7

Performance Step: 12 PERFORM independent verification of all calculations made on Attachment 2.

Standard: Requests Independent Verifier.

Evaluator Cue: **If necessary, repeat Initiating Cue: For the purpose of this examination, there will be no independent verification of your work.**

Comment: **Candidate may choose to check calculations.**

OST-1039 Section 7.3, Note prior to Step 8

Performance Step: 13 NOTE: The upper ratio or the lower ratio, whichever is greater, is the quadrant power tilt ratio (QPTR).

Standard: Reads and circle slashes note

Comment:

OST-1039 Section 7.3, Step 8

✓ **Performance Step: 14** RECORD QPTR:

Standard: Records QPTR value as 1.0469 to 1.0479 (N43 LOWER)
Identifies Lower as outside the band

Comment: **Acceptable band is +/- 5% (rounded to .0005).**
UPPER calculated band is 1.0291 to 1.0301
LOWER calculated band is 1.0469 to 1.0479

OST-1039 Section 7.3, Step 9

✓ **Performance Step: 15** CHECK QPTR is less than or equal to 1.02.

Standard: **Identifies Lower QPTR as greater than 1.02 and QPTR is unacceptable**

Comment:

OST-1039 Section 7.3, Note prior to Step 10

Performance Step: 16 NOTE: ERFIS turn on codes used to obtain ERFIS QPTR values include “QPTR” and “GD QPTR”.

Standard: Reads and circle slashes note

Comment:

OST-1039 Section 7.3, Step 10

Performance Step: 17 IF the ERFIS calculated QPTR value is available, THEN COMPARE OST-1039 results to the ERFIS QPTR calculated output as a quality check.

Standard: Request status of ERFIS calculated QPTR value, and N/A’s step 7.3.10 when notified ERFIS QPTR is not available.

Evaluator Note:	**This information is on the JPM Cue Sheet**

Evaluator Cue:

Comment:

OST-1039 Section 7.3, Step 11

Performance Step: 18 IF any ERFIS QPTR quality codes do not have a good quality code or the higher of ANM9112U or ANM9113L do not approximate the value for QPTR determined above, THEN CONTACT HNP IT to investigate.

Standard: Request if notification of the status of the ERFIS calculated QPTR value to HNP IT has been completed.

Evaluator Note:	**This information is on the JPM Cue Sheet**

Evaluator Cue:

Comment:

Terminating Cue:	**Once the determination that ERFIS Quality Codes do not have good quality code announce "End of the JPM":** **Evaluation on this JPM is complete.**

STOP Time: __________.

KEY

Record QPTR = **1.0474** Acceptable band is +/- 5% (rounded to .0005) 1.0469 to 1.0479

CHECK QPTR is less than or equal to 1.02 (circle) YES / [✓ NO]

	A	B	C	D
UPPER DETECTOR	UPPER DETECTOR CURRENT	UPPER 100% POWER NORMALIZED CURRENT	UPPER NORMALIZED FRACTION (NOTE 1)	AVERAGE UPPER NORMALIZED FRACTION
N-41	145.6	150.5	0.9674	0.9467
N-42	162.5	172.8	0.9403	
N-43	189.8	194.7	0.9748	
N-44	138.4	153.0	0.9045	
		SUM	3.7870	

$$\text{Upper Ratio} = \frac{\text{Maximum Upper Normalized Fraction}}{\text{Average Upper Normalized Fraction}} = \frac{0.9748}{0.9467} = 1.0296^{*}$$

* Standard for this calculation is 1.0291 to 1.0301

	A	B	C	D
LOWER DETECTOR	LOWER DETECTOR CURRENT	LOWER 100% POWER NORMALIZED CURRENT	LOWER NORMALIZED FRACTION (NOTE 1)	AVERAGE LOWER NORMALIZED FRACTION
N-41	159.6	167.1	0.9551	0.9400
N-42	172.1	191.1	0.9005	
N-43	205.3	208.5	0.9846	
N-44	165.2	179.6	0.9198	
		SUM	3.7600	

$$\text{Lower Ratio} = \frac{\text{Maximum Lower Normalized Fraction}}{\text{Average Lower Normalized Fraction}} = \frac{0.9846}{0.9400} = 1.0474^{**}$$

** Standard for this calculation is 1.0469 to 1.0479

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Admin JPM RO A2
Perform a Quadrant Power Tilt Ratio Surveillance

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 90% power when a rod in Control Bank 'A' (P-10) dropped. • The crew is performing AOP-001, MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM. • There are NO deficiency tags on PR NIs. • ERFIS points ANM9112U and ANM9113L have a BAD quality code. HNP IT has been notified and they are evaluating the ERFIS points.

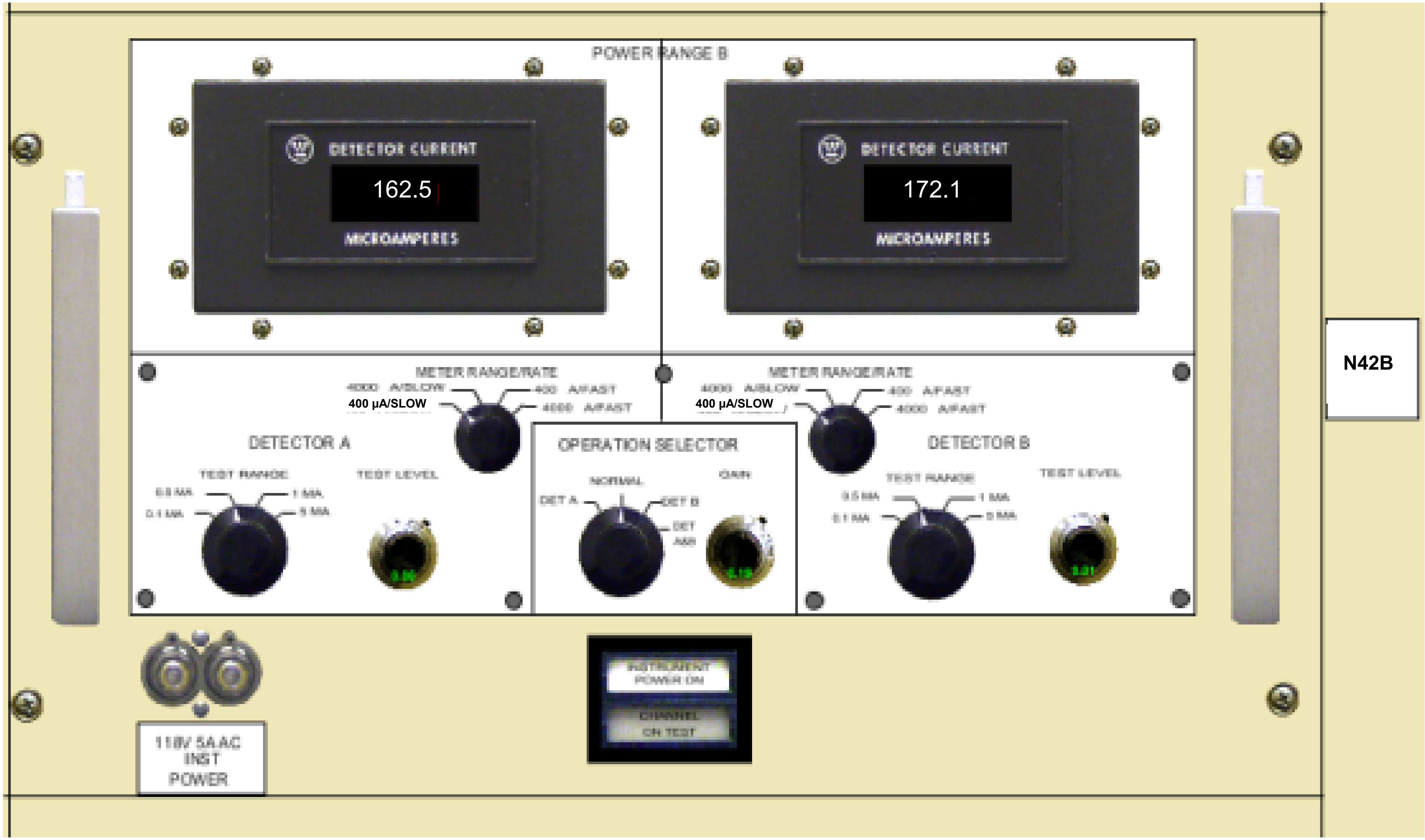
Initiating Cue:	The CRS has directed you to perform a manual QPTR IAW OST-1039, CALCULATION OF QPTR. The Power Range NIS indications are provided. After performing the calculation return your results to the evaluator. **For the purposes of the examination, there will be no independent verification. Show values of your work.**

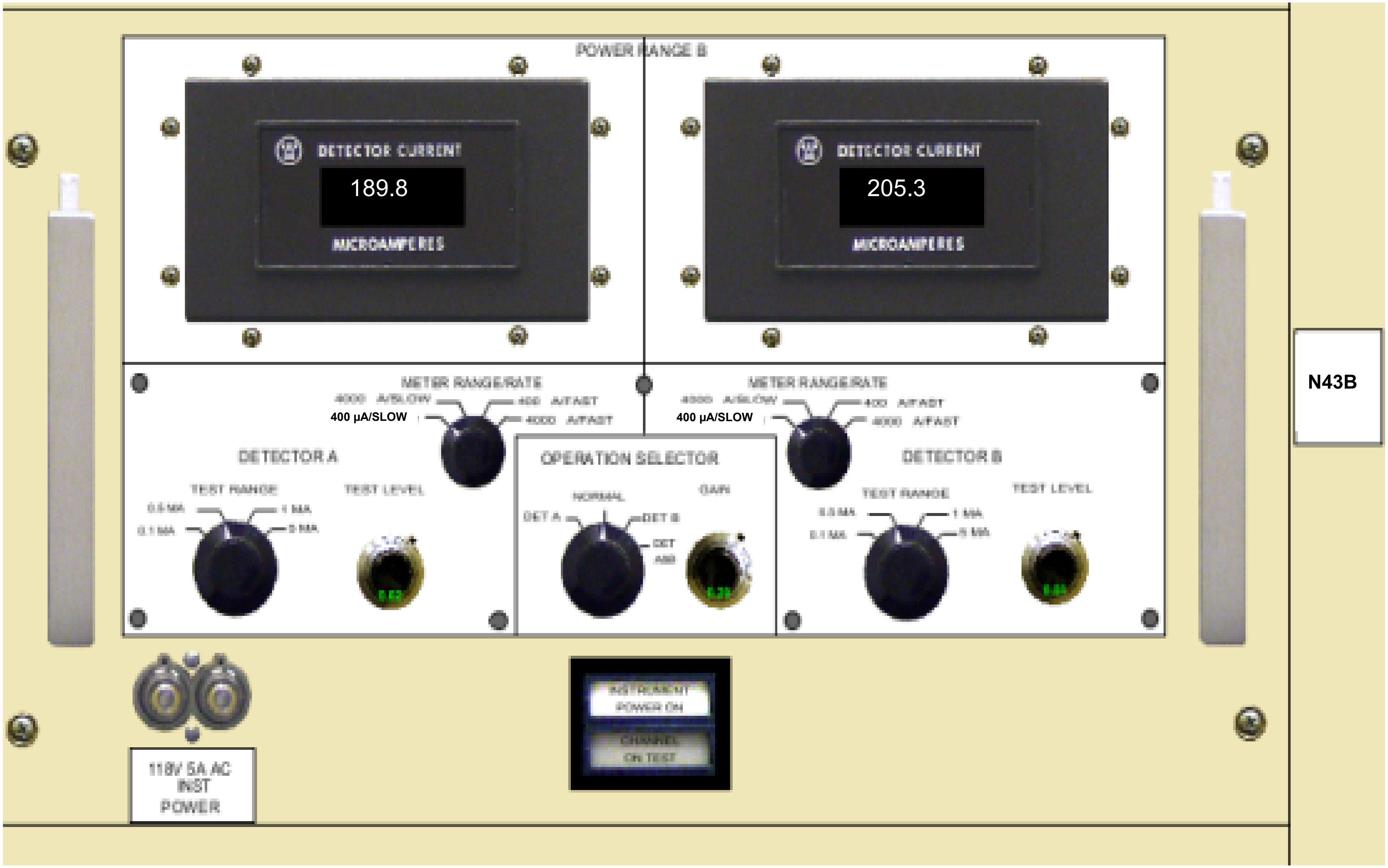
Name: ____

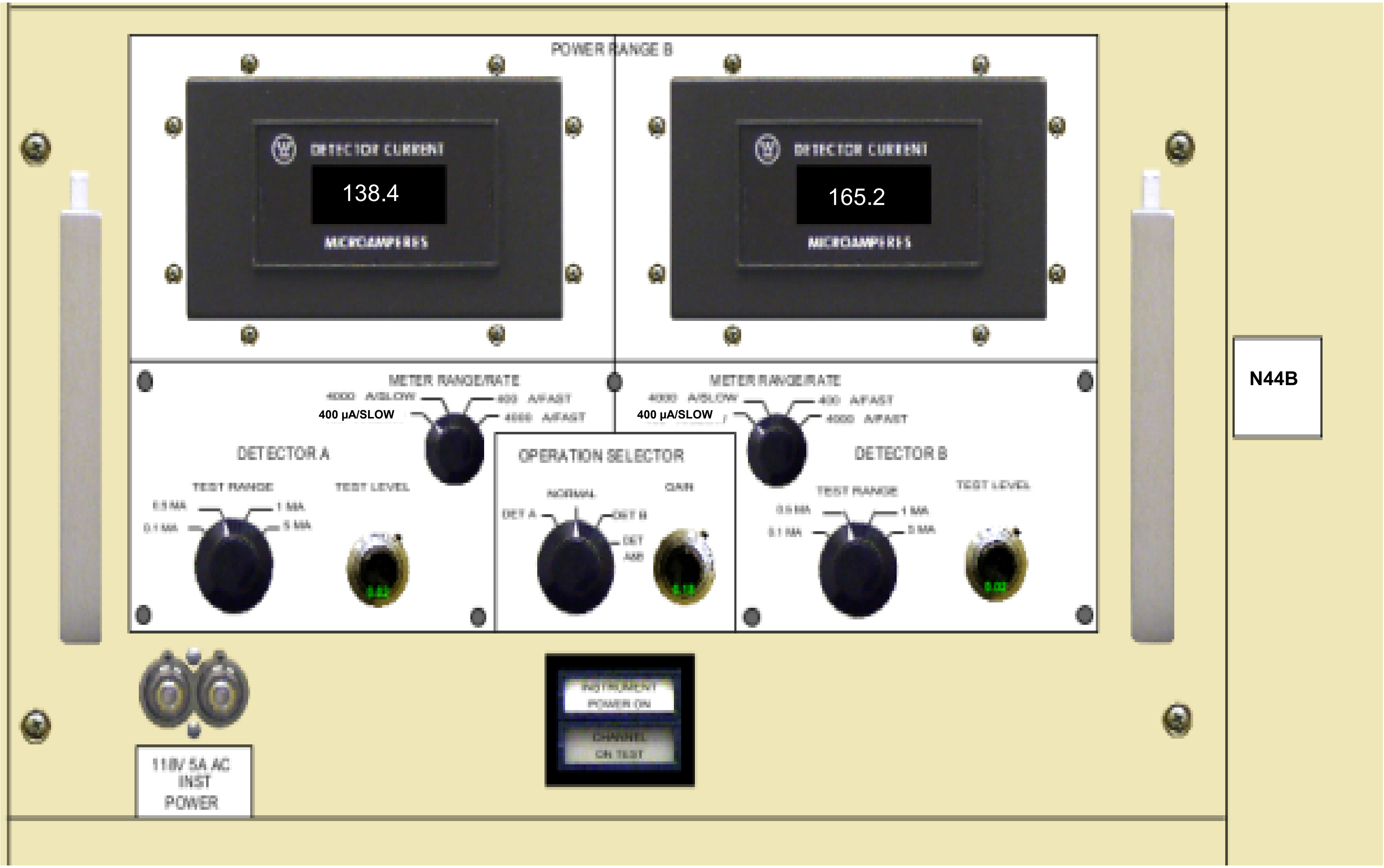
Date: ____________________

JPM CUE SHEET

JPM CUE SHEET




POWER RANGE B
DETECTOR CURRENT
189.8
MICROAMPERES
DETECTOR CURRENT
205.3
MICROAMPERES
METER RANGE/RATE
4000 A/SLOW
400 A/FAST
400 µA/SLOW
4000 A/FAST
DETECTOR A
TEST RANGE
0.5 MA
1 MA
0.1 MA
5 MA
TEST LEVEL
OPERATION SELECTOR
NORMAL
GAIN
DET A
DET B
DET ABS
METER RANGE/RATE
4000 A/SLOW
400 A/FAST
400 µA/SLOW
4000 A/FAST
DETECTOR B
TEST RANGE
0.5 MA
1 MA
0.1 MA
5 MA
TEST LEVEL
INSTRUMENT POWER ON
CHANNEL ON TEST
118V 5A AC INST POWER
N43B


POWER RANGE B
DETECTOR CURRENT
138.4
MICROAMPERES
DETECTOR CURRENT
165.2
MICROAMPERES
N44B
METER RANGE/RATE
4000 A/SLOW
400 A/FAST
400 µA/SLOW
4000 A/FAST
METER RANGE/RATE
4000 A/SLOW
400 A/FAST
400 µA/SLOW
4000 A/FAST
DETECTOR A
TEST RANGE
TEST LEVEL
0.5 MA
1 MA
0.1 MA
5 MA
OPERATION SELECTOR
NORMAL
GAIN
DET A
DET B
DET A&B
DETECTOR B
TEST RANGE
TEST LEVEL
0.5 MA
1 MA
0.1 MA
5 MA
INSTRUMENT POWER ON
CHANNEL ON TEST
118V 5A AC INST POWER

Facility:	Harris Nuclear Plant	Task No.:	119013H304
Task Title:	Using survey maps determine stay times	JPM No.:	2018 NRC Exam Admin JPM RO A3
K/A Reference:	G.2.3.4 RO 3.2 SRO 3.7	**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: Two Operators are being assigned to hang a clearance on 1CS-38, Letdown PCV Isol vlv and perform work in a radiological area. The clearance includes the following valves:

- 1CS-35
- 1CS-36
- 1CS-37
- 1CS-38
- 1CS-39
- 1CS-40
- 1CS-43

Operator 1 has an accumulated annual Whole Body dose of 1750 mrem (Duke Energy Progress).

Operator 2 has an accumulated annual Whole Body dose of 700 mrem (Duke Energy Progress) and worked at Nine Mile Point earlier this year where he has accumulated 2550 mrem.

In accordance with PD-RP-ALL-0001, the Radiation Protection Manager has authorized Operator 2 a dose extension to the limit that his signature authority is authorized to.

The ALARA group has determined that additional shielding is not warranted for this work.

Initiating Cue:	Using the supplied survey map, determine the maximum allowable individual stay times for each Operator that would prevent exceeding the Duke Energy Annual Administrative dose limit while performing these activities. Do not consider dose received during transit. The calculated dose should be ONLY what they would receive while working at the valves for the clearance. Complete the information below and return to the evaluator when complete.

Task Standard: Calculation of stay times based on survey maps, two hours for Operator 1, one hour and twelve minutes for Operator 2.

Required Materials: Survey map A45
RAB 236' LETDOWN & LETDOWN REHEAT HX &VG Map 21
SFD-5-S-1304

General References: PD-RP-ALL-0001 “Radiation Worker Responsibilities” Section 5.2 (Rev. 7)

LIMIT = 2 rem Duke Energy Progress dose not to exceed 3.4 rem total dose if non- Duke Energy Progress dose for the current year has been determined.

Time Critical Task: No

Validation Time: 10 minutes

Critical Task Justification	
Step 1	Must determine dose rates in order to calculate stay time
Step 2	Must determine available dose to determine stay time.
Step 3	IF incorrect calculation of stay time is made the individuals could exceed their dose limits.

START TIME: ____________

Evaluator Note:	**The order of performance does not matter** **IF THE APPLICANT ASKS FOR IT: Provide a copy of PD-RP-ALL-0001, Radiation Worker Responsibilities, Rev 3**

✓ **Performance Step: 1** Using Radiological Survey Record Map A45 and RAB 236' LETDOWN & LETDOWN REHEAT HX &VG Map 21, determines dose rates in the area where the clearance will be applied

Standard: Identifies that General Area Dose Rates are 125 mrem/hr

Comment:

✓ **Performance Step: 2** Determine the remaining dose for the year for each individual

Standard: Operator 1: **250 mrem**

2000 mrem - 1750 mrem = 250 mrem

Operator 2: **150 mrem**

3400 mrem - 700 mrem (DEP) - 2550 mrem (NMP) = 150 mrem

Comment:

✓	**Performance Step: 3**	Determine stay time for each operator (based on 1st Operator reaching 2 Rem and the 2nd Operator reaching 3.4 Rem - for the year)
	Standard:	Operator 1: **2 hours** 250 mrem ÷ 125 mrem/hr = 2 hours Operator 2: **1 hour and 12 minutes** 150 mrem ÷ 125 mrem/hr = 1 hour and 12 minutes
	Comment:	

Terminating Cue:	**After the stay time has been calculated, this JPM is complete.** **END OF JPM**

STOP TIME: ____________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Admin JPM RO A3 - Using survey maps determine stay times.
PD-RP-ALL-0001 Rev. 7

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:	Two Operators are being assigned to hang a clearance on 1CS-38, Letdown PCV Isol vlv and perform work in a radiological area. The clearance includes the following valves: • 1CS-35 • 1CS-36 • 1CS-37 • 1CS-38 • 1CS-39 • 1CS-40 • 1CS-43 Operator 1 has an accumulated annual Whole Body dose of 1750 mrem (Duke Energy Progress). Operator 2 has an accumulated annual Whole Body dose of 700 mrem (Duke Energy Progress) and worked at Nine Mile Point earlier this year where he has accumulated 2550 mrem. In accordance with PD-RP-ALL-0001, the Radiation Protection Manager has authorized Operator 2 a dose extension to the limit that his signature authority is authorized to administer. The ALARA group has determined that additional shielding is not warranted for this work.

Initiating Cue:	Using the supplied survey map, determine the maximum allowable individual stay times for each Operator that would prevent exceeding the Duke Energy Annual Administrative dose limit while performing these activities. Do not consider dose received during transit. The calculated dose should be ONLY what they would receive while working at the valves for the clearance. Complete the information below and return to the evaluator when complete.

Name ____

Record the maximum allowable stay time calculations below to the nearest hour and minute.

Operator 1: _________________________ Operator 2: ________________________

Facility:	Harris Nuclear Plant	Task No.:	341010H302
Task Title:	Perform Review of Daily Surveillance Requirements Log	JPM No.:	2018 NRC Exam Admin JPM SRO A1-1
K/A Reference:	G 2.1.18 RO 3.6 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, 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 operating at 100% power on Tuesday at 2300 • EDG “A” is synchronized to the grid for a post-maintenance test and has been running at 6.3 MW for 30 minutes. • The Daily Surveillance Logs (OST-1021, Attachment 3) have been completed. • ERFIS Pressurizer Pressures are unavailable.

Initiating Cue:	You are the CRS and have just completed an emergent watch relief due to illness of the scheduled person. Review the OST-1021, Attachment 3 logs. At the conclusion of your review, list any discrepancies or problems if applicable and identify the Tech Spec and applicable actions. Be prepared to discuss any findings with the evaluator.

Note: Ensure page 33 of OST-1021 (Handout) is in color for the red circle

Task Standard: All errors and TS actions identified

Required Materials: Perform in a location with TS or electronic access to TS available and PLP-114, Rev 26.

General References:

- OST-1021, DAILY SURVEILLANCE REQUIREMENTS, DAILY INTERVAL, MODE 1 AND 2
- Technical Specifications

Handouts:

- Copy of a completed OST-1021, Attachment 4.
- Substitute the following incorrect data:
  - Page 26 instruction line item 1 – N/A this line. Make 3 of the RCS Loop flows out of spec. There are 4 wrong things with this – there should be an initial in line item 1 not an N/A. Then the 3 channels should be identified as out of spec. There should also be a Tech Spec determination based on the out of spec readings.
  - Page 29 the Condensate Storage Tank Level (both channels) progressively lowering from 85% to 60% throughout the day. The readings are out of spec and there should also be a Tech Spec determination.
  - Page 33 the EDG Room Temperature channel TDG6903A at 121ºF (should be circled) and there should also be a PLP-114 determination.
  - Page 33 the Aux RSVR Level (both channels) progressively lowering from 250.2 ft to 249.7 ft throughout the day. The readings are out of spec and there should also be a Tech Spec determination.

Time Critical Task: No

Validation Time: 25 minutes

Critical Step Justification	
Step 2	Critical to comply with Technical Specification requirements.
Step 3	Critical to comply with Technical Specification requirements.
Step 4	Critical to comply with Relocated Technical Specification requirements.
Step 5	Critical to comply with Technical Specification requirements.

Start Time: __________.

Performance Step: 1 Obtain completed log.

Standard: Reviews handout.

Evaluator Cue:

- **Provide handout for NRC Exam Admin JPM SRO A1-1 after the Initial Conditions are reviewed and the Initiating Cue is provided.**
- **If necessary, after the applicant discusses each finding: What action, if any, is required relative to this reading?**

Evaluator Note:	**Only the incorrect items in the logs are identified in the JPM Steps.**

Comment:

√ **Performance Step: 2** Review OST-1021, Attachment 3 for approval.

Standard:

Page 26

- Instruction line item 1 – Should not be N/A. EST-708 needs to be performed due to several RCS flow readings not meeting Acceptance Criteria. (not critical)
- Identifies ONE of the three of the RCS Loop flows out of spec.
  - The 0800 - 1100 readings for FI-414 reads 98.6 which is lower than $\geq$ 99.3%
  - The 0800 - 1100 readings for FI-426 reads 98.9 which is lower than $\geq$ 99.3%
  - The 2000 - 2300 reading for FI-435 reads 98.9 which is lower than $\geq$ 99.3%
- Determines the required action for Technical Specification compliance:
  - Perform EST-708 due to the RCS flow readings not meeting Acceptance Criteria. (As directed by Instruction 1 at top of page.)

    OR
  - TS 3.2.5.c; 2 hrs to restore, or reduce thermal power to less than 5% of RTP w/in next 6 hrs

Comment:

√ **Performance Step: 3** Review OST-1021, Attachment 3 for approval.

Standard: **Page 29**

- Determines that the 2000 – 2300 reading for both Condensate Storage Tank Levels are below Acceptance Criteria.
- TS 3.7.1.3; within 4 hrs restore CST to operable status, or HSB in next 6 hrs; HSD in following 6 hrs.

Comment:

√ **Performance Step: 4** Review OST-1021, Attachment 3 for approval.

Standard: **Page 33**

- Identifies that the 2000 – 2300 reading for Diesel Generator Room 261 temperature, TDG6903A, exceeds Acceptance Criteria.
- PLP-114 Attachment 4, Area Temperature Monitoring; if out of spec for > 8 hr, prepare a report within 30 days to evaluate continued operability of affected equipment.

Evaluator Note:	**Reading does not meet criteria for declaring equipment in the room inoperable.**

Comment:

√ **Performance Step: 5** Review OST-1021, Attachment 3 for approval.

Standard: **Page 33**

- Determines that the 0800 - 1100 reading for Aux RSVR Level are 249.7 and 249.8 ft which is below the 250 ft Acceptance Criteria.
- TS 3.7.5; HSB in next 6 hrs; CSD in following 30 hrs.

Comment:

Terminating Cue:	**After all findings have been reviewed: Evaluation on this JPM is complete.**

Stop Time: __________.

KEY

Discrepancies or problems identified:

Page 26

- Instruction line item 1 – Should not be N/A. EST-708 needs to be performed due to several RCS flow readings not meeting Acceptance Criteria.
- Identifies three of the RCS Loop flows out of spec.
    - The 0800 - 1100 readings for FI-414 reads 98.6 which is lower than >99.3%
    - The 0800 - 1100 readings for FI-426 reads 98.9 which is lower than >99.3%
    - The 2000 - 2300 reading for FI-435 reads 98.9 which is lower than >99.3%
- Perform EST-708 due to the RCS flow readings not meeting Acceptance Criteria. (As directed by Instruction 1 at top of page.)
- TS 3.2.5.1

Page 29

- Determines the 2000 – 2300 reading for both Condensate Storage Tank Levels are below Acceptance Criteria.
- TS 3.7.1.3 Action a

Page 33

- Identifies that the 2000 – 2300 reading for Diesel Generator Room 261 temperature, TDG6903A, exceeds Acceptance Criteria.
- PLP-114 Attachment 4, Area Temperature Monitoring

Page 33

- Identifies that the 0800 - 1100 reading for both Aux RSVR Levels are below Acceptance Criteria.
- TS 3.7.5

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Admin Exam SRO A1-1
Perform Review of Daily Surveillance Requirements Log
OST-1021

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 on Tuesday at 2300 • EDG “A” is synchronized to the grid for a post-maintenance test and has been running at 6.3 MW for 30 minutes. • The Daily Surveillance Logs (OST-1021, Attachment 3) have been completed. • ERFIS Pressurizer Pressures are unavailable.

Initiating Cue:	You are the CRS and have just completed an emergent watch relief due to illness of the scheduled person. Review the OST-1021, Attachment 3 logs. At the conclusion of your review, list any discrepancies or problems if applicable and identify the Tech Spec and applicable actions. Be prepared to discuss any findings with the evaluator.

Name: ___________________________________

Date: _____________

IF any discrepancies or problems are identified, list page number and discrepancy here:

Facility: Harris Nuclear Plant

Task No.: 301005H401

Task Title: Determine Rod Misalignment Using Thermocouples and Evaluate Tech Specs

JPM No.: 2018 NRC Exam Admin JPM SRO A1-2

K/A Reference: G 2.1.7 RO 4.4 SRO 4.7

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 95% power, with a load reduction in progress. During the down power DRPI indication for rod F08 shows a difference of 24 steps higher than the group demand. • The load reduction has been stopped and AOP-001 was entered. • ALB-013-8-5, Computer Alarm Rod Dev/Seq NIS Power Range Tilts alarm is the only MCB alarm received. • I&C investigated and found no obvious electrical problems.

Initiating Cue:	With the information provided complete Attachment 2 of AOP-001, calculate the temperature difference between thermocouple(s) adjacent to the misaligned rod and the average of symmetric thermocouple(s). 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:	AOP-001, Attachment 1, Attachment 2, Rev. 48 Technical Specifications
Handouts:	JPM Cue Sheets Pages 11,12 AOP-001, Attachments 1 and 2, Rev. 48
Time Critical Task:	No
Validation Time:	15 minutes

Critical Task Justification	
Step 2	If the wrong thermocouples are used then none of the results will be correct
Step 4	If the wrong values are selected then none of the results will be correct
Step 5	If the calculation for the averages were incorrect the results will be incorrect
Step 6	If the differences are calculated incorrectly then the candidate may come to the wrong conclusion for Tech Specs
Step 7	If the wrong Tech Spec Action is selected an LCO action could be exceeded

Start Time: __________.

AOP-001

Performance Step: 1 OBTAIN PROCEDURE (provided with handout)

Standard: Obtains AOP-001 and refers to Attachments 1 and 2.

Comment:

AOP-001 Attachment 2 Step 1

✓ **Performance Step: 2** DETERMINE THERMOCOUPLE LOCATION(S) ADJACENT TO THE MISALIGNED ROD USING THE CORE GRID MAP (SHEET 1).

Standard: Using the core grid map (Attachment 2, page 1 of 3), Determines affected thermocouples to be E07, E08, F09, and G08.

Comment: **Note, page 47, AOP-001: E07 does not have symmetric thermocouple locations per Attachment 2.**

AOP-001 Attachment 2 Step 2

Performance Step: 3 CIRCLE LOCATION(S) IN TABLE ABOVE.

Standard: Circles E08, F09, and G08.on the table (Attachment 2, page 2 of 3).

Comment:

AOP-001 Attachment 2 Step 3

✓ **Performance Step: 4**

RECORD the following in the table below:

- Adjacent TC number
- Adjacent TC value using the RVLIS Console, ERFIS, or OSI-PI
- Symmetric TC numbers (not including adjacent TCs)
- Symmetric TC values for all OPERABLE TCs using the RVLIS Console, ERFIS, or OSI-PI

Standard:

Locates RVLIS Console and accesses T/C CORE MAP for Train A and Train B. **(Printout of RVLIS core map provided in handout)**

Records value for Affected TC E07(640°F) and Notes it does not have any Symmetric TC's.

Records value for Affected TC E08 (648°F) and Symmetric TCs H05 (644°F), H11 (652°F), and L08 (642°F).

Records value for Affected TC F09 (644°F) and Symmetric TCs G06 (640°F), and J10 (650°F).

Records value for Affected TC G08 (646°F) and Symmetric TC H09 (642°F).

Comment:

AOP-001 Attachment 2 Step 4

✓ **Performance Step: 5** DETERMINE THE AVERAGE OF SYMMETRIC THERMOCOUPLES, FOR EACH ADJACENT THERMOCOUPLE.

Standard:

Determines (646°F ± **2°F**) for E08's Symmetric TCs

Determines (645°F ± **2°F**) for F09's Symmetric TCs

Determines (642°F) for G08's Symmetric TC

Comment:

EXAMINERS NOTE: **If the candidate includes the adjacent TCs with the Symmetric TC numbers the averages will be wrong and the end result will be that a wrong final difference will be given:**

Determines (646.5°F) for E08's Symmetric TCs

Determines (644.7°F) for F09's Symmetric TCs

Determines (642°F) for G08's Symmetric TCs

AOP-001 Attachment 2 Step 5

✓ **Performance Step: 6** COMPARE EACH ADJACENT THERMOCOUPLE VALUE LISTED TO ITS SYMMETRIC THERMOCOUPLE AVERAGE FOR INDICATION OF A MISALIGNED ROD. (REFER TO ATTACHMENT 1.)

Standard: Calculates difference of **4°F**

2°F for TC E08

1°F for TC F09

4°F for TC G08

(± 2°F between all affected TCs and their symmetric TCs.)

Using AOP-001, Attachment 1 determines since the difference between thermocouples adjacent to the misaligned rod and the average of symmetric thermocouples is < 10°F that a malfunction of Rod Position Indication (DRPI) is occurring.

Circles 1. A malfunction of Rod Position Indication (DRPI) is occurring.

Comment:

Technical Specifications

✓ **Performance Step: 7** OBTAIN AND EVALUATE TECHNICAL SPECIFICATIONS

Standard: Obtains Technical Specifications and refers to LCO 3.1.3.2

Determines that ACTION a. is applicable. (See page 11)

Evaluator Cue:	**After the candidate has completed the calculation for the thermocouples and performed a Technical Specification evaluation.** **END OF JPM**

Terminating Cue:	Difference between each affected thermocouple and it's symmetric thermocouples has been calculated and the Technical Specifications evaluation completed .

Stop Time: _________

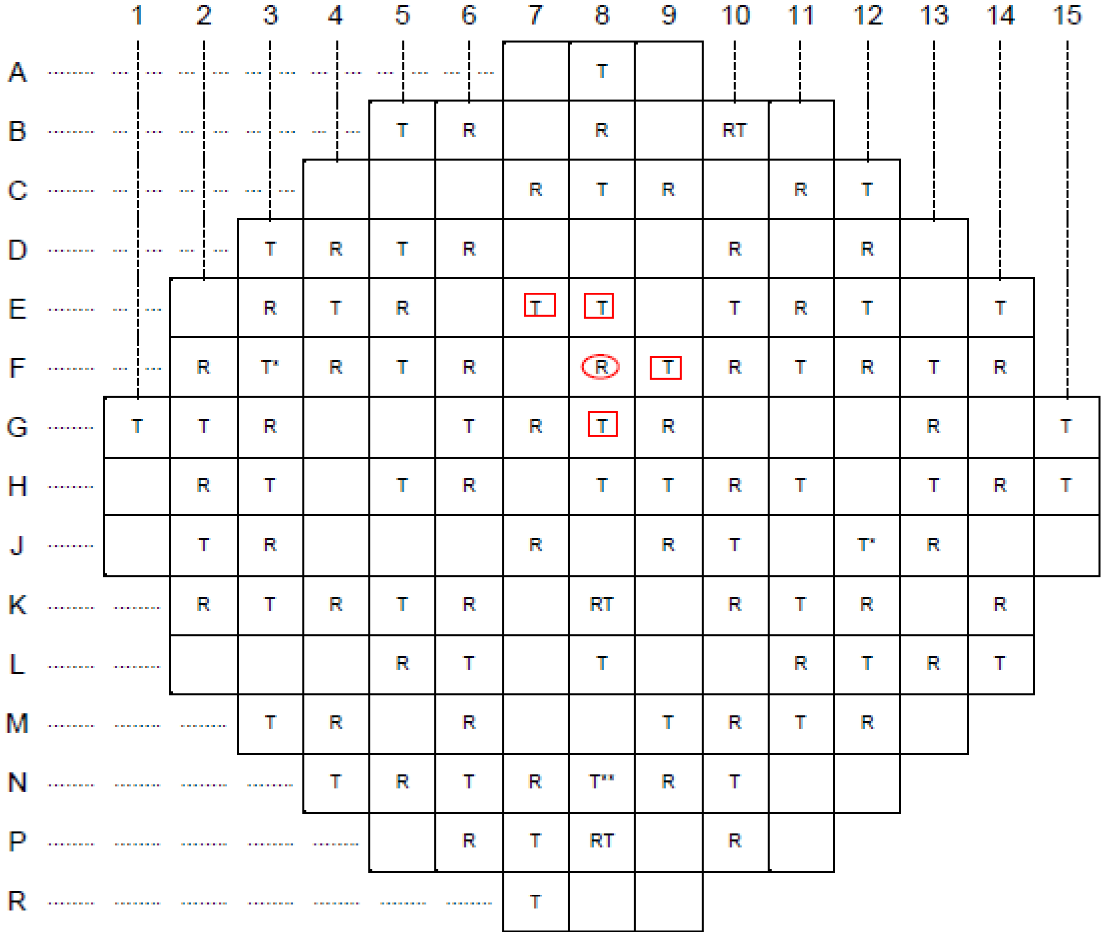
KEY

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 1 of 3

THERMOCOUPLE LOCATIONS



	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A								T							
B					T	R		R		RT					
C							R	T	R		R	T			
D			T	R	T	R				R		R			
E			R	T	R		T	T		T	R	T		T	
F		R	T*	R	T	R		R	T	R	T	R	T	R	
G	T	T	R			T	R	T	R				R		T
H		R	T		T	R		T	T	R	T		T	R	T
J		T	R				R		R	T		T*	R		
K		R	T	R	T	R		RT		R	T	R		R	
L					R	T		T			R	T	R	T	
M			T	R		R			T	R	T	R			
N				T	R	T	R	T**	R	T					
P						R	T	RT		R					
R							T								

R - Control Rod

T - Thermocouple

T* - Thermocouple(s) abandoned by EC 47997 (core location[s] F03, J12)

T** - Thermocouple(s) abandoned by EC 76393 (core location[s] N08)

KEY

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 2 of 3

NOTE

- B10, E07, H08, K08, and P08 have no symmetric locations.
- Symmetric thermocouples are those in the same row.

SYMMETRIC LOCATIONS

GRID	I		II		III		IV	
TRAIN	A	B	A	B	A	B	A	B
	A08				H15			
		G01		G15			R07	
S L	B05			E14		L14		
Y O		C08	H13				N08**	H03
M C		D03	C12				N04	M03
M A	E04	D05		E12	M11	L12		
E T			H11	E08		L08		H05
T I		F05	F11	E10	K11		K05	L06
R O		F03*	F13			N10	N06	K03
I N	G06		F09			J10		
C S		G08			H09			
	G02						J02	P07
					M09	J12*		

* - Thermocouple(s) abandoned by EC 47997 (core location[s] F03, J12)

** - Thermocouple(s) abandoned by EC 76393 (core location[s] N08)

☐ 1. **DETERMINE** thermocouple location(s) adjacent to the misaligned rod using core grid map (Sheet 1).

☐ 2. **CIRCLE** location(s) in Table above.

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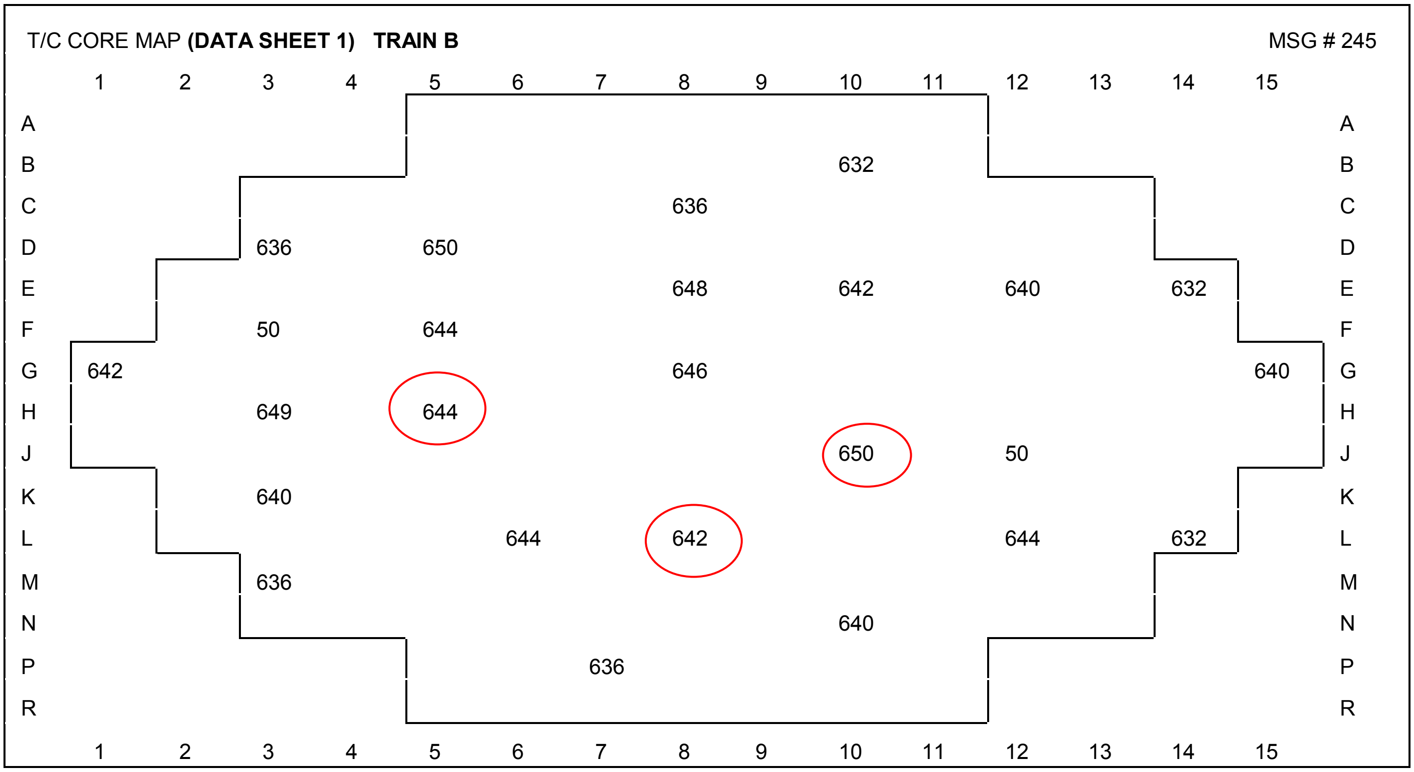
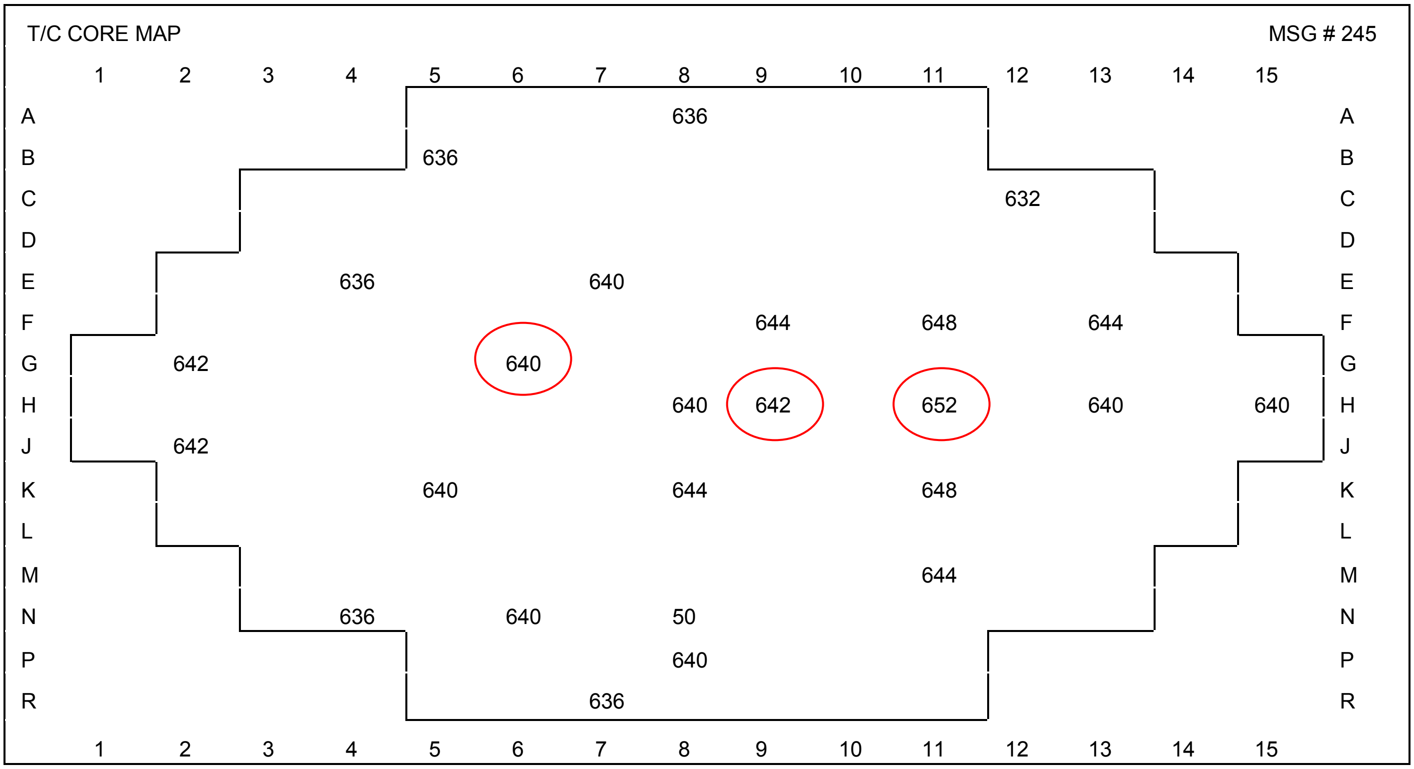
KEY



T/C CORE MAP **(DATA SHEET 1)** **TRAIN B** MSG # 245

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A																A
B										632						B
C								636								C
D			636		650											D
E								648		642		640		632		E
F			50		644											F
G	642							646							640	G
H			649		644											H
J										650		50				J
K			640													K
L						644		642				644		632		L
M			636													M
N										640						N
P							636									P
R																R
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	

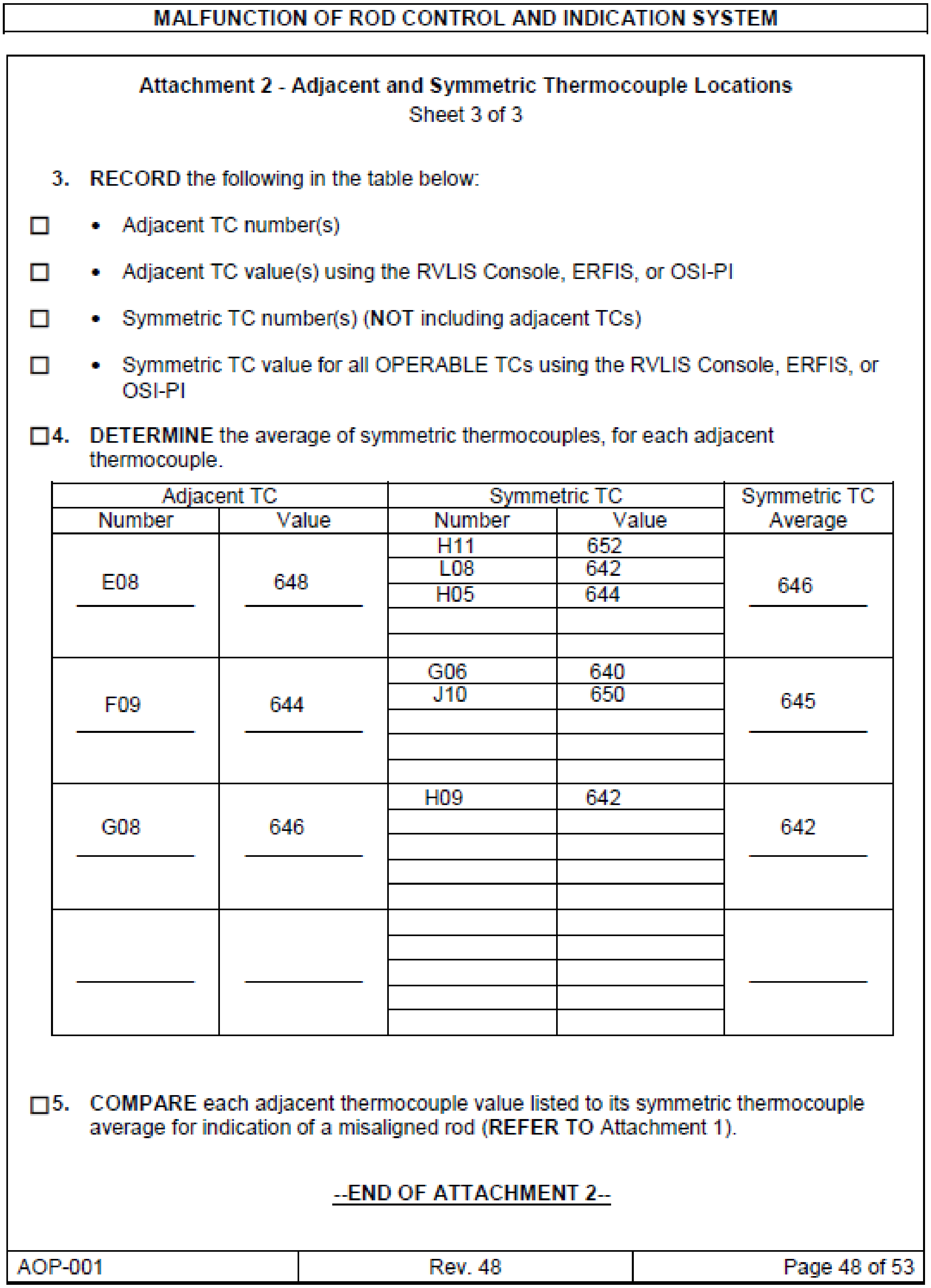
TRAIN A



T/C CORE MAP MSG # 245

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A								636								A
B					636											B
C												632				C
D																D
E				636			640									E
F									644		648		644			F
G		642				640										G
H								640	642		652		640		640	H
J		642														J
K					640			644			648					K
L																L
M											644					M
N				636		640		50								N
P								640								P
R							636									R
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	

KEY



MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations
Sheet 3 of 3

3. **RECORD** the following in the table below:

☐ • Adjacent TC number(s)

☐ • Adjacent TC value(s) using the RVLIS Console, ERFIS, or OSI-PI

☐ • Symmetric TC number(s) (**NOT** including adjacent TCs)

☐ • Symmetric TC value for all OPERABLE TCs using the RVLIS Console, ERFIS, or OSI-PI

☐ 4. **DETERMINE** the average of symmetric thermocouples, for each adjacent thermocouple.

Adjacent TC		Symmetric TC		Symmetric TC Average
Number	Value	Number	Value	
E08	648	H11	652	646
		L08	642	
		H05	644	
F09	644	G06	640	645
		J10	650	
G08	646	H09	642	642

☐ 5. **COMPARE** each adjacent thermocouple value listed to its symmetric thermocouple average for indication of a misaligned rod (**REFER TO** Attachment 1).

--END OF ATTACHMENT 2--

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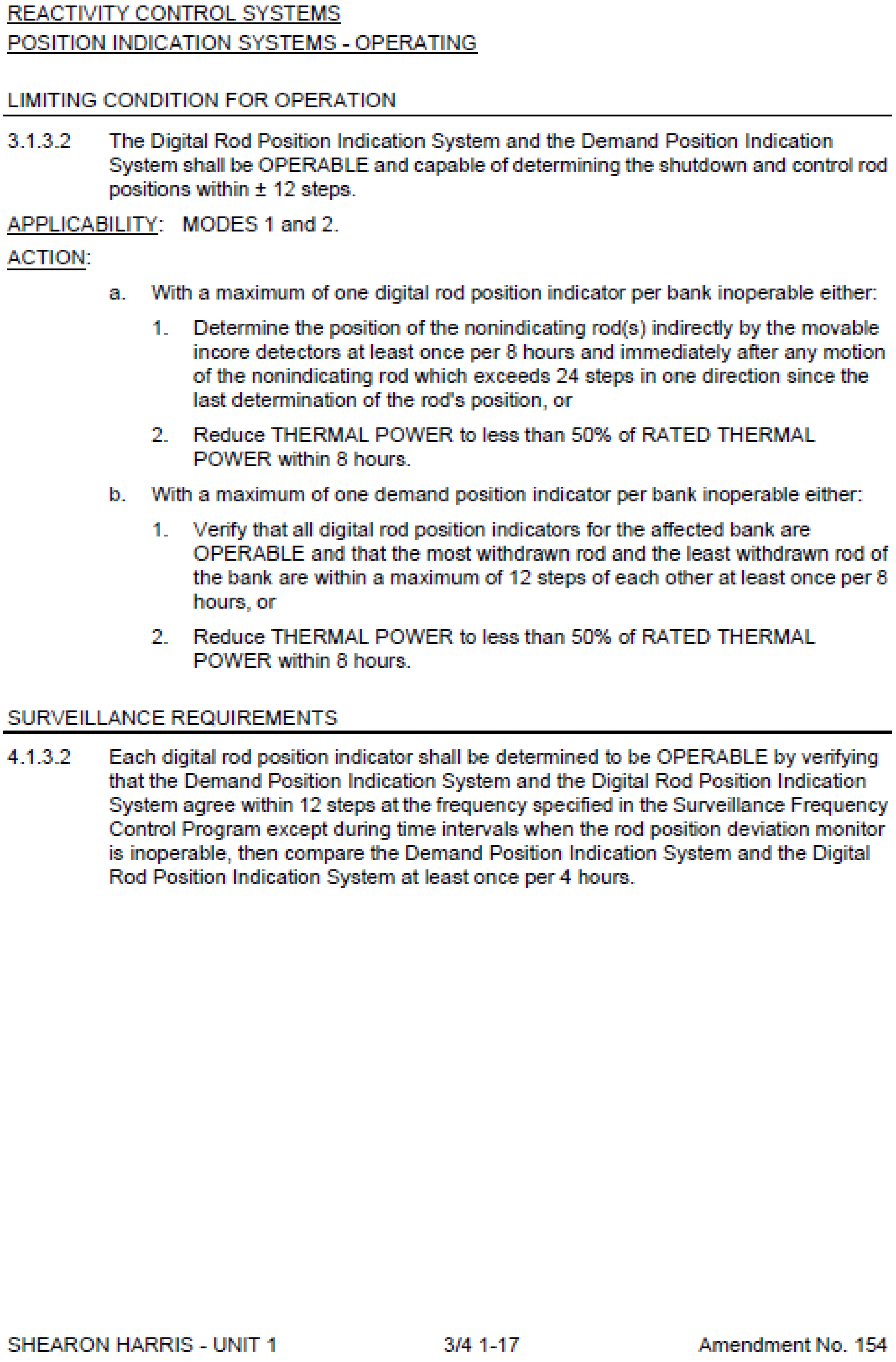
KEY

1. A malfunction of Rod Position Indication (DRPI) is occurring



REACTIVITY CONTROL SYSTEMS
POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the shutdown and control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

  1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

  2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

  1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

  2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at the frequency specified in the Surveillance Frequency Control Program except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

SHEARON HARRIS - UNIT 1 3/4 1-17 Amendment No. 154

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Admin Exam SRO A1-2
Determine Rod Misalignment Using Thermocouples and Evaluate Technical Specifications
AOP-001

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 was at 95% power, with a load reduction in progress. During the down power DRPI indication for rod F08 shows a difference of 24 steps higher than the group demand. - The load reduction has been stopped and AOP-001 was entered. - ALB-013-8-5, Computer Alarm Rod Dev/Seq NIS Power Range Tilts alarm is the only MCB alarm received. - I&C investigated and found no obvious electrical problems.

Initiating Cue:	With the information provided complete Attachment 2 of AOP-001, calculate the temperature difference between thermocouple(s) adjacent to the misaligned rod and the average of symmetric thermocouple(s). 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.

Name: ______________________________________

Date: __________________

Circle the correct response that applies:

1. A malfunction of Rod Position Indication (DRPI) is occurring
2. A Rod Misalignment is occurring

Technical Specification(s) and applicable LCO's that apply: ______________________

____

____

____

T/C CORE MAP **(DATA SHEET 1)** **TRAIN B** MSG # 245

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A																A
B										632						B
C								636								C
D			636		650											D
E								648		642		640		632		E
F			50		644											F
G	642							646							640	G
H			649		644											H
J										650		50				J
K			640													K
L						644		642				644		632		L
M			636													M
N										640						N
P							636									P
R																R
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	

TRAIN A

T/C CORE MAP MSG # 245

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A								636								A
B					636											B
C												632				C
D																D
E				636			640									E
F									644		648		644			F
G		642				640										G
H								640	642		652		640		640	H
J		642														J
K					640			644			648					K
L																L
M											644					M
N				636		640		50								N
P								640								P
R							636									R
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	

Attachment 1 - Indications of Misaligned Rod

Sheet 1 of 1

The table below indicates the variation in plant parameters which may be indicative of rod misalignment. This variation refers to relative changes in indication from a reference condition at which the suspect rod's position was known to be properly aligned. The reference case may be taken from prior operating records, or it may be updated each time the proper rod positioning is verified by in-core measurements. In general, greater misalignment will cause larger variations. Variations in NI channel indication are also affected by the core location of the suspect rod. For example, a misaligned rod that is closest to the N-44 detector should indicate that N-44 flux parameters are abnormal when compared with flux parameters of the other Power Range NI channels. If the parameters below exhibit no abnormal variations with an individual DRPI differing from its group step counter demand position by more than 12 steps, it is probably a rod position indication problem. Quadrant Power Tilt Ratio can be determined by accessing 'GD QPTR' or 'QPTR' and using the highest of ANM9112U - QPTR UPPER RATIO (ANM0112M-118M) or ANM9113L - QPTR LOWER RATIO (ANM0113M-119M).

PLANT PARAMETER	VALUE INDICATIVE OF ROD MISALIGNMENT
Quadrant Power Tilt Ratio (QPTR)	Greater than 1.02
Power Range Instrumentation	Greater than 2% difference between any two channels (**REFER TO** Attachment 4)
Delta Flux Indicators	Greater than 2% difference between any two channels (**REFER TO** Attachment 4)
Core Outlet Thermocouples	Greater than 10°F difference between thermocouples adjacent to the misaligned rod and the average of symmetric thermocouples (**PERFORM** Attachment 2)
Axial Flux Traces (in-core movable detector)	**CONSULT** Reactor Engineering **AND EVALUATE** using in-core movable detectors per EST-922, Control Rod Position Determination Via Incore Instrumentation

--END OF ATTACHMENT 1--

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 1 of 3

THERMOCOUPLE LOCATIONS

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A								T							
B					T	R		R		RT					
C							R	T	R		R	T			
D			T	R	T	R				R		R			
E			R	T	R		T	T		T	R	T		T	
F		R	T*	R	T	R		R	T	R	T	R	T	R	
G	T	T	R			T	R	T	R				R		T
H		R	T		T	R		T	T	R	T		T	R	T
J		T	R				R		R	T		T*	R		
K		R	T	R	T	R		RT		R	T	R		R	
L					R	T		T			R	T	R	T	
M			T	R		R			T	R	T	R			
N				T	R	T	R	T**	R	T					
P						R	T	RT		R					
R							T								

R - Control Rod

T - Thermocouple

T* - Thermocouple(s) abandoned by EC 47997 (core location[s] F03, J12)

T** - Thermocouple(s) abandoned by EC 76393 (core location[s] N08)

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 2 of 3

NOTE

- B10, E07, H08, K08, and P08 have no symmetric locations.
- Symmetric thermocouples are those in the same row.

SYMMETRIC LOCATIONS

GRID	I		II		III		IV	
TRAIN	A	B	A	B	A	B	A	B
SYMMETRIC LOCATIONS	A08				H15			
		G01		G15			R07	
	B05			E14		L14		
		C08	H13				N08**	H03
		D03	C12				N04	M03
	E04	D05		E12	M11	L12		
			H11	E08		L08		H05
		F05	F11	E10	K11		K05	L06
		F03*	F13			N10	N06	K03
	G06		F09			J10		
		G08			H09			
	G02						J02	P07
					M09	J12*		

* - Thermocouple(s) abandoned by EC 47997 (core location[s] F03, J12)

** - Thermocouple(s) abandoned by EC 76393 (core location[s] N08)

☐ 1. **DETERMINE** thermocouple location(s) adjacent to the misaligned rod using core grid map (Sheet 1).

☐ 2. **CIRCLE** location(s) in Table above.

MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM

Attachment 2 - Adjacent and Symmetric Thermocouple Locations

Sheet 3 of 3

3. **RECORD** the following in the table below:

☐ • Adjacent TC number(s)

☐ • Adjacent TC value(s) using the RVLIS Console, ERFIS, or OSI-PI

☐ • Symmetric TC number(s) (**NOT** including adjacent TCs)

☐ • Symmetric TC value for all OPERABLE TCs using the RVLIS Console, ERFIS, or OSI-PI

☐4. **DETERMINE** the average of symmetric thermocouples, for each adjacent thermocouple.

Adjacent TC		Symmetric TC		Symmetric TC Average
Number	Value	Number	Value	
__________	__________			__________
__________	__________			__________
__________	__________			__________
__________	__________			__________

☐5. **COMPARE** each adjacent thermocouple value listed to its symmetric thermocouple average for indication of a misaligned rod (**REFER TO** Attachment 1).

--END OF ATTACHMENT 2--

daFacility: Harris Nuclear Plant
Task No.: 015004H201

Task Title: Perform the Quadrant Power Tilt Ratio Surveillance
JPM No.: 2018 NRC Exam Admin JPM SRO A2

K/A Reference: G2.2.12 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:	• The plant is operating at 90% power when a rod in Control Bank ‘A’ (rod P-10) dropped. • The crew is performing AOP-001, Malfunction Of Rod Control And Indication System. • ERFIS points ANM9112U and ANM9113L have a BAD quality code. HNP IT has been notified and they are evaluating the ERFIS points.

Initiating Cue:	You have been directed to perform a manual QPTR in accordance with OST-1039, CALCULATION OF QPTR, AND evaluate the actions, if any, of the applicable Technical Specification. The Power Range NIS indications are provided. **For the purposes of the examination, there will be no independent verification. Show values of your work.**

Task Standard:	Calculations within required band. Correct Tech Spec actions are identified.
Required Materials:	Calculator
General References:	OST-1039, CALCULATION OF QPTR, Revision 17 Technical Specifications
Handouts:	• OST-1039 • Power Range NI – Current and Voltage Set point Table • Technical Specifications
Time Critical Task:	No
Validation Time:	15 minutes

Critical Step Justification	
Step 9	Must accurately determine the correct calculation based on collecting and inputting either provided data or visual inspection data. The calculation will yield an unsatisfactory QPTR.
Step 10	Must accurately determine the correct calculation based on collecting and inputting either provided data or visual inspection data. The calculation will yield an unsatisfactory QPTR.
Step 11	Must accurately determine the correct calculation based on collecting and inputting either provided data or visual inspection data. The calculation will yield an unsatisfactory QPTR.
Step 14	Must identify that the QPTR upper is outside the band which will make this overall results unsatisfactory.
Step 15	Must determine that QPTR is greater than 1.02 (which is a Tech Spec limit) and that the QPTR is unsatisfactory.
Step 19	Must accurately identify associated Technical Specifications with a QPTR that has exceeded the limits specified in HNP Technical Specifications.

Start Time: __________.

Performance Step: 1 Obtain procedure.

Standard: Reviews procedure.

Evaluator Cue:	**Provide OST-1039.**

Evaluator Note:	**A KEY is provided for your use on JPM prior to candidate pages.**

Comment:

Evaluator Note:	**NOTE: The NI curve numbers provided in this JPM are numbers from the 2018 NRC Exam Frozen Procedures Curve Book folder.**

Procedure Note: Precaution and Limitation 3.1.1 has guidance if performing this OST with one Power Range Channel inoperable.

Performance Step: 2 Completes Prerequisites section:

- Ensure instrumentation needed for the performance of this test is free of deficiencies that affect instrument indication.
- Ensure the most recent Curve F-x-8 is used in the performance of this procedure. (Reference 9.5.7 and 9.5.1)
- Obtain CRS permission to perform this OST.
- Obtain necessary tools and equipment from the following list
  - IBM PC or compatible

Standard:

- Logs F-20-8 revision number : 4
- Initials/signs all blocks

Comment:

Performance Step: 3

- IF Quadrant Power Tilt Ratio Calculation Computer Program is used, THEN PERFORM the following:
  - MARK Step 7.1. Step 2 N/A.
  - MARK Section 7.3 N/A.
  - PERFORM Section 7.2.
- IF manual calculation of the Quadrant Power Tilt Ratio is used, THEN PERFORM the following:
  - MARK Section 7.2 N/A.
  - PERFORM Section 7.3.

Standard:

- Marks Section 7.2 N/A
- Proceeds to Section 7.3

Comment:

OST-1039 Section 7.3 Note prior to step 1

Performance Step: 4

NOTE: The detector current meters on each power range channel drawer are designated as left-upper, right-lower.

Standard:

Reads and place keeps note

Comment:

OST-1039 Section 7.3, Step 1

Performance Step: 5

Prior to reading the value of detector current, VERIFY the meter range/rate switch is in the 400 μA/SLOW position.

Standard:

Prior to reading the value of detector current, VERIFIES the Meter Range/Rate switch is in the 400 μA/SLOW position.

Evaluator Note:	**This information is on the JPM Cue Sheet**

Comment:

OST-1039 Section 7.3, Step 2

Performance Step: 6 RECORD on Attachment 2, in column A, the upper and lower detector currents from all operable power range channels as read on the Nuclear Instrumentation Cabinet.

Standard: Transposes readings from PRNIS Readings Table onto Attachment 2.

Comment:

OST-1039 Section 7.3, Step 3

Performance Step: 7 RECORD on Attachment 2, in column B, the 100% power normalized current for each channel from Curve F-x-8.

Standard: Transposes TOP and BOTTOM 100% current values from the Curve Book provided.

Comment:

OST-1039 Section 7.3, Note prior to Step 4

Performance Step: 8 NOTE: When recording all fractions and ratios, record to four decimal places, dropping the fifth and subsequent decimal places.

Standard: Reads and place keeps note

Comment:

OST-1039 Section 7.3, Step 4

✓ **Performance Step: 9** Divide values in Column A by the respective normalized current in Column B and record the result in Column C as the Normalized Fraction.

Standard: Divides each Upper and Lower reading by the respective 100% normalized current value and records in Column C.

Comment:

OST-1039 Section 7.3, Step 5

✓ **Performance Step: 10** CALCULATE the average value for the upper and the lower Normalized Fractions as follows:

- ADD the Normalized Fraction in each section of column C, recording the sum in the space provided.
- DIVIDE the sum obtained in Step 7.3.5.a by the number of operable NI channels, recording the result in column D of Attachment 2.

Standard: Adds all Normalized Fractions for the same plane and records the sum in the space provided.

Divides by the sum by four and records result in Column D.

Comment:

OST-1039 Section 7.3, Step 6

✓ **Performance Step: 11** Using the formula and values from Attachment 2, CALCULATE the Upper and Lower Ratios.

Standard:

- Divides the Maximum Normalized Fraction by the Average Normalized Fraction on each plane.
- Determines the UPPER ratio is ≥ 1.02
- Determines the LOWER ratio is ≥ 1.02

Evaluator Note:	**The applicant may inform the CRS as soon as any calculation is > 1.02. If so, acknowledge and direct applicant to complete Attachment 2.**

Comment:

	OST-1039 Section 7.3, Step 7
Performance Step: 12	PERFORM independent verification of all calculations made on Attachment 2.
Standard:	Requests Independent Verifier.
Evaluator Cue:	**If necessary, repeat Initiating Cue: For the purpose of this examination, there will be no independent verification of your work.**
Comment:	**Candidate may choose to check calculations.**

	OST-1039 Section 7.3, Note prior to Step 8
Performance Step: 13	NOTE: The upper ratio or the lower ratio, whichever is greater, is the quadrant power tilt ratio (QPTR).
Standard:	Reads and place keeps note
Comment:	

	OST-1039 Section 7.3, Step 8
✓ **Performance Step: 14**	RECORD QPTR:
Standard:	Records QPTR value as 1.0469 to 1.0479 (N43 LOWER) **Identifies Lower as outside the band**
Comment:	**Acceptable band is +/- 5% (rounded to .0005).** **UPPER calculated band is 1.0291 to 1.0301** **LOWER calculated band is 1.0469 to 1.0479**

OST-1039 Section 7.3, Step 9

✓ **Performance Step: 15** CHECK QPTR is less than or equal to 1.02.

Standard: **Identifies Upper and Lower QPTR's are greater than 1.02 and QPTR is unacceptable**

Comment:

OST-1039 Section 7.3, Note prior to Step 10

Performance Step: 16 NOTE: ERFIS turn on codes used to obtain ERFIS QPTR values include "QPTR" and "GD QPTR".

Standard: Reads and place keeps note

Comment:

OST-1039 Section 7.3, Step 10

Performance Step: 17 IF the ERFIS calculated QPTR value is available, THEN COMPARE OST-1039 results to the ERFIS QPTR calculated output as a quality check.

Standard: Request status of ERFIS calculated QPTR value, and N/A's step 7.3.10 when notified ERFIS QPTR is not available.

Evaluator Note:	**This information is on the JPM Cue Sheet**

Evaluator Cue:

Comment:

OST-1039 Section 7.3, Step 11

Performance Step: 18 IF any ERFIS QPTR quality codes do not have a good quality code or the higher of ANM9112U or ANM9113L do not approximate the value for QPTR determined above, THEN CONTACT HNP IT to investigate.

Standard: Request if notification of the status of the ERFIS calculated QPTR value to HNP IT has been completed.

Evaluator Note:	**This information is on the JPM Cue Sheet**

Comment:

Evaluator Cue:	**Continue with the evaluation of the Technical Specification LCO(s) that is(are) in effect.**

✓ **Performance Step: 19** Identify the Technical Specification LCOs that would be in effect.

Standard: Identifies that Technical Specification 3.2.4, Quadrant Power Tilt Ratio has been exceeded

- Identifies the following ACTION statements to be implemented and the required time limitation (see page 12)
    - 3.2.4.a.1 1 hour

      a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

      1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.
    - 3.2.4.a.2 2 hours, reduce thermal power to ≤ 85%
        - (5% x 3% = 15% 100% - 15% = 85%)

      2. Within 2 hours either:

      a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.
    - 3.2.4.a.3 24 hours

      3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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; and

Evaluator Note:	Technical Specification 3.2.4.a.4 is not required to be identified since no direction is provided in the cue for raising thermal power.

Comment:

Terminating Cue:	**After the calculation and Tech Spec Evaluation has been completed:** **Evaluation on this JPM is complete.**

STOP Time: __________.

KEY

Record QPTR = **1.0474** Acceptable band is +/- 5% (rounded to .0005) 1.0469 to 1.0479

CHECK QPTR is less than or equal to 1.02 (circle) YES / [✓ NO]

	A	B	C	D
UPPER DETECTOR	UPPER DETECTOR CURRENT	UPPER 100% POWER NORMALIZED CURRENT	UPPER NORMALIZED FRACTION (NOTE 1)	AVERAGE UPPER NORMALIZED FRACTION
N-41	145.6	150.5	0.9674	0.9467
N-42	162.5	172.8	0.9403	
N-43	189.8	194.7	0.9748	
N-44	138.4	153.0	0.9045	
		SUM	3.7870	

$$\text{Upper Ratio} = \frac{\text{Maximum Upper Normalized Fraction}}{\text{Average Upper Normalized Fraction}} = \frac{0.9748}{0.9467} = 1.0296^{*}$$

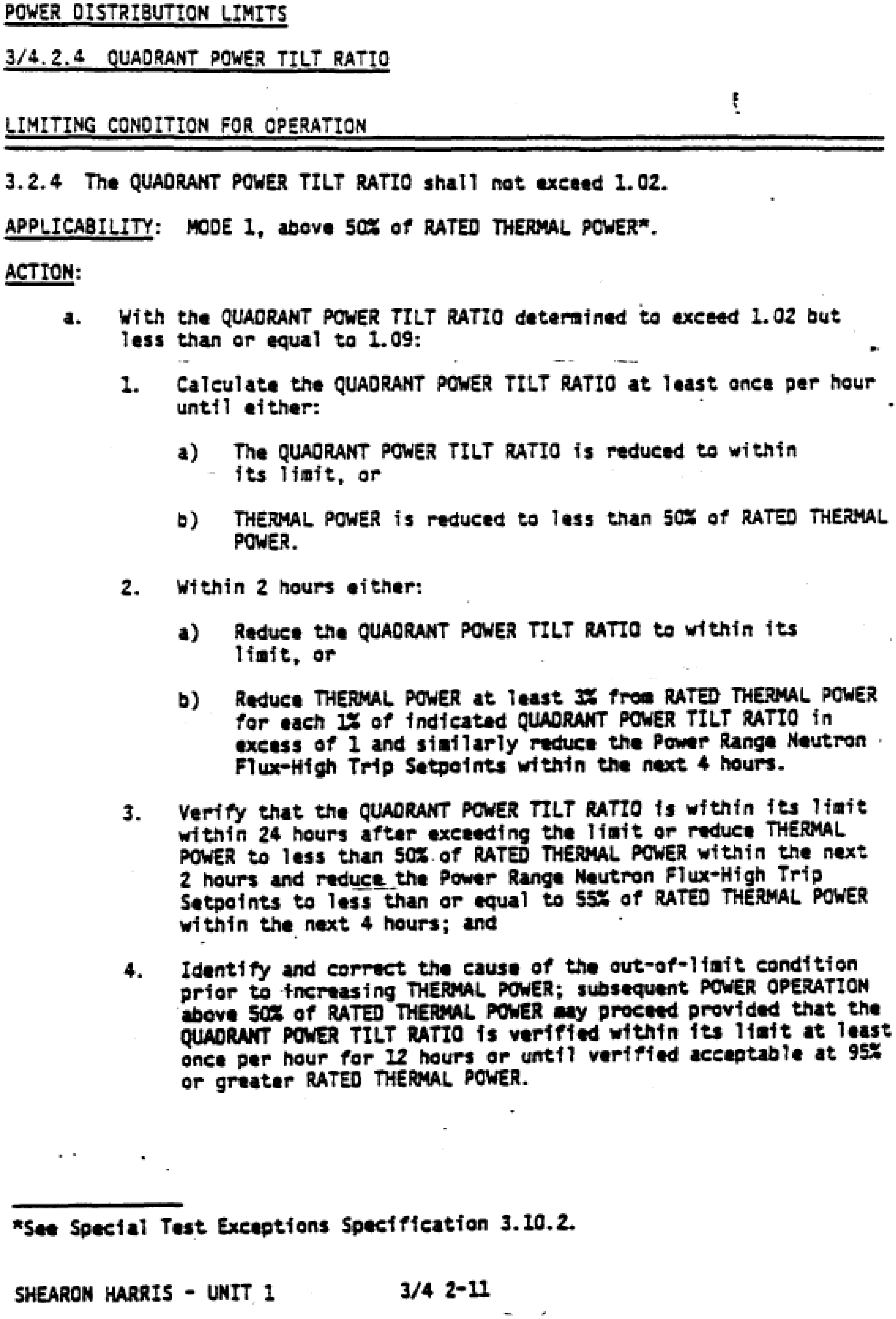
* Standard for this calculation is 1.0291 to 1.0301

	A	B	C	D
LOWER DETECTOR	LOWER DETECTOR CURRENT	LOWER 100% POWER NORMALIZED CURRENT	LOWER NORMALIZED FRACTION (NOTE 1)	AVERAGE LOWER NORMALIZED FRACTION
N-41	159.6	167.1	0.9551	0.9400
N-42	172.1	191.1	0.9005	
N-43	205.3	208.5	0.9846	
N-44	165.2	179.6	0.9198	
		SUM	3.7600	

$$\text{Lower Ratio} = \frac{\text{Maximum Lower Normalized Fraction}}{\text{Average Lower Normalized Fraction}} = \frac{0.9846}{0.9400} = 1.0474^{**}$$

** Standard for this calculation is 1.0469 to 1.0479

KEY



POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

    a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

    b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

    a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

    b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

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*See Special Test Exceptions Specification 3.10.2.

SHEARON HARRIS - UNIT 1 3/4 2-11

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Admin JPM SRO A2
Perform a Quadrant Power Tilt Ratio Surveillance

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 90% power when a rod in Control Bank 'A' (P-10) dropped. - The crew is performing AOP-001, MALFUNCTION OF ROD CONTROL AND INDICATION SYSTEM. - ERFIS points ANM9112U and ANM9113L have a BAD quality code. HNP IT has been notified and they are evaluating the ERFIS points.

Initiating Cue:	You have been directed to perform a <u>manual</u> QPTR in accordance with OST-1039, CALCULATION OF QPTR, AND evaluate the actions, if any, of the applicable Technical Specification (write response below). The Power Range NIS indications are provided. **For the purposes of the examination, there will be no independent verification. Show values of your work.**

Name: ____

Date: __________________

Technical Specification(s) and applicable LCO(s) that apply: ______________________

____

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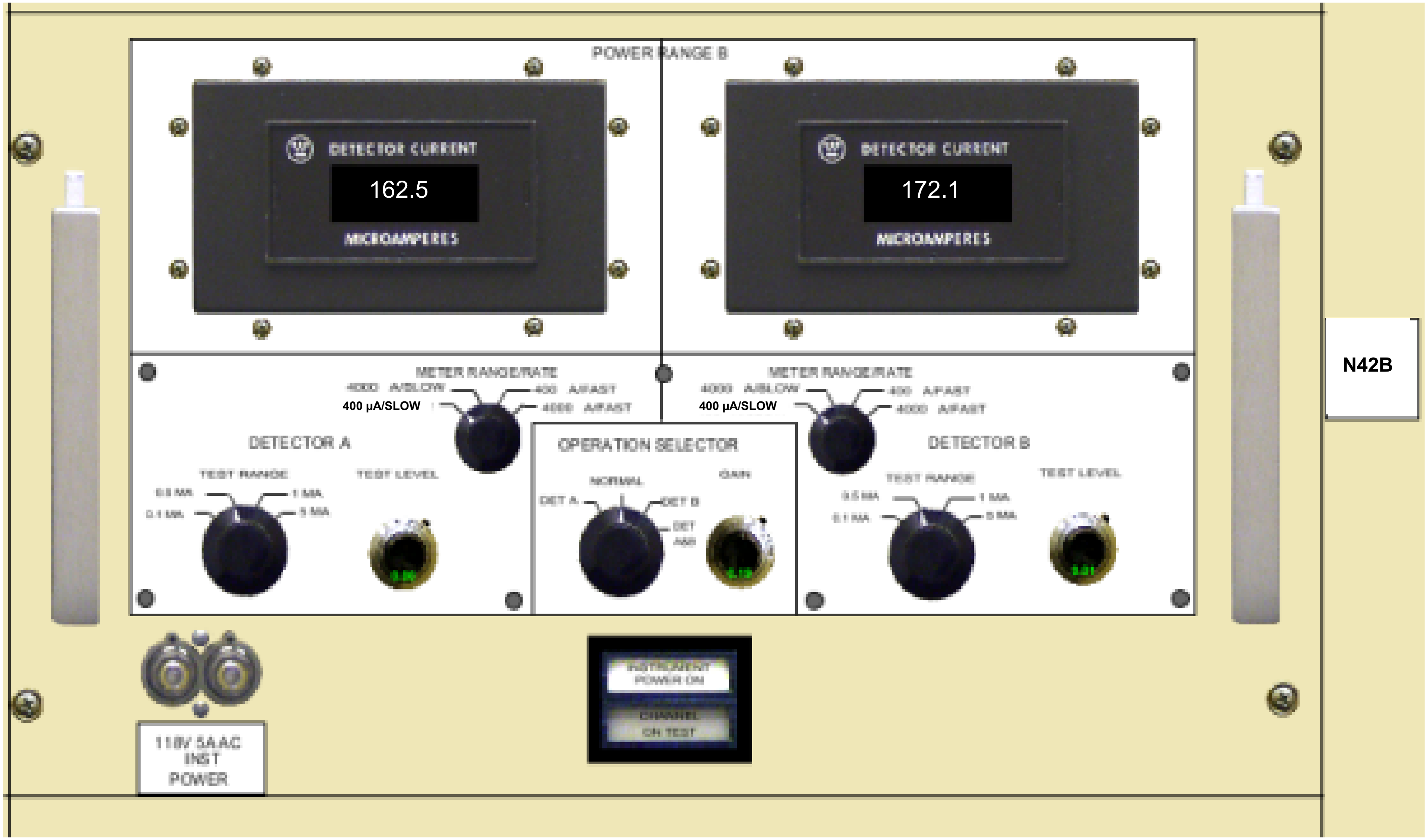
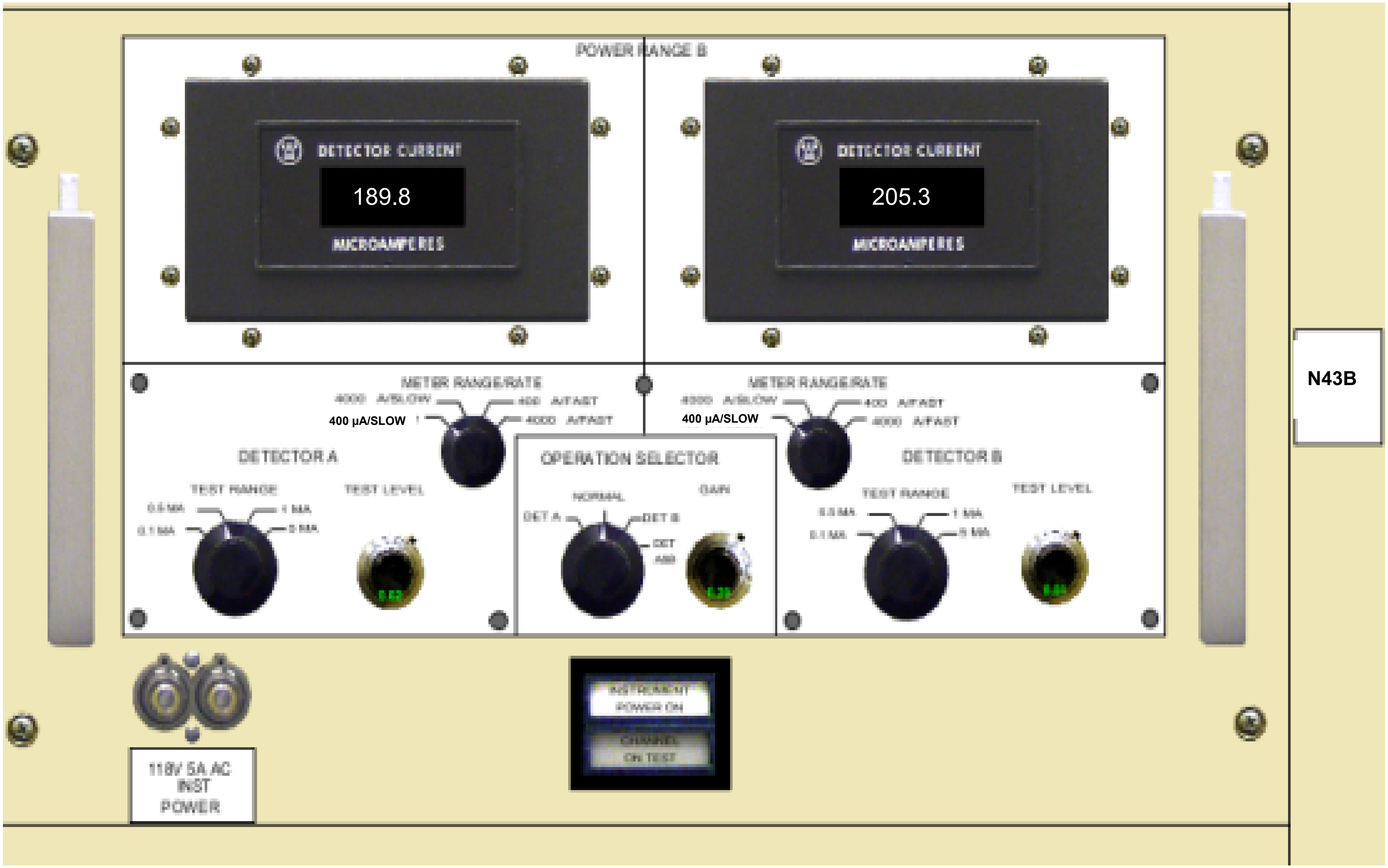
JPM CUE SHEET

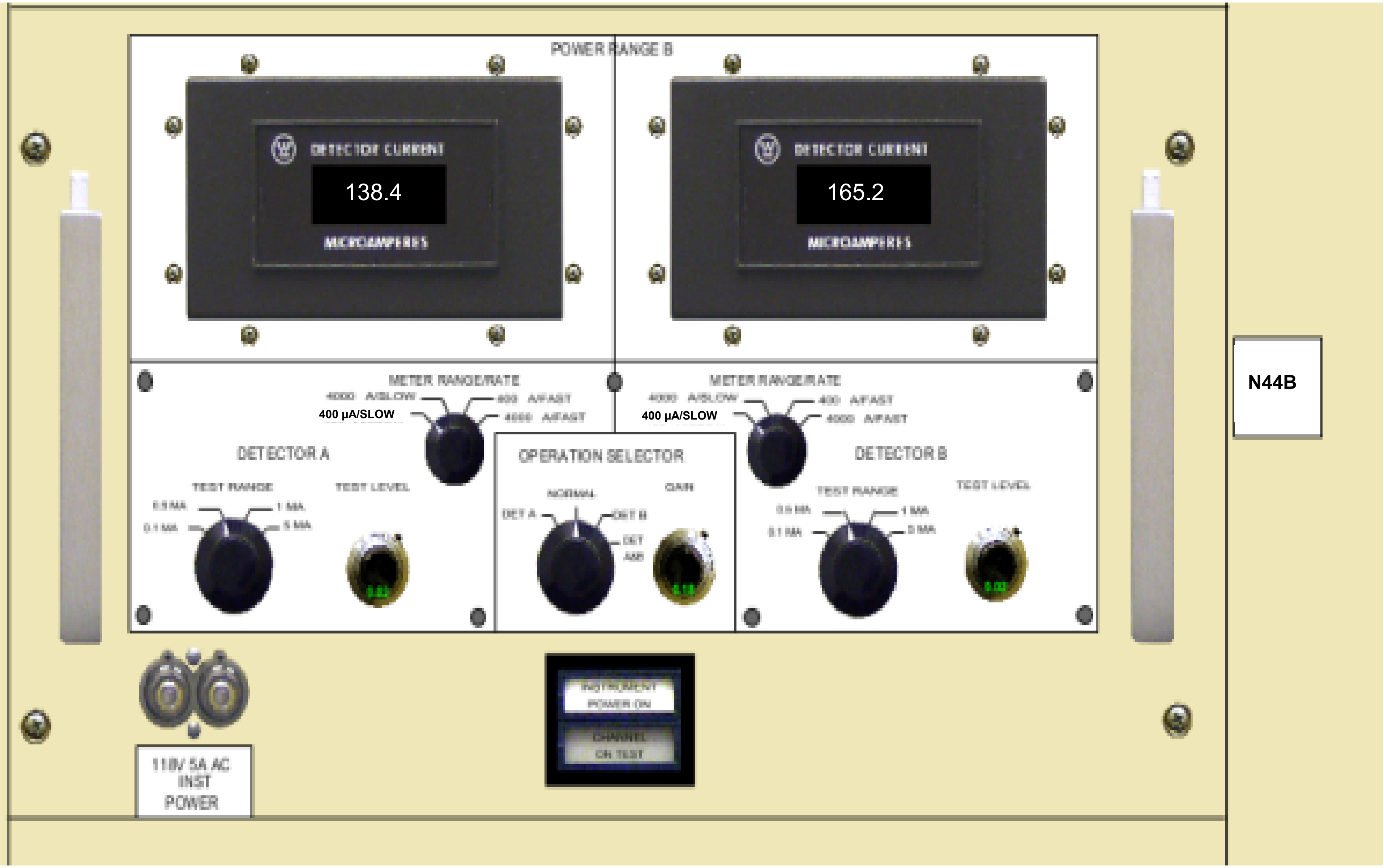
JPM CUE SHEET



JPM CUE SHEET



JPM CUE SHEET



Facility:	Harris Nuclear Plant	Task No.:	341021H102
Task Title:	Complete review and approval of OP-120.07, Attachment 3 Waste Gas Decay Tank Release Log	JPM No.:	2018 NRC Exam Admin JPM SRO A3
K/A Reference:	G.2.3.4 RO 3.2 SRO 3.7	**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 OP-120.07, Section 8.39, Venting a Gas Decay Tank with Waste Gas System in Service, is in progress. OP-120.07, Attachment 3, Waste Gas Decay Tank Release Log, pre-release section has just been completed

Initiating Cue:	You are the CRS and have been asked to review and approve the just completed copy of OP-120.07, Attachment 3. Completely review the procedure and note any problems in the spaces provided.

Task Standard:	All errors (3) identified
Required Materials:	None
General References:	OP-120.07, Attachment 3, Waste Gas Decay Tank Release Log
Handout:	Completed Batch Gaseous Effluent Permit Partially completed OP-120.07, Attachment 3 with the errors incorporated.
Time Critical Task:	No
Validation Time:	15 minutes

Critical Task Justification	
Step 2	Must determine proper flow rates in order to ensure the actual dose release remains less than the calculate amount of dose per the Effluent release permit. **(Item 1)**
Step 3	Must determine correct date and time the sample was taking in order to ensure the actual dose release remains less than the calculate amount of dose per the Effluent release permit. **(Item 2)**
Step 4	Must comply with procedure use and adherence standards and correctly perform the appropriate procedure section due to possibility of short-lived isotopes in the Waste Gas Header. **(Item 3)**

START TIME: ___________

Performance Step: 1 Reviews the completed OP-120.07 for a Waste Gas Decay Tank Release.

Standard: Ensures proper conditions, signatures/initials, and may verify the current revision of the procedure

Comment:

✓ **Performance Step: 2** Review the completed OP-120.07 and compares REM-3546 PIG Monitor Background reading to RM-11 display and determines the value was not properly transferred to Attachment 3. Identifies one of three (3) discrepancy items

Standard: **ITEM 1: Attachment 3 (Sheet 1 of 3 number 5) – REM-3546 PIG (4GG793) WPB Vent Stack 5 Monitor Background reading is <u>incorrectly transcribed from the information displayed on RM-11.</u>**

Comment:

✓ **Performance Step: 3** Review the completed OP-120.07 and determines that RCDT Vent position is incorrectly N/A'd

Standard: **ITEM 2: Attachment 3 (Sheet 1 of 3 number 15) – RCDT Vent position is <u>incorrectly N/A'd based on the OP-120.07 section in progress per the initial conditions.</u>**

Comment:

✓ **Performance Step: 4**	Review RM-11 RM-WV-3546-1 WRGM screen displays and compares reading to values from the Batch Gaseous Effluent Permit Pre-release data and determines the High (Max) Setpoint is incorrect entered in to RM-11. Identifies three of three (3) discrepancy items
Standard:	**ITEM 3: Attachment 3 (Sheet 2 of 3 number 19) – REM-3546-1 WRGM Permit Values is <u>incorrectly transcribed from the batch gaseous Effluent Permit provided by chemistry.</u>**
Comment:	
Performance Step: 5	Review the completed OP-120.07
Standard:	Returns the procedure unsigned and has identified three errors during the review for pre-release approval.
Comment:	

Evaluator Note and Terminating Cue:	**When the procedure is returned: Evaluation on this JPM is complete.**

STOP TIME: _______________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Admin JPM SRO A3 - Complete review and approval of OP-120.07, Attachment 3 Waste Gas Decay Tank Release Log

OP-120.07, Attachment 3

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 OP-120.07, Section 8.39, Venting a Gas Decay Tank with Waste Gas System in Service, is in progress. OP-120.07, Attachment 3, Waste Gas Decay Tank Release Log, pre-release section has just been completed

Initiating Cue:	You are the CRS and have been asked to review and approve the just completed copy of OP-120.07, Attachment 3. Completely review the procedure and note any problems in the spaces provided.

NAME ____ DATE ______________________

IF any discrepancies were identified in the review of OP-120.07 list them on the lines below

____

____

____

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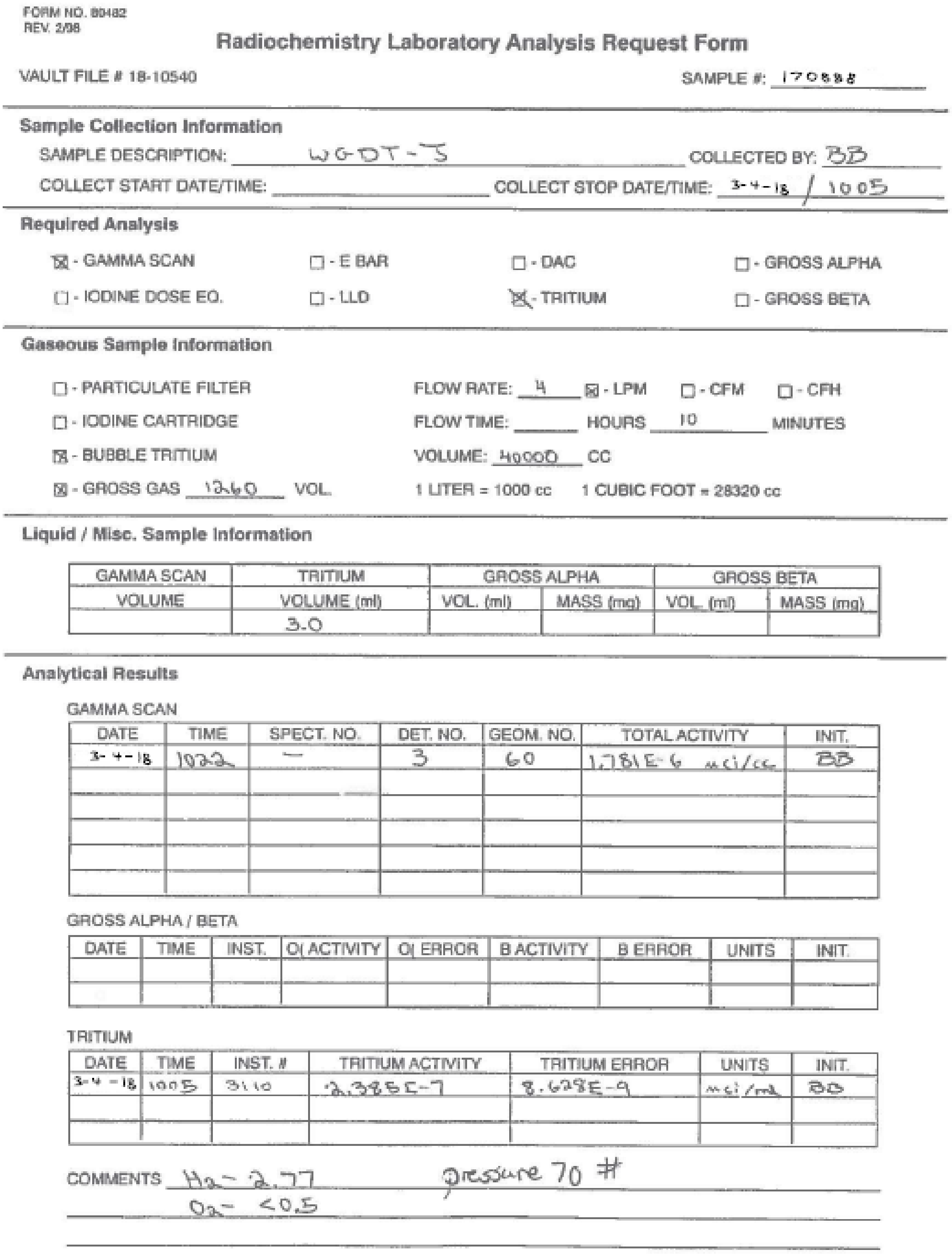
____

JPM CUE SHEET



FORM NO. 80482
REV. 2/98

Radiochemistry Laboratory Analysis Request Form

VAULT FILE # 18-10540

SAMPLE #: 170888

Sample Collection Information

SAMPLE DESCRIPTION: WGDT-J

COLLECTED BY: BB

COLLECT START DATE/TIME: ______

COLLECT STOP DATE/TIME: 3-4-18 / 1005

Required Analysis

☒ - GAMMA SCAN ☐ - E BAR ☐ - DAC ☐ - GROSS ALPHA

☐ - IODINE DOSE EQ. ☐ - LLD ☒ - TRITIUM ☐ - GROSS BETA

Gaseous Sample Information

☐ - PARTICULATE FILTER

☐ - IODINE CARTRIDGE

☒ - BUBBLE TRITIUM

☒ - GROSS GAS 1260 VOL.

FLOW RATE: 4 ☒ - LPM ☐ - CFM ☐ - CFH

FLOW TIME: ______ HOURS 10 MINUTES

VOLUME: 40000 CC

1 LITER = 1000 cc 1 CUBIC FOOT = 28320 cc

Liquid / Misc. Sample Information

GAMMA SCAN	TRITIUM	GROSS ALPHA		GROSS BETA	
VOLUME	VOLUME (ml)	VOL. (ml)	MASS (mg)	VOL. (ml)	MASS (mg)
	3.0				

Analytical Results

GAMMA SCAN

DATE	TIME	SPECT. NO.	DET. NO.	GEOM. NO.	TOTAL ACTIVITY	INIT.
3-4-18	1022	—	3	60	1.781E-6 µci/cc	BB

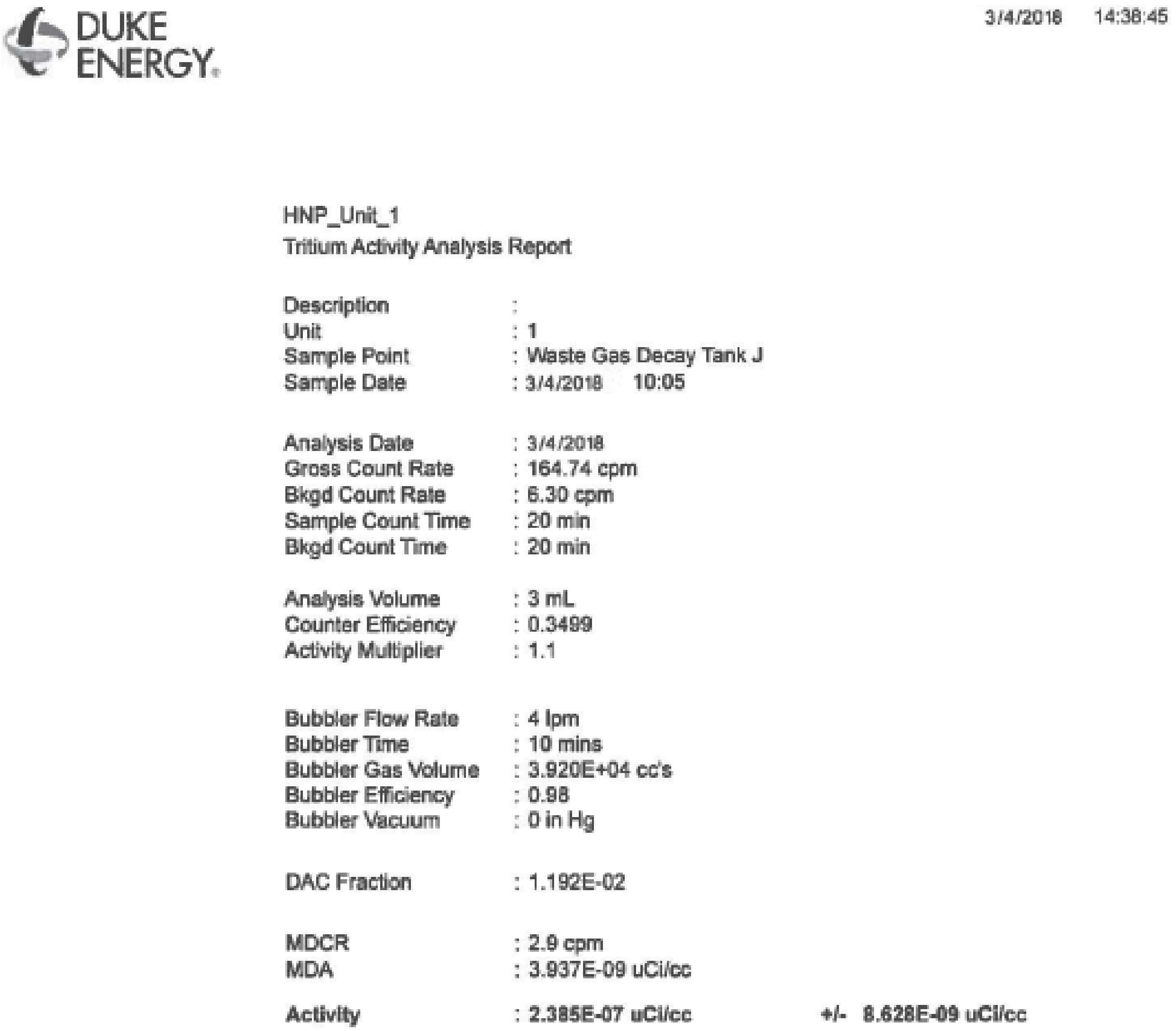
GROSS ALPHA / BETA

DATE	TIME	INST.	O(ACTIVITY	O(ERROR	B ACTIVITY	B ERROR	UNITS	INIT.

TRITIUM

DATE	TIME	INST. #	TRITIUM ACTIVITY	TRITIUM ERROR	UNITS	INIT.
3-4-18	1005	3110	2.385E-7	8.628E-9	µci/ml	BB

COMMENTS H_2 - 2.77 pressure 70 #
O_2 - <0.5



DUKE ENERGY

3/4/2018 14:38:45

HNP_Unit_1
Tritium Activity Analysis Report

Description :
Unit : 1
Sample Point : Waste Gas Decay Tank J
Sample Date : 3/4/2018 10:05

Analysis Date : 3/4/2018
Gross Count Rate : 164.74 cpm
Bkgd Count Rate : 6.30 cpm
Sample Count Time : 20 min
Bkgd Count Time : 20 min

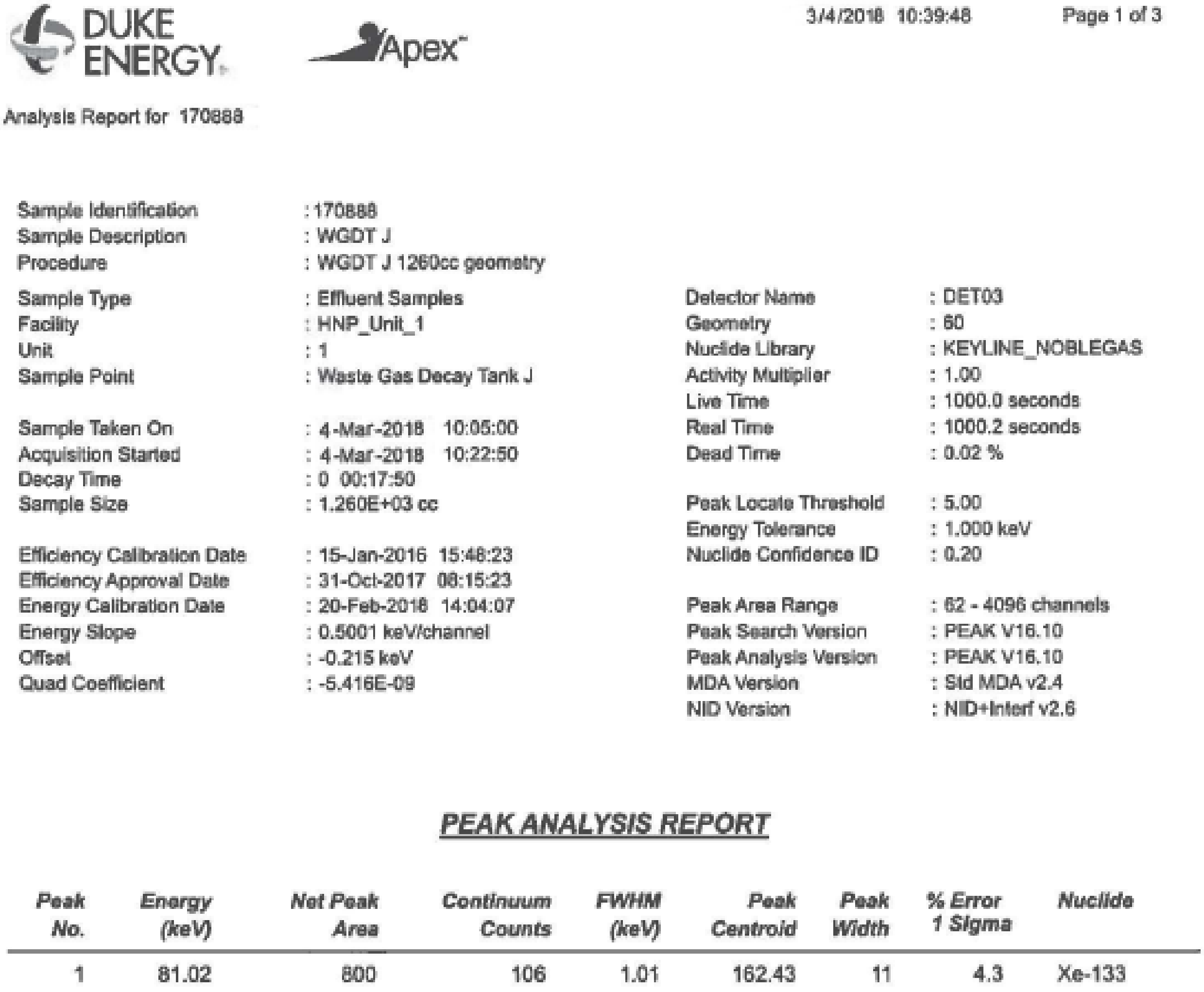
Analysis Volume : 3 mL
Counter Efficiency : 0.3499
Activity Multiplier : 1.1

Bubbler Flow Rate : 4 lpm
Bubbler Time : 10 mins
Bubbler Gas Volume : 3.920E+04 cc's
Bubbler Efficiency : 0.98
Bubbler Vacuum : 0 in Hg

DAC Fraction : 1.192E-02

MDCR : 2.9 cpm
MDA : 3.937E-09 uCi/cc

Activity : 2.385E-07 uCi/cc +/- 8.628E-09 uCi/cc



DUKE ENERGY

Apex™

3/4/2018 10:39:48 Page 1 of 3

Analysis Report for 170888

Sample Identification : 170888
Sample Description : WGDT J
Procedure : WGDT J 1260cc geometry

Sample Type : Effluent Samples
Facility : HNP_Unit_1
Unit : 1
Sample Point : Waste Gas Decay Tank J

Sample Taken On : 4-Mar-2018 10:05:00
Acquisition Started : 4-Mar-2018 10:22:50
Decay Time : 0 00:17:50
Sample Size : 1.260E+03 cc

Efficiency Calibration Date : 15-Jan-2016 15:48:23
Efficiency Approval Date : 31-Oct-2017 08:15:23
Energy Calibration Date : 20-Feb-2018 14:04:07
Energy Slope : 0.5001 keV/channel
Offset : -0.215 keV
Quad Coefficient : -5.416E-09

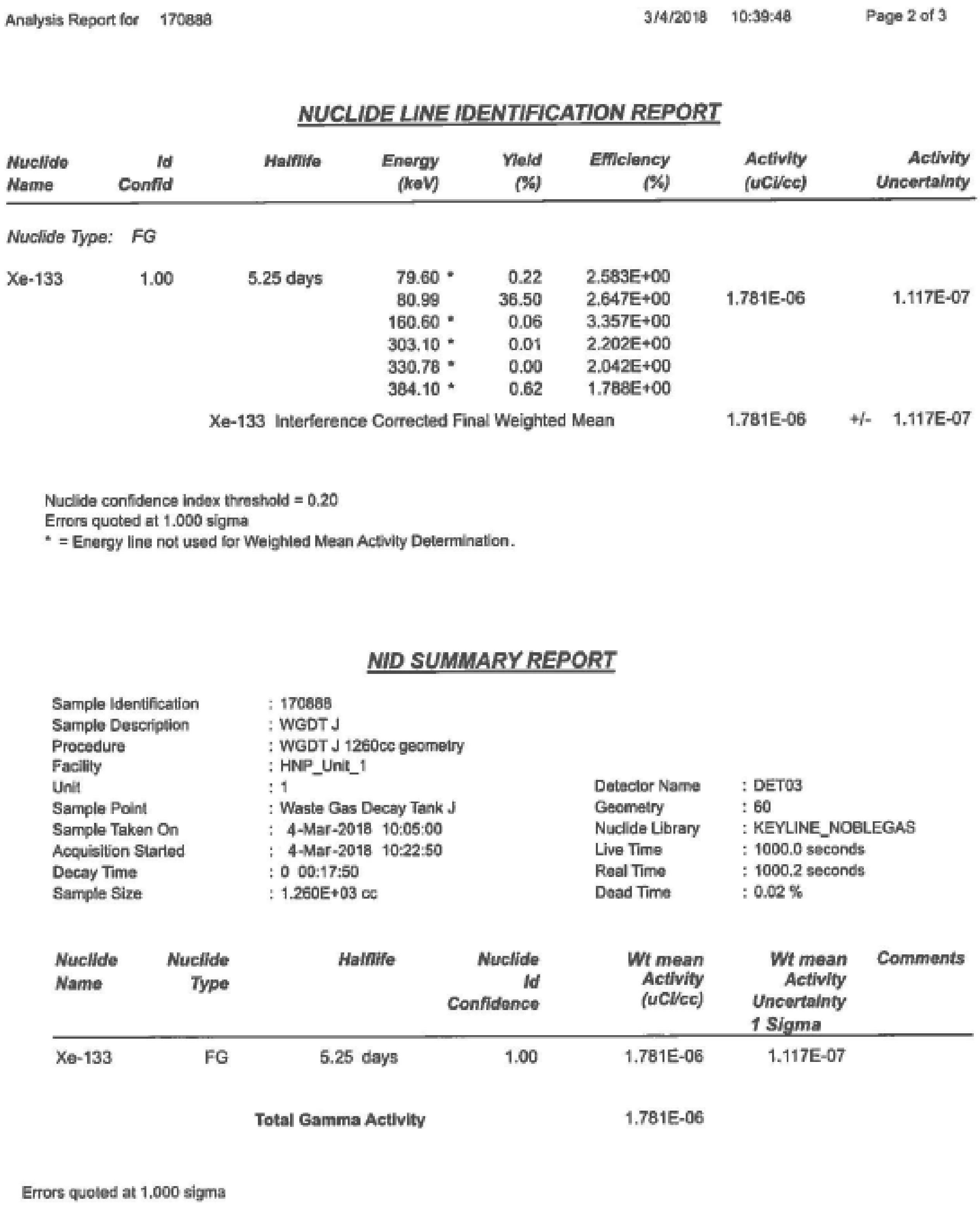
Detector Name : DET03
Geometry : 60
Nuclide Library : KEYLINE_NOBLEGAS
Activity Multiplier : 1.00
Live Time : 1000.0 seconds
Real Time : 1000.2 seconds
Dead Time : 0.02 %

Peak Locate Threshold : 5.00
Energy Tolerance : 1.000 keV
Nuclide Confidence ID : 0.20

Peak Area Range : 62 - 4096 channels
Peak Search Version : PEAK V16.10
Peak Analysis Version : PEAK V16.10
MDA Version : Std MDA v2.4
NID Version : NID+Interf v2.6

PEAK ANALYSIS REPORT

Peak No.	*Energy (keV)*	*Net Peak Area*	*Continuum Counts*	*FWHM (keV)*	*Peak Centroid*	*Peak Width*	*% Error 1 Sigma*	*Nuclide*
1	81.02	800	106	1.01	162.43	11	4.3	Xe-133



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Analysis Report for    170888                                    3/4/2018    10:39:48         Page 2 of 3


                            NUCLIDE LINE IDENTIFICATION REPORT

Nuclide        Id           Halflife        Energy      Yield     Efficiency       Activity        Activity
Name        Confid                           (keV)        (%)            (%)       (uCi/cc)     Uncertainty

Nuclide Type:  FG

Xe-133        1.00          5.25 days       79.60 *      0.22     2.583E+00
                                            80.99       36.50     2.647E+00      1.781E-06       1.117E-07
                                           160.60 *      0.06     3.357E+00
                                           303.10 *      0.01     2.202E+00
                                           330.78 *      0.00     2.042E+00
                                           384.10 *      0.62     1.788E+00

                    Xe-133  Interference Corrected Final Weighted Mean             1.781E-06    +/-  1.117E-07


       Nuclide confidence index threshold = 0.20
       Errors quoted at 1.000 sigma
       * = Energy line not used for Weighted Mean Activity Determination.


                                   NID SUMMARY REPORT

       Sample Identification        : 170888
       Sample Description           : WGDT J
       Procedure                    : WGDT J 1260cc geometry
       Facility                     : HNP_Unit_1
       Unit                         : 1                             Detector Name     : DET03
       Sample Point                 : Waste Gas Decay Tank J        Geometry          : 60
       Sample Taken On              :  4-Mar-2018  10:05:00         Nuclide Library   : KEYLINE_NOBLEGAS
       Acquisition Started          :  4-Mar-2018  10:22:50         Live Time         : 1000.0 seconds
       Decay Time                   : 0  00:17:50                   Real Time         : 1000.2 seconds
       Sample Size                  : 1.260E+03 cc                  Dead Time         : 0.02 %

       Nuclide      Nuclide          Halflife        Nuclide        Wt mean         Wt mean      Comments
       Name            Type                               Id       Activity        Activity
                                                   Confidence       (uCi/cc)     Uncertainty
                                                                                  1 Sigma

       Xe-133            FG          5.25  days          1.00      1.781E-06       1.117E-07

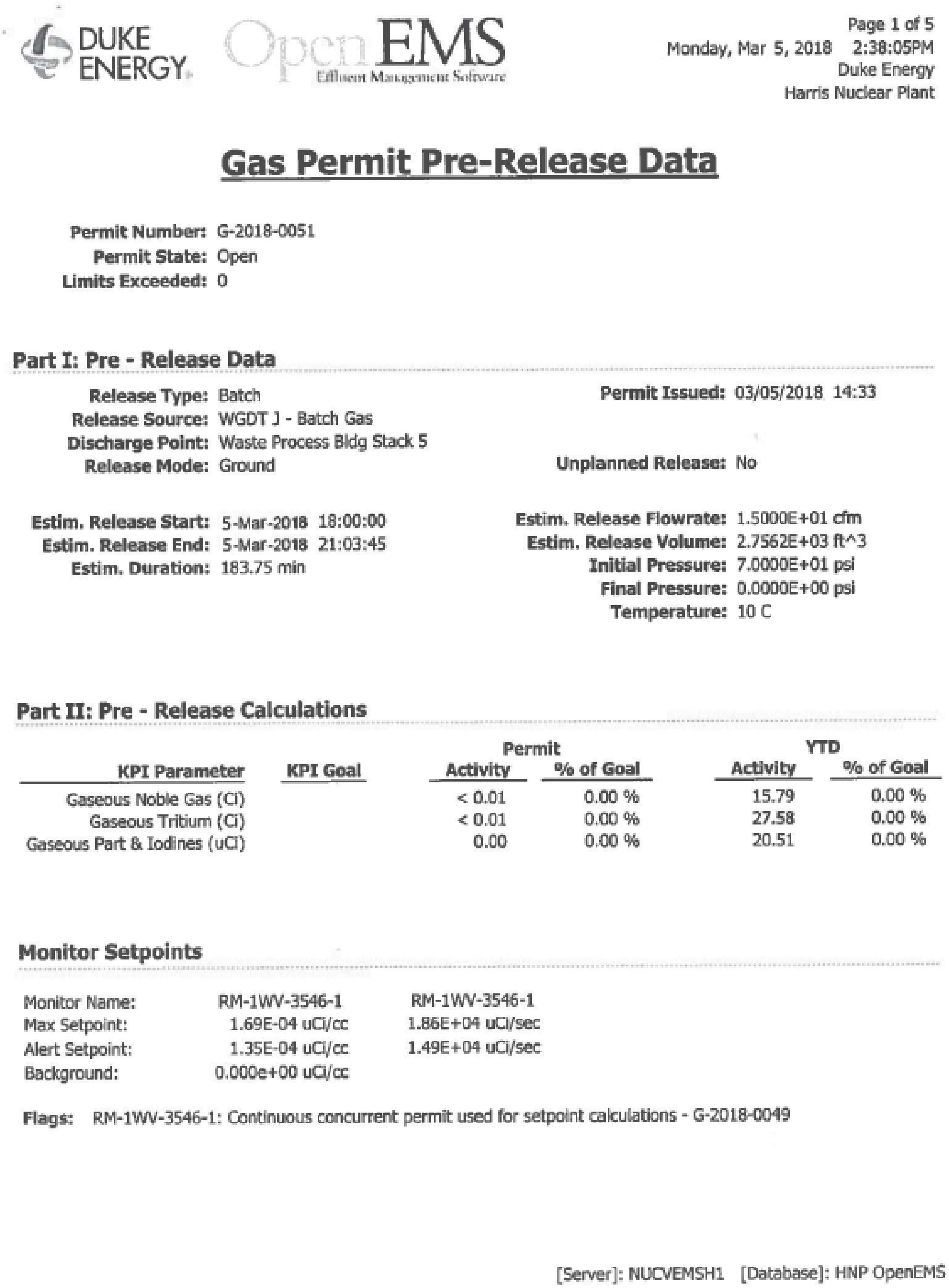

                         Total Gamma Activity                      1.781E-06


     Errors quoted at 1.000 sigma
```

Analysis Report for 170808 3/4/2018 10:39:46 Page 3 of 3

UNIDENTIFIED PEAK REPORT

No Unidentified Peaks Present



DUKE ENERGY

OpenEMS
Effluent Management Software

Page 1 of 5
Monday, Mar 5, 2018 2:38:05PM
Duke Energy
Harris Nuclear Plant

Gas Permit Pre-Release Data

Permit Number: G-2018-0051
Permit State: Open
Limits Exceeded: 0

Part I: Pre - Release Data

Release Type: Batch
Release Source: WGDT J - Batch Gas
Discharge Point: Waste Process Bldg Stack 5
Release Mode: Ground
Permit Issued: 03/05/2018 14:33
Unplanned Release: No

Estim. Release Start: 5-Mar-2018 18:00:00
Estim. Release End: 5-Mar-2018 21:03:45
Estim. Duration: 183.75 min

Estim. Release Flowrate: 1.5000E+01 cfm
Estim. Release Volume: 2.7562E+03 ft^3
Initial Pressure: 7.0000E+01 psi
Final Pressure: 0.0000E+00 psi
Temperature: 10 C

Part II: Pre - Release Calculations

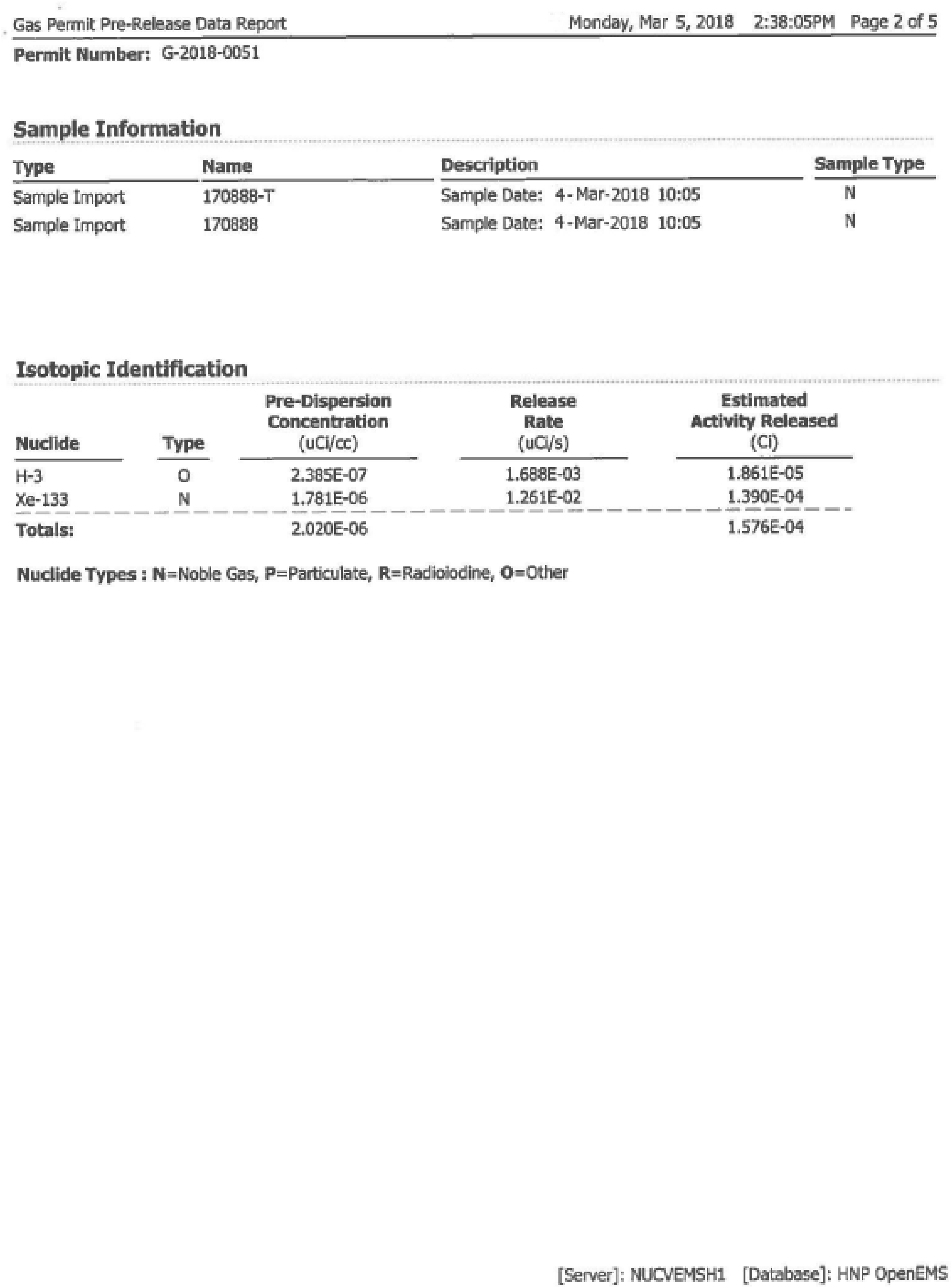
		Permit		YTD	
KPI Parameter	**KPI Goal**	**Activity**	**% of Goal**	**Activity**	**% of Goal**
Gaseous Noble Gas (Ci)		< 0.01	0.00 %	15.79	0.00 %
Gaseous Tritium (Ci)		< 0.01	0.00 %	27.58	0.00 %
Gaseous Part & Iodines (uCi)		0.00	0.00 %	20.51	0.00 %

Monitor Setpoints

Monitor Name:	RM-1WV-3546-1	RM-1WV-3546-1
Max Setpoint:	1.69E-04 uCi/cc	1.86E+04 uCi/sec
Alert Setpoint:	1.35E-04 uCi/cc	1.49E+04 uCi/sec
Background:	0.000e+00 uCi/cc	

Flags: RM-1WV-3546-1: Continuous concurrent permit used for setpoint calculations - G-2018-0049

[Server]: NUCVEMSH1 [Database]: HNP OpenEMS



Gas Permit Pre-Release Data Report Monday, Mar 5, 2018 2:38:05PM Page 2 of 5

Permit Number: G-2018-0051

Sample Information

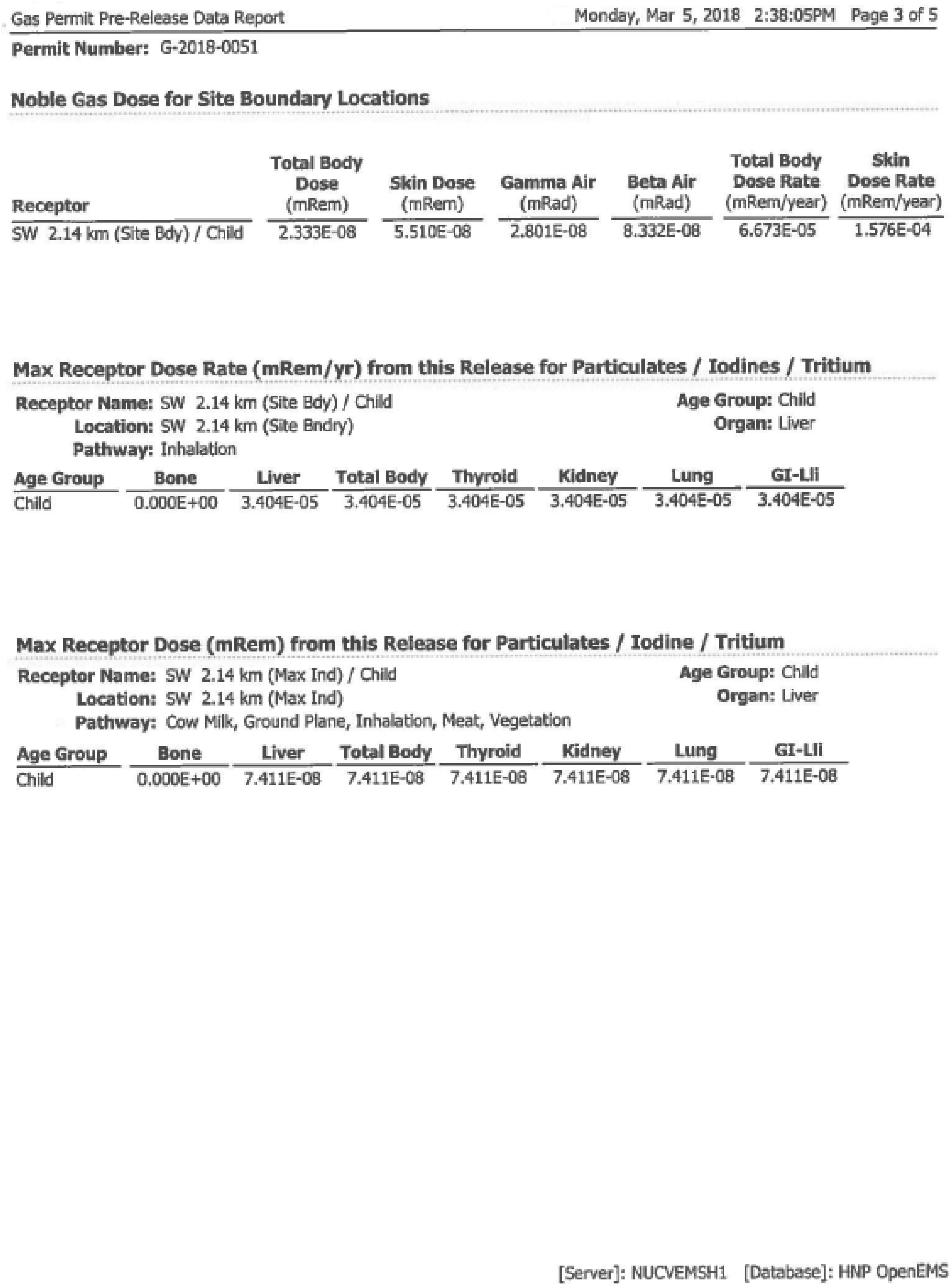
Type	Name	Description	Sample Type
Sample Import	170888-T	Sample Date: 4-Mar-2018 10:05	N
Sample Import	170888	Sample Date: 4-Mar-2018 10:05	N

Isotopic Identification

Nuclide	Type	Pre-Dispersion Concentration (uCi/cc)	Release Rate (uCi/s)	Estimated Activity Released (Ci)
H-3	O	2.385E-07	1.688E-03	1.861E-05
Xe-133	N	1.781E-06	1.261E-02	1.390E-04
Totals:		2.020E-06		1.576E-04

Nuclide Types : N=Noble Gas, **P**=Particulate, **R**=Radioiodine, **O**=Other

[Server]: NUCVEMSH1 [Database]: HNP OpenEMS



Gas Permit Pre-Release Data Report Monday, Mar 5, 2018 2:38:05PM Page 3 of 5

Permit Number: G-2018-0051

Noble Gas Dose for Site Boundary Locations

Receptor	Total Body Dose (mRem)	Skin Dose (mRem)	Gamma Air (mRad)	Beta Air (mRad)	Total Body Dose Rate (mRem/year)	Skin Dose Rate (mRem/year)
SW 2.14 km (Site Bdy) / Child	2.333E-08	5.510E-08	2.801E-08	8.332E-08	6.673E-05	1.576E-04

Max Receptor Dose Rate (mRem/yr) from this Release for Particulates / Iodines / Tritium

Receptor Name: SW 2.14 km (Site Bdy) / Child
Location: SW 2.14 km (Site Bndry)
Pathway: Inhalation
Age Group: Child
Organ: Liver

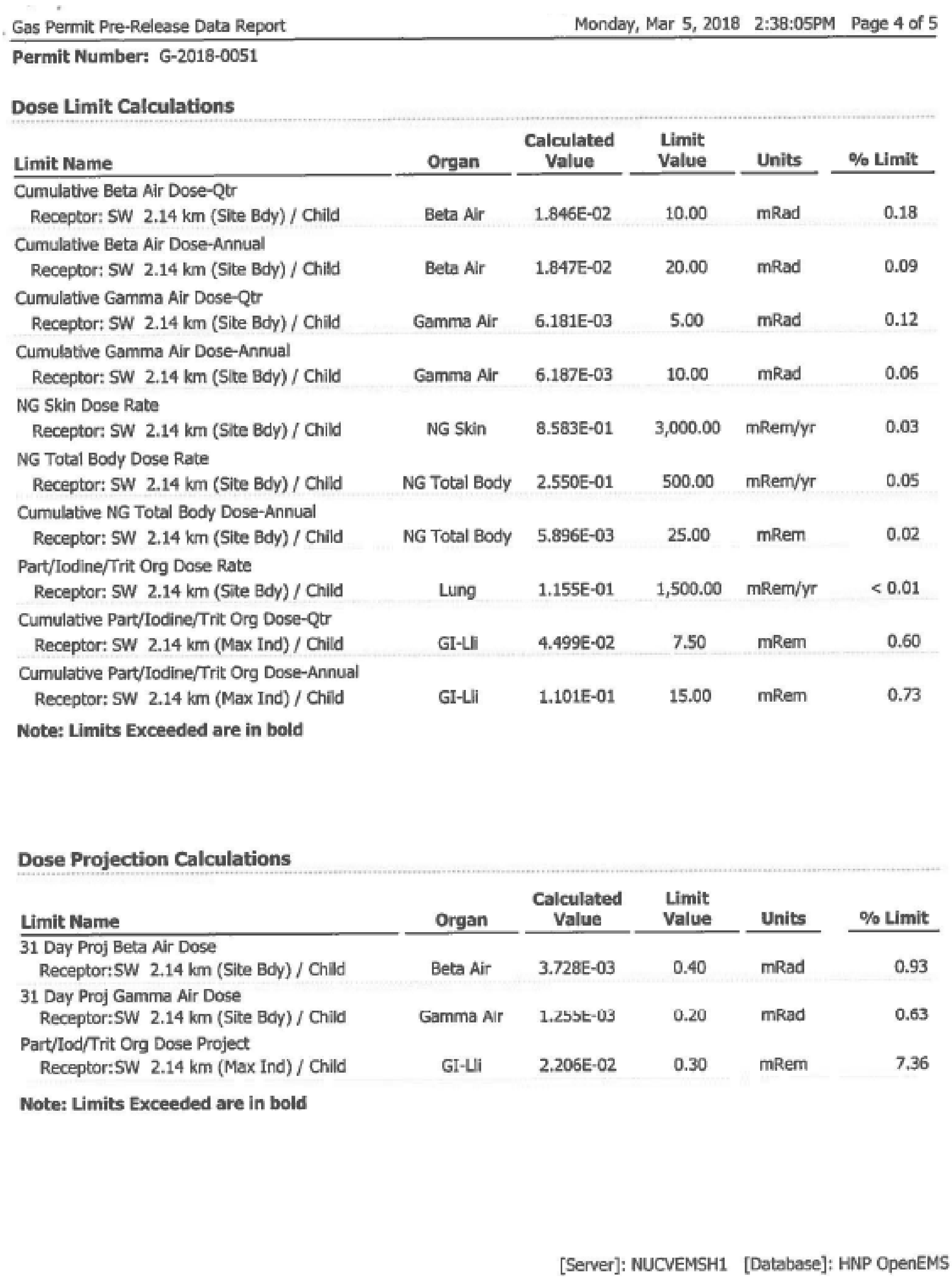
Age Group	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-Lli
Child	0.000E+00	3.404E-05	3.404E-05	3.404E-05	3.404E-05	3.404E-05	3.404E-05

Max Receptor Dose (mRem) from this Release for Particulates / Iodine / Tritium

Receptor Name: SW 2.14 km (Max Ind) / Child
Location: SW 2.14 km (Max Ind)
Pathway: Cow Milk, Ground Plane, Inhalation, Meat, Vegetation
Age Group: Child
Organ: Liver

Age Group	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-Lli
Child	0.000E+00	7.411E-08	7.411E-08	7.411E-08	7.411E-08	7.411E-08	7.411E-08

[Server]: NUCVEMSH1 [Database]: HNP OpenEMS



Gas Permit Pre-Release Data Report Monday, Mar 5, 2018 2:38:05PM Page 4 of 5

Permit Number: G-2018-0051

Dose Limit Calculations

Limit Name	Organ	Calculated Value	Limit Value	Units	% Limit
Cumulative Beta Air Dose-Qtr Receptor: SW 2.14 km (Site Bdy) / Child	Beta Air	1.846E-02	10.00	mRad	0.18
Cumulative Beta Air Dose-Annual Receptor: SW 2.14 km (Site Bdy) / Child	Beta Air	1.847E-02	20.00	mRad	0.09
Cumulative Gamma Air Dose-Qtr Receptor: SW 2.14 km (Site Bdy) / Child	Gamma Air	6.181E-03	5.00	mRad	0.12
Cumulative Gamma Air Dose-Annual Receptor: SW 2.14 km (Site Bdy) / Child	Gamma Air	6.187E-03	10.00	mRad	0.06
NG Skin Dose Rate Receptor: SW 2.14 km (Site Bdy) / Child	NG Skin	8.583E-01	3,000.00	mRem/yr	0.03
NG Total Body Dose Rate Receptor: SW 2.14 km (Site Bdy) / Child	NG Total Body	2.550E-01	500.00	mRem/yr	0.05
Cumulative NG Total Body Dose-Annual Receptor: SW 2.14 km (Site Bdy) / Child	NG Total Body	5.896E-03	25.00	mRem	0.02
Part/Iodine/Trit Org Dose Rate Receptor: SW 2.14 km (Site Bdy) / Child	Lung	1.155E-01	1,500.00	mRem/yr	< 0.01
Cumulative Part/Iodine/Trit Org Dose-Qtr Receptor: SW 2.14 km (Max Ind) / Child	GI-Lli	4.499E-02	7.50	mRem	0.60
Cumulative Part/Iodine/Trit Org Dose-Annual Receptor: SW 2.14 km (Max Ind) / Child	GI-Lli	1.101E-01	15.00	mRem	0.73

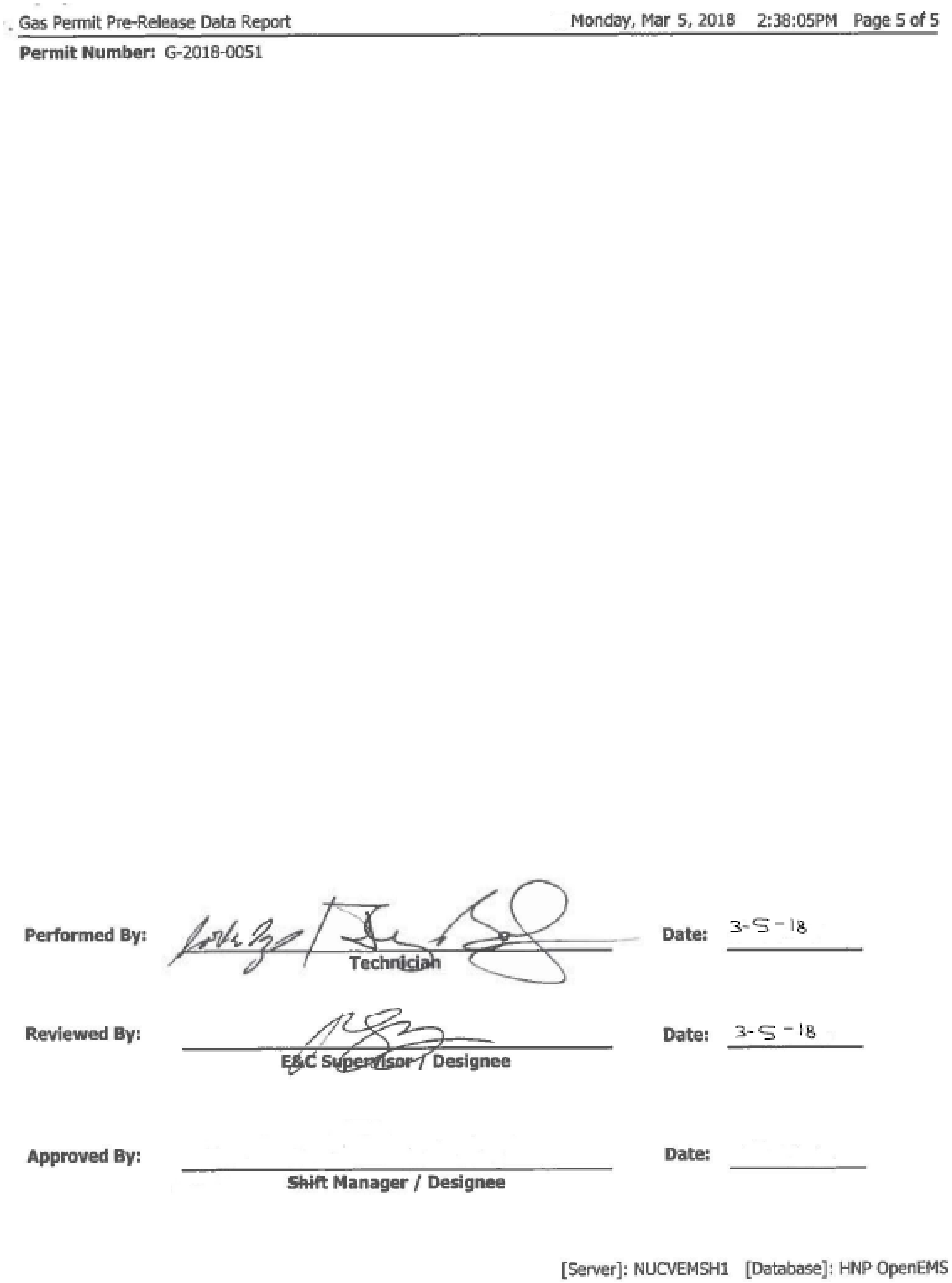
Note: Limits Exceeded are in bold

Dose Projection Calculations

Limit Name	Organ	Calculated Value	Limit Value	Units	% Limit
31 Day Proj Beta Air Dose Receptor:SW 2.14 km (Site Bdy) / Child	Beta Air	3.728E-03	0.40	mRad	0.93
31 Day Proj Gamma Air Dose Receptor:SW 2.14 km (Site Bdy) / Child	Gamma Air	1.255E-03	0.20	mRad	0.63
Part/Iod/Trit Org Dose Project Receptor:SW 2.14 km (Max Ind) / Child	GI-Lli	2.206E-02	0.30	mRem	7.36

Note: Limits Exceeded are in bold

[Server]: NUCVEMSH1 [Database]: HNP OpenEMS



Gas Permit Pre-Release Data Report Monday, Mar 5, 2018 2:38:05PM Page 5 of 5

Permit Number: G-2018-0051

Performed By: ______________________ **Date:** 3-5-18

Technician

Reviewed By: ______________________ **Date:** 3-5-18

E&C Supervisor / Designee

Approved By: ______________________ **Date:** __________

Shift Manager / Designee

[Server]: NUCVEMSH1 [Database]: HNP OpenEMS

JPM CUE SHEET

```
TREND 10 MIN X 24          3546 WPB EXH STACK 5                03/05/18   14:27:42

 NAME    TYPE      CHANNEL ID          DESCRIPTION               VALUE     UNITS
4EE797   GEFF     REM-*1WV-3546     3546 WPB EXH STACK 5       3.52E+02   UC/S

 SMPL FLOW1 =   3.00E+00    PROC FLOW N =   2.86E+05    HIGH ALARM AT 1.86E+04
 SMPL FLOW2 =   0.00E+00    PROC FLOW A =   0.00E+00   ALERT ALARM AT 1.49E+04
 BACKGROUND VALUE   =   0.00E+00 CPM
 BACKGROUND TIME    =   02/26/18 13:29:10
                             V          V          V          V          ALARM
 3.52E+02                                                                  V  V
  0.00E+00
  0.00E+00
  0.00E+00
  0.00E+00
  0.00E+00
  0.00E+00
  0.00E+00
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  0.00E+00
  0.00E+00
  0.00E+00
  0.00E+00

 FLOWING
          FILTER ADVANCING       PURGING        CHECKSOURCE     GRIDS
 POWERFAIL (HIST.)          C/S (CURR. VALUE)       ALERT ALARM
 QUESTIONABLE               NORMAL                  HIGH ALARM
                                                                 KEYPAD[  OFF-LINE  ]
                                                                 PRINTER  #2
                                                                  RM-11
```

JPM CUE SHEET

```
TREND 10 MIN X 24          3546-1 WPB WRGM EFFLUENT          03/05/18   14:25:50

 NAME    TYPE      CHANNEL ID            DESCRIPTION               VALUE     UNITS
4EX837   WEFF   RM-1WV-3546-1     3546-1 WPB WRGM EFFLUENT      3.38E-01  UC/S

SMPL FLOW1 =  1.41E+00     PROC FLOW N =  2.86E+05      HIGH ALARM AT 1.86E+04
SMPL FLOW2 =  0.00E+00     PROC FLOW A =  0.00E+00     ALERT ALARM AT 1.49E+04
BACKGROUND VALUE    =  0.00E+00 CPM
BACKGROUND TIME     =  02/26/18 14:18:26
                       V          V          V          V         ALARM
3.38E-01                                                          V  V

1.72E-01
1.87E-01
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
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0.00E+00
0.00E+00

                FILTER ADVANCING        PURGING        CHECKSOURCE
   POWERFAIL (HIST.)      O/S (CURR. VALUE)      ALERT ALARM       GRIDS
   QUESTIONABLE           NORMAL                 HIGH ALARM
                                                                   KEYPAD
                                                                   PRINTER OFF LINE
                                                                     RM-11 #3
```

```
TREND 10 MIN X 24          3546 WPB EXH STACK 5           03/05/18   14:27:07
 NAME    TYPE      CHANNEL ID          DESCRIPTION              VALUE    UNITS
 4GG793  GAS     REM-*1WV-3546     3546 WPB EXH STACK 5      2.58E-06  UC/ML
 SMPL FLOW1 =  3.00E+00    PROC FLOW N =  2.86E+05    HIGH ALARM AT 1.69E-04
 SMPL FLOW2 =  0.00E+00    PROC FLOW A =  0.00E+00   ALERT ALARM AT 1.35E-04
 BACKGROUND VALUE  =  1.47E+01 CPM
 BACKGROUND TIME   =  02/26/18 13:29:11
                      V          V          V          V        ALARM
 2.58E-06                                                       V V
 2.24E-06
 2.25E-06
 2.15E-06
 2.17E-06
 2.19E-06
 2.20E-06
 2.22E-06
 2.24E-06
 2.25E-06
 2.23E-06
 2.25E-06
 2.27E-06
 2.28E-06
 2.30E-06
 2.28E-06
 2.30E-06
 2.31E-06
 2.14E-06
 2.15E-06
 2.17E-06
 2.19E-06
 2.20E-06
 2.22E-06
 2.24E-06
  FLOWING
            FILTER ADVANCING       PURGING       CHECKSOURCE     GRIDS
   POWERFAIL (HIST.)          O/S (CURR. VALUE)      ALERT ALARM
   QUESTIONABLE               NORMAL                 HIGH ALARM
                                                               KEYPAD[     ]
                                                               PRINTER OFF-LINE
                                                                 RM-11 #2
```

```
TREND 10 MIN X 24          3546-1 WPB WRGM LOW RNG          03/05/18  14:24:27

 NAME    TYPE      CHANNEL ID          DESCRIPTION              VALUE     UNITS
4NX833   WGAS    RM-1WV-3546-1     3546-1 WPB WRGM LOW RNG     2.49E-09   UC/ML

SMPL FLOW1 =   1.41E+00     PROC FLOW N =  2.86E+05     HIGH ALARM AT 1.62E-04
SMPL FLOW2 =   0.00E+00     PROC FLOW A =  0.00E+00    ALERT ALARM AT 1.35E-04
BACKGROUND VALUE   =  1.26E+01 CPM
BACKGROUND TIME    =  02/26/18 08:43:44
                      V          V          V          V          ALARM
 2.49E-09                                                          V  V
 1.30E-09
 2.01E-08
 4.01E-07
 4.01E-07
 4.01E-07
 4.01E-07
 4.01E-07
 4.00E-07
 4.00E-07
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 4.01E-07

FLOWING
               FILTER ADVANCING          PURGING          CHECKSOURCE
 POWERFAIL (HIST.)                                                          GRIDS
 QUESTIONABLE          C/S (CURR. VALUE)          ALERT ALARM
                       NORMAL                     HIGH ALARM
                                                                       KEYPAD[         ]
                                                                       PRINTER OFF-LINE
                                                                         RM-11 #2
```

`Facility:	Harris Nuclear Plant	Task No.:	345010H602
Task Title:	Determine Initial Protective Action Recommendations	JPM No.:	2018 NRC Exam Admin JPM SRO A4
K/A Reference:	G2.4.38 RO 2.4 SRO 4.4 G2.4.44 RO 2.4 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.** A General Emergency has just been declared.

Initiating Cue:	Using the information provided, to determine Protective Action Recommendations.

Task Standard: Protective Action Recommendations determined within 15 minutes.

Required Materials: None

General References: PEP-110 EAL Matrix
PEP-110 Rev. 25

Handouts:

- Attached Initial Conditions
- PEP-110 Rev. 25
- PEP-110 EAL Matrix

Time Critical Task: YES – 15 minutes for classification.

Validation Time: 15 minutes for classification

CRITICAL STEP JUSTIFICATION	
Step 2	Protective Action Recommendations prevent or minimize exposure to the general public. Protective Action Recommendations are made to the State and County agencies that are responsible for implementing protective actions for the general public whenever PAGs are exceeded.
Step 4	Protective Action Recommendations must be determined within 15 minutes of the classification of a General Emergency.

Evaluator Cue:	**Start Time for this portion of JPM begins when the individual has been briefed.**

START TIME: ____________

Performance Step: 1 OBTAINS PEP-110.

Standard : Obtains PEP-110

Comments:

✓ **Performance Step: 2** Determine Protective Action Recommendations

Standard :

Uses PEP-110 and determines Table 1 applies:

- **Evacuate 2 mile radius**
- **Evacuate 10 miles downwind: Subzones A, B, E, L, M, N**
- **Shelter remaining subzones: C, D, F, G, H, I, J, K**

Recommend the consideration of KI use by the public - YES

Comments:

Performance Step: 3 Verify Protective Action Recommendations

Standard : Reviews PEP-110 to verify Protective Action Recommendations
Completes PAR and turns in results to the Evaluator

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 PAR record 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 ____________

VERIFICATION OF COMPLETION

Job Performance Measure No.: 2018 NRC Exam Admin JPM SRO A4
Classify an Event
PEP-110 and EP-EAL

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

Name: ______________________________

Date: ______________

Initial Conditions:	**This is a TIME CRITICAL JPM.** A General Emergency has just been declared.

Initiating Cue:	Using the information provided, determine the Protective Action Recommendations.

A General Emergency has just been declared with the following conditions:

- A LOCA inside Containment has occurred
- Safety Injection has actuated but flow has NOT been established
- Core Cooling CSFST is RED
- The crew is implementing EOP-FR-C.1, Response to Inadequate Core Cooling
- An unisolable pathway from Containment to the environment exists
- The Dose Assessment Team projects 1.2E3 mrem TEDE and 6E3 mrem CDE at the Site Boundary
- Wind direction is from 150˚

Protective Action Recommendations:

Evauate:____

____

____

____

Shelter:____

____

____

____

Recommend the consideration of KI use by the public: YES / NO (circle one)