Facility: SURRY Date of Exam: SEPTEMBER 2Ø16

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	1	2				2	2			1	9	2	2	4
	Tier Totals	4	4	5				5	5			4	27	5	5	10
2. Plant Systems	1	3	2	2	2	3	3	3	3	2	3	2	28	3	2	5
	2	1	1	1	1	1	1	1	Ø	1	1	1	10	2	3	3
	Tier Totals	4	3	3	3	4	4	4	3	3	4	3	38	5	3	8

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

Note:

1. BM Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 Radiation Control K/A is allowed if the K/A is replaced by a K/A from another Tier 3 Category). *)
2. BM 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. BM Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted with justification; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. BM 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. BM 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. BM Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. BM The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.
8. BM On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in a category other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. BM For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

G* Generic K/As

RO

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
007EG2.4.45	Reactor Trip - Stabilization - Recovery / 1	4.1	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to prioritize and interpret the significance of each annunciator or alarm.
008AK3.01	Pressurizer Vapor Space Accident / 3	3.7	4.4	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Why PZR level may come back on scale if RCS is saturated.
011EK2.02	Large Break LOCA / 3	2.6	2.7	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Pumps
015AK2.10	RCP Malfunctions / 4	2.8	2.8	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	RCP indicators and controls
022AG2.2.37	Loss of Rx Coolant Makeup / 2	3.6	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to determine operability and/or availability of safety related equipment
026AA1.03	Loss of Component Cooling Water / 8	3.6	3.6	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	SWS as a backup to the CCWS
029EK1.01	ATWS / 1	2.8	3.1	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Reactor nucleonics and thermo-hydraulics behavior
040AA1.18	Steam Line Rupture - Excessive Heat Transfer / 4	4.2	4.2	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Control rod position indicators
054AA2.03	Loss of Main Feedwater / 4	4.1	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Conditions and reasons for AFW pump startup
055EA2.03	Station Blackout / 6	3.9	4.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Actions necessary to restore power
056AK1.01	Loss of Off-site Power / 6	3.7	4.2	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Principle of cooling by natural convection

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
058AK3.01	Loss of DC Power / 6	3.4	3.7	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Use of dc control power by D/Gs
062AA2.03	Loss of Nuclear Svc Water / 4	2.6	2.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	The valve lineups necessary to restart the SWS while bypassing the portion of the system causing the abnormal condition
065AK3.08	Loss of Instrument Air / 8	3.7	3.9	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Actions contained in EOP for loss of instrument air
077AK1.03	Generator Voltage and Electric Grid Disturbances / 6	3.3	3.4	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Under-excitation
WE04EA1.3	LOCA Outside Containment / 3	3.8	4.0	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Desired operating results during abnormal and emergency situations.
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.
we11EG2.4.6	Loss of Emergency Coolant Recirc. / 4	3.7	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge symptom based EOP mitigation strategies.

RO

T1G2 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
001AK3.02	Continuous Rod Withdrawal / 1	3.2	4.3	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Tech-Spec limits on rod operability
036AA2.01	Fuel Handling Accident / 8	3.2	3.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	ARM system indications
037AG2.2.12	Steam Generator Tube Leak / 3	3.7	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of surveillance procedures.
051AA1.04	Loss of Condenser Vacuum / 4	2.5	2.5	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Rod position
059AA1.02	Accidental Liquid RadWaste Rel. / 9	3.3	3.4	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	ARM system
069AK3.01	Loss of CTMT Integrity / 5	3.8	4.2	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Guidance contained in EOP for loss of containment integrity
074EK2.09	Inad. Core Cooling / 4	2.6	2.6	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Controllers and positioners
WE03EK1.1	LOCA Cooldown - Depress. / 4	3.4	4.0	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Components, capacity, and function of emergency systems.
WE15EA2.2	Containment Flooding / 5	2.9	3.3	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
003A2.03	Reactor Coolant Pump	2.7	3.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Problems associated with RCP motors, including faulty motors and current, winding and bearing temperature problems
004A2.27	Chemical and Volume Control	3.5	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Improper RWST boron concentration
004K5.46	Chemical and Volume Control	2.5	2.9	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Reason for going solid in PZR (collapsing steam bubble): make sure no steam is in PRT when PORV is opened to drain RCS
005A1.03	Residual Heat Removal	2.5	2.6	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Closed cooling water flow rate and temperature
006A2.13	Emergency Core Cooling	3.9	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Inadvertent SIS actuation
006G2.1.28	Emergency Core Cooling	4.1	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the purpose and function of major system components and controls.
007A1.02	Pressurizer Relief/Quench Tank	2.7	2.9	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Maintaining quench tank pressure
008K2.02	Component Cooling Water	3.0	3.2	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	CCW pump, including emergency backup
010K1.08	Pressurizer Pressure Control	3.2	3.5	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	PZR LCS
010K6.01	Pressurizer Pressure Control	2.7	3.1	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Pressure detection systems
012K4.07	Reactor Protection	3.0	3.2	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	First-out indication

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
013A4.01	Engineered Safety Features Actuation	4.5	4.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	ESFAS-initiated equipment which fails to actuate
013K6.01	Engineered Safety Features Actuation	2.7	3.1	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Sensors and detectors
022A4.01	Containment Cooling	3.6	3.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	CCS fans
026A1.03	Containment Spray	3.5	3.5	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Containment sump level
026A4.01	Containment Spray	4.5	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	CSS controls
039A3.02	Main and Reheat Steam	3.1	3.5	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Isolation of the MRSS
039K5.08	Main and Reheat Steam	3.6	3.6	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Effect of steam removal on reactivity
059K4.18	Main Feedwater	2.8	3.0	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Automatic feedwater reduction on plant trip
061K2.03	Auxiliary/Emergency Feedwater	4.0	3.8	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	AFW diesel driven pump
062A3.01	AC Electrical Distribution	3.0	3.1	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Vital ac bus amperage
063G2.4.20	DC Electrical Distribution	3.8	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of operational implications of EOP warnings, cautions and notes.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
063K3.02	DC Electrical Distribution	3.5	3.7	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Components using DC control power
064K6.08	Emergency Diesel Generator	3.2	3.3	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Fuel oil storage tanks
073K5.02	Process Radiation Monitoring	2.5	3.1	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Radiation intensity changes with source distance
076K1.15	Service Water	2.5	2.6	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	FPS
078K3.03	Instrument Air	3.0	3.4	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Cross-tied units
103K1.02	Containment	3.9	4.1	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Containment isolation/containment integrity

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
015A1.02	Nuclear Instrumentation	3.5	3.6	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	SUR
016A3.02	Non-nuclear Instrumentation	2.9	2.9	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Relationship between meter readings and actual parameter value
027K2.01	Containment Iodine Removal	3.1	3.4	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Fans
028K5.04	Hydrogen Recombiner and Purge Control	2.6	3.2	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	The selective removal of hydrogen
034K6.02	Fuel Handling Equipment	2.6	3.3	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Radiation monitoring systems
071K3.04	Waste Gas Disposal	2.7	2.9	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Ventilation system
072A4.01	Area Radiation Monitoring	3.0	3.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	Alarm and interlock setpoint checks and adjustments
075G2.1.30	Circulating Water	4.4	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to locate and operate components, including local controls.
079K4.01	Station Air	2.9	3.2	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Cross-connect with IAS
086K1.02	Fire Protection	2.7	3.2	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Raw service water

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
G2.1.15	Conduct of operations	2.7	3.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of administrative requirements for temporary management directives such as standing orders, night orders, Operations memos, etc.
G2.1.3	Conduct of operations	3.7	3.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of shift or short term relief turnover practices.
G2.1.43	Conduct of operations	4.1	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to use procedures to determine the effects on reactivity of plant changes
G2.2.20	Equipment Control	2.6	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process for managing troubleshooting activities.
G2.2.22	Equipment Control	4.0	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of limiting conditions for operations and safety limits.
G2.2.7	Equipment Control	2.9	3.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process for conducting special or infrequent tests
G2.3.13	Radiation Control	3.4	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of radiological safety procedures pertaining to licensed operator duties
G2.4.11	Emergency Procedures/Plans	4.0	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of abnormal condition procedures.
G2.4.13	Emergency Procedures/Plans	4.0	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of crew roles and responsibilities during EOP usage.
G2.4.27	Emergency Procedures/Plans	3.4	3.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of "fire in the plant" procedures.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
029EA2.09	ATWS / 1	4.4	4.5	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Occurrence of a main turbine/reactor trip
038EG2.4.8	Steam Gen. Tube Rupture / 3	3.8	4.5	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
054AG2.1.23	Loss of Main Feedwater / 4	4.3	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to perform specific system and integrated plant procedures during all modes of plant operation.
055EA2.03	Station Blackout / 6	3.9	4.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Actions necessary to restore power
058AG2.4.20	Loss of DC Power / 6	3.8	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of operational implications of EOP warnings, cautions and notes.
WE12EA2.2	Steam Line Rupture - Excessive Heat Transfer / 4	3.4	3.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
061AG2.4.30	ARM System Alarms / 7	2.7	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of events related to system operations/status that must be reported to internal orginizations or outside agencies.
068AA2.08	Control Room Evac. / 8	3.9	4.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	S/G pressure
we08EG2.1.19	RCS Overcooling - PTS / 4	3.9	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to use plant computer to evaluate system or component status.
WE13EA2.1	Steam Generator Over-pressure / 4	2.9	3.4	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
012A2.01	Reactor Protection	3.1	3.6	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Faulty bistable operation
039A2.03	Main and Reheat Steam	3.4	3.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Indications and alarms for main steam and area radiation monitors (during SGTR)
062A2.04	AC Electrical Distribution	3.4	3.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Effect on plant of de-energizing a bus
076G2.1.32	Service Water	3.8	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to explain and apply all system limits and precautions.
103G2.2.44	Containment	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

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
014G2.2.22	Rod Position Indication	4.0	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of limiting conditions for operations and safety limits.
068A2.02	Liquid Radwaste	2.7	2.8	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Lack of tank recirculation prior to release

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
G2.1.41	Conduct of operations	2.8	3.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the refueling processes
G2.1.7	Conduct of operations	4.4	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation.
G2.2.13	Equipment Control	4.1	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of tagging and clearance procedures.
G2.2.40	Equipment Control	3.4	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to apply technical specifications for a system.
G2.3.12	Radiation Control	3.2	3.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of radiological safety principles pertaining to licensed operator duties
G2.4.46	Emergency Procedures/Plans	4.2	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to verify that the alarms are consistent with the plant conditions.
G2.4.9	Emergency Procedures/Plans	3.8	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.

Facility: _Surry_________________________	Date of Examination: _9/19/2016__
Examination Level: RO ☒ SRO ☐	Operating Test Number: SR 2016 301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, R	Calculate VCT Leakrate
Conduct of Operations	M, R	Calculate H2 Vent Time FR-I.3
Equipment Control	N, R	SA-AA-125 Requirements for MCR Chiller
Radiation Control		
Emergency Plan	N, S	Complete EPIP-2.02, SBLOCA, Alert

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

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

Facility: ______________________ Date of Examination: 9/19/2016

Examination Level: RO ☐ SRO ☑ Operating Test Number: SR 2016 301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, R	Calculate VCT Leakrate Evaluate Tech Specs
Conduct of Operations	M, R	Calculate H2 Vent Time FR-I.3
Equipment Control	N, R	Operable MCR Chiller combination IAW 0-OP-VS-006
Radiation Control	N, R	Dose Rate Calculation
Emergency Plan	N, S	Determine PAR

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

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

Facility: Surry	Date of Examination: 9/19/16
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐	Operating Test No.: SR 2016 301

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

System / JPM Title	Type Code*	Safety Function
a. Adjust the PRNIs IAW 1-OPT-RX-001 (001 AA2.05 4.4/4.6)	A,M,S	1
b. Isolate a Leaking RSHX (059 AK3.01 3.5/3.9)	D,L,S	9
c. Perform E-0, Attachment 4 (WE14 EA1.3 3.3/3.8)	N,L,EN,S	5
d. Transfer the SI System to Cold Leg Recirc (006 A3.08 4.2/4.3)	D,A,L,S	2
e. Respond to a Loss of the Operating RHR Pump (005 A2.03 2.9/3.1)	D,L,S	4P
f. Bypass Containment Detection ON 0-FP-MON-IMS-1 (086 A4.02 3.5 3.5)	D,L,S	8
g. Place H2 Analyzers In Service (011EK3.12 4.4/4.6)	A,L,M,S	3
h. Synchronize and Transfer Electrical Power Systems (062 A4.01 3.3/3.1)	D,S	6
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. Locally Isolate Flooding #3 MER (076 A2.01 3.5/3.7)	D,A,L,E	4S
j. Locally Establish RCS & SG Hi/Lo Interface Integrity (068 AA1.12 4.4/4.4)	D,L,E	8
k. Locally Emergency Borate (024 AA1.04 3.6/3.7)	A,D,R,E	1

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

* Type Codes	Criteria for RO
A)lternate path	5 (4 - 6)
(C)ontrol room	
(D)irect from bank	8 (≤ 9)
(E)mergency or abnormal in-plant	3 (≥ 1)
(EN)gineered safety feature	1 (≥ 1) (control room system)
(L)ow-Power / Shutdown	8 (≥ 1)
(N)ew or (M)odified from bank including 1(A)	3 (≥ 2)
(P)revious 2 exams	0 (≤ 3) (randomly selected)
(R)CA	1 (≥ 1)
(S)imulator	

Facility: Surry Date of Examination: 9/19/16

Exam Level: RO ☐ SRO-I ☒ SRO-U ☐ Operating Test No.: SR 2016 301

Control Room Systems:* 8 for RO; 7 for SRO-I; 2 or 3 for SRO-U		
System / JPM Title	Type Code*	Safety Function
a. Adjust the PRNIs IAW 1-OPT-RX-001 (001 AA2.05 4.4/4.6)	A,M,S	1
b. Isolate a Leaking RSHX (059 AK3.01 3.5/3.9)	D,L,S	9
c. Perform E-0, Attachment 4 (WE14 EA1.3 3.3/3.8)	N,L,EN	5
d. Transfer the SI System to Cold Leg Recirc (006 A3.08 4.2/4.3)	D,A,L,S	2
e. Respond to a Loss of the Operating RHR Pump (005 A2.03 2.9/3.1)	D,L,S	4P
f. Bypass Containment Detection ON 0-FP-MON-IMS-1 (086 A4.02 3.5 3.5)	D,L,S	8
g. Place H2 Analyzers In Service (011EK3.12 4.4/4.6)	A,L,M,S	3
h.		
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. Locally Isolate Flooding #3 MER (076 A2.01 3.5/3.7)	D,A,L,E	4S
j. Locally Establish RCS & SG Hi/Lo Interface Integrity (068 AA1.12 4.4/4.4)	D, L,E	8
k. Locally Emergency Borate (024 AA1.04 3.6/3.7)	A,D,R,E	1

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

* Type Codes	Criteria for SRO-I
A)lternate path	5 (4 – 6)
(C)ontrol room	
(D)irect from bank	7 (≤ 8)
(E)mergency or abnormal in-plant	3 (≥ 1)
(EN)gineered safety feature	1 (≥1) (control room system)
(L)ow-Power / Shutdown	8 (≥ 1)
(N)ew or (M)odified from bank including 1(A)	3 (≥ 2)
(P)revious 2 exams	0 (≤ 3) (randomly selected)
(R)CA	1 (≥ 1)
(S)imulator	

Facility: Surry	Date of Examination: 9/19/16
Exam Level: RO ☐ SRO-I ☐ SRO-U ■	Operating Test No.: SR 2016 301

Control Room Systems:* 8 for RO; 7 for SRO-I; 2 or 3 for SRO-U		
System / JPM Title	Type Code*	Safety Function
a. Adjust the PRNIs IAW 1-OPT-RX-001 (001 AA2.05 4.4/4.6)	A,M,S	1
b. Isolate a Leaking RSHX (059 AK3.01 3.5/3.9)	D,L,S	9
c. Perform E-0, Attachment 4 (WE14 EA1.3 3.3/3.8)	N,L,EN,S	5
d.		
e.		
f.		
g.		
h.		
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. Locally Isolate Flooding #3 MER (076 A2.01 3.5/3.7)	D,A,L,E	4S
k. Locally Emergency Borate (024 AA1.04 3.6/3.7)	A,D,R,E	1

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

* Type Codes	Criteria for SRO-U
A)lternate path	3 (2-3)
(C)ontrol room	
(D)irect from bank	3 (≤ 4)
(E)mergency or abnormal in-plant	2 (≥ 1)
(EN)gineered safety feature	1 (≥ 1) (control room system)
(L)ow-Power / Shutdown	3 (≥ 1)
(N)ew or (M)odified from bank including 1(A)	2 (≥ 1)
(P)revious 2 exams	0 (≤ 2) (randomly selected)
(R)CA	1 (≥ 1)
(S)imulator	

K/A Number: 001AK3.02: Continuous Rod Withdrawal / 1: Knowledge of the reasons for the following responses as they apply to the Continuous Rod Withdrawal: Tech-Spec limits on rod operability.

Level: RO Tier #: 1 Group #: 2 IR – RO: 3.2 IR-SRO: 4.3

Proposed Question: 1

Given the following**:**

- Unit 1 is at 100%.
- Rod H2 has dropped, and it took 5 seconds to drop into the core.

Which of the following describes:

1) The correct procedure to enter to mitigate the rod drop is ______.
2) Per Technical specifications, Control rods must have a drop time of ≤ ____ seconds.

A. 1) 0-AP-1.00, Rod Control System Malfunction
   2) 3.0

B. 1) 0-AP-1.00, Rod Control System Malfunction
   2) 2.4

C. 1) 0-AP-53.00, Loss of Vital Instrumentation/Controls
   2) 2.4

D. 1) 0-AP-53.00, Loss of Vital Instrumentation/Controls
   2) 3.0

Proposed Answer: B

Explanation: 1) A dropped control rod or rods is an entry condition for 0-AP-1.00, Rod Control System Malfunction, therefore 0-AP-1.00 is the correct procedure to enter. 2) Per Tech Specs 3.12.C rods are considered operable if: 1) trippable, 2) aligned within ± 12 steps or ± 24 steps as defined in TS 3.12.E.1.b, and 3) **have a drop time of ≤ 2.4 seconds** to dashpot entry.

Technical Reference: Surry Technical Specifications, Change 48. 0-AP-1.00, ROD CONTROL SYSTEM MALFUNCTION, Rev. 27.

Reference Provided to Applicant: NO

Learning Objective: ND-93.3-LP-3, ROD CONTROL, Objective J; Summarize the Technical Specifications associated with the Rod Control System. ND-93.4-LP-4, CERPI System, Objective D; Describe the Technical Specifications associated with the CERPI system, including for SRO candidates the basis behind these specifications.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.5,41.10 / 45.6 / 45.13)

Comments:

K/A Analysis: This question tests the Reactor Operator's knowledge of Tech Specs associated with Rod Operability, and the procedure that corrects the effects of a rod drop. A competent RO must understand what constitutes rod operability which in this case is evaluated by their knowledge of rod insertion, or rod drop times.

Distractor Analysis:

A. 1) 0-AP-1.00, Rod Control System Malfunction
2) 3.0

B. 1) 0-AP-1.00, Rod Control System Malfunction
2) 2.4

C. 1) 0-AP-53.00, Loss of Vital Instrumentation/Controls
2) 2.4

D. 1) 0-AP-53.00, Loss of Vital Instrumentation/Controls
2) 3.0

A. Incorrect. 1) Correct. 2) Incorrect because the tech spec required drop time is less than or equal to 2.4 seconds. Plausible because 3.0 seconds is within 25% of required 2.4 seconds.
B. Correct.
C. Incorrect. 1) Incorrect. Dropped rod is a direct entry condition for 0-AP-1.00. Plausible because 0-AP-53.00 does have actions for placing rod control in manual. 2) Correct.
D. Incorrect. 1) Incorrect. Dropped rod is a direct entry condition for 0-AP-1.00. Plausible because 0-AP-53.00 does have actions for placing rod control in manual. 2) Incorrect because the tech spec required drop time is less than or equal to 2.4 seconds. Plausible because 3.0 seconds is within 25% of required 2.4 seconds.

Tech Spec 0-AP-1.00, ROD CONTROL SYSTEM MALFUNCTION, ENTRY CONDITIONS

NUMBER	PROCEDURE TITLE	REVISION
0-AP-1.00	ROD CONTROL SYSTEM MALFUNCTION (WITH 7 ATTACHMENTS)	27
		PAGE 1 of 8

PURPOSE

To provide guidance to respond to Rod Control system malfunctions.

ENTRY CONDITIONS

1) Continuous rod insertion or withdrawal.
2) Dropped control rod or rods.
3) Failure of automatic control system.
4) Transition from Annunciator ( )G-B5, COMPU PRINTOUT ROD CONT SYS.
5) Transition from Annunciator ( )G-H2, RPI ROD BOTTOM $\leq$ 20 STEPS.
6) Transition from Annunciator ( )G-A6, ROD CONT SYS URGENT FAILURE.
7) Transition from Annunciator ( )G-H1, NIS DROPPED ROD FLUX DECREASE $\geq$ 5% PER 2 SEC.
8) Transition from Annunciator ( )G-G2, RPI ROD WRONG DIRECTION
9) Failure of Group Step Counter(s).

Reference, 0-AP-53.00, Loss of Vital Instrumentaion / Controls. Step 4. (P1 distractor)

NUMBER	PROCEDURE TITLE	REVISION
0-AP-53.00	LOSS OF VITAL INSTRUMENTATION / CONTROLS	21
		PAGE 3 of 13

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud.
- When the affected instrument/controller malfunction(s) has been addressed by this procedure, recovery actions should continue at Step 13.

*4.___ DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE:

- ☐ PRZR Pressure Control, Step 5
- ☐ NI Malfunction, ( )-AP-4.00
- ☐ SG Feed Flow, Step 7
- ☐ SG NR Level, Step 7
- ☐ SG Pressure, Step 7
- ☐ SG Steam Flow, Step 7
- ☐ Turbine First Stage Pressure, Step 8
- ☐ Median Tave, Step 9
- ☐ Loop Tave, Step 9
- ☐ Loop ΔT, Step 9
- ☐ Steam Dumps / SG PORVs, Step 10
- ☐ Turbine Controls, Step 10
- ☐ Turbine Valve Position, Step 10
- ☐ Turbine Monitoring Lights, Step 10
- ☐ PRZR Level Control, Step 11
- ☐ Reactor Coolant Flow, Step 12a
- ☐ PRZR Pressure Protection, Step 12b
- ☐ RCS Wide Range Pressure, Step 12f
- ☐ VCT Level, Step 12e
- ☐ CTMT Pressure, Step 12c
- ☐ RWST Level, Step 12d
- ☐ Underground FO Storage Tk, Step 12g
- ☐ CAT Level, Step 12h
- ☐ Emergency Condensate Makeup Tank Level, Step 12i
- ☐ FP/Domestic Water Level, Step 12j

Reference Tech Specs 3.12.C, Tech Spec requirements for rod operability.(P2 Correct Answer)

C. Control Rod Assemblies

1. To be considered OPERABLE during startup and POWER OPERATION each control rod assembly shall:

   1) be trippable,

   2) aligned within ± 12 steps or ± 24 steps of its group step demand position, as defined in Section 3.12.E.1.b, and

   3) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

K/A Number: 003A2.03: Reactor Coolant Pump / 4: Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems.

Level: RO Tier #: 2 Group #: 1 IR – RO: 2.7 IR-SRO: 3.1

Proposed Question: 2

Unit 1 is operating at 100% power.

- Annunciator 1B-D7, RCP 1A RSVR HI-LO level has been received.
- RCP Motor Upper Thrust Bearing Temperature indicates 197°F and rising.
- RCP Stator Winding Temperature indicates 235°F and rising.
- The Team has initiated 1-AP-9.00, RCP Abnormal Conditions.

In accordance with 1-AP-9.00, which ONE of the following identifies:

1) The _____ Temperature limit has been exceeded.
2) The _____ spray valve will need to be closed after the RCP is secured.

A. 1) RCP Motor Upper Thrust Bearing
   2) 1-RC-PCV-1455A

B. 1) RCP Motor Upper Thrust Bearing
   2) 1-RC-PCV-1455B

C. 1) RCP Stator Winding
   2) 1-RC-PCV-1455A

D. 1) RCP Stator Winding
   2) 1-RC-PCV-1455B

Proposed Answer: A.

Explanation: IAW 1-AP-9.00, Step 13 RNO directs monitoring of RCP parameters using Attachment 1. Attachment 1, will return the Team to Step 16 of AP-9.00 when temperature exceeds 195°F. With temperatures above limit and rising, the candidate will determine that the Unit must be tripped and the affected RCP secured. The associated spray valve is closed to prevent “short-cycling” spray to the PZR.

Technical Reference: 1-AP-9.00, RCP Abnormal Conditions, Rev 40.

Reference Provided to Applicant: NO

Learning Objective: ND-88.1-LP-6, Rx Coolant Pumps, Objective G, Summarize the content of Abnormal Procedures 9 and 16, and Annunciator Procedures for RCP CC Return Low Flow, RCP Vibrations, and RCS Low Flow, and the EP Continuous Actions Page which require stopping the RCPs.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5/ 45.3 / 45/13)

Comments:

K/A Match Analysis:

Question matches K/A, Candidate must consider implications of indications provided and determine the correct response IAW AP-9.00.

Distractor Analysis:

A. Correct. Both parts 1) and 2) are correct.

B. Incorrect. Part 1) is correct. Part 2) is incorrect. Plausible if Candidate confuses the applicable spray valve.

C. Incorrect. Part 1) is incorrect, 300°F is the limit for RCP operation. Plausible if Candidate applies 195°F from bearing temperature requirement. Part 2) is correct.

D. Incorrect. Part 1) is incorrect, 300°F is the limit for RCP operation. Plausible if Candidate applies 195°F from bearing temperature requirement. Part 2) is incorrect. Plausible if Candidate confuses applicable spray valve.

1-AP-9.00, RCP Abnormal Conditions

NUMBER	PROCEDURE TITLE	REVISION
1-AP-9.00	RCP ABNORMAL CONDITIONS	41
		PAGE 13 of 16

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Control Operations can monitor the RCP Speed Sensing Panel for additional information on RCP motor performance.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15.___	CHECK RCP STATOR TEMPERATURES - ANY GREATER THAN 300°F ☐ • PCS point T4014A, RCP A ☐ • PCS point T4015A, RCP B ☐ • PCS point T4016A, RCP C	☐ IF stator temperatures are less than 275°F, THEN GO TO Step 22. IF stator temperature(s) are greater than 275°F, THEN do the following: ☐ a) Continue to monitor stator temperatures. ☐ b) Monitor RCP parameters IAW Attachment 1. c) Investigate cause for high stator temperature: ☐ • CC leak to cooler ☐ • Loss of CC flow or high CC temperature ☐ • CTMT Ventilation problems ☐ d) Monitor pump vibration. ☐ e) IF vibration rises, THEN notify Shift Manager and System Engineering. ☐ f) GO TO Step 22.

NUMBER 1-AP-9.00	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 40	RCP PARAMETERS	PAGE 1 of 2

NOTE: If the Lower Bearing Seal Water temperature RTD is not operable, indirect monitoring of bearing temperature can be achieved by more frequent surveillance of Seal Water outlet temperature.

PARAMETERS	INSTRUMENT	ACTION LEVEL
RCP Stator Winding Temperature	PCS Point T4014A, RCP A PCS Point T4015A, RCP B PCS Point T4016A, RCP C	Greater than 300°F
RCP Motor Upper Thrust Brg Temperature	PCS Point T0414A, RCP A PCS Point T0434A, RCP B PCS Point T0454A, RCP C	Greater than 195°F
RCP Mtr Upper Radial Brg Temperature	PCS Point T0413A, RCP A PCS Point T0433A, RCP B PCS Point T0453A, RCP C	Greater than 195°F

K/A Number: 004A2.27: Chemical and Volume Control: Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:: Improper RWST boron concentration.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.5 IR-SRO: 4.2

Proposed Question: 3

Given the following:

- Unit 1 is at 100%, steady state.
- Annunciator 1A-A1, RWST TECH SPEC LO LVL came in earlier in the shift.
- The crew has just completed actions directed by ARP 1A-A1, and alarm is now clear.
- Chemistry department has completed sampling of the Unit 1 RWST, and reports that boron has lowered from 2340 ppm to 2240 ppm.

Which of the following choices completes the following statements:

1) The Technical Specification LCO for the boron injection system from the RWST requires a minimum boron of _____.
2) Procedure ______ should be used to restore this boron injection system to its previous boron level.

A. 1) 2250
   2) 1-OP-CS-004, Refueling Water Storage Makeup

B. 1) 2250
   2) 1-OP-CH-23, Boration Followed by a Manual Make-up using the Blender

C. 1) 2300
   2) 1-OP-CH-23, Boration Followed by a Manual Make-up using the Blender

D. 1) 2300
   2) 1-OP-CS-004, Refueling Water Storage Makeup

Proposed Answer: D

Explanation: 1) Per Technical Specifications 3.2.B.2.b.3, and 3.3.A.1.b the RWST must contain a boron concentration of at least 2300 ppm but not greater than 2500 ppm. Per Tech Specs 3.3.A.2.b the Safety Injection Accumulator must contain a boron concentration of at least 2250 ppm. 2) To restore boron concentration to the RWST procedure 1-OP-CS-004, Refueling Water Storage Makeup must be used. 1-OP-CH-23 is used for operating the blender to perform a boration to the RCS.

Technical Reference: Technical Specifications, Change 47. 1-OP-CS-004, Refueling Water Storage Makeup, Rev. 18

Reference Provided to Applicant: No

Learning Objective: ND-88.3-LP-9, Blender Ctrl Subsystem, Objective C, Using a sketch of the system drawn from memory, describe the flowpaths into and out of the blender. Objective D, Operate the blender control system during all modes of operation.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.5 / 43/5 / 45/3 / 45/5)

Comments:

K/A Analysis: The first part of the K/A requires the candidate to know the minimum RWST boron, which is asked in part 1, and affirms that in the scenario RWST boron level *dropped below the minimum required boron.* Part 2 of the K/A requires knowledge of procedures to correct control or mitigate the consequence. The candidate is asked for specific procedure that is used to makeup to the RWST and restore boron to the correct level.

Distractor Analysis:

A. 1) 2250
   2) 1-OP-CS-004, Refueling Water Storage Makeup

B. 1) 2250
   2) 1-OP-CH-23, Boration Followed by a Manual Make-up using the Blender

C. 1) 2300
   2) 1-OP-CH-23, Boration Followed by a Manual Make-up using the Blender

D. 1) 2300
   2) 1-OP-CS-004, Refueling Water Storage Makeup

A. Incorrect. 1) Incorrect, Per Tech Specs the RWST must contain a boron concentration of at least 2300 ppm. Plausible because per Tech Specs Safety Injection Accumulator must contain a boron concentration of at least 2250 ppm. 2) Correct.
B. Incorrect. 1) Incorrect, Per Tech Specs the RWST must contain a boron concentration of at least 2300 ppm. Plausible because per Tech Specs Safety Injection Accumulator must contain a boron concentration of at least 2250 ppm. 2) Incorrect procedure 1-OP-CS-004 is used to makeup to the RWST at proper blended boron. Plausible because 1-OP-CH-23 is used to perform a boration to the RCS.
C. Incorrect. 1) Correct. 2) Incorrect procedure 1-OP-CS-004 is used to makeup to the RWST at proper blended boron. Plausible because 1-OP-CH-23 is used to perform a boration to the RCS.
D. Correct.

Reference, Technical Specifications 3.2.B.2.b, (Correct Answer for part 1)

b. A subsystem supplying borated water from the refueling water storage tank via a charging pump to the Reactor Coolant System consisting of:

1. One OPERABLE flow path,
2. One OPERABLE charging pump,
3. The OPERABLE refueling water storage tank with:
   a. A minimum contained borated water volume of 387,100 gallons,
   b. A boron concentration of at least 2300 ppm but not more than 2500 ppm, and
   c. A maximum solution temperature of 45°F.

Reference, Technical Specifications 3.3.A.1,2 (Correct Answer for part 1, distractor for part 1)

3.3 SAFETY INJECTION SYSTEM

Applicability

Applies to the operating status of the Safety Injection System.

Objective

To define those limiting conditions for operation that are necessary to provide sufficient borated water to remove decay heat from the core in emergency situations.

Specifications

A. A reactor shall not be made critical unless:

1. The refueling water storage tank (RWST) is OPERABLE with:

   a. A contained borated water volume of at least 387,100 gallons.

   b. A boron concentration of at least 2300 ppm but not greater than 2500 ppm.

   c. A maximum solution temperature of 45°F.

2. Each safety injection accumulator is OPERABLE with:

   a. A borated water volume of at least 975 cubic feet but not greater than 1025 cubic feet.

   b. A boron concentration of at least 2250 ppm.

   c. A nitrogen cover-pressure of at least 600 psia.

   d. The safety injection accumulator discharge motor operated valve blocked open by de-energizing AC power and the valves's breaker locked, sealed or otherwise secured in the open position when the reactor coolant system pressure is greater than 1000 psig.

Reference 1-OP-CS-004, Refueling Water Storage Tank Makeup 1.0, 5.1 (correct answer for part 2)

DOMINION
Surry Power Station

1-OP-CS-004
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1.0 PURPOSE

1.1 To provide instructions for makeup to the RWST using the blender or the SFP.

DOMINION
Surry Power Station

1-OP-CS-004
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Init Verif

5.0 INSTRUCTIONS

5.1 Makeup to Unit 1 RWST from the Blender

______ 5.1.1 Notify the STA that Unit 1 RWST will be filled.

______ 5.1.2 Check that the VCT is at desired level and that the blender is not needed for normal makeup to the VCT.

______ 5.1.3 Check emergency borate flowpath is available. <u>IF</u> <u>NOT</u> available, <u>THEN</u> review Tech. Spec. 3.2.C.1.

______ 5.1.4 Check that the blender is not needed for normal boric acid control of the Reactor Coolant System.

Reference 1-OP-CH-23, Boration followed by Manual Makeup 1.0, 5.1 (part 2 distractor)

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1.0 PURPOSE

1.1 To provide instructions for operating the Blender to perform a boration to the RCS followed by a manual make-up.

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Init **Verif**

5.0 INSTRUCTIONS

5.1 Boration

NOTE:
- If Unit on Excess Letdown 1-OP-CH-007 should be used.
- This subsection will be used for the first boration followed by manual make-up of the shift. Attachment 1 will be used as a guide for further borations followed by manual make-ups for the remainder of the shift.

_______ 5.1.1 Notify Shift Supervision and STA of impending Boration followed by manual make-up. (**Reference 2.4.1**)

_______ 5.1.2 Place the MAKE-UP MODE CNTRL switch in the STOP position.

5.1.3 Adjust both of the following controllers for the flow rate and total gallons of Boric Acid for the boration. IF the **BA FLOW CNTRL** controller setpoint has previously been set, THEN enter N/A for that substep.

Reference ND-88.3-LP-2, Blender Ctrl page 23. Describes one of uses of blender in Manual (Bkgd)

4. MANUAL

   a. May be used to provide blended flow to VCT.

   b. May also be used to provide blended flow to spent fuel pit or RWST; details later.

   c. Does not open a flowpath to the VCT; operator must line up to VCT by placing CH-FCV-1113B control switch in OPEN.

   d. Procedure

      (1) The operator places the Make-Up Mode Control Switch to STOP.

K/A Number: 004K5.37: Chemical and Volume Control: Knowledge of the operational implications of the following concepts as they apply to the CVCS: Effects of boron saturation on ion exchanger behavior.

Level: RO Tier #: 2 Group #: 1 IR – RO: 2.6 IR-SRO: 3.1

Proposed Question: 4

Unit 1 is in CSD.

- Current RCS Boron concentration is 2435 ppm.
- 1-OP-CH-011, CVCS Mixed Bed Demin Operations, is in progress to place 1-CH-I-1A, Mixed Bed Demin 1A, in service.

Which ONE of the following identifies:

1) Whether 1-CH-I-1B, Mixed Bed Demin 1B, may be in service during the rinse in of 1-CH-I-1A, Mixed Ben Demin.

2) The effluent boron concentration that will allow 1-CH-I-1A flow to be directed to the VCT.

A. 1) Yes.
 2) 2385.

B. 1) Yes.
 2) 2420.

C. 1) No.
 2) 2385.

D. 1) No.
 2) 2420.

Proposed Answer: D.

Explanation: IAW 1-OP-CH-011, CVCS Mixed Bed Demin Operations, the “B’ demin must be removed from service prior to rinse in of the “A” demin since the demins share a common sample point to ensure that an accurate sample of “A” demin effluent may be obtained (P&L 4.5). Precaution and Limitation 4.3 of 1-OP-CH-011 requires influent and effluent boron concentration to be within 25 ppm before the demin stream is directed to the VCT.

Technical Reference: 1-OP-CH-011, CVCS Mixed Bed Demin Operations, Rev 16.

Reference Provided to Applicant: No.

Learning Objective:

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5/ 45.7)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must recall specific requirements for Demin operations and apply knowledge of influent and effluent boron concentration requirements to ensure a dilution will not occur when the demin is placed in service.

Distractor Analysis:

A. Incorrect. Part 1) is incorrect, “B” demin must be removed from service to ensure accurate effluent sample when “A” demin is rinsed in. Plausible should Candidate fail to recognize that Demins are piped in parallel and share a common effluent sample point. Part 2) is incorrect, effluent boron concentration must be within 25 ppm of the influent. Plausible if Candidate applies the maximum RCS boron concentration to place a Deborating Demin in service (50 PPM) with the maximum boron differential across a demin to return flow to the VCT.

B. Incorrect. Part 1) is incorrect, “B” demin must be removed from service to ensure accurate effluent sample when “A” demin is rinsed in. Plausible should Candidate fail to recognize that Demins are piped in parallel and share a common effluent sample point. Part 2) is correct, the value given is within the maximum 25 ppm difference between inflwunt and effluent.

C. Incorrect. Part 1) is correct, the “B” demin would have to be removed from service. Part 2) is incorrect, effluent boron concentration must be within 25 ppm of the influent. Plausible if Candidate applies the maximum RCS boron concentration to place a Deborating Demin in service (50 PPM) with the maximum boron differential across a demin to return flow to the VCT.

D. Correct. Both Parts 1) and 2) are correct.

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4.3 The effluent C_B and influent C_B must be within 25 ppm of each other before the demin stream is directed to the VCT.

4.5 An accurate sample of Mixed Bed Demineralizer effluent can only be obtained with one demineralizer in service, as the demineralizers share a common sample point.

Excerpt from ND-88.3-LP-2, Charging and Letdown

q. Deborating IXs

(1) Two per unit located on IX alley. Loaded with OH type resin.

(2) Used near the end of life for removing boron from the RCS. Generally used when Boron concentration is less than 50 ppm.

K/A Number: 005A1.03: Residual Heat Removal: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Closed cooling water flow rate and temperature.

Level: RO Tier #: 2 Group #: 1 IR – RO: 2.5 IR-SRO: 2.6

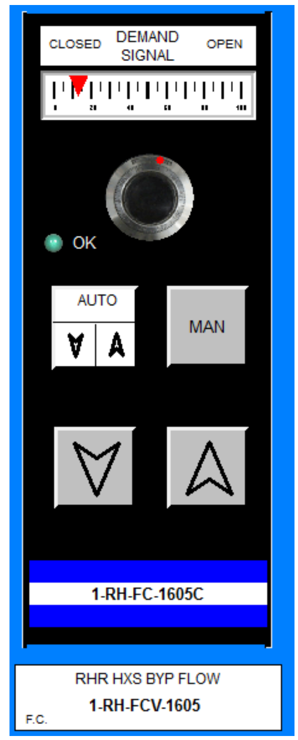
Proposed Question: 5

Initial Conditions:

- Unit 1 is at ISD with cooldown in progress with RHR in service.
- RHR total flow is 3500 gpm.
- 1-HCV-1758 is 10 % OPEN.
- The standby CC pump 1-CC-P-1A was started.

Current Conditions (5 minutes later):

- The RO adjusts the pot for 1-RH-FCV-1605 (shown to the right) in the counter-clockwise direction one full turn.



Which one of the following completes the statements below?

1) This adjustment will cause RHR HX CC Outlet temperature to ______.
2) CC flow through the RHR Heat Exchanger should be maintained less than 9500 gpm to avoid excessive ________.

A.	1) lower	2) pump amps in the CC pump
B.	1) rise	2) vibrations in the RHR heat exchanger
C.	1) lower	2) vibrations in the RHR heat exchanger
D.	1) rises	2) pump amps in the CC pump

Proposed Answer: B

Explanation: 1) Adjusting pot for 1-RH-FCV-1605 in the counter-clockwise direction will lower bypass flow and total RHR flow. This will also cause more flow through the RHR heat exchangers which will cause RCS temperature to lower (CC flow is fixed), and more flow through the RHR Heat exchanger will cause CC temperature to rise. 2) CC flow through the RHR heat exchanger must be maintained <

9500 gpm to limit flow induced vibrations in the RHR heat exchanger. CC pump amps should be < 80 amps after initial pump starting current decays or the pump shall be stopped.

Technical Reference: 1-OP-RH-001, RHR Operations, Rev. 27. OP-51.1, Component Cooling Subsystem, Rev. 8.

Reference Provided to Applicant: No

Learning Objective: ND-88.2-LP-1, RHR System, Objective C; “Assess the function and operation of each RHR system component.”

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 45.5)

Comments:

K/A Analysis: Question requires the operator to predict changes in various indications when operating 1-RH-FCV-1605. Part of the answer is to also determine which parameters (CC Flow) need to be monitored before exceeding design limits. In this case CC flow must be monitored and maintained < 9500 gpm in order to prevent damage caused by vibrations to the RHR heat exchanger.

Distractor Analysis:

A.	1) lower	2) pump amps in the CC pump
B.	1) rise	2) vibrations in the RHR heat exchanger
C.	1) lower	2) vibrations in the RHR heat exchanger
D.	1) rises	2) pump amps in the CC pump

A. Incorrect. 1) Incorrect. RHR heat exchanger CC outlet temperature will rise. Plausible if operator confuses operation of controller and believes *less* flow will flow through the RHR heat exchanger. 2) Incorrect. The limit of 9500 gpm is based on avoiding flow induced vibration of the RHR heat exchanger. Plausible because there is a limit for CC pump amps following initial start of the CC pump, after starting current decays, and if this limit is exceeded the pump should be stopped.
B. Correct.
C. Incorrect. 1) Incorrect. RHR heat exchanger CC outlet temperature will rise. Plausible if operator confuses operation of controller and believes *less* flow will flow through the RHR heat exchanger. 2) Correct.
D. Incorrect. 1) Correct. 2) Incorrect. The limit of 9500 gpm is based on avoiding flow induced vibration of the RHR heat exchanger. Plausible because there is a limit for CC pump amps following initial start of the CC pump, after starting current decays, and if this limit is exceeded the pump should be stopped.

Reference ND-88.2-LP-1. Show operation of components, 1-RH-FCV-1605, 1-RH-HCV-1758.

i. HCV-1758

(1) Air operated butterfly valve located on the RHR flats.

(2) Controls RHR flow through the in-service HX which in turn controls the RCS cooldown rate.

(3) Controlled by a Hagan half-station on BB 1-1.

(4) No automatic controls, no automatic trips.

(5) HCV-1758 fails OPEN.

j. FCV-1605

(1) Air operated butterfly valve located on the RHR flats.

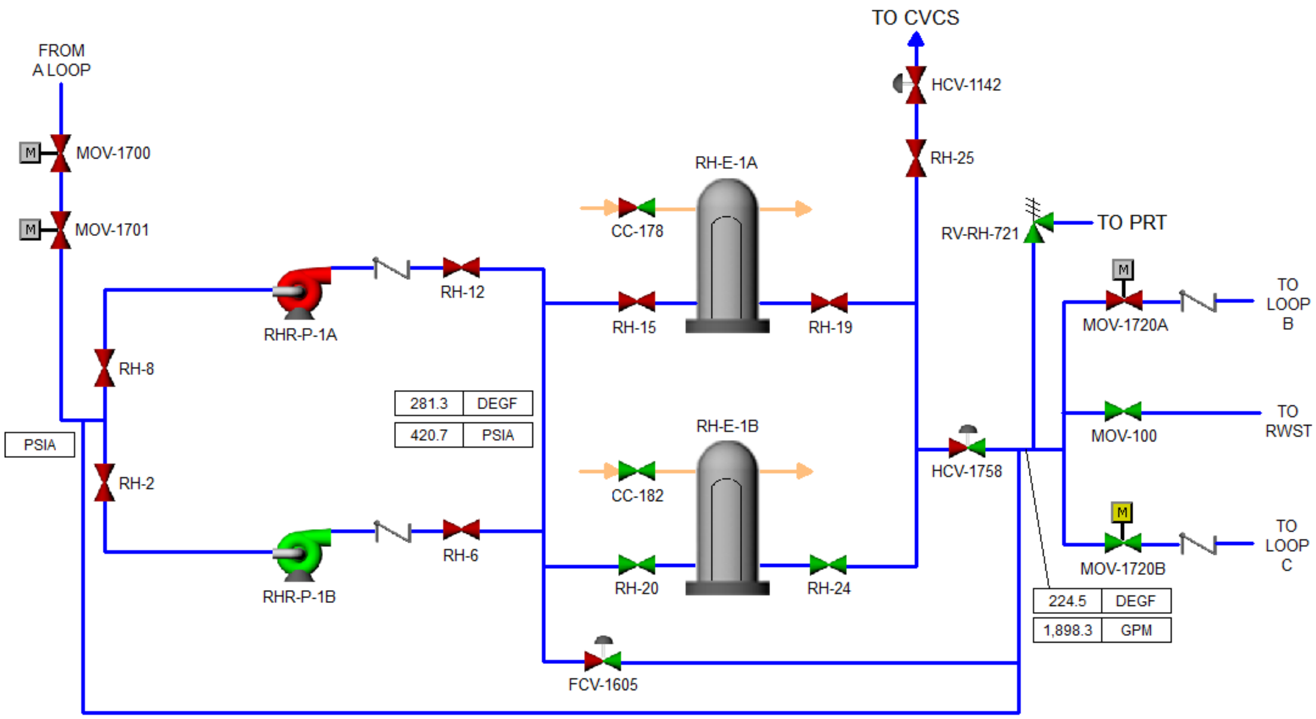
(2) Controls HX bypass flow from a signal generated by FT-1605.

(3) Controlled by Hagan full station on BB 1-1. In automatic, the valve will control flow at the setpoint on the controller using the feedback from FT-1605.

(4) When the position of HCV-1758 is changed, system flow rate is changed. This is sensed by FT-1605 and FCV-1605 is adjusted a corresponding amount to return total system flow to the previous value.

(5) FCV-1605 fails SHUT.

Simulator drawing of RHR showing major components.



When FCV 1605 is closed more flow is forced through RHR H/X. This causes RCS temperature to lower because of increased heat transfer. CC outlet temperature will rise, because of increased RHR flow through the heat exchanger.

Reference 1-OP-RH-001 showing limit of 9500 gpm (page 8) and distractor limit of 6000 gpm.

CAUTION

Sustained CC flow greater than 9500 gpm should be avoided due to flow induced vibration in the RHR HX and piping. (**Ref. 2.4.6**)

5.1.6 Establish CC flow through RHR Heat Exchanger A IAW the following instructions.

______ a. Notify Unit 2 RO that 1-CC-TV-109A will be opened and if necessary, the standby CC pump will be started. (**Ref. 2.4.5**)

______ b. IF the standby CC pump will be started, THEN check personnel are clear of the pump shaft and start the pump. Otherwise, enter N/A. (**Ref. 2.4.5**)

______ c. While monitoring 1-CC-FI-110A, RHR HX A CC Outlet Hdr A FLOW, open 1-CC-TV-109A, RHR HX A CC Rtn Hdr TV. (Flow should increase to approximately 4000 to 6000 gpm.) (**Ref. 2.4.5**)

Reference OP-51.1, Component Cooling Subsystem, showing limit of 80 amps for start of a CC pump.

5.0 INSTRUCTIONS

5.1 Placing in Operation

5.1.1 Throttle pump discharge valve for pump being started to 25 percent open.

5.1.2 Place the pump control mode selector switch for component cooling pump tc the START position. (✓)

_____ 1-CC-P-1A   _____ 1-CC-P-1B

_____ 1-CC-P-1C   _____ 1-CC-P-1D

5.1.3 Slowly open the pump discharge valve on the pump started.

5.1.4 Check pump ammeter and verify pump amps less than 80 after initial starting current decays. IF greater than 80 amps, THEN stop the pump.

5.1.5 Check pump flow indication less than 9000 gpm, flow will vary with the number of components being served.

K/A Number: 006A2.13: Emergency Core Cooling: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent SIS actuation.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.9 IR-SRO: 4.2

Proposed Question: 6

Unit 1 operating at 100% power.

- Vital Bus III/IIIA are lost.
- The Team initiated 1-AP-10.03, Loss of Vital Bus III.
- The Team tripped the reactor and secured the "A" RCP.
- Train "A" and "B" SI actuated.

Which ONE of the following identifies:

1) The SI signal that is actuated by this sequence of events?
2) The procedure used to reset SI.

A. 1) High Steam flow with Low Tave.
   2) 1-E-0, Reactor trip or Safety Injection.

B. 1) Header to line.
   2) 1-E-0, Reactor Trip or Safety Injection.

C. 1) High Steam flow with Low Tave.
   2) 1-AP-10.19, Resetting Safety Injection.

D. 1) Header to line.
   2) 1-AP-10.19, Resetting Safety Injection.

Proposed Answer: A.

Explanation: Loss of Vital Bus III/IIIA cause channel III Instruments to fails low. Channel III High Steam flow is developed for all 3 RCS loops, and the "C" RCS Loop Tave fails low. When the "A" RCP is secured, flow reverses in the "A" RCS Loop causing Low Tave. Thus a HSF with Low Tave SI signal actuates. This would be considered a "spurious" SI actuation.

Technical Reference: E-0, Reactor Trip or Safety Injection, Rev. 71. 1-AP-10.03, Loss of Vital Bus III, Rev. 19.

Reference Provided to Applicant: No

Learning Objective: ND-91-LP-3, SI Sys Operations, Objective A, List the five (5) Safety Injection actuation signals, including setpoints, coincidence, and purposes.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 45.5)

Comments:

K/A Match Analysis: Question matches K/A in that the Candidate must determine the cause for an inadvertent SI, and identify the procedure used to reset SI.

Distractor Analysis:

A. Correct. Both Part 1) and 2) are correct.

B. Incorrect. Part 1) is incorrect, HSF signal is actuated. Plausible if Candidate applies reversal of flow in “A” RCS loop to pressure reduction in “A” SG and addition of AFW to “A” SG could lead to a header to line SI signal. Part 2) is correct.

C. Incorrect. Part 1) is correct, HSF SI would be actuated for this sequence of events. Part 2) is incorrect; SI is reset in E-0, Step 15. Plausible should the Candidate fail to recall the major actions accomplished in E-0.

D. Incorrect. Part 1) is incorrect, HSF signal is actuated. Plausible if Candidate applies reversal of flow in “A” RCS loop to pressure reduction in “A” SG and addition of AFW to “A” SG could lead to a header to line SI signal. Part 2) is incorrect; SI is reset in E-0, Step 15. Plausible should the Candidate fail to recall the major actions accomplished in E-0.

NUMBER	PROCEDURE TITLE	REVISION
1-E-0	REACTOR TRIP OR SAFETY INJECTION	71
		PAGE 8 of 15

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. ___	CHECK IF SI FLOW SHOULD BE REDUCED:	
	☐ a) RCS subcooling based on CETCs - GREATER THAN 30°F	☐ a) GO TO Step 24.
	b) Secondary heat sink:	☐ b) GO TO Step 24.
	☐ • Total feed flow to SGs - GREATER THAN 350 GPM	
	OR	
	☐ • Narrow range level in at least one SG - GREATER THAN 12%	
	☐ c) RCS pressure - STABLE OR RISING	☐ c) GO TO Step 24.
	☐ d) PRZR level - GREATER THAN 22%	☐ d) Try to stabilize RCS pressure with normal PRZR spray. RETURN TO Step 13a.
14. ___	HAVE STA INITIATE MONITORING OF CRITICAL SAFETY FUNCTION STATUS TREES	☐ Assign a person to initiate monitoring of Status Trees.
15. ___	RESET BOTH TRAINS OF SI	

NUMBER	PROCEDURE TITLE	REVISION
1-AP-10.19	RESETTING SAFETY INJECTION (WITH 5 ATTACHMENTS)	1
		PAGE 1 of 4

PURPOSE

To provide guidance for resetting a Safety Injection signal that will not reset using the Main Control Room pushbuttons.

ENTRY CONDITIONS

Failure of either train of Safety Injection to reset from the Main Control Room as indicated by any of the following:

- Annunciator 1A-F3, SI INITIATED TRAIN A, fails to clear using the SI RESET TRAIN A pushbutton
- Annunciator 1A-F4, SI INITIATED TRAIN B, fails to clear using the SI RESET TRAIN B pushbutton
- Failure of components to realign after a Safety Injection

NUMBER	PROCEDURE TITLE	REVISION
1-AP-10.03	LOSS OF VITAL BUS III	19
		PAGE 4 of 12

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- Safety Injection is imminent if Loop A or B Tave drops below 543°F.
- If Safety Injection occurs, RCS temperature should be controlled below 543°F to prevent reoccurring SI signals.
- Most SI flow transmitters may be deenergized.

1-AP-10.03, CAUTION Prior to Step 10

* *

CAUTION: Excessive AFW flow to A SG may result in a Header to Line SI.

* *

K/A Number: 006G2.1.28: Emergency Core Cooling: Knowledge of the purpose and function of major system components and controls.

Level: RO   Tier #: 2   Group #: 1   IR – RO: 4.1   IR-SRO: 4.1

Proposed Question: 7

Which of the following describes:

1) LHSI pump suction is realigned to the containment sump during _____ of Recirculation Mode Transfer (RMT).
2) RMT is manually initiated using _____ push buttons.

A. 1) Phase I
   2) 2 of 2

B. 1) Phase II
   2) 2 of 2

C. 1) Phase I
   2) 1 of 2

D. 1) Phase II
   2) 1 of 2

Proposed Answer: B

Explanation: 1) Automatic swapover to the recirc mode should occur as 2/4 RWST level transmitters reach 13.5%. During phase I the first valves to reposition are the valves between the LHSI pump discharge and the charging pump suctions, 1-SI-MOV-1863 A & B. Therefore initially the LHSI pumps should be taking a suction from the RWST. One minute later, Phase II occurs and the HHSI pump suction valves from the RWST close and the LHSI pump suction valves from the containment sump, 1-SI-MOV1860A and B begin to OPEN. 2) The RMT design allows manual initiation of the system by depressing the initiation pushbuttons. Manual initiation of the system may be accomplished by pressing 2/2 pushbuttons for each train.

Technical Reference: ND-91-LP-3, SI System Operations. 1-ES-1.3, Transfer to Cold Leg Recirculation.

Reference Provided to Applicant: NO

Learning Objective: ND-91-LP-3, SI System Operations, Objective E; Explain the operation of the Safety injection system in response to an RMT signal, including RMT signal setpoint, coincidence, and purpose.

Question Source: New

Question History: Last 2 NRC Exams: No

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.5 / 41.10 / 45.6 / 45.13)

Comments:

K/A Match: Question requires knowledge of purpose and function of RMT. Specifically the candidate needs to have an understanding of how to manually initiate RMT, and the functions that should be taking place during phase 1 of RMT.

Distractor Analysis:

A. Incorrect. 1) Incorrect. Phase I is incorrect as the suction to the LHSI pumps at this time is the RWST, and the suction shifts to Containment sump during Phase II. Plausible if the candidate confuses the actions that occur at Phase I with Phase II. Since HHSI is partially realigned in Phase I, it is plausible that the candidate may believe that LHSI is aligned in Phase I, and HHSI is *fully* aligned in Phase II-which is false. 2) Correct.
B. Correct.
C. Incorrect. 1) Incorrect. Phase I is incorrect as the suction to the LHSI pumps at this time is the RWST, and the suction shifts to Containment sump during Phase II. Plausible if the candidate confuses the actions that occur at Phase I with Phase II. Since HHSI is partially realigned in Phase I, it is plausible that the candidate may believe that LHSI is aligned in Phase I, and HHSI is *fully* aligned in Phase II-which is false. 2) Incorrect as RMT requires 2/2 per train. Plausible if candidate confuses RMT manual initiation with other ESFAS initiations such as manual initiation of SI which only requires ½ for initiation.
D. Incorrect. 1) Correct. 2) Incorrect as RMT requires 2/2 per train. Plausible if candidate confuses RMT manual initiation with other ESFAS initiations such as manual initiation of SI which only requires ½ for initiation.

Reference: ES-1.3, Step 5. This shows requirement to push both RMT actuation pushbuttons for each train. Also shows required valve changes that occur at this time. Note: LHSI pumps still taking a suction from the RWST at this time.

NUMBER	PROCEDURE TITLE	REVISION
1-ES-1.3	TRANSFER TO COLD LEG RECIRCULATION	20
		PAGE 4 of 10

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If suction source is lost to any SI or spray pump, the pump should be stopped.

* *

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.___	ALIGN SI SYSTEM FOR RECIRC	
	a) Close CHG pump miniflow recirc valves ☐ • 1-CH-MOV-1275A ☐ • 1-CH-MOV-1275B ☐ • 1-CH-MOV-1275C	☐ a) Manually close 1-CH-MOV-1373. ☐ IF 1-CH-MOV-1373 does NOT close manually, THEN locally close.
	☐ b) RWST Level - LESS THAN 13%	☐ b) Do NOT continue. WHEN RWST level less than 13%, THEN GO TO Step 5c.
	c) Check Phase 1 - INITIATED ☐ 1) White Phase 1 Status light on bench board - LIT 2) LHSI discharge to HHSI - OPEN ☐ • 1-SI-MOV-1863A ☐ • 1-SI-MOV-1863B 3) LHSI recirc valves - CLOSED ☐ • 1-SI-MOV-1885A ☐ • 1-SI-MOV-1885B ☐ • 1-SI-MOV-1885C ☐ • 1-SI-MOV-1885D	c) Initiate RMT. ☐ Push both RMT actuation pushbuttons for each train. ☐ IF RMT has NOT actuated, THEN manually align valves.

NUMBER	PROCEDURE TITLE	REVISION
1-ES-1.3	TRANSFER TO COLD LEG RECIRCULATION	20
		PAGE 5 of 10

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	ALIGN SI SYSTEM FOR RECIRC (Continued)	
	d) Check Phase 2 - INITIATED ☐ 1) Amber Phase 2 Status light on bench board - LIT 2) LHSI suction from sump - OPEN ☐ • 1-SI-MOV-1860A ☐ • 1-SI-MOV-1860B 3) LHSI suction from RWST - CLOSED ☐ • 1-SI-MOV-1862A ☐ • 1-SI-MOV-1862B 4) CHG pump suction from RWST valves - CLOSED ☐ • 1-CH-MOV-1115B ☐ • 1-CH-MOV-1115D	☐ d) IF swap over does NOT occur after 1 minute time delay, THEN manually align valves.

Distractor for part 1. Suctions don't swap until phase 2.

Reference from ND-91-LP-3, SI System Operations. Shows manual operation of RMT, and manual operation of SI (used as distractor).

5. The initiation of RMT causes a pre-defined sequence of events to take place. The entire sequence for RMT swapover takes approximately 2 1/2 minutes. This sequence of events is as follows:

Refer to/display H/T-3.13, Recirc Mode and use with the following information.

a. The valves between the LHSI pump discharges and the charging pump suctions (MOV-863 A & B) begin to open while the LHSI pump recirc valves (MOV-885 A, B, C, & D) begin to close.

b. One minute later, the HHSI pump normal suction valves (LCV-115 B & D) close and the LHSI pump suction valves from the containment sump (MOV-860 A & B) begin to open.

c. When a fully open signal is received from MOV-860 A & B, the LHSI pump normal suction valves MOV-862A and B, receive a close signal. They are interlocked such that they will not go closed (automatically) until a full open signal is received from MOV-860 A & B.

d. There are two status lights associated with each RMT train.

(1) The white status light illuminates when RMT is initiated and begins valve repositioning (i.e., beginning of Phase I of swapover).

(2) The amber status light illuminates when the 1 minute timer has timed out and the HHSI normal suctions from the RWST and the LHSI suctions from the sump should begin repositioning.

6. The RMT design also allows manual initiation of the system by depressing the initiation pushbuttons.

Refer to/display H/T-3.14, RMT Instrumentation and use with the following information.

a. Manual initiation of the system may be accomplished by pressing 2/2 pushbuttons for each train.

A. Safety Injection Actuation Signals

1. MANUAL Initiation

a. The purpose of the manual SI initiation is to provide LOCA protection during heat-up and/or cooldown while SI is blocked. It also provides backup initiation capability for the other SI actuation signals.

b. Manual initiation is accomplished by depressing one of the two SI pushbuttons located on the vertical section of benchboard 1-1. Depressing either SI pushbutton will initiate both trains of SI.

c. The Manual SI signal cannot be blocked.

K/A Number: 007A1.02: Pressurizer Relief/Quench Tank: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining quench tank pressure.

Level: RO Tier #: 2 Group #: 1 IR – RO: 2.7 IR-SRO: 2.9

Proposed Question: 8

Unit 1 is in Cold Shutdown with the Pressurizer solid.

- The PRT has been drained to 7% in preparation to drain the RCS to a PRZR level of 22%.
- 1-RC-P-1C, “C” RCP, is running.
- An inadvertent safety injection occurs.

Which ONE of the following identifies:

1) The <u>First</u> component that operates to control RCS pressure.
2) The pressure in the PRT that will cause the Rupture disc to fail.

A. 1) PRZR PORV.
   2) 50 psig.

B. 1) PRZR PORV.
   2) 100 psig.

C. 1) RHR RV.
   2) 50 psig.

D. 1) RHR RV.
   2) 100 psig.

Proposed Answer: B.

Explanation: With solid plant pressure control, an inadvertent SI will cause CH and LD to isolate, and align the running HHSI pump to flow into the RCS. The first component to operate to limit the rise in RCS pressure will be the lowest set PRZR PORV at 365 psig (OPMS is enabled); the RHR relief valve is set to relieve at 600 psig; both RVs discharge to the PRT. The PRT rupture disc is set to relieve at 100 psig. 50 psig is established as a distractor since this is the highest pressure expected in the PRT on a design lift of a PRZR PORV.

Technical Reference: 1-AP-10.20, Rev. 15, Response to Spurious Safety Injection with RCS Temperature Less Than 350°F.

Reference Provided to Applicant: No

Learning Objective: ND-88.1-LP-3, PZR and Press Relief. Objective C, Describe the construction, location, and interrelationships of the following major pressurizer relief tank components: PRT vessel, Sparger pipe, Primary grade spray, Nitrogen, vent and drain lines, Instrumentation. ND-91-LP-3, SI Sys Operations, Objective H, Determine the response required if Safety Injection inadvertently actuates with plant conditions <350°F in accordance with 1-AP-10.20, response to Spurious Safety Injection with RCS Temperature Less than 350°F.

Question Source: Modified Bank (Surry, 2014, Question 7)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 45.5)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must assess plant conditions and determine the components that relieve to the PRT, (i.e., which would be expected to lift to control RCS pressure). Also, recall the setpoint of the PRT rupture disc.

Distractor Analysis:

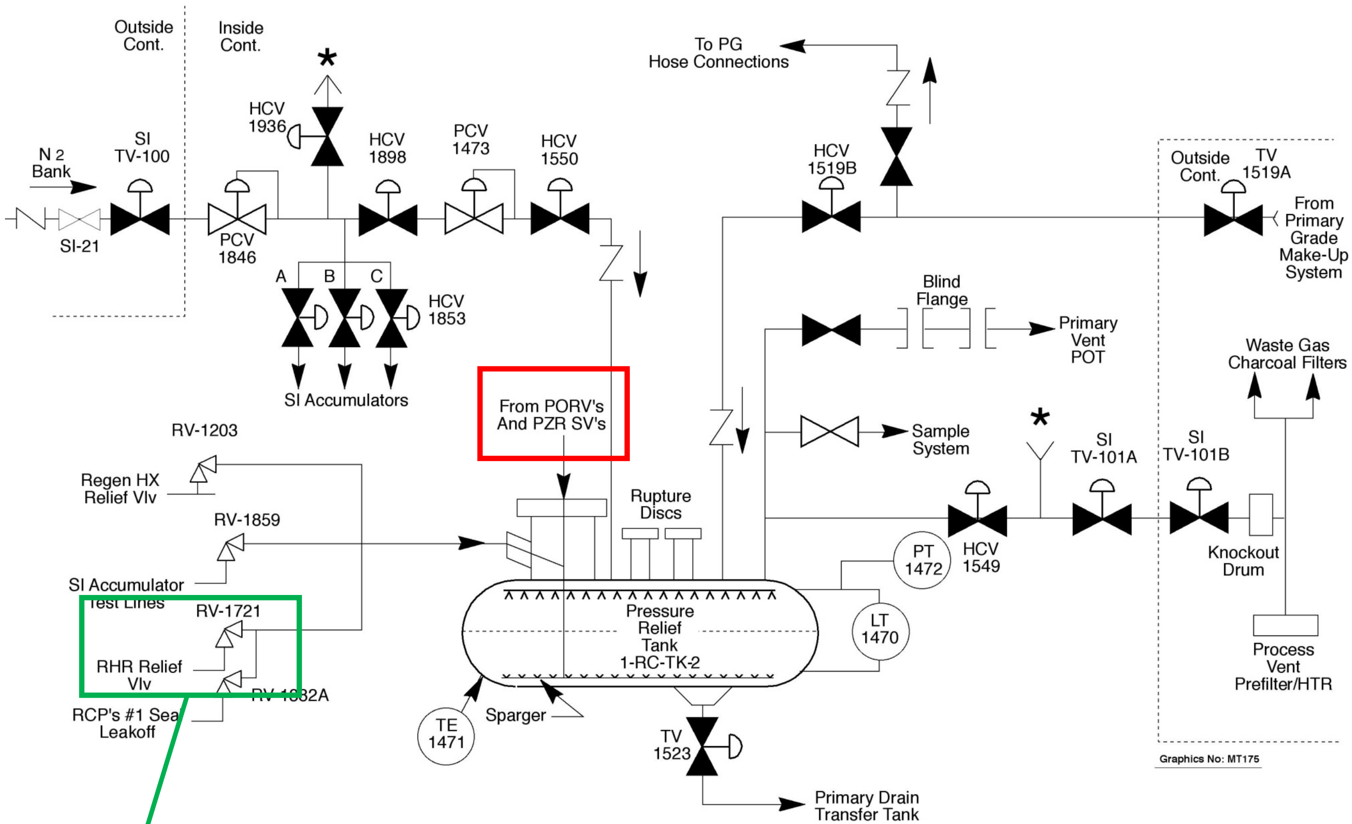
A. Incorrect. Part 1) is correct, the PORV would be expected to lift and control the increase in RCS pressure caused by the SI while solid. Part 2) is incorrect, the rupture disc fails at 100 psig. Plausible if Candidate confuses rupture setpoint with the expected pressure in the PRT following a design basis relief valve lift.

B. Correct. Both Parts 1) and 2 are correct.

C. Incorrect. Part 1) is incorrect, RHR relief valve lifts at 600 psig. Plausible if Candidate fails to account for OPMS being enabled under the plant condition given. Part 2) is incorrect, the rupture disc fails at 100 psig. Plausible if Candidate confuses rupture setpoint with the expected pressure in the PRT following a design basis relief valve lift.

D. Incorrect. Part 1) is incorrect, RHR relief valve lifts at 600 psig. Plausible if Candidate fails to account for OPMS being enabled under the plant condition given. Part 2) is correct.

References: Excerpt from ND-88.1-LP-3, PZR and Press Relief, Page 19:



PRESSURIZER RELIEF TANK DIAGRAM

RHR relief valve: 600 psig relief valve protects against RHR overpressure.

5. Rupture Disks. Overpressure protection for the PRT is provided by the rupture disks located on top of the tank.

   a. The rupture discs have a capacity slightly greater than the combined capacity of the pressurizer safety valves and are designed to rupture at 100 psig.

   b. The rupture pressure is twice that expected in the design discharge. This feature enables the discs to provide overpressure protection for the PRT while avoiding disc deformation during design operation.

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5.2 Preparations for Draining to 22% Pressurizer Level

5.2.4 Check or drain the Pressurizer Relief Tank to between 5% and 10% IAW 1-OP-RC-011, Pressurizer Relief Tank Operations.

Parent Question: Question 7 of NRC Exam 2014:

K/A Number: 007A1.02, Pressurizer Relief/Quench Tank / 5, Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including:, (CFR: 41.5 / 45.5) Maintaining quench tank pressure

Level: RO Tier #: 2 Group #: 1 IR – RO: 2.7 IR-SRO: 2.9

Proposed Question 7:

Given the following:

- Unit 2 is operating at 100% power.
- Annunciator 2C-F7, PRZR RELIEF TK HI PRESS is received
- PRT Pressure is 10.3 psig and rising slowly.
- PRT level is 63% and slowly rising.
- PRT Temperature is 92 °F.

Which ONE of the following describes:

1) The action required to restore PRT parameters in accordance with ARP 2C-F7 and 2-OP-RC-011, PRT Operations.
2) The lowest pressure at which the rupture disc will rupture.

A. 1) Drain the PRT 2) 50 psig

B. 1) Vent the PRT 2) 50 psig

C. 1) Drain the PRT 2) 100 psig

D. 1) Vent the PRT 2) 100 psig

Proposed Answer: D

NRC APPROVED. K/A Number: 007EG2.4.45: ReactorTrip - Stabilization - Recovery /1: Ability to prioritize and interpret the significance of each annunciator or alarm.

Level: RO Tier #: 1 Group #: 1 IR – RO: 4.1 IR-SRO: 4.3

Proposed Question: 9

Initial Conditions:

- Unit 1 was operating at 100% when a spurious Reactor Trip occurred.
- All equipment responded normally.

Current Conditions:

- Crew is currently performing 1-ES-0.1, Reactor Trip Response, step 8 “Initiate Boration of RCS”.

Which of the following lit alarms listed below is the highest priority, and MUST be addressed first?

A. 1D-A1, BA TK 1A HI-LO LVL CH1.

B. 1D-A3, BA TK 1A HI-LO TEMP CH1.

C. 1D-F2, STM GEN BD HI-LO-FLOW.

D. 1D-G1, VCT HI-LO LVL.

Proposed Answer: D

Explanation: VCT HI-LO LVL is the highest priority alarm. This alarm will require immediate actions in accordance with 0-AP-53.00, LOSS OF VITAL INSTRUMENTATION/CONTROLS if there is an instrumentation failure. This alarm could signify a Low level which if unattended to could cause damage to the Charging pumps which are an ESF. This alarm could also signify a LOCA. The other alarms 1D-A1, 1D-F2, and 1D-A5 are alarms that could actuate during extended boration and/or would be expected alarms following a reactor trip.

Technical Reference: 0-AP-53.00, Rev. 21. ARP 1D-A1, BA TK HI-LO LVL CH-1, Rev. 0. ARP 1D-A3, BA TK 1A HI-LO TEMP CH-1, Rev. 1. ARP 1D-F2, STM GEN BD HI-LO-FLOW, Rev. 0. ARP 1D-G1, VCT HI-LO LVL, Rev. 2. 1-ES-0.1, Reactor Trip Response, Rev. 51.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-4, ES-0.1, Objective B; Given a copy of ES-0.1, Reactor Trip Response, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition. ND-93.4-LP-12, Objective K; Given a set of plant conditions, determine the appropriate operator response in accordance with 0-AP-53.00, Loss of Vital Instrumentation Controls, to include the following: Immediate actions, Entry conditions, Major actions, Step Basis.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Comments:

K/A Analysis: The question requires the operator analyze the alarms that appeared simultaneously and determine which one has the highest priority. This meets the K/A.

Distractor Analysis:

A. 1D-A1, BA TK 1A HI-LO LVL CH1.

B. 1D-A3, BA TK 1A HI-LO TEMP CH1.

C. 1D-F2, STM GEN BD HI-LO-FLOW.

D. 1D-G1, VCT HI-LO LVL.

A. Incorrect. This alarm does not require immediate attention, the instrumentation is not considered vital and there is no threat to ESF equipment. Plausible because this alarm could actuate during extended boration and the Lo level corresponds to Tech Spec boration values of 6000 gal which is required if the reactor is critical (Plant is in Hot SD).
B. Incorrect. This alarm does not require immediate attention, the instrumentation is not considered vital and there is no threat to ESF equipment. Plausible because at power the minimum Boric acid temperature shall be 112°F, but the reactor is shutdown so this does not apply.
C. Incorrect. This alarm does not require immediate attention. Plausible because, at power, this alarm could indicate a BD line malfunction resulting in high or low BD flow. This is an expected alarm following a reactor trip since the BD trip valves close on AFW initiation.
D. Correct. This alarm could indicate a LOCA or failure of vital instrumentation, both of which requires immediate actions per AP-16.00 or AP-53.00.

Reference: 1D-G1, VCT HI-LO LEVEL (correct answer). Shows actions that could be indicative of a LOCA.

NUMBER	PROCEDURE TITLE	REVISION
1D-G1	VCT HI-LO LVL	2
		PAGE 2 of 4

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. ___	CHECK VCT LEVEL - LESS THAN OR EQUAL TO 24% ☐ • 1-CH-LI-1112 OR ☐ • 1-CH-LI-1115	☐ GO TO Step 7.
2. ___	VERIFY CHARGING AND LETDOWN FLOW BALANCED	☐ Align Letdown or Charging as necessary.

* *

CAUTION: At 13% in the VCT, charging pump suction should switch from the VCT to the RWST and annunciator 1D-H1, VCT LO-LO LVL, should actuate.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. ___	CHECK MAKEUP - INITIATED	☐ Manually initiate makeup.
4. ___	CHECK VCT LEVEL DIVERT VALVE 1-CH-LCV-1115A - DIRECTING FLOW TO THE VCT	☐ Position divert valve to flow to the VCT.
5. ___	CHECK RCS LEAK RATE - NORMAL	☐ GO TO 1-AP-16.00, EXCESSIVE RCS LEAKAGE.

AP-53.00. Shows actions that could be indicative of a failure of vital instrumentation.

NUMBER	PROCEDURE TITLE	REVISION
0-AP-53.00	LOSS OF VITAL INSTRUMENTATION / CONTROLS	21
		PAGE 3 of 13

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud.
- When the affected instrument/controller malfunction(s) has been addressed by this procedure, recovery actions should continue at Step 13.

*4.___ DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE:

- ☐ • PRZR Pressure Control, Step 5
- ☐ • NI Malfunction, ( )-AP-4.00
- ☐ • SG Feed Flow, Step 7
- ☐ • SG NR Level, Step 7
- ☐ • SG Pressure, Step 7
- ☐ • SG Steam Flow, Step 7
- ☐ • Turbine First Stage Pressure, Step 8
- ☐ • Median Tave, Step 9
- ☐ • Loop Tave, Step 9
- ☐ • Loop ΔT, Step 9
- ☐ • Steam Dumps / SG PORVs, Step 10
- ☐ • Turbine Controls, Step 10
- ☐ • Turbine Valve Position, Step 10
- ☐ • Turbine Monitoring Lights, Step 10
- ☐ • PRZR Level Control, Step 11
- ☐ • Reactor Coolant Flow, Step 12a
- ☐ • PRZR Pressure Protection, Step 12b
- ☐ • RCS Wide Range Pressure, Step 12f
- ☐ • VCT Level, Step 12e
- ☐ • CTMT Pressure, Step 12c
- ☐ • RWST Level, Step 12d
- ☐ • Underground FO Storage Tk, Step 12g
- ☐ • CAT Level, Step 12h
- ☐ • Emergency Condensate Makeup Tank Level, Step 12i
- ☐ • FP/Domestic Water Level, Step 12j

1D-A1 (distractor). If reactor was critical this alarm could provide indication of Tech Spec function not being met.

NUMBER	PROCEDURE TITLE	REVISION
1D-A1	BA TK 1A HI-LO LVL CH-1	0
		PAGE 2 of 3

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Tech Spec 3.2 requires 6000 gallons of boric acid solution to be available when Reactor is critical. The tank level corresponding to 6000 gallons is 79%.

* *

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK BORIC ACID TANK 1-CH-TK-1A LEVEL - LESS THAN OR EQUAL TO 83% ON LI-106	☐ GO TO Step 3.
2.___	MAKE PREPARATIONS TO FILL 1-CH-TK-1A AND GO TO STEP 8	
3.___	CHECK BORIC ACID TANK 1-CH-TK-1A LEVEL - GREATER THAN OR EQUAL TO 100% ON LI-106	☐ Initiate a work request and GO TO Step 8.

1D-A3 (distractor). If reactor was critical this alarm could provide indication of Tech Spec function not being met.

NUMBER	PROCEDURE TITLE	REVISION
1D-A3	BA TK 1A HI-LO TEMP CH-1	1
		PAGE 2 of 3

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Tech Spec 3.2 requires BA solution temperature to be at least 112°F.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. ___	LOCALLY CHECK BORIC ACID TANK 1-CH-TK-1A TEMPERATURE: ☐ • GREATER THAN OR EQUAL TO 155°F ON TI-107 OR ☐ • LESS THAN OR EQUAL TO 135°F ON TI-107	☐ Initiate a work request and GO TO Step 7.

1D-F2 (used as a distractor). Lower priority, expected alarm.

NUMBER	PROCEDURE TITLE	REVISION
1D-F2	STM GEN BD HI-LO FLOW	0
		PAGE 2 of 3

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: The affected PCV should close when BD flow is greater than or equal to 75 GPM OR blowdown temperature reaches 145°F and PCV is in AUTO. This should result in a 0 GPM flow indication.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. ___	CHECK BD FLOW - GREATER THAN OR EQUAL TO 75 GPM ON ONE OF THE FOLLOWING: • STM GEN A BD FLOW HI RNG FI-BD-104A • STM GEN B BD FLOW HI RNG FI-BD-104B • STM GEN C BD FLOW HI RNG FI-BD-104C	Do the following: a) IF blowdown flow normal, THEN initiate a work request AND GO TO Step 12. b) IF blowdown flow less than or equal to 10 GPM, THEN GO TO Step 6.
2. ___	CHECK CAUSE FOR HIGH FLOW - AFFECTED RELIEF VALVE LIFTING • 1-BD-RV-100A • 1-BD-RV-100B • 1-BD-RV-100C	Do the following: a) IF high flow due to equipment failure, THEN initiate a work request AND GO TO Step 12. b) IF high flow due to improper adjustments, THEN adjust flow to chemistry requirements AND GO TO Step 12.

ES 401-9 COMMENTS WITH CORRECTIONS SHOWN IN RED, 3/28/16

Window dressing. All that is required to know is plant trip, alarms arrive concurrently, and a boration is in progress. Do not give the list of alarms twice. Distractor "c" could indicate a reason for the VCT level problem (i.e. the boration flow may restore VCT level). There could be two defensible correct answers.

Cleaned up stem to remove 1st list of alarms. Changed distractor "C".

NRC APPROVED. K/A Number: 008AK3.01: Pressurizer Vapor Space Accident / 3: Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: Why PZR level may come back on scale if RCS is saturated.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.7 IR-SRO: 4.4

Proposed Question: 10

The following conditions exist on Unit 1:

- A loss of offsite power has occurred.
- 1-RC-SV-1551A, "A" PRZR Safety Valve, lifts and fails to reseat.
- RCS pressure is 700 psig.
- PRZR level is 5% and rising.
- The PRT has ruptured.

Which ONE of the following identifies:

1) The cause of the rapid change in PRZR level.
2) The sequence of procedure transition expected from E-0.

A. 1) SI accumulator injection.
   2) E-1, Loss of Reactor or Secondary Coolant.

B. 1) SI accumulator injection.
   2) ES-1.1, SI Termination.

C. 1) RCS voiding.
   2) E-1, Loss of Reactor or Secondary Coolant.

D. 1) RCS voiding.
   2) ES-1.1, SI Termination.

Proposed Answer: C.

Explanation: After the RCPs lose power, the RCS Th and Core area will reach saturation quickly. This will cause PRZR level to rise due to the bubble shifting from the PZR to the reactor head due to loss of head cooling flow. RC pressure will lower due to the inventory loss through the open SV. However, the SI accumulators are at ~600 psig and will not inject until RCS pressure drops below that value.

Technical Reference: ES-1.2, Background document, Section 2.3.

Reference Provided to Applicant: No.

Learning Objective: ND-95.2-LP-7, Loss of Coolant Accidents, Objective B. Assess the expected plant response to a small break loss of coolant accident.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.5,41.10 / 45.6 / 45.13)

Comments:

K/A Match analysis: Question matches K/A. Candidate must assess the event in progress and determine the cause for the rapid change in PRZR level, and determine the appropriate procedure to address the event after leaving E-0. Question is assessing Candidate understanding of TMI event.

Distractor Analysis:

A. Incorrect. Part 1) is incorrect, once the RCS reaches saturation, voiding begins in the Th legs and the vessel causing PRZR level to rapidly rise. Plausible if Candidate fails to consider that a PRZR SV failing open will cause saturation conditions to be reached in the TH/Vessel shortly after RCPs stop. Part 2) is correct.

B. Incorrect. Part 1) is incorrect, once the RCS reaches saturation, voiding begins in the Th legs and the vessel causing PRZR level to rapidly rise. Plausible if Candidate fails to consider that a PRZR SV failing open will cause saturation conditions to be reached in the TH/Vessel shortly after RCPs are secured. Part 2) is incorrect, terminating SI is incorrect since the PRZR level raises to 100% with inventory loss reduced but still occurring. Plausible should Candidate place too much emphasis on PRZR level rise leading difficulties in pressure control vice taking action to control RCS leak rate.

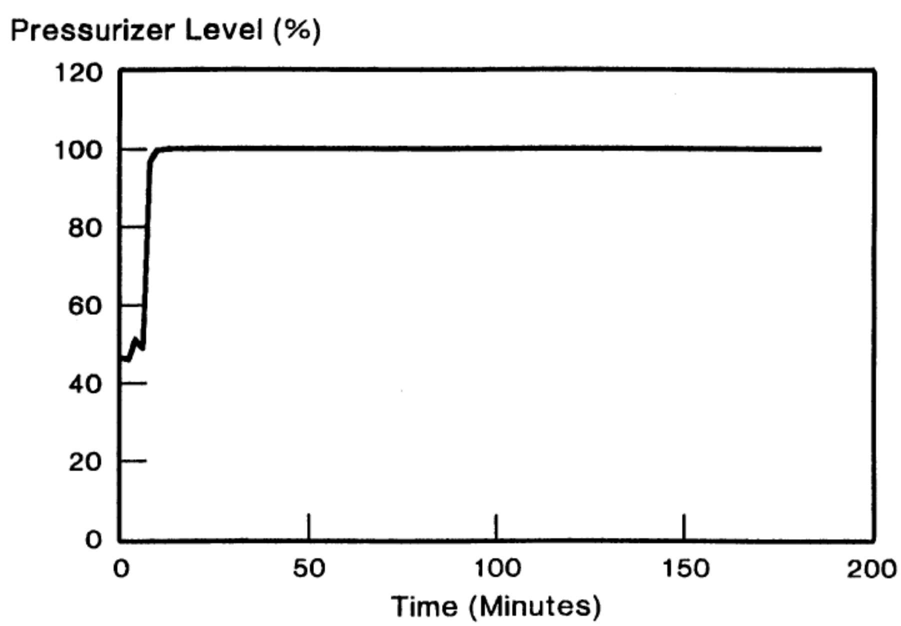
C. Correct. Both Parts 1) and 2) are correct.

D. Incorrect. Part 1) is correct, voiding is the predominant reason for the rapid rise in PRZR level. Part 2) is incorrect, terminating SI is incorrect since the PRZR level raises to 100% with inventory loss reduced but still occurring. Plausible should Candidate place too much emphasis on PRZR level rise leading difficulties in pressure control vice taking action to control RCS leak rate.

Excerpt from ES-1.2 Background Document, Section 2.3, Stuck Open PORV Case.

Again, TREAT does not model all the controls and actions possible in E-0 and E-1. The following is one possible scenario leading to the ES-1.2 entry. The first 19 steps in E-0 (mostly verifications) were assumed to be completed quickly (within several minutes after trip). At E-0, Step 20, it was assumed that the stuck-open PORV could not be blocked and a transition to E-1, Step 1, was made. The RCPs were then tripped (t= 5:58) since RCS pressure had dropped below 1300 psig and RCS subcooling was less than 20°F (assumed subcooling errors). Step 5 of E-1 provides another check on the PORVs. Again, it was assumed that the faulted PORV could not be blocked. In E-1, Step 6, RCS subcooling was too low; consequently, the SI termination criteria were not satisfied. Finally, at t= 20:00, the ES-1.2 transition from E-1, Step 13, was taken.

Figure 13. PRESSURIZER LEVEL FOR STUCK OPEN PORV WITH ES-1.2 GUIDELINE APPLICATION



ES-1.2 Background
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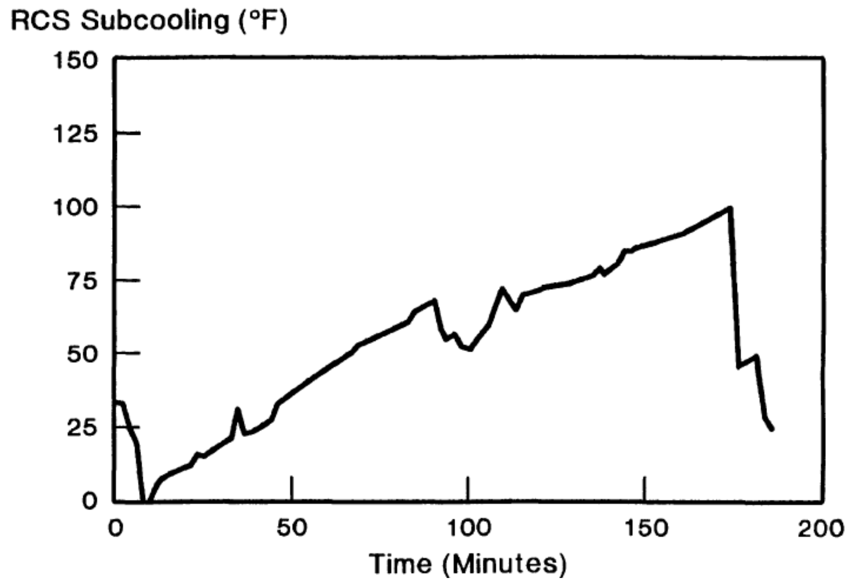
HP-Rev. 2, 4/30/2005

Figure 14. RCS SUBCOOLING FOR STUCK OPEN PORV WITH ES-1.2 GUIDELINE APPLICATION



ES 401-9 COMMENTS WITH CORRECTIONS SHOWN IN RED, 3/28/16

Asking for “predominant” cause is too vague. Both ECCS flow and voiding are contributing to level rise. Determining the relative contribution of each is beyond the scope of an operator’s job. Also, the crew may never arrive at E-1. If the diagnosis step occurs prior to PRT rupture, an E-1 transition may not occur. After the PZR is full, RCS pressure will probably rise, causing the plant to meet ECCS termination criteria. Window dressing. If the plant is at 100% power and a safety lifts and fails to reseat, it is not necessary to say that trip and SI result. RCP trip does not change Pzr level response.

Changed stem to state LOOP had occurred (RCPs lose power). Removed reference to reactor trip and SI. Stated that RCS pressure was 700 psig to make SI accumulator injection plausible, but incorrect. Added PRT rupture to ensure transition to 1-E-1 was correct path. Removed “predominant”. Changed distractor from “SI flow” to “SI accumulator injection”.

K/A Number: 008K2.02: Component Cooling Water: Knowledge of bus power supplies to the following: CCW pump, including emergency backup.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.0 IR-SRO: 3.2

Note: Discussion with Chief Examiner (B. Caballero): We discussed how the K/A does not limit them to testing the backup power supply. This was relevant because the CCW pumps do not have a backup power supply at Surry. We discussed how the K/A was inclusive of testing backup power supplies, but the K/A was not restricted to testing the backup power supply.

Proposed Question: 11

Initial Conditions:

- Unit 1, and Unit 2 are both operating at 100% power.
- "B" CC pump is running on Unit 1.

Current Conditions:

- An electrical fault in the switchyard causes the following Unit 1 annunciators to alarm:
  - 1K-D3, BUS 1D UV.
  - 1K-E3, BUS 1E UV.
  - 1K-G6, 4KV BUS RSV SUP BKR OPEN.
  - 1K-H3, BUS 1J UNDERVOLT.

Which ONE of the following completes the following statement:

The __1)__ CC pump is running on Unit 1, and is powered from ___2)___.

A. 1) "A" 2) RSST "C"

B. 1) "A" 2) EDG #1

C. 1) "B" 2) EDG #3

D. 1) "B" 2) RSST "A"

Proposed Answer: A

Explanation: The electrical fault results in loss of Transfer bus D, and E which in turn supply 4160V Emergency buses 1J and 2H. Unit 1 CC pump B is powered from bus 1J, therefore this pump will turn OFF. When power is lost the stub bus supplying CC pump B will trip open and require manual action to re-close. EDG 3 will auto start and auto load onto bus 1J but the stub bus breaker is open and requires manual action to close, therefore CC pump B will deenergize and will not be energized by the EDG until the stub bus breaker is manually closed. CC pump A, will auto start on low discharge pressure after CC pump B is deenergized. RSST "C" which powers Tranfer bus F and in turn powers 4160V bus 1H is unaffected, which allows the A CC pump to auto start with power from RSST C.

Technical Reference: ARP 1K-E3, BUS 1E UNDERVOLT. ARP 1K-D3, BUS 1D UNDERVOLT.

Reference Provided to Applicant: NO

Learning Objective: ND-88.5-LP-1 OBJECTIVE A, Describe the major system components including their function and power supply if applicable including: CC Surge Tank, Radiation Monitors, **CC Pumps,** Chemical Addition, CC Heat Exchangers.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7)

Comments:

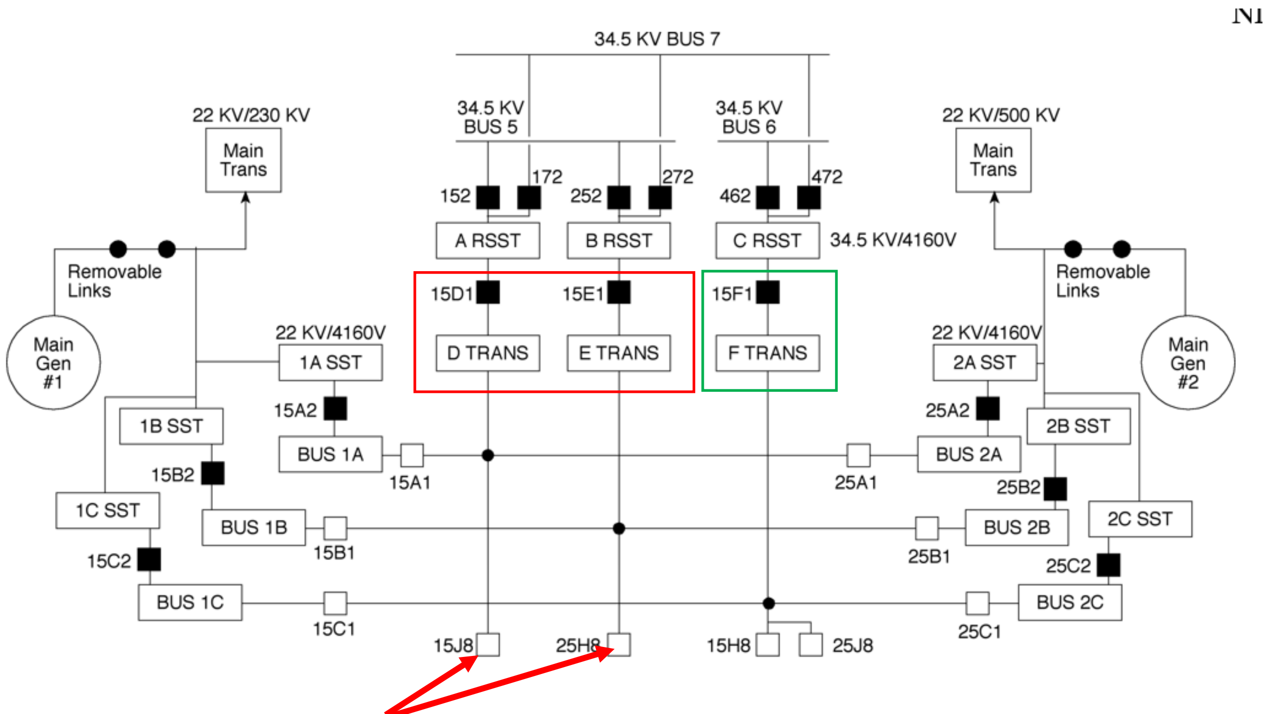
K/A Match Analysis: Question requires an understanding of the power supply from the switchyard to the CC pump itself. Question also requires an understanding of how the components in the power chain respond. An example of this are the stub busses tripping when power from the Transfer bus is lost.

Distractor Analysis:

A. 1) “A” 2) RSST “C”
B. 1) “A” 2) EDG #1
C. 1) “B” 2) EDG #3
D. 1) “B” 2) RSST “A”

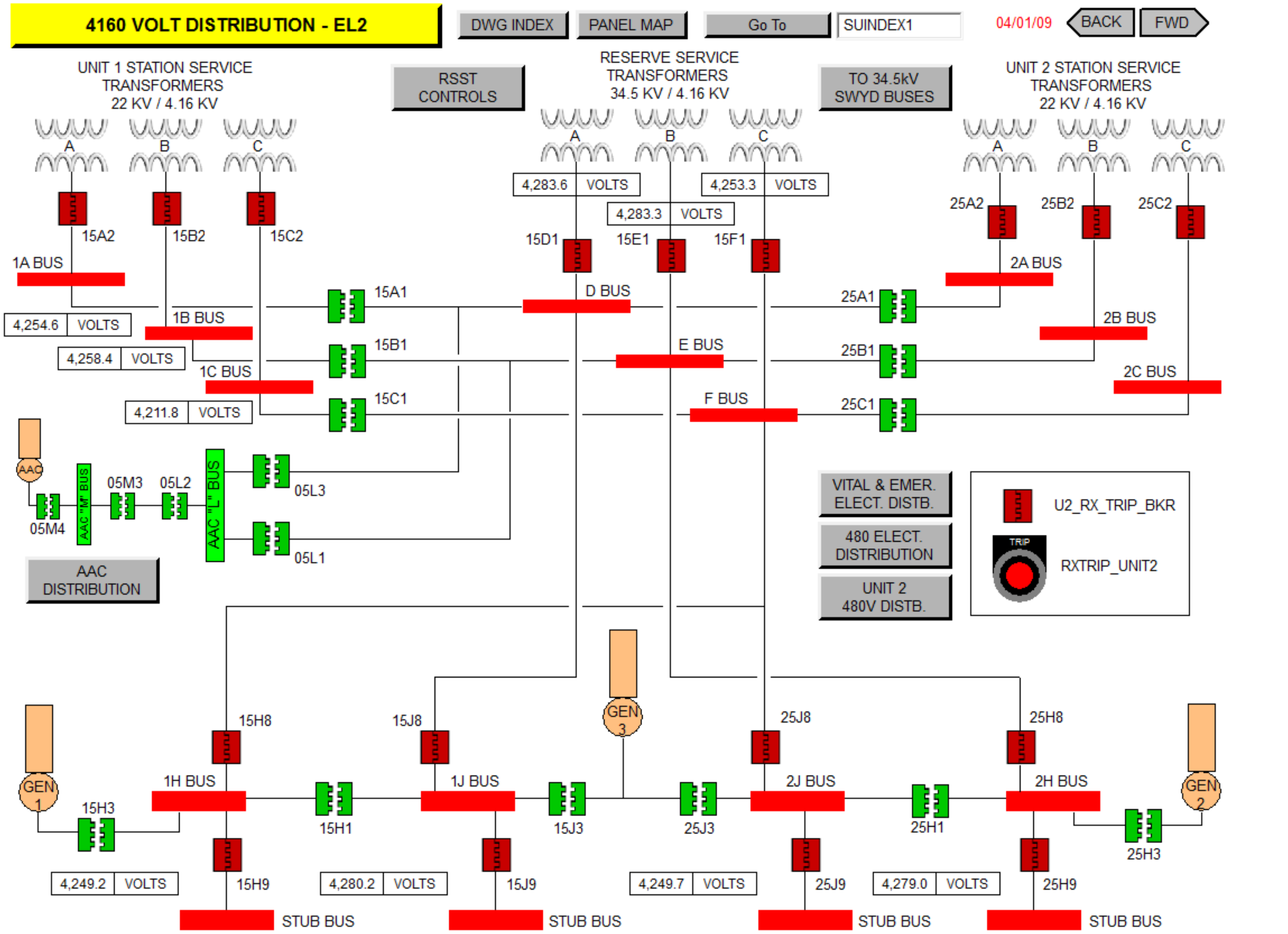
A. Correct.
B. Incorrect. 1) Correct. 2) Incorrect. Plausible if the candidate confuses the RSST power supply to the respective bus and believes the RSST supplying this pump is lost. Also plausible because EDG 1 is the emergency power supply to CC pump A.
C. Incorrect. 1) Incorrect. The pump running is not correct as this pump lost power. Plausible if the candidate confuses the RSST power supply to the respective bus and also believes the CC pump will ride the bus, when the EDGs repower the bus. 2) Incorrect. Plausible because EDG 3 will auto start and load onto bus 2J.
D. Incorrect. 1) Incorrect. The pump running is not correct as this pump lost power. Plausible if the candidate confuses the RSST power supply to the respective bus. 2) Incorrect. Plausible if the candidate believes this RSST is not affected because this is the normal RSST supply to CC Pump B.

Reference from ND-90.2-LP-2, Station Service Distribution. Red box shows the power source to the running CC pumps which is lost. The Green box shows the Transfer bus that is unaffected.

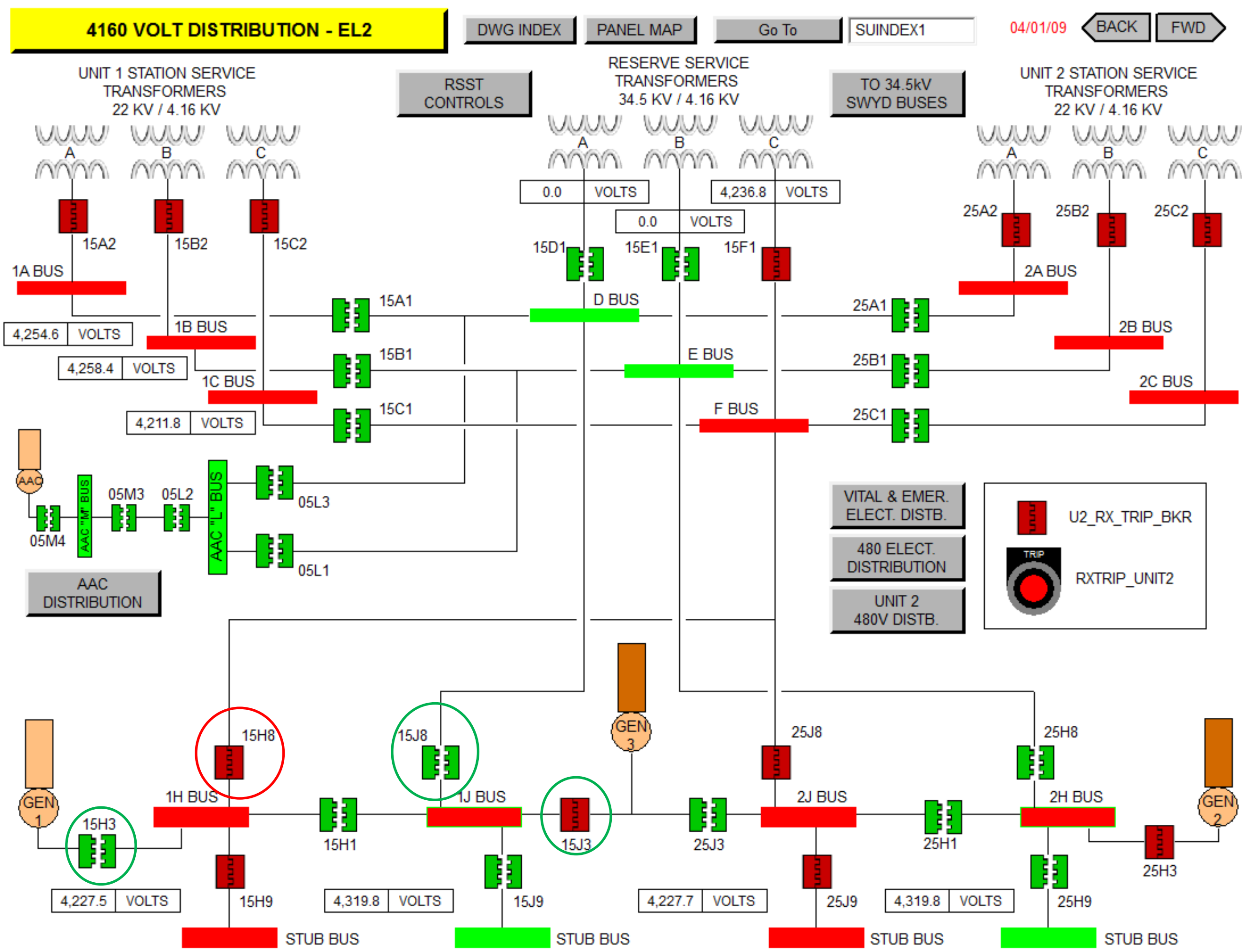


Shows power lost to 1J and 2H bus. This results in loss of power to running CC pumps B and C.

Simulator drawing (Classroom Open Sim) showing normal power supply at 100% (no failures). The "B" CC pump is powered from 1J stub bus which in turn receives power from Reserve Station Service Transformer A (RSST A)



Simulator drawing (Classroom Open Sim) showing power supply following switchyard failure that results in loss of power to RSST A and B. The stub bus for "B" CC pump is open and even though the 1J bus is powered from EDG 3, the B CC pump is OFF because the stub bus is open and requires manual action. The A CC pump starts on low discharge pressure and is unaffected as it's RSST C remains powered.



Reference from NCRODP 51-S, Component Cooling system showing power supplies to CC pumps. Power to 1H and 2J is unaffected.

TABLE 51-3
POWER SUPPLIES

Component	Power Supply
Component cooling pump 1A (Unit 1)	Bus 1H, Breaker 15H10
Component cooling pump 1B (Unit 1)	Bus 1J, Breaker 15J10
Component cooling pump 1C (Unit 2)	Bus 2H, Breaker 25H10
Component cooling pump 1D (Unit 2)	Bus 2J, Breaker 25J10
Charging pump cooling water pump 2A	MCC-1H1-1, Breaker 53
Charging pump cooling water pump 2B	MCC-1J1-1, Breaker 44
Chilled CC pump 1A	MCC-1A2-1, Breaker 52
Chilled CC pump 1B	MCC-1C2-1, Breaker 42
	MCC-2C2-1, Breaker 34
Chilled CC pump 1C	MCC-2A2-1, Breaker 41

Reference from NCRODP 51-S, Component Cooling system showing CC pump standby starting.

Component Cooling Pumps. The CC pumps provide the motive force for circulating cooling water through the CCHXs, individual subsystem loads, and back to the pumps' suction. Normally, two pumps (one per reactor unit) supply the required cooling water flow rate when both reactor units are in operation. Two standby pumps supply 100 percent backup capability. The standby pump for each reactor unit automatically starts on low discharge pressure. The pumps are operated from MCR Vertical Board 1-2. Pump controls are discussed in the Instrumentation and Controls section.

Reference from ND-90.3-LP-7, page 40, showing description of stub breaker trip which supplies CC and RHR pumps for respective Emergency bus.

bus?(K)

Answer: The following breakers receive a trip signal after 60 seconds of degraded voltage or 2 seconds of undervoltage:

- 15H9 Stub bus
- 15H10 CC pump
- 15H11 RHR pump
- 15H5 "A" charging pump if "C" normal is racked in and closed
- 15H6 "C" charging pump if 15H5 not open in 1 second

NUMBER	PROCEDURE TITLE	REVISION
1K-D3	BUS 1D UNDERVOLT	2
		PAGE 2 of 2

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. ___	CHECK BREAKER 15D1 - TRIPPED ☐ • Reserve Supply to Bus D	☐ IF undervoltage present, THEN trip 15D1 AND GO TO Step 2. ☐ IF undervoltage NOT present, THEN initiate a Work Request AND GO TO Step 4.
2. ___	VERIFY AUTOMATIC ACTIONS:	
	☐ a) Check EDG 3 - Supplying 1J Bus	☐ a) Initiate 0-AP-17.05, EDG 3 - EMERGENCY OPERATIONS.
	☐ b) Check 15A1, Station Service Reserve Supply Bkr Bus 1A - TRIPPED	☐ b) Manually trip breaker.
	☐ c) Check 25A1, Station Service Reserve Supply Bkr Bus 2A - TRIPPED	☐ c) Manually trip breaker.
	☐ d) Check 15J8, Normal Supply to 1J Emerg Bus - TRIPPED	☐ d) Manually trip breaker.

K/A Number: 010K1.08: Pressurizer Pressure Control: Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: PZR LCS.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.2 IR-SRO: 3.5

Proposed Question: 12

Unit 1 operating at 100% power.

- Pressurizer level selector switch in Position 1.
- 1-RC-LI-1460, PRZR Level CHNL 2 fails low.

Which ONE of the following identifies the effect of the failure on:

1) Charging flow.
2) Pressurizer heaters.

A. 1) Rises.
2) Pressurizer heaters energize.

B. 1) Rises.
2) Pressurizer Heaters de-energize.

C. 1) Lowers.
2) Pressurizer heaters energize.

D. 1) Lowers.
2) Pressurizer Heaters de-energize

.

Proposed Answer: D

Explanation: With the Pressure level selector switch in position 1; 1-RC-LI-1460 is the lower channel. When this channel fails low, pressurizer heaters are de-energized until the channel is de-selected. Letdown isolates, causing pressurizer level on 1-RC-LI-1459A to rise. The level control system will sense the rise in PRZR level and reduce CH flow.

Technical Reference: 0-AP-53.00, Rev 21.

Reference Provided to Applicant: No.

Learning Objective: ND-93.3-LP-7, Press Level Ctrl Sys, Objective C, Evaluate the consequences of a failure of a pressurizer level channel, include the operator actions necessary to mitigate the failure including the effects on the RCS if NO actions are taken.

Question Source: New

Question History: Last 2 NRC Exams: YES / NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Comments:

K/A Match Analysis: Question matches K/A since Candidate must demonstrate knowledge of the effect of a Level Control channel failure (PZR LCS) with the effect on PRZR heaters (PRZR PCS).

Distractor Analysis:

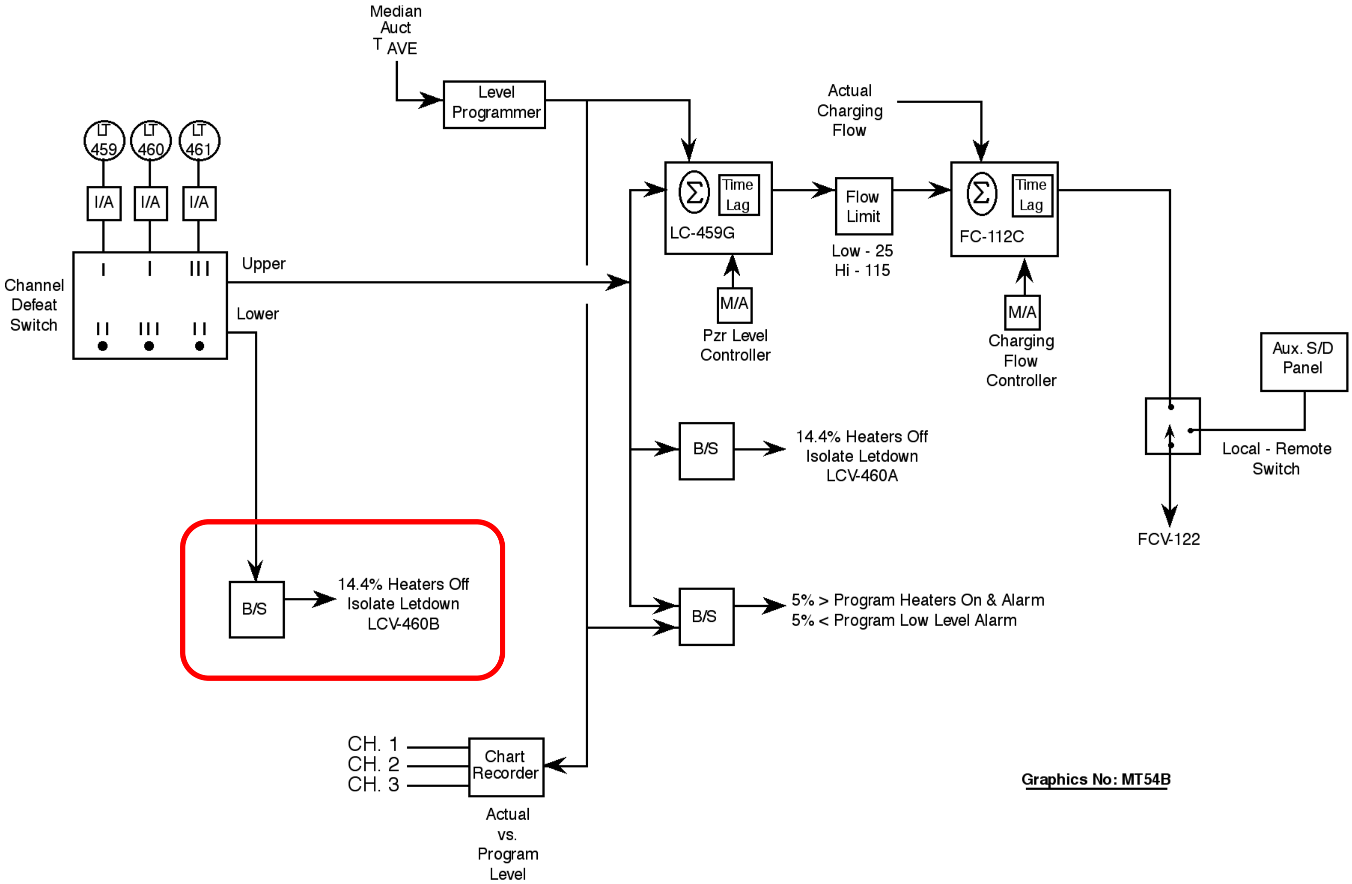
A. Incorrect. Part 1) is incorrect, charging flow will lower. Plausible if the Candidate confuses upper and lower control channel, since an upper channel failure would cause charging flow to rise. Part 2) is incorrect, pressurizer heaters would de-energize. Plausible if Candidate assumes rise in pressurizer level causes pressurizer heaters to energize in anticipation of PRZR insurge causing PRZR temperature to lower.

B. Incorrect. Part 1) is incorrect, charging flow will lower. Plausible if the Candidate confuses upper and lower control channel, since an upper channel failure would cause charging flow to rise. Part 2) is correct, lower channel failing low will de-energize the PRZR heaters.

C. Incorrect. Part1 is correct, charging flow will lower as PRZR level rises, as sensed by the upper channel. Part 2) is incorrect, pressurizer heaters would de-energize. Plausible if Candidate assumes rise in pressurizer level causes pressurizer heaters to energize in anticipation of PRZR insurge causing PRZR temperature to lower.

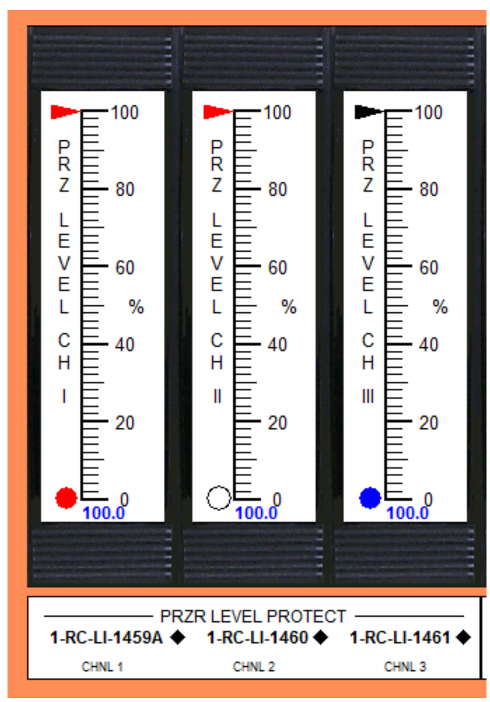
D. Correct. Both Parts 1) and 2) are correct.

From 0-AP-53.00, Attachment 3



PZR LEVEL CONTROL

With the lower controlling channel failing **LOW**, the end result is much the same as when it fails high. The lower control channel causes letdown to isolate when it receives the low level signal. Charging will go to a minimum as pzr level increases. The Rx trips at 88% level followed by the RCS going solid and pressure increasing.



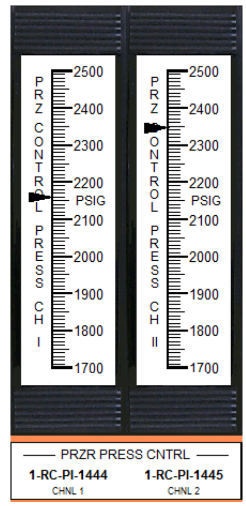
K/A Number: 010K6.01: Pressurizer Pressure Control: Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems.

Level: RO Tier #: 2 Group #: 1 IR – RO: 2.7 IR-SRO: 3.1

Proposed Question: 13

Given the following conditions:

- Unit 1 is at 50% power.
- The following annunciators have just alarmed:
  - 1C-D7, PRZR PWR RELIEF LINE HI TEMP.
  - 1C-F7, PRZR RELIEF TK HI PRESS.
  - 1C-F8, PRZR HI PRESS.
  - 1D-H4, PRZR SFTY V V PWR RELIEF V V OPEN.
- The RO reports pressure indication as shown to the right:
  - 1-RC-PI-1444 is lowering.
  - 1-RC-PI-1445 is rising.



Assuming **NO** operator action, which ONE of the following identifies:

1) Pressurizer PORV __(1)__ is open.
2) An SI due to low Pressurizer pressure __(2)__ occur.

	PZR PORV OPEN	Low Press. SI
A.	1) 1-RC-PCV-1456	2) will not
B.	1) 1-RC-PCV-1455C	2) will
C.	1) 1-RC-PCV-1456	2) will
D.	1) 1-RC-PCV-1455C	2) will not

Proposed Answer: A

Explanation: The output of 1-RC-PT-1445 provides pressure indication on 1-RC-PI-1445, and pressure signal to PORV 1-RC-PCV-1456. The other Pressurizer control channel, 1-RC-PT-1444 provide an input to 1-RC-PI-1444, and the Master pressurizer controller which, in turn controls the other PORV, 1-RC-PCV-1455C, Pressurize spray valves, and heaters. The indications given point to a failure of 1-RC-PT-1445 failing high. This in turn will send an open signal to PORV 1-RC-PCV-1456 which opens causing pressure to lower. Pressure indication on 1-RC-PI-1444 is lowering because actual pressure is lowering. The other PORV, and spray valves remains closed because they are controlled by

PT-1444. The PORV is closed once pressure is < 2000 psig due to 2/3 pressure protection interlock. There will be no Low pressure SI (1780 psig) because once the PORV is closed, pressure will rise.

Technical Reference: ND-93.3-LP-5, PZR Pressure Control, Rev. 18.

Reference Provided to Applicant: No

Learning Objective: ND-93.3-LP-5, PZR Pressure Control, Objective C; Given a set of plant conditions, determine the appropriate operator response in accordance with 1(2)-AP-31.00, Increasing or Decreasing RCS Pressure to include the following:

- Immediate Actions (if applicable)
- Entry Conditions
- Major Actions
- Step Bases

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7 / 45.7)

Comments:

K/A Analysis: Question poses a scenario whereby Pressure detection failure of 1-RC-PT-1445 fails HIGH. All conditions provide an indication of this and requires the candidate to determine the effect of this failure. Therefore this question matches the K/A.

Distractor Analysis:

A. Correct

B. Incorrect. 1) Incorrect. 1-RC-PCV-1456 will open. Plausible if candidate confuses components that are operated by 1-RC-PT-1445. 2) Incorrect. SI will not occur because PORV will close at 2000 psig which is above SI low-pressure setpoint. Plausible if candidate confuses this failure with failure of 1-RC-PT-1444 which will cause the spray valves to OPEN and pressure to drop to Rx. Trip and SI setpoint.

C. Incorrect. 1) Correct. 2) Incorrect. Rx. Trip or SI will not occur because PORV will close at 2000 psig which is above SI low-pressure setpoint. Plausible if candidate confuses this failure with failure of 1-RC-PT-1444 which will cause the spray valves to OPEN and pressure to drop to Rx. Trip and SI setpoint.

D. Incorrect. 1) Incorrect.1-RC-PCV-1456 will open. Plausible if candidate confuses components that are operated by 1-RC-PT-1445. 2) Correct.

References: ND-93.3-LP-5 PRZR Press Control, pages 10-12.

5. P-444 fails high - No operator action.

a. PORV-1455C fails open and both spray valves open.

b. RCS pressure lowers rapidly until pressure reaches 2000 psig where the PORV will close.

c. Pressure will continue to lower due to the spray valves being open.

d. The reactor will trip on low pressurizer pressure at ≈1885 psig if it had not already tripped on OTΔT.

e. Pressure will continue to lower to the SI setpoint of 1780 psig.

f. Pressure will lower until the rise in pressurizer level causes pressure to start rising.

g. When the pressurizer goes solid, pressure will rise. When pressure goes above 2000 psig PORV-1455C will open to lower pressure.

h. Pressure will stabilize around 2000 psig as the PORV opens and closes around the 2000 psig interlock.

6. P-444 fails low - No operator action.

   a. Spray valves close.

   b. With all heaters energized, pressure will rise to 2335 psig where PORV 1456 will cycle to maintain pressure at 2335 psig.

7. P-445 fail high

   a. PORV 1456 will open causing pressure to lower.

   b. When pressure reaches 2000 psig, the PORV will close.

   c. Pressure will stabilize around 2000 psig as the PORV opens and closes around the 2000 psig interlock.

K/A Number: 011EK2.02: Large Break LOCA / 3: Knowledge of the interrelations between the and the following Large Break LOCA: Pumps.

Level: RO　　Tier #: 1　　Group #: 1　　IR – RO: 2.6　　IR-SRO: 2.7

Proposed Question: 14

The operating Team is transferring the SI system from cold leg recirculation to hot leg recirculation during a Design Basis LBLOCA in accordance with 1-ES-1.4, Transfer to Hot Leg Recirculation.

Which ONE of the following completes the following statement:

LHSI flow is limited to a maximum of ____(1)____ gpm by the ____(2)____ when aligned to hot leg flow path.

A. 1) 3500
   2) operator

B. 1) 3500
   2) cavitating venturi

C. 1) 3100
   2) operator

D. 1) 3100
   2) cavitating venturi

Proposed Answer: A

Explanation: Step 1 of 1-ES-1.4 directs the Operator to limit LHSI flow to < 3500 gpm by throttling LHSI pump discharge MOV when aligning to hot leg recirc. Cavitating venturi's are used in the cold leg flowpath. 3100 gpm is a reference flow value for the LHSI pump at 120 psig RCS pressure.

Technical Reference: 1-ES-1.4, Transfer to Hot Leg Recirculation, Rev. 8.

Reference Provided to Applicant: No

Learning Objective: ND-91-LP-2, SI Sys Description, Objective C, Using a simplified one-line diagram drawn from memory showing SI system components, flowpaths, and interconnections, describe the major SI System components to include instrumentation and annunciators available in the control room.

ND-95.3-LP-11, ES-1.4, Objective B, Given a copy of ES-1.4, Transfer to Hot Leg Recirculation, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR 41.7 / 45.7)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must differentiate between flow reference value and procedurally driven flow limit during LBLOCA shift to Hot Leg Recirc flow mode. Further, Candidate must recall method of limiting LHSI flow during the transfer.

Distractor Analysis:

A. Correct. Both Parts 1) and 2) are correct.

B. Incorrect. Part 1) is correct. Part 2) is incorrect, cavitating venture used in cold leg flowpath. Plausible if Candidate fails to recall method of flow control used in the cold vs hot flowpath.

C. Incorrect. Part 1) is incorrect, 3100 gpm is a reference value provided to the Candidate for LHSI flow at 120 psi RCS pressure. Plausible if Candidate confuse reference flow value with procedurally driven limit. Part 2) is correct.

D. Incorrect. Part 1) is incorrect, 3100 gpm is a reference value provided to the Candidate for LHSI flow at 120 psi RCS pressure. Plausible if Candidate confuse reference flow value with procedurally driven limit. Part 2) is incorrect cavitating venture used in cold leg flowpath. Plausible if Candidate fails to recall method of flow control used in the cold vs hot flowpath.

Excerpt from ND-95.3-LP-11, ES-1.4, Page 6:

Ask: Why it is necessary to manually throttle the SI flow to less than 3500 gpm/pump when going to the hot legs?(K)

Answer: The hot leg lines do not have cavitating venturis installed to limit the flow (can go to ≈4000 gpm/pump to T_H) whereas, the cold leg lines have these restrictions installed, which limits the cold leg injection flow to ≈3425 gpm/pump. This action maintains adequate HHSI pump NPSH.

NUMBER	PROCEDURE TITLE	REVISION
1-ES-1.4	TRANSFER TO HOT LEG RECIRCULATION	8
		PAGE 2 of 6

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: If any hot leg injection MOV will not open, the TSC should be consulted to determine the optimal SI alignment.

1. ___ ALIGN LHSI FLOW PATH FOR HOT LEG RECIRCULATION:

ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ a) Check LHSI pump A - RUNNING	☐ a) GO TO Step 1e.
b) Close LHSI pump A discharge to cold legs:	
☐ • 1-SI-MOV-1864A	
c) Throttle open LHSI pump A discharge to hot legs to 3500 gpm:	
☐ • 1-SI-MOV-1890A	
d) Maintain LHSI pump A flow - LESS THAN 3500 GPM	

ND-91-H/T-2.8

LHSI PUMP FLOWRATES

PRESURE (PSIG)	FLOW (GPM)
200	0
160	1600
120	3100

K/A Number: 012K4.07: Reactor Protection: Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: First-out indication.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.0 IR-SRO: 3.2

Proposed Question: 15

Unit 1 was operating at 100% when RCP A tripped due to a faulty overcurrent trip relay.

- The following “First Out” Annunciators on Annunciator Panel ‘E’ alarmed, and started flashing.
  - 1E-A8, RX TRIPPED BY TURB TRIP.
  - 1E-A9, RX TRIP BKRS OPEN.
  - 1E-B10, LOSS OF COOL FLOW PWR > P8.
  - 1E-G10, STM GEN LO-LO LVL.

Which of the following completes the statement:

Of these First Out annunciators that are flashing, the alarm that would be flashing red is annunciator ______.

A. 1E-A8, RX TRIPPED BY TURB TRIP

B. 1E-A9, RX TRIP BKRS OPEN

C. 1E-B10, LOSS OF COOL FLOW PWR > P8

D. 1E-G10, STM GEN LO-LO LVL

Proposed Answer: C

Explanation: Alarm 1E-B10 is the first Reactor Trip annunciator that alarms because it is actuated when 2/3 flow detectors in 1/3 Reactor Coolant loops sense a low flow condition < 92% with power > P8 setpoint which is 35%. As this annunciator senses a reactor trip condition and it is the first Rx Trip alarm to actuate it will flash Red. The other alarms will occur following this alarm.

Technical Reference: 1E-B10, LOSS OF COOL FLOW PWR > P8. ND-93.4-LP-7, Miscellaneous Monitoring.

Reference Provided to Applicant: NO

Learning Objective: ND-93.4-LP-7, Miscellaneous Monitoring, Objective D; Describe the operation of the Main Control Room Annunciator Systems, to include the following: Hathaway System power supplies, Hathaway cabinet components and indications, alarm storage and sequence retrieval, annunciator bench board controls, annunciator alarms considered as Trip Indicators and/or Trip Demands

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7)

Comments:

K/A Analysis: K/A requires an understanding of the RPS First out feature. This question tests that understanding by posing a scenario that will cause multiple first out alarms to actuate, and then requiring the operator to determine which alarm would actuate first, which will cause that alarm to flash RED.

Distractor Analysis:

A. 1E-A8, RX TRIPPED BY TURB TRIP

B. 1E-A9, RX TRIP BKRS OPEN

C. 1E-B10, LOSS OF COOL FLOW PWR > P8

D. 1E-G10, STM GEN LO-LO LVL

A. Incorrect. This alarm will come in after the reactor has already tripped, when the turbine has tripped. This is not the first Trip Demand indicator. Plausible because this alarm is a FIRST OUT alarm requiring the operator to discriminate which alarm came in first.
B. Incorrect. This alarm will come in after the reactor has already sensed and sent trip signal to Reactor Trip breakers. This alarm is considered a First Out Trip Indicator. Plausible because this alarm is a FIRST OUT alarm requiring the operator to discriminate which alarm came in first.
C. Correct Answer.
D. Incorrect. This alarm will come in after the reactor has already tripped, due to S/G level shrinking Low. This is not the first Trip Demand indicator. Plausible because this alarm is a FIRST OUT alarm requiring the operator to discriminate which alarm came in first.

Reference: ND-93.4-LP-7, Misc. Monitoring Equipment Page, D.1.e (page 18)

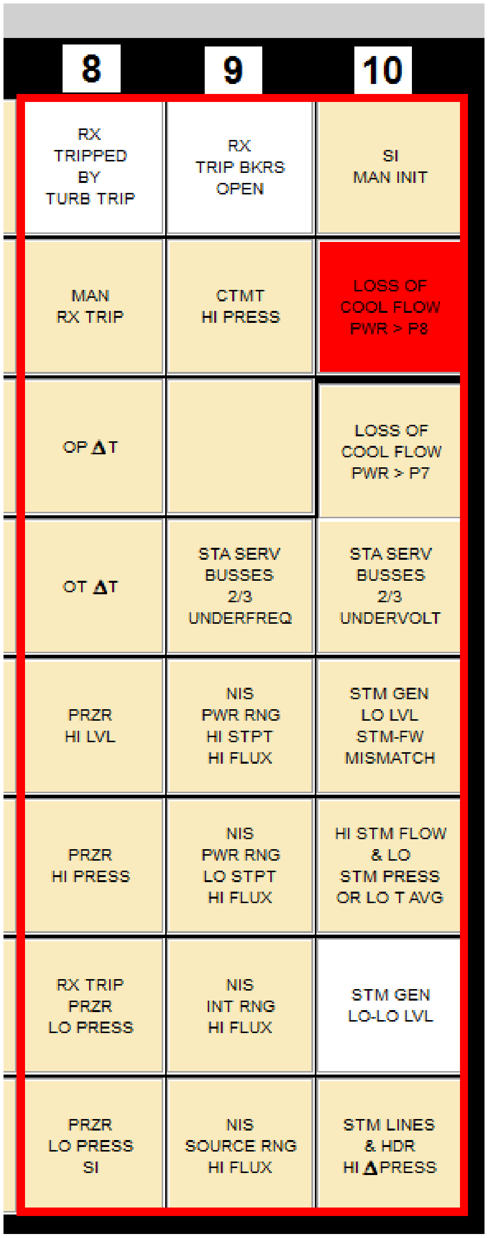
e. When an alarm condition is sensed by the SER, a signal is sent to the alarm modules to actuate the particular annunciator window associated with the parameter.

(1) Annunciator Panels A, B, C, D, G, H, J, and K are arranged in an 8X8 matrix (up to 64 alarms). Each window has two light bulbs to illuminate the window. These bulbs can be easily accessed for replacement by flipping down the annunciator window.

(2) Annunciator Panels E and F are arranged in an 8X10 matrix (up to 80 alarms). Each window has two normal bulbs. These Panels have a section outlined in red tape called the FIRST OUT section. The windows in this section have two red bulbs in addition to the two normal bulbs. The first alarm processed by the Hathaway system to actuate an alarm in the "E" or "F" first out section will cause the red bulbs to energize, all alarms processed after that will light the normal bulbs. This functions to alert the operator to the signal that caused a reactor or turbine trip.

(3) The FIRST OUT alarms are designated as either TRIP DEMAND or TRIP INDICATORS. The TRIP INDICATOR annunciators are E-A9 (RX TRIP BKRS OPEN) and F-A1 (TURB TRIPPED BY RX TRIP). These alarms indicated that the reactor has tripped since they are actuated by reactor trip breaker position.

Simulator First Out Annunciator panel (E panel with Red boundary). This shows all the alarms that have actuated 10 seconds after RCP trip. Note: Operator has acknowledged alarms. 1E-B10 was the “first” alarm that came in.



Reference 1E-B10 ARP. Setpoint is 92% flow.

NUMBER	PROCEDURE TITLE	REVISION
1E-B10	LOSS OF COOL FLOW PWR > P8	4
		PAGE 2 of 2

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- A Reactor Trip should occur if power is greater than 35% and 2/3 channels indicate low flow in Loop A.
- A Reactor Trip should occur if power is below 35% and 2/3 channels in 2/3 loops indicate low flow.
- The Reactor shall not remain critical with less than three RCPs running.
- If an RCP needs to be tripped with the Reactor critical, a Reactor trip must be performed before securing the RCP.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK THE FOLLOWING: ☐ • RX power - GREATER THAN 35% AND ☐ • RCS flow - 2/3 CHANNELS ON 1/3 LOOPS LESS THAN 92%	☐ IF Reactor Trip required, THEN GO TO Step 2. ☐ IF Reactor trip NOT required, THEN initiate a Condition Report AND GO TO Step 3.
2.___	GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION	

K/A Number: 013A4.01: Engineered Safety Features Actuation: Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate.

Level: RO Tier #: 2 Group #: 1 IR – RO: 4.5 IR-SRO: 4.8

Proposed Question: 16

Unit 2 initially operating at 100% power.

- An RCS leak develops which rapidly escalates to a LBLOCA.
- “B” CS pump, 2-CS-P-1B, fails to start.
- Containment pressure is 30 psia.

Which ONE of the following identifies:

1) The Recirc Spray pump that must be monitored for cavitation.
2) Action required if cavitation observed.

A. 1) “B” OSRS pump.
   2) Place control switch in PTL; locally open pump breaker.

B. 1) “B” OSRS pump.
   2) Place control switch in PTL.

C. 1) “B” ISRS Pump.
   2) Place control switch in PTL; locally open pump breaker.

D. 1) “B” ISRS Pump.
   2) Place control switch in PTL.

Proposed Answer: B.

Explanation: The “B” CS pump supplies water to suction of the “B” OSRS pump to reduce to potential for cavitation. Per E-0, Attachment 1, Step 9; and FR-Z.1, Sep 1 b) RNO, if cavitation is seen – the OSRS pump is stopped by placing the control switch in PTL.

Technical Reference: 2-E-0, Attachment 1, Step 9, Rev. 76; 2-FR-Z.1, Step 1 B) RNO, Rev. 19.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-48, Objective C, Given a copy of FR-Z.1, Response to Containment High Pressure, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last 2 NRC Exams: YES / NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7 / 45.5 to 45.8)

Comments:

K/A Match Analysis: Questions matches K/A. Candidate must apply knowledge of system to identify which component is affected when a CS pump trips, and action necessary to stop the affected pump.

Distractor Analysis:

A. Incorrect. Part 1) is correct. Part 2) is incorrect, OSRS pump does not require local tripping. Plausible if Candidate confuses which recirc spray pump must have CLS signal reset to stop the pump using the control switch.

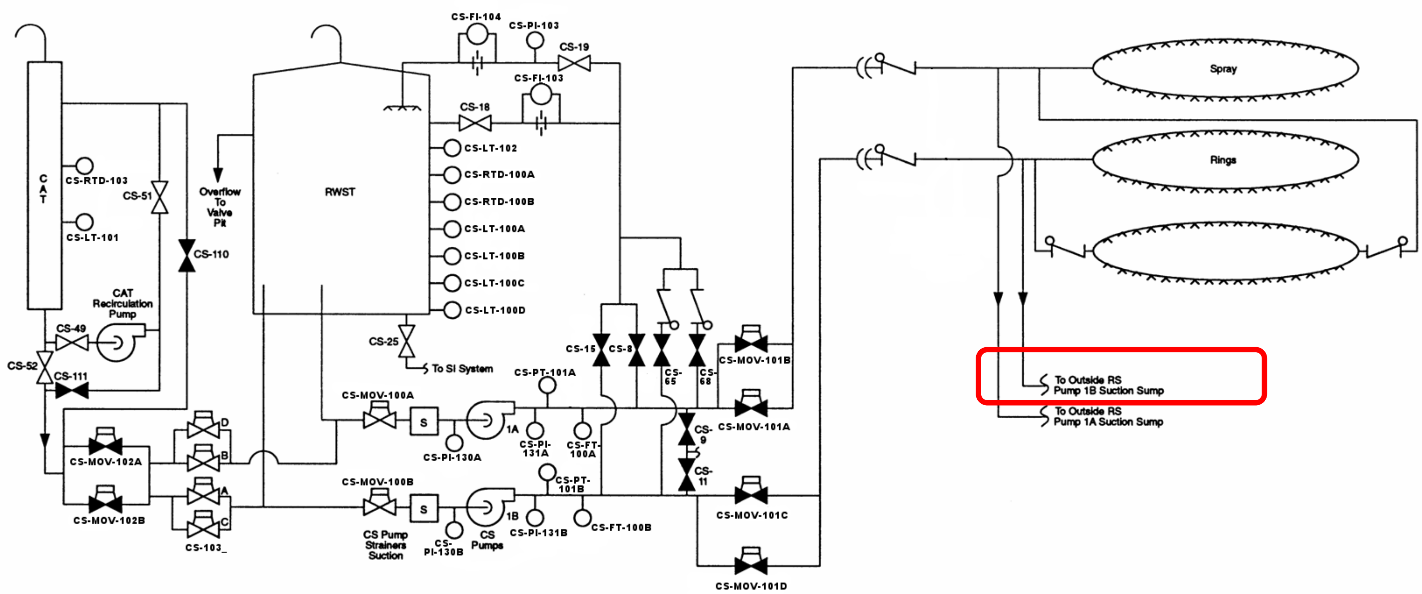
B. Correct. Both Parts 1) and 2) are correct.

C. Incorrect. Part 1) is incorrect, ISRS pumps supply their own NPSH by directing an amount of water cooled by the HX back to their own suction. Plausible if candidate confuses the method used to ensure NPSH for the RS pumps. Part 2) is incorrect, OSRS pump does not require local tripping. Plausible if Candidate confuses which recirc spray pump must have CLS signal reset to stop the pump using the control switch.

D. Incorrect. Part 1) is incorrect, ISRS pumps supply their own NPSH by directing an amount of water cooled by the HX back to their own suction. Plausible if candidate confuses the method used to ensure NPSH for the RS pumps. Part 2) is correct.

Excerpt from ND-95.3-LP-48, Objective C:

The Outside RS pump associated with any failed spray pump is also monitored for cavitation since flow from the spray pump is necessary to ensure adequate pump NPSH during the injection phase. If cavitation is apparent, indicated by fluctuations in pump flow or current, the affected OSRS pump is stopped.



NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 71	SYSTEM ALIGNMENT VERIFICATION	PAGE 5 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*9.___	CHECK IF RS REQURIED:	
	☐ a) Check RWST level - LESS THAN OR EQUAL TO 60%	a) Do the following: ☐ 1) Continue to monitor RWST level. 2) GO TO Step 10. IF RWST level lowers to less than or equal to 60%, THEN perform Step 9b through Step 9d.
	☐ b) Check ISRS pumps - RUNNING	☐ b) Manually Start Pumps.
	☐ c) Check OSRS pumps - RUNNING (Time Delayed)	☐ c) Manually Start Pumps.
	☐ d) Check OSRS pumps - NOT CAVITATING	☐ d) Put affected OSRS pump in PTL.

NUMBER	PROCEDURE TITLE	REVISION 19
2-FR-Z.1	RESPONSE TO CONTAINMENT HIGH PRESSURE	PAGE 2 of 10

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If 2-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, containment spray systems should be operated as directed by 2-ECA-1.1, instead of Step 1 below.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK IF CS REQUIRED:	
	☐ a) Check CTMT pressure - HAS INCREASED TO GREATER THAN 23 PSIA	☐ a) RETURN TO procedure and step in effect.
	☐ b) Verify CS pumps - RUNNING	☐ b) IF RWST level greater than 3%, THEN start CS pumps. IF any CS pump can NOT be started, THEN monitor OSRS pumps for cavitation. ☐ IF cavitation is indicated, THEN put affected OSRS pump in PTL.

K/A Number: 013K6.01: Engineered Safety Features Actuation: Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors.

Level: RO Tier #: 2 Group #: 1 IR – RO: 2.7 IR-SRO: 3.1

Proposed Question: 17

Given the following:

- Unit 1 is operating at 100% power.
- 1-RC-PT-1455, PZR Pressure Protection Channel I has failed to 1800 psig.
- The channel HAS NOT been placed in TRIP at this time.

1) If another PZR Pressure Protection channel fails low, a Reactor Trip _____ occur.
2) Which ΔT protection circuit (OPΔT or OTΔT) will this failure affect?

A. 1) will
2) OTΔT

B. 1) will not
2) OPΔT

C. 1) will not
2) OTΔT

D. 1) will
2) OPΔT

Proposed Answer: A

Explanation: With the failure of 1 channel of PZR Pressure protection to 1800 psig and the Reactor Trip setpoint of 1885 psig, the failure of an additional channel low will cause a Low Pressure Reactor Trip. The OTΔT protection circuit is the only one affected by a change in PZR pressure.

Technical Reference: 1E-G8, Rx Trip PRZR Lo Press, Revision 3.

Reference Provided to Applicant: No

Learning Objective: ND-91-LP-3, SI System Operations, Objective A; “List the five (5) Safety Injection actuation signals, including setpoints, coincidence, and purposes.”

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 45.5 to 45.8)

Comments:

K/A Analysis: Question poses a failure of one or more of the PZR Pressure protection channels. This requires candidate to determine the effect on ESFAS function: Reactor Trip.

Distractor Analysis:

A. Correct.
B. Incorrect. 1) Incorrect as Low Pressure Rx Trip setpoint is 1885 psig. 2) Incorrect as OPΔT is unaffected by changes in PZR pressure. Plausible if candidate mistakes OPΔT for OTΔT.
C. Incorrect. 1) Incorrect as Low Pressure Rx Trip setpoint is 1885 psig. 2) Correct.
D. Incorrect. 1) Correct. 2) Incorrect as OPΔT is unaffected by changes in PZR pressure. Plausible if candidate mistakes OPΔT for OTΔT.

Reference: ARP 1E-G8. Shows Reactor Trip logic.

NUMBER	PROCEDURE TITLE	REVISION
1E-G8	RX TRIP PRZR LO PRESS	3
		PAGE 1 of 2

REFERENCES 1E-19

1) UFSAR - Section 6.0, 7.0, 14.0
2) 0-DRP-004, Precautions, Limitations and Setpoints
3) Westinghouse Interconnecting Wiring Diagram 5965D35, 5965D36
4) DCP 07-047, Implement Requirements of TSCR 318

PROBABLE CAUSE

1) 2/3 Pressurizer Pressure Protection Channels reach the setpoint of 1885 psig as detected by LC-1- 455C, 456C, 457C and caused by one or more of the following:
   a. Loss of Coolant Accident.
   b. Steam Generator Tube Rupture.
   c. Steam Line Break.
   d. Spray Valve stuck open.
   e. Failed Annunciator.

Reference: COLR – S1C26: Shows OTΔT and OPΔT specifications.

3.2 Overtemperature ΔT (TS 2.3.A.2.d)

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2\left(\frac{1+t_1 s}{1+t_2 s}\right)(T - T') + K_3\boxed{(P - P')} - f(\Delta I)\right]$$

Where:

ΔT is measured RCS ΔT, °F.

ΔT_0 is the indicated ΔT at RATED POWER, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature (T_{avg}), °F.

T' is the nominal T_{avg} at RATED POWER, ≤ **573.0°F**.

P is the measured pressurizer pressure, psig.

P' is the nominal RCS operating pressure ≥ **2235 psig**.

3.3 <u>Overpower ΔT</u> (TS 2.3.A.2.e)

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5\left(\frac{t_3 s}{1 + t_3 s}\right)T - K_6(T - T') - f(\Delta I)\right]$$

Where:

ΔT is measured RCS ΔT, °F.

ΔT_0 is the indicated ΔT at RATED POWER, °F.

s is the Laplace transform operator, $sec^{-1.}$

T is the measured RCS average temperature (T_{avg}), °F.

T' is the nominal T_{avg} at RATED POWER, ≤ **573.0°F**.

K/A Number: 015A1.02: Nuclear Instrumentation: Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the NIS controls including: SUR.

Level: RO Tier #: 2 Group #: 2 IR – RO: 3.5 IR-SRO: 3.6

Proposed Question: 18

The following conditions exist:

- A Unit 1 Reactor Startup is in progress.
- 1-OP-RX-006, Withdrawal of Control Banks to Critical Conditions, Attachment 2, ICRR plot (1/M plot) data is as follows:
    - At 98 steps "C" Control bank – Source Range counts stabilize at 300 cps.
    - At 15 steps "D" Control bank – Source Range counts stabilize at 375 cps.
    - At 60 steps "D" Control Bank – Source Range counts stabilize at 545 cps.
    - At 90 steps "D" Control Bank – Source Range counts stabilize at 1000 cps.
    - A two minute wait to allow count rate and start up rate to stabilize is taken prior to data taking.

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

Based on the change in Source Range counts, the 1/M projected critical rod height is currently ________ on "D" Control Bank.

REFERENCE PROVIDED

A. 124 steps

B. 103 steps

C. 142 steps

D. 154 steps

Proposed Answer: A.

Explanation: 124 steps on "D" bank based on

Technical Reference: 1-OP-RX-006, Attachment 2, Inverse Count Rate Ratio Plot (Pull to Criticality). Rev. 36.

Reference Provided to Applicant: YES

Learning Objective: ND-86.2-LP-7B, Operational Aspects of Reactor Startup, Objective C., Calculate a 1/M plot (inverse multiplication factor plot).

Question Source: Bank

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 . 45.5)

Comments:

K/A match analysis: Question matches K/A. Candidate predicts criticality using a 1/M plot during a pull and wait reactor startup. Question provided with 1-OP-RX-006. Attachment 2, Page 1 and 2 only.

Distractor Analysis:

A. Correct.

B. Incorrect. Prediction based on dividing 545 cps by 1000 cps (last two SR counts), 1/M = 0.18, and plotting to "x" axis.

C. Incorrect. Prediction based on drawing line from 1/M =1 to 1/M = .3 and continuing to "x" axis.

D. Incorrect. Prediction based on using 2nd and 3rd calculated points and drawing to "x" axis.

DOMINION
Surry Power Station

1-OP-RX-006
Revision 36
Page 23 of 31

(Page 1 of 2)

Attachment 2

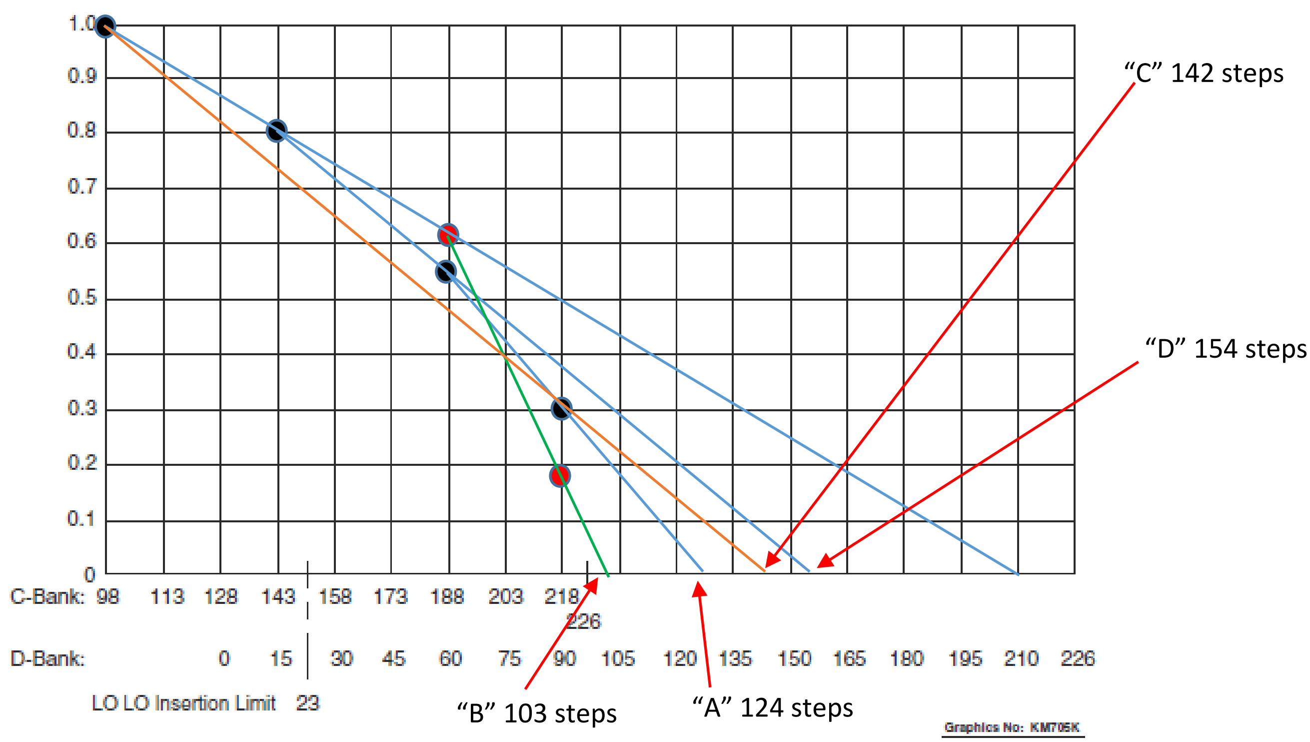
INVERSE COUNT RATE RATIO PLOT (PULL TO CRITICALITY)

Steps		Counts	1/M (correct)	1/M (incorrect)
C	D			
98	0	300	1.0	1.0
143	15	375	0.8	0.8
188	60	545	0.55	0.61
218	90	1000	0.3	0.18

(Page 2 of 2)

Attachment 2

INVERSE COUNT RATE RATIO PLOT (PULL TO CRITICALITY)



INVERSE COUNT RATE RATIO PLOT

K/A Number: 015AK2.10: RCP Malfunctions/4: Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP indicators and controls.

Level: RO Tier #: 1 Group #: 1 IR – RO: 2.8 IR-SRO: 2.8

Proposed Question: 19

Given the following conditions:

- Unit 1 is being started up following refueling.
- Vacuum-assist method was NOT used for RCS loop fill-and-vent.
- The first RCP start is being performed.

After starting "A" RCP, #1 seal delta-P is noted to be rapidly lowering.

In accordance with 1-OP-RC-001, Starting and Running any Reactor Coolant Pump, the RCP must be stopped when #1 seal delta-P reaches ___1)___. The basis for this action is ___2)___.

A. 1) 300 psid IF it is imminent that delta-P will lower to 240 psid
   2) to ensure adequate seal flow during pump coastdown

B. 1) 50 psid IF it is imminent that delta-P will lower to 25 psid
   2) to ensure adequate seal flow during pump coastdown

C. 1) 300 psid IF it is imminent that delta-P will lower to 240 psid
   2) to minimize the time required to repressurize the RCS

D. 1) 50 psid IF it is imminent that delta-P will lower to 25 psid
   2) to minimize the time required to repressurize the RCS

Proposed Answer: B

Explanation: 1-OP-RC-001, Starting and Running any Reactor Coolant Pump, Precaution and Limitation 4.10 states; "After start of an RCP, if any Seal Stage delta-P is rapidly decreasing and it is imminent that delta-P will decrease to less than 25 psid, stopping the RCP when delta-P reaches 50 psid will provide sufficient seal flow during pump coastdown.

Technical Reference: 1-OP-RC-001, Starting and Running any Reactor Coolant Pump; Rev. 26.

Reference Provided to Applicant: No

Learning Objective: ND-88.1-LP-6, Rx. Coolant Pumps, Objective F; List the conditions and interlocks necessary to start and operate a Reactor Coolant Pump and signals that will trip an RCP.

Question Source: Bank (RCP0064)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR 41.7 / 45.7)

Comments:

K/A Analysis: Question requires knowledge of key indications that must be utilized to ensure successful start of a Reactor Coolant Pump. In this case the candidate must have knowledge of the required delta-P prior to securing a Reactor Coolant Pump. This question meets the K/A.

Distractor Analysis:

A. Incorrect. 1) Incorrect. 1-OP-RC-001 states that an RCP should be stopped if delta-P decreases to 50 psig and it is imminent that delta-P will reach 25 psid. 300 psid is plausible because 240 psig used as minimum pressure to provide required NPSH. 2) Correct.
B. Correct.
C. Incorrect. 1) Incorrect. 1-OP-RC-001 states that an RCP should be stopped if delta-P decreases to 50 psig and it is imminent that delta-P will reach 25 psid. 300 psid is plausible because 240 psig used as minimum pressure to provide required NPSH. 2) Incorrect as the stated reason is to ensure sufficient seal flow during pump coastdown. Plausible because inadequate seal delta-P is indicative of inadequate venting, and this reason provides logical explanation for recovery to normal pressure.
D. Incorrect. 1) Correct. 2) Incorrect as the stated reason is to ensure sufficient seal flow during pump coastdown. Plausible because inadequate seal delta-P is indicative of inadequate venting, and this reason provides logical explanation for recovery to normal pressure.

Bank question RCP0064:

Given the following conditions:

- Unit 1 is being started up following refueling.
- Vacuum-assist method was NOT used for RCS loop fill-and-vent.
- The first RCP start is being performed.

After starting "A" RCP, #1 seal delta-P is noted to be rapidly decreasing.

In accordance with 1-OP-RC-001, Starting and Running any Reactor Coolant Pump, the RCP must be stopped when #1 seal delta-P reaches ___(1)___. The basis for this action is ___(2)___.

A.	(1) (2)	240 psid IF it is imminent that delta-P will decrease to 200 psid; to ensure adequate seal flow during pump coastdown
B.	(1) (2)	50 psid IF it is imminent that delta-P will decrease to 25 psid; to ensure adequate seal flow during pump coastdown
C.	(1) (2)	240 psid IF it is imminent that delta-P will decrease to 200 psid; to minimize the time required to repressurize the RCS
D.	(1) (2)	50 psid IF it is imminent that delta-P will decrease to 25 psid; to minimize the time required to repressurize the RCS

Answer:	B

Question 31 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	154785
User-Defined ID:	RCP0064
Cross Reference Number:	
Topic:	RCP0064
Num Field 1:	
Num Field 2:	
Text Field:	1.00
Comments:	ND-88.1-LP-6F [11466], [7881]

References: 1-OP-RC-001 P&L 4.10, 4.11:

______ • When three starts or attempted starts have been made within a two-hour period, then a FOURTH start should NOT be made until the motor has been allowed to cool by standing idle for at least one hour.

______ 4.7 If RCP vibration sensors are overranged by inputs in excess of full scale, the indicators may fail low. (Reference 2.3.8)

______ 4.8 A reactor coolant pump should not be operated in a loop with SG water level below the top of the U-tubes (70% WR).

______ 4.9 Charging Pumps should not be swapped when RCS pressure is less than 400 psig with RCP(s) running unless absolutely necessary. Swapping Charging Pumps at low RCS pressure can challenge the RCP seal package. (OE28830, Catawba)

______ 4.10 After start of an RCP, if any Seal Stage ΔP is rapidly decreasing and it is imminent that ΔP will decrease to less than 25 psid, stopping the RCP when ΔP reaches 50 psid will provide sufficient seal flow during pump coastdown.

______ 4.11 The minimum RCS pressure to provide required NPSH for running an RCP is 240 psig.

References: 1-OP-RC-001, CAUTION prior to step 5.2.31:

CAUTION

- If an RCP fails to start or trips due to speed sensing circuit, Control Operations Department must be contacted before any restart is attempted.
- After start of an RCP, if any Seal Stage ΔP is rapidly decreasing and it is imminent that ΔP will decrease to less than 25 psid, RCP must be stopped when ΔP reaches 50 psid. This will provide sufficient seal flow during pump coastdown.
- The minimum RCS pressure to provide required NPSH for running an RCP is 240 psig.

NOTE: Annunciator 1K-G8, BUS 1J DEGRADED VOLTAGE, may actuate when the RCP is started.

_____ 5.2.31 Start 1-RC-P-1A. IF 1-RC-P-1A fails to start or trips after start, THEN notify System Engineer AND contact Control Ops before any restart is attempted.

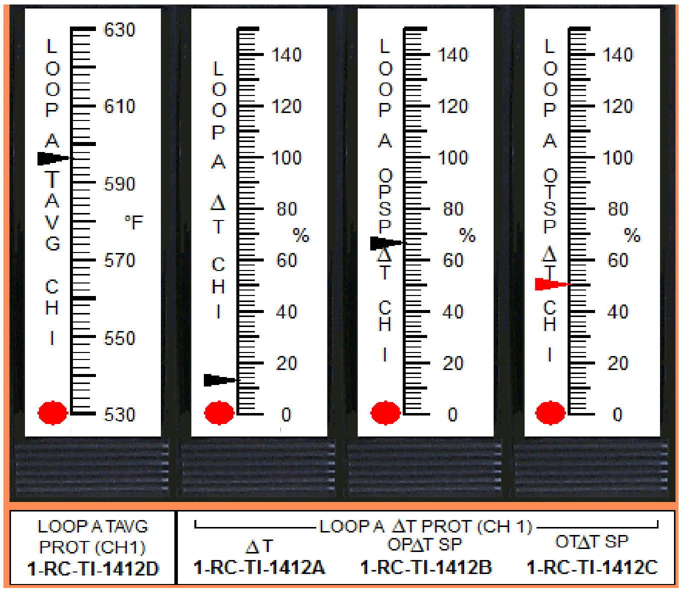
K/A Number: 016A3.02: Non-nuclear Instrumentation: Ability to monitor automatic operation of the NNIS, including: Relationship between meter readings and actual parameter value.

Level: RO Tier #: 2 Group #: 2 IR – RO: 2.9 IR-SRO: 2.9

Proposed Question: 20

The Unit is operating at 100% power. All loop Tave indications are 573ºF.

- A loop RTD has failed giving the indications below.



Which ONE of the following identifies:

1) Which NR RTD failing high will cause the indications shown above?
2) Will Median Tave change with this failure?

A. 1) T_H.
   2) Yes.

B. 1) T_H.
   2) No.

C. 1) T_C.
   2) Yes.

D. 1) T_C.
   2) No.

Proposed Answer: D.

Explanation: Tave = T_H + T_C/2. T_H is developed by averaging 3 T_H RTDs spaced 120° apart in the T_H leg. T_c uses a single RTD to develop the T_C input to calculating T_{ave}. A single T_H RTD failing high causes Loop T_{ave} to change from 573°F to 580.6°F, whereas a T_C RTD failing high will change T_{ave} from 573°F to 616.6°F; T_c would have a greater effect on calculated Loop T_{ave}. Median T_{ave} selects the Median or Middle indication of the three loops; on a change in a single loop input of this magnitude would cause the circuit to "kick-out" the deviating channel input and default to a "auctioneered high" mode. In this case, since all loop T_{ave}s were 573°F initially, Median T_{ave} will remain the same as before the failure.

Technical Reference: 0-AP-53.00, Rev. 21; ND-93.3-LP-2 – Delta T / T_{avg} Instrumentation Systems, Rev. 15.

Reference Provided to Applicant: No.

Learning Objective: ND-93.3-LP-2, Delta T_T_{avg} Instruments, Objective B, Describe the arrangement of the hot and cold leg RTDs. Objective E, Outlining all outputs and functions with a simplified one-line diagram, explain the operation of the T_{avg} Protection circuit.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7 / 45.5)

Comments:

K/A Match Analysis: Question matches K/.A. Candidate must assess effect of RTD failures on Loop T_{ave} indication and determine effect on Median Loop T_{ave} calculation/determination by circuitry.

Distractor Analysis:

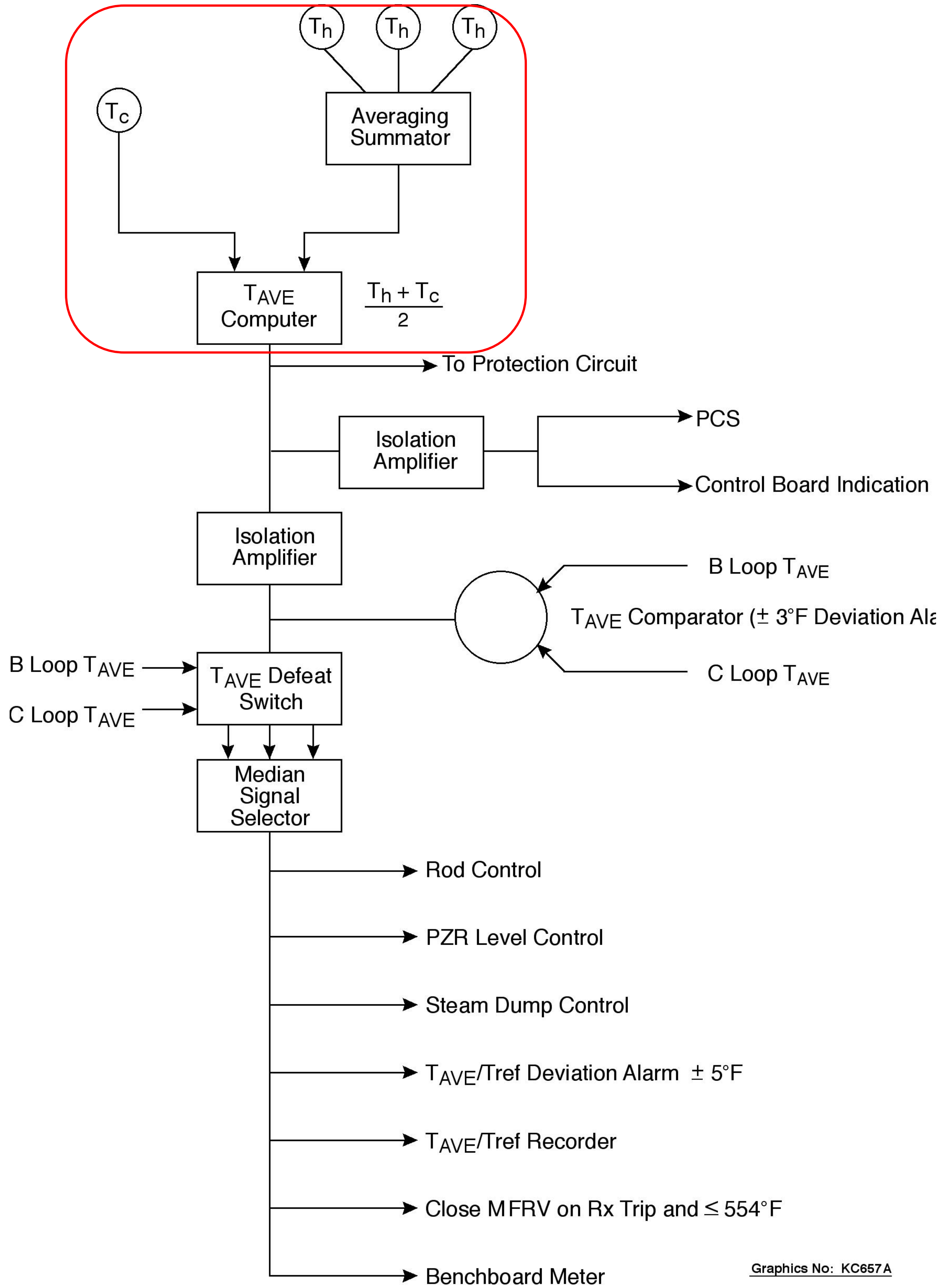
A. Incorrect. Part 1) is incorrect, T_c Failure will cause Loop T_{ave} to fail in the direction of the RTD failure and Loop ΔT to fail in the opposite direction. For a failure of a T_h instrument, both Loop Tave and Loop ΔT fail in the same direction. Part 2) is incorrect, calculated T_{ave} for the Loop with the failed RTD would be "kicked out" of the Median T_{ave} circuitry and default to an auctioneered high selection for Median T_{ave} indication. Plausible if Candidate confuses "Median" with "Average" and assumes affected Loop T_{ave} would be averaged with other Loop T_{ave}s, thus median T_{ave} indication would change to a higher value.

B. Incorrect. Part 1) is incorrect, T_c Failure will cause Loop T_{ave} to fail in the direction of the RTD failure and Loop ΔT to fail in the opposite direction. For a failure of a T_h instrument, both Loop

Tave and Loop ΔT fail in the same direction. Part 2) is correct, Median Tave indication would be unaffected.

C. Incorrect. Part 1) is correct. Part 2) is incorrect, calculated T_{ave} for the Loop with the failed RTD would be "kicked out" of the Median T_{ave} circuitry and default to an auctioneered high selection for Median T_{ave} indication. Plausible if Candidate confuses "Median" with "Average" and assumes affected Loop T_{ave} would be averaged with other Loop T_{ave}s, thus median T_{ave} indication would change to a higher value.

D. Correct. Both Parts 1) and 2) are correct.



T_{AVE} CONTROL CIRCUIT

Excerpt from ND-93.3-LP-2

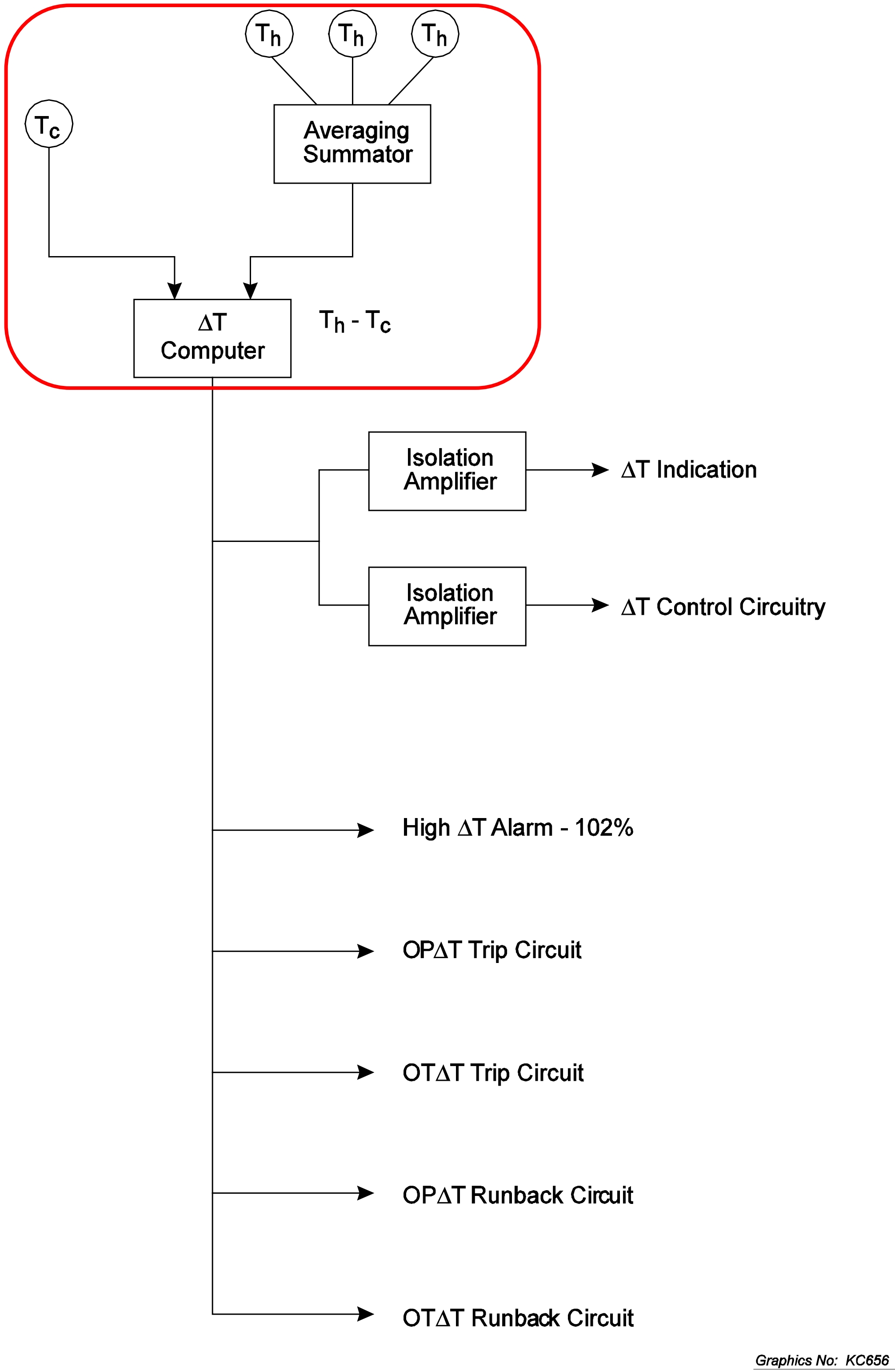
4. Median Tave Signal Selector

   a. The output from each loop's Tave computer inputs to the Median Signal Selector (MSS) after passing through the Tave Defeat Switch. The defeat switch is used only during Instrument Tech channel calibration and troubleshooting procedures.

   b. The MSS will select the median signal of the three loops and output that signal to the remainder of the circuitry.

   c. The use of a MSS ensures that the control systems are provided with a valid temperature signal when a single channel fails.

   d. If one of the input signals deviate excessively from the median, the circuit eliminates the deviated signal and becomes a high selector of the remaining valid inputs.



DELTA T PROTECTION CIRCUIT

K/A Number: 022A4.01: Containment Cooling: Ability to manually operate and/or monitor in the control room: CCS fans.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.6 IR-SRO: 3.6

Proposed Question: 21

Given the following:

- Unit 1 was operating at 100%.
- All three Containment Air Recirculation Fans are in operation.
- Auto Reactor Trip and SI initiation occurs due to a small break LOCA.
- Containment Pressure is 20 psia and slowly rising.
- Team is currently performing E-1, Loss of Reactor or Secondary Coolant.

Which ONE of the following describes:

1) Containment Air Recirculation Fan(s) ________ is (are) running.
2) The Unit 1 Containment Air Recirculation Fans are operated at the ______ panel.

A. 1) “C”
   2) Ventilation

B. 1) “A”, “B”, and “C”
   2) Ventilation Status Panel (VSP)

C. 1) “C”
   2) Ventilation Status Panel (VSP)

D. 1) “A”, “B”, and “C”
   2) Ventilation

Proposed Answer: D

Explanation: 1) When CTMT pressure exceeds 23.0 psia, Hi-Hi CLS actuates and secures the Emergency bus powered fans (CARF A&B). Therefore all three CARF fans will remain running. 2) The Containment Air Recirculation Fans are operated at the Ventilation Panel.

Technical Reference: 1-E-0, Reactor Trip or SI, Attachment 1 and 4; Rev. 71.

Reference Provided to Applicant: No

Learning Objective: ND-88.4-LP-6, CTMT Ventilation, Objective A; Describe the operation of the Containment Air recirculation System, including flowpaths, capacities, power supplies and trip signals.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 45.5 to 45.8)

Comments:

K/A Analysis: Question tests the candidate's knowledge of manual operation by asking the location, and also tests the understanding of the operation of the fans under accident conditions. This question meets the K/A.

Distractor Analysis:

A. Incorrect. 1) Incorrect because all fans will be running at this time. Plausible if operator confuses actions that occur at HI CLS (which has occurred) with actions that occur at HI-HI CLS (which has not occurred). This would be correct if Containment pressure was at the Hi-Hi CLS setpoint of 23 psia. 2) Correct.
B. Incorrect. 1) Correct. 2) Incorrect because the CARF fans are operated at the Ventilation panel. Plausible because there are other Ventilation fans operated at the VSP such as 1-VS-F-58A, and 1-VS-F-58B which are the Aux Building Cat I Filtration Fans.
C. Incorrect. 1) Incorrect because all fans will be running at this time. Plausible if operator confuses actions that occur at HI CLS (which has occurred) with actions that occur at HI-HI CLS (which has not occurred). This would be correct if Containment pressure was at the Hi-Hi CLS setpoint of 23 psia. 2) Incorrect because the CARF fans are operated at the Ventilation panel. Plausible because there are other Ventilation fans operated at the VSP such as 1-VS-F-58A, and 1-VS-F-58B which are the Aux Building Cat I Filtration Fans.
D. Correct.

References: ND-88.4-LP-6, CTMT Ventilation, page 4. LP that shows auto trip of CARF A and B at Hi-Hi CLS.

d. Cooling water for the cooling coils is supplied by component cooling water or chilled component cooling water. The switchover from CC to chilled CC is accomplished by placing the switch on the unit's ventilation panel from "norm" to "chill." Chilled CC is normally maintained to the coolers.

e. Power Supplies: 1-VS-F-1A - 480V 1H
1-VS-F-1B - 480V 1J
1-VS-F-1C - 480V 1C1

f. The air recirculation fans will operate in containment pressure up to 8.3 psig (23 psia). At this point the 1A and 1B fans are tripped by a hi-hi CLS signal. This is to protect the emergency buses from an overload condition caused by the starting and running of the pumps in the SI and CLS systems. The 1C fan will remain running, since it is not powered from an emergency bus.

1-E-0, Reactor Trip or SI Attachment 1. This directs performing Att 4 if Ctmt press > 23 psia (which it is not)

NUMBER 1-E-0	ATTACHMENT TITLE SYSTEM ALIGNMENT VERIFICATION	ATTACHMENT 1
REVISION 71		PAGE 4 of 7

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*8. ___	CHECK IF CS REQUIRED:	
	☐ a) CTMT pressure - HAS EXCEEDED 23 PSIA	a) Do the following: 1) IF CTMT pressure has exceeded 17.7 psia, THEN check or align the following valves: ☐ • 1-RM-TV-100A - CLOSED ☐ • 1-RM-TV-100B - CLOSED ☐ • 1-RM-TV-100C - CLOSED ☐ • 1-SV-TV-102 - CLOSED ☐ • 1-IA-TV-101A - CLOSED ☐ • 1-IA-TV-101B - CLOSED ☐ • 1-IA-AOV-103 - OPEN ☐ 2) GO TO Step 10.
	☐ b) Manually initiate HI HI CLS	
	☐ c) Trip all RCPs	
	☐ d) Check CS pumps - RUNNING	☐ d) Manually start pump(s).
	☐ e) Initiate Attachment 4	

1-E-0, Reactor Trip or SI Attachment 4. Used as distractor for part 1.

NUMBER 1-E-0	ATTACHMENT TITLE CLS COMPONENT VERIFICATION	ATTACHMENT 4
REVISION 71		PAGE 1 of 2

LOCATION: Vertical Board

VALVE POSITION: CLOSED
LIGHTS: GREEN

__ 1-RM-TV-100C

__ 1-RM-TV-100B

__ 1-RM-TV-100A

__ 1-CC-TV-105A

__ 1-CC-TV-105B __ 1-CC-TV-140A __ 1-CC-TV-110A

__ 1-CC-TV-105C __ 1-CC-TV-140B __ 1-CC-TV-110B __ 1-CC-TV-110C __ 1-IA-TV-100

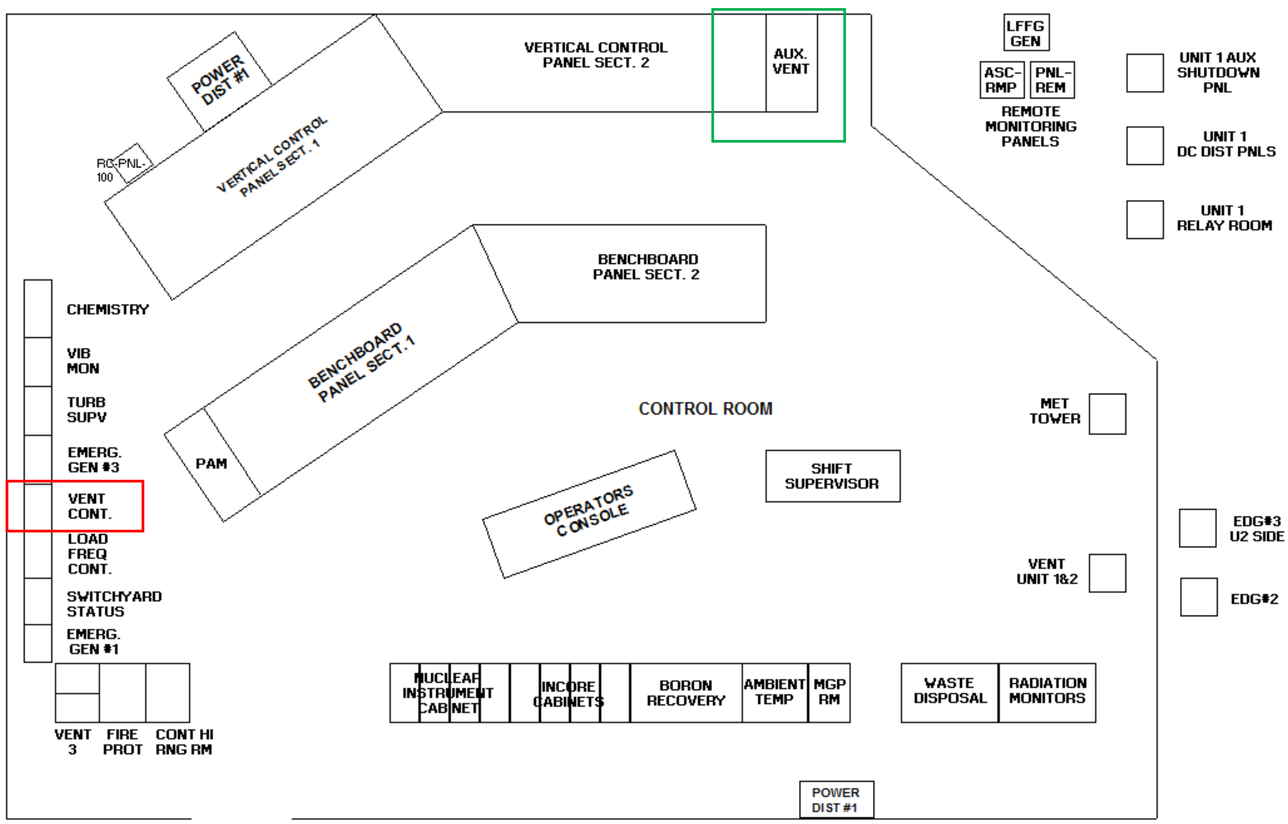
__ 1-SV-TV-102 __ 1-IA-TV-101A __ 1-IA-TV-101B

LOCATION: Unit 1 Vent Panel

RECIRC FAN STATUS: OFF
LIGHTS: AMBER

__ 1-VS-F-1A __ 1-VS-F-1B

Reference: Control Room layout showing location of Vent panel (red) and VSP (distractor) (green).



K/A Number: 022AG2.2.37: Loss of Rx Coolant Makeup / 2: Ability to determine operability and/or availability of safety related equipment.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.6 IR-SRO: 4.6

Proposed Question: 22

Unit 1 is at CSD.

You are the Reactor Operator performing the CRO Shift Relief Checklist.

U1 Control Room Operator Shift Relief Checklist
CSD/RSD

NDT Protection Status:

_______ Verify compliance with T.S. 3.1.g

Which of the following Charging Pump alignments satisfies compliance with Tech Spec 3.1.g, RCS Overpressure Mitigation?

	1-CH-P-1A	**1-CH-P-1B**	**1-CH-P-1C Norm**	**1-CH-P-1C Alt**
A.	Pull-to-Lock	Pull-to-Lock	Running	Pull-to-Lock
B.	Auto	Pull-to-Lock	Pull-to-Lock	Running
C.	Running	Pull-to-Lock	Auto	Pull-to-Lock
D.	Auto	Auto	Running	Pull-to-Lock

Proposed Answer: A.

Explanation: Prior to lowering RCS temperature below 350°F, a maximum of 1 charging pump shall be capable of injecting into the RCS IAW Tech Spec 3.1.G.1.b.(1).

Technical Reference: Tech Spec 3.1.G.1.b.(1); 1-OSP-ZZ-004, Rev 48.

Reference Provided to Applicant: No

Learning Objective: ND-88.3-LP-2, Charging and Letdown, Objective D, Describe the Technical Specifications associated with the CVCS System, including for SRO candidates, the basis behind these specifications.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 43.5 / 45.12)

Comments:
K/A Match Analysis: Question matches K/A. Candidate must verify compliance with TS-3.1.G regarding charging pump operability/availability while at CSD.

Distractor Analysis:

A. Correct.

B. Incorrect. Plausible since this would be a normal at-power alignment to tag out 1-CH-P-1B.

C. Incorrect, two charging pumps are capable of injecting to RCS upon auto start of the standby pump. Plausible if Candidate confuses “not running” with “not capable of injecting”.

D. Incorrect, two charging pumps are capable of injecting to RCS upon auto start of the standby pump. Plausible if Candidate confuses “not running” with “not capable of injecting”. This is a normal at-power alignment for the charging system

TS 3.1-23
09-01-95

G. Reactor Coolant System Overpressure Mitigation

Specification

1. The Reactor Coolant System (RCS) overpressure mitigating system shall be OPERABLE as described below:

   a. Whenever the RCS average temperature is greater than 350°F, a bubble shall exist in the pressurizer with the necessary sprays and heaters OPERABLE.

   b. Prior to decreasing RCS average temperature below 350°F, verify a maximum of one charging pump is capable of injecting into the RCS and that each accumulator is isolated. Thereafter, once per 12 hours:

      (1) Verify that a maximum of one charging pump is capable of injecting into the RCS.

      (2) Verify that each accumulator is isolated, if isolation is required.

   c. Whenever the RCS average temperature is less than or equal to 350°F and the reactor vessel head is bolted:

      (1) A maximum of one charging pump shall be OPERABLE and capable of injecting into the RCS. Two charging pumps may be in operation momentarily during transfer of operation from one charging pump to another.

Attachment 1

COLD SHUTDOWN CONDITIONS

EQUIPMENT	MIN REQ	D	N	TECH SPECS	REMARKS
Non-isolated RC Loops and/or non-isolated RHR Heat Removal Loops	1 operating and 1 operable			3.1.A.1.d	OU-AA-200 TS allows testing on the non-operating RHR loop
Secondary Cooling	1 SG capable of being fed and its RCS loop intact and full				Desired if RCS temp greater than 140°F.
Overpressure Mitigating System: • PORVs	2 operable			3.1.G.1.c.(4)	Including automatic actuation circuits and backup air supply
• PORV Block Valves (1-RC-MOV-1535 & 1-RC-MOV-1536)	2 Open			3.1.G.1.c.(4)	When PORVs are providing the overpressure protection and once per day thereafter.
RCS vented through one open PORV or equivalent size opening (Pzr SV)	1 Open			3.1.G.c.(5)	N/A if credit taken for PORVs in AUTO with OPMS setpoints.
• Pressurizer Conditions	Steam Bubble with a maximum narrow range level of 33%			3.1.G.1.c.(3)	Applicable only during the initial 72 hours following Reactor Shutdown. N/A if credit taken for PORVs above
• Charging Pumps	2 in PTL			3.1.G.1.b.(1)	Prior to decreasing RCS avg temp less than 350°F and once per 12 hrs; thereafter. Log time in Narrative Log
• SI Accumulators A, B and C	Discharge			3.1.G.1.b.(2)	Prior to decreasing RCS avg

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

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.5 IR-SRO: 3.5

Proposed Question: 23

Per 1-ECA-1.1 the Containment Spray pumps will be secured as soon as _________.

A. Containment Sump Level exceeds 4.0 feet

B. Containment Sump Level exceeds 6.0 feet

C. RWST level falls below 3.0%

D. RWST level falls below 6.0%

Proposed Answer: C

Explanation: Per 1-ECA-1.1 the Containment Spray pumps are secured once RWST lowers to 3%.

Technical Reference: 1-ECA-1.1, Loss of Emergency Coolant Recirculation; Rev. 39.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-20, ECA-1.1; Objective B, Given a copy of ECA-1.1, Loss of Emergency Coolant Recirculation, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 45.5)

Comments:

K/A Analysis: Question requires knowledge of required parameters, including Containment Sump Level for operation of Containment Spray and Recirc Spray equipment.

Distractor Analysis:

A. Incorrect. ECA-1.1 requires Containment Spray remain running unitl RWST < 3.0%. Plausible since this is value required for Recirc Spray equipment operation.

B. Incorrect. ECA-1.1 requires Containment Spray remain running unitl RWST < 3.0%. Plausible since this is value required for Recirc Spray equipment operation if candidate transposes RWST level of 6% with Containment sump level.
C. Correct.
D. Incorrect. ECA-1.1 requires Containment Spray remain running unitl RWST < 3.0%. Plausible because this is value that CHG and LHSI pumps taking suction from the RWST must be stopped.

Reference, 1-ECA-1.1 steps 8, and 9.

NUMBER	PROCEDURE TITLE	REVISION
1-ECA-1.1	LOSS OF EMERGENCY COOLANT RECIRCULATION	39
		PAGE 7 of 33

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Operation of an OSRS pump without the associated CS pump could cause cavitation as indicated by fluctuating amperage.

* *

NOTE: If CLS can NOT be reset, local breaker operation will be required to stop CS and ISRS pumps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.___	CHECK RECIRCULATION SPRAY SYSTEM:	
	a) Check for EITHER of the following: ☐ • Any CS pump - RUNNING OR REQUIRED OR ☐ • RWST level - LESS THAN 20%	☐ a) GO TO Step 10.
	☐ b) Check CTMT sump level - GREATER THAN 4.0 ft	b) Do the following: ☐ 1) Check CLS reset. IF NOT, THEN reset both trains of CLS. ☐ 2) Stop RS pumps. ☐ 3) WHEN sump level greater than 4.0 ft, THEN do Steps 8c and 8d. ☐ 4) GO TO Step 9.
	☐ c) Check SW aligned to at least two RS HXs	☐ c) Align SW to at least two RS HXs.
	☐ d) Start RS pumps associated with aligned RS HXs	

Step 9 on next page.

NUMBER	PROCEDURE TITLE	REVISION 39
1-ECA-1.1	LOSS OF EMERGENCY COOLANT RECIRCULATION	PAGE 8 of 33

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- CHG and LHSI pumps taking suction from the RWST must be stopped when level lowers to 6%.
- CS pumps taking suction from the RWST must be stopped when level lowers to 3%.

* *

9. ___ DETERMINE CS REQUIREMENTS:

☐ a) Determine number of CS pumps required:

CONTAINMENT PRESSURE	RS PUMPS RUNNING	CS PUMPS REQUIRED
GREATER THAN 60 PSIA	--------	2
BETWEEN 14 PSIA AND 60 PSIA	FEWER THAN 2	2
	2 OR MORE	1
LESS THAN 14 PSIA	--------	0

☐ b) CS pumps running - EQUAL TO NUMBER REQUIRED

b) Do the following:

☐ 1) Manually operate CS pump(s).

2) Close associated CS pump discharge MOVs for stopped pump(s):

☐ • 1-CS-P-1A, 1-CS-MOV-101A and 1-CS-MOV-101B

☐ • 1-CS-P-1B, 1-CS-MOV-101C and 1-CS-MOV-101D

Reference from ND-95.3-LP-20, ECA-1.1, steps 8 and 9.

STEP 8: CHECK RECIRCULATION SPRAY SYSTEM.

a. The purpose of this step is to determine if RS pumps can be started for RWST conservation.

b. The RS pumps take suction from the Ctmt sump and are the preferred means of Ctmt heat removal to conserve RWST inventory. In addition, the RS systems provide the long term heat sink during sump recirculation.

c. If no CS pump is running, securing spray systems is not needed to conserve RWST inventory and the team skips step 9 which establishes the number of pumps for required Ctmt heat removal.

d. Recirc sump cooling is also established if RWST is less than 20%, consistent with transfer to sump recirc in ES-1.3. Even if offsite power is not available, this allows SW to be restored to the CC HXs for normal RHR cooling in subsequent steps if sufficient inventory is still available in the RWST.

e. Adequate Ctmt sump level to support RS pumps is addressed in substep b). 4.0 feet is the minimum sump level necessary to prevent cavitation of the IRS and ORS pumps while operating.

f. Substeps c) and d) start RS pumps and verify SW to the RS HXs to establish sump cooling.

g. **This step is not pertinent for recovering from only a LOCA outside Ctmt** event since Ctmt conditions would be normal. **However, this step should still be performed. (rk).**

Step 9 on next page

CAUTION PRIOR TO STEP 9: CHG and LHSI pumps taking suction from the RWST must be stopped when level lowers to 6%.

a. The purpose of this note is to alert the team to stop any Safety Injection pumps before they lose suction from the RWST, below 6%.

b. This is to prevent damage to the affected pumps

CAUTION #2 PRIOR TO STEP 9: CS PUMPS TAKING A SUCTION FROM THE RWST MUST BE STOPPED WHEN LEVEL LOWERS TO 3%.

a. The purpose of this note is to alert the team to stop the Containment Spray pumps before they lose suction from the RWST, below 3%.

b. This is to prevent damage to the affected pumps.

K/A Number: 026A4.01: Containment Spray: Ability to manually operate and/or monitor in the control room: CSS controls.

Level: RO Tier #: 2 Group #: 1 IR – RO: 4.5 IR-SRO: 4.3

Proposed Question: 24

The breaker for 1-CS-P-1A, Unit 1 “A” Containment Spray Pump (14H5), is racked out for maintenance.

Which ONE of the following describes the effect on the following valves?

- CS Suction (1-CS-MOV-100A)
- CS Discharge (1-CS-MOV-101A and 1-CS-MOV-101C)

A. Suction and Discharge valves cannot be opened if closed.

B. Suction and Discharge valves cannot be closed if open.

C. Suction valve ONLY cannot be cycled open or closed.

D. Discharge valves ONLY cannot be cycled open or closed.

Proposed Answer: B.

Explanation: A contact in the “close” circuit is used to ensure that the affected CS pump breaker is racked in or to “test” to allow the suction/discharge valves to close.

Technical Reference: 11448-ESK-6BQ, Sheets 1 and 2; 11448-ESK-6AC

Reference Provided to Applicant: No

Learning Objective: ND-91-LP-5, CS System, Objective B, Using a simplified one-line diagram drawn from memory, describe the operation of the major components of the Containment Spray System, including the start/stop signals, the interlocks, and the Control Room instrumentation available.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 45.5 to 45.8)

Comments:

K/A Match Analysis:

Question Matches K/A. Candidate must determine if CS valves can be manually operated based on CS pump breaker position.

Distractor Analysis:

A. Incorrect. CS valves can be opened at any time as long as they have power. Plausible if Candidate incorrectly recalls interlock.

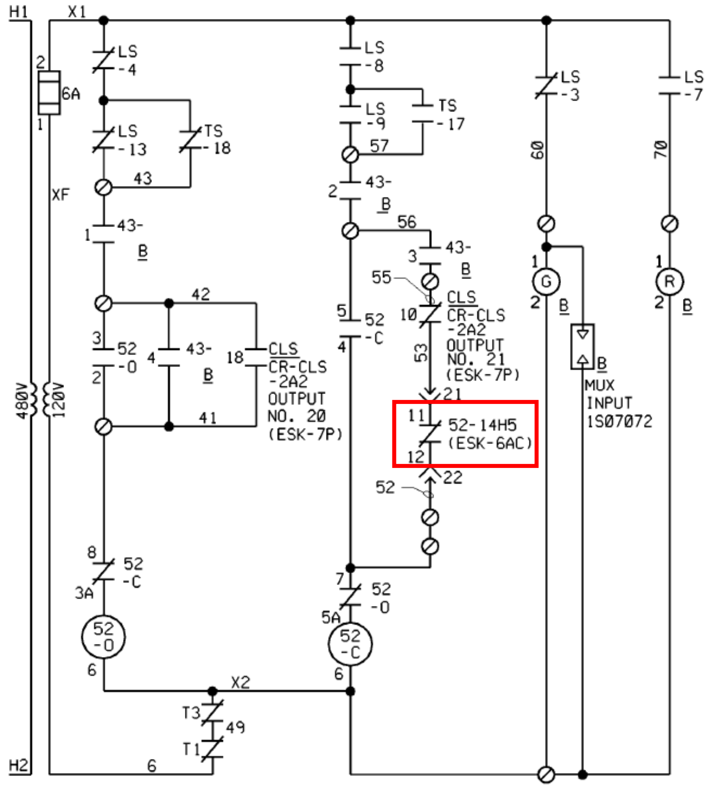
B. Correct based on ESK and LP description.

C. Incorrect. Both suction and discharge valves are affected. CS valves can be opened at any time as long as they have power. Plausible if Candidate incorrectly recalls interlock.

D. Incorrect. Both suction and discharge valves are affected. Plausible if Candidate incorrectly recalls interlock.

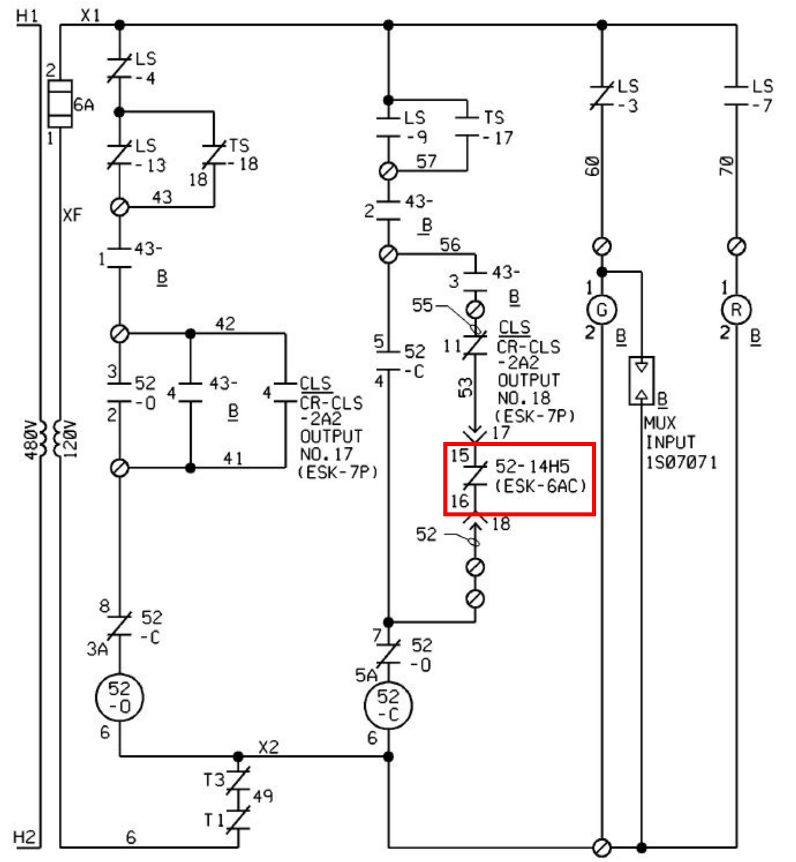
Excerpt from ND-91-LP-5, Containment Spray System

- CS-MOV-100A and B are controlled from benchboard 1-1 and have red/green indication lights for valve position.
- Valves are normally open but receive an open signal upon initiation of a HI-HI CLS to ensure the valve is open.
- These valves cannot be closed unless the respective CS pump breaker is open, racked in, and the CLS signal is reset. The pump breaker must be racked in (or to test) to enable the breaker position contacts to operate the interlock.
- In order to close a discharge MOV, these conditions must be met:
  - Handswitch to CLOSE
  - HI-HI CLS signal reset
  - Respective emergency bus powered CS pump breaker must be racked in (or in test) and OPEN. For example, the “H” powered discharge MOVs look at the “A” CS pump breaker and the “J” powered MOVs look at the “B” CS pump breaker. Note: Each pump has one discharge MOV powered from the opposite emergency bus.



CONTAINMENT SPRAY PUMP DISCH VALVE 01-CS-MOV-101A

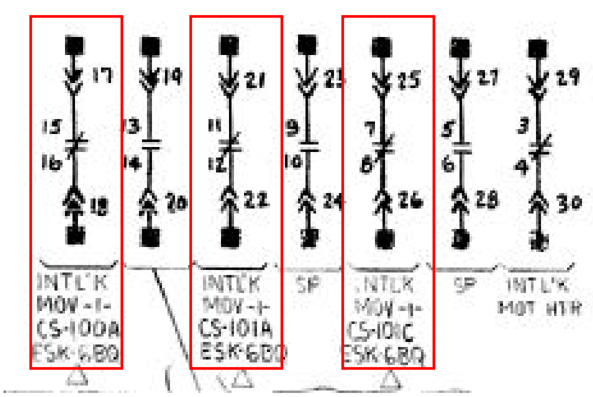
CKT:CS101A MCC 1H1-2 CUBICLE 1A



CONTAINMENT SPRAY PUMP SUCTION VALVE 01-CS-MOV-100A

CKT:CS100A MCC 1H1-2 CUBICLE 3B

From 11448-ESK-6AC



K/A Number: 026AA1.05: Loss of ComponentCooling Water / 8: Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: The CCWS surge tank, including level control and level alarms, and radiation alarm.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.1 IR-SRO: 3.1

Proposed Question: 25

Given the following:

Unit 1 is operating at 100%, Unit 2 is in Cold Shutdown.

- The following annunciators have just alarmed:
  - 0-RM-L5, CC HX A/B OUT ALERT/FAILURE.
  - 0-RM-L6, CC HX C/D OUT ALERT/FAILURE.
  - 0-RM-M5, 1-CC-RI-105 HIGH.
  - 0-RM-M6, 1-CC-RI-106 HIGH.
- 1-CC-LI-100, CC Surge Tank is reading 51% and is rising at 1%/minute.
- The BOP operator attempts to close the CC Surge Tank Vent Isolation valve by placing HCV-CC-100 to OFF.
- HCV-CC-100 remains OPEN.

Which one of the following indicates the NEXT action that shall be taken to attempt to close the CC Surge Tank Vent Isolation valve?

A. Close breaker 22 on Unit 1 Semi-vital bus.

B. Locally take the handswitch “HCV-CC-100” to the “CLOSE” position.

C. Take the handswitch “SOV-CC-200” on Unit 2 Vertical Board to the “OFF” position.

D. Locally turn the handwheel on HCV-CC-100 in the clockwise direction until valve is closed.

Proposed Answer: C

Explanation: HCV-CC-100 should have closed on High radiation from 1-CC-RI-105, or 1-CC-RI-106. Failing to auto close the operator is required to manually close HCV-CC-100. The operator attempted to close from unit 1 and the valve remains open. The operator next attempts to close the valve using Unit SOV-CC-200 handswitch.

Technical Reference: 0-RM-L5, CC HX A/B OUT ALERT/FAILURE, Rev. 3.

Reference Provided to Applicant: No

Learning Objective: ND-88.5-LP-1, Component Cooling, Objective C; Outline the instrumentation, alarms, and controls provided for the component cooling system, including; CC system Annunciators, **CC system trip valves,** Effect of auto start inhibit on CC pumps, Local operation of CC Pumps, Reg Guide 1.97 instrumentation, Vertical board 1-1 instrumentation, Vertical board 1-2 Instrumentation and controls.

Question Source: Modified Bank (LORP LCC00016)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.7 / 45.5 / 45.6)

Comments:

K/A analysis: Question requires the candidate to demonstrate knowledge of operation of HCV-CC-100. Therefore this meets the K/A.

Distractor Analysis:

A. Incorrect. This valve is a deenergize-to-close valve, therefore closing a breaker will not cause HCV-CC-100 to close. Plausible if operator is confused with how the valve operates.
B. Incorrect. There is no local handswitch. Plausible if operator is confused with how the valve operates.
C. Correct.
D. Incorrect. There is no local handwheel.This valve requires both unit 1 and unit 2 manual CS to be placed in same condition. Plausible if operator is confused with how this valve operates.

Parent question LCC00016 (LORP).

11	ID: LCC00016	Points: 1.00

Both CC radiation monitors have alarmed "high". The HCV-CC-100 (CC surge tank vent valve) indicates open on VB-2 and the RO has taken the handswitch to the "OFF" position.

Which ONE of the following indicates what other action(s) can be taken to attempt to close the CC surge tank vent path?

A.	Take the handswitch "HCV-CC-200" on unit 2 VB-2 to "OFF".
B.	Locally take the handswitch "HCV-CC-100" to the "CLOSE" position.
C.	Locally turn the handwheel on HCV-CC-100 in the clockwise direction.
D.	Open breaker 22 on Unit 1 Semi-vital bus.

Answer:	A

Question 11 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	91484
User-Defined ID:	LCC00016
Cross Reference Number:	
Topic:	LCC00016
Num Field 1:	
Num Field 2:	
Text Field:	1.00
Comments:	(B13DSG1E02); [S95-0854] (O), CC, ARP, RM M-5 CEP-121, CEP-901

Associated objective(s):

(P) Respond to a component cooling radiation monitor alarm in accordance with ARP RM H-3 or H-4.

(P) Determine the performance and design attributes of the Basic Electrical Distribution System.

Reference from ND-88.5-LP-1, slide 10, showing HCV-100 with two SOV's. Note there is no local handswitch, MOV handwheel or other manual handwheel for operation of this HCV.

Unit 1, SOV. Unit 2 SOV.

Reference from NCRODP 51-S, Component Cooling Water, page 31.

Major Control Valves

The following paragraphs describe the operation of the major control valves associated with the CC System. The major control valves are operated in the MCR from the benchboard, the vertical boards or the ventilation panel. Many of the components cooled by CC water have control valves in the CC discharge line. These valves are controlled by signals generated in the system being cooled by CC water; therefore, the operation of these valves is discussed in the applicable system module, rather than in this module. Where similar components are used in the system, only the "A" component and/or Unit 1 component is described.

The CC surge tank is normally vented to the process vent header via HCV-CC100. When the radiation levels in the CC subsystem are high (as sensed by RM-CC105 and -CC106), HCV-CC100 shuts to prevent radioactive gases from entering the process vent header.

HCV-CC-100 is a pilot solenoid, air-operated valve and is controlled from vertical board 1-2 in the MCR. The control switch is a two-position (OFF, ON) switch. Green and red lights above the switch inform the operator of the valve position. Both of the control switches must be in the "OPEN" position, or the valve will shut.

The trip valves associated with the CC System are all pilot-solenoid, air-actuated control valves, and are operated from the MCR. The valves are designed to fail shut on loss of electrical power to the pilot solenoid or loss of instrument air pressure except for the RCP thermal barrier TVs which fail

Reference from 0-RN-L5, required actions for closing HCV-CC-100.

NUMBER	PROCEDURE TITLE	REVISION
0-RM-L5	CC HX A/B OUT ALERT/FAILURE	3
		PAGE 3 of 4

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. ___	CONSULT WITH SHIFT SUPERVISION AND CHECK CLOSURE OF CC SURGE TANK VENT VALVE - DESIRED	☐ GO TO Step 4.
3. ___	CLOSE CC SURGE TANK VENT VALVE: ☐ a) Place HCV-CC-100 in OFF (Unit 1) ☐ b) Place SOV-CC-200 in CLOSE (Unit 2)	

K/A Number: 027K2.01: Containment Iodine Removal: Knowledge of bus power supplies to the following: Fans.

Level: RO Tier #: 2 Group #: 2 IR – RO: 3.1 IR-SRO: 3.4

Proposed Question: 26

Given the following conditions:

- Containment atmosphere cleanup on Unit 1 is in progress.
- 1-VS-F-3A and 1-VS-F-3B, Iodine Filtration Fans are running.

The loss of which ONE of the following electrical busses would cause 1-VS-F-3B to stop?

A. “A” Station Service bus.

B. “B” Station Service bus.

C. “H” Emergency bus.

D. “J” Emergency bus.

Proposed Answer: B. “B” Station Service bus.

Explanation: 1-VS-F-3B is powered from the 1B1-2E Station Service 480v MCC.

Technical Reference: Station Drawing 11448-FE-1N, Sh. 001

Reference Provided to Applicant: No

Learning Objective: ND-88.4-LP-6, CTMT Vent, Objective C, Describe the purpose and operation of the iodine filtration system.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7)

Comments:

KA Match Analysis: This question matches the KA in that the trainee must know that the Iodine Filtration fans are Station Service powered in order to answer the question correctly. The trainee must also know which particular bus powers each fan.

Distractor Analysis:

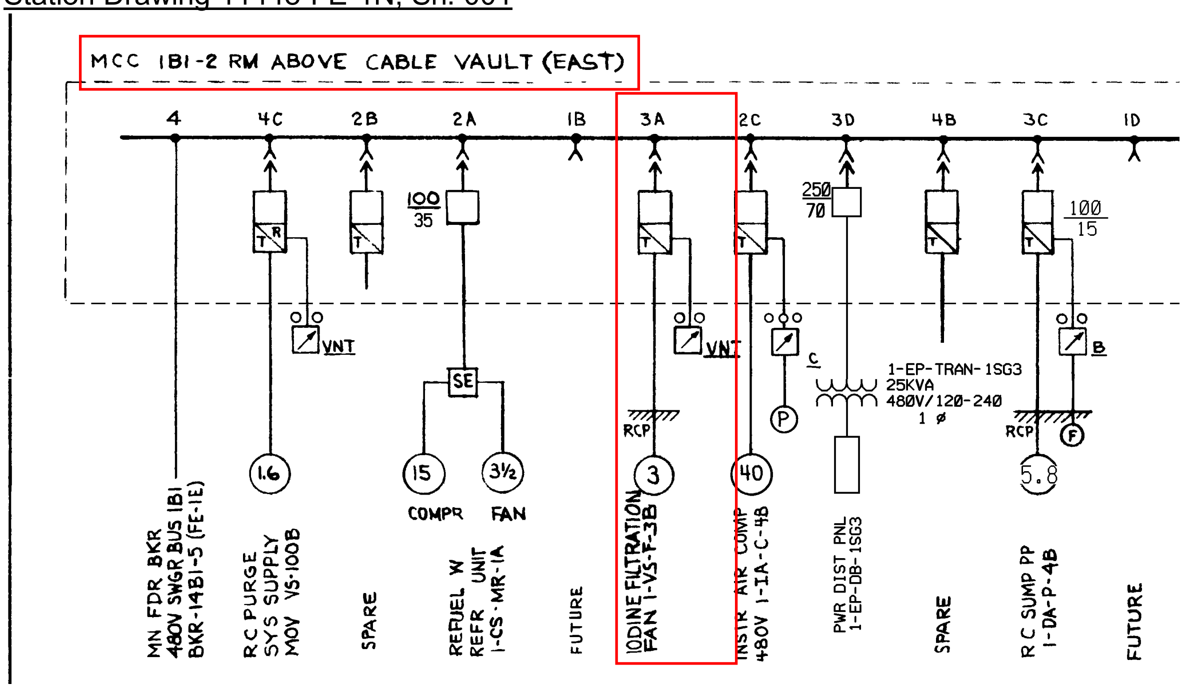
A. Incorrect but plausible since the candidate may correctly assume Station Service power, but wrong bus.

B. CORRECT.

C. Incorrect but plausible since the candidate may assume the fan is Emergency Bus powered.

D. Incorrect, but plausible if the candidate assumes that 1-VS-F-3B is powered from the 1J bus since the "B" train ESF equipment is "J" powered by convention.

Station Drawing 11448-FE-1N, Sh. 001



K/A Number: 028K5.04: Hydrogen Recombiner and Purge Control: Knowledge of the operational implications of the following concepts as they apply to the HRPS: The selective removal of hydrogen.

Level: RO Tier #: 2 Group #: 2 IR – RO: 2.6 IR-SRO: 3.2

Proposed Question: 27

A Large Break Loss of Coolant Accident (LBLOCA) has occurred on Unit 1 with the following conditions:

- Containment H_2 concentration is 2.8% as determined in 1-E-1, Step 26.
- The operating team is performing 1-E-1, Attachment 3 to place H_2 recombiners in service.

Which ONE of the following describes:

1) Whether the H_2 recombiners be placed in service at this H_2 concentration?
2) The temperature (°F) at which recombination occurs?

A. 1) Yes.
2) 1150°F - 1200°F

B. 1) Yes.
2) 2200°F

C. 1) No.
2) 1150°F - 1200°F

D. 1) No.
2) 2200°F

Proposed Answer: A. 1) Yes.

2) 1150°F - 1200°F

Explanation: Procedure 1-E-1, step 26 directs the team to determine Containment Hydrogen concentration and place the hydrogen recombiners in service if H_2 concentration is >0.5%, but < 4%. Directions in 1-E-1, Attachment 3 have the operator maintain thermocouple readings within a range of 1150°F and 1200°F. A NOTE in the attachment states that thermocouple temperature should not exceed 1400°F.

Technical Reference: 1-E-1, Loss of Reactor or Secondary Coolant, Rev 43

Reference Provided to Applicant: NO

Learning Objective: ND-93.4-LP-4, POST ACCIDENT EQUIPMENT Objective C, Describe the operation of the Electric Hydrogen Recombiner System.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.5 / 45.7)

Comments:

KA Match Analysis: This question asks the trainee if the recombiners may be placed in service at a specific condition (operational) and at what temperature H_2 recombines with free O_2 to for H_20 (selective removal).

Distractor Analysis:

A. CORRECT – Both are correct. Part 1 is correct in stating that the recombiners may be placed in operation at 2.8% H_2 (limit = 0.5% - 4%). Part 2 is correct in that recombination of H_2 occurs at 1150°F - 2000°F.

B. Incorrect – Part 1 is correct in stating that the recombiners may be placed in operation at 2.8% H_2 (limit = 0.5% - 4%). Part 2 is incorrect in that this is the temperature at which the Zirc Hydriding Reaction (production of H_2 from zirc-water reaction) markedly increases. It is plausible since this is a common number discussed about H2 production during an accident.

C. Incorrect – Part 1 is incorrect. It is plausible that the trainee will believe that the recombiners should be placed in service at a higher H2 concentration. Part 2 is correct in that recombination of H_2 occurs at 1150°F - 2000°F.

D. Incorrect – Part 1 is incorrect. It is plausible that the trainee will believe that the recombiners should be placed in service at a higher H2 concentration. Part 2 is incorrect in that this is the temperature at which the Zirc Hydriding Reaction (production of H_2 from zirc-water reaction) markedly increases. It is plausible since this is a common number discussed about H2 production during an accident.

NUMBER	PROCEDURE TITLE	REVISION
1-E-1	LOSS OF REACTOR OR SECONDARY COOLANT	43
		PAGE 27 of 29

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- This procedure should be continued while obtaining hydrogen sample in Step 26.
- Hydrogen analyzer heat tracing must be energized for twenty minutes before sample analysis.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
26.___	CHECK CTMT HYDROGEN CONCENTRATION:	
	☐ a) Put CTMT H2 analyzer in service IAW Attachment 2	
	☐ b) Check H2 concentration - LESS THAN 4%	☐ b) Consult with TSC for additional recovery actions. GO TO Step 27.
	☐ c) Check H2 concentration - LESS THAN 0.5%	☐ c) Turn on hydrogen recombiners IAW Attachment 3.

NUMBER 1-E-1	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 43	ELECTRIC HYDROGEN RECOMBINER OPERATION	PAGE 1 of 2

* *

CAUTION: Emergency Diesel Generator loading must not exceed 2675 KW before the Hydrogen Recombiner is energized.

* *

PLACING RECOMBINERS IN SERVICE

NOTE: Heater temperature as determined by any thermocouple should not exceed 1400°F.

1. ___ Turn the POWER OUT switch to ON.

2. ___ Record present CTMT pressure. ____________ PSIA

3. ___ Record pre-LOCA CTMT temperature. _________°F

4. ___ Determine the Pressure Factor (C_p) IAW Attachment 4.

5. ___ Calculate the Recombiner power setting using the following:

   ________ (C_p) x 38.1 KW = ________ (1A required power setting)

   ________ (C_p) x 46.2 KW = ________ (1B required power setting)

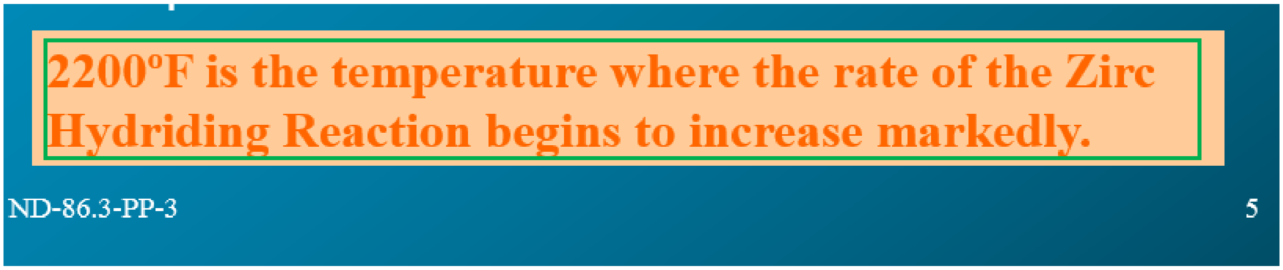
6. ___ Adjust the POWER ADJUST pot clockwise until 5 KW is obtained on the POWER OUT meter. Hold 5 KW for ten minutes.

7. ___ Adjust the POWER ADJUST pot until 10 KW is obtained on the POWER OUT meter. Hold 10 KW for ten minutes.

8. ___ Adjust the POWER ADJUST pot until 20 KW is obtained on the POWER OUT meter. Hold 20 KW for five minutes.

9. ___ Adjust the POWER ADJUST pot to obtain the required power setting calculated in Step 5. Allow the Recombiner to stabilize for 2 hours.

10. ___ Monitor the temperature of the three thermocouples. Adjust the POWER ADJUST pot to maintain an average thermocouple reading within a range of 1150°F to 1200°F. (An adjustment of 4 KW will result in a temperature change of approximately 75°F.)


2200ºF is the temperature where the rate of the Zirc Hydriding Reaction begins to increase markedly.
ND-86.3-PP-3
5

K/A Number: 029EK1.01: ATWS/1: Knowledge of the operational implications of the following concepts as they apply to the ATWS: Reactor nucleonics and thermo-hydraulics behavior.

Level: RO   Tier #: 1   Group #: 1   IR – RO: 2.8   IR-SRO: 3.1

Proposed Question: 28

The following plant conditions exist:

- An ATWS is in progress.
- All feedwater to the steam generators has been lost.
- Turbine trip has NOT occurred.
- Main Generator output breakers are closed.

Which ONE of the following would be expected indications several minutes into the event? (Assume no operator action is taken.)

	Reactor Power	PZR Pressure	PZR Level	Steam Pressure
A.	Stable	Lowering	Lowering	Lowering
B.	Lowering	Rising	Rising	Lowering
C.	Stable	Rising	Rising	Rising
D.	Lowering	Lowering	Lowering	Rising

Proposed Answer:

	Reactor Power	PZR Pressure	PZR Level	Steam Pressure
B.	Lowering	Rising	Rising	Lowering

Explanation: With a Loss of Feedwater ATWS and the turbine not tripped, the S/G's will boil dry resulting in lowering steam pressure. With no heat removal, this will also lead to rising RCS temperature. As RCS temperature rises, PZR level rises causing a rise in PZR pressure. The rise in RCS temperature also adds negative reactivity, causing reactor power to lower.

Technical Reference: FR-S.1 WOG Background Document, Rev 2

Reference Provided to Applicant: NO

Learning Objective: ND-95.1-LP-11 ANTICIPATED TRANSIENT WITHOUT TRIP (ATWT), Objective B, Diagnose the sequence of events for the most limiting ATWT event.

Question Source: Modified Bank (Surry ILT Bank TAA0081 – modified stem/distractors)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.8 / 41.10 / 45.3)

Comments:

KA Match Analysis: This question matches the KA in that the candidate must know how the reactor and the primary plant behave during an ATWT event.

Distractor Analysis:

A. Incorrect but plausible if candidate assumes that reactor power will remain stable due to steam demand remaining unchanged.

B. CORRECT.

C. Incorrect but plausible if the candidate assumes that reactor power will remain stable due to steam demand remaining unchanged and the RCS heatup causes S/G pressure to rise.

D. Incorrect but plausible if the candidate believes the reduction in reactor power leads to a reduction in RCS temperature, resulting in lowering pressure and level.

Parent Question TAA0081.

The following plant conditions exist:

- *An ATWS is in progress.*
- *All feedwater to the steam generators has been lost.*
- *The turbine generator has remained loaded and running.*

Which ONE of the following would be an indication of the above conditions several minutes after the ATWS occurred? (Assume all control systems are in AUTO and no operator action is taken.)

A. Reactor power decreases; pressurizer pressure decreases; pressurizer level decreases; steam pressure decreases.

B. Reactor power decreases; pressurizer pressure increases; pressurizer level increases; steam pressure decreases.

C. Reactor power remains stable; pressurizer pressure increases; pressurizer level increases; steam pressure increases.

D. Reactor power increases; pressurizer pressure decreases; pressurizer level decreases; steam pressure increases.

Answer: B

Figure 5. NUCLEAR POWER VERSUS TIME FOR A LOSS OF NORMAL FEEDWATER ATWS

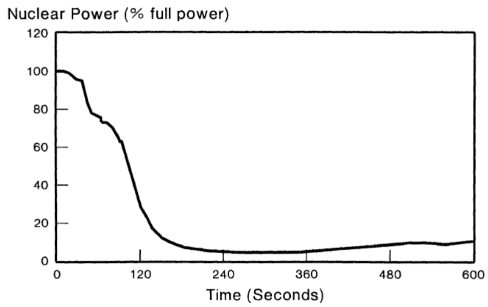


Figure 6. RCS AVERAGE COOLANT TEMPERATURE VERSUS TIME FOR A LOSS OF NORMAL FEEDWATER ATWS

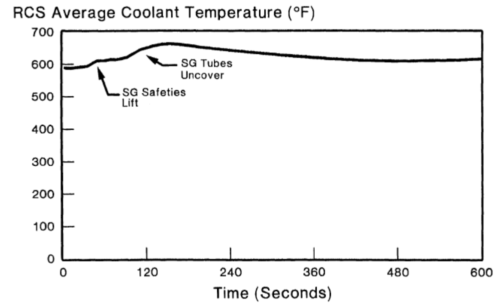


Figure 7. PRESSURIZER PRESSURE VERSUS TIME FOR A LOSS OF NORMAL FEEDWATER ATWS

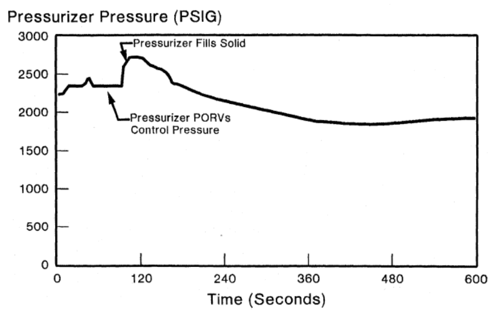
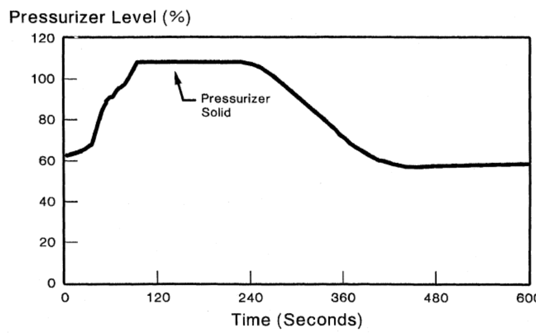


Figure 8. PRESSURIZER LEVEL VERSUS TIME FOR A LOSS OF NORMAL FEEDWATER ATWS



K/A Number: 034K6.02: Fuel Handling Equipment: Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System: Radiation monitoring systems.

Level: RO Tier #: 2 Group #: 2 IR – RO: 2.6 IR-SRO: 3.3

Proposed Question: 29

Which ONE of the following Radiation Monitors being removed from service would result in fuel movement being stopped?

A. New Fuel Storage Area Radiation Monitor (1-RM-RI-152).

B. Containment High Range Radiation Monitor (1-RM-RI-128).

C. Containment High Range Gamma Radiation Monitor (1-MS-RM-161).

D. Manipulator Crane Area Radiation Monitor (1-RM-RI-162).

Proposed Answer: D. Manipulator Crane Area Radiation Monitor (1-RM-RI-162).

Explanation: 1-OSP-ZZ-004, Unit 1 Safety Systems Status List for Cold Shutdown/Refueling Conditions and Tech Specs 3.10.A.3 require that Containment Particulate & Gas (RM-RI-159/160) and Manipulator Crane Area (RM-RI-162) be operable, therefore when one is inoperable, refueling activities must cease.

Technical Reference:

1-OSP-ZZ-004, Unit 2 Safety Systems Status List for Cold Shutdown/Refueling Conditions, Rev 48

Technical Specification 3.10, Refueling, Item A.3, and C

Reference Provided to Applicant: NO

Learning Objective: ND-92.5-LP-1, Refueling Overview, Objective D, Describe the Technical Specifications associated with refueling, including for SRO candidates, the basis behind these specifications.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 45.7)

Comments:

KA Match Analysis: This question meets the KA in that the candidate needs to have the knowledge that station procedures and Tech Specs requires that Manipulator Crane and CTMT Particulate/Gas RMs must all be operable for fuel movement to be allowed to take place.

Distractor Analysis:

A. Incorrect, but plausible since other rad monitors in the area, such as Fuel Bridge, would require fuel movement to stop.

B. Incorrect, but plausible if the operator confuses the CHRRMS radiation monitors with the 159/160 containment rad monitors.

C. Incorrect but plausible if the operator confuses the Containment High Range Gamma radiation monitor with the 159/160 containment rad monitors.

D. CORRECT – See explanation above.

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Attachment 8
REFUELING OPERATIONS REQUIREMENTS

EQUIPMENT	MIN REQ	D	N	TECH SPECS	REMARKS
Refueling Containment Integrity set	As Required			3.10.A.1	IAW 1-OP-FH-001
Radiation Monitors: • Manipulator Crane • Containment Gaseous • Containment Particulate • SFP Bridge • Vent-Vent Gaseous • Vent-Vent Particulate	 1 operable 1 operable 1 operable 1 operable 1 operable 1 operable			3.10.A.3 3.10.B.1	If the Containment Air Recirculation fans are not running then refer to Tech Spec 3.10 for actions.
[illegible]	[illegible]			[illegible]	

3.10 REFUELING

Applicability

Applies to operating limitations during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building.

Objective

To assure that no accident could occur during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building that would affect public health and safety.

Specification

A. During REFUELING OPERATIONS the following conditions are satisfied:

1. The equipment access hatch and at least one door in the personnel airlock shall be capable of being closed. For those penetrations which provide a direct path from containment atmosphere to the outside atmosphere, the containment isolation valves shall be OPERABLE or the penetration shall be closed by a valve, blind flange, or equivalent or the penetration shall be capable of being closed.

2. At least one source range neutron detector shall be in service at all times when the reactor vessel head is unbolted. Whenever core geometry or coolant chemistry is being changed, subcritical neutron flux shall be continuously monitored by at least two source range neutron detectors, each with continuous visual indication in the Main Control Room and one with audible indication within the containment. During core fuel loading phases, there shall be a minimum neutron count rate detectable on two operating source range neutron detectors with the exception of initial core loading, at which time a minimum neutron count rate need be established only when there are eight (8) or more fuel assemblies loaded into the reactor vessel.

3. The manipulator crane area monitors and the containment particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

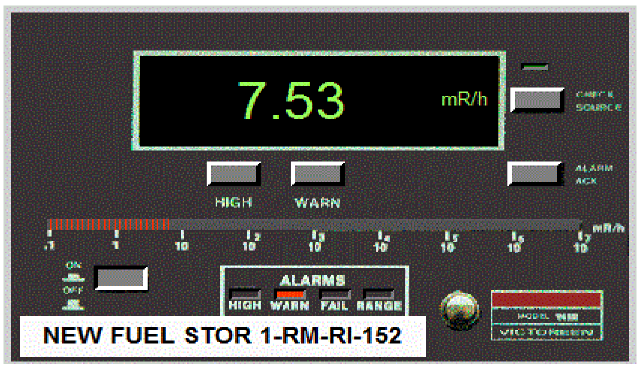
K/A Number: 036AA2.01: Fuel Handling Accident / 8: Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: ARM system indications.

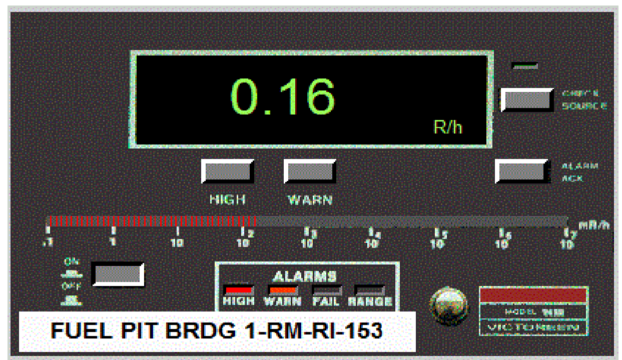
Level: RO Tier #: 1 Group #: 2 IR – RO: 3.2 IR-SRO: 3.9

Proposed Question: 30

The Operations Fuel Handling Group is loading a Spent Fuel Cask.

The following Radiation Monitor indications are received in the Main Control Room:





Which ONE of the following identifies:

1) The effect that this event will have on Fuel Handling Activities?
2) The actions (if any) required by the Main Control Room?

A. 1) Fuel Handling Activities must stop.

2) Secure normal MCR ventilation, then start one MCR Emergency Ventilation fan.

B. 1) Fuel Handling Activities may continue.

2) Align Fuel Building Ventilation to Filtered Exhaust via 1-VS-F-58A/B.

C. 1) Fuel Handling Activities must stop.

2) Align Fuel Building Ventilation to Filtered Exhaust via 1-VS-F-58A/B.

D. 1) Fuel Handling Activities may continue.

2) Secure normal MCR ventilation, then start one MCR Emergency Ventilation fan.

Proposed Answer: A. 1) Fuel Handling Activities must stop.

2) Secure normal MCR ventilation, then start one MCR Emergency Ventilation fan.

Explanation: The RM indications shown above point to a Fuel Handling incident in the Fuel Building. The ARPs for the RMs send the MCR team to 0-AP-22.00, which directs stopping FH operations, evacuating the area, and shift MCR ventilation to Emergency Ventilation.

Technical Reference: 0-RM-D3, 1-RM-RI-153 High, rev 7

0-AP-22.00, Fuel Handling Abnormal Conditions, Rev 24

Reference Provided to Applicant: NO

Learning Objective: ND-92.5-LP-7, Refueling APs – Objective C. Given a set of plant conditions, determine the appropriate operator response in accordance with AP-22.00, Fuel Handling Abnormal Conditions, AP-22.01, Loss of Refueling Cavity Level, and AP-22.02, Malfunction of Spent Fuel Pit Systems, to include the following:

- Immediate Actions (if applicable)
- Entry Conditions
- Major Actions
- Step Bases

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43.5 / 45.13)

Comments:

KA Match Analysis: This question matches the KA in that several RM indications are given and the candidate must diagnose that a Fuel Handling incident has occurred.

Distractor Analysis:

A. CORRECT: both parts are correct.
B. Incorrect: Part 1 is incorrect, but plausible if the candidate assumes that FH operations may continue to complete the Fuel Assembly reconstitution to stop the release. Part 2 is incorrect but plausible if the candidate believes that the FB exhaust must be aligned to Filtered Exhaust to mitigate the release to the environment.
C. Incorrect: Part 1 is correct. Part 2 is incorrect but plausible if the candidate believes that the FB exhaust must be aligned to Filtered Exhaust to mitigate the release to the environment.
D. Incorrect: Part 1 is incorrect, but plausible if the candidate assumes that FH operations may continue to complete the Fuel Assembly reconstitution to stop the release. Part 2 is correct.

NUMBER	PROCEDURE TITLE	REVISION
0-RM-D3	1-RM-RI-153 HIGH	7
		PAGE 3 of 3

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. ___	CHECK WITH FUEL HANDLING PERSONNEL - ABNORMAL CONDITION EXISTS	☐ GO TO Step 8.
7. ___	INITIATE 0-AP-22.00, FUEL HANDLING ABNORMAL CONDITIONS	

NUMBER	PROCEDURE TITLE	REVISION
0-AP-22.00	FUEL HANDLING ABNORMAL CONDITIONS	24
		PAGE 2 of 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK FUEL REPAIR - IN PROGRESS	☐ GO TO Step 4.
2.___	CHECK LOCAL RADIATION CONDITIONS - NORMAL	☐ GO TO Step 4.
3.___	GO TO STEP 20	
4.___	STOP FUEL HANDLING OPERATIONS	
5.___	EVACUATE THE AFFECTED AREA ☐ • Containment OR ☐ • Fuel Building	
6.___	CHECK MCR EMERGENCY VENTILATION - NOT IN SERVICE	☐ GO TO Step 10.
7.___	SECURE NORMAL MCR VENTILATION ☐ a) Close 1-VS-MOD-103A ☐ b) Close 1-VS-MOD-103B ☐ c) Close 1-VS-MOD-103C ☐ d) Close 1-VS-MOD-103D	
8.___	VERIFY STOPPED OR STOP MCR VENTILATION FANS ☐ • 1-VS-F-15 ☐ • 1-VS-AC-4	

NUMBER	PROCEDURE TITLE	REVISION
0-AP-22.00	FUEL HANDLING ABNORMAL CONDITIONS	24
		PAGE 3 of 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- Chilled water flow to the in-service Unit 1 MCR AHU must be throttled to at least 15 gpm when the Emergency Supply fan is started.
- Chilled water flow to the in-service Unit 2 MCR AHU must be throttled to at least 25 gpm when the Emergency Supply fan is started.
- An Emergency Supply Fan must not be started if the filter is wet.
- Only one Emergency Supply Fan must be started.

* *

9.___ IMMEDIATELY START ONE EMERGENCY SUPPLY FAN IAW THE FOLLOWING:
(1-VS-F-41 OR 2-VS-F-41 PREFERRED)

a) Start 1-VS-F-41 IAW the following:

☐ 1) Open 1-VS-MOD-104A, CONT RM EMERG SUP MOD

☐ 2) Start 1-VS-F-41

OR

b) Start 2-VS-F-41 IAW the following:

☐ 1) Open 2-VS-MOD-204A, CONT RM EMERG SUP MOD

☐ 2) Start 2-VS-F-41

OR

(STEP 9 CONTINUED ON NEXT PAGE)

K/A Number: 037AG2.2.12: Steam Generator Tube Leak / 3: Knowledge of surveillance procedures.

Level: RO Tier #: 1 Group #: 2 IR – RO: 3.7 IR-SRO: 4.1

Proposed Question: 31

The following Unit conditions exist:

- Unit 2 is at **17%** Turbine power with a ramp up in progress.
- 2A-A3, N-16 HIGH annunciator has just alarmed.
- N-16 RMs indicate:
  - A 200 gpd
  - B 9 gpd
  - C 9 gpd

Which ONE of the following actions is required per 2A-A3, N-16 HIGH Annunciator Response Procedure?

A. No action is required, the alarm is invalid.

B. Increase surveillance of Main Steam Line radiation monitors.

C. Initiate 0-OSP-RC-002, Steam Generator Primary to Secondary Leakage Monitoring.

D. Initiate 2-OPT-RC-10.0, Reactor Coolant Leakage – Computer Calculated.

Proposed Answer:

A. No action is required, the alarm is invalid.

Explanation: With reactor power below 25%, N-16 indications are not valid per ARP 2A-A3. Per the ARP if the reading is below high setpoint then notify I&C and submit a Work Reuest.

Technical Reference: 2A-A3, N-16 HIGH ARP, Rev 2.

Reference Provided to Applicant: NO

Learning Objective: ND-93.5-LP-3 – Post-TMI Radiation Monitoring System; Objective D. Determine the operation of the N-16 Radiation Monitoring System.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 45.13)

Comments:

KA Match Analysis: The question meets the KA in that the candidate must be familiar with surveillance procedures related to SGTL and when they are required.

Distractor Analysis:

A. CORRECT.
B. Incorrect, but plausible if the candidate believes that the indication is valid, but not above the high setpoint.
C. Incorrect, but plausible if candidate believes that the indication is valid and needs to look for a primary-to-secondary leak.
D. Incorrect, but plausible if candidate believes the indication is valid and that another method of looking for a primary to secondary leak is required.

NUMBER	PROCEDURE TITLE	REVISION
2A-A3	N-16 HIGH	2
		PAGE 2 of 3

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: • N-16 Radiation Monitor setpoints are available in the NI/Radiation Monitor information book.
• N-16 Radiation Monitor readings are invalid when Reactor power is less than 25%.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK REACTOR POWER - GREATER THAN 25%	☐ Terminate this procedure.
2.___	CHECK N-16 RECORDER - ANY MONITOR READING GREATER THAN OR EQUAL TO HIGH SETPOINT ☐ • 2-MS-RR-293	Do the following: ☐ a) Notify I&C. b) Locally check N-16 Local Processing and Display Units in Cable Spreading Room. ☐ • 2-MS-RI-290 ☐ • 2-MS-RI-291 ☐ • 2-MS-RI-292 ☐ c) IF any LPDU greater than high setpoint, THEN GO TO Step 3. ☐ d) Submit Work Request. e) Increase surveillance of the following: ☐ • Air Ejector activity ☐ • NRC MS Line RMs ☐ • RCS leak rate ☐ f) GO TO Step 5.

NUMBER	PROCEDURE TITLE	REVISION
2A-A3	N-16 HIGH	2
		PAGE 3 of 3

STEP	ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: A Steam Generator tube leak of 150 gpd equates to 0.1 gpm. Leaks of this size will probably not cause an observable change in primary system parameters.

3.___	CHECK RCS LEAK RATE:	Do the following:
	☐ • PRZR level - DECREASING OR ☐ • Annunciator 2D-E5, CHG PP TO REGEN HX HI-LO FLOW - LIT OR ☐ • A discernible negative change in VCT level trend has developed	☐ a) Initiate 0-OSP-RC-002, STEAM GENERATOR PRIMARY TO SECONDARY LEAKAGE MONITORING. b) Perform RCS leak rate IAW either of the following: ☐ • 2-OPT-RC-10.0, REACTOR COOLANT LEAKAGE - COMPUTER CALCULATED OR ☐ • 2-OPT-RC-10.01, REACTOR COOLANT LEAKAGE - MANUALLY CALCULATED ☐ c) GO TO Step 5.

K/A Number: 039A3.02: Main and Reheat Steam: Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.1 IR-SRO: 3.5

Proposed Question: 32

Initial Conditions:

- Startup in progress on Unit 1.
- Unit conditions are as follows:
  - Reactor Power – 25% with unit online
  - Containment pressure = 10.5 psia and stable
  - “A” Steam Flow = 0.88E6 lbm/hr.
  - “B” Steam Flow = 0.89E6 lbm/hr.
  - “C” Steam Flow = 0.88E6 lbm/hr.

Current Conditions:

- The following parameters are now noted:
  - Reactor Power – 4.8E-7 amps
  - Containment pressure = 10.5 psia and stable
  - “A” Steam Flow = 0 lbm/hr.
  - “B” Steam Flow = 0 lbm/hr.
  - “C” Steam Flow = 0 lbm/hr.

Which ONE of the following is correct regarding:

1) How Main Steam isolation is monitored from the MCR.

2) What is the signal that caused the isolation?

A. 1) Individual MSTV position indications on MCR Bench Board;
2) Header-to-Line SI.

B. 1) Individual MSTV position indications on MCR Bench Board;
2) High Steam Flow SI.

C. 1) Individual MSNRV position indications on MCR Vertical Board;
2) High Steam Flow SI.

D. 1) Individual MSNRV position indications on MCR Vertical Board;
2) Header-to-Line SI.

Proposed Answer:

B. 1) individual MSTV position indications on MCR benchboard; 2) High Steam Flow SI.

Explanation: In this case, all MSTVs would be expected to close on the High Steam Flow SI. Header-to-Line SI does not close MSTVs. Also, even though MSNRVs will isolate the Main Steam system, they are only closed manually.

Technical Reference: 0-DRP-004 Precautions, Limitations, and Setpoints, Rev 76

Reference Provided to Applicant: NO

Learning Objective: ND-89.1-LP-2, Main Steam System Objective B. Describe the major components of the Main Steam System, including the operation of that component and the specific reason for the component.

Question Source: Modified Bank (Point Beach 2005 #43 – modified stem/distractors)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 45.5)

Comments:

KA Match Analysis: The question matches the KA in that it addresses automatic isolation of the Main Steam System and the ability of the candidate to recognize when this isolation should occur.

Distractor Analysis:

A. Incorrect, but plausible if candidate believes Header-to-Line SI automatically closes MSTVs

B. CORRECT.

C. Incorrect, but plausible if candidate believes MSNRVs are part of automatic isolation.

D. Incorrect, but plausible if candidate believes MSNRVs are part of automatic isolation and Header-to-Line SI causes automatic isolation of Main Steam System.

Point Beach 2005 #43 – modified stem/distractors

43. 2005 ILT RO 43
Consider the following Unit 1 conditions:

- Unit was at 15% reactor power.
- Main Steam Line Break on 'B' SG has occurred inside containment.
- An automatic Reactor Trip and Safety Injection occurred.
- The following parameters are now noted:
  - Containment pressure - 7 PSIG and rising.
  - T_{cold} is 450°F and lowering.
  - 'B' Steam flow is 1.8×10^6 lbm/hr.
  - 'A' Steam flow is 1.0×10^6 lbm/hr.
  - 'B' MSIV is closed.
  - 'A' MSIV is open.

Which of the following is correct regarding the 'A' MSIV?

A. 'A' MSIV should have closed due to High Containment Pressure.

B. 'A' MSIV should have closed due to Low T_{avg} and High Steam Flow coincident with the SI signal.

C. 'A' MSIV is in the proper alignment but will close if 'A' High-High Steam Flow bistable actuates coincident with the SI signal.

D. 'A' MSIV is in the proper alignment but will close if containment reaches High Containment Pressure setpoint.

Answer: B

KA#
039.A3.02

Cog Level
High

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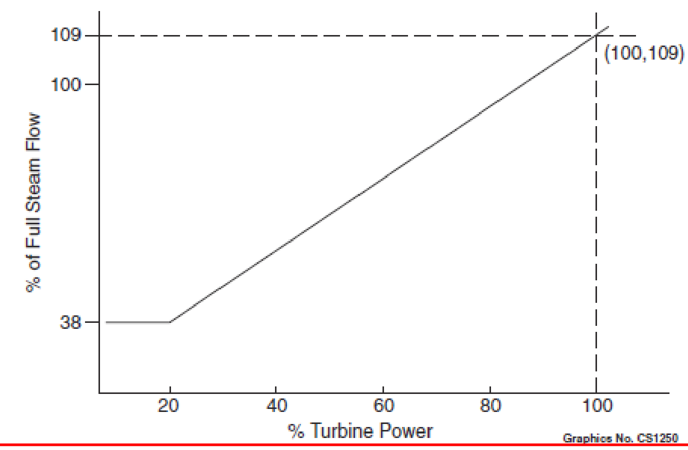
Attachment 1

REACTOR PROTECTION SYSTEM SETPOINTS



Attachment 1

REACTOR PROTECTION SYSTEM SETPOINTS

5.	Low Steam Line Pressure		
	(PC-474A, PC-485A, PC-496A)	525 psig	
	Lead time constant		
	(PM-474B, PM-485B, PM-496B)	14 seconds	
	Lag time constant		
	(PM-474B, PM-485B, PM-496B)	1.8 seconds	
6.	Steam Break/Low T_{ave}		
	(TC-412E, TC-422E, TC-432E)	543°F	
7.	Automatic Reset of Manual Block on High Pressurizer Pressure	2000 psig	(3)
	(PC-455B, PC-456B, PC-457B)		
8.	Containment High Pressure Contacts for PS-LM-100 (200);	3.0 psig 17.7 psia	(2)
	A3-1 B3-1 C3-1 D3-1		
9.	CLS Reset	14.2 psia	(1)

B. Steam Line Isolation

1.	High Steam Line Flow (See graph above)		
2.	High High Containment Pressure Contacts for PS-LM-100(200):	23 psia	
	A1-1 A1-2		
	B1-1 B1-2		
	C1-1 C1-2		
	D1-1 D1-2		

K/A Number: 039K5.08: Main and Reheat Steam: Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.6 IR-SRO: 3.6

Proposed Question: 33

A startup is in progress on Unit 1.

- The reactor is at 1E-8 amps and critical data is being taken.
- Main Steam Trip Valves and Non-Return Valves are open.
- 1-MS-TCV-105A (Condenser Steam Dump Valve) fails open.

Assuming NO OPERATOR ACTION, which ONE of the following describes the immediate effect on the plant?

	RCS Temperature	Reactor Power
A.	RISES	RISES
B.	LOWERS	RISES
C.	LOWERS	LOWERS
D.	RISES	LOWERS

Proposed Answer:

	RCS Temperature	Reactor Power
B.	LOWERS	RISES

Explanation: With the reactor critical below the POAH, as steam is drawn off RCS temperature lowers. This adds positive reactivity, causing reactor power to rise.

Technical Reference:

Reference Provided to Applicant: NO

Learning Objective: ND-86.2-LP-2 Moderator Temperature Coefficient and Defect, Objective C. Justify how and why the moderator temperature coefficient changes with moderator temperature, boron, and control rods.

Question Source: Modified Bank (Surry LORP Bank LRX00108)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 45.7)

Comments:

<u>KA Match Analysis</u>: This question matches the KA in that the candidate must determine the effect on the reactor (reactivity) from an excessive steam demand.

<u>Distractor Analysis:</u>

A. Incorrect: Plausible if candidate mistakes the cause-and-effect of the steam demand.

B. CORRECT: This would be the expected response for a negative MTC.

C. Incorrect: This would be the expected response for a positive MTC.

D. Incorrect: Plausible if candidate mistakes the cause-and-effect of the steam demand.

Parent Question (LRX00108)

87	ID: LRX00108	Points: 1.00

Which ONE of the following completes the statement below? The rector is critical at 10,000 cps when a Steam Generator PORV fails open. Assume no operator actions, no reactor trip occurs, and end-of-core-life conditions. When the reactor stabilizes, Tave will be __(1)__ than initial Tave, and reactor power will be __(2)__ the point of adding heat.

A. (1) less; (2) above

B. (1) greater; (2) at

C. (1) greater; (2) above

D. (1) less; (2) at

Answer: A

NRC APPROVED. K/A Number: 040AA1.18: Steam Line Rupture - Excessive Heat Transfer / 4: Ability to operate and / or monitor the following as they apply to the Steam Line Rupture: Control rod position indicators.

Level: RO   Tier #: 1   Group #: 1   IR – RO: 4.2   IR-SRO: 4.2

Proposed Question: 34

Given the following conditions:

- Unit 1 is stable at 90% power.
- The main turbine is in IMP OUT.
- Bank D rods are at 200 steps with the Rod Control Mode Selector switch in AUTO.
- “A” S/G Main Steam PORV (1-MS-RV-101A) fails open due to controller failure.

Which ONE of the following describes:

(1) The effect on Control Rod Position?

(2) The expected procedure used to locally isolate the PORV?

A. 1) Rods step OUT;
2) 0-AP-53.00 – Loss of Vital Instrumentation/Controls.

B. 1) Rods step IN;
2) 0-AP-53.00 – Loss of Vital Instrumentation/Controls.

C. 1) Rods step IN;
2) 1-AP-38.00 – Main Steam System Control Malfunction.

D. 1) Rods step OUT;
2) 1-AP-38.00 – Main Steam System Control Malfunction.

Proposed Answer:

D. 1) Rods step OUT;
2) 1-AP-38.00 – Main Steam System Control Malfunction.

Explanation: With the turbine in IMP OUT, an open PORV will cause Tave to lower enough that rods begin to step out in AUTO.

Technical Reference: 0-AP-53.00 – Loss of Vital Instrumentation/Controls, Rev 21.

1-AP-38.00, Main Steam System Control Malfunction, Rev 6

Reference Provided to Applicant: NO

Learning Objective: ND-89.1-LP-2 – Main Steam System; Objective F. Outline the causalities AP-38.00 (Main Steam System Control Malfunction) is designed to mitigate, and explain the type of actions taken in response to each failure.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.7 / 45.5 / 45.6)

Comments:

<u>KA Match Analysis</u>: The question matches the KA in that the operator must understand the reactivity effect of a steam leak/rupture and how the Rod Control System will respond. The operator must also have knowledge of which procedure would be used to mitigate the effects of this event.

<u>Distractor Analysis</u>:

A. Incorrect, but plausible if the candidate believes that AP-53.00 will isolate the PORV. AP-53.00 will transition the team to AP-38 for isolation.

B. Incorrect, but plausible if the candidate believes that rods will step in due to a power mismatch signal. This would be plausible if the turbine was in IMP IN. Also, the candidate may assume that AP-53.00 will perform the isolation.

C. Incorrect, but plausible if the candidate believes that rods will step in due to a power mismatch signal. This would be plausible if the turbine was in IMP IN.

D. CORRECT.

NUMBER	PROCEDURE TITLE	REVISION
0-AP-53.00	LOSS OF VITAL INSTRUMENTATION / CONTROLS	21
		PAGE 8 of 13

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If Reactor power has been affected by a secondary transient, Turbine adjustment may be needed to control Tave.

* *

10. ___ CHECK STEAM SYSTEM CONTROLS - NORMAL

- ☐ • Steam Dumps / PORVs
- ☐ • Turbine Controls
- ☐ • Turbine Valve Positions
- ☐ • Turbine Monitoring Lights

No

RNO: ☐ Initiate ( )-AP-38.00, MAIN STEAM SYSTEM CONTROL MALFUNCTION.

NUMBER	PROCEDURE TITLE	REVISION 6
1-AP-38.00	MAIN STEAM SYSTEM CONTROL MALFUNCTION	PAGE 3 of 7

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. ___	CHECK SG PORVS - CLOSED No →	☐ IF SG pressure greater than desired pressure, THEN check PORV(s) close when SG pressure lowers below desired pressure AND GO TO Step 3. IF SG pressure less than desired pressure, THEN do the following: ☐ a) Place affected PORV controller in Manual and close valve. b) IF any SG PORV NOT closed, THEN do either of the following: • In the Cable Vaults, on the Key Switch panel, place key switch for MS PRESS CONT VLV FIRE EMERG CLOSE to EMERG CLOSE position: ☐ • 1-MS-RV-101A ☐ • 1-MS-RV-101B ☐ • 1-MS-RV-101C OR • Locally close isolation valve(s): ☐ • SG A, 1-MS-86 ☐ • SG B, 1-MS-119 ☐ • SG C, 1-MS-157

ES 401-9 COMMENTS WITH CORRECTIONS SHOWN IN RED, 3/28/16

“KA Match Analysis” refers to loss of condenser vacuum that does not seem relevant here.

Changed KA Match Analysis.

K/A Number: 051AA1.04: Loss of Condenser Vacuum /4: Ability to operate and / or monitor the following as they apply to the Loss of Condenser Vacuum: Rod position.

Level: RO Tier #: 1 Group #: 2 IR – RO: 2.5 IR-SRO: 2.5

Proposed Question: 35

Initial Conditions:

- Unit 1 is at 100%.
- Condenser vacuum is 27 in and lowering.
- The team has entered 1-AP-14.00, Loss of Main Condenser Vacuum.

Current conditions:

- The turbine is being ramped down at 1% per minute IAW 0-AP-23.00, Rapid Load Reduction.
- Control rods are stepping in automatic.
- The Rod-to-Rod Deviation light on the CERPI display is lit.

Which of the following completes the statements below:

1) __________ contains guidance to adjust IRPI channels.
2) The CERPI Rod-to-Rod Deviation light will come on as soon as any rod indicates greater than _____ steps from another rod in the group.

A. 1) 1-AP-14.00 – Loss of Main Condenser Vacuum
   2) 12

B. 1) 0-AP-23.00, Rapid Load Reduction
   2) 20

C. 1) 0-AP-23.00, Rapid Load Reduction
   2) 12

D. 1) 1-AP-14.00 – Loss of Main Condenser Vacuum.
   2) 20

Proposed Answer:

C. 1) 0-AP-23.00, Rapid Load Reduction
   2) 12

Explanation: 1) With a slowly lowering vacuum, attempts to reduce load to compensate should continue per AP-23.00. 1-AP-14.00 will direct a rapid load reduction if Condenser heat sink is not normal. 0-AP-23.00 contains a note prior to step 6 to contact I & C to provide assistance with adjusting IRPIs. 2) The CERPI Rod-to-Rod Deviation light will come on once any rod is greater than 12 steps from any other rod in the group.

Technical Reference: 1-AP-14.00 – Loss of Main Condenser Vacuum, Rev 16;

0-AP-23.00 – Rapid Load Reduction, 41; ND-93.3-LP-4, CERPI System, Rev. 10.

Reference Provided to Applicant: No

Learning Objective: ND-95.1-LP-6 – Loss of Main Condenser Vacuum; Objective D. Assess the expected sequence of events and the operator actions associated with a loss of Main Condenser Vacuum. ND-93.3-LP-4, CERPI System; Objective F. Explain the overall integrated operations of the Computer Enhanced Rod Position Indication System, including its purpose, location, readouts, and component operation.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.7 / 45.5 / 45.6)

Comments:

<u>KA Match Analysis</u>: Question matches the KA in that the candidate must monitor Rod Position Indication during a Loss of Condenser Vacuum and subsequent ramp, have knowledge of the different indications that provide information to the operator, and take the appropriate action.

<u>Distractor Analysis</u>:

A. Incorrect. 1) Incorrect, AP-23.00 contains a Note stating to the operator that I&C should be contacted as necessary to adjust IRPIs. Plausible because 1-AP-14.00 *directs* the operator to ramp per AP-23.00. 2) Correct.
B. Incorrect. 1) Correct. 2) Incorrect, the rod-to-rod deviation illuminates <u>*as soon as*</u> there is a 12 step deviation between rods. Plausible because there are other alarm light indications that provide status of IRPIs such as Any rod on bottom which comes on at 20 steps.
C. CORRECT.
D. Incorrect. 1) Incorrect, AP-23.00 contains a Note stating to the operator that I&C should be contacted as necessary to adjust IRPIs. Plausible because 1-AP-14.00 *directs* the operator to ramp per AP-23.00. 2) Incorrect, the rod-to-rod deviation illuminates <u>*as soon as*</u> there is a 12 step deviation between rods. Plausible because there are other alarm light indications that provide status of IRPIs such as Any rod on bottom which comes on at 20 steps.

Reference: 1-AP-14.00. Shows step to ramp per AP-23.00.

NUMBER	PROCEDURE TITLE	REVISION
1-AP-14.00	LOSS OF MAIN CONDENSER VACUUM	16
		PAGE 6 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. ___	CHECK GLAND STEAM PRESSURE - BETWEEN 5 PSIG AND 12 PSIG	☐ Adjust GS supply or isolate Spillover, if affecting Condenser vacuum performance.
14. ___	CHECK CONDENSER HEAT SINK - NORMAL	Do the following:
	☐ • All Waterboxes fully in service	☐ a) Raise flow through waterboxes, as required.
	☐ • Intake canal level trend	☐ b) <u>IF</u> heat removal is insufficient, <u>THEN</u> initiate 0-AP-23.00, RAPID LOAD REDUCTION.
	☐ • Conditions at High Level Intake	
	☐ • Waterbox differential temperature No	
	☐ • Hotwell level	
	☐ • Station Vacuum Priming	

Reference 0-AP-23.00, Rapid Load Reduction. Shows note to contact I&C for IRPI adjustment.

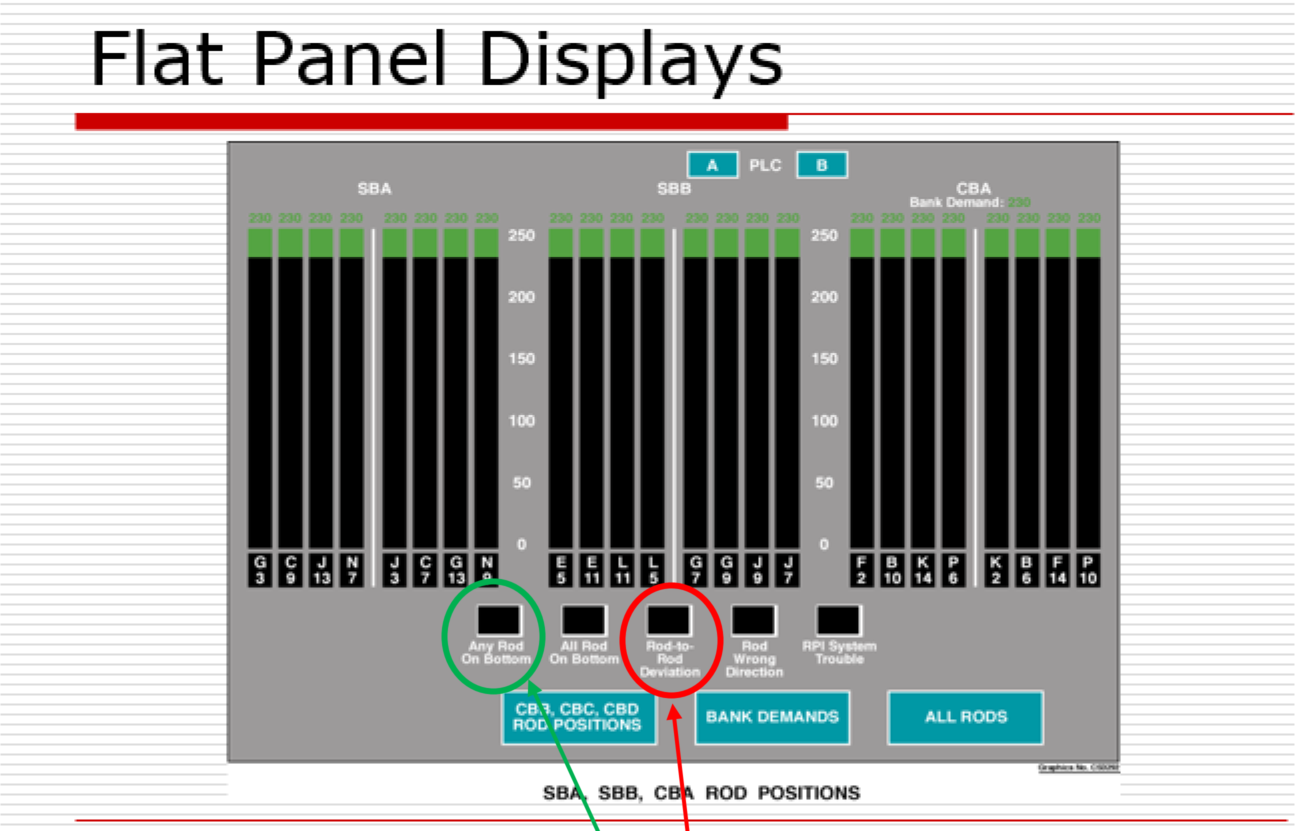
NUMBER	PROCEDURE TITLE	REVISION
0-AP-23.00	RAPID LOAD REDUCTION	41
		PAGE 5 of 18

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- If at any time plant conditions no longer require rapid load reduction, actions should continue at Step 35.
- RCS Tave must be maintained less than or equal to 577°F and RCS pressure must be maintained greater than or equal to 2205 psig. Tech Spec 3.12.F.1 should be reviewed if either parameter is exceeded.
- I & C should be contacted to provide assistance with adjusting IRPIs.

6. ___ CONTROL RAMP RATE TO MAINTAIN RCS PRESSURE GREATER THAN 2205 PSIG

Reference ND-93.3-LP-4, slide 9. Shows status lights for IRPI.



a. One screen is SBA, SBB, CBA positions. This screen shows the position and number of each of the rods in these banks. Also, at the top the PLC which is being used is display with a selection button to go to the other PLC. At the bottom the operator can select the other three screens. Also on the bottom are the following five alarms:

- Any Rod on the Bottom- any rod less than 20 steps
- All Rods on the Bottom
- Rod-To-Rod- Deviation any rod greater than 12 steps from another in group
- Rod Wrong Direction rods moving in opposite demand direction
- RPI System Trouble computer monitors for trouble. Alarm will be discussed in further detail later in lesson plan.

K/A Number: 054AA2.03: Loss of Main Feedwater / 4: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Conditions and reasons for AFW pump startup.

Level: RO Tier #: 1 Group #: 1 IR – RO: 4.1 IR-SRO: 4.2

Proposed Question: 36

Initial conditions:

- A plant startup was in progress.
- Power level was at **35%**.

Current conditions:

- An automatic reactor trip has occurred.
- SG blowdown isolation valves (1-BD-TV-100A through -100F) are closed.
- Current SG narrow range levels (all slowly rising):
  - A – 15%
  - B – 16%
  - C – 18%.

Which ONE of the following states:

(1) The initiating event that caused the trip.

(2) The expected automatic actions based on these conditions.

A. 1) The operating MFP tripped;
2) ONLY the motor driven AFW pumps have a start signal.

B. 1) The operating MFP tripped;
2) BOTH the turbine driven AND motor driven AFW pumps have a start signal.

C. 1) All feed water regulating valves drifted closed, actuating AMSAC;
2) BOTH the turbine driven AND motor driven AFW pumps have a start signal.

D. 1) All feed water regulating valves drifted closed, actuating AMSAC;
2) ONLY the motor driven AFW pumps have a start signal.

Proposed Answer:

B. 1) The operating MFP tripped;
2) BOTH the turbine driven AND motor driven AFW pumps have a start signal.

Explanation: The only thing that could have happened to cause the Reactor trip was a trip of the running MFP. Coupled with the indications that all BD trip valves are closed, this indicates an AFW signal is present. Since 2/3 S/G NR levels are <17%, all AFW pumps are expected to start. AMSAC is operationally blocked <37%.

Technical Reference: SPS UFSAR Chapter 10, Rev 47

Reference Provided to Applicant: No

Learning Objective: ND-95.1-LP-4, Loss of Feedwater; Objective B. Describe the plant response to a complete loss of feedwater.

Question Source: Modified Bank (VC Summer January 2006 - #11; modified stem/distractors)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43.5 / 45.13)

Comments:

KA Match Analysis: Question matches KA in that the candidate must interpret given unit conditions to determine the actuating event for AFW pump start.

Distractor Analysis:

A. Incorrect, but plausible if candidate misses the S/G levels and relies on the 1-of-2 MFP breakers open auto-start of the MDAFWPs.
B. CORRECT.
C. Incorrect, but plausible if candidate assumes that AMSAC is available at this power level. An AMSAC initiation will give and AFW auto-start signal, resulting in all AFW pumps starting and BD TVs closing.
D. Incorrect, but plausible if candidate assumes AMSAC is available at this power level and relies on the 1-of-2 MFP breakers open auto-start of the MDAFWPs.

Parent Question: VC Summer January 2006 - #11

The following conditions exist:

- *A plant startup was in progress.*
- *Power level was at 38%*
- *The reactor tripped*
- *SG blowdown isolation valves (PVG-503A(B)(C), A(B)(C) ISOL) closed*
- *Current SG narrow range levels in "A", "B", and "C" SGs are 8%, 10%, and 10%, respectively, and decreasing*

Which ONE of the following correctly states the initiating event that caused the trip and the expected automatic actions based on these conditions?

A. *The operating MFP tripped and ONLY the motor driven EFW pumps have a current start signal.*

B. *The operating MFP tripped and BOTH the turbine driven AND motor driven EFW pumps have a current start signal.*

C. *All SG flow control valves drifted closed and AMSAC should have actuated.*

D. *All SG flow control valves drifted closed and ONLY the turbine driven EFW pump has a current start signal.*

Answer: B

and associated piping are installed in a tornado-protected area adjacent to the containment so that their use can be relied upon during any loss-of-station power accident.

The automatic initiation signals and circuits for the auxiliary feedwater system comply with the single-failure criterion of IEEE Standard 279-1971, with exceptions. The following signals are used for automatic initiation of the auxiliary feedwater system:

1. Turbine-driven auxiliary feed pump
   a. Low-low steam generator level (two out of three)
   b. Undervoltage on the reactor coolant pump buses (two out of three)
   c. AMSAC initiation
2. Motor-driven auxiliary feed pumps
   a. Low-low level from any one steam generator
   b. Loss of reserve station power (station blackout)
   c. Trip of both main feedwater pumps
   d. Safety injection
   e. AMSAC initiation

K/A Number: 055EA2.03: Station Blackout / 6: Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.9 IR-SRO: 4.7

Proposed Question: 37

The following conditions exist:

- An earthquake has occurred, causing a complete loss of offsite power.
- Both Units have Safety Injected.

Placing the __1)__ #3 EDG bypass (Giveaway) switch in BYPASS will energize the __2)__ bus.

A. 1) Unit 2
2) 2H

B. 1) Unit 2
2) 1J

C. 1) Unit 1
2) 1J

D. 1) Unit 1
2) 2H

Proposed Answer:

B. 1) Unit 2
2) 1J

Explanation: The bypass switches on the respective Units' #3 EDG panel acts as a "give-away" switch to override the interlock that loads the EDG on an accident unit. With both units in an accident situation (SI), the EDG will be running unloaded. By placing the Unit 2 bypass/give-away switch in BYPASS, the #3 EDG will load onto Unit 1 "J" bus.

Technical Reference: 1-ECA-0.0 – Loss of All AC Power, rev 40

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-17 – ECA-0.0, Loss of All AC Power; Objective D. Given actual or simulated plant conditions requiring implementation of ECA-0.0, Loss of All AC Power, successfully transition through the procedure, applying step background knowledge as required, to safely place the plant in the required optimal recovery condition.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 43.5 / 45.13)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must choose the correct switch to restore power to an emergency bus.

Distractor Analysis:

A. Incorrect, but plausible if the candidate confuses the capability of being powered by the AAC Diesel Generator with #3 EDG.
B. CORRECT.
C. Incorrect, but plausible since the switch and bus are same-unit. However, the switch acts as a “give-away” switch, so the other unit takes the EDG.
D. Incorrect, but plausible if the candidate confuses the capability of being powered by the AAC Diesel Generator with #3 EDG.

NUMBER	PROCEDURE TITLE	REVISION
1-ECA-0.0	LOSS OF ALL AC POWER	40
		PAGE 7 of 30

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

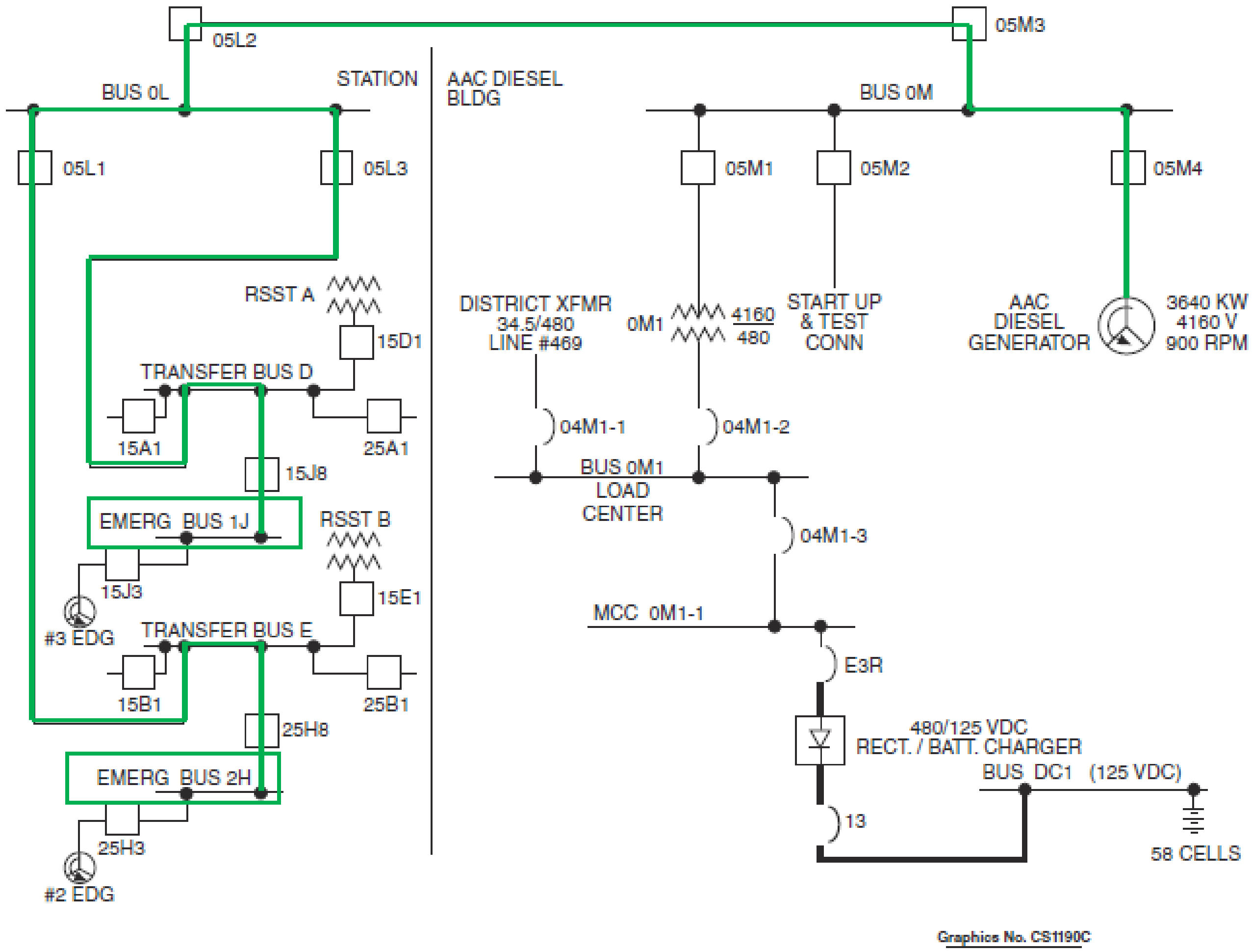
5. TRY TO RESTORE POWER TO ANY AC EMERGENCY BUS (Continued)

3) IF both buses NOT energized, AND desired to transfer EDG 3 to Bus 1J, THEN do the following:

- ☐ a. Place 25J3 in PTL.
- ☐ b. Place UNIT 2 PNL 3-2 SWITCH 43-15J3 IN BYP COMMITS NO. 3 EDG TO UNIT 1 in BYP. (Key No. 19)
- c. IF ACB 15J3 is NOT closed, THEN do the following.
  - ☐ 1. Turn SYNCH ACB-15J3 to ON.
  - ☐ 2. Close ACB-15J3.
  - ☐ 3. Turn SYNCH ACB-15J3 to OFF.

☐ 4) Stop EDG in EXERCISE for any unloaded EDG.

☐ 5) IF both Unit 1 Emergency Buses still deenergized, THEN initiate 0-AP-17.06, AAC DIESEL GENERATOR - EMERGENCY OPERATIONS, AND GO TO Step 6.



SURRY POWER STATION
STATION BLACKOUT EDG DISTRIBUTION

K/A Number: 056AK1.01: Loss of Off-site Power / 6: Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Principle of cooling by natural convection.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.7 IR-SRO: 4.2

Proposed Question: 38

Given the following:

- The plant trips from full power following a loss of offsite power.
- The crew has entered 1-AP-39.00, Natural Circulation of RCS.

In accordance with 1-AP-39.00, which ONE of the following indicates that natural circulation is in progress?

A. RCS Subcooling indication stable or lowering.

B. Steam Generator water levels stable or rising.

C. RCS cold leg temperature stable or rising.

D. Steam Generator pressures stable or lowering.

Proposed Answer:

D. Steam Generator pressures stable or lowering.

Explanation: Several indications are listed in 1-AP-39.00 to verify NC flow is occurring. Subcooling – >30°F; CETCs – stable or lowering; S/G pressures – stable or lowering; T_{hot} – stable or lowering; T_{cold} – at saturation temp for S/G pressure.

Technical Reference: 1-AP-39.00, Natural Circulation of RCS, Rev 7.

Reference Provided to Applicant: No

Learning Objective: ND-86.3-LP-4, Natural Circulation; Objective C. State four (4) indications available to the reactor operator for verifying the existence of natural circulation including an explanation of how these indications respond to a loss of natural circulation.

Question Source: Bank (Farley 8/2004; #40)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.8 / 41.10 / 45.3)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must recognize indications that NC flow is occurring.

Distractor Analysis:

A. Incorrect, but plausible if candidate believes lowering subcooling indicates good NC flow.
B. Incorrect, but plausible if candidate believes that SG levels rising is a good indication of NC flow occurring.
C. Incorrect, but plausible if candidate assumes T_{cold} temperatures rising is an indication of adequate NC flow.
D. CORRECT.

Farley 8/2004; #40

The plant was operating at 10% Reactor power when a Loss of Off-Site Power caused the RCP's to trip. All core-exit thermocouples are operable. ESP-0.2, Natural Circulation Cooldown to Prevent Reactor Vessel Head Steam Voiding, is being performed.

Which one of the following describes the indication(s) that could be used to verify natural circulation cooling is adequate IAW ESP-0.2?

A.✓ Steam Generator pressures stable or falling.

B. RCS cold leg temperatures stable or rising.

C. Steam Generator water levels stable or rising.

D. Sub Cooled Margin Monitor indication stable or falling.

Reference: 1-AP-39.00, Natural Circulation of RCS.

NUMBER 1-AP-39.00	ATTACHMENT TITLE NATURAL CIRCULATION INFORMATION	ATTACHMENT 1
REVISION 7		PAGE 1 of 1

NOTE: Saturation conditions may be checked using Attachment 2.

NATURAL CIRCULATION VERIFICATION

- ☐ • RCS subcooling based on CETCs - GREATER THAN 30°F.
- ☐ • CETCs - STABLE OR SLOWLY LOWERING.
- ☐ • SG pressure - STABLE OR SLOWLY LOWERING.
- ☐ • WR Hot Leg temperatures - STABLE OR SLOWLY LOWERING.
- ☐ • WR Cold Leg temperatures - AT SATURATION FOR SG PRESSURES.

K/A Number: 058AK3.01: Loss of DC Power 6: Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by D/Gs.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.4 IR-SRO: 3.7

Proposed Question: 39

A loss of EDG DC Control Power will cause a loss of ____.

A. immersion heater control

B. the turbo soak back pump

C. field flash control

D. louver control

Proposed Answer:

C. field flash control

Explanation: EDG control power supplies the initial field flash for the EDG. Therefore a loss of EDG DC control will (among other items) cause a loss of field flashing for the EDG.

Technical Reference: ND-90.3-LP-1, EDG. Rev. 25.

Reference Provided to Applicant: No

Learning Objective: ND-90.3-LP-1, EDG Objective D; Describe the power sources required for EDG operation to include:

- 480v MCC power sources and associated loads
- 125v DC power sources and associated loads
- Generator exciter output and loads

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR 41.5,41.10 / 45.6 / 45.1)

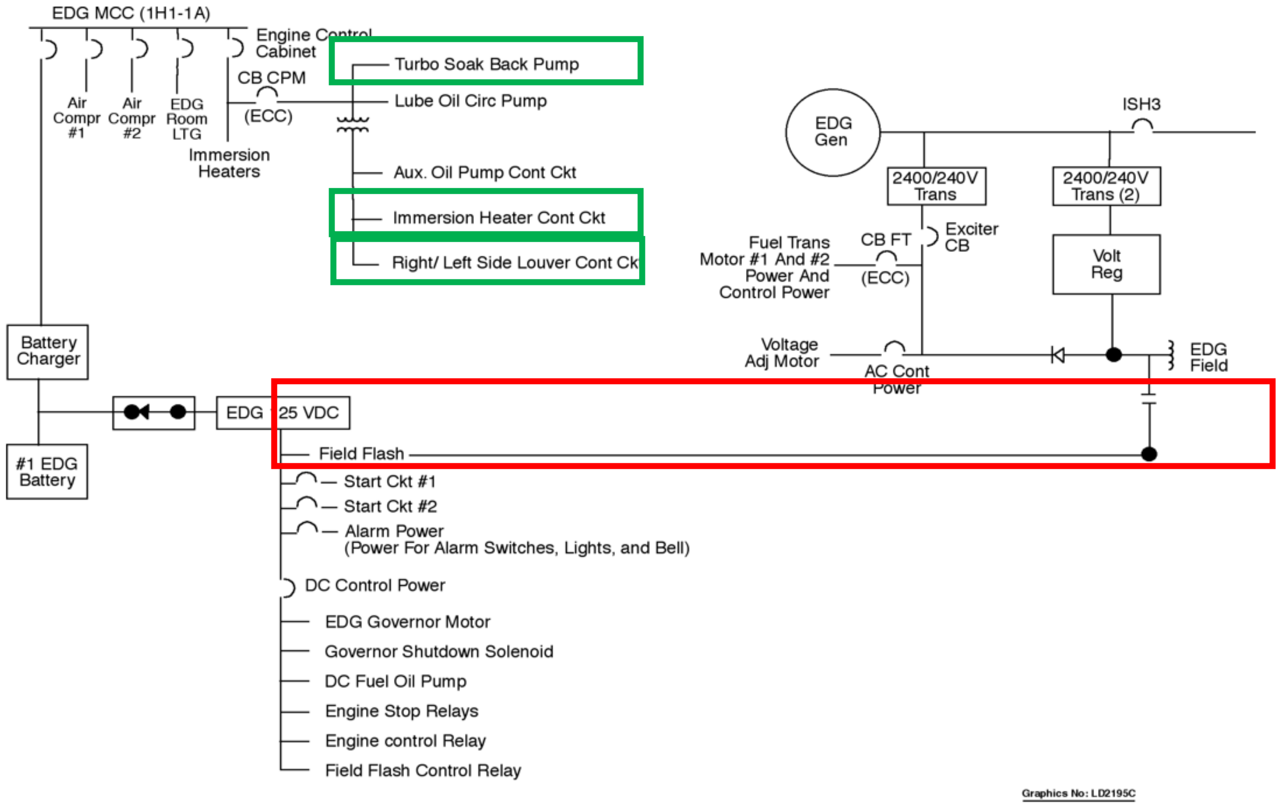
Comments:

KA Match Analysis: The question meets the KA in that the candidate must understand how the loss of a DC bus affects EDG operation.

Distractor Analysis:

A. Incorrect, because immersion heater control is powered from the EDG MCC, 1H1-1A. Plausible if the candidate confuses EDG DC control power with EDG 480V MCC.
B. Incorrect, because the turbo soak back pump is powered from the EDG MCC, 1H1-1A. Plausible if the candidate confuses EDG DC control power with EDG 480V MCC.
C. CORRECT.
D. Incorrect, because louver control is powered from the EDG MCC, 1H1-1A. Plausible if the candidate confuses EDG DC control power with EDG 480V MCC.

Reference: ND-90.3-LP-1, EDG, FIG. ND-90.3-H/T-1.15, EMERG GEN POWER SUPPLIES



EMERGENCY GENERATOR POWER SUPPLY DIAGRAM

K/A Number: 059AA1 .02: Accidental Liquid RadWaste Rel. / 9: Ability to operate and / or monitor the following as they apply to the Accidental Liquid Radwaste Release: ARM system.

Level: RO Tier #: 1 Group #: 2 IR – RO: 3.3 IR-SRO: 3.4

Proposed Question: 40

Given the following conditions:

- Chemistry has requested to sample RCS Hot Legs on Unit 1.
- You have just opened 1-SS-TV-106A/B, RC HOT LEG I/S and O/S TVs.
- The AUX BLDG SAMPLE Radiation Monitor 1-RM-156 alarms in ALERT, rapidly followed by HIGH.
- The Chemistry Technician reports a broken sample line in the Sample Room.

What is your Immediate Operator Action in accordance with 0-AP-35.01, Radioactive Liquid Release Control?

A. Notify Health Physics to evacuate the Auxiliary Building.

B. Close 1-SS-TV-106A/B to stop the release.

C. Have the Chemistry Technician quantify leak rate.

D. Secure Auxiliary Building Central ventilation system.

Proposed Answer:

B. Close 1-SS-TV-106A/B to stop the release.

Explanation: The Immediate Action of 0-AP-35.01, Radioactive Liquid Release Control is STOP THE RELEASE. Quantifying the leak rate, notifying HP and possibly securing ventilation are secondary actions.

Technical Reference: 0-AP-35.01 - Radioactive Liquid Release Control, rev 3.

Reference Provided to Applicant: No

Learning Objective: ND-93.4-LP-12- Abnormal Procedures; Objective C. Given a set of plant conditions, determine the appropriate operator response in accordance with 0-AP-35.00, Oil and Hazardous Substance Release, 0-AP-35.01, Radioactive Liquid Release Control, 0-AP-35.02, Response to Potential Toxic, Corrosive, Asphyxiant, or Flammable Atmosphere, and 0-AP-12.03, Response to Oil Spill in the James River, to include the following:

- Immediate Actions (if applicable)
- Entry Conditions
- Major Actions
- Step Bases

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.7 / 45.5 / 45.6)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must use the ARM indication to determine that there is a liquid release in the Sample Room.

Distractor Analysis:

A. Incorrect, but plausible since the ARP for 1-RM-RI-156 High, RM-F4 directs evacuation of the affected area and HP notifications. HP notifications are also performed in 0-AP-35.01, but after the IAs.
B. CORRECT.
C. Incorrect, but plausible since 0-AP-35.01 will require a volume released or a release rate. These actions are performed after the immediate action of stopping the leak.
D. Incorrect, but plausible if the candidate believes that securing the area ventilation to stop the release is a prudent act. It may be, but only after the liquid release itself has been isolated or otherwise stopped.

NUMBER	PROCEDURE TITLE	REVISION
0-AP-35.01	RADIOACTIVE LIQUID RELEASE CONTROL	3
		PAGE 2 of 3

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Any indication which can aid in determining the volume or rate of the release should be closely monitored.

[1] ___	STOP THE RELEASE	☐ IF release can NOT be stopped, THEN reduce and contain the release using any suitable method.
2. ___	TAKE ACTION TO CONTAIN THE RELEASE	
3. ___	DIRECT HEALTH PHYSICS TO INITIATE THE FOLLOWING PROCEDURES: ☐ • HP-3010.023, UNPLANNED LIQUID RELEASE ☐ • HP-3051.020, GROUNDWATER PROTECTION PROGRAM	
4. ___	CHECK RELEASE VOLUME - DETERMINED	Do the following. ☐ a) Initiate Attachment 1. ☐ b) GO TO Step 6.
5. ___	RECORD RELEASE VOLUME _______________	

NUMBER	PROCEDURE TITLE	REVISION
0-RM-F4	1-RM-RI-156 HIGH	3
		PAGE 2 of 3

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK ALARM - READING ON MONITOR OR CHART RECORDER GREATER THAN OR EQUAL TO HIGH SETPOINT ☐ • 1-RM-RI-156 ☐ • 1-RM-RR-175B, Pen 2	Do the following: ☐ a) Evaluate entry into 0-AP-10.13, LOSS OF MAIN CONTROL ROOM ANNUNCIATORS. ☐ b) Initiate a Condition Report. ☐ c) GO TO Step 6.
2.___	EVACUATE AFFECTED AREA	
3.___	CHECK PRIMARY SAMPLING - IN PROGRESS	☐ GO TO Step 5.
4.___	CLOSE ASSOCIATED SAMPLE SYSTEM TRIP VALVES	

K/A Number: 059K4.18: Main Feedwater: Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: Automatic feedwater reduction on plant trip.

Level: RO Tier #: 2 Group #: 1 IR – RO: 2.8 IR-SRO: 3.0

Proposed Question: 41

Which ONE of the following conditions would prevent the Main Feedwater Regulating Valves from closing during a Reactor Trip?

A. A failure of Median T_{ave} such that it remains at 573°F.

B. One Reactor Trip Breaker remains closed.

C. Two-of-three S/G narrow range levels in two-of-three S/G's remain >17%.

D. AMSAC fails to initiate.

Proposed Answer:

A. A failure of Median T_{ave} such that it remains at 573°F.

Explanation: Automatic closure of the MFRVs occurs with reactor trip breakers open and Median Tave <554°F.

Technical Reference: Technical Specifications Table 4.1-A

0-DRP-004 –Precautions, Limitations, and Setpoints; rev 76

SPS UFSAR Chapter 10 – Steam and Power Conversion, rev 47 (*used as distractor*)

Reference Provided to Applicant: No

Learning Objective: ND-93.3-LP-16, Permissive/Bypass/Trip Status Lights; Objective A. When given the permissive number (i.e., P-1, P-2), explain the function of the permissive including the setpoints, and coincidence.

Question Source: Modified Bank (Ft. Calhoun 2005 - #45; modified stem/distractors)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must know the auto-closure signal for the MFRVs on a reactor trip (P-4 interlock).

Distractor Analysis:

A. CORRECT.
B. Incorrect, but plausible if candidate believes that both RTBs must be open to satisfy P-4.
C. Incorrect, but plausible if candidate confuses AFW auto start signal with FRV close signal.
D. Incorrect, but plausible if candidate confuses AFW auto start signal with FRV close signal.

Ft. Calhoun 2005 - #45; modified stem/distractors

Which one of the following conditions would prevent the feedwater regulating valves from ramping down following a reactor trip from full power?

A. The steam dump valve AUTO/INHIBIT switch has been placed in INHIBIT.
B. An Auxiliary Feedwater Actuation Signal (AFAS)
C. Feedwater Override switches have been placed in OVERRIDE
D. The FW/43 switch is placed in the OFF position

Answer:

A. The steam dump valve AUTO/INHIBIT switch has been placed in INHIBIT.

Reference: Tech Specs Table 4.1-A

TABLE 4.1-A
REACTOR TRIP SYSTEM AND ENGINEERED SAFEGUARDS ACTION INTERLOCKS

DESIGNATION	CONDITION	FUNCTION
Reactor Trip (P-4)	1 of 2 breakers open	Reactor tripped - actuates turbine trip, allows auto closing of main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal.

DOMINION
Surry Power Station

0-DRP-004
Revision 76
Page 59 of 85

(Page 9 of 19)
Attachment 2
REACTOR CONTROL SYSTEM SETPOINTS

5. Feedwater Control

A. High Level Override for Feedwater Valve Closure, Turbine Trip, and Feedwater pump trip
(LC-474, LC-475, LC-476)
(LC-484, LC-485, LC-486) 75% of level span
(LC-494, LC-495, LC-496)

B. Low T_{ave}, Reactor Trip Override for Feedwater Valve Closure (TC-408K)
valve closure on low Tave 554°F

and associated piping are installed in a tornado-protected area adjacent to the containment so that their use can be relied upon during any loss-of-station power accident.

The automatic initiation signals and circuits for the auxiliary feedwater system comply with the single-failure criterion of IEEE Standard 279-1971, with exceptions. The following signals are used for automatic initiation of the auxiliary feedwater system:

1. Turbine-driven auxiliary feed pump
   a. Low-low steam generator level (two out of three)
   b. Undervoltage on the reactor coolant pump buses (two out of three)
   c. AMSAC initiation
2. Motor-driven auxiliary feed pumps
   a. Low-low level from any one steam generator
   b. Loss of reserve station power (station blackout)
   c. Trip of both main feedwater pumps
   d. Safety injection
   e. AMSAC initiation

The steam generator level signals and the input signals from the safety injection system are both redundant and independent. Undervoltage on the reactor coolant pump buses, main feed pump breaker trip, and loss of reserve station power are considered operational signals for economic (non-public safety) protection and are therefore not required to meet the single failure criterion of IEEE Standard 279-1971.

The AMSAC signal is provided as a means, diverse from the reactor protection system, to automatically initiate the auxiliary feedwater system. This back-up signal was provided in accordance with the requirements of 10 CFR 50.62. The AMSAC logic circuit power supplies are normally powered from non-safety related sources independent of the RPS and are capable of operating on a loss of offsite power. They can be powered from EDG #1 (Unit 1) and EDG #2 (Unit 2) by manual action. (Section 7.2.3.2.7)

K/A Number: 061K2.02: Auxiliary/Emergency Feedwater: Knowledge of bus power supplies to the following: AFW electric drive pumps.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.7 IR-SRO: 3.7

Proposed Question: 42

What is the breaker id that supplies power to Aux Feedwater pump, 1-FW-P-3B?

A. 1-EP-BKR-14H5

B. 1-EP-BKR-15H4

C. 1-EP-BKR-15J4

D. 1-EP-BKR-14J5

Proposed Answer:

C. 1-EP-BKR-15J4

Explanation: 1-FW-P-3B is powered from 1J 4160V bus, breaker 15J4.

Technical Reference: Station Drawing 11448-FE-1A1 – Unit 2 4160V, rev 29

Reference Provided to Applicant: No

Learning Objective: ND-89.3-LP-4 – Auxiliary Feedwater; Objective B. B. [Describe the design features and operation of the following AFW system components:

- Emergency Condensate Storage Tank (1-CN-TK-1)
- Emergency Makeup Tank (1-CN-TK-3)
- Auxiliary Feedwater Pumps (1-FW-P-2, 3A/B)
- AFW Booster Pumps
- AFW Crosstie
- MOV-FW-151A through F

- Stop Check Valves
- Flow Transmitters
- Cavitating Venturis
- Radiation Monitoring

SOER 82-08, Recommendation 4]

Question Source: Bank (Point Beach 2005, Question #46)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must know the power supplies to the MDAFW pump.

Distractor Analysis:

A. Incorrect, but plausible if candidate confuses H and J buses as well as what is powered from 480V and 4160V.
B. Incorrect, but plausible if candidate confuses H and J buses.
C. CORRECT.
D. Incorrect, but plausible if candidate confuses what is powered from 480V and 4160V.

Parent Question; Point Beach 2005, Question #46

46. 2005 ILT RO 46
Which of the following is the power supply to P-38B, Motor-Driven AFW pump?

A. 1B-03

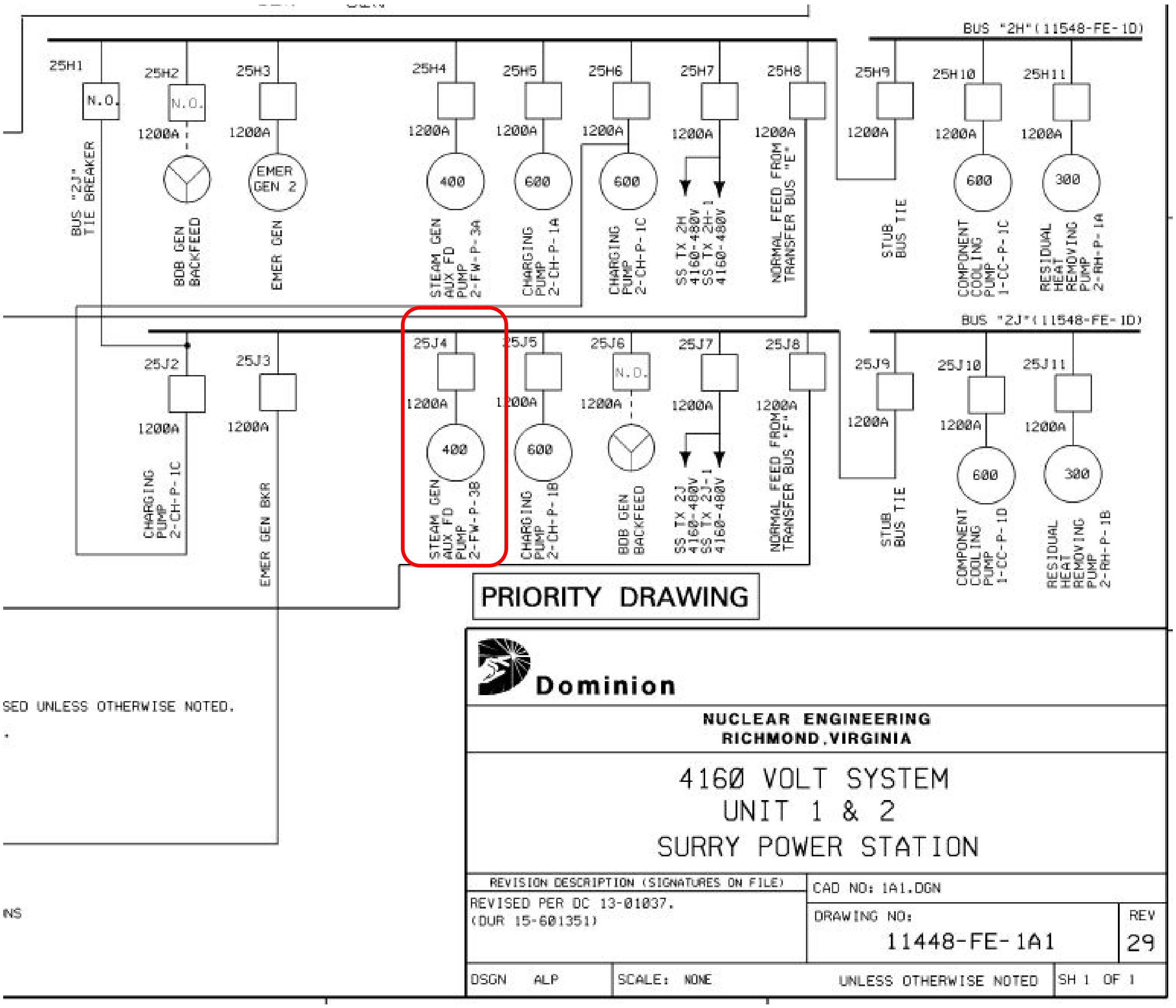
B. 1B-04

C. 2B-03

D. 2B-04

Answer: D

KA#
061.K2.02


BUS "2H" (11548-FE-1D)
25H1
N.O.
BUS "2J" TIE BREAKER
25H2
N.O.
1200A
BDB GEN BACKFEED
25H3
1200A
EMER GEN 2
EMER GEN
25H4
1200A
400
STEAM GEN AUX FD PUMP 2-FW-P-3A
25H5
1200A
600
CHARGING PUMP 2-CH-P-1A
25H6
1200A
600
CHARGING PUMP 2-CH-P-1C
25H7
1200A
SS TX 2H 4160-480V
SS TX 2H-1 4160-480V
25H8
1200A
NORMAL FEED FROM TRANSFER BUS "E"
25H9
1200A
STUB BUS TIE
25H10
1200A
600
COMPONENT COOLING PUMP 1-CC-P-1C
25H11
1200A
300
RESIDUAL HEAT REMOVING PUMP 2-RH-P-1A
BUS "2J" (11548-FE-1D)
25J2
1200A
CHARGING PUMP 2-CH-P-1C
25J3
1200A
EMER GEN BKR
25J4
1200A
400
STEAM GEN AUX FD PUMP 2-FW-P-3B
25J5
1200A
600
CHARGING PUMP 2-CH-P-1B
25J6
N.O.
1200A
BDB GEN BACKFEED
25J7
1200A
SS TX 2J 4160-480V
SS TX 2J-1 4160-480V
25J8
1200A
NORMAL FEED FROM TRANSFER BUS "F"
25J9
1200A
STUB BUS TIE
25J10
1200A
600
COMPONENT COOLING PUMP 1-CC-P-1D
25J11
1200A
300
RESIDUAL HEAT REMOVING PUMP 2-RH-P-1B
PRIORITY DRAWING
SED UNLESS OTHERWISE NOTED.
NS
Dominion
NUCLEAR ENGINEERING
RICHMOND, VIRGINIA
4160 VOLT SYSTEM
UNIT 1 & 2
SURRY POWER STATION
REVISION DESCRIPTION (SIGNATURES ON FILE)
REVISED PER DC 13-01037.
(DUR 15-601351)
CAD NO: 1A1.DGN
DRAWING NO:
11448-FE-1A1
REV
29
DSGN ALP
SCALE: NONE
UNLESS OTHERWISE NOTED
SH 1 OF 1

K/A Number: 062A3.01: AC Electrical Distribution: Ability to monitor automatic operation of the ac distribution system, including: Vital ac bus amperage.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.0 IR-SRO: 3.1

Proposed Question: 43

Given the following:

- 1K-A8, UPS SYSTEM 1A TROUBLE is LIT.
- An Operator has been dispatched in accordance with the ARP.
    - Normal and alternate 480V supply breakers are closed.
    - At UPS 1A1, the Inverter Output Low and Fuse Blown yellow lights are lit.

Which ONE of the choices below correctly completes the following statement?

Vital Bus 1-I is currently being supplied from the __1)__ and vital bus amperage indication is located on __2)__.

A. 1) Regulating Line Conditioner
   2) Main Control Room ammeter

B. 1) Station Battery 1A
   2) UPS 1A1 ammeter

C. 1) Station Battery 1A
   2) Main Control Room ammeter

D. 1) Regulating Line Conditioner
   2) UPS 1A1 ammeter

Proposed Answer:

D. 1) Regulating Line Conditioner
   2) UPS 1A1 ammeter

Explanation: A failure of the UPS Inverter, but not the battery charger will cause the UPS system trouble annunciator, but not the Battery system trouble annunciator. This is due to the alternate source picking up the load on the Vital Bus associated with the affected UPS. The static switch will seamlessly supply the VB via the Alternate Source (RLC).

Technical Reference: ARP 1K-A7 – BATT SYSTEM 1A TROUBLE, rev 8

ARP 1K-A8 – UPS SYSTEM 1A TROUBLE, rev 6

Reference Provided to Applicant: No

Learning Objective: ND-90.3-LP-5 – Vital and Semi-Vital Bus Distribution; Objective B. [Describe the components and indications associated with an Uninterruptable Power Supply (UPS). SOER 83-03, Recommendation 11].

Question Source: Modified Bank (North Anna 2014 #46)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7 / 45.5)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must understand how the UPS operates to automatically maintain VB voltage, frequency, and current. The candidate must also know where VB current is monitored.

Distractor Analysis:

A. Incorrect, but plausible if candidate believes MCR meters indicate VB amperage. MCR meters for the vital buses monitor voltage, not amperage. VB amperage is only available locally at the UPS.
B. Incorrect, but plausible if candidate assumes that since the battery charger section is working (normal & alternate 480v supply breakers closed) that the battery is supplying the load.
C. Incorrect, but plausible if candidate believes MCR meters indicate VB amperage. MCR meters for the vital buses monitor voltage, not amperage. VB amperage is only available locally at the UPS. Also, if candidate may assume that since the battery charger section is working (normal & alternate 480v supply breakers closed) that the battery is supplying the load.
D. CORRECT.

Parent Question: North Anna 2014 #46

46. 46 - 062A3.01 001/NEW/N/A/HIGH/2/3.0/3.1//

Given the following:

- Battery Charger 1-I has failed due to an overcurrent fault
- 1-AR-H-B1, Battery Chgr 1-I Trouble is LIT
- 1-AR-H-A1, Vital Bus 1-I Invert Troublp, is NOT LIT
- In accordance with the Annuciator Response Procedure an operator has been dispatched
  - At 1-VB-INV-01, Vital Bus Distribution Panel 1-I Inverter, "Low AC Output Voltage" is NOT LIT

Which ONE of the choices below completes the following statement?

Vital AC Bus 1-I is currently being supplied from___(1)____ and the vital bus indications will be verified normal by checking__(2)___.

A. (1) Voltage Regulating Transformer 1-I
(2) Amps in the Main Control Room

B. (1) Voltage Regulating Transformer 1-I
(2) Amps at the Inverter Panel, 1-VB-INV-01

C✓ (1) Station Battery 1-I
(2) Amps at the Inverter Panel, 1-VB-INV-01

D. (1) Station Battery 1-I
(2) Amps in the Main Control Room

Reference; 1K-A8

NUMBER	PROCEDURE TITLE	REVISION
1K-A8	UPS SYSTEM 1A TROUBLE	6
		PAGE 1 of 5

REFERENCES 1K-8

1) UFSAR 8.0
2) Tech Spec 3.16
3) 11448-FE-1A2
4) NRC Generic Letter 91-11
5) 1-DRP-001, Vital Bus Breaker List
6) DR S-2000-0454, Inadvertent UPS Power Supply Swapping
7) PI-S-2004-2430, UPS Inverter Clock

PROBABLE CAUSE

1) Alarm actuates when inverter output voltage is less than 116.4 VAC or greater than 123.6 VAC.
2) Alarm actuates when the Static Switch is on Alternate Source.
3) Alarm actuates when Alternate Source voltage is less than 114.0 VAC or greater than 126.0 VAC.
4) UPS trouble may be caused by one or more of the following:
   - Blown fuse on inverter, static switch or battery charger
   - Loss of inverter
   - Inverter output voltage drift
   - Supply breaker trip
   - Regulating transformer output voltage drift
   - Testing or maintenance in progress

K/A Number: 062AA2.03: Loss of Nuclear Svc Water / 4: Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The valve lineups necessary to restart the SWS while bypassing the portion of the system causing the abnormal condition.

Level: RO Tier #: 1 Group #: 1 IR – RO: 2.6 IR-SRO: 2.9

Proposed Question: 44

Initial conditions:

- Both units operating at 100% power.
- The 1D and 2C SW Supply Headers are in service.

Current conditions:

- Service Building Inside Operator reports indications that the 2C SW Supply Header is clogging.
- The operating crew has initiated 0-AP-12.00, Service Water System Abnormal Conditions.

Which ONE of the following correctly indicates the order in which the Third SW Supply Header is placed in service?

REFERENCE PROVIDED

A. 1) Open/check open 2-SW-11; 2) Place both MER 5 headers in service; (3) Open 2-SW-477; (4) Open/check open 1-SW-500 and 2-SW-478.

B. 1) Open/check open 2-SW-477; 2) Open/check open 1-SW-500 or 2-SW-478; (3) Place both MER 5 headers in service; (4) Open 2-SW-11.

C. 1) Open/check open 1-SW-500 and 2-SW-478; 2) Open 2-SW-477; (3) Verify 2C MER 5 supply in service; (4) Open 2-SW-11.

D. 1) Open/check open 2-SW-11; 2) Verify 2C MER 5 supply in service; (3) Open 2-SW-477; (4) Open/check open 1-SW-500 or 2-SW-478.

Proposed Answer:

A. 1) Open/check open 2-SW-11; 2) Place both MER 5 headers in service; (3) Open 2-SW-477; (4) Open/check open 1-SW-500 and 2-SW-478.

Explanation: To ensure continued SW supply to MER 3 and MER 5 chillers and Charging Pump SW pumps, a third header is placed in service prior to isolating the troubled one. Using the provided drawing, the candidate must logically place the 3rd header in service.

Technical Reference: 0-AP-12.00, rev 14

Reference Provided to Applicant: Yes – 0-AP-12.00, rev 14 – Attachment 1 page 3 of 3.

Learning Objective: ND-89.5-LP-2, Service Water; Objective H. Explain the system interconnections associated with the Service Water Systems, flowpaths affected and safety implications of stop logging, and abnormal procedures associated with the systems.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43.5 / 45.13)

Comments:

KA Match Analysis: The question matches the KA in that the candidate uses the reference provided to develop a valve lineup to place the third SW header in service. This act bypasses the portion of the system causing the problem.

Distractor Analysis:

A. CORRECT
B. Incorrect, but plausible if candidate does not use logic to develop a valve lineup to place the third SW header in service. Both 1-SW-500 and 2-SW-478 must be open to ensure SW supply to both MER 3 and MER 4.
C. Incorrect, but plausible if candidate does not use logic to develop a valve lineup to place the third SW header in service.
D. Incorrect, but plausible if candidate does not use logic to develop a valve lineup to place the third SW header in service. Both 1-SW-500 and 2-SW-478 must be open to ensure SW supply to both MER 3 and MER 4.

NUMBER 0-AP-12.00	ATTACHMENT TITLE PLACING THIRD SW HEADER IN SERVICE	ATTACHMENT 1
REVISION 14		PAGE 1 of 3

* *

CAUTION: This Attachment must be performed to ensure that three SW headers are in service before removal of any service water header.

* *

NOTE:
- Page 3 of this Attachment contains a graphic of the three Service Water headers and the respective isolation valves.
- If maintenance activities are currently being performed on waterboxes associated with SW headers, placing the third SW header in service should be performed IAW 0-OP-SW-49.3, SWAPPING CONTROL ROOM CHILLER AND CHARGING PUMP SW SUPPLY AND DISCHARGE HEADERS, instead of this Attachment.

1. ___ Check no current/recent maintenance activities on the three SW headers.

2. ___ Check open or open the following SW isolation:

   ___ a. 1-SW-495, 1D Waterbox MER 4 SW Sup Isol

   ___ b. 2-SW-11, 2A Waterbox MER 4 and 5 SW Sup Isol

   ___ c. 2-SW-474, 2C Waterbox MER 4 and 5 SW Sup Isol

3. ___ Place both MER 5 SW Supply Headers in service by performing the following:

   ___ a. Check open or open 2-SW-532, MER 5 Sup from 2-SW-11 Hdr Upstr Isol.

   ___ b. Vent line using 2-SW-572 until air is vented and water issues from the vent.

   ___ c. Check open or slowly open 2-SW-533, MER 5 Sup from 2-SW-11 Hdr Dnstr Isol.

   ___ d. Check open or open 2-SW-530, MER 5 Sup from 2-SW-474 Hdr Upstr Isol.

   ___ e. Vent line using 2-SW-571 until air is vented and water issues from the vent.

   ___ f. Check open or slowly open 2-SW-531, MER 5 Sup from 2-SW-474 Hdr Dnstr Isol.

4. ___ Vent the 1D, 2A, and 2C SW Headers using the following valves until air is vented and water issues from the vent:

   ___ a. 1-SW-737, 1D SW Header Vent

   ___ b. 2-SW-504, 2A SW Header Vent

   ___ c. 2-SW-505, 2C SW Header Vent

NUMBER 0-AP-12.00	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 14	PLACING THIRD SW HEADER IN SERVICE	PAGE 2 of 3

5. ___ Place SW Header 1D in service by checking open or slowly opening 1-SW-499, Unit 1 SW Sup Hdr MER 4 Manifold Isol.

6. ___ Place SW Header 2A in service by performing the following substeps.

   ___ a. Check open or open 2-SW-477, 2A Waterbox SW Sup Hdr MER 4 Manifold Isol.

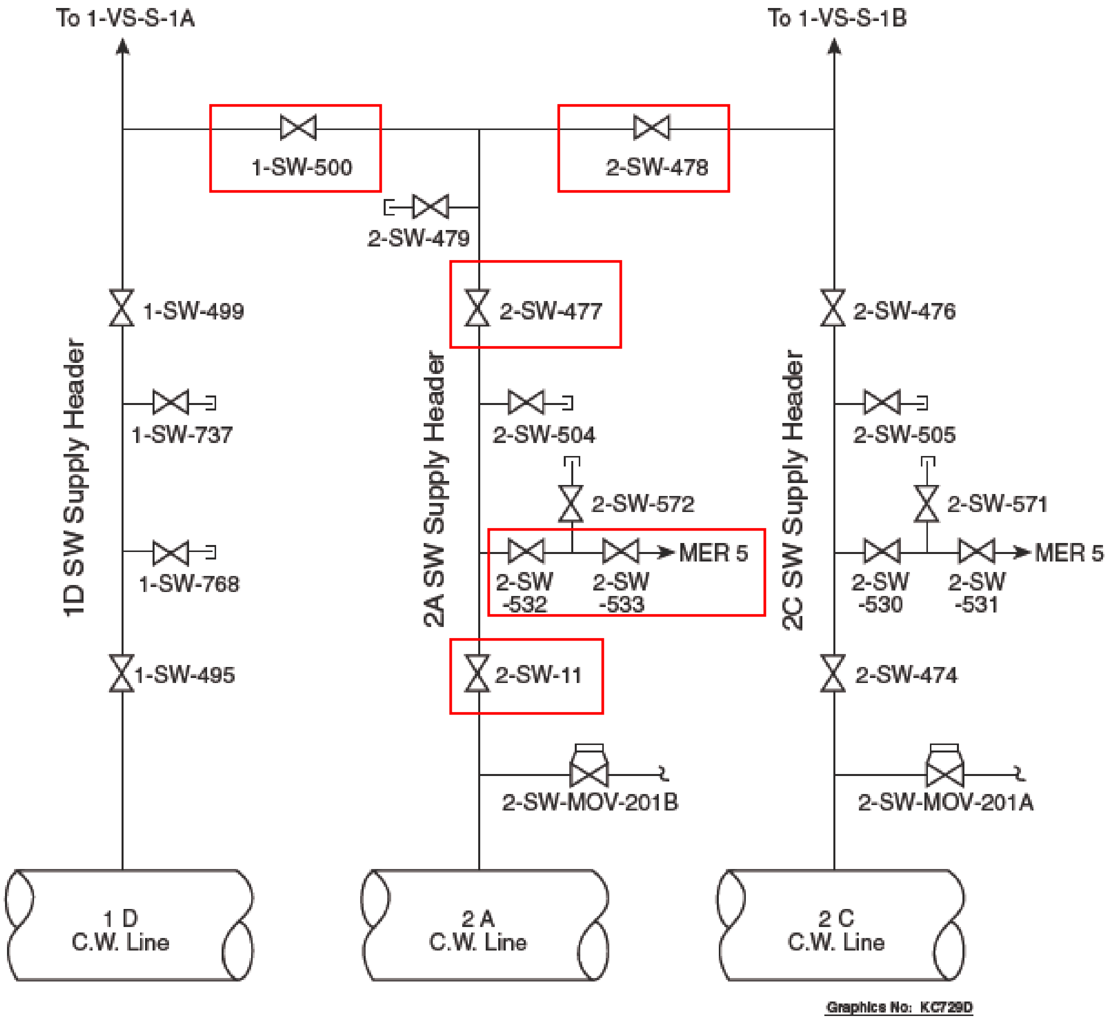
   ___ b. Vent line using 2-SW-479 until air is vented and water issues from the vent.

   ___ c. Check open or slowly open 1-SW-500, MER 4 SW Sup Hdr Manifold Xconn.

   ___ d. Check open or slowly open 2-SW-478, MER 4 SW Sup Hdr Manifold Xconn.

7. ___ Place SW Header 2C in service by checking open or slowly opening 2-SW-476, 2C Waterbox SW Sup Hdr MER 4 Manifold Isol.

NUMBER 0-AP-12.00	ATTACHMENT TITLE PLACING THIRD SW HEADER IN SERVICE	ATTACHMENT 1
REVISION 14		PAGE 3 of 3



CONTROL ROOM CHILLER AND CHARGING PUMP SERVICE WATER SUPPLY LINES

K/A Number: 063G2.4.20: DC Electrical Distribution: Knowledge of operational implications of EOP warnings, cautions and notes.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.8 IR-SRO: 4.3

Proposed Question: 45

A loss of the 1A DC bus will cause the following:

- The __1)__ EDG will automatically start.
- The __2)__ 4160 V bus breakers will lose control power.

A. 1) #3
   2) 1J bus

B. 1) #3
   2) 1H Bus

C. 1) #1
   2) 1J bus

D. 1) #1
   2) 1H bus

Proposed Answer:

D. 1) #1
   2) 1H bus

Explanation: The loss of DC Power procedure (1/2-AP-10.06) contains notes that #1 EDG will auto start on a loss of 1A DC bus and the 4160V breakers on the 1H bus will lose control power. The #3 EDG and 1J bus are affected on a loss of the 1B DC bus.

Technical Reference: 1-AP-10.06, Rev. 21

Reference Provided to Applicant: No

Learning Objective: ND-90.3-LP-6, 125 VDC Distribution; Objective C. Describe the indications and alarms associated with the DC and Black Battery distribution systems.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must know the NOTE in 1/2-AP-10.06 regarding EDG auto starts and loss of breaker control power.

Distractor Analysis:

A. Incorrect, but plausible if candidate fails to recall note or believes that the opposite train is affected.
B. Incorrect, but plausible if candidate fails to recall note or believes that the opposite train is affected.
C. Incorrect, but plausible if candidate fails to recall note or believes that the opposite train is affected.
D. CORRECT.

NUMBER	PROCEDURE TITLE	REVISION 21
1-AP-10.06	LOSS OF DC POWER	PAGE 2 of 6

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- The EDG 1 auto-starts when 1A DC Bus is lost.
- The EDG 3 auto-starts when 1B DC Bus is lost.
- AFW flow must be throttled to idle loops to prevent an inadvertent Header to Line SI.

NUMBER 1-AP-10.06	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 21	SWITCHGEAR WITHOUT CONTROL POWER LOSS OF 1A DC BUS	PAGE 1 of 1

1. Evaluate starting the following pumps prior to locally tripping the redundant equipment.

- ☐ 1-CH-P-1B or 1-CH-P-1C (Alt)
- ☐ 1-RH-P-1B
- ☐ 1-CC-P-1B

2. Evaluate locally tripping the following equipment breakers.

NO CONTROL POWER - BUS STILL ENERGIZED

- ☐ 15H3 #1 EDG
- ☐ 15H4 1-FW-P-3A
- ☐ 15H5 1-CH-P-1A
- ☐ 15H6 1-CH-P-1C (Norm)
- ☐ 15H8 1H Normal supply
- ☐ 15H9 Stub Bus supply
- ☐ 15H10 1-CC-P-1A
- ☐ 15H11 1-RH-P-1A

- 14H2 PRZR Heaters
- 14H3 1-SI-P-1A
- 14H4 1-RS-P-1A
- 14H5 1-CS-P-1A
- 14H7 1-RS-P-2A
- 14H8 1-VS-F-1A
- 14B1 1-WT-P-100A
- 14C2 1-MG-2
- 14C12 PRZR Heaters
- 14C16 1-VS-F-1C
- 14C17 1-CD-P-4C

NUMBER 1-AP-10.06	ATTACHMENT TITLE SWITCHGEAR WITHOUT CONTROL POWER LOSS OF B DC BUS	ATTACHMENT 3
REVISION 21		PAGE 1 of 1

1. Evaluate starting the following pumps prior to locally tripping the redundant equipment.

- ☐ 1-CH-P-1A OR 1-CH-P-1C (Norm)
- ☐ 1-RH-P-1A
- ☐ 1-CC-P-1A

2. Evaluate locally tripping the following equipment.

NO CONTROL POWER - BUS STILL ENERGIZED

- ☐ 15J2 1-CH-P-1C
- ☐ 15J3 #3 EDG
- ☐ 15J4 1-FW-P-3B
- ☐ 15J5 1-CH-P-1B
- ☐ 15J8 1J Bus supply from RSS
- ☐ 15J9 Stub Bus supply
- ☐ 15J10 1-CC-P-1B
- ☐ 15J11 1-RH-P-1B

- 14J-3 1-SI-P-1B
- 14J-4 1-RS-P-1B
- 14J-5 1-CS-P-1B
- 14J-7 1-VS-F-1B
- 14J-8 1-RS-P-2B
- 14J-9 PRZR Heater

K/A Number: 063K3.02: DC Electrical Distribution: Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: Components using DC control power.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.5 IR-SRO: 3.7

Proposed Question: 46

Which ONE of the following describes the reason a reactor trip occurs upon a complete loss of DC power?

A. Loss of voltage to reactor trip breakers causes these breakers to open and provide a direct reactor trip to meet the single failure mode criteria.

B. Loss of either DC bus would cause all main feedwater regulating valves and bypass valves to fail closed and a low S/G level will occur.

C. The turbine generator output breaker trips open and a reactor trip is required to protect the main turbine from overspeed.

D. The reactor coolant pumps trip off and a reactor trip is initiated prior to losing all flow through the core.

Proposed Answer: A. Loss of voltage to reactor trip breakers causes these breakers to open and provide a direct reactor trip to meet the single failure mode criteria.

Explanation: Upon a loss of DC power to their UV coils, the reactor trip and bypass breakers will open causing the reactor trip. The FRV and Feed bypass controllers are powered from the vital buses. The RCP breakers have lost trip power, so will be unable to open except locally with the pushbutton on the breaker itself. The generator output breakers get their control power from batteries in the switchyard.

Technical Reference: 1-AP-10.06, Loss of DC Power, rev 21

Reference Provided to Applicant: No

Learning Objective: ND-90.3-LP-6, 125 VDC Distribution; Objective B. Given a total loss of either Station Battery bus, summarize the effects on station operation.

Question Source: Bank (LORP Bank – LAOP0059)

Which ONE of the following is the reason a reactor trip will occur upon a complete loss of DC power?

A. Loss of voltage to reactor trip breakers causes these breakers to open and provide a direct reactor trip to meet the single failure mode criteria.

B. Loss of either DC bus would cause all main feedwater regulation valves and feedwater bypass valves to fail closed and a low S/G level will occur.

C. The turbine generator output breaker trips open and a reactor trip is required to protect the main turbine from overspeed.

D. The reactor coolant pumps trip off and a reactor trip is initiated prior to losing all flow through the core.

Answer: A

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 45.6)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must know which components use DC control power and how the loss of DC affects the component.

Distractor Analysis:

A. CORRECT.
B. Incorrect, but plausible if candidate believes that FRV and bypass controllers utilize 125 VDC control power from the DC bus. These controllers utilize VB control power.
C. Incorrect, but plausible if the candidate believes that main generator output breaker control power comes from the station DC buses. The generator output breakers will open as usual as a result of the reactor trip.
D. Incorrect, but plausible if the candidate believes that the RCP breakers open on loss of control power. They will actually lose all remote operation and protection due to loss of tripping power and must be opened locally at the breaker if necessary.

NUMBER	PROCEDURE TITLE	REVISION
1-AP-10.06	LOSS OF DC POWER	21
		PAGE 2 of 6

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- The EDG 1 auto-starts when 1A DC Bus is lost.
- The EDG 3 auto-starts when 1B DC Bus is lost.
- AFW flow must be throttled to idle loops to prevent an inadvertent Header to Line SI.

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK LOSS OF DC BUS: ☐ • Reactor Trip Breakers - One open and one de-energized AND ☐ • 1-MS-SOV-102A OR 1-MS-SOV-102B - DE-ENERGIZED (OPEN) AND ☐ • RX Trip BRK A and BYP BRK B OR RX Trip BRK B and BYP BRK A indicating lights - OFF	Do the following: ☐ a) Review appropriate station drawings for loss of individual circuits. ☐ b) RETURN TO procedure and step in effect.
2.___	CHECK TURBINE TRIP:	
	☐ a) Manually trip the turbine.	☐ a) Locally trip the turbine.
	☐ b) Check all turbine stop valves - CLOSED	☐ b) IF turbine will NOT trip, THEN close MSTVs.
	c) Isolate reheaters by closing MSR steam supply SOV ☐ • 1-MS-SOV-104	☐ c) IF reheater FCVs will NOT close, THEN close MSR steam supply MOVs.
3.___	CHECK THE MAIN GENERATOR OUTPUT BREAKERS - OPEN ☐ • OCB-G102 ☐ • OCB-G1T240	☐ Manually open breakers.

NUMBER 1-AP-10.06	ATTACHMENT TITLE	ATTACHMENT 8
REVISION 21	CONTINGENCY ACTIONS - LOSS OF DC BUS 1A	PAGE 1 of 1

* *

CAUTION: The 4 KV Switchgear Breakers without DC control power will NOT trip automatically and can NOT be cycled electrically.

* *

K/A Number: 064K6.08: Emergency Diesel Generator: Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.2 IR-SRO: 3.3

Proposed Question: 47

Current Condition on Unit 1:

- 1-EP-BKR-15H8 (1H Bus Normal Supply from RSS) tripped open spuriously.
- #1 EDG has started but failed to load.
- A field operator reports the exciter field circuit breaker on #1 EDG is tripped.

The Fuel Oil System _(1)_ be able to transfer fuel oil to the **Base Tank** and _(2)_ be able to transfer fuel to the **Wall Tank**.

A. (1) will
(2) will

B. (1) will not
(2) will

C. (1) will not
(2) will not

D. (1) will
(2) will not

Proposed Answer: C. (1) will not
(2) will not

Explanation: Power supply to the EDG transfer pumps that feed the wall tank are 1H1-1 and 1H1-2. These MCCs will not have power if EDG does not load. Also, if the EDG field is not flashed there will be no power to 1-EE-P-2A/2B. These pumps transfer fuel oil from the wall tank to the base tank.

Technical Reference: 0-AP-17.04, EDG 1 or EDG 2 – Emergency Operations, Rev. 27.

Reference Provided to Applicant: No

Learning Objective: ND-90.3-LP-1, Emergency Diesel Generator; Objective B. Describe the components and flowpaths of the EDG Air Start, Fuel Oil, Cooling, and Lubricating Oil Systems.

Question Source: Bank (Surry LORP LEDG0062)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 45.7)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must know the effect on the fuel oil system if the Diesel Exciter field breaker does not close. In this case the Diesel will not have the ability to pump fuel from the Wall tanks to the Base mounted tanks.

Distractor Analysis:

A. Incorrect, but plausible if the candidate does not know the power supplies to the EDG FO transfer pumps.
B. Incorrect, but plausible if the candidate does not know the power supplies to the EDG FO transfer pumps.
C. CORRECT.
D. Incorrect, but plausible if the candidate does not know the power supplies to the EDG FO transfer pumps.

Parent Question

	ID: LEDG0062	Points: 1.00

Current Condition on Unit 1:

- 1-EP-BKR-15H8 (1H Bus Normal Supply from RSS) tripped open spuriously.
- #1 EDG has started but failed to load.
- A field operator reports the exciter field circuit breaker on #1 EDG is tripped.

Based on the current conditions, which ONE of the following describes the impact this will have on the ability to transfer fuel to the #1 EDG?

The Fuel Oil System __(1)__ be able to transfer fuel oil to the Base Tank and __(2)__ be able to transfer fuel to the Wall Tank.

A. (1) will not
(2) will not

B. (1) will not
(2) will

C. (1) will
(2) will

D. (1) will
(2) will not

Answer: A

Answer Explanation:

- Power supply to the diesel transfer pumps that feed the wall tank are 1H1-1 and 1H1-2, which will not have power if #1 EDG does not load.
- If the generator field is not flashed, power will not be available to 1-EE-P-2A/2B to transfer fuel oil to the base tank.

NUMBER	PROCEDURE TITLE	REVISION
0-AP-17.04	EDG 1 OR EDG 2 - EMERGENCY OPERATIONS	27
		PAGE 6 of 10

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. ___	MONITOR EDG ( ) PARAMETERS IAW ATTACHMENT 5	
12. ___	GO TO STEP 28	

* *

CAUTION: Continued operation of the EDG without field voltage will deplete the base tank fuel supply.

* *

K/A Number: 065AK3.08: Loss of Instrument Air/ 8: Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Actions contained in EOP for loss of instrument air.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.7 IR-SRO: 3.9

Proposed Question: 48

Given the following conditions:

- Unit 2 is experiencing a sustained loss of Instrument Air.
- The operating team has tripped the reactor and initiated 2-E-0, Reactor Trip or Safety Injection.
- 0-AP-40.00, Non-Recoverable Loss of Instrument Air has also been initiated.

Which ONE of the following completes the statements below?

1) Why are Reactor Coolant Pumps tripped after the Reactor Trip?
2) How is decay heat removed from the RCS during this event?

A. 1) Reduce heat input to RCS.
   2) Manual operation of Steam Dumps.

B. 1) Component Cooling is lost.
   2) Manual operation of Steam Dumps.

C. 1) Component Cooling is lost.
   2) Local operation of S/G PORVs.

D. 1) Reduce heat input to RCS.
   2) Local operation of S/G PORVs.

Proposed Answer:

C. 1) Component Cooling is lost.
   2) Local operation of S/G PORVs.

Explanation: On a sustained loss of IA, the outside CTMT trip valves fail closed. This results in a complete loss of Component Cooling flow to the RCPs. Also, MSTVs will fail closed isolating the Steam Dumps. There are local, dedicated air bottles to locally operate the S/G PORVs for decay heat removal. The S/G PORVs are sized to remove decay heat as well as RCP heat.

Technical Reference: 0-AP-40.00 - Non-Recoverable Loss of Instrument Air, rev. 30

Reference Provided to Applicant: No

Learning Objective: ND-95.1-LP-9, Loss of Instrument Air; Objective D. Assess the expected sequence of events and the operator actions associated with a Loss of Instrument Air.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.5,41.10 / 45.6 / 45.13)

Comments:

KA Match Analysis: The question meets the KA in that the candidate must know actions required to compensate for the Loss of IA while in the EOP network.

Distractor Analysis:

A. Incorrect, but plausible if the candidate does not recall that RCPs should be secured due to loss of CC flow IAW 0-AP-40.00 and also believes that Steam Dumps are available (MSTVs have closed on loss of IA).
B. Incorrect. MSTVs have closed on loss of IA. Plausible if the candidate believes that Steam Dumps are available.
C. CORRECT.
D. Incorrect, but plausible if the candidate does not recall that RCPs should be secured due to loss of CC flow IAW 0-AP-40.00.

NUMBER	PROCEDURE TITLE	REVISION 30
0-AP-40.00	NON-RECOVERABLE LOSS OF INSTRUMENT AIR	PAGE 3 of 25

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	CHECK UNIT - AT POWER PRIOR TO LOSS OF INSTRUMENT AIR (Continued)	
		IF Unit NOT on RHR, THEN do the following: a) Consult with the following: ☐ • OM on call ☐ • STA ☐ • Shift Supervision ☐ b) Review Steps 2 through 46 to determine steps to be performed. ☐ c) Perform steps required and initiate appropriate procedures as required.
2. ___	CHECK TRIPPED OR TRIP REACTOR AND CHECK INITIATED OR INITIATE ( )-E-0, REACTOR TRIP OR SAFETY INJECTION	

* *

CAUTION: Component Cooling will be lost to Reactor Coolant pumps.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. ___	TRIP REACTOR COOLANT PUMPS AND INITIATE ( )-AP-39.00, NATURAL CIRCULATION OF RCS	

NUMBER	PROCEDURE TITLE	REVISION
0-AP-40.00	NON-RECOVERABLE LOSS OF INSTRUMENT AIR	30
		PAGE 7 of 25

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: The MSTV bypass valves should be opened together to equalize steam and feed rates of the steam generators.

* *

NOTE:
- RCS temperature and SG pressure should be monitored for excessive or unexpected cooldown.
- RCS temperature should be stabilized until RCS boration is commenced IAW Step 17.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. ___	LOCALLY ESTABLISH ALTERNATE STEAM RELEASE:	
	☐ a) Check SG narrow range levels - GREATER THAN 12% IN AT LEAST ONE SG	a) Do the following: ☐ 1) Maintain AFW flow greater than 350 gpm until narrow range level greater than 12% in at least one SG. ☐ 2) WHEN narrow range level greater than 12% in at least one SG, THEN perform Step 12b.
	b) Establish steam release path: ☐ • For Unit 1, locally use SG PORV(s) IAW Attachment 10 ☐ • For Unit 2, locally use SG PORV(s) IAW Attachment 11	

K/A Number: 069AK3.01: Loss of CTMT Integrity / 5: Knowledge of the reasons for the following responses as they apply to the Loss of Containment Integrity: Guidance contained in EOP for loss of containment integrity.

Level: RO Tier #: 1 Group #: 2 IR – RO: 3.8 IR-SRO: 4.2

Proposed Question: 49

Given the following conditions:

- Unit 1 was operating at 100% when a small break LOCA occurred.
- The team is in 1-E-1 at Step 17.
- 1-IA-446 and 1-IA-447 were opened to align Turbine Building Instrument Air to containment.
- An operator is standing by, briefed in accordance with 1-OP-IA-005, Administrative Control of Unit 1 Instrument Air to Unit 1 Containment Valves 1-IA-446 and 1-IA-447.

The operator must immediately close 1-IA-446 and 1-IA-447 within ____ of Safety Injection initiation to restore Containment Integrity.

A. 120 seconds

B. 10 minutes

C. 45 minutes

D. 40 seconds

Proposed Answer:

D. 40 seconds

Explanation: 1-IA-446 and -447 are manually-operated containment isolation valves. They are normally maintained in the locked-closed position. If opened when containment integrity is required, an administrative control must be established. The procedure 1-OP-IA-005 specifies the time requirement of 40 seconds to close the valves in the event of an SI signal or containment isolation is required.

Technical Reference: 1-OP-IA-005, Administrative Control of Unit 1 Instrument Air to Unit 1 Containment Valves 1-IA-446 and 1-IA-447, Rev. 2.

Reference Provided to Applicant: No

Learning Objective: ND-92.1-LP-1, Station Air Systems; Objective B. [Determine the system flowpaths and components associated with the Instrument Air System. SOER 88-01, Recommendation 2&3].

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR 41.5,41.10 / 45.6 / 45.13)

Comments:

KA Match Analysis: The question meets the KA in that the candidate must know the administrative control requirements to restore containment integrity with IA and CTMT IA crosstied.

Distractor Analysis:

A. Incorrect, but plausible since this is the time limit to close a Main Feed Regulating Valve that's "on-the-jack" per 1(2)-MOP-FW-015.
B. Incorrect, but plausible since this is the time limit to close 1-FP-36 (motor-driven fire pump recirculation valve) IAW 0-OP-FP-003.
C. Incorrect, but plausible since this is the time limit to establish Containment Closure IAW 1-OPT-CT-210.
D. CORRECT.

Init **Verif**

5.0 INSTRUCTIONS

5.1 Establishing Administrative Control of 1-IA-446 and 1-IA-447

______ 5.1.1 Perform a prejob briefing covering the following requirements:

______ a. The person assigned Administrative Control of 1-IA-446 and 1-IA-447 understands that no other concurrent job responsibilities have been or shall be accepted during performance of this procedure.

______ b. The person assigned Administrative Control of 1-IA-446 and 1-IA-447 understands the requirement to remain in the immediate vicinity of the valves at all times when 1-IA-446 and 1-IA-447 are open.

______ c. The person assigned Administrative Control of 1-IA-446 and 1-IA-447 understands that continuous communication capability shall be maintained with the Main Control Room at all times by use of either a Portable Radio (primary method) or Gai-Tronics (secondary method).

______ d. The person assigned Administrative Control of 1-IA-446 and 1-IA-447 understands that 1-IA-446 and 1-IA-447 shall be immediately closed within 40 seconds of Safety Injection initiation or upon determination that containment isolation is required.

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5.2 **Administrative Control of 1-FP-36**
(Reference 2.3.4, **2.4.1**)

______ 5.2.1 Perform a prejob briefing covering the following requirements. (✓)

____ The person assigned Administrative Control of 1-FP-36, Recirc For 1-FP-P-1, understands that no other concurrent responsibilities have been or shall be accepted during performance of this procedure.

____ The person assigned Administrative Control of 1-FP-36 understands that 1-FP-36 must be fully closed within 10 minutes of notification that 1-FP-P-1 is required for a fire control function.

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6.89 **Verification of Conditions for Penetrations**

SRO 6.89.1 Review Attachment 2, Penetration Breach Log.

SRO 6.89.2 Check the following for each open penetration.

- Closure Actions are specified and all actions are external to Containment. (with exception of hatches)
- Containment Closure Team has been established and briefed.
- Penetrations are capable of being closed within 45 minutes, when acceptable radiological conditions exist for closure team personnel.

______ 4.17 When automatic closure of the MFRV is defeated, administrative controls ensure Operators are assigned to manually close the proper Main Feedwater MOV and/ or MFRV within two minutes to satisfy Time Critical Operator Action. (Ref. 2.3.9)

K/A Number: 071K3.04: Waste Gas Disposal: Knowledge of the effect that a loss or malfunction of the Waste Gas Disposal System will have on the following: Ventilation system.

Level: RO Tier #: 2 Group #: 2 IR – RO: 2.7 IR-SRO: 2.9

Proposed Question: 50

Given the following information:

- The "A" Waste Gas Decay Tank is being released.
- 1-GW-RM-131A, Process Vent Radiation Monitor (Particulate) alarms in HIGH.

The Radiation Monitor HIGH alarm automatically closes all of the following valves EXCEPT ___.

A. 1-GW-FCV-160, CTMT Vacuum Isolation, Unit 1

B. 1-GW-FCV-260, CTMT Vacuum Isolation, Unit 2

C. 1-GW-HCV-106, Aerated Vent Isolation

D. 1-GW-FCV-101, Decay Tank Bleed Isolation

Proposed Answer:

C. 1-GW-HCV-106, Aerated Vent Isolation

Explanation: There are 3 valves that are automatically closed on a Process Vent HIGH particulate or gas alarm. They are the CTMT vacuum pump isolations on each unit as well as the WGDT bleed isolation valve. GW-HCV-106 is manually closed during the performance of the ARP to help identify the source of the leak.

Technical Reference: 0-RMA-C6 – Process Vent Part Alert/Hi, Rev. 3

Reference Provided to Applicant: No

Learning Objective: ND-92.4-LP-1 – Gaseous and Liquid Waste Processing Systems; Objective B. Determine the operation of the Gaseous Waste System.

Question Source: Modified Bank (Sequoyah 2007 - #50)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 45.6)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must know how the ventilation system realigns on a malfunction of the Gaseous Waste system.

Distractor Analysis:

A. Incorrect, but plausible if the candidate does not recall that the CTMT vacuum pumps on each unit (1-FCV-160/260) are isolated by PV High alarm.
B. Incorrect, but plausible if the candidate does not recall that the CTMT vacuum pumps on each unit (1-FCV-160/260) are isolated by PV High alarm.
C. CORRECT.
D. Incorrect, but plausible if the candidate does not recall that the WGDT bleed (1-GW-FCV-101) is isolated by PV High alarm.

Parent Question, Sequoyah 2007 - #50

50. Given the following plant conditions:

- A Gas Decay tank release is in progress with A- A ABGTS running for dilution flow.
- A leak occurs on the waste gas compressor which results in a gas release to the Auxiliary Building.
- 0-RE-90-101, Auxiliary Building Vent Monitor, is in alarm.

Which ONE (1) of the following indicates the effect this leak will have on the plant? (Assume no operator actions)

A. Gas Decay Tank release will be terminated; ONLY A-A ABGTS will be running.

B. Gas Decay Tank release will be terminated; BOTH A-A and B-B ABGTS will be running.

C. Gas Decay Tank release will continue; ONLY A-A ABGTS will be running.

D. Gas Decay Tank release will continue; BOTH A-A and B-B ABGTS will be running.

NUMBER	PROCEDURE TITLE	REVISION
0-RMA-C6	PROCESS VENT PART ALERT / HI	3
		PAGE 3 of 7

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: When CTMT Vacuum Pump Discharge Isolation valve 1-GW-FCV-160 or 1-GW-FCV-260 is closed, the associated Vacuum Pumps must be placed in OFF.

* *

NOTE: If a high alarm has actuated, the automatic functions associated with that monitor shall be verified or performed.

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.___	CHECK AUTOMATIC ACTIONS:	
	☐ a) Decay Tank Bleed Isolation valve 1-GW-FCV-101 - CLOSED	☐ a) Manually close valve(s).
	b) Check the following	b) Do the following:
	☐ • CTMT Vacuum Pump Discharge Isolation valve 1-GW-FCV-160 - CLOSED	1) Check or place Ctmt Vacuum pumps in OFF
	☐ • CTMT Vacuum Pump Discharge Isolation valve 1-GW-FCV-260 - CLOSED	☐ • 1-CV-P-1A
		☐ • 1-CV-P-1B
		☐ • 2-CV-P-1A
		☐ • 2-CV-P-1B
		☐ 2) Manually close valve(s).
		☐ 3) GO TO Step 6.

NUMBER	PROCEDURE TITLE	REVISION
0-RMA-C6	PROCESS VENT PART ALERT / HI	3
		PAGE 6 of 7

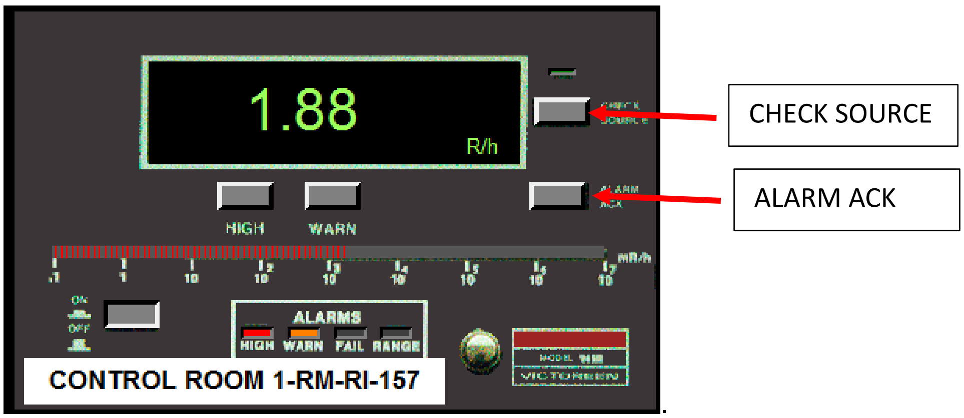
STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. ___	IDENTIFY SOURCE OF RELEASE:	
	☐ a) Close trip valve 1-GW-HCV-106 Time closed __________	
	☐ b) Check activity decreased within 15 minutes of time in Step 11.a	☐ b) Open trip valve 1-GW-HCV-106 AND GO TO Step 11.f.
	☐ c) Close isolation valves IAW Attachment 1	
	☐ d) Open trip valve 1-GW-HCV-106	
	☐ e) Open isolation valves closed in Attachment 1 individually to identify source of release	
	☐ f) Check source of release isolated	f) Do the following: ☐ 1) Monitor systems with relief valves discharging to process vent. ☐ 2) Isolate release.
	☐ g) Return unaffected systems to service	

K/A Number: 072A4.01: Area Radiation Monitoring: Ability to manually operate and/or monitor in the control room: Alarm and interlock setpoint checks and adjustments.

Level: RO Tier #: 2 Group #: 2 IR – RO: 3.0 IR-SRO: 3.3

Proposed Question: 51

Given the following:



1) When the CHECK SOURCE button is pushed, an actual source ____ exposed to the detector.

2) When the ALARM ACK button on the faceplate is pushed, the 1-RM-RI-157 HIGH annunciator (0-RM-H3) _____ silence and lock in.

A. 1) is not
 2) will not

B. 1) is not
 2) will

C. 1) is
 2) will not

D. 1) is
 2) will

Proposed Answer: A

Explanation: For Area Radiation Monitors an electronic “check source” is used vice using an actual source (used for process radiation monitors).

Technical Reference: 0-OPT-RM-001, Radiation Monitoring Equipment Check, Rev. 46.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-1, Pre-TMI Rad Monitor, Objective A, Determine the operation of the Victoreen Area Radiation Monitoring System.

Question Source: Modified Bank (2014 NRC exam Q 50) Part 2 is new.

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 45.5 to 45.8)

Comments:

K/A Analysis: Question requires an understanding of the proper indications that should be observed when conducting a source check of the Control Room Area Radiation Monitor. This meets the K/A.

Distractor Analysis:

A. Correct.
B. Incorrect. The ALARM ACK button only acknowledges the alarm on the faceplate. The annunciator buttons on the panel are used to acknowledge/silence/reset the RM annunciators. Plausible because this is partially correct.
C. Incorrect. Plausible because this type of check source would be present for Process Radiation Monitors.
D. Incorrect. Plausible because this type of check source would be present for Process Radiation Monitors. The ALARM ACK button only acknowledges the alarm on the faceplate. The annunciator buttons on the panel are used to acknowledge/silence/reset the RM annunciators.

2014 NRC Exam Question 50:

Question 50

Unit 1 and 2 are operating at 100% power.

0-OPT-RM-001, Radiation Monitor Equipment Check is in progress.

1-RM-RI-151, Control Room Rad Monitor, is to be source checked.

When the CHECK SOURCE pushbutton is depressed a(n) ___(1)____ is applied to the detector, and the Radiation Monitor will return to normal indication after ____(2)____.

A. 1) radioactive source 2) 2 to 3 seconds

B. 1) radioactive source 2) 3 minutes

C. 1) electronic signal 2) 2 to 3 seconds

D. 1) electronic signal 2) 3 minutes

Proposed Answer: C. 1) electronic signal 2) 2-3 seconds

Reference 0-OPT-RM-001 Attachment 3 (Area Rad Monitors) Correct answer.

Attachment 3
AREA RADIATION MONITORS SOURCE CHECK AND ALARM CHECK

NOTE: A readout during source checking that is prefixed with an "E" indicates a failed source check. (Reference 2.3.8)

NOTE: When the source check is performed the Alert/Failure and High Alarms will actuate for all area monitors listed below except 1-RM-RI-162.

1. Perform a source check for Digital Area Monitors as follows.
   a. Record the Current Reading of the respective AREA radiation monitor.
   b. For each AREA radiation monitor, depress the CHECK SOURCE pushbutton and check an increased readout in R/hr which does not begin with an "E."
   c. Depress the ALARM ACK pushbutton on the ratemeter.
   d. Acknowledge the alarm on the Radiation Monitoring Panel.
   e. For each AREA radiation monitor, depress the CHECK SOURCE pushbutton a second time and check the reading returns to background.
   f. Reset the alarms on the Radiation Monitoring panel.

MONITOR DESIGNATION	CURRENT READING	Increased Indication in R/hr With No "E"	Indication Returned to Background (INIT)
1-RM-RI-151 DECON BLDG			
1-RM-RI-152 NEW FUEL STOR			
1-RM-RI-153 FUEL PIT BRDG			
1-RM-RI-154 AUX BLDG CONT			
1-RM-RI-155 AUX BLDG DRUM			
1-RM-RI-156 AUX BLDG SAMPLE			
1-RM-RI-157 CONTROL ROOM			

0-OPT-RM-001, Attachment 1, Source Check for Process Rad Monitors.

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Attachment 1

PROCESS RADIATION MONITORS SOURCE CHECK AND SETPOINT DATA

1. Perform a source check of applicable monitors by performing the following substeps.
   a. Depress button marked Check Source, on monitor and hold in for 3 to 5 seconds or until an upscale deflection or readout is indicated.
   b. Check increased readout or upscale deflection of the meter. IF increased readout or upscale deflection of the meter is NOT evident, THEN purge the system or advance the paper AND repeat Steps a and b.
2. Check Alarm Setpoints of applicable monitors by performing the following substeps.
   a. Record High Alarm Setpoint value.
   b. Depress HIGH pushbutton.
   c. Read digital readout and record reading in Actual Alarm Setpoint.

ND-95.3-LP-1, Pre-TMI Rad Monitor, Operation of Check source pushbutton.

e. Check source pushbutton – Does one of two things to check monitor operation:

- For an AREA radiation monitor, an electronic signal is input to the detector. The operator should see an increase of ~5 R/hr.
- For the PARTICULATE & GAS radiation monitors, a radioactive source is exposed to the detector. The operator should observe an increased indication.

ND-95.3-LP-1, Pre-TMI Rad Monitor, Operation of faceplate alarm acknowledge. This ack pushbutton only acknowledges the alarm on the individual rad monitor.

c. Alarm status indicators - Will light up when any of the five alarm conditions exist.

(1) High alarm - Will give the alarm whenever the high setpoint is reached. The high alarm light will come on and the annunciator will alarm. The bargraph color will change to red. The alarm will flash until the alarm acknowledge pushbutton is pushed. The high alarm setpoint can be checked by pushing the HIGH pushbutton.

K/A Number: 073K5.02: Process Radiation Monitoring: Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Radiation intensity changes with source distance.

Level: RO Tier #: 2 Group #: 1 IR – RO: 2.5 IR-SRO: 3.1

Proposed Question: 52

Given the following conditions:

- 1-CH-RI-118 and 1-CH-RI-119 (RCS Letdown RMs) are indicating 100 mRem/hr.
- Health Physics reports a hot-spot physically located 4 feet away from the RMs and is the source of the elevated reading.
- An operator must hang a tag on a valve that is located 8 feet from the hot spot.

What is the dose rate in the area where the operator will be hanging the tag?
(ASSUME THE HOT SPOT IS A POINT SOURCE)

A. 50 mRem/hr

B. 25 mRem/hr

C. 400 mRem/hr

D. 71 mRem/hr

Proposed Answer: B.

Explanation: With a point source, the equation $D_1r_1^2 = D_2r_2^2$ is used, where D is Dose Rate and r is radius (or distance). Given the data above, the operator would be expected to be in a radiation field of 25 mRem/hr.

Technical Reference:

Reference Provided to Applicant: No

Learning Objective: ND-93.5-LP-1, Pre-TMI Rad Monitor, Objective B, Determine the operation of the Victoreen Process Radiation Monitoring System.

Question Source: Modified Bank (Harris, 2004, Question 63, Attached)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 45.7)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must identify how radiation intensity changes with distance from source.

Distractor Analysis:

A. Incorrect. Plausible if Candidate uses line source equation of $D_1r_1 = D_2r_2$. 100mRem/hr x (4 feet) ÷ (8 feet) = 50 mRem/hr.

B. Correct. 100mRem/hr x $(4\text{ feet})^2 \div (8\text{ feet})^2$ = 25 mRem/hr.

C. Incorrect. Plausible if Candidate switches distances. 100mRem/hr x $(8\text{ feet})^2 \div (4\text{ feet})^2$ = 400 mRem/hr.

D. Incorrect. Plausible if Candidate uses square root vice square of distance $D_1\sqrt{r_1} = D_2\sqrt{r_2}$. 100 mRem/hr x √4 feet ÷ √8 feet = 70.7 mRem/hr (rounded to 71).

Parent question (Harris, 2004, Question 63)

QCESTION: 63

Given the following conditions:

- After transferring resin, it is noted that RM-1WR-3644A, SPENT RESIN PUMP 1-44., radiation monitor is indicating 10mRem/hr.
- The monitor is physically located 20 feet away from a suspected clog in the pipe which is the source of the monitor indication.
- An operator must hang a clearance on a valve that is located 5 feet from the suspected clog in the pipe.

What is the dose rate in the area where the operator will be hanging the clearance?
(ASSUME THE CLOG IN THE PIPE IS A POINT SOURCE)

a. 20 mRem/hr

b. 40 mRem/hr

c. 80 mRem/hr

d. 160mRem/hr

ANSWER
d. 160mRem/hr

K/A Number: 074EK2.09: Inad. Core Cooling / 4: Knowledge of the interrelations between the and the following Inadequate Core Cooling: Controllers and positioners.

Level: RO Tier #: 1 Group #: 2 IR – RO: 2.6 IR-SRO: 2.6

Proposed Question: 53

Given the following:

- A LOCA inside containment occurred on Unit 2.
- CETCs are at 708ºF and rising.
- Full scale RVLIS is at 44% rising slowly.
- 2-SI-MOV-2867C and D, HHSI to COLD LEGS, fail to open.

Which of the following completes the statements below?

1) When an SI occurs, a contact closes in the Open contactor circuit of 2-SI-MOV-1867C and 2-SI-MOV-2867D to open the valves. The contact remains closed until __1)__.
2) Valves 2-SI-MOV-2867C and 2-SI-MOV-2867D __2)__ throttleable.

A. 1) SI is reset
2) are

B. 1) SI is blocked
2) are not

C. 1) SI is blocked
2) are

D. 1) SI is reset
2) are not

Proposed Answer: A

Explanation: Conditions given meet entry requirements for FR-C.1, Inadequate Core Cooling. The first step in 2-FR-C.1 is to check SI valve alignment, and since 2-SI-MOV-2867C and D failed to open, the procedure will direct manually opening the HHSI to cold leg MOVs. 2-SI-MOV-2867C, and 2-SI-MOV-1867D have no seal-in for the Open or Close direction when using the control switch. 1) When an SI occurs, the SI signal closes a contact in the Open contactor circuit to open the valves fully on an SI signal. The contact remains closed until SI is reset; at minimum 60 seconds must elapse before SI may be reset. 2) The seal-in function of the HHSI MOVs, 2-SI-MOV-1867C and D was removed to enhance system reliability during a case where only a single train has power during the recirculation phase of SI, and flow must be shifted.

Technical Reference: 2-FR-C.1, Response to Inadequate Core Cooling, Rev. 35. ND-91-2-LP-2, SI System Description.

Reference Provided to Applicant: NO

Learning Objective: ND-91-2-LP-2, SI System Description, Objective C, Using a simplified one-line diagram drawn from memory showing SI system components, flowpaths, and interconnections, describe the major SI System components to include instrumentation and annunciators available in the control room.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 41.7 / 45.7)

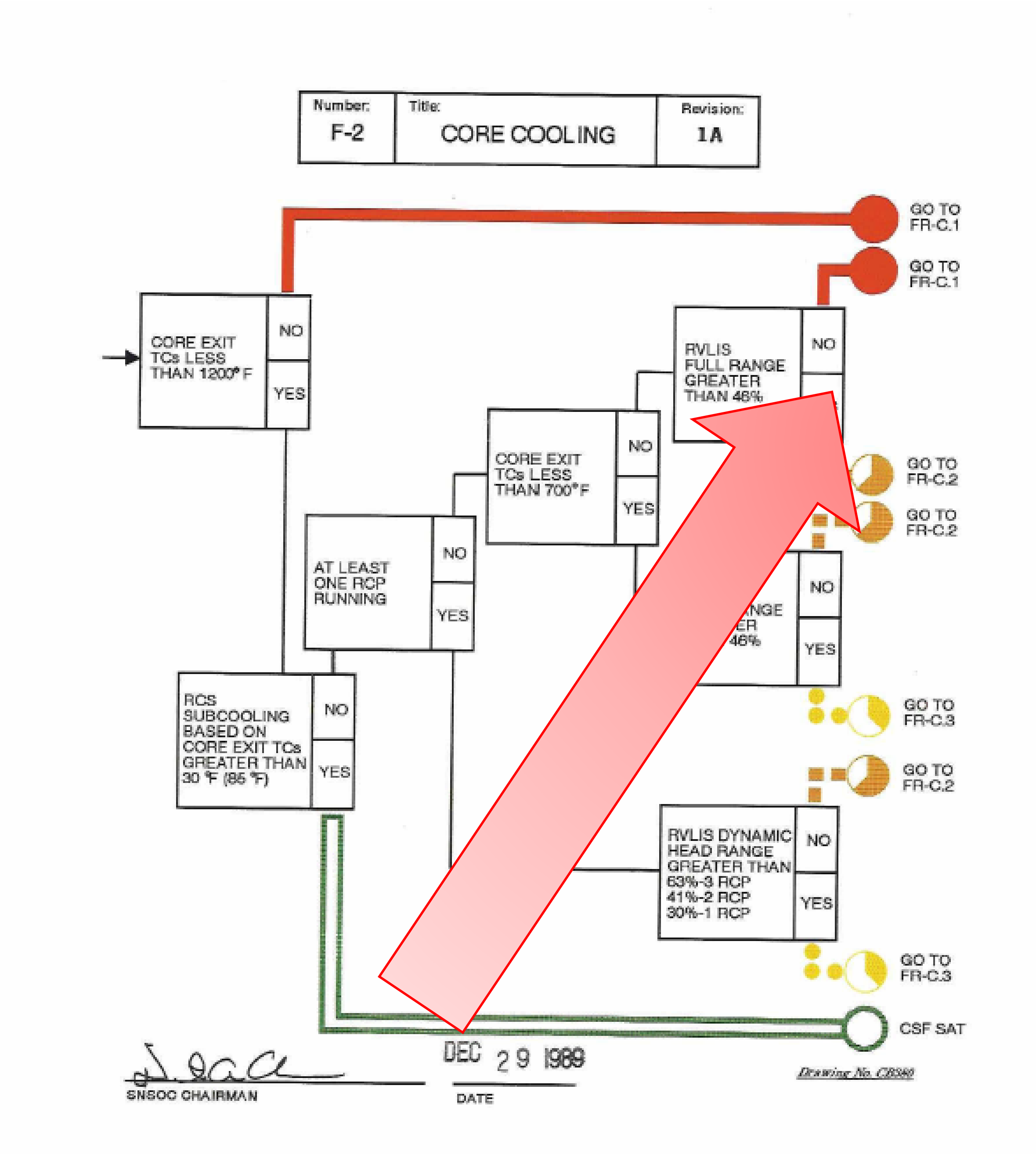
Comments:

K/A Analysis: KA requires knowledge of interrelations between inadequate core cooling and controllers and positioners. During a LBLOCA the MOV controllers for MOV 2867C and D must function properly in order to provide HHSI to the core. The operator must understand how these controllers operate and in order to provide proper manual operation in the event of failure.

Distractor Analysis:

A. Correct.
B. Incorrect. 1) Incorrect, as the seal-in Open remains until SI is reset. Plausible because there are some components (CS pumps and discharge valves) that cannot be manually re-positioned from the control room until the initiation condition is clear. In the case of CS pumps and valves, CLS must be reset (initiation condition clear) before the operator can manually reposition MOVs. 2) Incorrect. Seal-in function is clear. Plausible because many ESF components are seal-in open. Example of this is the Cont Spray suction and discharge MOVs.
C. Incorrect. 1) Incorrect, as the seal-in Open remains until SI is reset. Plausible because there are some components (CS pumps and discharge valves) that cannot be manually re-positioned from the control room until the initiation condition is clear. In the case of CS pumps and valves, CLS must be reset (initiation condition clear) before the operator can manually reposition MOVs. 2) Correct.
D. Incorrect. 1) Correct. 2) Incorrect. Seal-in function is clear. Plausible because many ESF components are seal-in open. Example of this is the Cont Spray suction and discharge MOVs.

Reference 0-F-2, Core Cooling. This shows that entry conditions are met for question conditions.



2-FR-C.1, Response to Inadequate Core Cooling step 1.

NUMBER	PROCEDURE TITLE	REVISION 35
2-FR-C.1	RESPONSE TO INADEQUATE CORE COOLING	PAGE 2 of 19

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If RWST level lowers to less than 20%, the SI system should be aligned for cold leg recirculation using 2-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.

* *

1.___ CHECK SI VALVE ALIGNMENT: — ☐ Manually align valves.

a) CHG pump suction from RWST MOVs - OPEN

- ☐ 2-CH-MOV-2115B
- ☐ 2-CH-MOV-2115D

b) CHG pump suction from VCT MOVs - CLOSED

- ☐ 2-CH-MOV-2115C
- ☐ 2-CH-MOV-2115E

c) CHG line isolation MOVs - CLOSED

- ☐ 2-CH-MOV-2289A
- ☐ 2-CH-MOV-2289B

d) HHSI to cold legs - OPEN

- ☐ 2-SI-MOV-2867C
- ☐ 2-SI-MOV-2867D

e) LHSI suction from RWST - OPEN

- ☐ 2-SI-MOV-2862A
- ☐ 2-SI-MOV-2862B

Reference ND-91-LP-2, page 20 showing part 1 and part 2 correct answer.

d. SI MOV open/close seal-in functions:

(1) Both Units: 1(2)-SI-MOV-1(2)890A/B (LHSI to Hot legs MOVs) and 1(2)-SI-MOV-1(2)864A/B (LHSI to Cold Leg MOVs) , 1(2)-SI-MOV-1(2)842 (HHSI to Cold Leg, Alternate header MOV), and 1(2)SI-MOV-1(2)869 A/B (HHSI to Hot leg MOVs), throttle in Open or Close direction (no seal-ins)

(2) 1(2)-SI-MOV-1(2)867C/D (HHSI to Cold Leg MOVs), have no seal-in for the Open or Close direction when using the control switch. When an SI occurs, the SI signal closes a contact in the Open contactor circuit to open the valves fully on an SI signal. The contact remains closed until SI is reset; at minimum, 60 seconds must elapse before SI may be reset.

(3) The seal-in function of the HHSI MOVs was removed to enhance system reliability during a case where only a single train has power during the recirculation phase of SI, and flow must be shifted from Cold leg to Hot leg injection, or from Hot back to Cold leg injection. Further discussion will be provided on background and procedures

Reference, NCRODP-53-S, Containment Spray (DISTRACTOR)

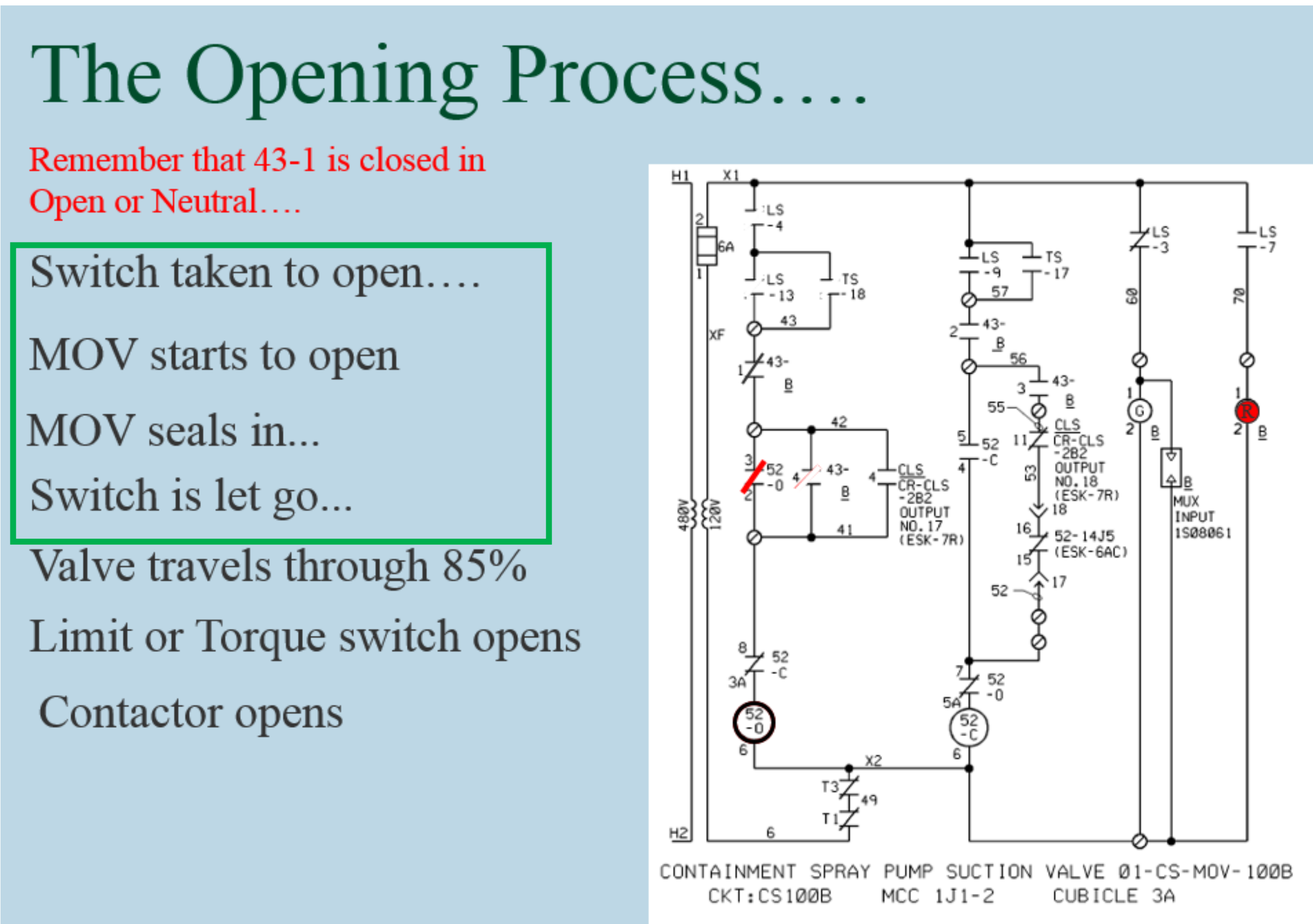
Valve Instrumentation and Controls

CS pump suction isolation valves. The CS pump suction Isolation Valves MOV-CS-100A and B are controlled by OPEN-CLOSE handswitches on Benchboard 1-1. Open and shut lights on the handswitch indicate the valve position. Each valve is normally open but receives an open signal upon initiation of a CLS to ensure its open status. The valve cannot be shut unless the respective CS pump breaker is open, racked in, and the CLS signal is reset.

CS pump discharge isolation valves. The CS pump discharge Isolation Valves MOV-CS-101A/B/C/D are controlled by OPEN-CLOSE handswitches

NCRODP-53-S Page 20 03/19/10

Reference; ND-91-LP-5, CS, Slide 35 showing Opening process (seal-in Open)



K/A Number: 075G2.1.30: Circulating Water: Ability to locate and operate components, including local controls.

Level: RO Tier #: 2 Group #: 2 IR – RO: 4.4 IR-SRO: 4.0

Proposed Question: 54

CW pumps are remotely started from the ___1)___ RO console, with the traveling screen in ___2)___ speed, in accordance with OP-48.1.1, Starting Any Circulating Water Pump.

A. 1) Unit 1
   2) slow

B. 1) Unit 2
   2) fast

C. 1) Unit 2
   2) slow

D. 1) Unit 1
   2) fast

Proposed Answer: D.

Explanation: CW pumps are started remotely using soft controls on the Unit 1 PCS with the traveling screen running in fast speed IAW OP-48.1.1, Starting any Circulating Water Pump.

Technical Reference: OP-48.1.1, Starting Any Circulating Water Pump

Reference Provided to Applicant: No

Learning Objective: ND-89.5-LP-1, Circulating Water System, Objective B, B. Describe the components associated with the circulating water system: Circ water pumps

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.7 / 45.7)

Comments:

K/A Match: Question matches K/A. Candidate must recall location CW is operated from and speed of screen when CW pump started.

Distractor Analysis:

A. Incorrect. Part 1) is correct. Part 2) is incorrect, procedure step 5.2.6 directs the screen drive to be placed in fast for pump start. Plausible since the Screen has a slow and fast speed.

B. Incorrect. Part 1) is incorrect, CW pumps are started from the Unit 1 RO PCS ONLY. Plausible since Candidate may recall that the CW PCS screen may be accessed from the Unit 2 RO console for monitoring. Part 2) is correct.

C. Incorrect. Part 1) is incorrect, CW pumps are started from the Unit 1 RO PCS ONLY. Plausible since Candidate may recall that the CW PCS screen may be accessed from the Unit 2 RO console for monitoring. Part 2) is incorrect, procedure step 5.2.6 directs the screen drive to be placed in fast for pump start. Plausible since the Screen has a slow and fast speed.

D. Correct. Both Parts 1) and 2) are correct.

Excerpt from OP-48.1.1, Starting Any Circulating Water Pump

5.2.6 Check or place the Travelling Water Screen Drive Slow-Off-Fast switch in FAST for the pump to be started AND check rotation. (✓)

Excerpt from NCRODP-12(S), Circulating Water System Module

Circulating Water System ***Low-Level Intake Structure***

head; and each is driven by a 2000 HP, 4160V ac, squirrel-cage induction motor.

No automatic start features are provided on these pumps. They are manually controlled remotely from the Unit 1 (U1) Plant Computer System (PCS) Operator Work Stations (OWS) in the MCR or from the Low-Level Control House. Manual start permissive features must be satisfied before the pumps can be started. Manual stop and automatic trip features are also provided. These features are described in the Instrumentation and Controls section.

K/A Number: 076K1.19: Service Water: Knowledge of the physical connections and/or cause- effect relationships between the SWS and the following systems: SWS emergency heat loads.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.6 IR-SRO: 3.7

Proposed Question: 55

Initial Conditions:

- Unit 1 is operating at 100%.
- Unit 1 "A" High Level stop log for maintenance.

Current Conditions:

- A Large Break LOCA occurs inside containment on Unit 1.
- All ESF equipment respond as designed.

Which one of the following describes the RSHXs that have SW flowing through them?

A. RSHX A and B.

B. RSHX C and D.

C. RSHX A and D.

D. RSHX B and C.

Proposed Answer: C

Explanation: With a stop log placed in the 1A High Level, RSHXs B and C will be isolated. Therefore following a Hi-Hi CLS RSHX A and D will have flow through them.

Technical Reference: ND-89.5-LP-2, Service Water Lesson Plan.

Reference Provided to Applicant: No

Learning Objective: ND-89.5-LP-2, Service Water, Objective C; "State in general the flowpath of service water from the high level intake to the discharge tunnel for the systems served by service water.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Comments:

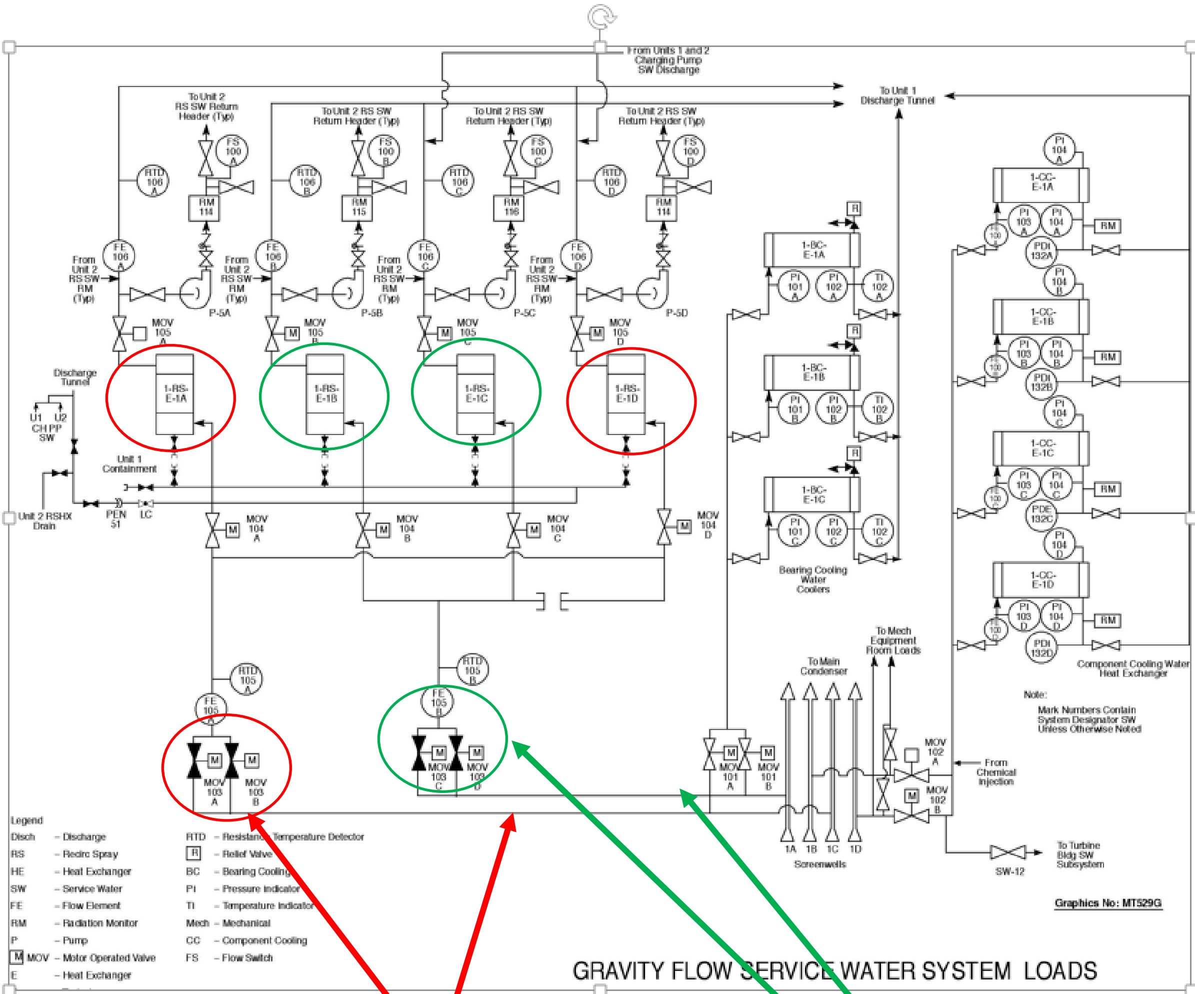
K/A analysis: K/A specifies knowledge of the physical connections and/or Cause-effect relationships between the Service Water System and the SWS emergency heat loads. This question tests the interconnections between the Service Water system supply (High level intake) and the RSHXs which are the emergency heat loads.

Distractor analysis:

A. Incorrect. Plausibility based on candidate confusing high level intake supply with the RSHXs that are supplied. This is common mistake since the 1A high level then flows through 1-SW-MOV-103 A and B, and then flows through RSHX B and C.
B. Incorrect. Plausibility based on candidate confusing high level intake supply with the RSHXs that are supplied. This is common mistake since the 1A high level then flows through 1-SW-MOV-103 A and B, and then flows through RSHX B and C.
C. Correct.
D. Incorrect. Plausibility based on candidate confusing high level intake supply with the RSHXs that are supplied. This is common mistake since the 1A high level then flows through 1-SW-MOV-103 A and B, and then flows through RSHX B and C.

Reference: ND-89.5-H-2.4, Service Water, Gravity Flow SW loads.

ND-89.5-H-2.4



High level 1C flows thru MOV 103A and B,then RSHXs A and D (Correct answer).	High Level 1 A isolated, MOV 103C and D, RXHXs B and C have no SW flow.

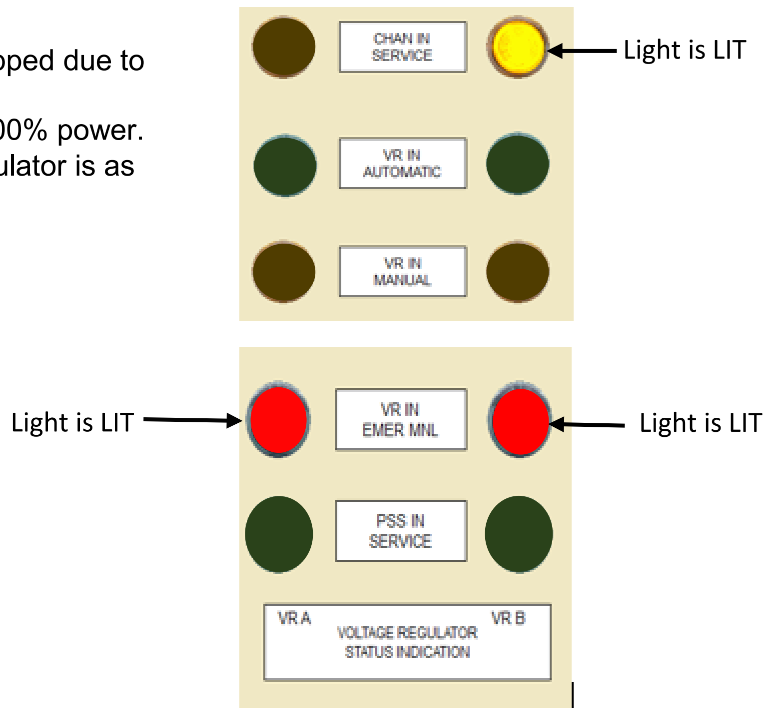
NRC APPROVED. K/A Number: 077AK1.03: Generator Voltage and Electric Grid Disturbances / 6: Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: Under-excitation.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.3 IR-SRO: 3.4

Proposed Question: 56

The following conditions exist:

- North Anna Unit 1 has tripped due to 500kV grid instability.
- Both Surry Units are at 100% power.
- Surry Unit 2 Voltage Regulator is as shown.



Which ONE of the following completes the statement below?

1) Unit 2 Voltage Regulator ____ prevent the operator from exceeding excitation limits.
2) In accordance with 0-AP-10.18, Response to Grid Instability, the Systems and Marketing Operations Center (SOC/MOC) ____ required to be notified

A. 1) will not
   2) is

B. 1) will not
   2) is not

C. 1) will
   2) is

D. 1) will

2) is not

Proposed Answer: A.

Explanation: 1) Per 2-OP-TM-001, Turbine-Generator Startup to 20%-25% Turbine Power P&L 4.19, there are no excitation limits except those imposed by the operator when the VR is in the Emergency Manual mode. 2) In accordance with 0-AP-10.18, Response to Grid Instability, step 9, the Systems Operations Center is required to be notified anytime the voltage regulator is not in Auto. The RO is normally the individual who contacts SOC/MOC so they need to understand who to contact.

Technical Reference: 2-OP-TM-001, Turbine-Generator Startup to 20%-25% Turbine Power, Rev 58.

0-AP-10.18, Response to Grid Instability, Rev. 21

Reference Provided to Applicant: No

Learning Objective: ND-90.1-LP-6, Voltage Regulator, Objective C, Describe the controls, indications, and alarms associated with the Main Generator.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must select response based on knowledge of Unit 2 voltage regulator operation to control generator voltage to prevent over- and under-excitation. Operational implications are also tested by the question regarding notification to the SOC/MOC.

Distractor Analysis:

A. Correct. Both Parts 1) and 2) are correct.

B. Incorrect. Part 1) is correct. Part 2) is incorrect. The SOC and MOC are required to be notified when the Voltage Regulator is NOT in AUTO per 0-AP-10.18. This is Plausible because it is a common misconception regarding which organization should be notified, and there are circumstances when the SOC and/or MOC are not required to be notified (Ref. Transmission agreement).

C. Incorrect. Part 1) is incorrect per 2-OP-TM-001, P&L 4.19. Plausible if Candidate confuses effect of Generator Voltage Regulator being in Manual vs. Emergency Manual. Part 2) is correct.

D. Incorrect. Part 1) is incorrect per 2-OP-TM-001, P&L 4.19. Plausible if Candidate confuses effect of Generator Voltage Regulator being in Manual vs. Emergency Manual. Part 2) is incorrect. The SOC and MOC are required to be notified when the Voltage Regulator is NOT in AUTO per 0-AP-10.18. This is Plausible because it is a common misconception regarding which organization should be notified, and there are circumstances when the SOC and/or MOC are not required to be notified (Ref. Transmission agreement).

Reference 2-OP-TM-001, Turbine Generator startup to 20-25%, P&Ls

DOMINION
Surry Power Station

2-OP-TM-001
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______ 4.17 The instantaneous change of steam temperature to the LP Turbine inlet(s) should be limited to a maximum of 100°F.

______ 4.18 The maximum allowable LP Turbine inlet temperature change must be limited to 250°F/hr.

______ 4.19 The Voltage Regulator has redundant channels. When one channel initiates excitation outside of the generator capability curve or loses sensing quantities, that channel is removed from service and the redundant channel is placed in service. If the redundant channel also initiates excitation outside the generator capability curve or loses sensing quantities, the regulator is automatically placed in Manual. In Manual mode, the operator can control the voltage regulation, however, the automatic functions are lost except that the unit maintains a minimum excitation limit which the operator can not exceed. If the units fail in the manual mode, the unit is automatically placed in Emergency Manual. In Emergency Manual mode there are no excitation limits except those imposed by the operator. In this case, the units will have to be manually reset at the voltage regulator cabinet.

Reference 0-AP-10.18, step 9. Shows Part 2 correct answer.

NOTE:
- Fluctuations in Megawatts (spikes and dips) or Megawatt swings indicate grid disturbance. Voltage Regulator failures have little to no affect on Megawatt output.
- The Voltage Regulators should be kept in Auto during grid disturbances.

9. ___ CHECK BOTH UNITS VOLTAGE REGULATORS - IN AUTO

☐ Place Voltage Regulator(s) in Auto if possible.

IF Voltage Regulator NOT able to be put in Auto, THEN notify the following within 30 minutes:

☐ System Operations Center (SOC)

☐ Market Operations Center (MOC)

Reference Nuclear Switchyard Interface Agreement Shows example of when NOT to contact. Supports Part 2 distractor.

4.4.2 The Plant Control Room shall notify the MOC of the following:

PJM requires 30-minute notification prior to any of the following planned events: outage, derate, generator connecting to or disconnecting from the grid, even if the events are previously scheduled. The 30-minute requirement applies to the event start and end (e.g., before starting a derate and before ramping up above station service to put power on the grid). Notify MOC in advance of the planned events to allow them to make the 30-minute notification to PJM. For any unplanned event, notify as soon as practicable.

Do **NOT** communicate any transmission data to MOC. If generator limitation is due to transmission-related activity, simply pass on the impact to MW output and the expected duration. The cause can be stated as non-generator related.

ES 401-9 COMMENTS WITH CORRECTIONS SHOWN IN RED, 3/28/16

This question only requires knowledge of VAR limit. There is nothing here about generator voltage or electric grid disturbances.

Changed question to deal with generator voltage regulator control during grid disturbances.

ES 401-9 Comments, 3/30/16: The second part is fine. The first part does not discriminate. It is reasonable to guess the left is A and right is B

Switched Part 2 to Part 1. Asked new Part 2 to better test operational implications, by asking if it is required to notify SOC.

K/A Number: 078K3.03: Instrument Air: Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Cross-tied units.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.0 IR-SRO: 3.4

Proposed Question: 57

Initial Conditions:

- Both units are at 100% power.
- Instrument and Service Air headers are in a normal alignment.
- 1-IA-C-1, Instrument Air Compressor is tagged out for repairs.
- Annunciator 1B-E5, SA COMPR TRBL alarms. An Operator is dispatched to investigate.

Current Conditions:

- The Operator reports that 1-SA-C-4A, SA Compressor has tripped and the other SA Compressors have failed to start.
- MCR Service Air and Instrument Air Indications are 98 psig and lowering.

With NO Operator actions, which ONE of the following statements is correct?

1) 2-IA-C-1, Instrument Air Compressor will auto start at ____ psig.
2) After 2-IA-C-1 starts Unit 1 Instrument Air _____ re-pressurize.

A. 1) 90
   2) will

B. 1) 80
   2) will not

C. 1) 80
   2) will

D. 1) 90
   2) will not

Proposed Answer: D

Explanation: Initially the Unit 1 and Unit 2 Instrument Air systems are supplied from the Service Air Compressors. Service Air is cross-tied with Instrument Air, and the Unit 1 and Unit 2 Instrument Air systems are operated split. 1) Once the SA Compressors trips, MCR Instrument Air and Service Air pressures start to lower. At an Instrument Air header pressure of 90 psig, the Unit 2 Instrument Air compressor, 2-IA-C-1 will auto start (Unit 1 is tagged out). After 2-IA-C-1 auto starts the Unit 2 Instrument air header will re-pressurize. The Unit 1 Instrument air compressor is tagged out, and because the Instrument Air headers are normally operated split, Unit 1 Instrument air header will not re-pressurize.

Technical Reference: ARP 1B-E6, IA LO HDR PRESS/IA COMPR1 TRBL, Revision17

Reference Provided to Applicant: No

Learning Objective: ND-92.1-LP-1, Station Air Systems, Objective B; Determine the system flowpaths and components associated with the Instrument Air System.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7 / 45.6)

Comments:

K/A Analysis: K/A requires knowledge of the effect of a malfunction of the Instrument Air System with respect to cross-tied units. Instrument Air is cross-tied with Service Air, but Unit 1 Instrument air is split from Unit 2 Instrument air. This question requires knowledge of this fact by posing a scenario that results in loss of Service Air Compressors.

Distractor Analysis:

A. Incorrect. 1) Correct. 2) Incorrect because in this case the IA cross tie valves 1-IA-44, and 2-IA-44 are closed. Plausible if the candidate confuses which crossties are normally open.
B. Incorrect. 1) Incorrect. Incorrect because the auto start setpoint for the Instrument air compressor is 90 psig. Plausible because 80 psig is the alarm setpoint for the alarm, 1B-E6, IA LO HDR PRESS/IA COMPR TRBL. 2) Correct.
C. Incorrect. 1) Incorrect. Incorrect because the auto start setpoint for the Instrument air compressor is 90 psig. Plausible because 80 psig is the alarm setpoint for the alarm, 1B-E6, IA LO HDR PRESS/IA COMPR TRBL. 2) Incorrect because in this case the IA cross tie valves 1-IA-44, and 2-IA-44 are closed. Plausible if the candidate confuses which crossties are normally open.
D. Correct.

Reference, ND-92.1-LP-2 (page 13) Shows auto start of IA compressors.

a. Instrument Air compressors

(1) The compressors are controlled by a local control panel. When started manually, the compressor motor runs continuously, the compressor loads and unloads at 100 and 110 psig, respectively. In AUTO, the compressor starts if pressure reaches 90 psig; load and unload setpoints are the same.

Reference 1B-E6, IA LO HDR PRESS/IA COMPR 1 TRBL setpoint

NUMBER	PROCEDURE TITLE	REVISION
1B-E6	IA LO HDR PRESS/IA COMPR 1 TRBL	17
		PAGE 1 of 5

REFERENCES 1B-38

1) UFSAR - Sections 9.8.2 and 10.3.9.3
2) 11448-ESK-6DA, 10B, 10AE
3) 1-DRP-005, Instrumentation Setpoints
4) DCP-86-03-C, IA Dryer Modification
5) DC SU-08-0024, Compressed Air Modification

PROBABLE CAUSE

1) Alarm actuates when 1-IA-PS-120 senses IA header pressure less than or equal to 80 PSIG.

Reference, ND-92.1-LP-2 (page 9). Shows that SA main source for IA and that IA headers normally split but can be cross-tied.

c. The main source of air to the Instrument Air System is from the Service Air System.

d. Each unit has a compressor (similar to service air compressor with lower capacity), an air receiver, and an air dryer.

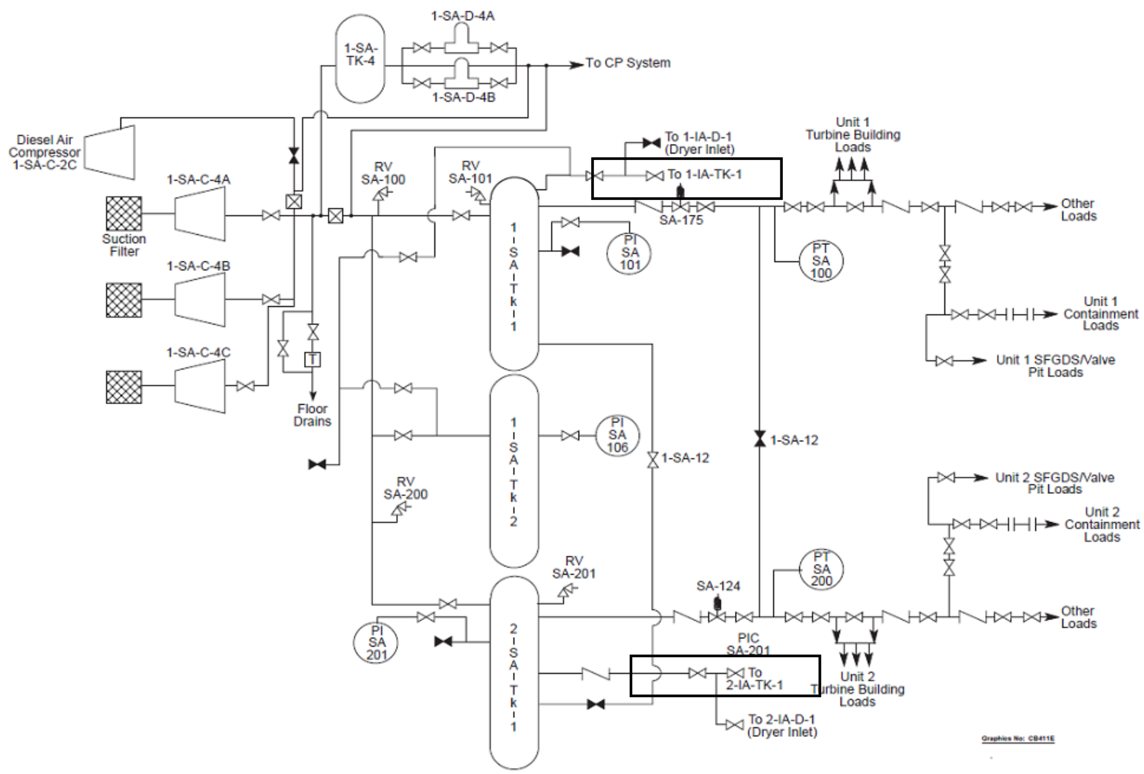
e. Air flows from the compressor to the air receiver.

f. Outlet of each receiver is directed to a desiccant type air dryer which removes moisture by air flowing through the desiccant vessels, absorbing the water vapor. Also particulate matter is removed by inlet and outlet filters.

g. From the dryer, the compressed air is directed to the various areas served.

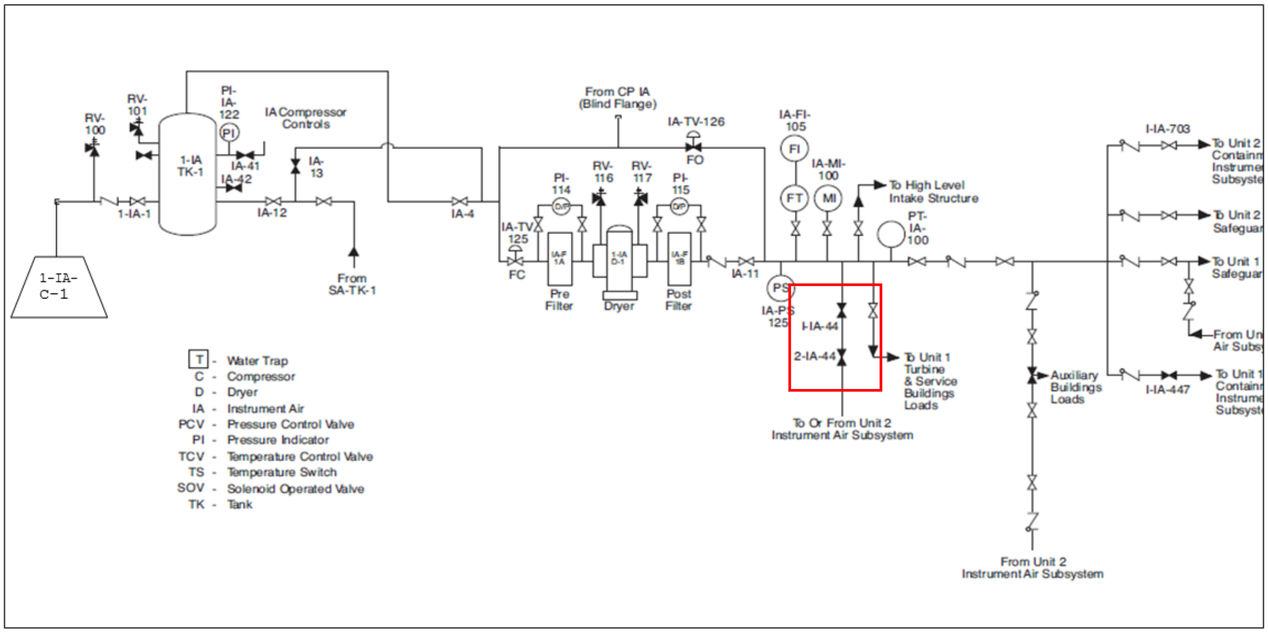
h. The two unit's instrument air supplies can be cross-tied, if necessary, by opening 1-IA-44 and 2-IA-44.

Reference ND-92.1-LP-2, SA System, Note Normal supply to IA system.



Service Air System

Reference ND-92.1-LP-2, IA System, Note cross-tied normally closed to opposite units IA system.



ND-92.1-H-1.3

Instrument Air Subsystem

K/A Number: 079K4.01: Station Air: Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: Cross-connect with IAS.

Level: RO Tier #: 2 Group #: 2 IR – RO: 2.9 IR-SRO: 3.2

NRC NOTE: (B Caballero)Same K/A as on 2014 exam (question 56). I encouraged them to attempt to write a question or modify the Q from the 2014 exam. They have installed a new instrument air system at the site, which is more robust and simpler than the previous system which may make question writing difficult. This is just a heads up that the exam writer may ask for relief from this K/A in the future.

Proposed Question: 58

Current Conditions:

- Both Units have experienced a Loss of All AC Power.
- An operator has been dispatched to start and align the Temporary Diesel Driven Air Compressor.

The __1)__ systems for both units are supplied by the Temporary Diesel Driven Air Compressor. The Temporary Diesel Driven Air Compressor __2)__ have its own air dryers.

A. 1) Instrument Air and Service Air
2) does

B. 1) Instrument Air and Service Air
2) does not

C. 1) Instrument Air ONLY
2) does

D. 1) Instrument Air ONLY
2) does not

Proposed Answer: A. 1) Instrument Air and Service Air; 2) does

Explanation: 1) The service air and instrument air systems are hard piped together. Both units' SA systems are normally crosstied, and independently provide IA for the respective unit. The Temporary Diesel Driven Air Compressor is provided to supply dry, filtered air to both units via 1-SA-223. When aligned, this will provide full SA and IA pressure for both units. 2) Temporary Air Dryers integral to the compressor skid are used to dry and filter the air. The normal IA dryers will bypass on a Loss of All AC Power.

Technical Reference: 1-ECA-0.0, Rev 40; 0-OP-SA-001. Rev 18.

Reference Provided to Applicant: No

Learning Objective: ND-92.1-LP-1, Station Air Systems, Objective A, Describe the system flowpaths and components associated with the Service Air System. ND-92.1-LP-1, Station Air Systems, Objective B, [Describe the system flowpaths and components associated with the Instrument Air System. SOER 88-01, Recommendation 2&3].

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

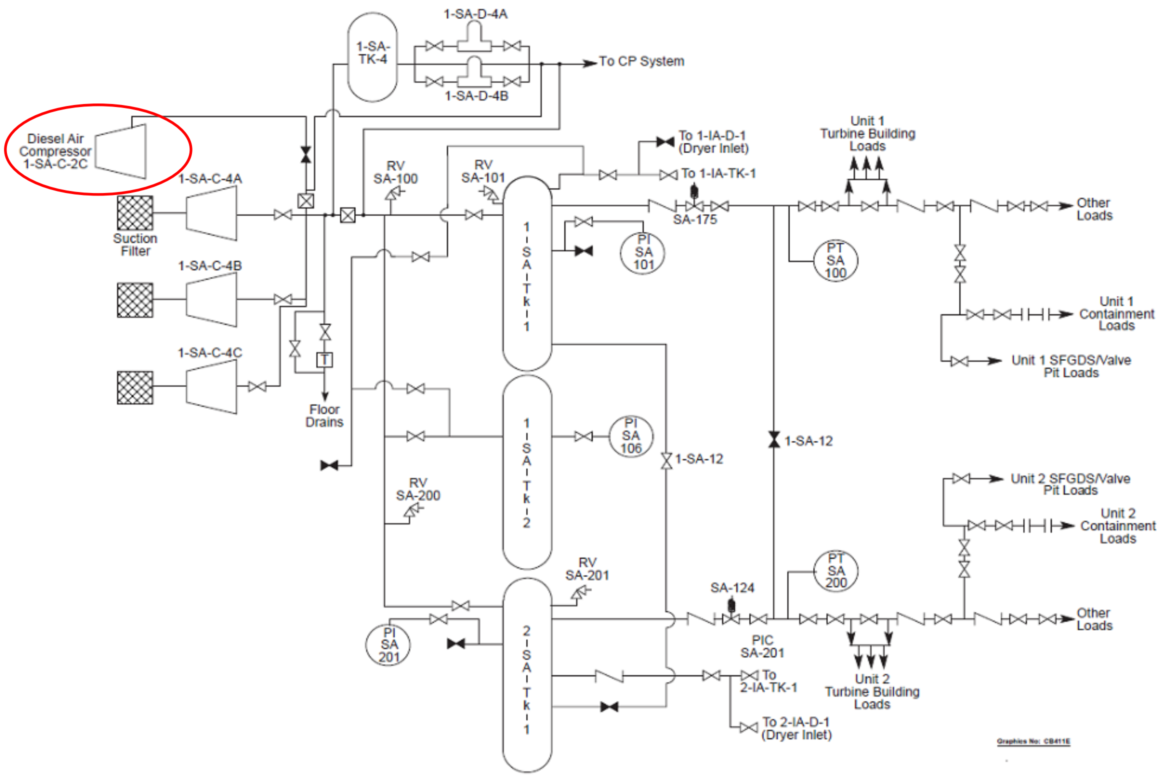
10 CFR Part 55 Content: (CFR: 41.7)

Comments:

Distractor Analysis:

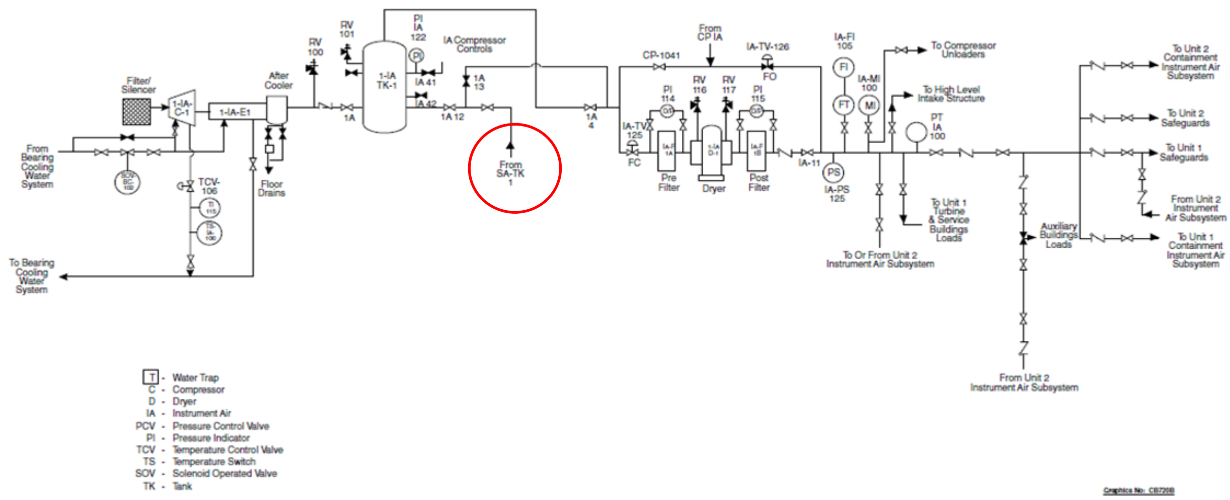
A. Correct.
B. Incorrect - 1) Correct. 2) Incorrect. Plausible if the candidate confuses the normal IA dryers with the dryers provided with the temporary air compressor.
C. Incorrect – 1) Incorrect. Instrument Air is normally provided via the SA system. Plausible if the candidate assumes that IA ONLY is pressurized since it is the higher priority system. 2) Correct.
D. Incorrect – 1) Incorrect. Instrument Air is normally provided via the SA system. Plausible if the candidate assumes that IA ONLY is pressurized since it is the higher priority system. 2) Incorrect. Plausible if the candidate confuses the normal IA dryers with the dryers provided with the temporary air compressor.

Reference for SA drawing from ND-92.1-LP-1.



Service Air System

Reference showing Instrument Air drawing. Normally SA supplies Instrument Air. With lowering SA Tank pressure, IA will also lower (shown in red) causing an alarm.



Instrument Air Subsystem

K/A Number: 086K1.03: Fire Protection: Knowledge of the physical connections and/or cause effect relationships between the Fire Protection System and the following systems: AFW system.

Level: RO Tier #: 2 Group #: 2 IR – RO: 3.4 IR-SRO: 3.5

Proposed Question: 59

The Crew is responding to a loss of secondary heat sink in accordance with 1-FR-H.1, Response to Loss of Secondary Heat Sink, and must align a low pressure water source to feed SGs.

- Emergency Condensate Storage Tank, 1-CN-TK-1 is at 10% and lowering rapidly.
- 1-CN-150, Emerg. CST Inlet is mechanically stuck in CLOSED position.
- AFW Booster pumps, 1-FW-P-4A, and 1-FW-P-4B are both tagged out for corrective maintenance.
- The Unit 2 RO reports he is unable to open AFW Crosstie MOVs, 1-FW-MOV-160A and B.

Which ONE of the following describes the required low pressure source that should be aligned in accordance with 1-FR-H.1, Attachment 1, Alternate AFW Sources?

A. 1-CN-TK-2, Condensate Storage Tank.

B. 1-CN-TK-3, Emergency Makeup Tank.

C. Fire Protection System using the Diesel driven fire pump, 1-FP-P-2.

D. 2-CN-TK-2, Condensate Storage Tank.

Proposed Answer: C

Explanation: Per 1-FR-H.1, Attachment 1, Alternate AFW Sources, the only remaining source that can be aligned is the Fire Protection System using the Diesel driven fire pump, 1-FP-P-2. 1-CN-TK-2, or 2-CN-TK-2, cannot be aligned because it requires 1-CN-150 to be opened and the valve is mechanically stuck in the closed position. 1-CN-TK-3, cannot be aligned because it requires the AFW booster pumps to be available and they are both tagged out. And finally Attachment 1 of 1-FR-H.1 specifies the Fire protection system using the Diesel Driven Fire pump.

Technical Reference: 1-FR-H.1, Response to Loss of Secondary Heat Sink. Revision 37.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-41, FR-H.1; Objective C, Given a copy of FR-H.1, Response to Loss of Secondary Heat Sink, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Comments:

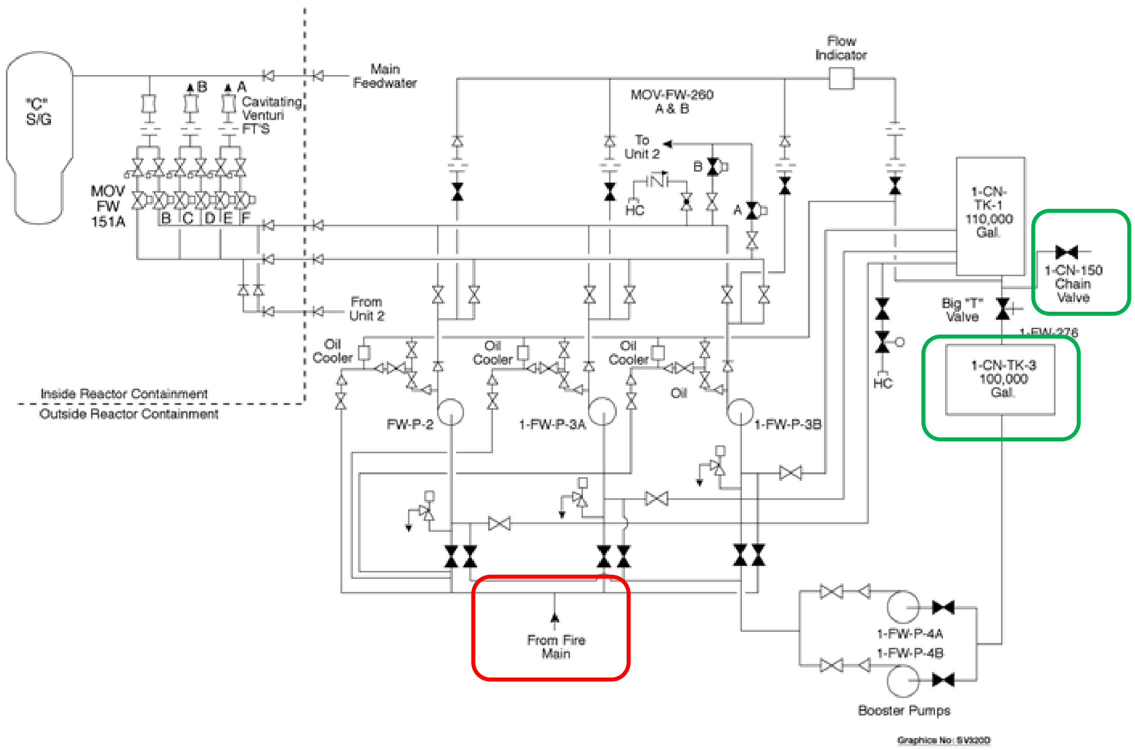
K/A analysis: Question requires knowledge of FR-H.1, Attachment 1 including knowledge of Fire Protection system alignment to AFW.

Distractor Analysis:

A. Incorrect. Plausible because this is a possible lineup, but not with present conditions. 1-CN-TK-2 cannot be aligned because it requires 1-CN-150 to be opened and the valve is mechanically stuck in the closed position.
B. Incorrect. Plausible because this is a possible lineup, but not with present conditions. 1-CN-TK-3, cannot be aligned because it requires the AFW booster pumps to be available and they are both tagged out.
C. Correct.
D. Incorrect. Plausible because this is a possible lineup, but not with present conditions. 1-CN-TK-2 and 2-CN-TK-2 are normally cross-tied, but 2-CN-TK-2 cannot be aligned because it also requires 1-CN-150 to be opened and the valve is mechanically stuck in the closed position.

Reference, ND-89.3-LP-4, AFW Drawing.

▪ ND-89.3-H/T-4.2¶



AUXILIARY FEEDWATER SYSTEM ¶

Reference 1-FR-H.1 Steps 1 and 2 (Distractors A and B)

NUMBER 1-FR-H.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 37	ALTERNATE AFW SOURCES	PAGE 1 of 3

Select one of the following methods as an alternate source of Auxiliary Feedwater.

1. Makeup to 1-CN-TK-1 from 1-CN-TK-2

☐ To fill the ECST from the 300,000 gallon tank, locally open 1-CN-150.

2. Alignment of AFW Booster Pump(s)

___ a. Check 1-CN-TK-3 level greater than 20%.

___ b. Locally open booster pump discharge to available AFW pump(s):

1-FW-P-2	1-FW-P-3A	1-FW-P-3B
☐ 1-FW-283	☐ 1-FW-284	☐ 1-FW-285

___ c. Locally check selected booster pump suction valve open:

1-FW-P-4A	1-FW-P-4B
☐ 1-FW-277	☐ 1-FW-280

___ d. Locally open selected booster pump discharge valve:

1-FW-P-4A	1-FW-P-4B
☐ 1-FW-279	☐ 1-FW-282

___ e. Locally start aligned booster pump(s).

Reference 1-FR-H.1 Step 4 Showing correct answer.

NUMBER 1-FR-H.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 37	ALTERNATE AFW SOURCES	PAGE 3 of 3

4. Fire Water Alignment to AFW pumps

___ a. Locally close telltale drain valve, 1-FW-119.

___ b. Locally open fire main isolation valves:

☐ • 1-FW-120
☐ • 1-FW-185

___ c. Locally start diesel driven fire pump.

NOTE: The following steps should be used as needed to align fire water to each available AFW pump one at a time and the MCR informed after each pump alignment is complete.

___ d. Locally open fire water suction valves to available AFW pump(s):

1-FW-P-2	1-FW-P-3A	1-FW-P-3B
☐ 1-FW-154	☐ 1-FW-169	☐ 1-FW-184

___ e. Locally check AFW pump suction pressure.

___ f. Locally close normal AFW pump suction valves:

1-FW-P-2	1-FW-P-3A	1-FW-P-3B
☐ 1-FW-153	☐ 1-FW-168	☐ 1-FW-183
☐ 1-FW-283	☐ 1-FW-284	☐ 1-FW-285

K/A Number: 103K1.02: Containment: Knowledge of the physical connections and/or cause effect relationships between the containment system and the following systems: Containment isolation/containment integrity.

Level: RO Tier #: 2 Group #: 1 IR – RO: 3.9 IR-SRO: 4.1

Proposed Question: 60

Initial Conditions:

- Unit 1 was at 100% power.
- Loss of coolant accident has occurred.
- Peak CTMT pressure reached 19 psia.

Current Conditions:

- CTMT pressure is 16 psia and lowering slowly.
- All systems functioned as designed.
- The Operators have not manually reset any safety features.

For this condition, which ONE of the following gives the status of the CTMT isolation signals?

	Phase I (SI)	**Phase II (Hi CLS)**	**Phase III (Hi Hi CLS)**
A.	Actuated	Actuated	NOT Actuated
B.	NOT Actuated	Actuated	Actuated
C.	Actuated	Actuated	Actuated
D.	NOT Actuated	NOT Actuated	NOT Actuated

Proposed Answer: A

Explanation: With containment pressure rising from ~10.5 psia to a peak of 19 psia with a LOCA in progress, the candidate could assume that two SI signals have been generated; SI and Hi CLS. With containment pressure above 17.7 psia, Hi CLS - phase 2 containment isolation would occur. Phase III isolation occurs at 23 psia.

Technical Reference:

Reference Provided to Applicant: No

Learning Objective: ND-91-LP-3, SI Operations, Objective A, List the five (5) Safety Injection actuation signals, including setpoints, coincidence, and purposes. ND-91-LP-5, CS System, Objective A, Describe the operations of the Consequence Limiting Safeguards System, including coincidence, logics, setpoints and actions occurring upon system actuation.

Question Source: Bank (LCONT0007, Attached)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Comments:

K/A Match Analysis: Question matches K/A, Candidate must recall containment pressure setpoints and determine whether phase I, II, or III containment isolation have/has occurred

Distractor Analysis:

A. Correct. Phase I and II containment isolation would have occurred.

B. Incorrect. Plausible if Candidate disregards stem where signals have not been reset (SI) and fails to recall setpoint of Hi-Hi CLS actuation (phase III).

C. Incorrect. SI and HI CLS are actuated; HI-HI has not. Plausible if Candidate fails to recall HI-HI CLS setpoint.

D. Incorrect. SI and HI CLS are actuated, HI-HI-CLS is not. Plausible if Candidate fails to consider that two signals available to actuate SI, and relies on current containment pressure vice peak pressure reached in determine HI and HI-HI CLS have not actuated.

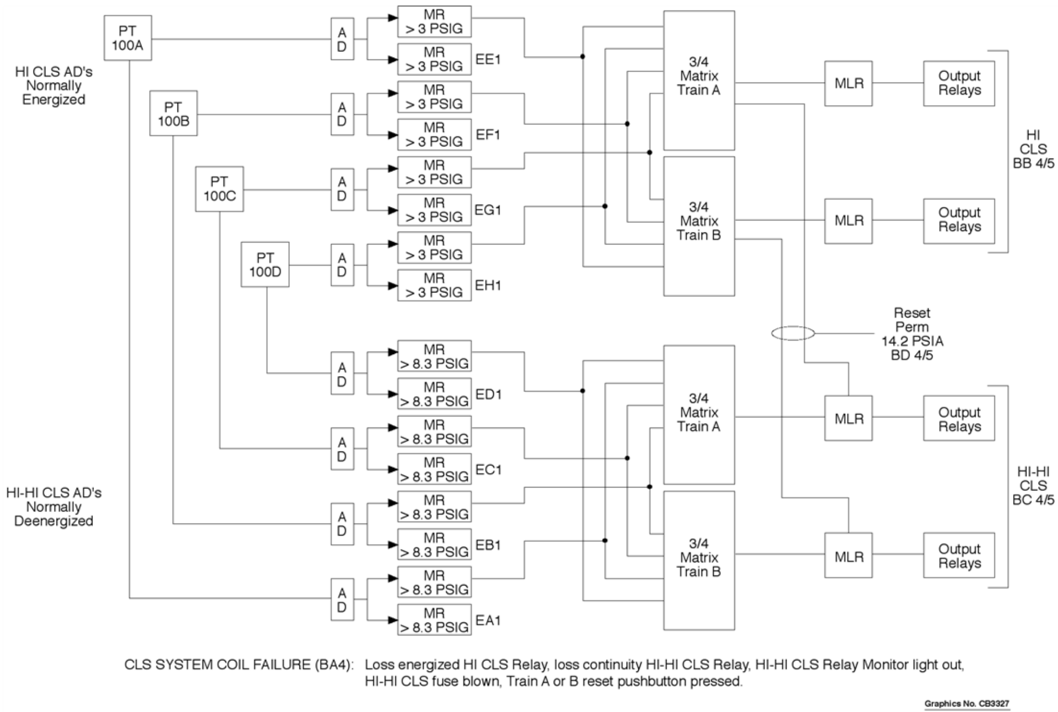
Excerpt from SI Operations LP:

A. Safety Injection Actuation Signals

HI CLS Initiation

LOW PRESSURIZER PRESSURE

From CS LP:



CLS CIRCUIT

7 **ID: LCONT007** **Points: 1.00**

Unit 1 was at 100% power with all systems in normal configuration when a RCS loss of coolant accident occurs. Peak CTMT pressure reached 19 psia and is currently 16 psia and reducing slowly. All systems functioned as designed. The Operators have not manually reset any safety features (SI, Hi CLS, Hi Hi CLS).

For this condition, which ONE of the following gives the status of the CTMT isolation signals?

A. Phase I (safety injection) CTMT is actuated; Phase II (Hi CLS) is actuated; Phase III (Hi Hi CLS) is NOT actuated.

B. Phase I (safety injection) CTMT is NOT actuated; Phase II (Hi CLS) is actuated; Phase III (Hi Hi CLS) is actuated.

C. Phase I (safety injection) CTMT is actuated; Phase II (Hi CLS) is actuated; Phase III (Hi Hi CLS) is actuated.

D. Phase I (safety injection) CTMT is NOT actuated; Phase II (Hi CLS) is NOT actuated; Phase III (Hi Hi CLS) is NOT actuated.

Answer: A

K/A Number: E11EG2.4.6: Loss of Emergency Coolant Recirc. / 4: Knowledge symptom based EOP mitigation strategies.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.7 IR-SRO: 4.7

Proposed Question: 61

Given the following conditions:

- A LOCA occurred on Unit 1.
- The team is performing 1-ECA-1.1, Loss of Emergency Coolant Recirculation.
- The team is SLOWLY depressurizing intact SGs to inject the SI accumulators.

Which ONE of the following identifies the basis for stopping the SG depressurization at 200 psig?

A. Allows time for refilling the RWST.

B. Prevents injection of nitrogen into the RCS.

C. Allows the RCP to remain running.

D. Extends the time until the accumulator fully depletes.

Proposed Answer: B

Explanation: SGs are depressurized relatively slowly such that accumulator water injection is minimized, extending the time to depletion of the accumulators. SG depressurization is stopped at 200 psig to **prevent an uncontrolled nitrogen injection into the RCS**, which would disrupt natural circulation flow through the core.

Technical Reference: 1-ECA-1.1, Loss of Emergency Coolant Recirculation. Rev. 39. ND-95.3-LP-20, ECA-1.1

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-20, ECA-1.1, Objective B; Given a copy of ECA-1.1, Loss of Emergency Coolant Recirculation, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: Modified Bank: changed answer (EOP0418)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)

Comments:

K/A Analysis: Question requires knowledge of mitigation strategies by testing the candidate's understanding of the basis behind key EOP steps.

Distractor Analysis:

A. Incorrect. Stopping depressurization at 200 psig will not significantly affect time for RWST refill. Plausible because this is one of major goals of ECA-1.1, and is associated with other steps contained in ECA-1.1 (step 6, Est one train of SI flow).
B. Correct.
C. Incorrect. This is not the reason for stopping depressurization at 200 psig. Plausible because RCP operation is desirable and candidate can rationalize that by stopping the depressurization at 200 psig, this will maintain pressure high enough to prevent the formation of steam bubble in vessel.
D. Incorrect. This is not the reason for stopping depressurization. Plausible because this is the reason for controlling the rate of depressurization. Also this is plausible because this is in same step as the requirement to stop depressurizing SGs at 200 psig.

Bank Question, and Question Info. EOP0418

Given the following conditions:

- A LOCA occurred on Unit 1.
- The team is performing 1-ECA-1.1, Loss of Emergency Coolant Recirculation.
- One RCP is running.
- RWST level is <3%.
- The team is SLOWLY depressurizing intact SGs to inject the SI accumulators.

Which ONE of the following identifies the basis for depressurizing SGs **at a reduced rate**?

A.	Allows time for refilling the RWST.
B.	Extends the time until accumulator depletion.
C.	Allows the RCP to remain running.
D.	Prevents injection of nitrogen into the RCS.

Answer:	B

Question 59 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	155004
User-Defined ID:	EOP0418
Cross Reference Number:	
Topic:	EOP0418
Num Field 1:	
Num Field 2:	
Text Field:	1.00
Comments:	ND-95.3-LP-20B [Defect 7881]

Reference: ECA-1.1 step 6 and supporting basis (Distractor A)

6. ___ ESTABLISH ONE TRAIN OF SI FLOW:

☐ a) Check CHG pumps - ONLY ONE RUNNING

a) IF both LHSI pumps stopped due to CTMT Sump blockage, THEN do either of the following:

ND-95.3-LP-20 (page 10)

STEP 6: ESTABLISH ONE TRAIN OF SI FLOW.

a. The purpose of this step is to restrict injection flow to the capacity of one pump.

b. This step reduces the amount of flow injecting into the RCS in order to delay RWST depletion. It has the team establish only one train of SI flow, which is one Chg/SI pump and one LHSI pump (if pressure is below LHSI pp SOH).

Reference: ECA-1.1 step 18 and supporting basis (Distractor C)

18. ___ CHECK IF AN RCP SHOULD BE STARTED:

☐ a) All RCPs - STOPPED	a) Do the following: ☐ 1) Stop RCP(s) NOT required for normal PRZR spray. ☐ 2) Close spray valve(s) on stopped RCPs. ☐ 3) GO TO Step 19.
☐ b) RCS subcooling based on CETCs - GREATER THAN 30°F [85°F]	☐ b) GO TO Step 19.
c) Try to start an RCP to provide normal PRZR spray: ☐ 1) Establish conditions for starting an RCP IAW 1-OP-RC-001, STARTING AND RUNNING ANY RCP ☐ 2) Start RCP C	☐ c) Try to start RCPs A and B to provide normal spray. (must use 1-RC-PCV-1455A)

ND-95.3-LP-20 (page 22)

31. **<u>STEP 18</u>: CHECK IF AN RCP SHOULD BE STARTED.**

a. The purpose of this step is to determine the appropriate RCP operation.

b. Forced flow is the preferred mode of operation to allow for normal RCS cooldown and provide pzr spray. If RCPs have not been tripped, all but one are now stopped to minimize heat input into the RCS. The RCP started or left running should be the one that can provide normal pzr spray. If no RCP is running, RCS subcooling and RCP support conditions are checked before starting an RCP.

c. Depressurization of the RCS may generate a steam bubble in the vessel upper head region if no RCP is running. This bubble could rapidly condense during pump start, drawing liquid from the pzr and reducing RCS subcooling. If pzr inventory is not sufficient, level may drop off span. In addition, local flashing of reactor coolant could occur if subcooling is not

Reference: ECA-1.1 step 38 and supporting basis (Distractor D, AND Correct Answer)

38. ___ DEPRESSURIZE ALL INTACT SGs TO INJECT ACCUMULATORS:

☐ a) Dump steam to condenser to maintain appropriate RVLIS indication:

RCPs RUNNING	RVLIS INDICATION	
	Full Range	Dynamic Range
0	GREATER THAN 63%	-------
1	-------	GREATER THAN 36%
2	-------	GREATER THAN 51%
3	-------	GREATER THAN 82%

a) Dump steam from intact SG(s):

☐ • Manually use SG PORV(s).

OR

☐ • Locally use SG PORV(s) IAW Attachment 6.

☐ b) Check SG pressures - LESS THAN 200 psig

☐ b) RETURN TO Step 38a.

☐ c) Stop SG depressurization

ND-95.3-LP-20 (page 36)

STEP 38: DEPRESSURIZE ALL INTACT SGS TO INJECT ACCUMULATORS.

a. The purpose of this step is to inject the accumulators into the RCS by reducing pressure at a controlled rate.

b. As previously mentioned, the RCS will be saturated and therefore, RCS pressure will be approximately equal to SG pressure.

c. In this step, the intact SGs are depressurized (and thus the RCS) to inject the accumulators. Steam is dumped as necessary to maintain appropriate RVLIS indication.

d. **SGs are depressurized relatively slowly such that accumulator water injection is minimized, extending the time to depletion of the accumulators and preventing an uncontrolled nitrogen injection into the RCS. (rk)**

e. When SG pressures reach 200 psig, the accumulator water will have been discharged into the RCS and the SG depressurization is stopped to prevent N_2 injection into the RCS.

K/A Number: G2.1 .43: Conduct of operations: Ability to use procedures to determine the effects on reactivity of plant changes such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Level: RO Tier #: 3 Group #: IR – RO: 4.1 IR-SRO: 4.3

Proposed Question: 62

In accordance with 1-AP-31.00, Increasing or Decreasing RCS pressure, a reduction in RCS pressure can cause RCS Tave to ___(1)___ due to the reactivity inserted by the moderator ____(2)____ coefficient.

A. 1) rise
   2) void

B. 1) rise
   2) pressure

C. 1) lower
   2) void

D. 1) lower
   2) pressure

Proposed Answer: D.

Explanation: A note prior to Step 6 in AP-31.00 states that RCS Tave may lower due to negative reactivity inserted by the moderator pressure coefficient. This section of the procedure deals with response to a reduction of RCS pressure.

Technical Reference: 1-AP-31.00, Increasing or Decreasing RCS pressure, Rev. 20. Note prior to Step 6.

Reference Provided to Applicant: No

Learning Objective: ND-93.3-LP-5, Pzr Pressure Control, Objective C, Given a set of plant conditions, determine the appropriate operator response in accordance with 1(2)-AP-31.00, Increasing or Decreasing RCS Pressure to include the following: Immediate Actions (if applicable), Entry Conditions, Major Actions, Step Bases

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.10 / 43.6 / 45.6)

Comments:

K/A Match Analysis: Questions matches K/A, Candidate must recall note from AP-31.00 concerning effect of RCS pressure change on RCS Tave.

Distractor Analysis:

A. Incorrect. Part 1) is incorrect, RCS Tave expected to lower due to negative reactivity inserted by pressure reduction. Plausible fail to recall expected response during pressure reduction. Part 2) incorrect, procedure refers to effect of pressure coefficient vice void coefficient. Plausible that candidate confuses effect of changes to moderator density caused by pressure change with effect of void change.

B. Incorrect. Part 1) is incorrect, RCS Tave expected to lower due to negative reactivity inserted by pressure reduction. Plausible fail to recall expected response during pressure reduction. Part 2) is correct.

C. Incorrect. Part 1) is correct as stated in AP-31.00. Part 2) is incorrect, procedure refers to effect of pressure coefficient vice void coefficient. Plausible that candidate confuses effect of changes to moderator density caused by pressure change with effect of void change.

D. Correct. Both Parts 1) and 2 are correct.

NUMBER	PROCEDURE TITLE	REVISION
1-AP-31.00	INCREASING OR DECREASING RCS PRESSURE	20
		PAGE 3 of 7

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.___	CHECK ALL PRZR HEATERS - ENERGIZED	☐ Check or place all PRZR Heaters in ON if necessary to control RCS pressure.

* *

CAUTION: Lowering RCS pressure will cause the OT ΔT setpoint to lower.

* *

NOTE:
- Attachment 1 may be referred to for a diagram of the pressure control system.
- RCS pressure lowering will cause a slight reduction in RCS Tave due to negative reactivity from the moderator pressure coefficient.

NRC APPROVED. K/A Number: G2.1.15: Conduct of operations: Knowledge of administrative requirements for temporary management directives such as standing orders, night orders, Operations memos, etc.

Level: RO Tier #: 3 Group #: IR – RO: 2.7 IR-SRO: 3.4

Proposed Question: 63

Which of the following completes the following statement:

In the event a TRM requirement is determined to be inadequate, a ___(1)___ order may be issued to impose more restrictive requirements on an expedited basis. An order imposing such administrative controls must be presented to the ___(2)___ for review prior to implementation.

A. 1) Shift (Night)
2) Facility Safety Review Committee (FSRC)

B. 1) Standing
2) Corrective Action Review (CARB)

C. 1) Standing
2) Facility Safety Review Committee (FSRC)

D. 1) Shift (Night)
2) Corrective Action Review (CARB)

Proposed Answer: C

Explanation: Per OP-AA-100, Conduct of Operations, Attachment 7: 1) Standing orders are issued to impose more restrictive requirements. Shift orders are short term in nature and deal with day-to-day issues such as administrative directives, special plant conditions or special data collection requirements. 2) Standing orders shall be presented to the FSRC prior to implementation.

Technical Reference: OP-AA-100, Conduct of Operations, Rev 29.

Reference Provided to Applicant: No

Learning Objective: SROU-02, Administrative Procedures, Objective A: For a Tier 1 procedure, discuss the following: purpose of the procedure, responsibilities, knowledge items applicable to operators.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.10 / 45.12)

Comments:

K/A Analysis: K/A requires knowledge of administrative requirements for temporary management directives. This question tests that knowledge by requiring the candidate to understand when standing orders are used and who is required to review/approve the standing order.

Distractor Analysis:

A. 1) Shift (Night)
   2) Facility Safety Review Committee (FSRC)

B. 1) Standing
   2) Corrective Action Review (CARB)

C. 1) Standing
   2) Facility Safety Review Committee (FSRC)

D. 1) Shift (Night)
   2) Corrective Action Review (CARB)

A. Incorrect. 1) Incorrect Plausible because shift orders are also a temporary order but with a different purpose (day-to-day operational issues). 2) Correct.
B. Incorrect. 1) Correct. 2) Incorrect, because the FSRC is specifically directed to review. Plausible because the CARB is sometimes used in lieu of the FSRC for review of other management directives such as an Operational Decision Making.
C. Correct.
D. Incorrect. 1) Incorrect Plausible because shift orders are also a temporary order but with a different purpose (day-to-day operational issues). 2) Incorrect, because the FSRC is specifically directed to review. Plausible because the CARB is sometimes used in lieu of the FSRC for review of other management directives such as an Operational Decision Making.

Reference, OP-AA-100, Conduct of Operations, Attachment 7 Temporary orders (Part 1 answer and distractor, part 2 answer)

ATTACHMENT 7
(Page 1 of 2)
Temporary Orders

Operations Management ensures that temporary (shift and standing) orders and required reading materials are removed when they **NO** longer apply. Temporary orders are used to amplify or clarify operational information of a temporary nature; they are **NOT** used to change procedures or support operability, except as noted below. The duration of such temporary orders is minimized.

In the event that requirements in the Technical Specifications (TS) or the Technical Requirements Manual (TRM) are determined to be improper or inadequate, a Standing Order may be issued to impose more restrictive requirements on an expedited basis. A Standing Order imposing such administrative controls shall be presented to the Facility Safety Review Committee (FSRC) prior to implementation. A TS or TRM change shall be processed in a timely manner to clear a Standing Order issued to address improper or inadequate TS or TRM requirements. (Ref. 5.4.1)

Shift and Standing Orders are temporary instructions from Operations management.

Temporary orders are:

- Written in a clear and concise manner
- Approved by the Operations management
- Consistent with approved procedures
- Reviewed by shift Operators as part of the shift turnover process
- Available in hard-copy or electronically
- Reviewed by Security for Target Set impact

Shift Orders are short-term in nature and expire upon the issuance of succeeding Shift Orders. They deal with day-to-day operational issues such as special plant conditions, administrative directives, or special data collection requirements. When issued, applicability is generally weekends or nights.

Reference OP-AA-101, Operations Decision Making section 3.7 (part 2 distractor)

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3.7 **Periodic Reviews of ODMs**

3.7.1 **REVIEW** plant status with respect to the active ODM action plans at least monthly.

NOTE: Independent reviews may be performed after ODM approval and implementation. In order to ensure a thorough review by knowledgeable personnel, the independent review may be performed by the Facilities Safety Review Committee or the Corrective Action Review Board.

3.7.2 **ENSURE** new ODMs are reviewed independently.

3.7.3 **PERFORM** periodic reviews of all open ODMs at least quarterly for their aggregate impact to safety.

3.7.4 **APPROVE** modifications to an ODM implementation action plan.

ES 401-9 COMMENTS WITH CORRECTIONS SHOWN IN RED, 3/28/16

The paragraph beginning “Given the following” is window dressing. The question stands fine if it begins “which of the following”. It is not necessary to say “correctly completes”. The applicant already knows only one answer is correct.

Removed initial paragraph with “Given the following…” and changed “correctly completes” to “completes”.

K/A Number: G2.1.3: Conduct of operations: Knowledge of shift or short term relief turnover practices.

Level: RO Tier #: 3 Group #: IR – RO: 3.7 IR-SRO: 3.9

Proposed Question: 64

Given the following:

- It is currently 06:30 on September 12
- An individual is preparing to relieve the RO.
- The individual has been offsite since September 5.

Which ONE of the choices below completes the following statement?

In accordance with OP-AA-100, Conduct of Operations, to complete his turnover the on-coming RO is required to review applicable logs and temporary orders since at least _____?

A. September 5
B. September 6
C. September 9
D. September 11

Proposed Answer: C

Explanation: Per OP-AA-100, Rev. 29, a watch stander is required to review logs and temporary orders for the duration of absence or 3 days, whichever is less.

Technical Reference: OP-AA-100, Attachment 2, Expectations 8.1 Routine Turnover, 4th bullet. Page 43 of 79.

Reference Provided to Applicant: No

Learning Objective: SROU-02, Objective A, Tier 1 Procedure, OP-AA-100.

Question Source: Modified Bank (From NAPS, Question 66, 2014)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.10 / 45.13)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must recall time requirement for log review from OP-AA-100, Conduct of Operations.

Distractor Analysis:

A. Incorrect. This is the last day on-site, Plausible if Candidate recalls that logs must be reviewed since last time on-site.

B. Incorrect. Correlates to day following last day onsite. Plausible if Candidate recalls that logs must be reviewed since last time on-site.

C. Correct. IAW OP-AA-100, Conduct of Operations, shift relief requires the review of logs and temporary orders since last on site or 3 days, whichever is less.

D. Incorrect. This would be correct if individual was on-site yesterday. Plausible if Candidate confuses relief following absence with practices during the shift week.

66. 66 - G2.1.3 001/NEW/N/A/HIGH/3/3.7/3.9//

Given the following:

• It is currently 06:30 on July 10

• A Reactor Operator is the on-coming watchstander and is preparing to assume his duties

• He has been offsite since July 2

Which ONE of the choices below completes the following statement?

In accordance with OP-AA-100, Conduct of Operations, to complete his turnover the on-coming RO is required to review applicable logs and temporary orders since at least _____?

July 2

July 3

July 7

July 9

K/A Number: G2.2.20: Equipment Control: Knowledge of the process for managing troubleshooting activities.

Level: RO Tier #: 3 Group #: IR – RO: 2.6 IR-SRO: 3.8

Proposed Question: 65

Given the following:

- Unit 1 is operating at 100%, Unit 2 is at Hot Shutdown.
- Troubleshooting is needed for Unit 2 TD-AFW to determine why the Trip Throttle valve tripped.
- A troubleshooting team consisting of Maintenance and Operations personnel is being assembled. The team leader will be a Maintenance First Line Supervisor.
- The Terry Turbine is tagged out.
- This activity is a Simple, Cat IV (No Risk) troubleshooting, because the troubleshooting activities do not affect the safety of the plant.

Which of the following describes.

The highest level Supervisor/Manager that is required to approve this troubleshooting plan.

A. Troubleshooting Team Leader.

B. Maintenance Manager.

C. Shift Manager.

D. FSRC Chair.

Proposed Answer: C

Explanation: This troubleshooting activity is categorized as Cat IV because the equipment is removed from service. Troubleshooting activities do not affect the operation or safety of the plant. Per MA-AA-103, section 3.7. the Shift Manager is the highest level manager that has to approve this troubleshooting.

Technical Reference: MA-AA-103, Conduct of Troubleshooting, Rev. 13

Reference Provided to Applicant: No

Learning Objective: SROU-02, Admin Procedures, Objective B; “For a Tier-2 procedure, discuss the following: purpose of the procedure, requirements for operations personnel.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)

Comments:

K/A Analysis: K/A requires knowledge of the troubleshooting process, and this question test the candidate's knowledge of troubleshooting.

Distractor Analysis:

A. Incorrect. Plausible because Troubleshooting Team Leader approval is only required for Complex troubleshooting, and there is no risk to this troubleshooting
B. Incorrect. Plausible because Maintenance Manager approval is required for Low or Medium Risk levels.
C. Correct.
D. Incorrect. Plausible because the FSRC Chair is required for High Risk levels.

Reference: MA-AA-103, 3.7, Approval of Troubleshooting Plan.

3.8.8 **DOCUMENT** any required equipment restoration actions. (Attachment 2)

3.9 Approval of Troubleshooting Plan

TT 3.9.1 **ENSURE** PRA risk impact of the troubleshooting has been assessed and entered into the PRA for the applicable dates by Operations or O&P and **DOCUMENT** on Troubleshooting Sheet (Attachment 2).

TTL 3.9.2 **DETERMINE** from the troubleshooting approval matrix, the approval required for the troubleshooting plan and **DOCUMENT** approvals on Troubleshooting Sheet (Attachment 2).

Troubleshooting Approval Matrix

		Risk Level			
		I (High)	II (Medium)	III (Low)	IV (No)
Rigor Category	A	Plant Manager and FSRC	Maintenance Manager	Maintenance Manager	N/A
	B	Plant Manager and FSRC	Maintenance Manager	Maintenance Manager	N/A
	C	Plant Manager	Shift Manager	Shift Manager	Shift Manager
	D	Plant Manager	Shift Manager	Shift Manager	Shift Manager

Note: Matrix to be used to determine the highest level of review/approval required for the troubleshooting.

Reference MA-AA-103, 5.2.7. Shows when TTL may need to approve Troubleshooting plan.

5.2.7 **Troubleshooting Team Manager (TTM)**

a. Ensuring that the complex troubleshooting process is executed properly and completely using standardized problem solving techniques. This includes assisting the TTL with the identification/correction of the problem and the validation of assumptions and conclusions (see step 3.1.18). **(Ref. 5.4.15 and 5.4.16)**

b. Acting as the primary interface between the complex troubleshooting team and Station leadership.

c. Communicating progress of Troubleshooting status periodically to Plant Management

d. Ensuring required additional support and resource needs are made available.

e. Ensuring technical and procedural support is provided for the work activity.

f. Approving the Complex Troubleshooting Plan in conjunction with the Troubleshooting Team Lead (TTL) and confirms that it gets closed out properly.

g. When the OCC is staffed, acting as the primary point of contact for communication between the OCC and the complex troubleshooting team.

h. Coordinating a formal shift turnover meeting between shifts of complex troubleshooting team.

5.2.8 **Troubleshooting Team Lead (TTL)**

a. Communicating troubleshooting results and involving Engineering, Operations, or other Maintenance groups when assistance is required.

b. Coordinating the development of the troubleshooting plan and its field execution.

c. Ensuring technical and procedural support is provided for the work activity.

Reference: MA-AA-103, 5.2.7, Attachment 2, page 3 of 4.

Troubleshooting Sheet

MA-AA-103 – Attachment 2 **Page 3 of 4**

Identify the Impact of the Troubleshooting on Plant Equipment (Alarms, Lost Indication, Lost Function, system flow changes, affects on adjacent equipment/systems, potential to affect reactivity by isolation of feedwater heating/control rod movement/boron dilution change or other means, etc. (Refer to MA-AA-103 Attachment 1 for additional risk and rigor consideration.)		
Describe the expected results.		
Identify any decision or stop points to evaluate progress or subsequent actions.		
FSRC review required? ☐ Yes ☐ No	PRA Risk evaluated by Operations or O&P? ☐ Yes	
Troubleshooting Team Lead (TTL) Approval, if required (Print Name)	Troubleshooting Team Lead (TTL) Approval, if required (Signature)	Date
Maintenance Manager/Designee Review/Approval (Mark N/A if Rigor Category C or D) (Print Name)	Maintenance Manager/Designee Review/Approval (Mark N/A if Rigor Category C or D) (Signature)	Date
Troubleshooting Team Manager (TTM) Approval, if required (Print Name)	Troubleshooting Team Manager (TTM) Approval, if required (Signature)	Date
Plant Manager (Nuclear) Approval, if required (Print Name)	Plant Manager (Nuclear) Approval, if required (Signature)	Date
FSRC Chair Approval, if required (Print Name)	FSRC Chair Approval, if required (Signature)	Date
Shift Manager Approval (Print Name)	Shift Manager Approval (Signature)	Date

Form No. 730600 (Apr 2014)

Reference: SROU-02, Admin Procedures. Shows MA-AA-103 as a Tier 2 procedure (SRO Only).

Tier 2:

- This is a procedure that contains some items ***that are applicable to operators***, but the procedure, as a whole, does not require memorization. Trainees should be familiar with:
    - Purpose of the procedure
    - Requirements for operations personnel.
- Studying a tier 2 procedure would NOT consist of bulk memorization, other than the items that are specific to operations personnel. This also would include who is required to authorize items applicable to operations.
- If a tier 2 procedure describes a process then the trainee should be able to describe the purpose of that process.

Tier 2 Procedures:

- CM-AA-TDC-204, Temporary Modifications
- LI-AA-700, Fatigue Management and Work Hour Limits for Covered Workers
- MA-AA-103, Conduct of Troubleshooting (SRO)
- MA-AA-105, Scaffolding (SRO)Ob
- OP-AA-101, Operational Decision Making
- OP-AA-102, Operability Determination (SRO- emphasis on immediate determination)
- OP-AA-201, Lockout
- OP-AA-1300, Quarantine
- OP-AA-1500, Operational Configuration Control
- OP-AA-900, Authentication
- OP-SU-601, Protected Equipment
- OU-SU-201 Shutdown Safety Assessment Checklist (SRO)
- PI-AA-200, Corrective Action
- PI-AA-5000, Human Performance
- SA-AA-125, Electrical Safety
- AD-AA-100, Technical Procedure Process Control
- VPAP-1101, Test Control
- VPAP-2003, Post Maintenance Testing Program
- VPAP-2101, Radiation Protection Program (Dose limits/Area Postings/RWP process and types of RWPs)
- VPAP-2103S, Offsite Dose Calculation Manual (Surry) (SRO)
- VPAP-2105, Temporary Shielding Program (SRO)
- VPAP-2802, Notifications and Reports (SRO) – (Ensure memorization of 1 hour or less notifications)
- WM-AA-100, Work Management

K/A Number: G2.2.22: Equipment Control: Knowledge of limiting conditions for operations and safety limits.

Level: RO Tier #: 3 Group #: IR – RO: 4.0 IR-SRO: 4.7

Proposed Question: 66

With a Unit operating at 100% power, which ONE of the following identifies:

1) The Safety Limit for RCS Pressure.
2) The time requirement to reach Hot Shutdown if the Safety Limit is exceeded.

A. 1) 2735 psig.
   2) 15 minutes.

B. 1) 2735 psig.
   2) 1 hour.

C. 1) 2485 psig.
   2) 15 minutes.

D. 1) 2485 psig.
   2) 1 hour.

Proposed Answer: B.

Explanation: IAW TS section 2.2, Safety Limit, Reactor Coolant System Pressure, 2735 psig is the RCS Safety Limit with Fuel in the vessel; if exceeded the Unit must be in HSD in 1 hour.

Technical Reference: TS 2.2.A and B.

Reference Provided to Applicant: No

Learning Objective: ND-88.1-LP-9, Tech Specs, Objective G, Apply the purpose and specification for the Safety Limits IAW section 2 of Tech Specs including for SRO candidates, the basis behind these specifications.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.5 / 43.2 / 45.2)

Comments:

K/A Match: Question matches K/A. Candidate must select the choice corresponding to the RCS Pressure safety limit and time requirement to reach HSD if limit is exceeded.

Distractor Analysis:

A. Incorrect. Part 1) is correct. Part 2) is incorrect, requirement is 1 hour. Plausible if Candidate applies TS time for Delta Flux out-of-band.

B. Correct. Both Parts 1) and 2) are correct.

C. Incorrect. Part 1) is incorrect, 2485 correlates to the primary lift setpoint. Plausible if Candidate confuses Safety lift setpoint with safety limit. Part 2 is incorrect, requirement is 1 hour. Plausible if Candidate applies TS time for Delta Flux out-of-band.

D. Incorrect. Part 1) is incorrect, 2485 correlates to the primary lift setpoint. Plausible if Candidate confuses Safety lift setpoint with safety limit. Part 2) is correct.

TS 2.2-1
09-15-05

2.2 SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the maximum limit on Reactor Coolant System pressure.

Objective

To maintain the integrity of the Reactor Coolant System.

Specification

A. The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

B. In the event the Safety Limit is violated, the facility shall be placed in at least HOT SHUTDOWN within 1 hour.

Excerpt from ND-88,1-LP-3, PZR and Press Relief:

5. Safety Valves

a. Safety Valves Design Bases - The combined capacity of the pressurizer safety valves is designed to achieve the following:

(1) Sufficient capacity (293,330 lb/hr each) to limit the RCS pressure to within the 2735 psig safety limit, following a complete loss of load while at rated power. For this, it is assumed no reactor trip occurs until the high pressurizer pressure trip is reached. Also assumed is no operation of the PORVs, steam dump valves, reactor control, pzr level control, or pzr spray (secondary steam reliefs do operate);

d. The safety valves are spring loaded: self actuated relief valves with back pressure compensation. The valves are set to relieve at 2485 psig with a capacity of 293,330 lbm/hr (each). The safety valves discharge to the pressurizer relief tank which is normally pressurized to 3 psig.

K/A Number: G2.2.7: Equipment Control: Knowledge of the process for conducting special or infrequent tests.

Level: RO Tier #: 3 Group #: IR – RO: 2.9 IR-SRO: 3.6

Proposed Question: 67

Given the following:

- Unit 1 Reactor Startup in progress.
- 1-NPT-RX-008, Startup Physics Testing (ICCE II) is scheduled later in the shift.
- Personnel have assembled to conduct a Pre-job briefing for 1-NPT-RX-008, Startup Physics Testing (ICCE II).

Which one of the following correctly describes the requirements per OP-AA-106, Infrequently Conducted or Complex Evolutions for the following?

1) A Senior Operations Manager is assigned for the test. This individual shall _____.
2) A review of test procedure instructions and acceptance criteria ____ required to be briefed to the team per Attachment 2 – Detailed Pre-Test Briefing Checklist.

A. 1) coordinate the test
2) is not

B. 1) be familiar with the test
2) is

C. 1) coordinate the test
2) is

D. 1) be familiar with the test
2) is not

Proposed Answer: B

Explanation: Per OP-AA-106, Infrequently Conducted or Complex Evolutions, before the start of each ICCE, **ENSURE** that a Senior Operations Manager is assigned to provide oversight of the test or evolution. 1)This individual shall be familiar with the test or evolution to the extent of knowing the general sequence, objectives, reactor safety considerations, portions of the test or evolution most susceptible to difficulty, and criteria for terminating the test or evolution. The Manager conducts a briefing of the team per Attachment 3 – Management Expectations Briefing Checklist. This briefing is a high level briefing that focuses on the need to exercise caution, and conservatism during the test. 2) The ICCE briefing does include specific test procedure instructions and acceptance criteria. This item is covered by the Test Coordinator using Attachment 2, Detailed Pre-Test Briefing Checklist.

Technical Reference: OP-AA-106, Infrequently Conducted or Complex Evolutions, Revision 9.

Reference Provided to Applicant: No

Learning Objective: SROUTP-SDS-02, Administrative Procedures, objective A; For a Tier 1 procedure, discuss the following:

- o Purpose of the procedure
- o Responsibilities, as it applies to operations personnel (Operations Personnel, Shift Operators, Reactor Operators, etc

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.10 / 43.3 / 45.13)

Comments:

K/A Match Analysis: Question requires knowledge of ICCE process requirements including; duties of the Senior Operations Manager, required management briefing items, therefore the question matches the K/A.

Distractor Analysis:

A. Incorrect. 1) Incorrect. This individual shall be familiar with the test or evolution to the extent of knowing the **general sequence, objectives**, reactor safety considerations, portions of the test or evolution most susceptible to difficulty, and criteria for terminating the test or evolution. Plausible if candidate confuses duties of Senior Operations Manager, because it is the Test Coordinator who directly coordinates the test. 2) Incorrect. This item is required to be briefed using Attachment 2 and the brief is conducted by the Test Coordinator. Plausible if the team confuses the briefing done by the Senior Operations Manager with the briefing done by the Test Coordinator.
B. Correct.
C. Incorrect. 1) Incorrect. This individual shall be familiar with the test or evolution to the extent of knowing the **general sequence, objectives**, reactor safety considerations, portions of the test or evolution most susceptible to difficulty, and criteria for terminating the test or evolution. Plausible if candidate confuses duties of Senior Operations Manager, because it is the Test Coordinator who directly coordinates the test. 2) Correct.
D. Incorrect. 1) Correct. 2) Incorrect. This item is required to be briefed using Attachment 2 and the brief is conducted by the Test Coordinator. Plausible if the team confuses the briefing done by the Senior Operations Manager with the briefing done by the Test Coordinator.

References: OP-AA-106, 3.2.2.b, Senior Operations Manager duties (Part 1 correct answer)

Manager Nuclear Operations

b. Before the start of each ICCE, **ENSURE** that a Senior Operations Manager is assigned to provide oversight of the test or evolution. This individual shall:

Senior Operations Manager/ OMOC

1. **ENSURE** that tests or evolutions are conducted in a manner that maximizes the margin of safety of the Unit.
2. **OVERSEE** the test or evolution to ensure the Station is operated safely without becoming involved in the details.
3. **BE** familiar with the test or evolution to the extent of knowing the general sequence, objectives, reactor safety considerations, portions of the test or evolution most susceptible to difficulty, and criteria for terminating the test or evolution.
4. **ATTEND** at least one pre-job briefing.
5. **EXERCISE** authority through the Shift Manager without relieving the Shift Manager's responsibility for safe Station operation.
6. **REMAIN** at the Station during critical portions of the test or evolution, in a location where oversight and responsibility can be exercised effectively.

References: OP-AA-106 , 3.2.4, Test Coordinator duties (part 1 distractor).

3.2.4 **Test Coordinator**

Senior Station Management

a. **ASSIGN** a Test Coordinator to support testing that is Category I or II ICCE. A Test Coordinator is optional for Category III ICCE.

Test Coordinator

b. **UNDERSTAND** and **COORDINATE** the various aspects of the test.

References: OP-AA-106, Attachment 2, Detailed Pre-Test Briefing checklist (Part 2 correct answer).

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Dominion

Detailed Pre-Test Briefing Checklist

OP-AA-106 Attachment 2 **Page 2 of 2**

Checklist

The following items were covered during the pre-test briefing (mark all that apply):

[ ] Test objectives.
[ ] Station and test organization.
[ ] Establish the specific position or person primarily responsible for the test.
[ ] Duties and responsibilities of individual personnel on the test team. Ensure that each participant understands their individual and team responsibilities.
[ ] Establish the lines of communication steering the test.
[ ] Personnel special certifications and qualifications required.
[ ] Discuss the initial conditions of the test, including Station and system status.
[ ] Anticipated Unit performance.
[ ] Termination guidance for specific portions of the test and contingency plans for unexpected occurrences.
[ ] Risks associated with the test.
[ ] Discuss interactions that may cause a Unit transient.
[ ] Discuss whether test will cause a pressurizer insurge and/or outsurge transient.
[ ] Discuss any significant industrial safety, radiological hazards, or environmental concerns.
[ ] Review of test procedure precautions and limitations.
[ ] Review of test procedure instructions and acceptance criteria.
[ ] Ensure that personnel have read and understand the necessary work procedures and are aware of specific hold points.
[ ] Administratively approved process for deviating from the test procedure, if it becomes necessary.
[ ] Discuss the scheduled activities, schedule restraints, and the impact of other work in the area.
[ ] Discuss the specific job or departmental administrative controls required, including RWPs, Tagouts, Flame Permits, Fire Watches.
[ ] Discuss the required materials for the activity and ensure their availability, including parts and tools.
[ ] Discuss barriers, "Operating Experience," and "Good Practices" noted from previous similar tests.
[ ] Discuss the work area conditions, including any additional services required (e.g., service air, ventilation, lighting).
[ ] Discuss the expected level of housekeeping, including cleanliness requirements, foreign material exclusion, and post-test cleanup.
[ ] Discuss the need for a post-test review and what will be required of each participant to close out the test.
[ ] Discuss the final condition of Station.

Brief Performed by (Signature)	Date

Key: RWPs-Radiation Work Permits

References: OP-AA-106, Attachment 3, Management expectations briefing (Part 2 distractor).

Dominion

Management Expectations Pre-Job Briefing Checklist for an ICCE

OP-AA-106 - Attachment 3 **Page 1 of 1**

Checklist

The following items were covered during the Management Expectations briefing (check all that apply):

- ☐ The need to exercise caution and conservatism during the ICCE, particularly when uncertainties are encountered.
- ☐ Emphasis on maintaining the highest margins of safety to place proper perspective on any sense of urgency that may otherwise prevail.
- ☐ Assigned responsibilities for the activity and any deviation from normal shift duties and accountabilities.
- ☐ The need for open communication.
- ☐ Lessons learned from pertinent in-house and industry operating experience to assist Operations Department and support personnel in internalizing the lessons.
- ☐ The need to stop the ICCE when unexpected conditions arise or unexpected behavior is experienced.
- ☐ Strict compliance with procedure details.
- ☐ Criteria for terminating the ICCE.

Brief Performed and Permission to Conduct the ICCE Obtained (Signature of Assigned Senior Operations Manager))	Date

Key: ICCE-Infrequently Conducted or Complex Evolution

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

Level: RO Tier #: 3 Group #: IR – RO: 3.4 IR-SRO: 3.8

Proposed Question: 68

In accordance with VPAP-0106, Subatmospheric Containment Entry, which ONE of the following identifies:

1) The minimum number of personnel designated as the Containment Emergency Team.
2) The minimum number of Containment Emergency Team members who must be First Aid Qualified.

A. 1) 3.
 2) All.

B. 1) 3.
 2) 1.

C. 1) 2.
 2) 1.

D. 1) 2.
 2) All.

Proposed Answer: C.

Explanation: IAW VPAP-0106, Rev. 13, Step 6.2.5, the Containment Emergency Team shall consist of at least 2 members, with at least one member first aid qualified.

Technical Reference: VPAP-0106, Subatmospheric Containment Entry, Rev. 13.

Reference Provided to Applicant: No

Learning Objective: SROUTP-SDS-02, Admin Procs, Objective A, Tier 1 Procedures.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.12 / 43.4 / 45.9 / 45.10)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must select the appropriate response based on knowledge of containment entry requirements as found in VPAP-0106.

Distractor Analysis:

A. Incorrect. Part 1) is incorrect, minimum number is 2. Plausible if Candidate assumes Emergency Team membership must meet the requirements of Entry Team minimum (2) plus a member that is first aid qualified. Part 2) is incorrect, minimum is 1 person. Plausible if Candidate assumes all members of Team must be first aid qualified.

B. Incorrect. Part 1) is incorrect, minimum number is 2. Plausible if Candidate assumes Emergency Team membership must meet the requirements of Entry Team minimum (2) plus a member that is first aid qualified. Part 2) is correct.

C. Both Parts 1) and 2) are correct.

D. Incorrect. Part 1) is correct. Part 2) is incorrect, minimum is 1 person. Plausible if Candidate assumes all members of Team must be first aid qualified.

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6.2.5 Containment Emergency Teams shall:

- Consist of a minimum of two members
- Have at least one member first aid qualified
- Satisfy the requirements of Steps 6.2.1, 6.2.2, and 6.2.4

K/A Number: G2.4.11: Emergency Procedures/Plans: Knowledge of abnormal condition procedures.

Level: RO Tier #: 3 Group #: IR – RO: 4.0 IR-SRO: 4.2

Proposed Question: 69

Which one of the following correctly state the requirements for communicating Notes and Cautions when using Abnormal Operating Procedures (AOP).

1) A Note or Caution that is not applicable ______.
2) A Note or Caution that is applicable and is repeated ______.

A. 1) need not be verbalized
2) may be paraphrased

B. 1) may be paraphrased
2) must be read verbatim

C. 1) may be paraphrased
2) may be paraphrased

D. 1) need not be verbalized
2) must be read verbatim

Proposed Answer: A

Explanation: OP-AP-104, Emergency and Abnormal Operating Procedures, step 3.7.2 states the following:

- A Note or Caution is read verbatim when encountered for the first time during an event.
- If not applicable, a Note or Caution need not be verbalized.
- If encountered repeatedly, a Note or Caution may be paraphrased.

Technical Reference: OP-AP-104, Emergency and Abnormal Operating Procedures. Rev. 3

Reference Provided to Applicant: No

Learning Objective: ND-93.5-LP-2, EP Writer, Objective F; Given actual or simulated EOP implementation, apply the management standards and other good practices applicable to EOP usage.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)

Comments:

K/A Analysis: Question specifically test candidate's knowledge of parallel procedure usage regarding AOP and EOP. This directly meets the K/A.

Distractor Analysis:

A. Correct.
B. Incorrect. 1) Incorrect as OP-AP-104 specifically states, "If not applicable, a Note or Caution need not be verbalized." Plausible if candidate confuses this requirement with the other requirements regarding Notes and Cautions, such as the requirement for frequently encountered Notes and Cautions. 2) Incorrect as OP-AP-104 states: "If encountered repeatedly, a Note or Caution may be paraphrased." Plausible if candidate confuses this requirement with the other requirements regarding Notes and Cautions, such as the requirement to read verbatim a Note or Caution that is encountered for the first time.
C. Incorrect. 1) Incorrect as OP-AP-104 specifically states, "If not applicable, a Note or Caution need not be verbalized." Plausible if candidate confuses this requirement with the other requirements regarding Notes and Cautions, such as the requirement for frequently encountered Notes and Cautions. 2) Correct.
D. Incorrect. 1) Correct. 2) Incorrect as OP-AP-104 states: "If encountered repeatedly, a Note or Caution may be paraphrased." Plausible if candidate confuses this requirement with the other requirements regarding Notes and Cautions, such as the requirement to read verbatim a Note or Caution that is encountered for the first time.

Reference OP-AP-104, Section 3.7.2 (Correct Answer)

3.7 Abnormal Operating Procedures

> NOTE: Except at Millstone, immediate actions are identified clearly as such in Abnormal Operating Procedures.

Procedure User 3.7.1 **PERFORM** immediate actions from memory.

3.7.2 **APPLY** the following communication techniques when using Abnormal Operating Procedures:

- A NOTE or CAUTION is read verbatim when encountered for the first time during an event
- If not applicable, a NOTE or CAUTION need not be verbalized
- If encountered repeatedly, a NOTE or CAUTION may be paraphrased
- A NOTE or CAUTION may be directed to applicable crew members; in such cases, the applicable crew member will paraphrase or acknowledge the NOTE or CAUTION

> NOTE: The priority of procedures depends upon the events in progress. Some abnormal operating procedures must be implemented while emergency operating procedures are in effect. In cases of parallel procedure usage, the EOP receives priority and immediate actions are completed before parallel procedure usage. When using an AOP in parallel with the EOP, only those steps in the AOP that ensure success of the EOP are required to be performed.

K/A Number: G2.4.13: Emergency Procedures/Plans: Knowledge of crew roles and responsibilities during EOP usage.

Level: RO Tier #: 3 Group #: IR – RO: 4.0 IR-SRO: 4.6

Proposed Question: 70

In accordance with OP-AP-104, Emergency and Abnormal Operating Procedures:

1) The ______ is responsible for directing the actions of the Reactor Operators during transient conditions.

2) When a RED or ORANGE path is encountered, CSF Status Trees shall be monitored at a maximum time interval of ______.

A. 1) Shift Manager
   2) every 15 minutes

B. 1) Shift Manager
   2) continuously (3-5 minutes)

C. 1) Unit SRO
   2) every 15 minutes

D. 1) Unit SRO
   2) continuously (3-5 minutes)

Proposed Answer: D.

Explanation: IAW OP-AP-104, the Unit SRO is responsible for directing the EOP/AOP actions of the ROs. The SM maintains an overview of the crew response. Also, if a RED or ORANGE terminus is encountered, the CSFSTs shall be monitored continuously (every 3-5 minutes).

Technical Reference: OP-AP-104, Emergency and Abnormal Operating Procedures, Rev. 2, Section 3.6, Critical Safety Function Status Trees.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-26, CSFST, Objective D, Given a specific plant condition, apply the Critical Safety Function rules of prioritization to determine the applicable procedure.

Question Source: Modified Bank (VC Summer, 2007, Question #73, Attached)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 45.12)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must utilize knowledge of CSFST usage and requirements of OP-AP-104 administrative procedure to determine the correct choice for the event in progress.

Distractor Analysis:

A. Incorrect. Part 1) is incorrect. Plausible if Candidate believes that the SM may direct crew response. Part 2) is incorrect, CSFSTs shall be monitored continuously. Plausible since 15-minute monitoring is normally expected during EOP implementation.

B. Incorrect. Part 1) is incorrect. Plausible if Candidate believes that the SM may direct crew response. Part 2) is correct.

C. Incorrect. Part 1) is correct. Part 2) is incorrect, CSFSTs shall be monitored continuously. Plausible since 15-minute monitoring is normally expected during EOP implementation.

D. Correct. Both Parts 1) and 2) are correct.

Reference: OP-AP-104

3.6 Except at Millstone Unit 2, Critical Safety Function (CSF) Status Trees

Monitoring of CSF Status Trees is a crew responsibility. In the absence of a STA, a licensed Operator is assigned to monitor the CSF Status Trees.

Millstone Unit 2

Monitor Resource Assessment Trees (RATs) in accordance with:

- EOP 2540-002 Resource Assessment Trees
- EOP 2540 Safety Function Tracking Page

Procedure User

3.6.1 **MONITOR** Critical Safety Status carefully.

a. Monitoring and implementation of the CSF Status Trees shall begin when directed by the initial emergency response procedure, or when a transition is made to another emergency procedure unless otherwise directed.

b. Upon power restoration following a loss of all AC event, once any equipment has been placed in a disabled condition recovery efforts should be in accordance with the Loss of All AC guidelines. This is necessary because actions taken after equipment has been placed in disabled must be carefully restored to prevent damage to equipment and to align the emergency equipment as required for further EOP implementation. FRs will not apply until procedurally specified.

INFORMATION USE

DOMINION

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c. The CSF Status Trees have different rules of usage than emergency operating procedures and are monitored, at least every 15 minutes, in parallel with the performance of the emergency operating procedures.

d. CSF Status Tree monitoring shall be continuous (approximately every 3 to 5 minutes) when a RED or ORANGE terminus is encountered.

Given the following plant conditions:

- A reactor trip and safety injection have occurred.
- Off-Site power is lost subsequent to safety injection actuation.
- Equipment failures during performance of EOP-2.0, *Loss of Reactor or Secondary Coolant*, resulted in the following conditions:
    - Bus 1DA is de-energized due to a fault.
    - CSF Status Trees indicate as follows:
        - Subcriticality GREEN
        - Core Cooling ORANGE
        - Heat Sink RED
        - Integrity GREEN
        - Containment YELLOW
        - Inventory YELLOW

Which ONE (1) of the following describes the requirement for Critical Safety Function Status Tree Monitoring in accordance with OAP-103.04, *EOP/AOP User's Guide*?

A. Continuous monitoring is required.

B. Monitor every 5 - 10 minutes unless a change in status occurs.

C. Monitor every 10 - 20 minutes unless a change in status occurs.

D. Monitored for information only.

A. identified as correct.

K/A Number: G2.4.27: Emergency Procedures/Plans: Knowledge of “fire in the plant” procedures.

Level: RO Tier #: 3 Group #: IR – RO: 3.4 IR-SRO: 3.9

Proposed Question: 71

Given the following:

- Both units are operating at 100%
- A large fire occurs in the Main Control Room (MCR) and is NOT brought under control.
- The Shift Manager orders the crew to don SCBAs due to the large amount of toxic fumes from the fire.

Which of the following describes:

1) The initial procedure that shall be performed by the crew.
2) If Control Room evacuation is necessary, due to the fire, which procedure will direct the actions necessary to evacuate the MCR.

A. 1) 0-FCA-1.00, Limiting MCR Fire.
2) 0-AP-20.00, Main Control Room Inaccessibility.

B. 1) 0-AP-48.00, Fire Protection – Operations Response.
2) 0-FCA-1.00, Limiting MCR Fire.

C. 1) 0-AP-48.00, Fire Protection – Operations Response.
2) 0-AP-20.00, Main Control Room Inaccessibility.

D. 1) 0-FCA-1.00, Limiting MCR Fire.
2) 0-FCA-1.00, Limiting MCR Fire.

Proposed Answer: B

Explanation: 0-AP-48.00 is the direct entry procedure for ANY fire on site. If control room evacuation is in progress then 0-AP-48.00 will direct transfer to 0-FCA-1.00 which in turn will direct the actions necessary to evacuate the MCR.

Technical Reference: 0-AP-48.00, Fire Protection – Operations Response; Rev. 34. 0-FCA-1.00, Limiting MCR Fire; Rev. 49. 0-AP-20.00, Main Control Room Inaccessibility.

Reference Provided to Applicant: No

Learning Objective: ND-95.6-LP-3, Fire Areas FCAs, Objective A, and B. Objective A: Examine the operations response to a station fire, implementation of AP 48.00. Objective B: Examine the operations involved in achieving plant control at the auxiliary shutdown panel to stabilize the unit at HSD and CSD conditions following implementation of FCA-1.00, Limiting Main Control Room Fire.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)

Comments:

K/A Analysis: Question requires knowledge of fire procedures 0-AP-48.00, Fire Protection – Operations Response, and 0-FCA-1.00, Limiting MCR fire. Specifically this question tests the operators' knowledge of which procedure to use when. 0-AP-20.00, Main Control Room Inaccessibility also included to enhance plausibility.

Distractor Analysis:

A. 1) 0-FCA-1.00, Limiting MCR Fire.
   2) 0-AP-20.00, Main Control Room Inaccessibility.

B. 1) 0-AP-48.00, Fire Protection – Operations Response.
   2) 0-FCA-1.00, Limiting MCR Fire.

C. 1) 0-AP-48.00, Fire Protection – Operations Response.
   2) 0-AP-20.00, Main Control Room Inaccessibility.

D. 1) 0-FCA-1.00, Limiting MCR Fire.
   2) 0-FCA-1.00, Limiting MCR Fire.

A. Incorrect. 1) Incorrect. 0-AP-48.00 is the entry procedure for all fires. Plausible because 0-FCA-1.00 does provide actions necessary for Fire in the MCR. 2) Incorrect. 0-FCA-1.00 provides directions for MCR evacuation. Plausible because 0-AP-20.00 provides similar actions for toxic fumes NOT caused by Fire.
B. Correct.
C. Incorrect. 1) Correct. 2) Incorrect. 0-FCA-1.00 provides directions for MCR evacuation. Plausible because 0-AP-20.00 provides similar actions for toxic fumes NOT caused by Fire.
D. Incorrect. 1) Incorrect. 0-AP-48.00 is the entry procedure for all fires. Plausible because 0-FCA-1.00 does provide actions necessary for Fire in the MCR. 2) Correct.

Reference: ND-95.6-LP-3, Fire Areas FCAs. (0-AP-48.00, 0-FCA-1.00)

0-AP-48.00 (partial)

A. AP 48.00, Fire Protection - Operations Response

Review AP-48.00 with trainees, highlighting the following:

1. Steps 1 and 2 sounds the fire alarm and an announcement is made as to the fire location and the Fire Team members' beepers are activated.

2. Steps 3 through 7 determine if the MCR must be evacuated due to the fire. If conditions worsen to the point of MCR abandonment, then FCA 1.00 is initiated.

0-FCA-1.00 (partial)

B. FCA 1.00, Limiting MCR Fire

1. The purpose of FCA 1.00, Limiting Main Control Room Fire, is to shift plant control to the auxiliary shutdown panels and direct local operations.

2. When implemented, the procedure will first stabilize the unit at hot shutdown. The steps in this group will address the immediate plant concerns.

Reference, 0-AP-48.00 (Entry page, steps 4-7). P1 Correct Answer.

Dominion

SURRY POWER STATION

ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
0-AP-48.00	FIRE PROTECTION - OPERATIONS RESPONSE (WITH 5 ATTACHMENTS)	34
		PAGE 1 of 9

PURPOSE

To provide guidance in the event of a fire.

ENTRY CONDITIONS

1. Notification of a fire in progress.

NUMBER	PROCEDURE TITLE	REVISION
0-AP-48.00	FIRE PROTECTION - OPERATIONS RESPONSE	34
		PAGE 4 of 9

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- If SCBAs are needed, changeout of the bottles must be coordinated so that the bottles do not all run out at the same time. Spare air bottles are stored in the Auxiliary Boiler Room and behind 1-VS-AC-5. Loss Prevention should be called in to recharge air bottles as necessary.
- If MCR evacuation is necessary, consideration should be given to taking SCBAs to the Auxiliary Shutdown Panel based on the potential for degraded habitability conditions in the ESGR. Personnel who are required to wear respirator glasses shall take them to the ESGR.

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.___	TRY TO PREVENT MCR EVACUATION:	
	a) Terminate the hazard	
	b) Secure Control Room AHUs at local switch, as directed by Shift Supervision: ☐ • 1-VS-AC-1 ☐ • 1-VS-AC-2 ☐ • 2-VS-AC-8 ☐ • 2-VS-AC-9	b) Locally open breaker(s): ☐ • 1-VS-AC-1, 1H1-1-4A1 ☐ • 1-VS-AC-2, 1J1-1-3A2 ☐ • 2-VS-AC-8, 2H1-1-4B2 ☐ • 2-VS-AC-9, 2J1-1-3A2
	☐ c) Wear breathing apparatus as directed by Shift Supervision	
	d) Secure MCR ventilation, as directed by Shift Supervision: ☐ 1) Stop 1-VS-F-15 ☐ 2) Close 1-VS-MOD-103C and 1-VS-MOD-103D	
*5.___	DETERMINE IF MCR EVACUATION - NECESSARY DUE TO UN-INHABITABILITY OR LOSS OF SAFE SHUTDOWN FUNCTION (REFER TO ATTACHMENT 3)	☐ GO TO Step 9.
6.___	CHECK AUXILIARY SHUTDOWN PANEL - AVAILABLE	Do the following: ☐ a) Initiate LFFG1, Operations Response. ☐ b) GO TO Step 9.

NUMBER	PROCEDURE TITLE	REVISION
0-AP-48.00	FIRE PROTECTION - OPERATIONS RESPONSE	34
		PAGE 5 of 9

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.___	INITIATE 0-FCA-1.00, LIMITING MCR FIRE	

Reference 0-FCA-1.00 (Entry page, steps 12, 13). P1 distractor, P2 Correct Answer.

Dominion

SURRY POWER STATION

FIRE CONTINGENCY ACTION

NUMBER	PROCEDURE TITLE	REVISION
0-FCA-1.00	LIMITING MCR FIRE (WITH 20 ATTACHMENTS)	49
		PAGE 1 of 33

PURPOSE

To provide guidance to Operations personnel for response to the consequences of a limiting fire in the Main Control Room.

TIME CRITICAL ACTIONS

ENTRY CONDITIONS

Transition from 0-AP-48.00, Fire Protection - Operations Response

NUMBER	PROCEDURE TITLE	REVISION 49
0-FCA-1.00	LIMITING MCR FIRE	PAGE 6 of 33

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

12.___ GET THE FOLLOWING ITEMS:

- ☐ • Appendix R Key Box
- ☐ • FCA Procedures
- ☐ • CRO Shift Relief Checklists
- ☐ • FR Procedures

13.___ EVACUATE MCR AND GO TO ESGRs

Reference 0-AP-20.00 (Caution and Note on page 2). Used as P2 distractor.

NUMBER	PROCEDURE TITLE	REVISION
0-AP-20.00	MAIN CONTROL ROOM INACCESSIBILITY	19
		PAGE 2 of 13

STEP	ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Operators should observe posted signs limiting the use of portable radios.

* *

NOTE:
- MCR evacuation due to a fire is addressed by 0-FCA-1.00, LIMITING MCR FIRE.
- Alarms on 1-VS-02A-1 or 2-VS-02A-1 are addressed by 0-AP-20.01, MAIN CONTROL ROOM OXYGEN MONITOR - ALARM OR MALFUNCTION.

STEP	ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	DETERMINE IF FIRE IN MCR	☐ GO TO Step 3.
2.___	GO TO 0-AP-48.00, FIRE PROTECTION - OPERATIONS RESPONSE	

K/A Number: WE03EK1.1: LOCA Cooldown - Depress. / 4: Knowledge of the operational implications of the following concepts as they apply to the (LOCA Cooldown and Depressurization): Components, capacity, and function of emergency systems.

Level: RO Tier #: 1 Group #: 2 IR – RO: 3.4 IR-SRO: 4.0

Proposed Question: 72

Unit 2 initially operating at 100% power.

- The Team tripped the reactor and manually actuated safety injection due to a SBLOCA inside containment.
- Off-site power was lost shortly following the reactor trip.
- The Team is currently performing 2-ES-1.2, Post LOCA Cooldown and Depressurization.

Which ONE of the following identifies the components used to:

1) Establish a cooldown of the RCS in 2-ES-1.2.
2) Depressurize the RCS to refill the pressurizer.

A. 1) SG PORVs.
   2) Pressurizer Spray.

B. 1) SG PORVs.
   2) Pressurizer PORV.

C. 1) Steam dumps.
   2) Pressurizer Spray.

D. 1) Steam dumps.
   2) Pressurizer PORV.

Proposed Answer: B.

Explanation: On a loss of of-site power, CW pumps will be lost requiring isolation of CW flow through the condenser and loss of the use of the steam dumps. Loss of the RCPs will prevent use of pressurizer spray.

Technical Reference: 2-ES-1.2, Rev. 47, Steps 11 and 14.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-9, ES-1.2, Objective B, Given a copy of ES-1.2, Post-LOCA Cooldown and Depressurization, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.8 / 41.10 / 45.3)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must assess Plant conditions to determine the method that will be used to Cooldown and depressurize the RCS.

Distractor Analysis:

A. Incorrect. Part 1) is correct, SG PORVs must be used since steam dump cannot be used due to isolation of CW flow to condenser on loss of off-site power. Part 2) is in correct, RCPs are unavailable for spray and Aux spray could not be used since SI would still be in service at this time. Plausible should the Candidate fail to consider RCPs lost or that Aux Spray would also be unavailable at this time.

B. Correct. Both Parts 1) and 2) are correct.

C. Incorrect. Part 1) is incorrect, steam dumps would be non-functional. Plausible if Candidate fails to account for isolation of CW flow through condenser due to loss of CW pumps. Part 2) is in correct, RCPs are unavailable for spray and Aux spray could not be used since SI would still be in service at this time. Plausible should the Candidate fail to consider RCPs lost or that Aux Spray would also be unavailable at this time.

D. Incorrect. Part 1) is incorrect, steam dumps would be non-functional. Plausible if Candidate fails to account for isolation of CW flow through condenser due to loss of CW pumps. Part 2) is correct.

NUMBER	PROCEDURE TITLE	REVISION
2-ES-1.2	POST LOCA COOLDOWN AND DEPRESSURIZATION	47
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11. ___ INITIATE RCS COOLDOWN TO CSD:

- ☐ a) Maintain cooldown rate in RCS cold legs - LESS THAN 100°F/HR
- ☐ b) Use RHR, if in service
- ☐ c) Dump steam to condenser from intact SG(s)

☐ c) Dump steam using intact SG(s) PORV.

2-AP-10.07 would be initiated due to the loss of off-site power.

14. ___ DEPRESSURIZE RCS TO REFILL PRZR:

- ☐ a) Use normal PRZR spray
- ☐ b) PRZR level - GREATER THAN 35% [63%]
- ☐ c) Stop RCS depressurization

☐ a) Use one PRZR PORV.

☐ b) GO TO Step 15. WHEN level greater than 35% [63%], THEN stop RCS depressurization.

K/A Number: WE04EA1.3: LOCA Outside Containment / 3: Ability to operate and / or monitor the following as they apply to the (LOCA Outside Containment): Desired operating results during abnormal and emergency situation.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.8 IR-SRO: 4.0

Proposed Question: 73

Initial Conditions:

- Unit 2 operating at 100% power.
- A LOCA has occurred in Unit 2 Safeguards.
- Unit 2 is tripped and safety injection actuated.

Current Conditions:

- RCS pressure is 990 psig and lowering.
- The team has transitioned to 2-ECA-1.2, LOCA Outside Containment.
- The team has just verified LHSI to Hot Leg MOVs closed, 2-SI-MOV-2890A and 2-SI-MOV-2890B.
- The team is ready to perform step 2 of 2-ECA-1.2, “Try to Identify and Isolate Break”.

Which of the following answers the questions below:

1) What is the first component the crew will attempt to close as per 2-ECA-1.2, step 2?
2) What parameter and change will the crew check to determine if the break is isolated?

A. 1) 2-SI-MOV-2890C, LHSI to Cold legs.
2) RCS subcooling > 30°F.

B. 1) 2-SI-MOV-2864A, 2-SI-MOV-2864B, LHSI pump Discharge MOVs.
2) RCS subcooling > 30°F.

C. 1) 2-SI-MOV-2864A, 2-SI-MOV-2864B, LHSI pump Discharge MOVs.
2) RCS pressure rising.

D. 1) 2-SI-MOV-2890C, LHSI to Cold legs.
2) RCS pressure rising.

Proposed Answer: D

Explanation: 1) 2-SI-MOV-2890C is the first component that the crew will close in step 2 of 2-ECA-1.2. Immediately after closing 2-SI-MOV-2890C the crew will check RCS pressure increasing. 2) If RCS pressure is rising then the break is isolated, and the crew will then place the LHSI pumps in PTL and close the LHSI pump suctions 2-SI-MOV-2862A and B.

Technical Reference: 2-ECA-1.2, LOCA Outside Containment. Rev. 8.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-21, ECA-1.2, Objective B; Given a copy of ECA-1.2, LOCA Outside Containment, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7 / 45.5 / 45.6)

Comments:

K/A Analysis: Question requires candidate to have knowledge of specific components that are operated in ECA-1.2 and also the desired operating results, which in this case is rising RCS pressure.

Distractor Analysis:

A. 1) 2-SI-MOV-2890C, LHSI to Cold legs.
2) RCS subcooling > 30ºF.

B. 1) 2-SI-MOV-2864A, 2-SI-MOV-2864B, LHSI pump Discharge MOVs.
2) RCS subcooling > 30ºF.

C. 1) 2-SI-MOV-2864A, 2-SI-MOV-2864B, LHSI pump Discharge MOVs.
2) RCS pressure rising.

D. 1) 2-SI-MOV-2890C, LHSI to Cold legs.
2) RCS pressure rising.

A. Incorrect. 1) Correct. 2) Incorrect. RCS subcooling > 30ºF is an indicator typically used to verify that SI flow is not required. This is plausible if the operator confuses equilibrium indications conditions that no longer require SI flow.
B. Incorrect. 1) Incorrect. 2-SI-MOV-2890C is the component directed to isolate next since this is the MOV closest to the RCS boundary. 2-SI-MOV-2864 A and B plausible because it is next MOV upstream of 2-SI-MOV-2890C. 2) Incorrect. RCS subcooling > 30ºF is an indicator typically used to verify that SI flow is not required. This is plausible if the operator confuses equilibrium indications conditions that no longer require SI flow.
C. Incorrect. 1) Incorrect. 2-SI-MOV-2890C is the component directed to isolate next since this is the MOV closest to the RCS boundary. 2-SI-MOV-2864 A and B plausible because it is next MOV upstream of 2-SI-MOV-2890C. 2) Correct.
D. Correct

Reference: 2-ECA-1.2 Steps 1,2.

NUMBER	PROCEDURE TITLE	REVISION
2-ECA-1.2	LOCA OUTSIDE CONTAINMENT	8
		PAGE 2 of 4

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: Depending on break location, higher than normal dose levels should be expected in the Auxiliary Building and the Safeguards after a LOCA outside CTMT.

* *

1. ___ VERIFY PROPER VALVE ALIGNMENT:

a) Locally unlock and close the following breakers:

- ☐ 2H1-2N 8A for 2-SI-MOV-2890A
- ☐ 2J1-2E 8B for 2-SI-MOV-2890B
- ☐ 2H1-2N 9A for 2-SI-MOV-2890C

b) LHSI to hot legs - CLOSED

- ☐ 2-SI-MOV-2890A
- ☐ 2-SI-MOV-2890B

RNO: ☐ b) Manually close valves. IF valves can NOT be manually closed, THEN locally close valves.

c) SI accumulator test valves - CLOSED

- ☐ HCV-SI-2850A
- ☐ HCV-SI-2850B
- ☐ HCV-SI-2850C
- ☐ HCV-SI-2850D
- ☐ HCV-SI-2850E
- ☐ HCV-SI-2850F

RNO: ☐ c) Manually close valves.

2. ___ TRY TO IDENTIFY AND ISOLATE BREAK:

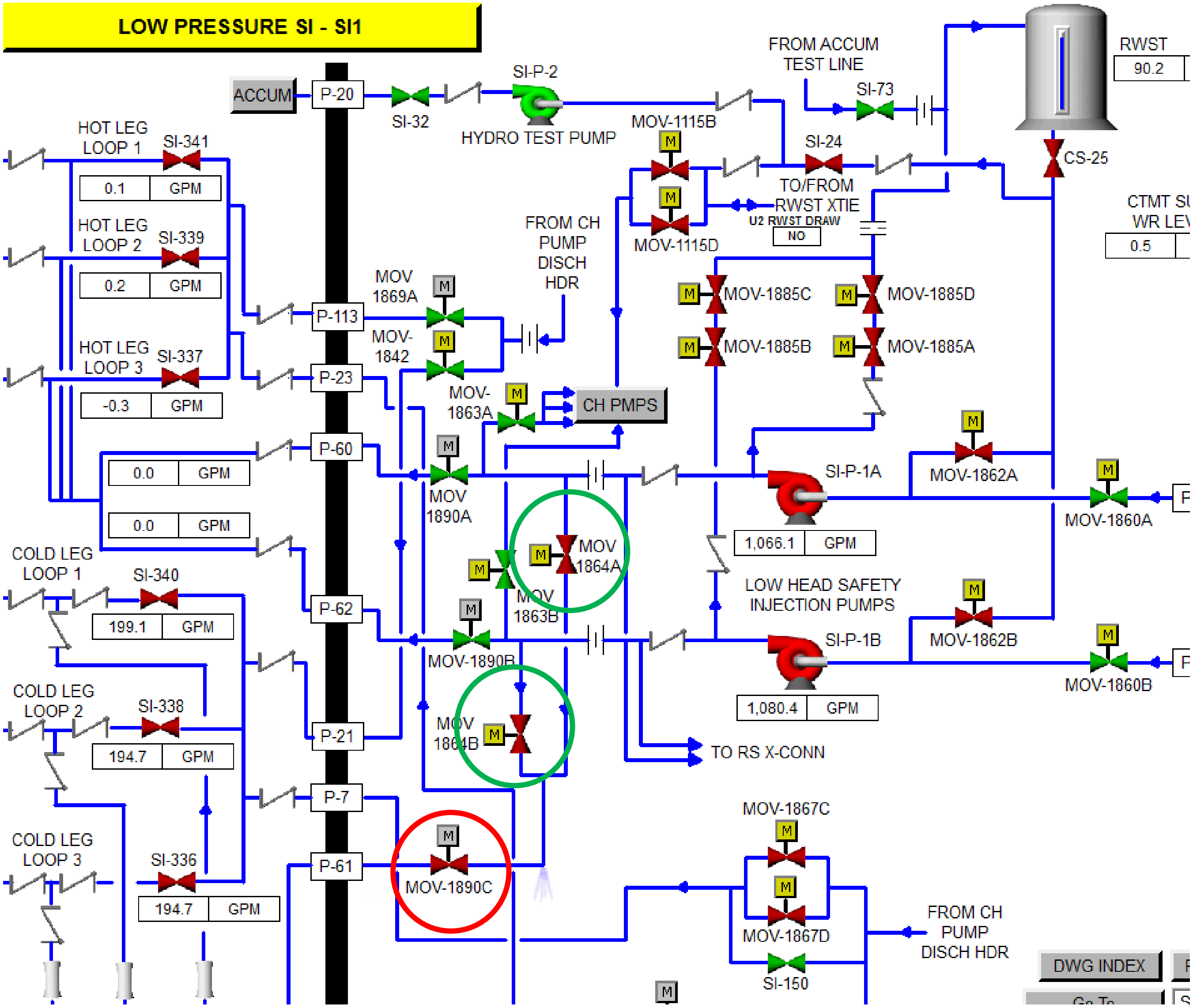
a) Close LHSI to cold legs

- ☐ 2-SI-MOV-2890C

☐ b) Check RCS pressure - INCREASING

RNO: b) Open 2-SI-MOV-2890C AND GO TO Step 3.

Reference Simulation drawing of SI system. Correct shown in red, distractor shown in green.



Reference: ND-95.3-LP-21, ECA-1.2, Basis for step 2.

3. **<u>STEP 2</u>: TRY TO IDENTIFY AND ISOLATE BREAK.**

   a. The purpose of this step is to attempt to identify and isolate the LOCA outside containment.

   b. This step instructs the team to close 1-SI-MOV-1890C and check RCS pressure increasing. IF pressure increasing, secure leakage from RWST and go to 1-E-1, Loss of Reactor or Secondary Coolant.

   c. IF pressure not increasing, reopen 1890C and continue in 1-ECA-1.2.

Reference 2-ES-1.1, SI Termination, Step 10 SI Flow not required.

NUMBER	PROCEDURE TITLE	REVISION
2-ES-1.1	SI TERMINATION	53
		PAGE 6 of 29

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. ___	CHECK IF LHSI PUMPS SHOULD BE STOPPED:	
	☐ a) Check LHSI pumps - ANY RUNNING WITH SUCTION ALIGNED TO RWST	☐ a) GO TO Step 10.
	☐ b) Stop LHSI pumps and put in Auto	
10. ___	CHECK SI FLOW NOT REQUIRED:	
	☐ a) RCS subcooling based on CETCs - GREATER THAN 30°F [85°F]	a) Do the following: ☐ 1) Manually start CHG pumps and align HHSI flow path to RCS cold legs. ☐ 2) GO TO 2-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.
	☐ b) Control charging flow to maintain PRZR level greater than 22% [50%]	☐ b) Control charging flow to raise PRZR level. IF PRZR level can NOT be raised, THEN do the following: ☐ 1) Manually start CHG pumps and align HHSI flow path to RCS cold legs. ☐ 2) GO TO 2-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.

K/A Number: WE05EK2.2: Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4: 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.

Level: RO Tier #: 1 Group #: 1 IR – RO: 3.9 IR-SRO: 4.2

Proposed Question: 74

The following conditions exist:

- FR-H.1, Response to Loss of Secondary Heat Sink, is in effect.
- All SG levels at 10% WR.
- CTMT pressure is 21 psia and stable.
- CETCs are increasing slowly.
- WR Hot Leg temperature is 560°F.
- AFW crosstie from Unit 2 has just been restored.

Which ONE of the following gives the required method of feeding the SGs?

A. One SG should be fed at the maximum available rate.

B. All SGs should be fed at greater than 540 gpm total.

C. All SGs should be fed at greater than 450 gpm total.

D. One SG should be fed at 100 gpm.

Proposed Answer: A.

Explanation: Conditions as described meet the “Hot/Dry SG” feed rate IAW the FR-H.1 Continuous Actions Page.

Technical Reference: FR-H.1, Continuous Actions Page, Rev. 37.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-41, FR-H.1, Objective C, Given a copy of FR-H.1, Response to Loss of Secondary Heat Sink, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: Modified Bank (LEOP0753, Attached)

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7 / 45.7)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must determine based on plant conditions that SG are Hot/Dry and select the appropriate feed rate based on conditions.

Distractor Analysis:

A. Correct. Th is greater than 550°F and CETC temperatures are rising meeting the hot/dry criteria, requiring One SG to be feed at maximum rate.

B. Incorrect. 540 gpm is the rate associated with feeding of SGs with SI not in service and at least one RCP running. Plausible for Candidate to assume RCPs still in operation and confuse feed rate necessary for minimum heat sink and that required to remove decay heat soon after a reactor trip.

C. Incorrect. 450 gpm total is the minimum required for heat sink under adverse Containment conditions if SG are not Hot/Dry. Plausible should Candidate not identify Hot/dry condition of SGs.

D. Incorrect. 100 gpm is the minimum required flow rate in FR-H.1 if CETC are stable or lowering and SG levels are < adverse containment minimum WR level. Plausible should Candidate confuse the feed requirements if CETC stable or lowering with those where CETCs are rising.

CONTINUOUS ACTIONS PAGE FOR 1-FR-H.1

4. RESTORATION OF AUXILIARY FEED FLOW TO HOT DRY SG
(SG with WR level less than 7% [25%] and RCS hot leg temperature greater than 550°F)

- If CETCs are STABLE or LOWERING, feed flow should be limited to 60 gpm [100 gpm] until WIDE RANGE SG level greater than 7% [25%].
- If CETCs are rising, feed flow should be established to ONE SG at the maximum available rate.

NUMBER	PROCEDURE TITLE	REVISION
1-FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	37
		PAGE 4 of 22

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	TRY TO ESTABLISH AFW FLOW TO AT LEAST ONE SG: (Continued)	
	d) Check AFW - AVAILABLE	☐ d) IF minimum feed flow NOT established, THEN do the following:
☐	1) Start AFW Pumps	☐ 1) Stop ALL RCPs.
☐	2) Check total flow to SGs - GREATER THAN 350 GPM [450 GPM]	☐ 2) IF desired to transfer EDG 3 to Bus 2J to restore power to 2-FW-P-3B, THEN GO TO Attachment 2.

NUMBER	PROCEDURE TITLE	REVISION
1-ES-0.1	REACTOR TRIP RESPONSE	51
		PAGE 3 of 16

☐ e) Check total AFW flow - GREATER THAN 540 GPM (350 GPM W/O RCPs)	☐ e) IF SG narrow range level greater than 12% in any SG, THEN control feed flow to maintain narrow range level. ☐ IF SG narrow range level less than 12% in all SGs, THEN manually start pumps AND align valves to establish at least minimum AFW flow.

107	ID: LEOP0753	Points: 1.00

Unit 1 experienced a SBLOCA inside Containment and the team performed E-0, Reactor Trip or Safety Injection, and Safety Injected at step 4. The team transitioned from E-0 to E-1, Loss of Reactor or Secondary Coolant. Following the trip and SI, a loss of all AFW occurred.

The following conditions exist:

- FR-H.1, Response to Loss of Secondary Heat Sink, is in effect.
- All SG levels at 10% WR.
- CTMT pressure is 21 psia and stable.
- CETCs are increasing slowly.
- WR Hot Leg temperature is 560°F.
- AFW crosstie from Unit 2 has just been restored.

Which ONE of the following gives the required method of feeding the SGs?

A. One SG should be fed at the maximum available rate.

B. All SGs should be fed at greater than 450 gpm total.

C. All SGs should be fed at greater than 350 gpm total.

D. One SG should be fed at 100 gpm.

Answer: A

K/A Number: WE15EA2.2: Containment Flooding / 5: Ability to determine and interpret the following as they apply to the (Containment Flooding): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Level: RO Tier #: 1 Group #: 2 IR – RO: 2.9 IR-SRO: 3.3

Proposed Question: 75

Initial Conditions:

- A large break LOCA has occurred on Unit 1 simultaneous with a loss of off-site power.
- All systems function as designed except 1-CS-P-1A tripped 30 seconds after it started.
- The team is performing E-1.

Current Conditions:

- SI is injecting with the RWST at 24%.
- RCS pressure is at containment pressure, which is 40 psia.
- The average of the five highest CETCs is 688°F.
- RVLIS Full Range level is 64%.
- Containment sump level is 7.3 feet and increasing due to a service water leak.

Which ONE of the following gives the correct procedural transition?

A. Go to FR-C.2, Response to Degraded Core Cooling.

B. Go to FR-Z.1, Response to Containment High Pressure.

C. Go to ES-1.3, Transfer to Cold Leg Recirculation.

D. Go to FR-Z.2, Response to Containment Flooding.

Proposed Answer: D

Explanation: FR-Z.2 is the highest priority procedure required to enter. FR-Z.1 is not required because one Containment Spray pump is operating (must lose 2 pumps). FR-C.2 does not apply with RVLIS at 64% (must be < 46%). ES-1.3 does not apply because RWST level is at 24% (must be < 20%).

Technical Reference: 0-F-5, Containment Critical Safety Function Status Tree, Rev. 2. 0-F-2, Core Cooling Critical Safety Function Status Tree, Rev. 1. E-1, Loss of Reactor or Secondary Coolant, Rev. 43.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-49, FR-Z.2, Objective A. Given the Major Action Categories associated with FR-Z.2, Response to Containment Flooding, explain the purpose of FR-Z.2, the transition criteria for entering and exiting FR-Z.2 and the types of operator actions that will occur within each category.

Question Source: Modified Bank. Modified stem, and 1 distractor from LORP LEOP0364

Question History: Last 2 NRC Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43.5 / 45.13)

Comments:

K/A Analysis: K/A requires adherence to appropriate procedures and operation within the limits. The question poses a scenario with multiple indications and the candidate must determine that FR-Z.2, Response to Containment Flooding is the priority procedure.

Distractor Analysis:

A. Incorrect. FR-Z.2 is the highest priority procedure that must be entered. Plausible because some of the conditions for FR-C.2 are in effect; Low subcooling, and no RCPs running. However FR-C.2 does not apply with RVLIS at 64% (must be < 46%).
B. Incorrect. FR-Z.2 is the highest priority procedure that must be entered. Plausible because some of the conditions for FR-Z.1 is met, High Containment pressure. However FR-Z.1 is not required because one Containment Spray pump is operating (must lose 2 pumps).
C. Incorrect. FR-Z.2 is the highest priority procedure that must be entered. Plausible because this is the success path, but ES-1.3 transition conditions are not met because RWST level must be < 20%.
D. Correct.

Reference LORP Exam question LEOP0364. Changed items shown with in red.

356	ID: LEOP0364	Points: 1.00

Initial Conditions:

- A large break LOCA has occurred on Unit 1 simultaneous with a loss of off-site power.
- All systems function as designed except 1-CS-P-1A tripped 30 seconds after it started.
- The team is performing E-1.

Current Conditions:

- SI is injecting with the RWST at 28%.
- RCS pressure is at containment pressure, which is 40 psia.
- The average of the five highest CETCs is 748°F.
- RVLIS Full Range level is 64%.
- Containment sump level is 7.3 feet and increasing due to a service water leak.

Which ONE of the following gives the correct procedural transition?

A. Go to FR-C.2, Response to Degraded Core Cooling.

B. Go to FR-Z.1, Response to Containment High Pressure.

C. Go to FR-Z.2, Response to Containment Flooding.

D. Go to FR-C.1, Response to Inadequate Core Cooling.

Answer: A

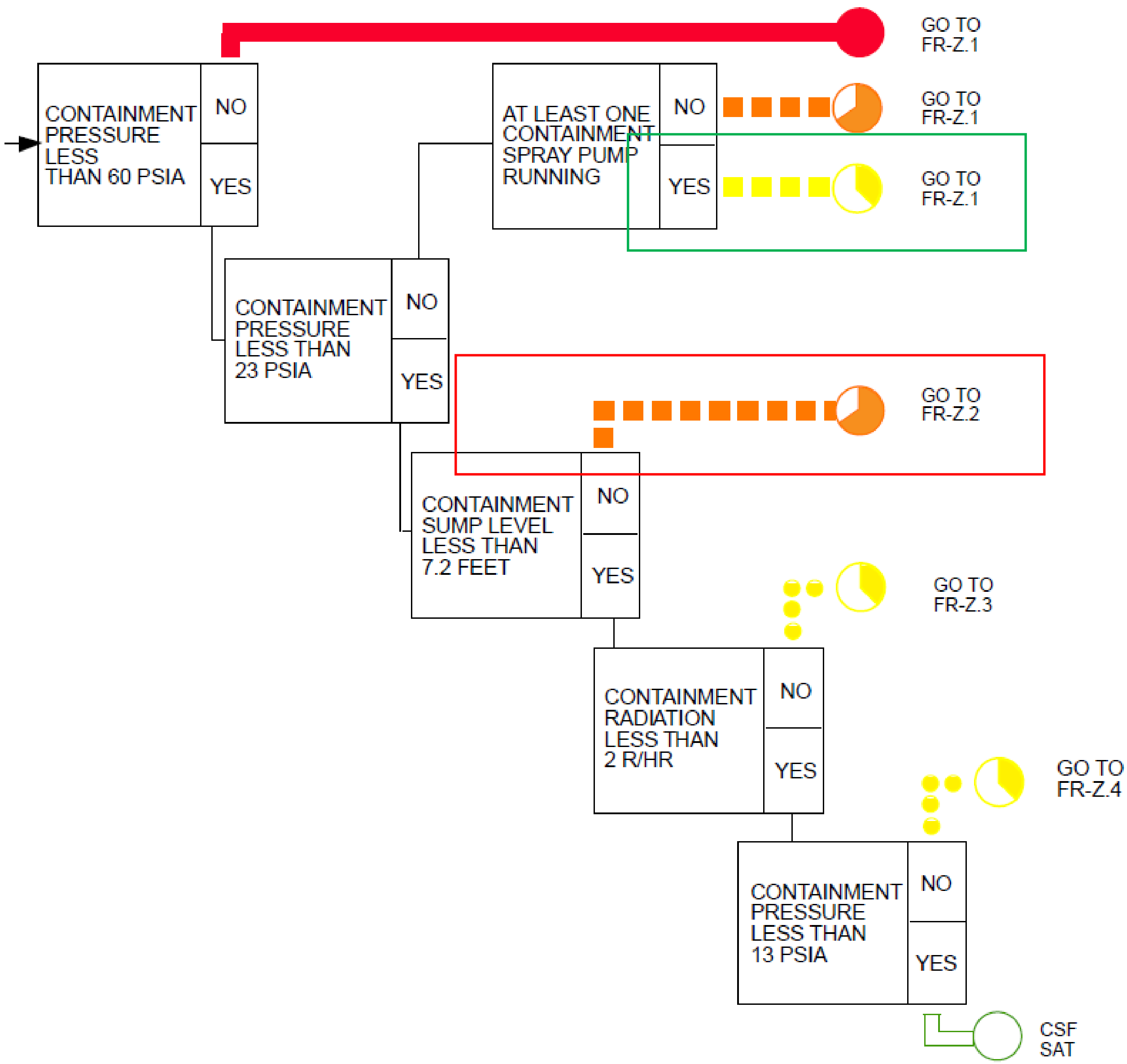
Answer Explanation:

Question requires operator to analyze status trees. 1-CS-P-1A would not require entry into FR-Z.1 (must lose 2 pumps). FR-C.1 does not apply with RVLIS at 64%. FR-C.2 is applicable and takes a higher priority than FR-Z.2.

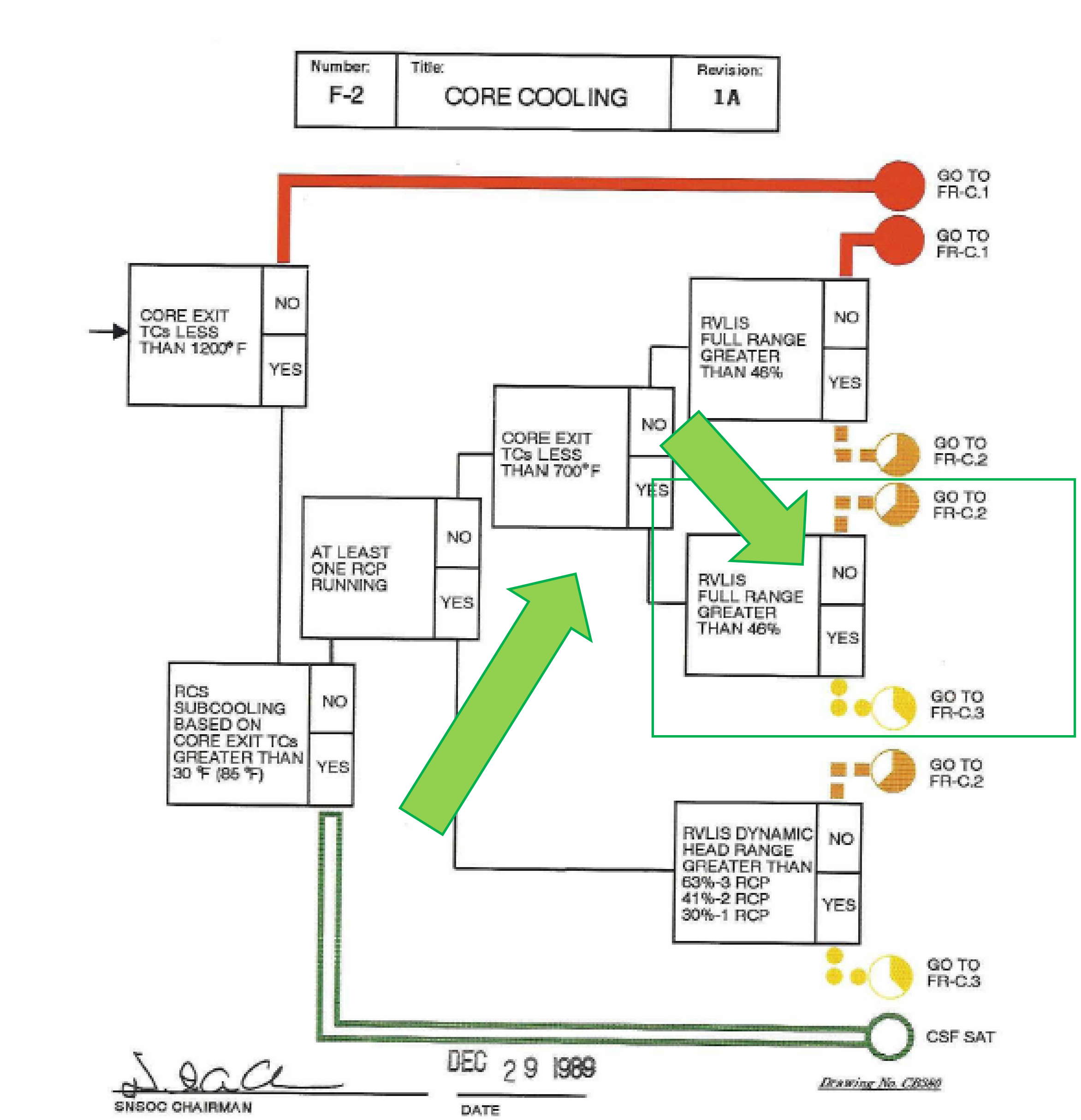
Question 356 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	92832
User-Defined ID:	LEOP0364
Cross Reference Number:	
Topic:	LEOP0364
Num Field 1:	
Num Field 2:	
Text Field:	1.00
Comments:	(B059); [TIR-7542], [TIR-1846], [S05-2356] (O), EOP, E-1, CSFST, FR-C.1, RO-09.6-ST-3, Status Trees, FR-C.2, Not in 2012 Sample Plan CEP-223, CEP-241 (12.2-5)(13.6)

Reference F-5 Containment Status Tree. Shows correct answer and FR-Z.1 distractor.

Number:	Title:	Revision:
F-5	CONTAINMENT	Rev-2



Reference F-2 Core Cooling Status Tree. Shows FR-C.2distractor.



Reference E-1. Shows Transition criteria from E-1 foldout page for ES-1.3.

CONTINUOUS ACTIONS PAGE FOR 1-E-1

1. RCP TRIP CRITERIA

   Trip all RCPs if BOTH conditions listed below occur:

   a. Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS

   b. RCS Subcooling - LESS THAN 30°F [85°F]

2. SI REINITIATION CRITERIA

   Following SI termination or SI flow reduction, manually start SI pumps and align valves as necessary if EITHER condition listed below occurs:

   - RCS subcooling based on CETCs - LESS THAN 30°F [85°F]
   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 22% [50%]

3. MINIFLOW RECIRC CRITERIA

   a. CLOSED - When RCS pressure is less than 1275 psig [1475 psig] AND RCP Trip Criteria are met (RCPs OFF).

   b. OPEN - When RCS pressure is greater than 2000 psig.

4. ADVERSE CONTAINMENT CRITERIA

   Use Adverse Containment setpoints if EITHER condition listed below occurs:

   - Containment Pressure - GREATER THAN 20 PSIA
   - Containment Radiation - GREATER THAN 1.0E5 R/HR

5. SECONDARY INTEGRITY CRITERIA

   Manually start SI pumps as necessary and GO TO 1-E-2, FAULTED STEAM GENERATOR ISOLATION, if any SG pressure is lowering in an uncontrolled manner or has completely depressurized, and has not been isolated.

6. E-3, TRANSITION CRITERIA

   Manually start SI pumps as necessary and GO TO 1-E-3, STEAM GENERATOR TUBE RUPTURE, if any SG level rises in an uncontrolled manner or any SG has abnormal radiation.

7. COLD LEG RECIRCULATION SWITCHOVER CRITERIA

   GO TO 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, if RWST level lowers to less than 20%.

8. AFW SUPPLY SWITCHOVER CRITERIA (Refer to Attachment 5)

   Transfer to one of the following alternate AFW water supplies if ECST level lowers to less than 20%.

   a. 1-CN-TK-2, using 1-CN-150.

   b. 1-CN-TK-3, using AFW Booster Pumps.

   c. AFW Crosstie.

   d. Firemain.

K/A Number: : ::

Level: RO Tier #: Group #: IR – RO: IR-SRO:

Proposed Question:

Proposed Answer:

Explanation:

Technical Reference:

Reference Provided to Applicant:

Learning Objective:

Question Source: Bank

Modified Bank

New

Question History: Last 2 NRC Exams: YES / NO

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content:

Comments:

K/A Number: 002A2.04, Reactor Coolant, 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.

Level: SRO Tier #: 2 Group #: 2 IR – RO: 4.3 IR-SRO: 4.6

Proposed Question: 76

Given the following conditions:

- Unit 1 was at 100% power.
- The reactor tripped due to a loss of offsite power.
- A loss of all feedwater occurred and the crew transitioned to 1-FR-H.1, Response to Loss of Secondary Heat Sink.
- Operators have not been able to restore a source of feedwater and are establishing RCS bleed and feed in accordance with 1-FR-H.1.
- Operators were able to open ONLY ONE PRZR PORV.

Based on these conditions, complete the following:

1) The RCS bleed path is ___________.
2) Once the heat sink is restored the crew will transition to _______.

A. 1) inadequate
2) 1-ES-1.1, SI Termination.

B. 1) adequate
2) 1-E-1, Loss of Reactor or Secondary Coolant, then 1-ES-1.2, Post-LOCA Cooldown and Depressurization.

C. 1) adequate
2) 1-ES-1.1, SI Termination.

D. 1) inadequate
2) 1-E-1, Loss of Reactor or Secondary Coolant, then 1-ES-1.2, Post-LOCA Cooldown and Depressurization.

Proposed Answer: A

Explanation: 1) Per 1-FR-H.1, step 19 the RCS bleed path is inadequate if both PRZR PORVs are not open. Step 19 RNO requires the crew to maintain open any PRZR PORV that is open, open Reactor and PRZR head vents, and locally align any available low pressure water source to the SGs. 2) Per 1-FR-H.1, once all conditions are met for Heat Sink restoration, at step 38 the crew will transiton to 1-ES-1.1.

Technical Reference: 1-FR-H.1, RESPOND TO LOSS OF SECONDARY HEAT SINK. Rev. 37.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-41, FR-H.1 RESPOND TO LOSS OF SECONDARY HEAT SINK, Objective C; “Given a copy of FR-H.1, Response to Loss of Secondary Heat Sink, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.”

Question Source: Modified Bank (EOP0386, format change and modified C2, and D2 for distractor plausibility).

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Comments:

K/A Analysis: The question requires the candidate to assess the impact on the viability of the RCS bleed path with only one PRZR PORV open and determine if the bleed path is adequate or inadequate. This meets the first part of the K/A “predict the impact…” The question goes on to further require the candidate to choose from four separate distractors and determine what actions are needed to correct, or mitigate the consequence of the malfunction. Each part 2 choice provides a viable option to correct or mitigate the situation posed.

Distractor Analysis:

A. Correct.
B. Incorrect. 1) Incorrect, 1-FR-H.1 requires two PORVS for an adequate bleed path. Plausible if candidate doesn’t fully understand basis of step. 2) Incorrect, 1-FR-H.1 states that once Heat Sink is met, the crew can transition to 1-ES-1.1. Plausible because 1-E-1 followed by 1-ES-1.2, Post-LOCA Cooldown and Depressurization, is a normal follow-up procedure for a loss of heat sink.
C. Incorrect. 1) Incorrect, 1-FR-H.1 requires two PORVS for an adequate bleed path. Plausible if candidate doesn’t fully understand basis of step. 2) Correct.
D. Incorrect. 1) Correct. 2) Incorrect, 1-FR-H.1 states that once Heat Sink is met, the crew can transition to 1-ES-1.1. Plausible because 1-E-1 followed by 1-ES-1.2, Post-LOCA Cooldown and Depressurization, is a normal follow-up procedure for a loss of heat sink.

Parent question EOP 0386.

*** **SRO ONLY** ***

Given the following conditions:

- Unit 1 was at 100% power
- The reactor tripped due to a loss of offsite power
- A loss of all SG feedwater occurred, and operators transitioned from 1-E-0, Reactor Trip or Safety Injection to 1-FR-H.1, Response to Loss of Secondary Heat Sink
- Operators could not restore a source of feedwater, and are establishing RCS bleed and feed in accordance with 1-FR-H.1
- Operators were able to open ONLY ONE PRZR PORV.

Based on these plant conditions, the RCS bleed path is _____, and the crew should _____.

A.	inadequate; open Reactor and PRZR vents and align a low-pressure water source to feed SGs.
B.	adequate; depressurize SGs to less than 550 psig while continuing efforts to re-establish a high pressure source to feed SGs.
C.	adequate; open Reactor and PRZR vents and align a low-pressure water source to feed SGs.
D.	inadequate; depressurize SGs to less than 550 psig while continuing efforts to re-establish a high pressure source to feed SGs.

Answer:	A

Question 81 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	124914
User-Defined ID:	EOP0386
Cross Reference Number:	
Topic:	EOP0386
Num Field 1:	
Num Field 2:	
Text Field:	1.00
Comments:	ND-95.3-LP-41C [Defect 2478]

References: 1-FR-H.1 Step 19. Shows correct answer.

19. ___ CHECK ADEQUATE RCS BLEED PATH:	Do the following:
☐ • PRZR PORVs - BOTH OPEN	☐ a) Maintain any PRZR PORV and associated block valve open.
☐ • PRZR PORV block valves - BOTH OPEN	☐ b) Open PRZR and RX head vents.
	☐ c) Locally align any available low pressure water source to the SG(s). IF no low pressure water source available, THEN GO TO Step 20.
	☐ • Fire water
	☐ • AFW booster pumps

1-FR-H.1 Step 24. Used for distractor C2. Note: This would be one of steps performed IF bleed path was adequate.

24. ___ CONTINUE ATTEMPTS TO ESTABLISH SECONDARY HEAT SINK IN AT LEAST ONE SG USING HIGH PRESSURE MAKEUP	Do the following:
☐ • AFW flow (IAW Attachment 6)	☐ a) Depressurize one intact SG to atmospheric pressure using SG PORV.
☐ • MFW flow (IAW Attachment 7)	b) Locally align any available low pressure water source to the depressurized SG:
☐ • Condensate flow (IAW Attachment 8)	☐ • AFW Booster pumps (Attachment 1)
	☐ • Fire water

1-FR-H.1, Step 9 (partial). Used for distractor B2

9.____ TRY TO ESTABLISH FEED FLOW FROM CONDENSATE SYSTEM BY DEPRESSURIZING ONE INTACT SG:

☐ a) Select one intact SG to depressurize

☐ b) Close MSTVs on the non-selected SGs

c) Align CHG pump suction to the RWST:

1) Check open or open CHG pump suction from RWST MOVs

☐ • 1-CH-MOV-1115B
☐ • 1-CH-MOV-1115D

2) Check closed or close CHG pump suction from VCT MOVs

☐ • 1-CH-MOV-1115C
☐ • 1-CH-MOV-1115E

d) Check closed or close letdown isolation valves

☐ • 1-CH-LCV-1460A
☐ • 1-CH-LCV-1460B

☐ e) Control charging flow to maintain PRZR level

☐ f) Dump steam to condenser at maximum rate to depressurize one intact SG to less than 550 psig

f) Dump steam from intact SG:

☐ • Manually use SG PORV.

OR

☐ • Locally use SG PORV IAW Attachment 5.

☐ IF at least one SG can NOT be depressurized, THEN GO TO Step 11.

Reference 1-FR-H.A steps 36-38. Shows both xfer to 1-ES-1.1 and 1-E-1.

NUMBER	PROCEDURE TITLE	REVISION
1-FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	37
		PAGE 22 of 22

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
36.___	CHECK IF LHSI PUMPS SHOULD BE STOPPED:	
	☐ a) Check LHSI pumps - ANY RUNNING WITH SUCTION ALIGNED TO RWST	☐ a) GO TO Step 37.
	b) Check RCS pressure:	☐ b) GO TO 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.
	☐ 1) Pressure - GREATER THAN 250 PSIG [400 PSIG]	
	☐ 2) Pressure - STABLE OR RISING	
	☐ c) Stop LHSI pumps and put in AUTO	

* *

CAUTION: Sufficient charging flow must be maintained when Pressurizer is water solid to prevent a drop in RCS pressure which could lead to a loss of RCS subcooling.

* *

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

K/A Number: 012A2.01, Reactor Protection, Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulty bistable operation.

Level: SRO Tier #: 2 Group #: 1 IR – RO: 3.1 IR-SRO: 3.6

Proposed Question: 77

Initial Conditions:

- Unit 1 is operating at 100%.
- During performance of Channel II testing, Pressurizer Pressure Channel II, 1-RC-PT-1456, High Pressure Reactor trip bistable failed.

In accordance with Tech Spec Table 3.7-1 (see excerpt below);

TABLE 3.7-1
REACTOR TRIP
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
5. Overtemperature ΔT*	3	2	2		6

With the number of operable channels less than __(1)__, the inoperable channel must be placed in __(2)__ with 72 hours.

A. (1) minimum

(2) trip

B. (1) total

(2) trip

C. (1) minimum

(2) bypass

D. (1) total

(2) bypass

Proposed Answer: B.

Explanation: With 1-RC-PT-1456 failed, this places the unit in a 72 hour clock to place the channel in Trip IAW TS Table 3.7-1, Item 5, OA 6. See below for Operator Action 6 requirements. With bistable failed on 1-RC-PT-1456, the SRO must determine, using TS, the correct Operator Action to continue Power Operation. Certain OA's require the bistable to be placed in a bypass condition (RWST) vice trip condition.

Technical Reference: TS 3.7, Item 5, Operator Actions 6.

Reference Provided to Applicant: No

Learning Objective: ND-93.3-LP-10, Rx Protection, Objective C, Explain all RPS reactor trip signals, including setpoints, coincidences, and purpose. ND-88.1-LP-9, Tech Specs, Objective H, Apply the RCS Tech Specs, including for the SRO candidate, the basis behind each specification.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Comments:

K/A Match Analysis: Question matches K/A in that Candidate must recognize the effect of failures and direct the mitigation of the failure.

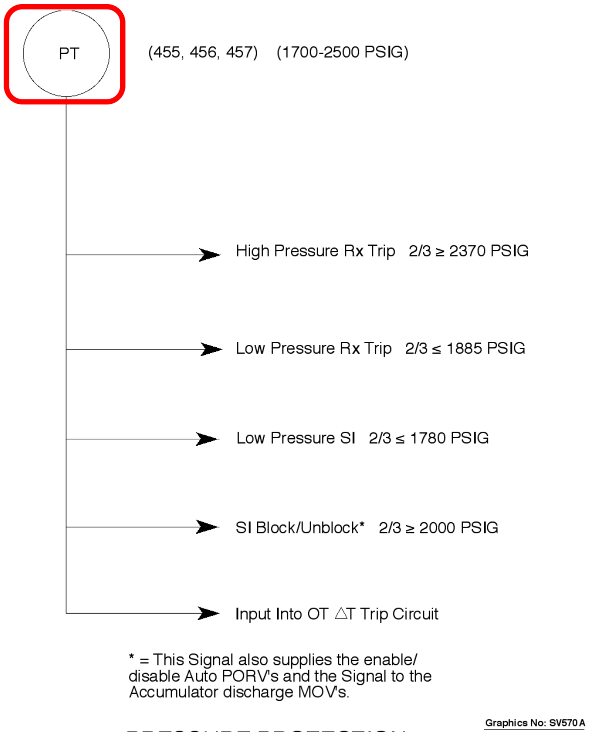
Distractor Analysis:

A. Incorrect. Operator Action 6 states that action is required for less than TOTAL number of channels vice MINIMUM. Plausible if candidate confuses OA6 with other OAs.

B. Correct.

C. Incorrect. Bypass of inoperable channel is allowed for RWST bistables IAW TS 3.7-2, Item 7a, OA 25.

D. Incorrect. Bypass of inoperable channel is allowed for RWST bistables IAW TS 3.7-2, Item 7a, OA 25.



PRESSURE PROTECTION

ACTION 6. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 72 hours.

2. The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1.

If the conditions are not satisfied in the time permitted, be in at least HOT SHUTDOWN within 6 hours.

ACTION 25. With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the bypassed condition within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?

Yes → RO question

No ↓

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes → RO question

No ↓

Can question be answered *solely* by knowing the TS Safety Limits?

Yes → RO question

No ↓

Does the question involve one or more of the following for TS, TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(2) for SRO-only

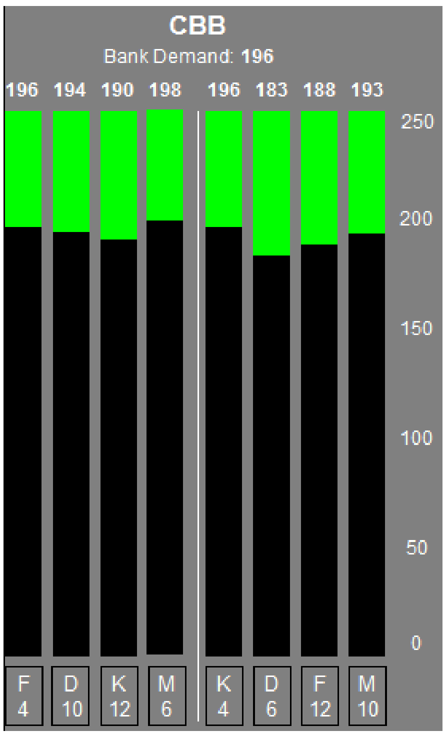
K/A Number: 014G2.2.22, Rod Position Indication, Knowledge of limiting conditions for operations and safety limits.

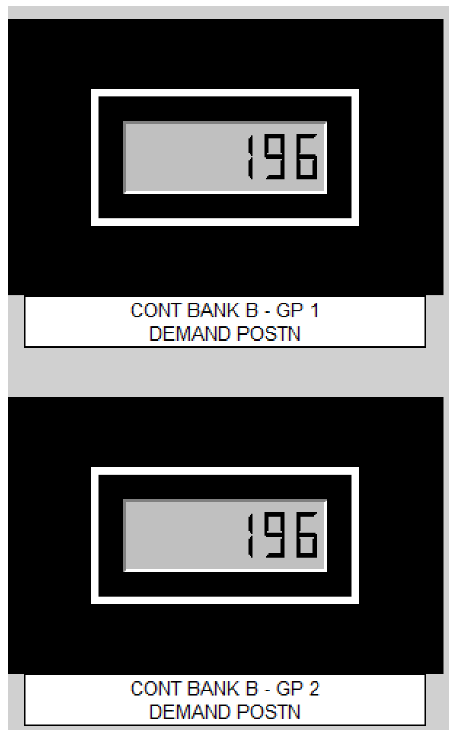
Level: SRO Tier #: 2 Group #: 2 IR – RO: 4.0 IR-SRO: 4.7

Proposed Question: 78

Given the following conditions:

- Unit 1 is performing physics testing following a refueling outage.
- Isothermal Temperature Coefficient determination is in progress.
- RCS temperature is currently being lowered by 3°F.
- MCR indications are:





Which ONE of the following completes the statements:

1) Control Rod D-6 ____ operable.
2) One of the bases for Rod Position Indication is to _____.

A. 1) is not 2) provide required SDM

B. 1) is not 2) limit the reactivity inserted by an ejected rod

C. 1) is 2) limit the reactivity inserted by an ejected rod

D. 1) is 2) provide required SDM

Proposed Answer:

D. is provide required SDM

Explanation: 1) From criticality up to 50% power Rod position indication must be within 24 steps of their respective group demand step counters. From 50% to 100% rod position must be within 12 steps of their group demand step counters. Also, the candidate must also know the basis for the limit.

Technical Reference: Surry Technical Specifications 3.12.E

Reference Provided to Applicant: NO

Learning Objective: ND-93.3-LP-4, CERPI System; Objective D. Describe the Technical Specifications associated with the CERPI System, including for SRO candidates, the basis behind these specifications

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.2 / 45.2)

Comments:

KA Match Analysis: This question matches the KA in that the candidate is required to know the Limiting Condition for Operation for control rod position and the basis for Rod Position Indicating System Limits.

Distractor Analysis:

A. Incorrect. 1) Incorrect as rod D-6 is within 24 steps of its group demand counter. Plausible if the candidate confuses 24 step limit (which applies at this power level) with the 12 step limit. 2) Correct.
B. Incorrect. 1) Incorrect as rod D-6 is within 24 steps of its group demand counter. Plausible if the candidate confuses 24 step limit (which applies at this power level) with the 12 step limit. 2) Incorrect. The basis for maintaining Rod deviation is based on SDM. Plausible because part of the basis for rod insertion limits is based on limiting rod ejection reactivity.
C. Incorrect 1) Correct. 2) Incorrect. The basis for maintaining Rod deviation is based on SDM. Plausible because part of the basis for rod insertion limits is based on limiting rod ejection reactivity.

D. CORRECT.

Tech Spec 3.12 Basis and limits for Rod Position indication.

TS 3.12-11
07-28-11

E. Rod Position Indication System and Bank Demand Position Indication System

1. From movement of control banks to achieve criticality and with the REACTOR CRITICAL, rod position indication shall be provided as follows:

   a. Above 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 12 steps of their respective group step demand counter indications.

   b. From movement of control banks to achieve criticality up to 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 24 steps of their respective group step demand counter indications for a maximum of one hour out of twenty-four, and to within ± 12 steps otherwise.

   c. From movement of control banks to achieve criticality and with the REACTOR CRITICAL, the Bank Demand Position Indication System shall be OPERABLE and capable of determining the group demand positions to within ± 2 steps.

rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step demand counter for that group. The Bank Demand Position Indication System is considered highly precise (± 2 steps).

The Rod Position Indication System provides an accurate indication of actual rod position, but at a lower precision than the group step demand counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. The Rod Position Indication System is capable of monitoring rod position within at least ± 12 steps during steady state temperature conditions and within ± 24 steps during transient temperature conditions. Below 50% RATED POWER, a wider tolerance on indicated rod position for a maximum of one hour in every 24 hours is permitted to allow the system to reach thermal equilibrium. This thermal soak time is available both for a continuous one hour period or several discrete intervals as long as the total time does not exceed 1 hour in any 24 hour period and the indicated rod position does not exceed 24 steps from the group step demand counter position.

The requirements on the rod position indicators and the group step demand counters are only applicable from the movement of control banks to achieve criticality and with the REACTOR CRITICAL, because these are the only conditions in which the rods can affect core power distribution and in which the rods are relied upon to provide required shutdown margin. The various action statement time requirements are based on operating experience and reflect the significance of the circumstances with respect to verification of rod position and potential rod misalignment. Reduction of RATED POWER to less than or equal to 50% puts the core into a condition where rod position is not significantly affecting core peaking factors. Therefore, during operation below 50% RATED POWER, no special monitoring is required. In the shutdown conditions, the operability of the shutdown banks and control banks has the potential to affect the required shutdown margin, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

Reference TS 3.12-13, Basis for Rod Insertion limits (P2 distractor).

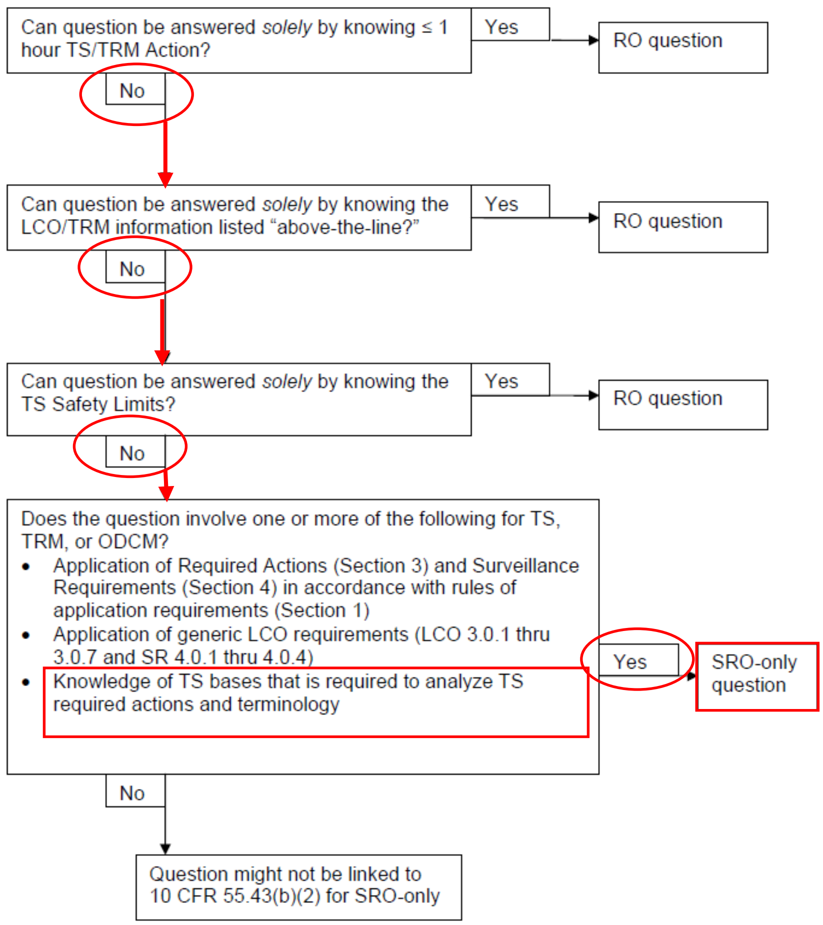
TS 3.12-13
07-28-11

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to COLD SHUTDOWN) are compensated for by changes in the soluble boron concentration. During POWER OPERATION, the shutdown control rod assemblies are fully withdrawn and control of power is by the control banks. A reactor trip occurring during POWER OPERATION will place the reactor into HOT SHUTDOWN. The control rod assembly insertion limits provide for achieving HOT SHUTDOWN by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted control rod assembly worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement.

SRO Justification:

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



NRC APPROVED. K/A Number: 029EA2.09, ATWS / 1, Ability to determine or interpret the following as they apply to a ATWS: Occurrence of a main turbine/reactor trip.

Level: SRO Tier #: 1 Group #: 1 IR – RO: 4.4 IR-SRO: 4.5

Proposed Question: 79

With the plant at 100% power, a loss of all feedwater occurs.

- The SGs reach the LO-LO level reactor trip setpoint, but the reactor fails to trip.
- The crew is NOT successful at tripping the reactor manually.
- The crew enters FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION - ATWS.

Which of the following completes the following statements:

1) The operator completes step 2, “Manually Trip The Turbine” by verifying ALL Turbine Stop valves closed or _________.

2) The Shift Manager will declare a (an) _______ for this event.

(REFERENCE PROVIDED)

A. 1. reducing load using the limiter
2. Alert (SA2.1)

B. 1. checking all Turbine Governor valves closed
2. Alert (SA2.1)

C. 1. reducing load using the limiter
2. Site Area Emergency (SS2.1)

D. 1. checking all Turbine Governor valves closed
2. Site Area Emergency (SS2.1)

Proposed Answer: C

Explanation: 1) Per FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION – ATWS, step 2 is accomplished by verifying all turbine stop valves closed or reducing load using the limiter. 2) Per Surry EAL; if the reactor fails to trip automatically and there is also a failure to manually trip the reactor from the Main Control Room then a Site Area Emergency shall be declared.

Technical Reference: 1-FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION – ATWS, Rev. 26. Surry EAL, Rev. 4.

Reference Provided to Applicant: Yes (SURRY EAL CHART)

Learning Objective: ND-95.1-LP-11, ATWT, Assess the expected sequence of events and the operator actions associated with an Anticipated Transient Without Trip (ATWT). ND-95.5-LP-2, SEM. Objective C, Using EPIP-1.01, Emergency Manager Controlling Procedure, analyze plant situations and determine the appropriate classification utilizing the EAL charts (both HOT and COLD conditions). **(STA/SRO only)**

ND-95.3-LP-36, FR-S.1, Objective B Given the Major Action Categories associated with FR-S.1, Response to Nuclear Power Generation/ATWS, explain the purpose of FR-S.1, the transition criteria for entering and exiting FR-S.1, and the types of operator actions that will occur within each category.

Question Source: NEW

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: ND-95.5-LP-2, SEM. Objective C, **(STA/SRO only)**

Comments:

K/A Analysis: This question tests the candidate's understanding of the proper indications and operator actions to verify/perform a manual turbine trip as required by FR-S.1. Furthermore the question test the SRO candidate's ability to classify the event.

Distractor Analysis:

A. 1. reducing load using the limiter
   2. Alert (SA2.1)

B. 1. checking all Turbine Governor valves closed
   2. Alert (SA2.1)

C. 1. reducing load using the limiter
   2. Site Area Emergency (SS2.1)

D. 1. checking all Turbine Governor valves closed
   2. Site Area Emergency (SS2.1)

A. Incorrect. 1) Correct. 2) Incorrect because EAL would be classified as a SAE. Plausible because the Alert EAL would be met if the Crew would have been able to manually trip the reactor from the MCR.
B. Incorrect. 1) Incorrect because FR-S.1 states to reduce load using the limiter. Plausible because this is the RNO for E-0 step 2. 2) Incorrect because EAL would be classified as a SAE. Plausible because the Alert EAL would be met if the Crew would have been able to manually trip the reactor from the MCR.
C. Correct.
D. Incorrect. 1) Incorrect because FR-S.1 states to reduce load using the limiter. Plausible because this is the RNO for E-0 step 2. 2) Correct.

References from 1-FR-S.1, Step 2, MANUALLY TRIP THE TURBINE (Part 1 Correct Answer)

NUMBER	PROCEDURE TITLE	REVISION
1-FR-S.1	RESPONSE TO NUCLEAR POWER GENERATION/ATWS	26
		PAGE 2 of 9

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: RCPs should not be tripped with Reactor power greater than 5%.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1] ___	VERIFY REACTOR TRIP:	☐ Verify or place control rods in Auto.
	☐ a) Manually trip Reactor	
	b) Check the following:	
	☐ • All Rods On Bottom light - LIT	
	☐ • Reactor Trip and Bypass Breakers - OPEN	
	☐ • Neutron Flux - DECREASING	
[2] ___	MANUALLY TRIP THE TURBINE:	☐ IF turbine will NOT trip, THEN reduce load using limiter. IF turbine load can NOT be reduced, THEN close MSTVs.
	☐ • Verify all turbine stop valves - CLOSED	

References from 1-E-0, Step 2 (Part 1 distractor)

[2]___ CHECK TURBINE TRIP:

☐ a) Manually trip the turbine	
☐ b) Check all turbine stop valves - CLOSED	b) Check either of the following: ☐ • All Turbine Governor Valves - CLOSED OR ☐ • Turbine speed - LOWERING OR ☐ • Generator Motoring - INITIATED ☐ IF turbine will NOT trip, THEN close MSTVs.
c) Isolate reheaters by closing MSR steam supply SOV ☐ • 1-MS-SOV-104	☐ c) IF reheater FCVs will NOT close, THEN close MSR steam supply MOVs.

References from SURRY EAL chart. (Correct answer shown in red, distractor shown in green)

SITE AREA EMERGENCY	ALERT
SS1a Loss of **all** offsite power and loss of **all** onsite AC power to emergency buses 1 2 3 4	SA1 AC power capability to emergency buses reduced to a single power source for greater than 15 minutes such that **any** additional single failure would result in loss of **all** AC power to emergency buses 1 2 3 4
SS1.1 Loss of **all** offsite and onsite AC power to Unit ( ) 4160V emergency buses H and J for > 15 min. (Note 3)	**SA1.1** AC power capability to Unit ( ) 4160V emergency buses H and J reduced to a single power source for > 15 min. (**any** additional single failure would result in loss of **all** AC power to the emergency buses) (Note 3)
SS1b Loss of **all** vital DC power	
SS1.2 Loss of **all** vital DC power based on < 105 volt DC bus voltage indications for > 15 min. (Note 3)	
SS2 Automatic trip fails to shutdown the reactor and manual actions taken from the reactor control console are **not** successful in shutting down the reactor 1 2	SA2 Automatic trip fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor 1 2
SS2.1 An automatic trip failed to shutdown the reactor and manual actions (i.e. trip pushbuttons) taken at the Main Control Room (MCR) Bench Board **do not** shutdown the reactor as indicated by reactor power $\geq$ 5%	**SA2.1** An automatic trip failed to shutdown the reactor and manual actions (i.e. trip pushbuttons) taken at the Main Control Room (MCR) Bench Board successfully shutdown the reactor as indicated by reactor power < 5%

ES 401-9 COMMENTS (WITH CORRECTIONS SHOWN IN RED), 3/28/16

Use of the SRO only flowchart for procedure selection is only valid when the associated question requires the applicant to select a procedure. This one doesn't.

Changed Part 2 to EAL question to better match SRO level. This changed question from a Modified Bank to a NEW question. Also changed the cog level from Memory to Comprehension because new question now requires analysis. Added SRO Only objective to 10CFR PART 55 section. Removed flowchart.

K/A Number: 038EG2.4.8, Steam Gen. Tube Rupture / 3, Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Level: SRO Tier #: 1 Group #: 1 IR – RO: 3.8 IR-SRO: 4.5

Proposed Question: 80

Initial Conditions:

Unit 2 is operating at 100% power.

- 2-AP-16.00, Excessive RCS Leakage, in progress due RCS leakage of 30 gpm into the “C” SG.
- Conditions degrade, and the RO is directed to trip the reactor and perform 2-E-0 due to a rise in RCS leakage.

Current Conditions:

- Over the next 15 minutes RCS leakage is as follows:

At Rx Trip	10 min. after Rx Trip	20 min. after Rx Trip	30 min. after Rx Trip
60 gpm	70 gpm	165 gpm	300 gpm

Using the procedures below, which of the following is the correct procedure sequence following performance of 2-E-0 immediate actions?

2-ES-0.1	Reactor Trip Response
2-E-3	Steam Generator Tube Rupture
2-AP-24.00	Minor SG Tube Leakage
2-AP-24.01	Large SG Tube Leak

A. 1) 2-ES-0.1, 2-AP-24.00, 2-E-3.

B. 1) 2-AP-24.01, 2-E-0, 2-E-3.

C. 1) 2-ES-0.1, 2-AP-24.01, 2-E-0, 2-E-3.

D. 1) 2-AP-24.00, 2-AP-24.01, 2-E-0, 2-E-3.

Proposed Answer: C.

Explanation: Once RCS leakage into “C” SG reaches 50 gpm then 2-AP-16.00 will require a reactor trip. The operator performs the immediate actions of 2-E-0, and at step 4 (no SI required) the operator will enter 2-ES-0.1. In 2-ES-0.1 a Note will direct performance of 2-AP-24.01 following a tube leak of < 150 gpm. When performing steps in 2-AP-24.01, once leakage exceeds the capacity of the charging pumps SI initiation criteria will be met due to the lowering subcooling and the approach to the Low Przr pressure setpoint. 2-AP-24.01 will direct the operator to initiate SI and go to 2-E-0. 2-E-0 steps will progress to the point where 2-E-3 is entered.

Technical Reference: 2-ES-0.1, Reactor Trip Response, Rev. 54. 2-AP-24.01, Large Steam Generator Tube Leak, Rev. 33.

Reference Provided to Applicant: No

Learning Objective: ND-95.2-LP-6, SGTR, Objective D, Given a set of plant conditions, determine the appropriate operator response in accordance with AP-24.00 and AP-24.01 to include the following:

- Immediate Actions (if applicable)
- Entry Conditions
- Major Actions
- Step Bases.

Question Source: NEW

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43(b)5)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must recall specific knowledge of APs and EOP network to determine the sequence of usage in an Emergency involving a Steam Generator Tube Rupture.

Distractor Analysis:

A. Incorrect. Following Reactor Trip, E-0 directs performing ES-0.1 at step 4, and ES-0.1 directs performing AP-24.01 at step 1. AP-24.00 is plausible as it is easily confused with AP-24.01. It is also possible that the crew may have been in this procedure since in AP-16.00, step 7RNO the crew is directed to perform AP-24.00 if reactor trip is not required. Transition to E-3 is plausible since that is the final EOP performed for SGTR.

B. Incorrect. Following Reactor Trip, E-0 directs performing ES-0.1 at step 4, and ES-0.1 directs performing AP-24.01 at step 1. Plausible since AP-24.01 will be the main procedure that is performed since ES-0.1 is a momentary entry.

C. Correct.

D. Incorrect. Following Reactor Trip, E-0 directs performing ES-0.1 at step 4, and ES-0.1 directs performing AP-24.01 at step 1. Plausible since it is entirely possible that crew was performing steps in AP-24.00 at time of Rx. Trip, and AP-24.00 has a transition to AP-24.01.

Reference 2-E-0 showing first transition to 2-ES-0.1 following performance of Immediate actions

[4]___ CHECK IF SI INITIATED:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a) Check SI indications: ☐ • LHSI pumps - RUNNING • SI annunciators - LIT ☐ • A-F-3 ☐ • A-F-4	a) Check if SI is required or imminent as indicated by any of the following: ☐ • Low PRZR pressure ☐ • High CTMT pressure ☐ • High steamline differential pressure ☐ • High steam flow with low Tave or low line pressure ☐ IF SI is required, THEN GO TO Step 4b. ☐ IF SI is NOT required, THEN GO TO 2-ES-0.1, REACTOR TRIP RESPONSE.

Reference 2-ES-0.1 showing second transition to 2-AP-24.01

NUMBER	PROCEDURE TITLE	REVISION
2-ES-0.1	REACTOR TRIP RESPONSE	54
		PAGE 2 of 16

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: If this procedure is being entered from 2-E-0, REACTOR TRIP OR SAFETY INJECTION, following a tube leak of less than 150 gpm, 2-AP-24.01, LARGE STEAM GENERATOR TUBE LEAK, should be used for guidance instead of this procedure.

Reference 2-AP-24.01, showing third transition to 2-E-0 once SGTR rises.

NUMBER	PROCEDURE TITLE	REVISION
2-AP-24.01	LARGE STEAM GENERATOR TUBE LEAK	33
		PAGE 2 of 36

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If the leak rate rises to 150 gpm, 2-E-0, Reactor Trip or Safety Injection, must be implemented.

* *

Reference 2-AP-16.00, showing transition to 2-AP-24.00

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*4.___ CHECK REACTOR TRIP - REQUIRED ☐ • Leak rate - GREATER THAN 50 GPM <u>OR</u> ☐ • Adequate makeup not being provided by blender	☐ GO TO Step 7.
7.___ CHECK SECONDARY RADIATION - NORMAL OR STABLE IF THERE IS PRE-EXISTING TUBE LEAK ☐ • Air Ejector Rad Monitor ☐ • SG Blowdown Rad Monitors ☐ • Main Steam Line Rad Monitors ☐ • Secondary sample ☐ • N-16 Rad Monitors	Do the following: ☐ a) Consult with Shift Manager. ☐ b) <u>IF</u> Reactor trip <u>NOT</u> required, <u>THEN</u> initiate 2-AP-24.00, MINOR SG TUBE LEAK.

Reference 2-AP-24.00 showing transition step to AP-24.01.

NUMBER	PROCEDURE TITLE	REVISION
2-AP-24.00	MINOR SG TUBE LEAK	9
		PAGE 2 of 4

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*1. ___	CHECK REACTOR TRIP - REQUIRED	Do the following: ☐ a) Initiate 0-AP-23.00, RAPID LOAD REDUCTION. (must be in HSD within six hours) ☐ b) GO TO Step 4.
2. ___	INITIATE 2-E-0, REACTOR TRIP OR SAFETY INJECTION	
3. ___	GO TO 2-AP-24.01, LARGE STEAM GENERATOR TUBE LEAK	

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

K/A Number: 039A2.03, Main and Reheat Steam, Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Indications and alarms for main steam and area radiation monitors (during SGTR).

Level: SRO Tier #: 2 Group #: 1 IR – RO: 3.4 IR-SRO: 3.7

Proposed Question: 81

Initial conditions:

- Unit 1 is at 100%
- Przr PORVS 1-RC-PCV-1455C and 1-RC-PCV-1456 are isolated due to Leak by.
- Air ejector Rad Monitor is alarming in ALERT and HIGH.
- N-16 Rad Monitors are alarming in ALERT and HIGH.
- 1-MS-RR-193, N16 Recorder, indicates 200 gpd on all 3 SGs.
- Reactor Trip and SI actuated 30 seconds after Rad Monitors alarmed.

Current conditions (3 minutes from Rx. Trip):

- All SG pressure indications are approximately 980 psig and rising slowly.
- Trend of SG levels from the time of Rx Trip to Now:

SG	Rx Trip	Trip + 1 min.	Trip + 2 min.	Trip + 3 min.
SG 'A' WR	55.3%	44.1%	45.0%	46.1%
SG 'B' WR	56.5%	46.1%	48.5%	51.0%
SG 'C' WR	56.3%	43.2%	44.5%	45.8%

Which of the following answers the questions below:

1) How many Steam Generators are ruptured?
2) Assuming no other failures, which procedure will be performed following 1-E-3, Steam Generator Tube Rupture?

A. 1) Three.
   2) 1-ES-3.1, POST SGTR COOLDOWN USING BACKFILL.

B. 1) Three.
   2) 1-ECA-3.3, SGTR without PRZR PRESS. CONTROL.

C. 1) One.
   2) 1-ES-3.1, POST SGTR COOLDOWN USING BACKFILL.

D. 1) One.
   2) 1-ECA-3.3, SGTR without PRZR PRESS. CONTROL.

Proposed Answer: C. 1-ES-3.1, POST SGTR COOLDOWN USING BACKFILL.

Explanation: 1) The indications given are for a large SGTR. Initially N-16 RR shows high leakage on all 3 SGs, but this is normal indication for a SGTR. The trend in SG level shows the 'B' SG rising much faster. Therefore there is only one S/G that is ruptured, B S/G. 2) The preferred procedural sequence is to perform E-3, and then ES-3.1. This is done to minimize radiological release to the general public. 1-ECA-3.3 is entered if unable to depressurize RCS. That is not the case as RCPs and Pzr spray is available.

Technical Reference: 1-E-3, SGTR, Rev. 51

Reference Provided to Applicant: NO

Learning Objective: ND-95.3-LP-13 – SGTR; Objective B. Given a copy of E-3, Steam Generator Tube Rupture, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43(b)5)

Comments:

KA Match Analysis: This question meets the KA in that the candidate must choose a procedure flowpath to put the unit in a condition required by EOPs to mitigate the consequences of SGTR.

Distractor Analysis:

A. Incorrect 1) Incorrect but plausible if the candidate concludes that there is no intact SG and assumes that all 3 SGs are ruptured because they are all rising. 2) Correct.
B. Incorrect. 1) Incorrect but plausible if the candidate concludes that there is no intact SG and assumes that all 3 SGs are ruptured because they are all rising. 2) Incorrect because Spray flow is available. Plausible if the candidate concludes that with no PORV that depressurization will not be possible at maximum rate.
C. CORRECT.
D. Incorrect. 1) Correct. 2) Incorrect because Spray flow is available. Plausible if the candidate concludes that with no PORV that depressurization will not be possible at maximum rate.

E. Reference ND-95.3-LP-13, showing 1-ES-3.1, preferred method for final depressurization.

NUMBER	PROCEDURE TITLE	REVISION
1-E-3	STEAM GENERATOR TUBE RUPTURE	51
		PAGE 40 of 40

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
49.___	GO TO APPROPRIATE POST-SGTR COOLDOWN METHOD AS DIRECTED BY TSC OR PLANT STAFF:	
	☐ • GO TO 1-ES-3.1, POST-SGTR COOLDOWN USING BACKFILL	
	OR	
	☐ • GO TO 1-ES-3.2, POST-SGTR COOLDOWN USING BLOWDOWN	
	OR	
	☐ • GO TO 1-ES-3.3, POST-SGTR COOLDOWN USING STEAM DUMP	

- END -

F.

c. Three alternate means of performing this post-SGTR cooldown have been developed. Each of the alternate methods has their advantages and limitations.

(1) In general, post-SGTR cooldown using backfill is the preferred method since it minimizes radiological releases and facilitates processing of contaminated coolant. However, this process is slow if no RCP is running. In addition, chemistry of the secondary side water should be considered with respect to potential boron dilution and adverse effects on primary components prior to initiating backflow of secondary fluid. **(rk)**

(2) The SG blowdown method also minimizes radiological releases. In addition, boron dilution and adverse secondary side water chemistry effects are eliminated. However, the storage and processing capabilities of the blowdown system are very limited, and, similar to the backfill method, RCS depressurization is likely to proceed slowly. **(rk)**

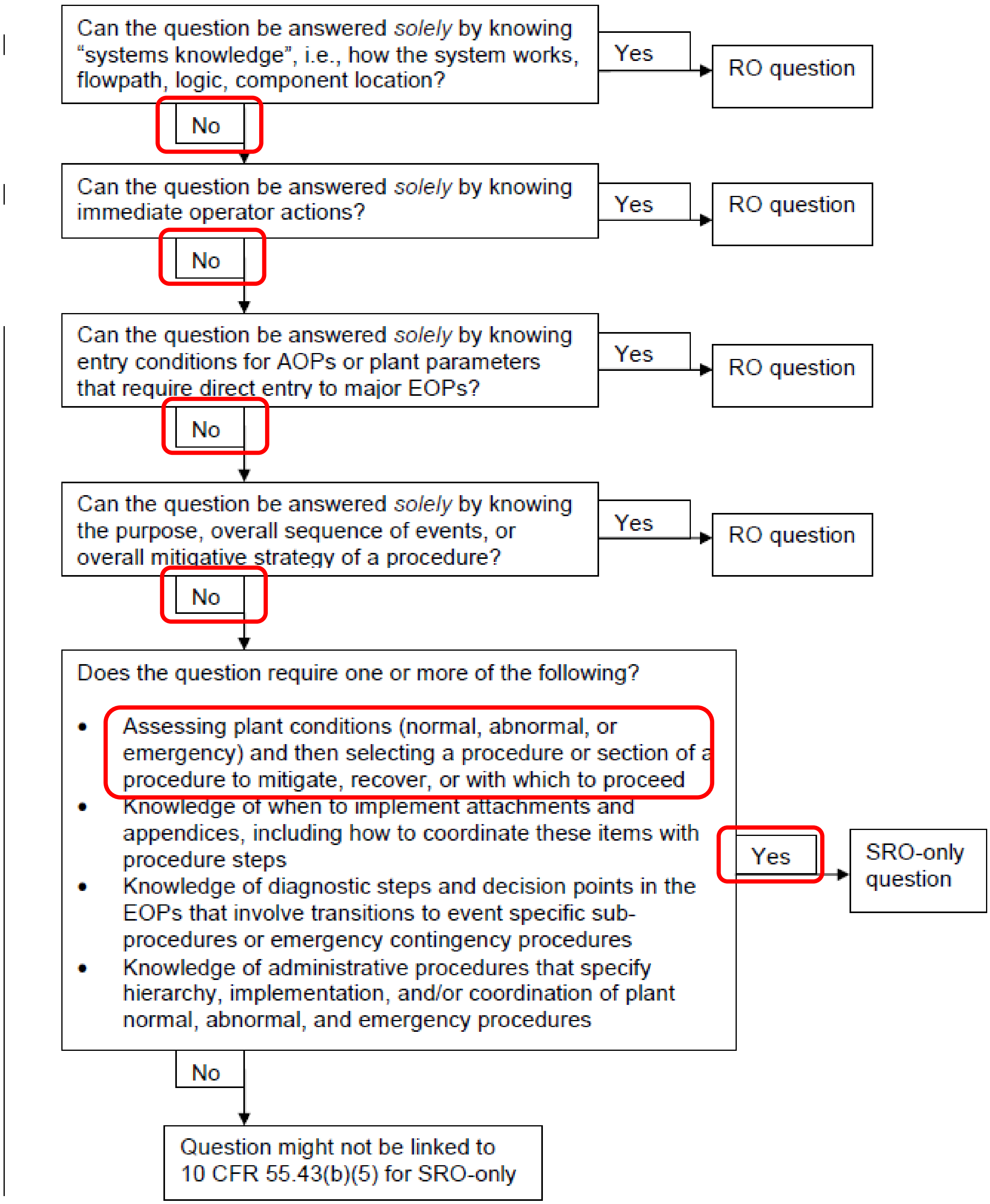
(3) The third alternate method requires steam release from the ruptured SG. This method provides the fastest means of depressurizing the RCS which may be important particularly if FW supply is limited. However, the radiological consequences must be considered particularly if steam dump to the condenser is unavailable (also consider the number of steam leaks present in the steam system). In

Reference 1-E-3 showing transition to 1-ECA-3.3 assuming that depressurization cannot occur because of PORV isolations. Spray is first priority and is available.

NUMBER	PROCEDURE TITLE	REVISION
1-E-3	STEAM GENERATOR TUBE RUPTURE	51
		PAGE 15 of 40

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.___	DEPRESSURIZE RCS TO MINIMIZE BREAK FLOW AND REFILL PRZR:	
	a) Check normal spray - AVAILABLE	☐ a) GO TO Step 20.
☐	• RCP C AND 1-RC-PCV-1455B - BOTH AVAILABLE OR	
☐	• RCPs A and B, AND 1-RC-PCV-1455A - BOTH AVAILABLE	
☐	b) Spray PRZR with maximum available spray	
20.___	DEPRESSURIZE RCS USING PRZR PORV TO MINIMIZE BREAK FLOW AND REFILL PRZR:	
☐	a) PRZR PORV - AT LEAST ONE AVAILABLE	☐ a) GO TO 1-ECA-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL.
	b) Open one PRZR PORV until ANY of the following conditions satisfied: (Attachment 3 lists conditions)	

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



K/A Number: 054AG2.1.23, Loss of Main Feedwater / 4, Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Level: SRO Tier #: 1 Group #: 1 IR – RO: 4.3 IR-SRO: 4.4

Proposed Question: 82

Unit 1 is at 100% Power.

- 1-FW-P-1A and 1-FW-P-1B (A & B Main Feed Pumps) trip.
- A Reactor Trip occurs.
- Neither MDAFW pump (1-FW-P-3A and 1-FW-P-3B) automatically starts.
- All other components operate as designed.

1) Following transition out of 1-E-0, what procedure will address manually starting the MDAFW pumps?
2) In accordance with the Tech Spec excerpt below, _____ hours are allowed to restore to operable status.

TABLE 3.7-2 (Continued)
ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
3. AUXILIARY FEEDWATER (continued)					
e. Trip of main feedwater pumps - start motor driven pumps	2/MFW pump	1/MFW pump	2-1 each MFW pump		24

A. 1) 1-ES-0.1.
 2) 48

B. 1) 1-FR-H.1.
 2) 48

C. 1) 1-ES-0.1.
 2) 72

D. 1) 1-FR-H.1
 2) 72

Proposed Answer: A

Explanation: 1) Since all other components operated as designed, the TDAFWP should start and provide minimum AFW flow, therefore, FRs are not warranted. 2) OA24 of TS states that the AFW pumps need to be restored to operable w/in 48 hours or be in a 12-hour clock to <350°F/450 psig.

Reference Provided to Applicant: No

Learning Objective: ND-95.1-LP-4, Loss of Feedwater; Objective G. Explain the expected plant response for a loss of feedwater accident.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.2 / 45.6)

Comments:

K/A Analysis: The K/A requires demonstration of an ability to perform a specific function related to Loss of Main Feedwater. The question poses a scenario dealing with a loss of Feedwater and requires the candidate to demonstrate specific knowledge as to the procedure flowpath to start AFW pumps that failed to start on an auto-start signal. Therefore this meets the K/A.

Distractor Analysis:

A. Correct.
B. Incorrect. 1) Incorrect. Since the TDAFW pump is supplying the AFW flow necessary for heat sink, 1-FR-H.1 is inappropriate. 2) Correct.
C. Incorrect. 1) Correct. 2) Incorrect. TS Table 3.7-2 OA 24 states that the AFW pumps need to be restored to operable w/in 48 hours or be in a 12-hour clock to <350°F/450 psig.
D. Incorrect. 1) Incorrect. Since the TDAFW pump is supplying the AFW flow necessary for heat sink, 1-FR-H.1 is inappropriate. 2) Incorrect. TS Table 3.7-2 OA 24 states that the AFW pumps need to be restored to operable w/in 48 hours or be in a 12-hour clock to <350°F/450 psig.

TABLE 3.7-2 (Continued)
ENGINEERED SAFEGUARDS ACTION
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Actions
3. AUXILIARY FEEDWATER (continued)					
e. Trip of main feedwater pumps - start motor driven pumps	2/MFW pump	1/MFW pump	2-1 each MFW pump		24

TS 3.7-24
08-31-01

TABLES 3.7-2 ANDS 3.7-3 (Continued)

TABLE NOTATIONS

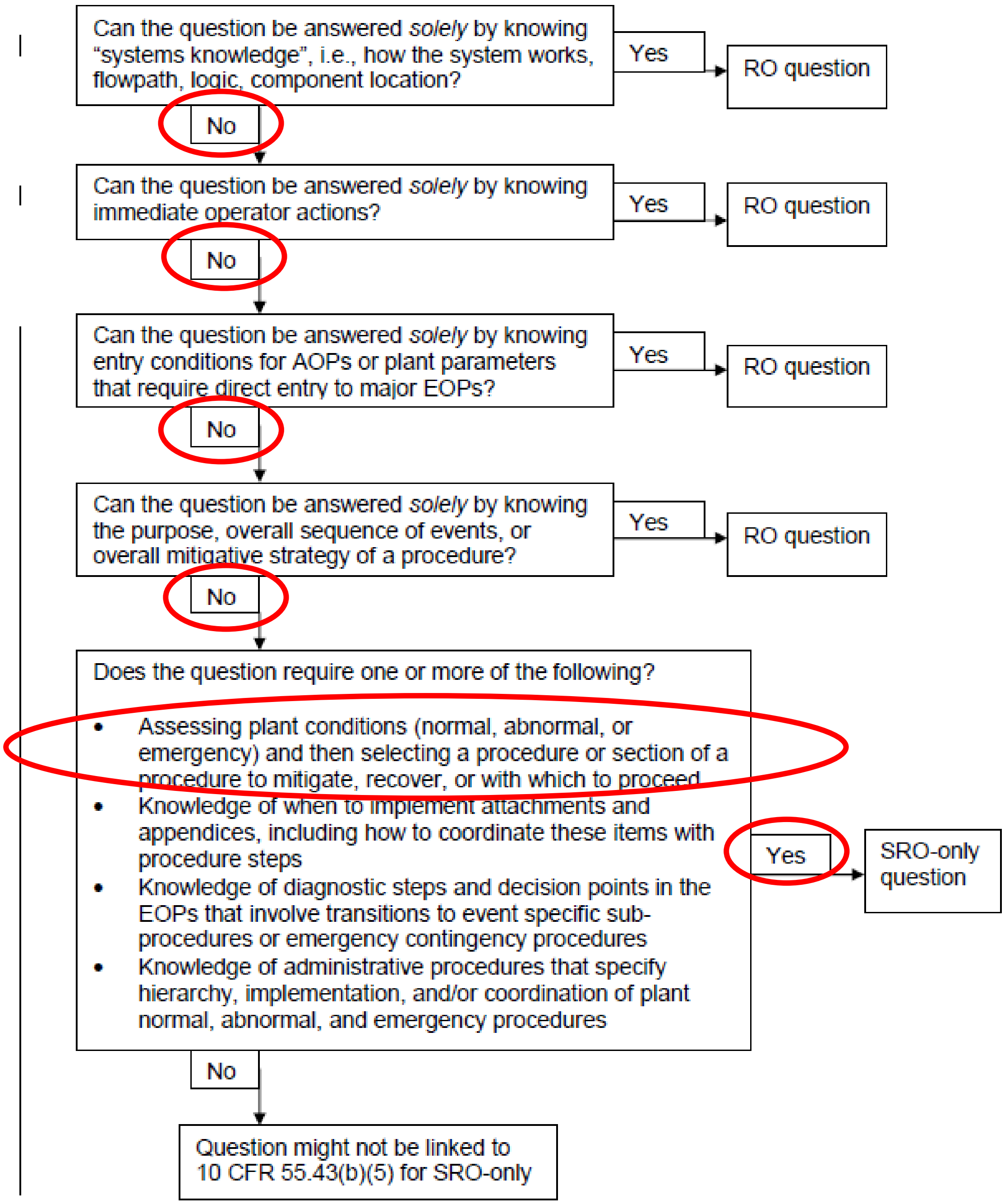
ACTION 21. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 22. With the number of OPERABLE channels one less than the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 6 hours and reduce pressure and temperature to less than 450 psig and 350° within the following 12 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1 provided the other channel is OPERABLE.

ACTION 23. With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

ACTION 24. With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or reduce pressure and temperature to less than 450 psig and 350°F

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



K/A Number: 055EA2.03, Station Blackout / 6, Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power.

Level: SRO Tier #: 1 Group #: 1 IR – RO: 3.9 IR-SRO: 4.7

Proposed Question: 83

Both Units operating at 100% power.

- A loss of off-site power has caused the trip of both Units.
- A number of equipment failures has led to the implementation of 1-ECA-0.0, Loss of All AC Power, on Unit 1.
- The Team has energized 1J emergency buses in accordance with 0-AP-17.06, AAC Diesel Generator – Emergency Operations.
- CTMT pressure is 17.6 psia.
- RCS subcooling is 42 °F and stable.
- PRZR level is 26% and stable.

Which ONE of the following identifies:

1) The <u>First</u> procedure attempted to load EDG #1 is ____1)____.
2) Which procedure, will the Team transition to under these conditions?

A. 1) ECA-0.0, Loss of All AC Power
   2) ECA-0.2, Loss of All AC Power With SI Required.

B. 1) 0-AP-17.04, EDG 1 or EDG 2 – Emergency Operations
   2) ECA-0.2, Loss of All AC Power With SI Required.

C. 1) ECA-0.0, Loss of All AC Power
   2) ECA-0.1, Loss of All AC Power Without SI Required.

D. 1) 0-AP-17.04, EDG 1 or EDG 2 – Emergency Operations
   2) ECA-0.1, Loss of All AC Power Without SI Required.

Proposed Answer: C.

Explanation: ECA-0.0, Step 5 will make the first attempt to start a non-running EDG and attempt to load a running EDG. 0-AP-17.04 is initiated after the attempt to start/load an EDG has failed and the EDG has been shutdown. Caution prior to Step 12 of ECA-0.0 will have the Team go to Step 33 when 1J bus is energized by the AAC Diesel Generator. Transition criteria to ECA-0.1 or 0.2 are listed in step 36.

Technical Reference: 1-ECA-0.0, Loss of All AC Power, Rev. 40.

Reference Provided to Applicant: No.

Learning Objective: ND-95.3-LP-17, ECA-0.0, Objective B, Given a copy of ECA-0.0, Loss of All AC Power, explain the basis of each step of the procedure.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR 43.5 / 45.13)

Comments:

K/A Match Analysis: Question matches the K/A. Candidate must recall ECA-0.0, Step 5 flow to answer Part 1); and the must assess plant condition and apply detailed knowledge of ECA-0.0 to select the appropriate procedure to implement on transition from ECA-0.0.

Distractor Analysis:

A. Incorrect. Part 1) is correct, ECA-0.0, Step 5 directs and attempt to start a non-running EDG and load EDG if/when running but not loaded. Part 2) is incorrect, Caution prior to Step 12 of ECA-0.0 requires the Team to Go To step 33 if 1J bus is energized by the AAC Diesel Generator, and the AAC DG is not needed for Unit 2. Step 33 will lead the Team to Step 36, where the determination of transition to ECA-0.1 or 0.2 is appropriate. Plausible if Candidate determines that SI would be imminent based on PRZR level lowering and choose ECA-0.2 as the appropriate transition.

B. Incorrect. Part 1) is incorrect, ECA-0.0 will direct initiating 0-AP-17.04 to correct the cause of EDG failure. Plausible if the Candidate assumes the EDG is unrecoverable if it fails to respond to an auto start signal. Part 2) is incorrect, Caution prior to Step 12 of ECA-0.0 requires the Team to Go To step 33 if 1J bus is energized by the AAC Diesel Generator, and the AAC DG is not needed for Unit 2. Step 33 will lead the Team to Step 36, where the determination of transition to ECA-0.1 or 0.2 is appropriate. Plausible if Candidate determines that SI would be imminent based on PRZR level lowering and choose ECA-0.2 as the appropriate transition.

C. Correct. Both Parts 1) and 2) are correct.

D. Incorrect. Part 1) is correct, ECA-0.0, Step 5 directs and attempt to start a non-running EDG and load EDG if/when running but not loaded. Plausible if the Candidate assumes the EDG is unrecoverable if it fails to respond to an auto start signal. Part 2) is correct.

NUMBER	PROCEDURE TITLE	REVISION
1-ECA-0.0	LOSS OF ALL AC POWER	40
		PAGE 5 of 30

5. ___ TRY TO RESTORE POWER TO ANY AC EMERGENCY BUS

☐ a) Check EDG - RUNNING

a) Do the following:

☐ 1) Put EDG in EXERCISE.

☐ 2) Start EDG.

☐ 3) Check established or establish generator voltage by depressing Field Flash pushbutton.

☐ 4) Place EDG in AUTO.

☐ 5) GO TO Step 5c.

☐ b) Check EDG in AUTO

5. TRY TO RESTORE POWER TO ANY AC EMERGENCY BUS (Continued)

☐ c) Check AC emergency buses - AT LEAST ONE ENERGIZED

c) Do the following:

☐ 1) IF both EDGs NOT running, THEN initiate 0-AP-17.06, AAC DIESEL GENERATOR - EMERGENCY OPERATIONS, AND GO TO Step 6.

2) Do the following for any running unloaded EDG:

☐ a. Put EDG in EXERCISE.

☐ b. Turn Sync key switch to ON for affected breaker.

☐ c. Check established or establish generator voltage by depressing Field Flash pushbutton.

☐ d. Close EDG output breaker.

☐ e. Turn Sync key switch to OFF for affected breaker.

☐ f. IF either emergency bus energized, THEN GO TO Step 5d.

4) Stop EDG in EXERCISE for any unloaded EDG.

12.___ TRY TO LOCALLY RESTORE AC POWER:

- ☐ a) Initiate AP-17 series procedures to restore EDGs
- ☐ b) Initiate 0-AP-10.08, STATION POWER RESTORATION, to restore power to transfer buses
- ☐ c) Initiate backfeed alignment

* *

CAUTION: • When power is restored to either AC emergency bus from an offsite source or the associated EDG, recovery actions should continue, starting with Step 33.

• If the AAC Diesel Generator is supplying only Bus 1J and is not required by Unit 2, recovery actions should continue, starting with Step 33.

* *

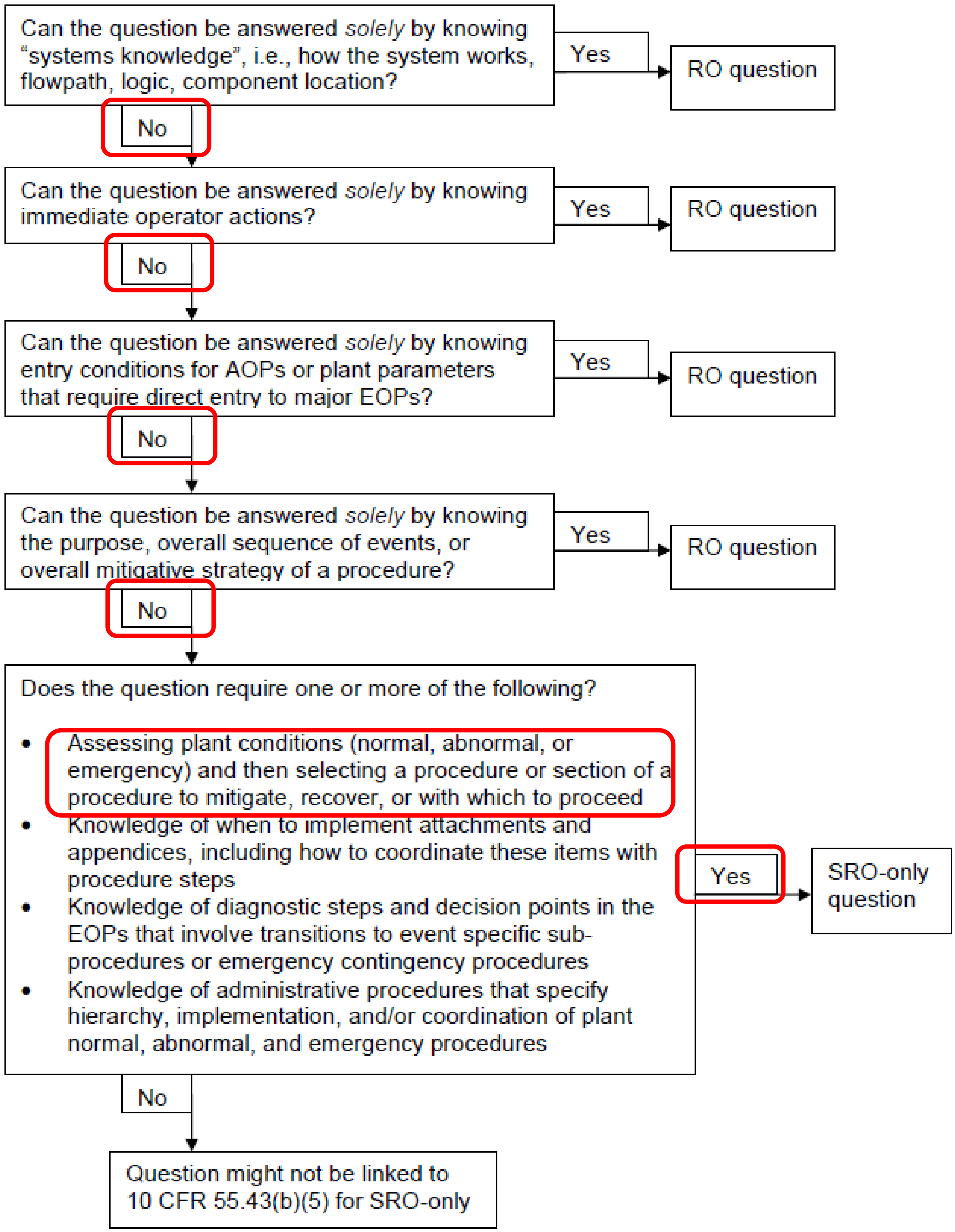
13.___ CHECK STATUS OF THE AAC DIESEL GENERATOR - SUPPLYING BOTH BUSES 1J AND 2H

- ☐ IF the AAC Diesel Generator is supplying only Bus 1J, THEN GO TO Step 33.
- ☐ IF neither Emergency bus energized, THEN GO TO Step 14.

36.___ IDENTIFY RECOVERY PROCEDURE:

- ☐ a) Check RCS subcooling based on CETCs - GREATER THAN 30°F [85°F]
  - ☐ a) GO TO 1-ECA-0.2, LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED.
- ☐ b) Check PRZR level - GREATER THAN 22% [50%]
  - ☐ b) GO TO 1-ECA-0.2, LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED.
- c) Check SI equipment status:
  - ☐ • SI equipment - HAS REMAINED SECURED UPON AC POWER RESTORATION
  - ☐ c) GO TO 1-ECA-0.2, LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



K/A Number: 058AG2.4.20, Loss of DC Power / 6, Knowledge of operational implications of EOP warnings, cautions and notes.

Level: SRO Tier #: 1 Group #: 1 IR – RO: 3.8 IR-SRO: 4.3

Proposed Question: 84

Given the following conditions:

- A Loss of Offsite Power has occurred.
- #1 EDG and #3 EDG have failed to start.
- The team is performing 1-ECA-0.0.
- "A" and "B" Station Batteries have reached 105 VDC.

A complete loss of a DC bus could occur within __(1)__ minutes.

A __(2)__ Fire Watch must be established in the areas served by LP CO_2 within 1 hour due to loss of fire protection.

(REFERENCE PROVIDED)

A. (1) 30; (2) Continuous

B. (1) 30; (2) Shiftly

C. (1) 120; (2) Continuous

D. (1) 120; (2) Shiftly

Proposed Answer: A.

Explanation: ECA-0.0 contains NOTE addressing battery life when voltage reaches 105 VDC (20-30 minutes). The TRM must be referenced to determine the fire watch requirements.

Technical Reference: 1-ECA-0.0 – Loss of All AC Power, rev 40. Surry TRM, Rev. 36

Reference Provided to Applicant: Yes, Surry Technical Requirements Manual

Learning Objective: ND-90.3-LP-6, 125 VDC Distribution; Objective D. Given a total loss of either Station Battery bus, summarize the effects on station operation.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)

Comments:

KA Match Analysis: This question matches the KA in that the candidate must recall a NOTE in 1-ECA-0.0 specifically addressing that once DC bus voltage reaches 105 VDC, a complete loss of the bus is expected within 20-30 minutes. The candidate must be able to use the TRM to determine appropriate fire watch requirements.

Distractor Analysis:

A. CORRECT
B. Incorrect, but plausible if the candidate believes shiftly fire watch is required due to EDG room not Appendix R required.
C. Incorrect, but plausible if the candidate believes normal battery discharge time (2 hours) is based on 105 VDC vice 132 VDC.
D. Incorrect, but plausible if the candidate believes normal battery discharge time (2 hours) is based on 105 VDC vice 132 VDC and a shiftly fire watch is required due to EDG room not Appendix R required.

NUMBER	PROCEDURE TITLE	REVISION
1-ECA-0.0	LOSS OF ALL AC POWER	40
		PAGE 18 of 30

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: When Station batteries or Black batteries reach a voltage of 105 VDC, the voltage will begin to lower exponentially. A complete loss of the bus could occur within 20 to 30 minutes.

22.___ CHECK DC BUS LOADS

a) Notify electricians to monitor the following:

- ☐ Station batteries
- ☐ Black batteries

☐ b) Check ELAP - IN PROGRESS — RNO: ☐ b) GO TO Step 23.

c) Check Vital Instruments - AVAILABLE

- ☐ Any DC Bus Voltage - GREATER THAN 105 VDC
- ☐ Vital Instruments - REQUIRED INSTRUMENTS AVAILABLE

RNO: ☐ c) Perform 1-FSG-7, LOSS OF VITAL INSTRUMENTATION OR CONTROL POWER.

NOTE: If an ELAP is in progress and Vital Bus Stripping has occurred, local monitoring of the ECST will be required.

23.___ CHECK ECST LEVEL - GREATER THAN 20%

RNO: ☐ IF ELAP is in progress AND ECST is available, THEN perform 1-FSG-6, ALTERNATE ECST MAKEUP.

Locally make up from CST using 1-CN-150. IF CST NOT available, THEN consider use of one of the following IAW Attachment 9.

- ☐ Crosstie from Unit 2
- ☐ Fire water

1. There are two (2) battery banks.

   a. Each battery bank provides 132 VDC power for 1800 amp hours, should a loss of emergency bus or UPS battery chargers occur. This is sufficient power to supply vital DC loads for a two (2) hour time period, after which it

is assumed that power will be available to supply the DC loads. This basis is a carryover from non-nuclear stations (no EDGs).

3.7 PLANT SYSTEMS

3.7.3 Low Pressure CO_2 System

TR 3.7.3 The Low Pressure CO_2 System shall be FUNCTIONAL for:

Cable tray rooms (Appendix R - Yes),
Cable vaults and tunnels (Appendix R - Yes),
Safety related charcoal filter banks 3A and 3B (Appendix R - Yes),
Emergency diesel generator rooms (Appendix R - No), and
Normal switchgear rooms (Appendix R - Yes).

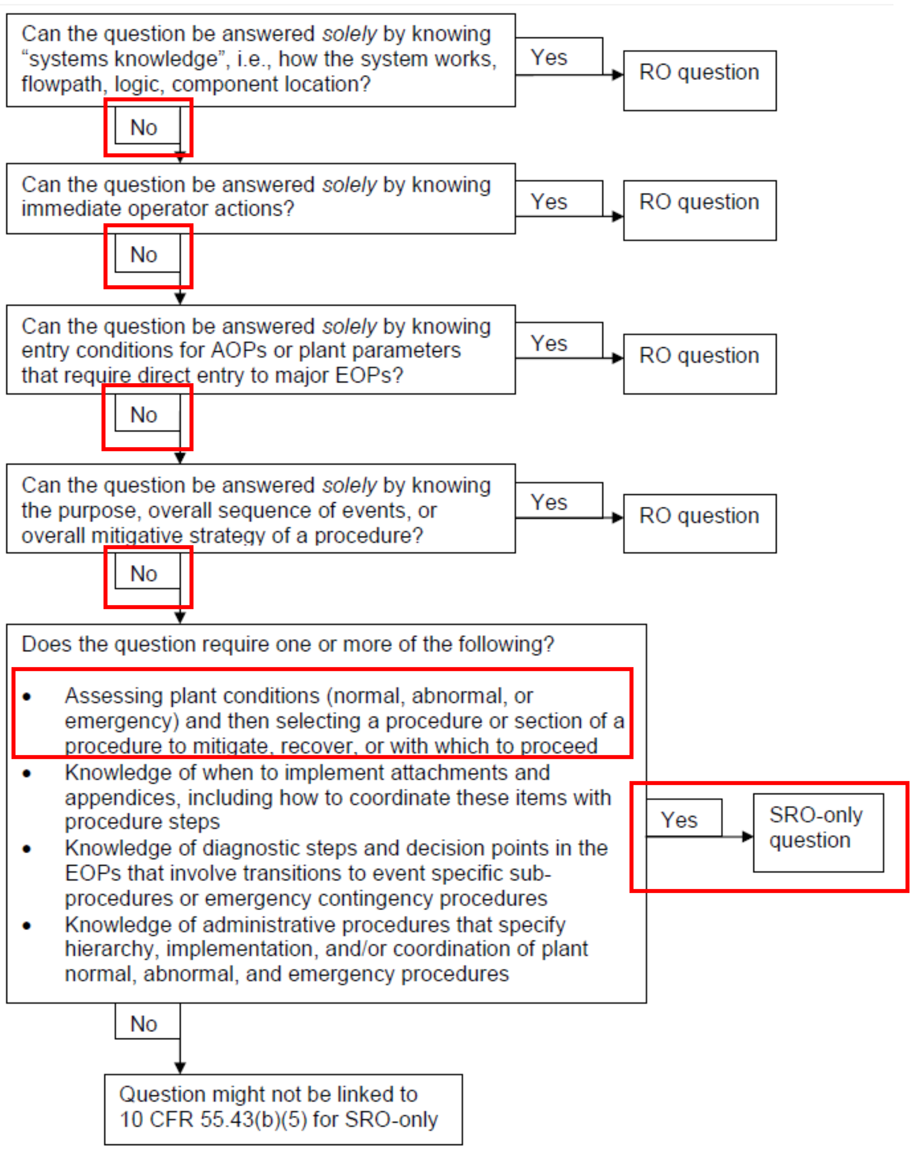
5.2 APPENDIX R AND FIRE PROTECTION COMPENSATORY MEASURES/FIRE WATCH REQUIREMENTS

APPENDIX R AND NON-APPENDIX R FIRE PROTECTION FEATURE NONFUNCTIONALITY AND FIRE WATCH REQUIREMENTS

NONFUNCTIONALITY CONDITION	FIRE WATCH FREQUENCY
If any of the following are nonfunctional, establish a fire watch at the specified frequency.	With one of the noted Appendix R nonfunctionality Conditions, establish an hourly fire watch.[a]
· Fire Detection Instrumentation (3.3.1) · Appendix R Instrumentation (3.3.2) · Suppression Systems ·· Cable Tunnel Spray/Sprinklers (3.7.2) ·· Low Pressure CO_2 System (3.7.3) ·· High Pressure CO_2 System (3.7.4) ·· Halon System (3.7.5)	With two or more of the noted Appendix R nonfunctionality Conditions requiring fire watches in the same fire area, establish a continuous fire watch in that fire area.[a]
· Fire Barriers (3.7.8) (fire-rated wall, floor, ceiling, penetration seal; fire door; fire damper; etc.) · Appendix R Alternate Shutdown Equipment (3.7.9)	With one of the noted Appendix R nonfunctionality Conditions and one or more of the noted non-Appendix R nonfunctionality Conditions requiring fire watches in the same fire area, establish a continuous fire watch in that fire area.[a]
· Passive Fire Protection Items (3.7.14) (radiant energy shields, cable tray covers/firestops, fire retardant coatings, etc.)	With one of the noted non-Appendix R nonfunctionality Conditions, establish a once per shift fire watch.[a]
Fire Watch Frequency Summary App R = Hourly App R + App R = Continuous App R + Non-App R = Continuous Non-App R = Shiftly Non-App R + Non-App R = Hourly	With two or more of the noted non-Appendix R nonfunctionality Conditions requiring fire watches in the same fire area, establish an hourly fire watch in that fire area.[a]

a. Refer to CM-AA-FPA-100 Attachment 2 Paragraph 3.9.6 for guidance with respect to alternative compensatory measures.

SRO Justification: Requires application of procedural notes in order to select a required course of action.



NRC APPROVED. K/A Number: 061AG2.4.30, ARM System Alarms /7, Knowledge of events related to system operations/status that must be reported to internal orginizations or external agencies, such as the State, the NRC, or the transmission system operator.

Note to NRC Chief Examiner: Following review of this question Senior Operations Manager *requests* adding the third bullet below. This is based on perception during validation that question is *leading* the SRO candidate to only consider VPAP-2802, Reportability procedure.

Level: SRO	Tier #: 1	Group #: 2	IR – RO: 2.7	IR-SRO: 4.1

Proposed Question: 85

Chemistry reports the following related to a Unit 1 RCS sample:

- Area Radiation measurements 1 foot away from a 1ml RCS sample indicate 13.9 mr/hr.
- RCS Activity also rose to 10.2 µCi/cc and has now been > 1.0 µCi/cc for the last 36 hours.
- Shift Manager has entered EPIP-1.01, Emergency Manager Controlling Procedure.

Which of the following completes the following statements?

1) Per Technical Specifications the reactor __(1)__ required to be shut down and cooled to 500°F or less within 6 hours.
2) The NRC must be notified of this event within a maximum time of __(2)__ hour(s).

(REFERENCE PROVIDED)

A. 1) is
   2) four

B. 1) is not
   2) four

C. 1) is not
   2) one

D. 1) is
   2) one

Proposed Answer: D

Explanation: 1) Per Tech Specs the specific activity may exceed 1.0 µCi/cc for up to 48 hours as long as the activity does not exceed 10.0 µCi/cc. The activity has just been determined to be > 10.0 µCi/cc. Tech Specs states that the reactor is required to be shutdown and cooled to 500°F or less within 6 hours. 2) Exceeding 13.7 mr/hr at one foot from a 1 ml RCS meets EAL threshold of FA1.1, therefore an EAL must be declared and the NRC notified within 1 hour maximum.

Technical Reference: Surry Technical Specifications, Change 47. Surry EAL, Revision 4.

Reference Provided to Applicant: Yes; VPAP-2802, Notifications and Reports. Surry EAL Matrix.

Learning Objective: ND-95.5-LP-2, SEM, Objective C; Using EPIP-1.01, Emergency Manager Controlling Procedure, analyze plant situations and determine the appropriate classification utilizing the EAL charts (both HOT and COLD conditions)**(STA/SRO only)**.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: ND-95.5-LP-2, SEM; objective C **(STA/SRO only)**

Comments:

K/A Analysis: Question poses a scenario where Chemistry reports local area radiation readings near an RCS sample. This meets the K/A since answer results in an EAL declaration which requires a maximum of 1 hour report to the NRC.

Distractor Analysis:

A. Incorrect. 1) Correct. 2) Incorrect as conditions requires an EAL declaration which requires NRC notification within one hour. Plausible since a tech spec required shutdown is a 4-hour notification per VPAP-2802 and conditions already require a shutdown.
B. Incorrect. 1) Incorrect. Tech specs requires immediate shutdown if Activity exceeds 10.0 µCi/cc. Plausible because the other Tech spec requires shutdown if RCS activity exceeds 1.0 µCi/cc for > 48 hours. The time Activity exceeds 1.0 µCi/cc is 36 hours. 2) Incorrect as conditions requires an EAL declaration which requires NRC notification within one hour. Plausible since a tech spec required shutdown is a 4-hour notification per VPAP-2802 and conditions already require a shutdown.
C. Incorrect. 1) Incorrect. Tech specs requires immediate shutdown if Activity exceeds 10.0 µCi/cc. Plausible because the other Tech spec requires shutdown if RCS activity exceeds 1.0 µCi/cc for > 48 hours. The time Activity exceeds 1.0 µCi/cc is 36 hours. 2) Correct.
D. Correct.

Technical Specifications showing actions for High activity. (correct answer-red, distractor-green)

D. Maximum Reactor Coolant Activity

Specifications

1. The total specific activity of the reactor coolant due to nuclides with half-lives of more than 15 minutes shall not exceed $100/\bar{\bar{E}}$ μCi/cc whenever the reactor is critical or the average temperature is greater than 500°F, where $\bar{\bar{E}}$ is the average sum of the beta and gamma energies, in Mev, per disintegration. If this limit is not satisfied, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection. Should this limit be exceeded by 25%, the reactor shall be made subcritical and cooled to 500°F or less within 2 hours after detection.

2. The specific activity of the reactor coolant shall be limited to ≤ 1.0 μCi/cc DOSE EQUIVALENT I-131 whenever the reactor is critical or the average temperature is greater than 500°F.

3. The requirements of D-2 above may be modified to allow the specific activity of the reactor coolant > 1.0 μCi/cc DOSE EQUIVALENT I-131 but less than 10.0 μCi/cc DOSE EQUIVALENT I-131. Following shutdown, the unit may be restarted and/or operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. With the specific activity of the reactor coolant > 1.0 μCi/cc DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 10.0 μCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection.

Reference showing Surry EAL for fuel element failure.

Any loss or any potential loss of either Fuel Clad or RCS	Any loss or any potential loss of Containment
1 2 3 4	1 2 3 4
FA1.1 Any loss or any potential loss of either Fuel Clad or RCS (Table F-1)	**FU1.1** Any loss or any potential loss of Containment (Table F-1)

	Fuel Cladding Barrier	
	Loss	**Potential Loss**
A.CSFST	1. CSFST Core Cooling-RED	1. CSFST Core Cooling-ORANGE **OR** CSFST Heat Sink-RED and heat sink required
B.Core Exit TCs	2. Core exit TCs > 1,200°F	2. Core exit TCs > 700°F
C.Radiation	3. Containment High Range Radiation Monitor > Table F-3 Fuel Clad Loss threshold 4. Dose rate at one foot from **EITHER**: 1 ml RCS sample ≥ 13.7 mR/hr **OR** 20 ml RCS sample ≥ 250 mR/hr **OR** 250 ml RCS sample ≥ 2,800 mR/hr 5. With letdown in service, Reactor Coolant Letdown Radiation Monitor CH-RI-( )18 or CH-RI-( )19 > 5.0 x 10^6 cpm	None

Reference showing VPAP-2802 (part 2 distractor)

6.3.4 **Four-hour Notifications**

NOTE: Some conditions, indicated by "See EPIP-1.01," may exceed an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure. If a condition exceeds an EAL, EPIPs control State and Federal agency notifications. If an event or condition does not exceed an EAL, it may still be reportable in accordance with this procedure.

a. As soon as practical, but within four hours, the Shift Manager shall notify the NRC Operations Center via the ENS of:

NOTE: If a unit enters a limiting condition for operation (LCO) and a unit shutdown is started due to the LCO, the event is reportable even if shutdown is not completed. LCOs terminated by a unit shutdown for an unrelated reason are still reportable if the condition would not have been corrected within the LCO time limit for shutdown.

1. Initiation of plant shutdown (reduction of power or temperature) required by Technical Specifications. The initiation of plant shutdown does not include mode changes required by Technical Specifications if initiated after the plant is already in a shutdown condition. See EPIP-1.01. **[10 CFR 50.72(b)(2)(i), 10 CFR 50.36 (c)(2)(i), NUREG 1022 Item 3.2.1]**

ES 401-9 COMMENTS (WITH CORRECTIONS SHOWN IN RED), 3/28/16

Window dressing. I think the applicant can correctly answer the question if the stem begins with the Chem Tech report. This history is not relevant.

Removed the Initial Conditions, and the first bullet from current conditions.

ES 401-9 COMMENTS, 3/30/16: The question is fine. The SRO justification is inappropriate. This is not a TS bases question.

Removed Figure 1 from References section.

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

Level: SRO Tier #: 2 Group #: 1 IR – RO: 3.4 IR-SRO: 3.1

Proposed Question: 86

Initial conditions:

- Unit 1 and 2 are operating at 100% power.
- Annunciator VSP-L7, LLIS TROUBLE alarms.
- The RO reports that 4160 V bus 1G has lost power.

Current conditions:

- Intake Canal level is 26 feet and lowering.
- The Electricians report there is a failed trip coil for breaker 15-G1, Supply to bus 1G.
- Power has just been restored to bus 1G per 0-AP-12.01, Loss of Intake Canal Level.

Which ONE of the following identifies:

1) Per 0-AP-12.01 the crew ____ required to wait 5 minutes before starting a circ water pump that has just tripped.
2) The minimum intake canal level required per TS 3.14 (Circulating and SW) Bases to provide adequate design flow through RS SW heat exchangers is ____ feet.

A. 1) is
2) 23.5

B. 1) is not
2) 23.5

C. 1) is
2) 17.2

D. 1) is not
2) 17.2

Proposed Answer: C.

Explanation: 1) 0-AP-12.01, Loss of Intake Canal Level will restore power to 1G bus by operating the cross-tie breaker. Attachment 3 states that "if starting circ water pumps that have just tripped, 5 minutes must be allowed for coastdown prior to starting. 2) Canal Level minimum level of 17.2 feet toprovide adequate design flow for RS SW heat exchangers is described in TS 3.14, Circulating and Service Water Systems Basis.

Technical Reference: TS 3.1.A.6 (Relief Valves) d. Inoperable PRZR PORV Block valve. TS 3.16.A.3 non-compliance, requiring invocation of TS 3.01.

Reference Provided to Applicant: No

Learning Objective: ND-90.3-LP-7, SS and Emergency Protection and Control, Objective H, Describe the requirements of Tech Spec section 3.6, 3.9, and 3.16 concerning the electrical distribution system, including for SRO candidates, the bases behind these specifications. ND-88.1-LP-9, Tech Specs, Objective H, Apply the RCS Tech Specs, including for the SRO candidate, the basis behind each specification.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Comments:

K/A Match Analysis: Question matches K/A. Candidate must predict the impact of a 1G Bus loss, and determine how it impacts the Circ Water pumps with regard to restart criteria for starting a tripped CW pump.

Distractor Analysis:

A. Incorrect. Part 1) is correct. Part 2) is incorrect, 17.2 feet is the minimum per Tech Specs 3.14 Basis. Plausible 23.5 feet is also discussed in TS Basis as the value that will initiate a Turbine trip on both units.

B. Incorrect. Part 1) is incorrect, per 0-AP-12.01, 5 minutes is minimum amount of time that the operator must wait on before attempting a restart of a circ water pump that tripped. Plausible because this is not a requirement in OP-48.1.1. Part 2) is incorrect, 17.2 feet is the minimum per Tech Specs 3.14 Basis. Plausible 23.5 feet is also discussed in TS Basis as the value that will initiate a Turbine trip on both units.

C. Correct. Both Parts 1) and 2) are correct.

D. Incorrect. Part 1) is incorrect, per 0-AP-12.01, 5 minutes is minimum amount of time that the operator must wait on before attempting a restart of a circ water pump that tripped. Plausible because this is not a requirement in OP-48.1.1. Part 2) is correct, a 6/30 is in effect, but it is not the most limiting. Plausible in that the clock is in effect.

Reference 0-AP-12.01, Attachment 3.

NUMBER 0-AP-12.01	ATTACHMENT TITLE REMOTE START OF CIRCULATING WATER PUMPS	ATTACHMENT 3
REVISION 35		PAGE 1 of 1

1. STARTING CIRCULATING WATER PUMPS FROM THE MCR OR LLIS

NOTE: If starting circ water pumps that have just tripped, 5 minutes must be allowed for coastdown before starting.

- ☐ a. Check 4160 VAC G bus energized by Normal Supply or X-Tie.
- ☐ b. Check the LOCAL CONTROL indication for the pump to be started is NOT LIT. IF remote start is inoperable, THEN check CW pumps to be started are in LOCAL.
- ☐ c. Select Soft Control for the pump to be started. N/A if local start to occur.
- ☐ d. Enable Soft Control and check red border. N/A if local start to occur.
- ☐ e. Start the selected Circulating Water Pump by pushing the START button. IF local start to occur due to inoperable PCS, THEN start CW pumps locally.
- ☐ f. Check amps indicated for pump started.
- ☐ g. Direct Outside Operator to perform local operational checks IAW OP-48.1.1.

Reference 0-OP-48.1.1,

5.7 **Circulating Water Pump Start**

CAUTION

To provide CW Pump cooling and spray to Screen, the associated Fish Screen Wash Pump must be started as soon as possible if this is the first CW Pump that will be started.

______ 5.7.1 Identify the Circulating Water Pump (CW Pump) that will be started. (✓)

____ 1-CW-P-1A (1-EP-BKR-15G2)

____ 1-CW-P-1B (1-EP-BKR-15G3)

____ 1-CW-P-1C (1-EP-BKR-15G6)

____ 1-CW-P-1D (1-EP-BKR-15G7)

____ 2-CW-P-1A (2-EP-BKR-25G2)

____ 2-CW-P-1B (2-EP-BKR-25G3

____ 2-CW-P-1C (2-EP-BKR-25G6)

____ 2-CW-P-1D (2-EP-BKR-25G7)

5.7.2 Start the CW Pump locally IAW the following. IF the CW Pump will be started remotely, THEN N/A Substeps 5.7.2.a through 5.7.2.e AND GO TO Step 5.7.3.

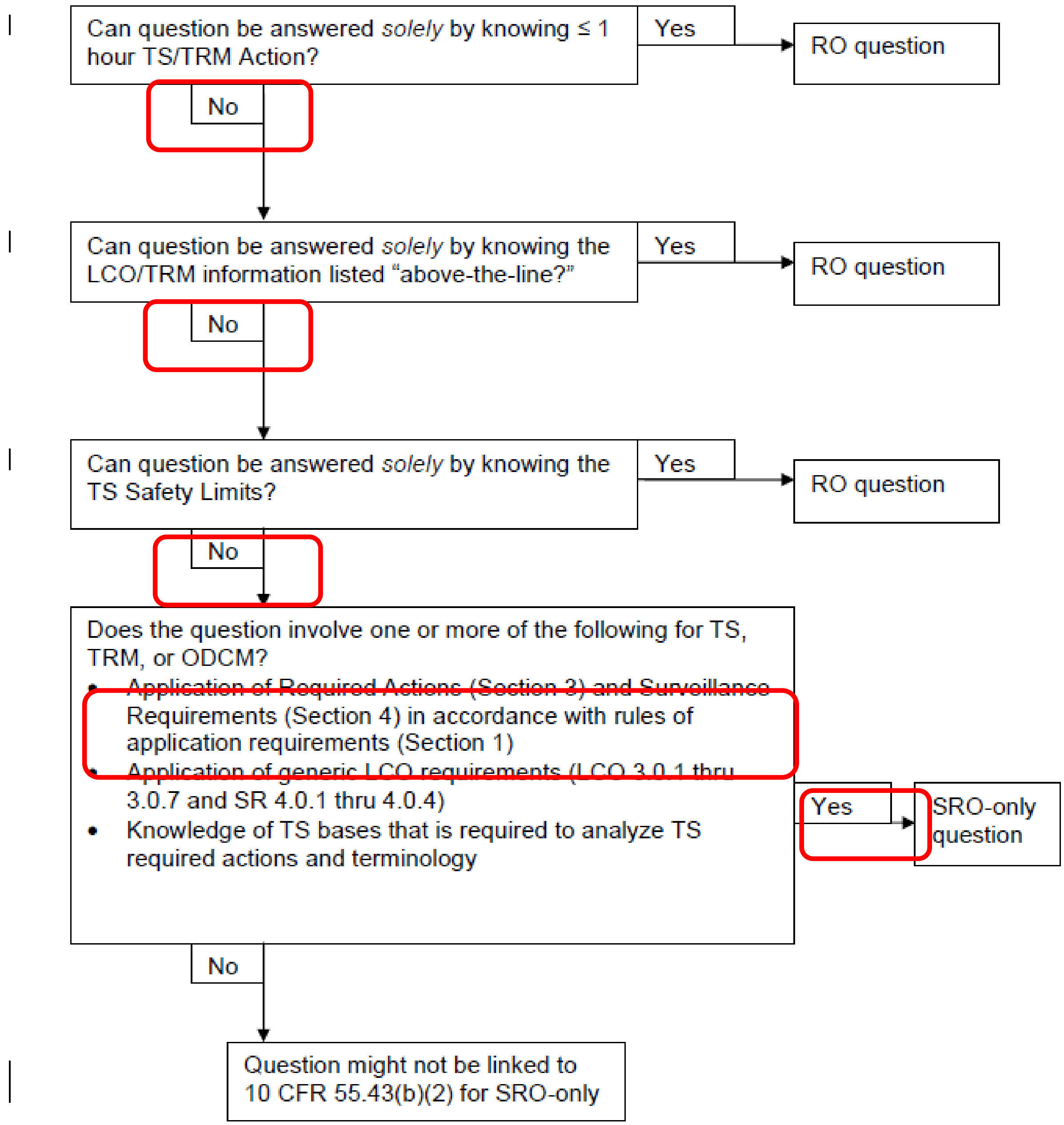
Reference: TS 3.14 Basis.

including replacement of an Emergency Service Water pump without forcing dual unit outages, yet limits the amount of operating time without the specified number of pumps.

When one Unit is in Cold Shutdown and the heat load from the shutdown unit and spent fuel pool drops to less than 25 million BTU/HR, then one Emergency Service Water pump may be removed from service for the subsequent time that the unit remains in Cold Shutdown due to the reduced residual heat removal and hence component cooling requirements.

A minimum level of +17.2 feet in the High Level Intake canal is required to provide design flow of Service Water through the Recirculation Spray heat exchangers during a loss-of-coolant accident for the first 24 hours. If the water level falls below +23' 6", signals are generated to trip both unit's turbines and to close the nonessential Circulating and Service Water valves. A High Level Intake canal level of +23' 6" ensures actuation prior to canal level falling to elevation +23'. The Circulating Water and Service Water isolation valves which are required to close to conserve Intake Canal inventory are periodically verified to limit total leakage flow out of the Intake Canal. In addition, passive vacuum breakers are installed on the Circulating Water pump discharge lines to assure that a reverse siphon is not continued for canal levels less than +23 feet when Circulating Water pumps are de-energized. The remaining six feet of canal level is provided coincident with ESW pump operation as the required source of Service Water for heat loads following the Design Basis Accident.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



K/A Number: 017A2.02, In-Core Temperature Monitoring System (ITM), Ability to (a) predict the impacts of the following malfunctions or operations on the ITM System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Core damage.

Level: SRO	Tier #: 2	Group #: 2	IR – RO: 3.6	IR-SRO: 4.1

Proposed Question: 87

Initial Conditions:

- A LOCA on Unit 1 coincident with a loss of offsite power has occurred
- The crew enters 1-ECA-0.0, Loss of All AC Power.

Current Conditions (20 minutes):

- Crew is holding at ECA-0.0 step 33 Check if 4160V AC Emergency Power is restored.
- Substation personnel estimate re-energizing Bus 5 and Bus 6 within 15 minutes.
- RCS pressure is 1800 psig.
- Core Exit Thermocouples are 1205°F and rising.
- RVLIS full range level is 29% and rising slowly.

Which of the following procedures is required to be entered?

A. 1-FSG-1, Long Term RCS Inventory Control.

B. 1-SACRG-1, Severe Accident Control Room Guideline Initial Response.

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

D. 1-SACRG-2, Severe Accident Control Room Guideline for Transients After TSC is Functional.

Proposed Answer: B. 1-SACRG-1, Severe Accident Control Room Guideline Initial Response.

Explanation: With CETCs >1200°F and no power, 1-SACRG-1 is required. Step 18 of 1-ECA-0.0 states that an ELAP is declared if power cannot be restored within 45 minutes. The question stem estimates power restoration before 45 minutes. ECA-0.0 requires that an ELAP be declared and other conditions met in order to perform 1-FSG-1.

Technical Reference: ECA-0.0, Loss of All AC Power, Rev. 40. F-2 – Core Cooling CSFST, rev 1A

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-17 – ECA-0.0; Objective D. Given actual or simulated plant conditions requiring implementation of ECA-0.0, Loss of All AC Power, successfully transition through the procedure, applying step background knowledge as required, to safely place the plant in the required optimal recovery condition.

Question Source: Modified Bank (Diablo Canyon 4-2007 #91). Modified question stem and two choices.

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Comments:

KA Match Analysis: Question matches the KA in that the candidate must utilize In Core Temperature monitoring indications, and determine that core damage will require transitioning to procedure SACRG-1 for mitigation.

Distractor Analysis:

A. Incorrect. An ELAP has not been declared therefore it is not required to implement FSG-1. Plausible because some of the conditions are met such as Przr level, and < 16 hours since Loss of All AC.
B. CORRECT.
C. Incorrect. Plausible because if power was available, then FR-C.1 entry condition would be met and this would be the correct procedure to enter.
D. Incorrect. SACRG-2 is entered following SACRG-1 and when TSC is manned. Plausible it would be expected to enter SACRG-2 once TSC is manned and at this time the TSC should be manned or in the progress of being manned.

Original Question: Diablo Canyon 4-2007 #91

GIVEN:

- *The crew has transitioned from E-0, Reactor Trip or Safety Injection, to E-1, Loss of Reactor or Secondary Coolant.*
- *A loss of offsite power has occurred.*
- *Emergency Diesel Generators are providing power to all vital buses.*
- *RCS pressure is 1800 psig.*
- *Core Exit Thermocouples are 1125°F*
- *RVLIS full range level is 29% and increasing slowly*

Which of the following actions should be taken by the Shift Foreman?

A. Core cooling is adequate, remain in E-1.

B. The core in uncovered and an inadequate core cooling condition exists, entry into FR-C.1, Response to Inadequate Core Cooling is required.

C. A degraded core cooling condition exists, entry into FR-C.2, Response to Degraded Core Cooling is required.

D. A severe accident and core damage has probably occurred, entry into SACRG-1, Severe Accident Control Room Guideline Initial Response is required.

Proposed Answer:

B. The core in uncovered and an inadequate core cooling condition exists, entry into FR-C.1, Response to Inadequate Core Cooling is required.

Reference: 1-ECA-0.0 showing correct transition out of ECA-0.0 (Correct Answer)

30. ___	CHECK CETCs - LESS THAN 1200°F	☐ IF CETC temperature rising, THEN GO TO 1-SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE.

References: ECA-0.0 step 18 (Distractor A). Time too early to declare ELAP

*18. ___	CHECK IF AC POWER CAN BE RESTORED TO AT LEAST ONE EMERGENCY BUS WITHIN 45 MINUTES OF INITIAL LOSS OF POWER	Do the following: ☐ a) Declare ELAP. ☐ b) Initiate 1-FSG-4, ELAP DC BUS LOAD SHED/MANAGEMENT. ☐ c) Initiate 0-FSG-5, INITIAL ASSESSMENT AND FLEX EQUIPMENT STAGING.

References: ECA-0.0 step 33 (Distractor A)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
33.	CHECK IF 4160V AC EMERGENCY POWER IS RESTORED: (Continued)	
		3) IF ELAP is in progress, THEN implement the following strategies as necessary:
		☐ RCS INVENTORY CONTROL IF pressurizer level is less than 35% [63%] AND time and personnel are available OR RVLIS is less than 78% AND RCS pressure is less than 400 psig, THEN perform 1-FSG-1, LONG TERM RCS INVENTORY CONTROL.

Reference, 1-SACRG-2 (Distractor D)

Dominion

DOMINION POWER
SURRY POWER STATION
SEVERE ACCIDENT CONTROL ROOM GUIDELINE

NUMBER	PROCEDURE TITLE	REVISION
1-SACRG-2	SEVERE ACCIDENT CONTROL ROOM GUIDELINE FOR TRANSIENTS AFTER TSC IS FUNCTIONAL (WITH 5 ATTACHMENTS)	2
		PAGE 1 of 6

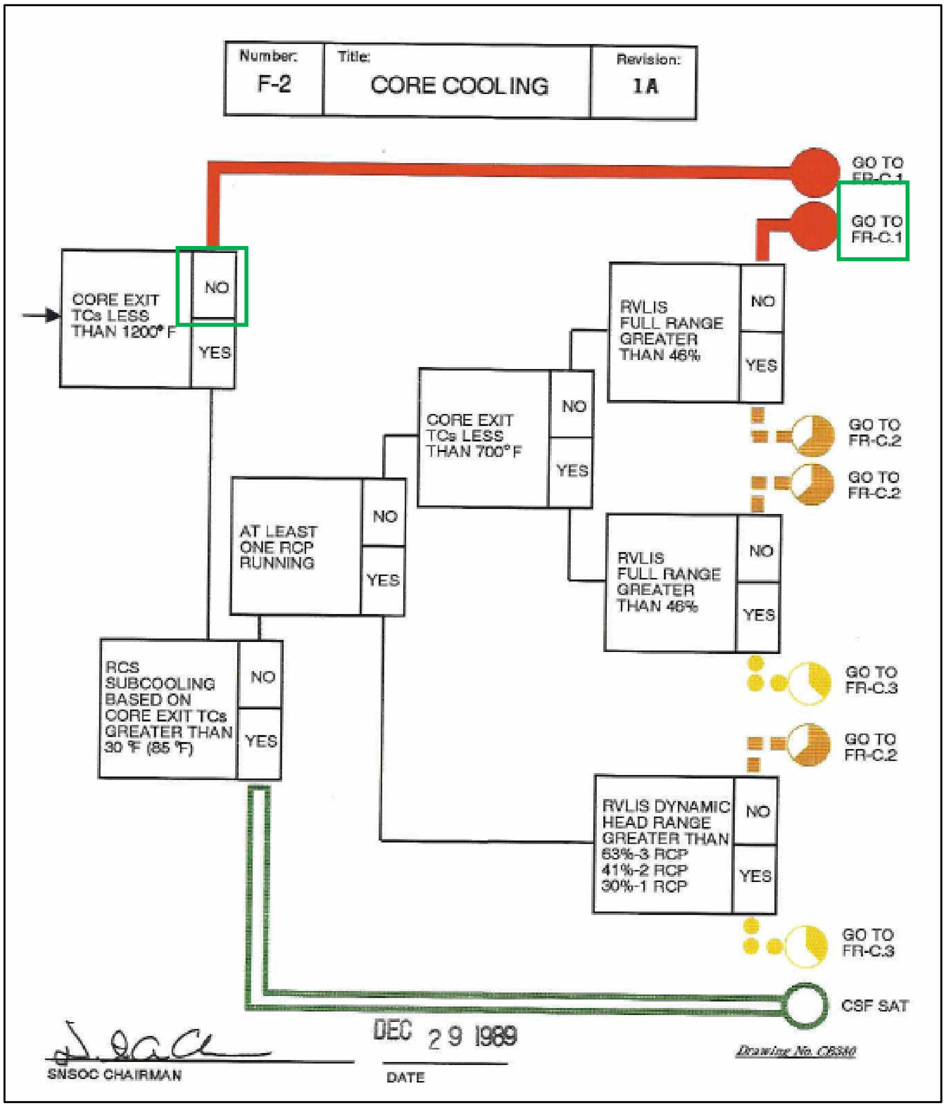
PURPOSE

This guideline provides actions to respond to a severe accident in which the core might be damaged. It is used when the TSC is functional and is monitoring the SAMGs.

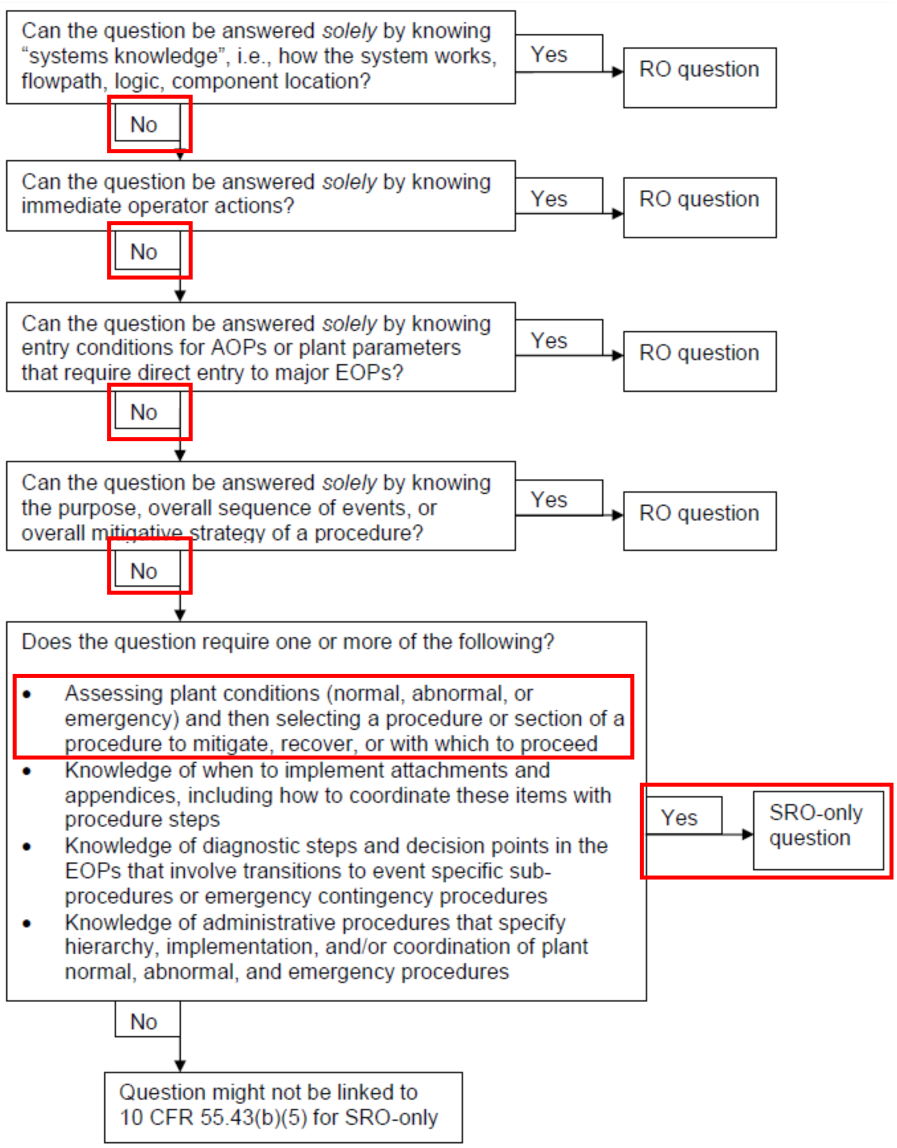
ENTRY CONDITIONS

This guideline is entered from 1-SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE, when the TSC is functional and is monitoring the SAMGs.

Reference: F-2, Core Cooling (Distractor C)



SRO Justification:



K/A Number: 068AA2.08, Control Room Evac. / 8, Ability to determine and interpret the following as they apply to the Control Room Evacuation: S/G pressure.

Level: SRO Tier #: 1 Group #: 2 IR – RO: 3.9 IR-SRO: 4.1

Proposed Question: 88

Given the following:

- Chemicals are being off-loaded in Unit 1 Alleyway.
- A leak in the lines causes Toxic fumes to enter the Main Control Room.
- Two ROs pass out and require offsite medical treatment.
- The SM has directed evacuation of MCR.
- The SM has also contacted outside agencies including the Sheriff to expedite an ambulance be sent to the station.

Which of the following describes:

1) Per VPAP-2802, Notifications and Reports, what is the most limiting report?
2) When control has been transferred to the Aux Shutdown panel, SG pressure is monitored at the __(2)__.

(REFERENCE PROVIDED)

A. 1) 4 hour
   2) Remote Monitoring panel

B. 1) Immediate
   2) Aux Shutdown panel

C. 1) 4 hour
   2) Aux Shutdown panel

D. 1) Immediate
   2) Remote Monitoring panel

Proposed Answer: D

Explanation: 1) Per VPAP-2802, 6.3.2.h, an immediate notification is required If an incident occurs during transport (including loading, unloading and temporary storage of hazardous materials in which a person is killed or requires hospitalization. In this case during offload of chemicals a leak caused two Licensed operators to require hospitalization. 2) Per 0-FCA-1.00 the SG PORVs are operated locally using the air bottles to control steam flow. Many components are operated from the Aux Shutdown Panel that relate to SG control such as the Aux Feed Pumps, but the SG PORVs are operated locally.

Technical Reference: 0-FCA-1.00, LIMITING MCR FIRE, Rev. 49.

Reference Provided to Applicant: NO

Learning Objective: ND-95.6-LP-3, Fire Area FCAs, Objective B; Examine the operations involved in achieving plant control at the auxiliary shutdown panel to stabilize the unit at HSD and CSD conditions following implementation of FCA-1.00, Limiting Main Control Room Fire.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: SROU-02, Objective B **For SRO Only**

Comments:

K/A Analysis: K/A requires knowledge pertaining to interpretation of SG pressures during CR Evacuation. Question poses a Fire in MCR requiring evacuation, and a Faulted SG that the candidate must interpret. This meets the K/A in that the SRO candidate must interpret the SG pressure indications given and determine actions necessary per 0-FCA-1.00.

Distractor Analysis:

A. Incorrect. 1) Incorrect. Two ROs were injured during offloading, therefore an immediate notification is required. Plausible because contacting outside agencies is a 4-hour report, but in this case an immediate report is more limiting. 2) Correct.
B. Incorrect. 1) Correct. 2) Incorrect as 0-FCA-11.0 directs an operator to go to the Cable Spreading room, and set up the Remote Monitoring panel for operation. Plausible because the Aux Shutdown Panel is the first location personnel are sent to.
C. Incorrect. 1) Incorrect. Two ROs were injured during offloading, therefore an immediate notification is required. Plausible because contacting outside agencies is a 4-hour report, but in this case an immediate report is more limiting. 2) Incorrect as 0-FCA-11.0 directs an operator to go to the Cable Spreading room, and set up the Remote Monitoring panel for operation. Plausible because the Aux Shutdown Panel is the first location personnel are sent to.
D. Correct.

Reference VPAP-2802, 6.3.2.h (immediate report). VPAP-2802, 6.3.4.a.5 (distractor-4hour report)

DOMINION

VPAP-2802
REVISION 40
PAGE 80 OF 232

h. If an incident occurs during transport (including loading, unloading, and temporary storage) of:

- Radioactive materials in which fire, breakage, spillage, or suspected radioactive contamination occurs (see also Step 6.29.3) [49 CFR 171.15(b)(2)]
- Hazardous materials in which any of the following is a direct result of the hazardous materials: [49 CFR 171.15(b)(1)]
  - A person is killed
  - A person requires hospitalization because of injuries
  - An evacuation of the general public occurs lasting one or more hours
  - One or more major transportation arteries or facilities are closed or shut down for one hour or more
  - The operational flight pattern or routine of an aircraft is altered
- A situation exists (e.g., a continuing danger to life exists at the scene of the incident) that, in the judgment of the carrier or Dominion, should be reported even though it does not meet one of the previous criteria [49 CFR 171.15(b)(5)]

NOTE: "Notification to other government agencies has been or will be made" is not necessarily an automatic notification to the NRC. Refer to NUREG – 1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, for discussions and examples (e.g., newsworthy events, environmental events, spurious, emergency siren actuations) or contact Station Licensing if clarification is needed. [NUREG-1022, Section 3.2.12]

5. Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned, or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials. **[Commitment 3.2.12]** [10 CFR 50.72(b)(2)(xi)]

Reference 0-FCA-1.00, Step 15. Shows direction to initiate remote monitoring per 0-FCA-11.0.

NUMBER	PROCEDURE TITLE	REVISION
0-FCA-1.00	LIMITING MCR FIRE	49
		PAGE 6 of 33

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

12. ___ GET THE FOLLOWING ITEMS:
    - ☐ • Appendix R Key Box
    - ☐ • FCA Procedures
    - ☐ • CRO Shift Relief Checklists
    - ☐ • FR Procedures

13. ___ EVACUATE MCR AND GO TO ESGRs

14. ___ SEND THE FOLLOWING PERSONNEL TO THE TSC:
    - ☐ • One SRO
    - ☐ • Emergency Communicators
    - ☐ • STA (With Shift Manager discretion)

15. ___ SEND PERSONNEL TO INITIATE THE FOLLOWING:
    - ☐ a) 0-FCA-12.00, EMERGENCY DIESEL GENERATOR OPERATION
    - ☐ b) 0-FCA-14.00, ESTABLISHING STABLE RCS MAKEUP FLOWPATHS
    - ☐ c) 0-FCA-11.00, REMOTE MONITORING

Reference 0-FCA-11.00, Step 6, and Attachment 4. Show steps and location for SG pressure monitoring.

NUMBER	PROCEDURE TITLE	REVISION
0-FCA-11.00	REMOTE MONITORING	6
		PAGE 2 of 3

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK CABLE SPREADING ROOM VENTILATION - RUNNING ☐ • 1-VS-RAF-1, Unit 1 ☐ • 1-VS-RAF-2, Unit 2	☐ Start fans as necessary.

NOTE:
- Remote instrumentation power supplies are referenced in Attachment 3.
- The Sound Powered Phone circuit may be used for communications if the communication link is intact.
- Portable radios may be used while communicating with the MCR from PNL REM and ASC RMP Panel.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.___	TAKE A PORTABLE RADIO AND GO TO CABLE SPREADING ROOM	
3.___	LOCATE PNL REM AND ASC RMP PANELS AND ESTABLISH COMMUNICATIONS WITH ROs AND SROs	

ATTACHMENT 4

NUMBER 0-FCA-11.00	ATTACHMENT TITLE REMOTE MONITORING LOG SHEET	ATTACHMENT 5
REVISION 6		PAGE 1 of 2

UNIT 1

NOTE: Enter time readings taken across top row. Readings should be taken at approximately 30 minute intervals.

TIME

PARAMETER										
SG A WR LVL										
SG B WR LVL										
SG C WR LVL										
NEUTRON FLUX Source Range										
NEUTRON FLUX Power Range										
RCS COLD LEG Loop A										
RCS COLD LEG Loop B										
RCS HOT LEG Loop A										
RCS HOT LEG Loop B										
RCS WR Press										
PRZR LVL										
SG A PRESS										
SG B PRESS										

K/A Number: 076G2.1.32, Service Water, Ability to explain and apply all system limits and precautions.

Level: SRO Tier #: 2 Group #: 1 IR – RO: 3.8 IR-SRO: 4.0

Proposed Question: 89

Given the following:

- Unit 1 is at 100%, Unit 2 is at 300°F, 400 psig, and is cooling down for a Refueling outage.
- Annunciator 0-VSP-M6, ESW PP HSE LO TEMP is received.
- An Operator is dispatched and reports that ESW pump house temperature is 38°F, the heating system has failed, and no ESW pumps ae running.

With the present conditions which of the following completes the below statements:

1) The ESW pump starting batteries are __(1)___.
2) The minimum number of ESW pumps that must be operable per Tech Specs are __(2)__ ESW pumps.

A. 1) inoperable
   2) three

B. 1) operable
   2) three

C. 1) inoperable
   2) two

D. 1) operable
   2) two

Proposed Answer: A.

Explanation: 1) OP-AA-102, Attachment 1, Immediate Operability Guidelines, states that if an SSC is unable to perform its required functions under all environmental conditions then the equipment is inoperable. IAW ARP 0-VSP-M6, The starting batteries for the ESW pumps are Inoperable if house temperature is less than 45 °F. 2) Per Tech Specs 3.14, three emergency service water pumps are required to be operable when > 350°F/450 psig. TS 3.14.B does allow two ESW pumps if one unit is in CSD with unit decay heat loads < 25 million BTU/hr. In this case Unit 2 is NOT in CSD therefore per TS 3.14.B three ESW pumps are required to be operable.

Technical Reference: ARP 0-VSP-M6, Rev.4, Caution 1 and 2 prior to Step 1. OP-AA-102, Operability Determination, Attachment 1. Technical Specifications Rev. 48

Reference Provided to Applicant: No.

Learning Objective:

- ND-89.5-LP-2, Service Water, Objective G, Evaluate the technical specifications associated with the Service Water Systems, including for SRO candidates, the basis behind these specifications.
- SROU-02, Admin Procedures, Objective B; For a Tier 2 procedure, discuss the following; Purpose of the procedure, Requirements for Operations personnel. OP-AA-102, Operability Determination (SRO- emphasis on immediate determination)

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: CFR55.43.(b)(2): Tech Specs.

SRO ONLY Objective: SROU-02; Purpose and requirements of Operability Determination, (SRO-only emphasis on immediate determination.

Comments:

K/A match analysis: Question matches K/A, Candidate must apply limitations of diesel driver and starting batteries to determine Operability of ESW pumps with Low temperature in the ESW pump house.

Distractor Analysis:

A. Correct. Both Pars 1) and 2) are correct.

B. Incorrect. Part 1) is incorrect, battery is Inoperable below 45°F. Plausible fail to identify parameter as below minimum for operability. Part 2) is correct.

C. Incorrect. Part 1) is correct. Part 2) is incorrect, three ESW pumps are required to be operable. Plausible if candidate confuses specific mode and decay heat requirements for two ESW pumps; unit needs to be in Cold shutdown and decay heat rate needs to be < 25 million BTU/hr.

D. Incorrect. Part 1) is incorrect, battery is Inoperable below 45°F. Plausible fail to identify parameter as below minimum for operability. Part 2) is incorrect, three ESW pumps are required to be operable. Plausible if candidate confuses specific mode and decay heat requirements for two ESW pumps; unit needs to be in Cold shutdown and decay heat rate needs to be < 25 million BTU/hr.

Reference: OP-AA-102, Attachment 1, Immediate Operability Determination Guidelines.

ATTACHMENT 1

(Page 1 of 5)

Immediate Operability Determination Guidelines

1. **Relationship of Operability and Qualification**

   Operability and qualification are closely related concepts. However, the fact that a system is NOT fully qualified does NOT, in all cases, render that system unable to perform its specified function if called upon. As per the definition of operability, a safety or safety support system or structure must be capable of performing its specified function(s) of prevention or mitigation as described in the current licensing basis, particularly the TS bases or FSAR.

   The purpose of an operability determination is to provide a basis for making a timely decision on plant operation when a degraded or nonconforming condition is discovered. Corrective actions taken to restore full qualification should be addressed through the corrective action process. The treatment of operability as a separate issue from the restoration of full qualification emphasizes that the operability determination process is focused on safe plant operation and should NOT be impacted by decisions or actions necessary to plan and implement corrective action (i.e., restore full qualification). *(IMC 06.03)*

   When a degraded or nonconforming condition exists, there is a lack of full qualification. Qualification concerns, whether it is a lack of required quality or loss of quality because of degradation, can and should be promptly considered to determine the effect on the operability of the system. *(IMC 03.40)*

2. **Conditions which render equipment inoperable**

   a. SSC is unable to perform its specified safety function due to gross failure, damage, or malfunction or due to being removed from service (tagged out).

   b. SSC fails to start upon receipt of a valid safety signal.

   c. SSC fails to meet the prescribed requirements of TS or of surveillance testing verifying compliance with TS.

8. Electrical / I&C Operability Issues

a. Setpoint and calibration tolerance

- IF an equipment setpoint or calibration is determined to exceed that required by TS, THEN the equipment is inoperable.
- IF an equipment loop is determined (by test and/or calculation) to be unable to perform its intended specified safety function within its required TS limits, THEN the loop is inoperable.

b. Equipment with automatic or manual actuation capability

IF the automatic or manual actuation capability is required (by TS, TRM, UFSAR, and EOPs) to fulfill a specified function / specified safety function and it is lost, THEN the equipment is inoperable.

c. Environmental Qualification (EQ)

- IF equipment is installed and maintained in accordance with the Environmental Qualification Program, THEN from an EQ standpoint, the equipment is operable (i.e., it is environmentally qualified or has "environmental qualification").
- IF, for equipment that is required to be environmentally qualified, a condition exists that obviously would NOT allow performance of a specified safety function under all postulated service conditions, THEN the equipment is inoperable.
- IF, for equipment that is required to be environmentally qualified, a condition exists that may compromise its environmental qualification, but it is NOT obvious whether its specified safety function would be performed under all postulated service conditions, THEN the condition may require an OD evaluation.

Reference: ARP 0-VSP-M6

NUMBER	PROCEDURE TITLE	REVISION
0-VSP-M6	ESW PP HSE LO TEMP	4
		PAGE 3 of 4

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- The starting batteries for the ESW Pumps are considered inoperable if pump house temperature drops to 45°F.
- If pump house temperature drops to 45°F and the diesels are not running, the diesel drivers for the ESW Pumps are considered inoperable.
- If the diesels or their starting batteries become inoperable, an LCO clock will be entered IAW Tech Spec 3.14.B.

* *

References Technical Specifications 3.14.A (two unit operation), and 3.14.B (Allowed operability).

3.14 CIRCULATING AND SERVICE WATER SYSTEMS

Applicability

Applies to the operational status of the Circulating and Service Water Systems.

Objective

To define those limiting conditions of the Circulating and Service Water Systems necessary to assure safe station operation.

Specification

A. The Reactor Coolant System temperature or pressure of a reactor unit shall not exceed 350° F or 450 psig, respectively, or the reactor shall not be critical unless:

1. The high level intake canal is filled to at least elevation +23.0 feet at the high level intake structure.

2. Unit subsystems, including piping and valves, shall be operable to the extent of being able to establish the following:

   a. Flow to and from one bearing cooling water heat exchanger.

   b. Flow to and from the component cooling heat exchangers required by Specification 3.13.(*)

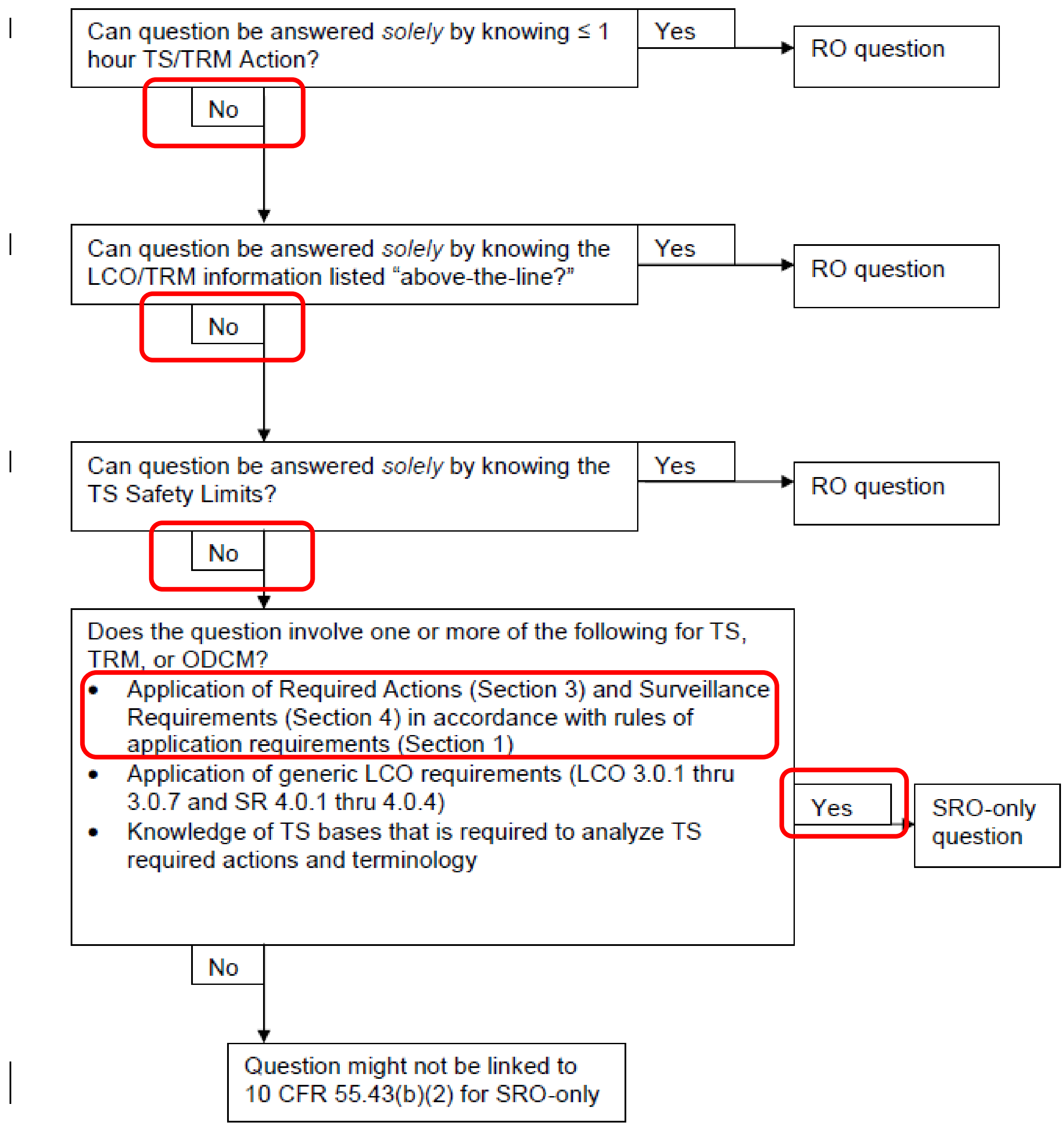
3. At least two circulating water pumps are operating or are operable.

4. Three emergency service water pumps are operable; these pumps will service both units simultaneously.

B. The requirements of Specification 3.14.A.4 may be modified to allow one Emergency Service Water pump to remain inoperable for a period not to exceed 7 days. If this pump is not OPERABLE in 7 days, then place both units in HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the next 30 hours.

The requirements of 3.14.A.4 may be modified to have two Emergency Service Water pumps OPERABLE with one unit in COLD SHUTDOWN with combined Spent Fuel pit and shutdown unit decay heat loads of 25 million BTU/HR or less. One of the two remaining pumps may be inoperable for a period not to exceed 7 days. If this pump is not OPERABLE in 7 days, then place the operating unit in HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the next 30 hours.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



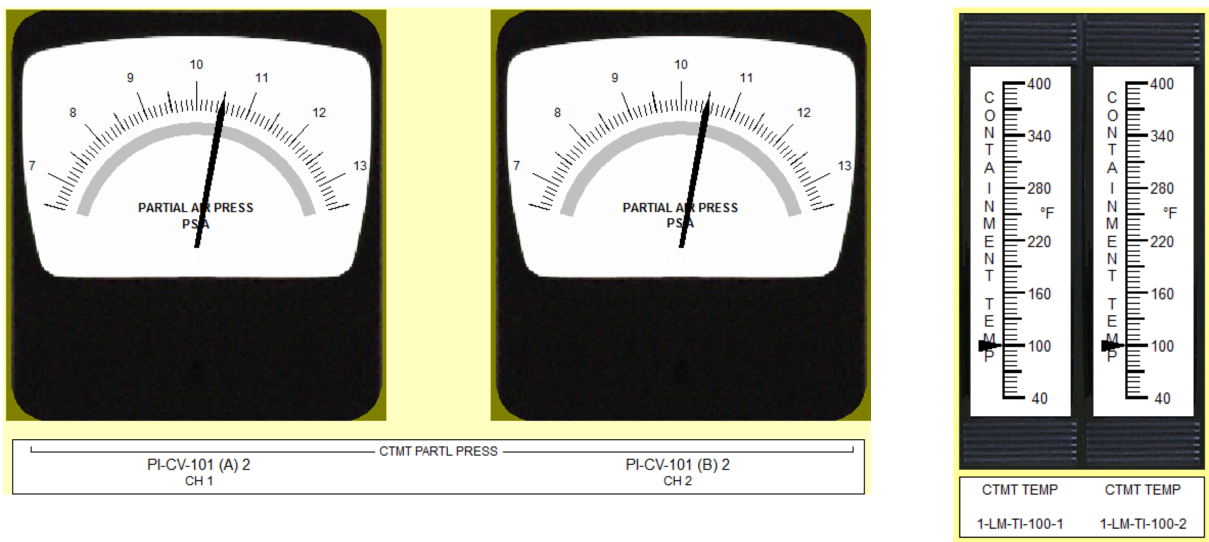
K/A Number: 103G2.2.44, Containment, Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Level: SRO Tier #: 2 Group #: 1 IR – RO: 4.2 IR-SRO: 4.4

Proposed Question: 90

The following conditions exist on Unit 1:

- Circulating/Service Water temperature is 68°F.
- Containment conditions are as indicated below:



Given the above conditions:

- The mode change to 350°F/450 psig __(1)__ (can/cannot) be made.
- The basis for the maximum Containment air partial pressure is __(2)__.

REFERENCE PROVIDED

A. (1) cannot; (2) MSLB peak calculated pressure criteria

B. (1) can; (2) MSLB peak calculated pressure criteria

C. (1) cannot; (2) LOCA depressurization criteria

D. (1) can; (2) LOCA depressurization criteria

Proposed Answer: B. (1) can; (2) MSLB peak calculated pressure criteria

Explanation: In accordance with 1-GOP-1.7, Unit Startup, RCS Heatup from Ambient to HSD Containment temperature must be above 75°F before exceeding 350°F/450 psig. Technical Specifications (TS-3.8) basis section addresses the bases behind the shape of the allowed partial pressure curve (Fig. 3.8-1).

Technical References: 1-GOP-1.7 Unit Startup, RCS Heatup from Ambient to HSD, rev 30;

1-DRP-003, Attachment 4, rev 124; TS 3.8.D and Basis

Reference Provided to Applicant: YES (Tech Spec Figure 3.8-1)

Learning Objective: ND-88.4-LP-2, CTMT Vessel; Objective E. Employ the technical specifications associated with the containment structure, including for the SRO candidates, the basis behind these specifications.

Question Source: Modified (Surry 2015 NRC Exam - #81)

Question History: Last NRC 2 Exams: YES (Surry 2015 NRC Exam - #81)

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.12)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must interpret given MCR indications and understand how operator directives affect the plant and system.

Distractor Analysis:

A. Incorrect, but plausible if the candidate does not understand that the mode change can be made since containment temperature is >75°F.

B. CORRECT.

C. Incorrect, but plausible if the candidate does not understand that the mode change can be made since containment temperature is >75°F and confuses the basis behind the shape of the curve.

D. Incorrect, but plausible if the candidate confuses the basis behind the shape of the curve.

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Revision 30
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______ 5.5.19 Check that Release Permit to support use of a SG PORV has been received from HP. Enter N/A if NOT required.

______ 5.5.20 Check compliance with TS 3.8 for CTMT temperature and pressure IAW 1-DRP-003, Attachment 4 or 5.

NOTE: RCS heatup will begin when RHR is removed from service. At this point, RCS heatup rate is controlled by the combination of Feedwater flow to the SGs and SG Blowdown flow, or steam release. Raising SG Blowdown flow by itself will not reduce or stop an RCS heatup, but will only reduce SG level. Minimizing SG Blowdown flow and Feedwater flow will maximize RCS heatup. Raising SG Blowdown flow and Feedwater flow will minimize RCS heatup. (**Reference 2.4.34**)

5.5.21 Remove the RHR System from service by performing the following.

______ a. Operators have been briefed and are taking actions to maintain containment temperature greater than 75°F (T.S. Figure 3.8-1) while securing RHR.

______ b. Remove the RHR System from service IAW 1-OP-RH-001, RHR Operations.

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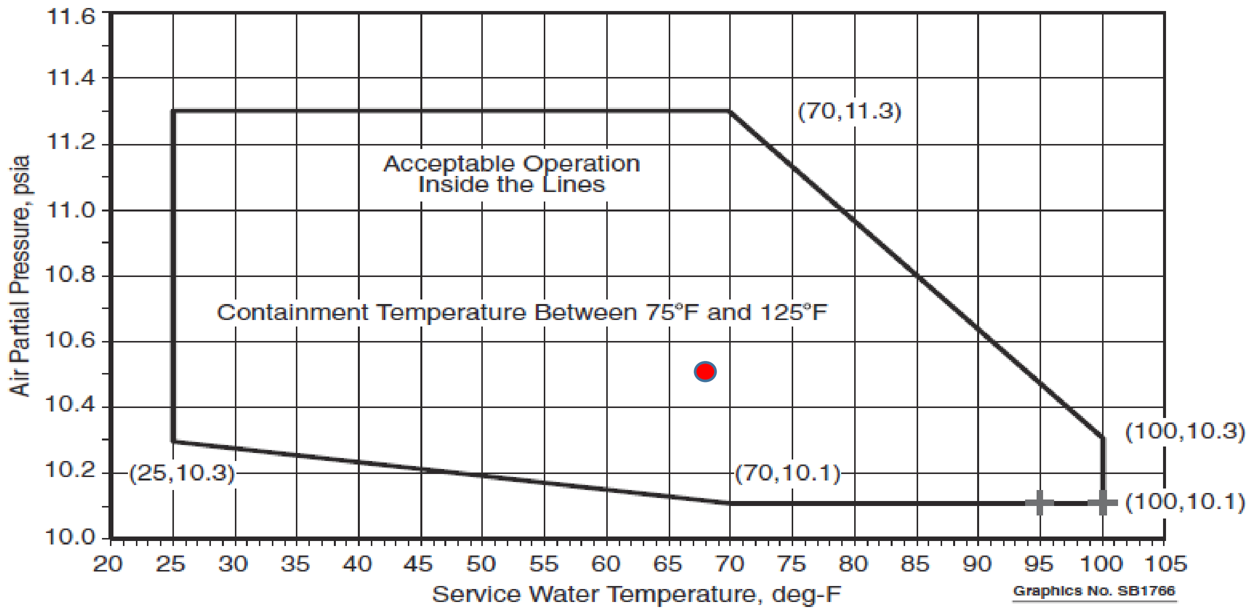
(Page 1 of 1)

Attachment 4

CONTAINMENT ALLOWABLE AIR PARTIAL PRESSURE INDICATION VS. SERVICE WATER TEMPERATURE



Note: Operation On or Outside the Line Requires Entry into TS 3.8.D.1.a

TS 3.8-3
11-08-04

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Otherwise, place the unit in HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

D. Internal Pressure

1. Containment air partial pressure shall be maintained within the acceptable operation range as identified in Figure 3.8-1 whenever the Reactor Coolant System temperature and pressure exceed 350°F and 450 psig, respectively.

a. With the containment air partial pressure outside the acceptable operation range, restore the air partial pressure to within acceptable limits within 1 hour or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TS 3.8-4
06-17-08

(3) assuring that environmental conditions will not preclude access to close the valves and 4) that this administrative or manual action will prevent the release of radioactivity outside the containment.

The Reactor Coolant System temperature and pressure being below 350°F and 450 psig, respectively, ensures that no significant amount of flashing steam will be formed and hence that there would be no significant pressure buildup in the containment if there is a loss-of-coolant accident. Therefore, the containment internal pressure is not required to be subatmospheric prior to exceeding 350°F and 450 psig.

The allowable value for the containment air partial pressure is presented in TS Figure 3.8-1 for service water temperatures from 25 to 100°F. The RWST water shall have a maximum temperature of 45°F.

The horizontal upper limit line in TS Figure 3.8-1 is based on MSLB peak calculated pressure criteria, and the sloped line from 70°F to 100°F service water temperatures is based on LOCA depressurization criteria.

<u>Parent Question (Surry 2015 #81)</u>

<u>Initial Conditions:</u>

- *Unit 1 and Unit 2 operating at 100% power.*
- *Unit 2 is performing 2-PT-41.1, CC Pump Performance.*
- *CC is **<u>split out</u>** in the Turbine Building.*
- *1-CC-E-1A, "A" CC HX, has been isolated due to a through wall SW leak.*

<u>Current Conditions:</u>

- *Annunciators 1B-A7 and 1B-B7, Channel 1 and Channel 2, CTMT PART +.1 PSI are received.*
- *CTMT pressure is 10.6 PSIA and rising slowly.*
- *The operating Team is performing ARP 1B-A7, and has raised SW flow to the "C" CC Heat Exchanger.*

Which ONE of the following describes:

1) The effect on Unit 1 Containment Temperature.
2) The basis of the sloped line from 70 °F to 100 °F on the Containment Allowable Air Partial Pressure VS. Service Water Temperature Curve (Figure TS-3.8-1) is __________.

(REFERENCE PROVIDED)

A. 1) No change.
2) MSLB Peak Calculated Pressure

B. 1) Lower.
2) LOCA depressurization

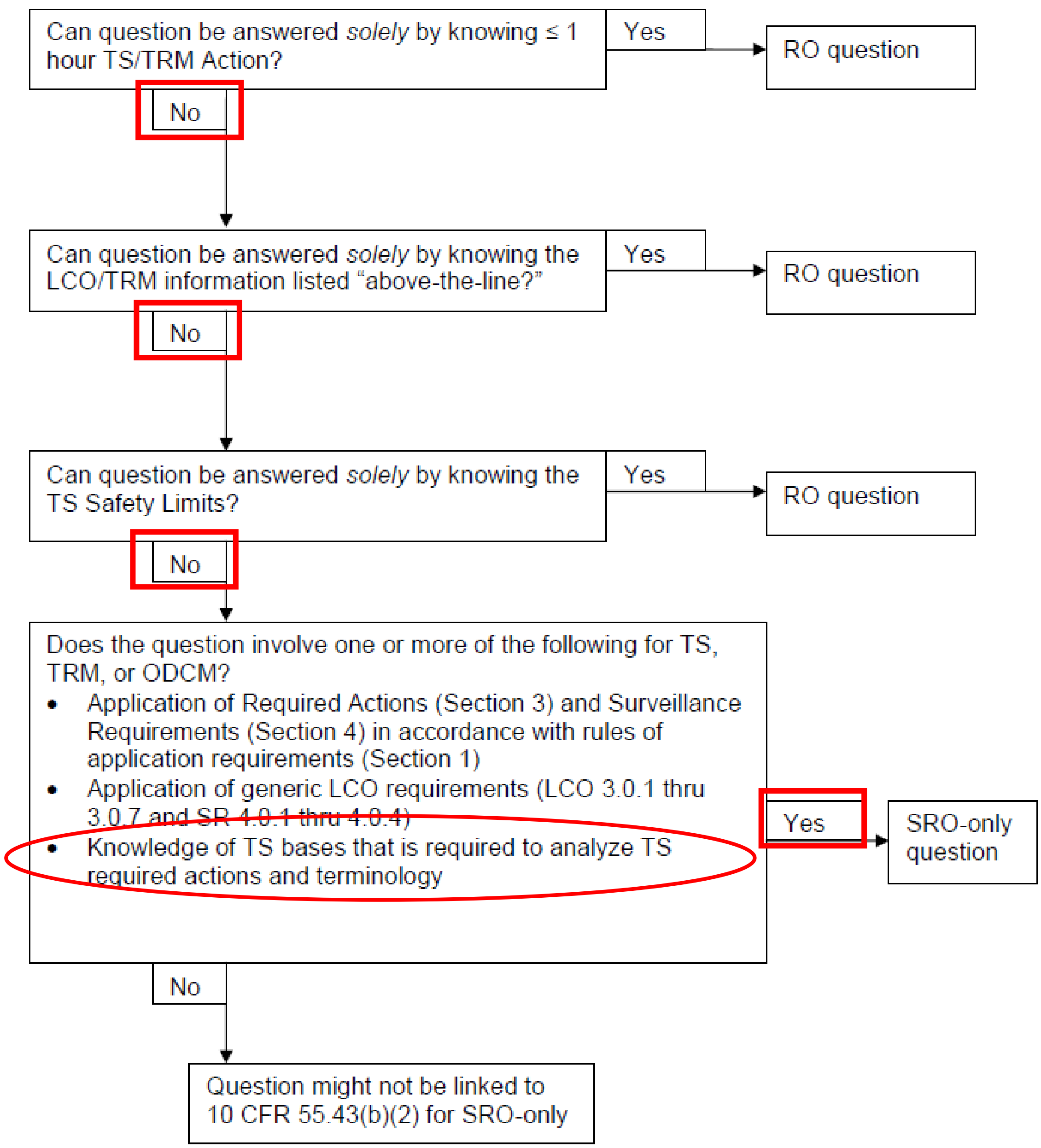
C. 1) No change.
2) LOCA depressurization

D. 1) Lower.
2) MSLB Peak Calculated Pressure

Proposed Answer: C

SRO Justification: Requires knowledge of Tech Spec Bases to answer question correctly.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



NRC APPROVED. K/A Number: G2.1.41, Conduct of operations, Knowledge of the refueling processes.

Level: SRO Tier #: 3 Group #: IR – RO: 2.8 IR-SRO: 3.7

Proposed Question: 91

In accordance with 1-OSP-ZZ-004, which one of the following will require Fuel movement to cease:

A. RCS Cavity level equal to 23.5 feet.

B. Process Vent Gaseous RM in ALERT.

C. Equipment Hatch is opened.

D. Source Range Detector N-31 Failed Low.

Proposed Answer: D

Explanation: Per 1-OSP-ZZ-004, both Source Range Detectors must be Operable.

Technical Reference: 1-OSP-ZZ-004, Unit 1 Safety Systems Status For Refueling, Revision 45.

Reference Provided to Applicant: No

Learning Objective: ND-92.5-LP-1, Refueling Overview; Objective D, Unit 1 Safety Systems Status For Refueling.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: CFR: 55.43(b)(7) Fuel Handling facilities and procedures.

Comments:

K/A Analysis: Question evaluates knowledge of the requirements for refueling by testing the applicant’s knowledge of the specific checks the SRO makes each shift to verify refueling requirements are met.

Distractor Analysis:

A. RCS Cavity level at 23.5 feet.

B. Process Vent Gaseous RM in ALERT.

C. Equipment Hatch is opened.

D. Source Range Detector N-31 Failed Low.

A. Incorrect. The Attachment 8 requirement is that Cavity level is > 23 feet. Normally the cavity is filled to between 26 ft to 27 ft. and PCS low level alarm calibrated to 26 feet. Plausible because value is close to minimum required value and at this level both the PCS alarm and the Spent Fuel Pit low level alarm would be in alarm.

B. Incorrect. Incorrect because Attachment 8 specifies the Vent-Vent Gaseous and Particulate are operable which is a different effluent exhaust point then Process Vent. Plausible because Process Vent Gaseous would indicate a problem with gaseous discharge from a different location.

C. Incorrect. 1) Incorrect. 1-OSP-ZZ-004 specifies that Containment integrity is met per Tech Specs and 1-OP-FH-001. The Equipment hatch can be opened as long as the Equipment hatch is capable of being closed. Plausible because the candidate can misconstrue that with the equipment hatch opened containment integrity is not met.

D. Correct.

References: 1-OSP-ZZ-004, Attachment 8 showing correct answer (red) and distractors (green)

(Page 1 of 2)

Attachment 8
REFUELING OPERATIONS REQUIREMENTS

EQUIPMENT	MIN REQ	D	N	TECH SPECS	REMARKS
Refueling Containment Integrity set	As Required			3.10.A.1	IAW 1-OP-FH-001
Radiation Monitors: • Manipulator Crane • Containment Gaseous • Containment Particulate • SFP Bridge • Vent-Vent Gaseous • Vent-Vent Particulate	 1 operable 1 operable 1 operable 1 operable 1 operable 1 operable			3.10.A.3 3.10.B.1	If the Containment Air Recirculation fans are not running then refer to Tech Spec 3.10 for actions.
Source Range Detectors (audible indication in CTMT must be checked operable)	2 operable			3.10.A.2	
Cavity level > 23 feet	23 feet			3.10.A.6	OU-SU-201, should be maintained as high as possible. No fuel movement permitted if < 23 feet in Cavity
RHR pump and Heat Exchanger: • Cavity Level > 23 feet • Cavity Level < 23 feet	 1 operable 2 operable			 3.10.A.4 3.10.A.5	
Direct communication between the Control Room and Manipulator Crane	Yes			3.10.A.8	When changing core geometry
RCS Boron concentration	≥ 2350 PPM (Admin limit)			CY-AP-PRI-100	RCS must be sampled at least once every 24 hours if the head is unbolted (Not required if defueled and cavity is drained below flange level. (Ref 2.3.15)
RHR Temperature	≤140 °F			1.0.C.1	
Reactor shutdown greater than 100 hours	100 hours			3.10.A.9	For movement of irradiated fuel

References: 1-OP-FH-001, Step 5.4.5 Shows requirement for Equipment Hatch.

5.4.4 Check that 1-OPT-CT-210, Refueling Containment Integrity, has been completed.

5.4.5 Check one of the following conditions exist for the Equipment Hatch. (✓)

____ Equipment Hatch is held in place by eight bolts and that **no** atmosphere to atmosphere openings exist at the escape hatch temporary blank.

____ Equipment Hatch remains open IAW 1-OP-CT-002, Containment Penetration Breach Log.

References: Tech Specs 3.10.A.1. Shows requirement for Equipment Hatch.

3.10 REFUELING

Applicability

Applies to operating limitations during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building.

Objective

To assure that no accident could occur during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building that would affect public health and safety.

Specification

A. During REFUELING OPERATIONS the following conditions are satisfied:

1. The equipment access hatch and at least one door in the personnel airlock shall be capable of being closed. For those penetrations which provide a direct path from containment atmosphere to the outside atmosphere, the containment isolation valves shall be OPERABLE or the penetration shall be closed by a valve, blind flange, or equivalent or the penetration shall be capable of being closed.

References: 1-OP-FH-001, Step 5.3.2. Shows requirement for PCS Cavity low level alarm when < 26 feet.

5.3.2 Check I & C has started 1-IPM-RC-L-459/460/461, Pressurizer Level Protection Loop L-1-459/460/461 Cold Calibration For Refueling Cavity Level Monitoring, to calibrate the respective PRZR LO LVL annunciator to alarm at less than 26 ft. cavity level.

WARNING

- A significant rise in work area dose rates will be experienced during the lifting and movement of the Reactor Vessel upper internals. Stop work dose rate criteria will be discussed prior to the lift and any Stop Work Order issued by HP shall be followed. **(Ref. 2.4.5) (Ref. 2.4.27)**
- For maximum shielding during movement of the upper internals to the storage stand, the cavity level should be maintained at approximately 26 ft 6 in. **(Ref. 2.4.27)**

ES 401-9 COMMENTS (WITH CORRECTIONS SHOWN IN RED), 3/28/16

The SRO flowchart is for selection and assessment of procedures, but this question does not required procedure selection.

Changed question to test knowledge of refueling requirements to continue refueling. This meets SRO requirements per 10CFR: 55.43(b)(7) Fuel Handling facilities and procedures. Added CFR 55.43(b)(7) to 10CFR block. Removed flowchart.

ES 401-9 COMMENTS, 3/30/16: Everything above “In accordance with” is window dressing.

NRC APPROVED. K/A Number: G2.1.7, Conduct of operations, Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation.

Level: SRO Tier #: 3 Group #: IR – RO: 4.4 IR-SRO: 4.7

Proposed Question: 92

Initial Conditions

Unit 1 is stable at 100% reactor power.

- Control Rod E-5, Shutdown Bank "B" dropped, "0" steps indicated.
- The Team responds per 0-AP-1.00, Rod Control System Malfunction.
- 0-AP-1.00, Attachment 6, Calculation of Excore Quadrant Power Tilt Ratios, has been completed with a result of 9% calculated quadrant power tilt.
- Power is stabilized at 95% of rated power.

Current Conditions

- The control rod is not recovered and quadrant power tilt does not change,

1. After one hour elapses, the maximum allowed Tech Spec reactor power level is ________.
2. After four hours elapse, the maximum allowed value of the high flux power range trip setpoint is _____.

REFERENCE PROVIDED

A. 1) 77%
   2) 89%.

B. 1) 75%
   2) 85%.

C. 1) 77%
   2) 85%.

D. 1) 75%
   2) 89%.

Proposed Answer: B.

Make this open reference. Give the applicants only the Quadrant power tilt and control rod assembly sections of TS. Reference provided for this question would be TS 3.12, Pages 7, 8 and 9. These pages are shown at the end of this document.

Explanation: Tech Spec 3.12.C, inoperable rod, requires reactor power be reduced to <75% within 1 hour, and Power Range Hi Φ Trip setpoint reduced <85% within the following 4 hours. Tech spec

3.12.B.6, flux tilt, requires reactor power be reduced 2% for % of tilt (100% - 18% = 82%) and power range Hi Φ flux trip be similarly reduced (2% for each % of tilt) within the next 4 hours; 107% -18% = 89%. If the Candidate assumes reactor power must be reduced from Current power level of 95%: 95% -18% = 77% required power level. Stipulate in Part 1) that a maximum Tech Spec power level, 0-AP-1.00 requires reactor power reduction to 70%-74% within 1 hour (Administrative power level limit to prevent exceeding TS power level of 75%), Candidate is required to answer question bounded by Tech Spec requirements vice Administrative requirement in 0-AP-1.00.

Technical Reference: TS 3.12.B and 3.12.C.

Reference Provided to Applicant: Yes (Page 3.12 - 7, 8, 9)

Learning Objective: ND-93.2-LP-4, PR NI, Objective E, Explain the steps necessary to perform a quadrant power tilt calculation, including the limits imposed upon a tilt by Technical Specifications.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10 CFR 55.43 (b)(2) Facility Operating limitations in the technical specifications and their basis.

Comments: Based on calculated quadrant power tilt, reactor power must be reduced to a value of 77%. **The dropped rod is considered inoperable**, which TS requires power be reduced <75% within 1 hour. The Hi Φ trip setpoint must be reduced to 89% based on Quadrant power tilt, and 85% based on the misaligned rod.

K/A Match Analysis: Question matches K/A. Candidate must assess provided calculated quadrant power tilt, differentiate between the required power reduction limits imposed for flux tilt and an Inoperable rod, and select the appropriate setting for the PR Hi Φ trip based on the most limiting Tech Spec requirement.

Distractor Analysis:

A. Incorrect. Part 1) is incorrect, 75% is the limiting reactor power for the inoperable rod. Plausible if the Candidate fails to consider the dropped rod as inoperable, and calculates reactor power limit due to flux tilt based on reduction from Current Power vice Rated power. Part 2) is incorrect, inoperable rod Hi Φ limit is 85%. Plausible Candidate fails to consider the dropped rod as inoperable, and calculates Hi Φ reactor trip limit based on the indicated flux tilt (107% - 18% = 89%).

B. Correct. Both Parts 1) and 2) are correct.

C. Incorrect. Part 1) is incorrect, 75% is the limiting reactor power for the inoperable rod. Plausible if the Candidate fails to consider the dropped rod as inoperable, and calculates reactor power limit due to flux tilt based on reduction from Current Power vice Rated power. Part 2) is correct.

D. Incorrect. Part 1) is correct, 75% is the reactor power limit for an inoperable rod. Part 2) is incorrect, inoperable rod Hi Φ limit is 85%. Plausible Candidate fails to consider the dropped rod as inoperable, and calculates Hi Φ reactor trip limit based on indicated flux tilt (107% - 18% = 89%).

TS 3.12-7
06-25-09

5. The allowable QUADRANT POWER TILT is 2.0% and is only applicable while operating at THERMAL POWER > 50%.

6. If, except for operation at THERMAL POWER < 50% or for physics and control rod assembly surveillance testing, the QUADRANT POWER TILT exceeds 2%, then:

   a. Within 2 hours, either the hot channel factors shall be determined and the power level adjusted to meet the requirement of Specification 3.12.B.1, or

   b. The power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

   c. If the QUADRANT POWER TILT exceeds 10%, the power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT within the next 30 minutes. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

TS 3.12 Basis:

TS 3.12-20
12-04-14

A 2% QUADRANT POWER TILT allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod assembly and an error allowance. The value 1.02 (2%) was selected because the purpose of the specification is to limit, or require detection of, gross changes in core power distribution between monthly incore flux maps. In addition, it is the lowest value of quadrant power tilt that can be used for an alarm without spurious actuation.

The QPTR limit must be maintained during power operation with THERMAL POWER > 50% of RATED POWER to prevent core power distributions from exceeding the design limits.

TS 3.12-8
06-25-09

c. If the hot channel factors are not determined, then the Overpower ΔT and Overtemperature ΔT trip setpoints shall be reduced by the equivalent of 2% power for every 1% QUADRANT POWER TILT within the next 4 hours, and the Nuclear Regulatory Commission shall be notified.

C. Control Rod Assemblies

1. To be considered OPERABLE during startup and POWER OPERATION each control rod assembly shall:

1) be trippable,

2) aligned within ± 12 steps or ± 24 steps of its group step demand position, as defined in Section 3.12.E.1.b, and

3) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

3. Startup and POWER OPERATION may continue with one control rod assembly inoperable provided that within one hour either:

a. The control rod assembly is restored to OPERABLE status, as defined in Specification 3.12.C.1 and 2, or

b. the shutdown margin requirement of Specification 3.12.A.3.c is satisfied. POWER OPERATION may then continue provided that:

1) either:

(a) power shall be reduced to less than 75% of RATED POWER within one (1) hour, and the High Neutron Flux trip setpoint shall be reduced to less than or equal to 85% of RATED POWER within the next four (4) hours, or

TS 3.12-11
07-28-11

E. Rod Position Indication System and Bank Demand Position Indication System

1. From movement of control banks to achieve criticality and with the REACTOR CRITICAL, rod position indication shall be provided as follows:

a. Above 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 12 steps of their respective group step demand counter indications.

NUMBER	PROCEDURE TITLE	REVISION
0-AP-1.00	ROD CONTROL SYSTEM MALFUNCTION	27
		PAGE 5 of 8

Ensure Candidate is bounded by Tech Spec Requirements, Not limit imposed by 0-AP-1.00.

18. ___ CHECK REACTOR POWER - LESS THAN OR EQUAL TO 75%	Do the following: ☐ a) Reduce Reactor Power to between 70% - 74% within one hour. ☐ b) Reduce NIS High Flux trip setpoints to less than or equal to 85% within the following four hours. ☐ c) WHEN Reactor Power has been reduced, THEN GO TO Step 19.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?
- Yes → RO question
- No ↓

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"
- Yes → RO question
- No ↓

Can question be answered *solely* by knowing the TS Safety Limits?
- Yes → RO question
- No ↓

Does the question involve one or more of the following for TS, TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(2) for SRO-only

ES 401-9 COMMENTS (WITH CORRECTIONS SHOWN IN RED), 3/28/16

The only required TS knowledge is the LCO. The required actions in this question do not seem to come from TS. No procedure is selected.

Annotated CFR reference in ES-401-5 with reference to SRO only objective. Added List of Tier 2 procedures to References. Removed Figure 1 [CFR55.43(b)(2)], and Figure 2 [CFR55.43(b)(5)].

ES 401-9 COMMENTS, 3/30/16: Subsets. If C/D1 is true, I suspect A/B1 is also true. In the first part specify that the action is required by (relevant procedure). The second part "time limit" should have an answer of time The word 'report' is not appropriate".

Amended Part 1 by adding "in accordance with OP_RX-006". In Part 2 modified the question by making a fill in blank to delete the word "report" from choices.

Per telephone conversation on 3/31/16, changed Part 1) to state that the Team was opening the Reactor Trip breakers and asking if E-0 would or would not be required. Also added references to 1-GOP-1.7, Unit Startup procedure P&L for manual opening of the Reactor Trip breakers and Startup Certification Lesson Plan ND-87-LP-1 where the difference between requirement to initiate E-0 or not is taught.

"The Team has opened" is not correct. The breakers are clearly still closed.

Rev 71 of Surry E-0 says "This procedure is applicable when RCS temperature is greater than or equal to 350°F assuming the RHR system is not in service and SI is operable." I am not willing to test an excerpt from a lesson plan when it seems to contradict an EOP.

Neither of the contents of "which of the following" is a question. A question mark at the end is inappropriate.

If the premise of the question is to recognize from indications that the reactor is critical below the RIL, then I suggest making a distractor consistent with the belief that the critical position is acceptable.

Revised to remove "The team has opened…" Changed the question to require candidate to determine reportability IAW VPAP-2802. Candidate must still analyze indications given and realize the reactor was critical below MIL. The fact that one trip button did not work but the other one did makes the 1- and 24-hour reports plausible. Correct answer changed from "D" to "B".

ES-401-9 Comment 4-12-16, Unsat, K/A Match.

This question does not require the applicant to determine that the reactor is critical below the RIL or what to do about it, and, as such does not test "operational judgments based on . . . instrument interpretation". All the applicant needs to know is that the crew tripped a critical reactor.

I suggest you construct a question such that there is a distractor that would be appropriate if the RIL requirement is met. Asking about a report gives away that some requirement is not met.

Question completely rewritten to change focus to TS requirement and basis for quadrant power tilt.

Question further edited based on feedback from Exam Chief on 4/15/16.

Exam Chief Feedback 4-15-16:

K/A Number: G2.1.7, Conduct of operations, Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation.

Level: SRO Tier #: 3 Group #: IR – RO: 4.4 IR-SRO: 4.7

Proposed Question: 92

Initial Conditions

Unit 1 is stable at 100% reactor power.

- Control Rod E-5, Shutdown Bank "B" dropped, "0" steps indicated.
- The Team responds per 0-AP-1.00, Rod Control System Malfunction.
- 0-AP-1.00, Attachment 6, Calculation of Excore Quadrant Power Tilt Ratios, has been completed with a result of 11% calculated quadrant power tilt.
- Power is stabilized at 95% of rated power.

If the control rod is not recovered and quadrant power tilt does not change,

3. after one hour elapses, the maximum allowed reactor power level is ________.
4. after four hours elapse, the maximum allowed value of the high flux power range trip setpoint is _____.

A. 1) 73%
   2) 87%.

B. 1) 75%
   2) 85%.

C. 1) 73%
   2) 85%.

D. 1) 75%
   2) 87%.

Proposed Answer: B.

Make this open reference. Give the applicants only the Quadrant power tilt and control rod assembly sections of TS.

REFERENCE PROVIDED

5. The allowable QUADRANT POWER TILT is 2.0% and is only applicable while operating at THERMAL POWER > 50%.

6. If, except for operation at THERMAL POWER < 50% or for physics and control rod assembly surveillance testing, the QUADRANT POWER TILT exceeds 2%, then:

   a. Within 2 hours, either the hot channel factors shall be determined and the power level adjusted to meet the requirement of Specification 3.12.B.1, or

   b. The power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

   c. If the QUADRANT POWER TILT exceeds 10%, the power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT within the next 30 minutes. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

7. If, except for operation at THERMAL POWER < 50% or for physics and control rod assembly surveillance testing, after a further period of 24 hours, the QUADRANT POWER TILT in Specification 3.12.B.5 above is not corrected to less than 2%:

   a. If the design hot channel factors for RATED POWER are not exceeded, an evaluation as to the cause of the discrepancy shall be made and a special report issued to the Nuclear Regulatory Commission.

   b. If the design hot channel factors for RATED POWER are exceeded and the power is greater than 10%, then the high neutron flux, Overpower ΔT and Overtemperature ΔT trip setpoints shall be reduced 1% for each percent the hot channel factor exceeds the RATED POWER design values within the next 4 hours, and the Nuclear Regulatory Commission shall be notified.

c. If the hot channel factors are not determined, then the Overpower ΔT and Overtemperature ΔT trip setpoints shall be reduced by the equivalent of 2% power for every 1% QUADRANT POWER TILT within the next 4 hours, and the Nuclear Regulatory Commission shall be notified.

C. Control Rod Assemblies

1. To be considered OPERABLE during startup and POWER OPERATION each control rod assembly shall:

   1) be trippable,

   2) aligned within ± 12 steps or ± 24 steps of its group step demand position, as defined in Section 3.12.E.1.b, and

   3) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

2. To be considered OPERABLE during shutdown modes, each control rod assembly shall:

   1) be trippable, and

   2) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

3. Startup and POWER OPERATION may continue with one control rod assembly inoperable provided that within one hour either:

   a. The control rod assembly is restored to OPERABLE status, as defined in Specification 3.12.C.1 and 2, or

   b. the shutdown margin requirement of Specification 3.12.A.3.c is satisfied. POWER OPERATION may then continue provided that:

      1) either:

(a) power shall be reduced to less than 75% of RATED POWER within one (1) hour, and the High Neutron Flux trip setpoint shall be reduced to less than or equal to 85% of RATED POWER within the next four (4) hours, or

(b) the remainder of the control rod assemblies in the group with the inoperable control rod assembly are aligned to within 12 steps of the inoperable rod within one (1) hour while maintaining the control rod assembly sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT; the THERMAL POWER level shall be restricted pursuant to Specification 3.12.A during subsequent operation.

2) the shutdown margin requirement of Specification 3.12.A.3.c is determined to be met within one hour and at least once per 12 hours thereafter.

3) the hot channel factors are shown to be within the design limits of Specification 3.12.B.1 within 72 hours. Further, it shall be demonstrated that the value of $F_{xy}(Z)$ used in the Constant Axial Offset Control analysis is still valid.

4) a reevaluation of each accident analysis of Table 3.12-1 is performed within 5 days. This reevaluation shall confirm that the previous analyzed results of these accidents remain valid for the duration of operation under these conditions.

K/A Number: G2.2.13, Equipment Control, Knowledge of tagging and clearance procedures.

Level: SRO Tier #: 3 Group #: IR – RO: 4.1 IR-SRO: 4.3

Proposed Question: 93

A Tagout is being generated using a Relief valve as part of the boundary with two breakers that may require grounding devices.

In accordance with OP-AA-200, Equipment Clearances:

1) Using relief valves as part of the boundary requires ____ approval.
2) The minimum voltage that requires grounding devices when working on electrical conductors, are those conductors that operate greater than ____ volts.

A. 1) Licensed SRO
   2) 600

B. 1) Licensed SRO
   2) 150

C. 1) Operations Manager on Call (OMOC)
   2) 600

D. 1) Operations Manager on Call (OMOC)
   2) 150

Proposed Answer: C

Explanation: 1) When using relief valves as part of the boundary the Manager Nuclear Ops Approval provides approval. An OMOC qualified individual may provide this permission as long as it is documented on Tagging record or the Narrative Log. 2) For conductors that operate greater than 600 volts grounding devices are required.

Technical Reference: OP-AA-200, Equipment Clearance, rev 23

Reference Provided to Applicant: NO

Learning Objective:

RO/SROSROUTP-SDS-02, Administrative Procedures

A. For a Tier 1 procedure, discuss the following:

- Purpose of the procedure
- Responsibilities, as it applies to operations personnel (Operations Personnel, Shift Operators, Reactor Operators, etc.)
- Who is required to authorize specific plant operations or process changes (for example, procedure modifications). For example, who is required to authorize the use of an air operated valve as an isolation boundary on a tagout. This is important because you, as a member of the control room staff, need to understand what authorizations are required BEFORE you direct an action in the plant.
- Knowledge items applicable to operators. For example, in OP-AA-100, how valve verifications are performed. This is specific to operators, and operators need to have this knowledge item in the carrying out of their daily duties.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: SRO ONLY TASK: Task # D784, Authorize placement/clearance of a tagging report.

Comments:

KA Match Analysis: This question matches the KA in that it requires the candidate to have in-depth knowledge of the tagging process, specifically whose permission is required for use of relief valves and requirements for placement of grounding devices.

Distractor Analysis:

A. Incorrect. 1) Incorrect because OP-AA-200, specifically requires the Manager Nuclear Operations, or OMOC to approve use of relief valves. Plausible because the Shift Manager or designee (Licensed SRO) is authorized to approve clearances. 2) Correct.
B. Incorrect. 1) Incorrect because OP-AA-200, specifically requires the Manager Nuclear Operations, or OMOC to approve use of relief valves. Plausible because the Shift Manager or designee (Licensed SRO) is authorized to approve clearances. 2) Incorrect because OP-AA-200 specifically requires grounding devices to be used for conductors that operate greater than 600 volts. Plausible because grounding devices are optional (not required) for voltages < 600 volts. Also plausible if candidate confuses voltage requirement for grounding devices, with voltage requirement for lifting jumpers that require plant manager permission (150 volts).
C. CORRECT
D. Incorrect. 1) Correct. 2) Incorrect because OP-AA-200 specifically requires grounding devices to be used for conductors that operate greater than 600 volts. Plausible because grounding devices are optional (not required) for voltages < 600 volts. Also plausible if candidate confuses voltage requirement for grounding devices, with voltage requirement for lifting jumpers that require plant manager permission (150 volts).

Reference OP-AA-200 page 33, shows correct answer for part 1.

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Operations

c. **Non-pneumatic Pressure Operated Valves (Safety/Relief Valves)**

1. Do **NOT USE** non-pneumatic pressure operated valves as isolation boundary valves unless:

Method 1

- The valve is physically restrained in the required position through the use of an approved gagging/blocking device that is tagged and listed on the Tagging Record.

Manager Nuclear Operations

- Manager Nuclear Operations approval to use Safety/Relief Valves as a boundary may be performed by an OMOC qualified individual. When permission is given, **DOCUMENT** on Tagging Record or in the Narrative Log.

Engineering

- **PERFORM** an engineering assessment to evaluate the gagging/block effects on the system/valve and code requirements

Operations

Method 2

- An operational assessment determines that no additional risk of personnel injury exists. This assessment should include but **NOT** be limited to:
  - Probability of the relief valve lifting
  - System(s) pressure (and control of pressure)
  - System fluid type
  - System temperature

Manager Nuclear Operations

- Manager Nuclear Operations approval to use Safety/Relief Valves as a boundary may be performed by an OMOC qualified individual. When permission is given, **DOCUMENT** on Tagging Record or in the Narrative Log.

Reference OP-AA-200 page 13, shows distractor for part 1.

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3.2.4 **Authorization of Tagging Record to be Hung**

Shift Manager or Designee

a. After the SRO finishes the completeness and accuracy review, **DETERMINE** whether the Tagout can be authorized by the following:

1. **ENSURE** that any required testing is initiated prior to tagging.
2. **ENSURE** work is ready to begin.
3. **ENSURE** compliance with Technical Specifications requirements, including the following:
   - Operability of any required redundant equipment and satisfactory completion of testing.
   - Compliance with applicable LCOs
4. **ENSURE** effects of the Tagout on indications, instrumentation, or controls outside the boundaries have been evaluated and necessary actions taken to provide compensation for any loss.
5. **ENSURE** compliance with Technical Requirements Manual (TRM), Maintenance Rule, and other requirements.
6. **ENSURE** required Tagout attributes and notes are referred to and are completed.
7. **NOTIFY** Control Room personnel of Tagout to be hung, as applicable.

b. IF conditions for authorizing the Tagout are satisfactory, THEN the Shift Manager or designee shall **PERFORM** the following:

1. **APPROVE** the Tagging record for execution and **SIGN IN** the computer for Tagout to be hung.
2. **ENSURE** tags are hung as soon as possible after authorizing/printing to minimize potential tag sharing discrepancies in eSOMS.

c. IF the conditions for the Tagout are NOT satisfactory, THEN the Shift Manager or designee shall **PLACE** the Tagging Record on hold until conditions permit.

Reference OP-AA-200 page 25, shows correct answer for part 2.

DOMINION

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3.3.5 **Installation and Control of Grounds**

Electrical Maintenance

a. **ENSURE** that all equipment or conductors are de-energized prior to installing grounding devices and placing ground tags.

User

b. **USE** grounding devices when working on electrical conductors that normally operate greater than 600 volts. Equipment grounding for less than 600 volts is optional.

Tagging Office/ Maintenance Craft Requiring the Grounds

c. Work orders requiring grounds should be specified by the tagging request. WHEN required by the Tagging Office, THEN GENERATE a separate work order to install grounds. The maintenance craft requiring the grounds shall REQUEST the ground tags.

Reference OP-AA-200 page 27, shows distractor for part 2

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d. **Bus/Line Potential Fuses**

1. **REMOVE** the fuses. **STORE** inside the breaker or fuse panel.
2. **TAPE** the stub or a copy of the Danger Tag to the removed fuses.
3. **ATTACH** the Danger Tag to the outside of the fuse cabinet or drawer, as appropriate.

3.3.7 **Miscellaneous Electrical Practices**

Electrical Maintenance/ Plant Manager (Nuclear)

a. **ISOLATE** electrical circuits before starting work, IF applicable. IF a circuit must remain live to support maintenance, THEN **TAKE** precautions to prevent electrical shock and loss of safety related equipment. Excluding test equipment, **OBTAIN** permission to lift or jumper live circuits above 150 volts from Plant Manager (Nuclear) or station approved procedures/documents.

K/A Number: G2.2.40, Equipment Control, Ability to apply technical specifications for a system.

Level: SRO Tier #: 3 Group #: IR – RO: 3.4 IR-SRO: 4.7

Proposed Question: 94

Given the following:

- Unit is operating at 100%. Current date is 4/1.
- 0745: Operator on rounds discovers a Tech Spec component is potentially inoperable.
- 0800: Shift Manager declares component inoperable. The particular Tech Spec LCO does not specify any time limit with unit at 100% (no "B" spec).
- 0900: Unit Shutdown commenced.
- 1200: Hot Shutdown mode achieved.

Which of the following state the MAXIMUM time the team has to achieve Cold Shutdown?

A. 4/2, 1800.

B. 4/2, 2000.

C. 4/2, 2100.

D. 4/3, 0000.

Proposed Answer: B

Explanation: In the event an LCO does not specify any time limit with the Unit at 100%, the unit shall be in Hot Shutdown within 6 hours and cold shutdown within the following 30 hours. The following guidance is included in the Tech Spec basis: "The application of 6 hours to hot shutdown plus the following 30 hours to cold shutdown for a total of 36 hours from initial action entry is appropriate; thus if 3 hours are used to attain hot shutdown, 33 hours remain to attain cold shutdown. Therefore 36 hours from time SSC is declared inoperable would require Cold shutdown to be attained no later than 4/2, 2000.

Technical Reference: Tech Spec 3.01, Rev. 47.

Reference Provided to Applicant: No

Learning Objective:

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.2 / 43.5 / 45.3)

Comments:

K/A Analysis: The question poses a generic statement that requires the SRO candidate to apply Tech Spec 3.01 including Bases knowledge regarding time to complete Cold shutdown.

Distractor Analysis:

A. Incorrect. Total time to achieve cold shutdown is 36 hours from time SSC was declared inoperable. Plausible if Candidate applies 30 hours from the time HSD was achieved, vice total of 36 hours. This is a common misconception.
B. Correct.
C. Incorrect. Total time to achieve cold shutdown is 36 hours from time SSC was declared inoperable. Plausible if Candidate applies 36 hours from the time shutdown started, vice total of 36 hours. This is a common misconception.
D. Incorrect. Total time to achieve cold shutdown is 36 hours from time SSC was declared inoperable. Plausible if Candidate applies 36 hours from the time HSD was achieved, vice total of 36 hours. This is a common misconception.

References Tech Spec 3.01, and Bases.

3.0 LIMITING CONDITIONS FOR OPERATION

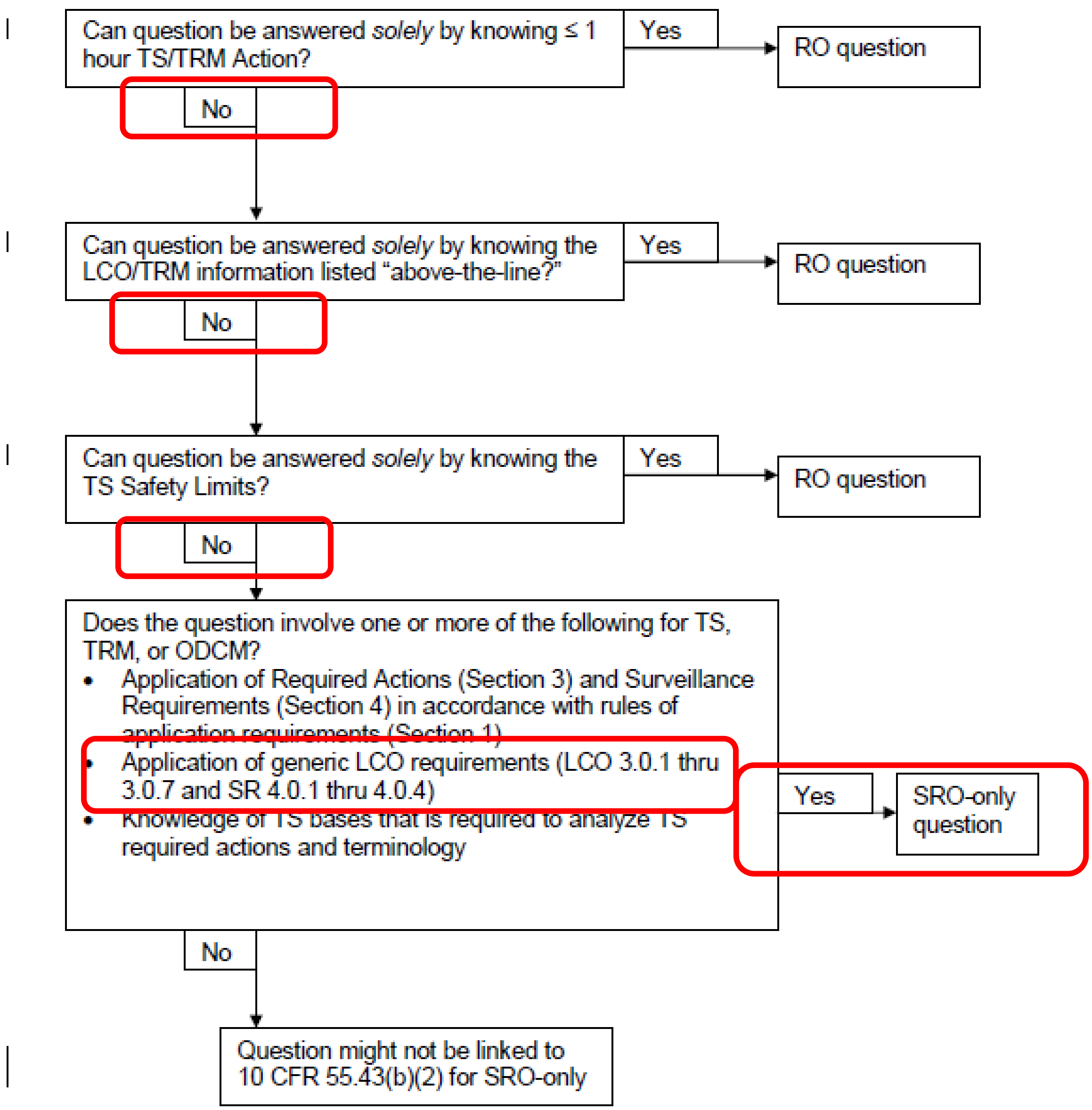
3.0.1 In the event a Limiting Condition for Operation and/or associated modified requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least hot shutdown within 6 hours and in at least cold shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

Basis

3.0.1 This specification delineates the action to be taken for circumstances not directly provided for in the action statements and whose occurrence would

Regarding action statements structured in the manner of TS 3.0.1 (i.e., in at least hot shutdown within 6 hours and in at least cold shutdown within the following 30 hours), the following application guidance is provided. The application of 6 hours to hot shutdown plus the following 30 hours to cold shutdown for a total of 36 hours from initial action statement entry is appropriate; thus, if 3 hours are used to attain hot shutdown, 33 hours remain to attain cold shutdown. In addition, the time provided for achieving a reduction in operational mode is not applied if the inoperability is discovered in a lower operational mode; therefore, if an inoperable condition is discovered in hot shutdown, the 6 hours provided to achieve hot shutdown is not included in the total time to achieve cold shutdown (30 hours).

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



K/A Number: G2.3.12, Radiation Control, Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Level: SRO Tier #: 3 Group #: IR – RO: 3.2 IR-SRO: 3.7

Proposed Question: 95

Initial Conditions:

- Unit 1 is in Refueling Shutdown with defueling operations in progress.
- The tenth fuel assembly has been lifted into the Manipulator crane.
- A catastrophic failure of the cavity seal results in cavity level lowering.
- Cavity level is 26 feet and lowering rapidly.
- Annunciator 1-RM-J8 Manipulator crane Alert/Failure has just alarmed.

Current Conditions:

- All Containment personnel have been evacuated.
- Fuel assembly has been place in vessel, and all fuel assemblies are covered.
- The following Annunciators have just alarmed:
  - 1-RM-K8, 1-RM-RI-162, Manipulator Crane HIGH.
  - 1-RM-M7, 1-RM-RI-163, Rx. Containment HIGH.

Which ONE of the following identifies:

1) The maximum allowable dose that can be authorize by the SEM to repair the cavity seal, per EPIP-4.04, Emergency Personnel Radiation Exposure is __(1)__.
2) The classification for this event in accordance with the EAL matrix is __(2)__.

REFERENCE PROVIDED

A. 1) 10 REM
   2) Alert, RA2.1

B. 1) 10 REM.
   2) NOUE, RU2.1

C. 1) 25 REM.
   2) NOUE, RU2.1

D. 1) 25 REM.
   2) Alert, RA2.1

Proposed Answer: B.

Explanation: 10 REM is the maximum allowed dose that can be authorized by the SEM to limit releases to the public. The EAL classification is NOUE RU-2.1, because there is an unexpected increase in plant radiation (Manipulator crane, RI-162 ALERT), and a valid low level in the reactor cavity. The Alert RA 2.1 is not correct because there is no fuel damage or uncovering of fuel.

Technical Reference: EPIP-4.4. Emergency Personnel Radiation Exposure, Attachment 1, Rev. 9. EAL Matrix, Rev.2.

Reference Provided to Applicant: Yes (EAL Matrix, Rev. 4.)

Learning Objective: ND-81.2-LP-3, External Exposure. Objective C: "Explain the federal exposure limits for Total Effective Dose Equivalent, extremity, skin, and Lens of the Eye doses including necessary requirements and limitations for extensions." ND-95.5-LP-2, SEM, Objective C, Using EPIP-1.01, Emergency Manager Controlling Procedure, analyze plant situations and determine the appropriate classification utilizing the EAL charts (both HOT and COLD conditions).**(STA/SRO Only)**

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: ND-95.5-LP-2, SEM. Objective C, **(STA/SRO only)**

CFR 55.43(b)(7), FUEL HANDLING

Comments:

K/A match Analysis: Question matches K/A. Candidate must assess Unit condition and determine radiological consequences of dose received by an operator dispatched to perform a function, and determine the classification for the event in progress.

Distractor Analysis:

A. Incorrect. Part 1) is correct, 10 REM is allowed per EPIP-4.40 to isolate a release that may affect the public. Part 2) is incorrect, NOUE RU-2.1, is the correct answer because there is an unexpected increase in plant radiation (Manipulator crane, RI-162 ALERT), and a valid low level in the reactor cavity. RA 2.1 is plausible because that EAL is partially correct, in that a valid HIGH alarm is in BUT the fuel is not damaged or uncovered.

B. Correct. Both Parts 1) and 2 are correct.

C. Incorrect. Part 1) is incorrect, 10 REM is the limit. Plausible should Candidate confuse the requirements for Emergency Dose allowed for equipment control and lifesaving activities. Part 2) is correct.

D. Incorrect. Part 1) is incorrect, 10 REM is the limit. Plausible should Candidate confuse the requirements for Emergency Dose allowed for equipment control and lifesaving activities. Part 2) is incorrect, NOUE RU-2.1, is the correct answer because there is an unexpected increase in plant radiation (Manipulator crane, RI-162 ALERT), and a valid low level in the reactor cavity. RA 2.1 is plausible because that EAL is partially correct, in that a valid HIGH alarm is in BUT the fuel is not damaged or uncovered.

NUMBER EPIP-4.04	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 9	EMERGENCY EXPOSURE LIMITS	PAGE 1 of 1

TABLE 1: EPA-400 EMERGENCY EXPOSURE LIMITS			
ACTIVITY	TEDE (Rem)	LDE (Rem)	SDE, THY, CDE, OR OTHER ORGAN (Rem)
GENERAL EMERGENCY EXPOSURE ACTIVITIES	5	15	50
PROTECTING VALUABLE PROPERTY (1)	10	30	100
LIFESAVING OR PROTECTION OF LARGE POPULATIONS (2)	25	75	250
LIFESAVING OR PROTECTION OF LARGE POPULATIONS (3)	> 25 Only on a voluntary basis to persons fully aware of the risks involved.	> 75	> 250

(1) Protecting Valuable Property:

- To save valuable equipment.
- To limit off-site releases.

(2) Lifesaving Activity:

- For search and rescue, first aid, and removal of injured personnel where there is reasonable expectation that the individual(s) is alive within the affected area.
- For entry to correct conditions which, if left uncorrected, could result in on-site or off-site injury.

Reference EAL MATRIX; RU 2.1 (CORRECT), RA 2.1 (INCORRECT)

RA2a Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel 1 | 2 | 3 | 4 | 5 | 6 | DEF	RU2 Unexpected increase in plant radiation 1 | 2 | 3 | 4 | 5 | 6 | DEF
RA2.1 Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel resulting in a valid high alarm on any of the following radiation monitors: - RM-RI-152 New Fuel Storage Area - RM-RI-153 Fuel Pit Bridge - RM-RI-( )62 Manipulator Crane - RM-RI-( )63 Reactor Containment - RM-RI-( )60 Containment Gas - RM-RI-( )59 Containment Particulate - VG-RI-131-(A,B,C) - Vent #2	**RU2.1** Valid low water level alarm indicating uncontrolled water level decrease in the refueling cavity, spent fuel pit or fuel transfer canal with all irradiated fuel assemblies remaining covered by water **AND** Unplanned valid direct area radiation reading increases resulting in a valid alert alarm on any of the following radiation monitors: - RM-RI-152 New Fuel Storage Area - RM-RI-153 Fuel Pit Bridge - RM-RI-( )62 Manipulator Crane - RM-RI-( )63 Reactor Containment
RA2.2 A water level drop in the reactor refueling cavity, spent fuel pit or fuel transfer canal that will result in irradiated fuel becoming uncovered	**RU2.2** Unplanned valid direct area radiation monitor reading increases by a factor of 1000 over normal* levels * Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value
RA2b Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown	
RA2.3 Valid radiation monitor or survey readings > 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions: Control Room RM-RI-157 **OR** Central Alarm Station (CAS)	

K/A Number: G2.4.46, Emergency Procedures/Plans, Ability to verify that the alarms are consistent with the plant conditions.

Level: SRO Tier #: 3 Group #: IR – RO: 4.2 IR-SRO: 4.2

Proposed Question: 96

- Unit 1 has the following indications:
  - "A" S/G – 200 psig, lowering
  - "B" S/G – 210 psig, lowering
  - "C" S/G – 205 psig, lowering
  - CETC - 397°F, lowering
  - RCS pressure – 900 psig, lowering
  - CTMT sump level – 1.8 feet, rising
  - CTMT pressure – 58 psia, lowering
- The following annunciators are lit:
  - 1E-B9, CTMT HI PRESS – RED
  - 1A-A1, RWST TECH SPEC LO LVL
    1B-A3, CTMT SUMP HI LVL
  - 1B-C4, CLS HI-HI TR A
  - 1B-C5, CLS HI-HI TR B
  - 1C-B8, PRZR LO PRESS
  - 1G-B1, APPROACH TO SATURATION TEMP ALARM

1) Are the listed annunciators consistent with the plant event in progress?
2) What are the procedure transitions for the plant event in progress?

REFERENCE PROVIDED

A. 1) No.
   2) E-0 → E-2 → ECA-2.1.

B. 1) No.
   2) E-0 → E-1 → ES-1.3 → E-1.

C. 1) Yes.
   2) E-0 → E-2 → ECA-2.1.

D. 1) Yes.
   2) E-0 → E-1 → ES-1.3 → E-1.

Proposed Answer:

A. 1) No.
   2) E-0 → E-2 → ECA-2.1.

Explanation: For the event described, Approach to Saturation is NOT expected. In fact, due to the cooling effect of the steam break subcooling should rise significantly. The team would be expected to enter E-0, then go to E-2 to isolate the faulted S/G, then ECA-2.1 to mitigate the effects of three faulted S/G's. Once AFW flow is throttle to minimum, transition to 1-E-1 is made.

Technical Reference: 1G-B1, APPROACH TO SATURATION TEMP ALARM, Rev. 1; 1-E-0, Reactor Trip or Safety Injection, Rev. 71; 1-E-2, Faulted Steam Generator Isolation, Rev. 20; 1-ECA-2.1, Uncontrolled Depressurization of All Steam Generators, Rev. 41.

Reference Provided to Applicant: Yes (Steam Tables)

Learning Objective: ND-95.2-LP-3 – Over Cooling Transients; Objective A. Break down the expected general plant responses to secondary break transients.

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Comments:

KA Match Analysis: The question matches the KA in that the candidate must analyze the indications and verify the alarms are consistent with conditions/indications.

Distractor Analysis:

A. CORRECT.
B. Incorrect, but plausible if the candidate assumes that a LOCA is in progress. The RWST TS low level alarm is an expected alarm on a reactor trip w/SI.
C. Incorrect, but plausible if the candidate miscalculates subcooling or is distracted by the sump level rise and ctmt pressure rise.
D. Incorrect, but plausible if the candidate assumes that a LOCA is in progress. The Approach to Saturation alarm is expected on a LOCA of significant size.

1G-B1 Excerpt

NUMBER	PROCEDURE TITLE	REVISION
1G-B1	APPROACH-TO SATURATION TEMP ALARM	1
		PAGE 1 of 2

REFERENCES 1G-9

1) UFSAR 7.10
2) 1-DRP-011, Emergency Operating Procedure Setpoints
3) 11448-ESK-10G

PROBABLE CAUSE

1) Alarm actuates when computer calculation equates to less than or equal to 30°F subcooling.

   Saturation temperature alarm is caused by depressurization of the RCS.

2) Instrumentation failure has occurred.

NUMBER	PROCEDURE TITLE	REVISION
1-E-0	REACTOR TRIP OR SAFETY INJECTION	71
		PAGE 6 of 15

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: Seal injection flow should be maintained to all RCPs.	
*8. ___	CHECK RCP TRIP AND MINIFLOW RECIRC CRITERIA:	
	☐ a) Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS	☐ a) GO TO Step 9.
	☐ b) RCS subcooling - LESS THAN 30°F [85°F]	☐ b) GO TO Step 9.
	☐ c) Stop all RCPs	
	☐ d) RCS pressure - LESS THAN 1275 psig [1475 PSIG]	☐ d) GO TO Step 9.
	e) Close CHG pump miniflow recirc valves: ☐ • 1-CH-MOV-1275A ☐ • 1-CH-MOV-1275B ☐ • 1-CH-MOV-1275C	☐ e) Close 1-CH-MOV-1373.
9. ___	CHECK IF SGs ARE NOT FAULTED: • Check pressures in all SGs: ☐ • STABLE OR RISING AND ☐ • GREATER THAN 100 PSIG	NO ☐ IF any SG pressure lowering in an uncontrolled manner OR is completely depressurized, THEN GO TO 1-E-2, FAULTED STEAM GENERATOR ISOLATION.
11. ___	CHECK RCS - INTACT INSIDE CTMT ☐ • CTMT radiation - NORMAL ☐ • CTMT pressure - NORMAL ☐ • CTMT RS sump level - NORMAL	NO ☐ GO TO 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.

NUMBER	PROCEDURE TITLE	REVISION
1-E-2	FAULTED STEAM GENERATOR ISOLATION	20
		PAGE 2 of 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

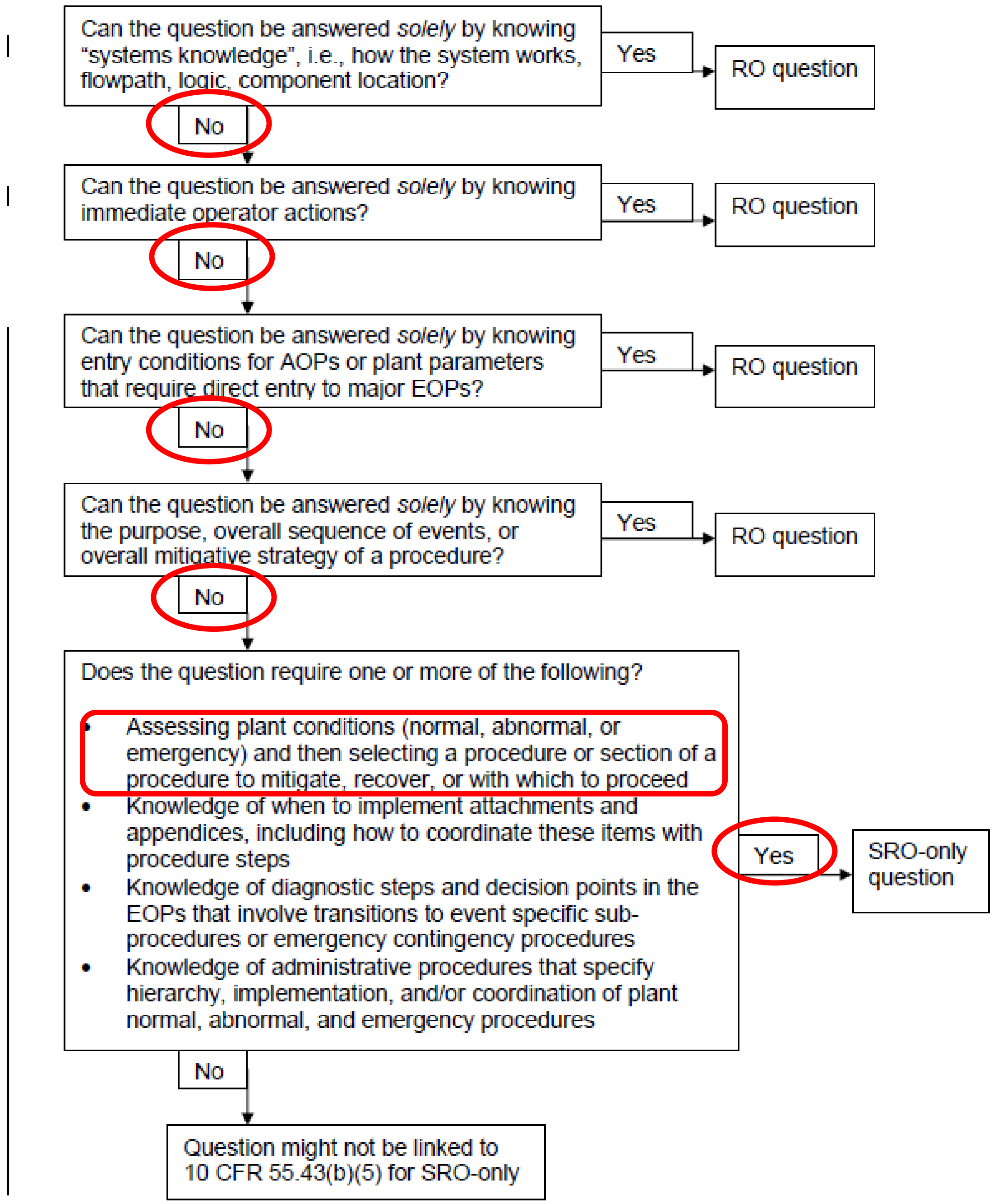
CAUTION:
- At least one SG must be maintained available for RCS cooldown.
- Any faulted SG or secondary break should remain isolated during subsequent recovery actions unless needed for RCS cooldown.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK MSTV AND BYPASS VALVE ON AFFECTED SG(s) - CLOSED	☐ Manually close MSTVs and locally close bypass valves. ☐ IF MSTV can NOT be closed, THEN close MS NRV.
2.___	CHECK IF ANY SG SECONDARY SIDE IS INTACT: ☐ • Check pressures in all SGs - ANY STABLE OR RISING	NO → ☐ • IF all SG pressures lowering in an uncontrolled manner, THEN GO TO 1-ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS.

SRO Justification:

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



NRC APPROVED. K/A Number: G2.4.9, Emergency Procedures/Plans, Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.

Level: SRO Tier #: 3 Group #: IR – RO: 3.8 IR-SRO: 4.2

Proposed Question: 97

Initial Conditions:

- Unit 1 RCS is solid, making preparations to draw a bubble in accordance with 1-GOP-1.1, Unit Startup from Ambient to 195°F.
- RCS temperature is 190°F and rising.
- RCS pressure is 320 psig and stable.
- 'A' RHR pump is in service.
- Loss of 'E' and 'F' Transfer busses occurs.

Current Conditions (12 minutes later):

- RCS temperature is 205°F and rising.

Which ONE of the following identifies:

1) The procedure that the crew shall use to restore RCS temperature to < 200°F is ______.
2) The EAL that shall be declared is NOUE ____.

(REFERENCE PROVIDED)

A. 1) 1-AP-10.27, Loss of All AC Power While on RHR
 2) CU 3.1

B. 1) 1-AP-27.00, Loss of Decay Heat Removal Capability
 2) CU 1.1

C. 1) 1-AP-27.00, Loss of Decay Heat Removal Capability
 2) CU 3.1

D. 1) 1-AP-10.27, Loss of All AC Power While on RHR
 2) CU 1.1

Proposed Answer: C

Explanation: Loss of Transfer busses E and F will result in a loss of Emergency busses 1H, 2J, and 2H. Emergency Diesel #1 will start and load onto 1H Emergency Bus. 1J bus is powered from D transfer bus so it is unaffected.1) The correct procedure to use is 1-AP-27.00 because bus 1J is unaffected. If both busses were lost then 1-AP-10.27 would be apliccable 2) The rise in RCS temperature is be due to a loss of the 'A' RHR pump. The fact that RCS temperature is > 200°F constitutes EAL CU3.1, Loss of decay heat.

Technical Reference: SEAL, Rev. 4

Reference Provided to Applicant: Yes (SEAL chart)

Learning Objective: ND-95.5-LP-2, SEM, Objective C; Using EPIP-1.01, Emergency Manager Controlling Procedure, analyze plant situations and determine the appropriate classification utilizing the EAL charts (both HOT and COLD conditions). **(STA/SRO only)**

Question Source: New

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: ND-95.5-LP-2, SEM; objective C **(STA/SRO only)**

Comments:

K/A ANALYSIS: This question requires knowledge of the emergency bus power supplies and the ability to classify per Emergency Plan Implementing procedures.

Distractor Analysis:

A. Incorrect. 1) Incorrect because AP-10.27 is only used if power is lost to both emergency AC busses. Plausible if candidate doesn't recognize that bus 1J is unaffected and bus 1H only had power lost momentarily. 2) correct.
B. Incorrect. 1) Correct. 2) CU1.1 is incorrect because there is more than one power supply to H and J busses. Plausible because candidate could interpret this loss as a loss to both H and J busses such as the case if D and F transfer busses were lost instead of E and F transfer busses.
C. Correct.
D. Incorrect. 1) Incorrect because AP-10.27 is only used if power is lost to both emergency AC busses. Plausible if candidate doesn't recognize that bus 1J is unaffected and bus 1H only had power lost momentarily. 2) CU1.1 is incorrect because there is more than one power supply to H and J busses. Plausible because candidate could interpret this loss as a loss to both H and J busses such as the case if D and F transfer busses were lost instead of E and F transfer busses.

Reference 1-AP-27.00, LOSS OF DECAY HEAT CAPABILITY (Part 1 correct answer).

ABNORMAL PROCEDURE

NUMBER	PROCEDURE TITLE	REVISION
1-AP-27.00	LOSS OF DECAY HEAT REMOVAL CAPABILITY (WITH 12 ATTACHMENTS)	26
		PAGE 1 of 18

PURPOSE

To provide guidance when the RHR System fails to remove decay heat.

ENTRY CONDITIONS

1) No RHR pumps running due to failure or loss of power.

2) Air-binding of the operating RHR pump as indicated by any of the following:
   - Motor amperage oscillations
   - Flow oscillations
   - Excessive pump noise
   - RHR HX LO FLOW annunciator, 1B-G6

Reference 1-AP-27.00, LOSS OF DECAY HEAT CAPABILITY Step 1 Caution (Part 1 distractor).

NUMBER	PROCEDURE TITLE	REVISION
1-AP-27.00	LOSS OF DECAY HEAT REMOVAL CAPABILITY	26
		PAGE 2 of 18

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- Loss of RHR due to a total loss of IA is addressed by 0-AP-40.00, NON-RECOVERABLE LOSS OF IA.
- Loss of RHR due to a total loss of AC Power is addressed by 1-AP-10.27, LOSS OF ALL AC POWER WHILE ON RHR.
- Loss of RHR may cause CTMT radiological and heat stress conditions to degrade. Local actions in CTMT should be coordinated with HP.
- During solid plant operation, inadvertent actuation of the OPMS may occur if letdown is isolated.
- If RCS boiling occurs, non-essential personnel should be evacuated from CTMT.

* *

NOTE: EPIPs may be applicable.

Reference 1-AP-10.27, LOSS OF ALL AC POWER WHILE ON RHR, Entry Conditions (P1 distractor).

NUMBER	PROCEDURE TITLE	REVISION
1-AP-10.27	LOSS OF ALL AC POWER WHILE ON RHR (WITH 7 ATTACHMENTS)	2
		PAGE 1 of 29

PURPOSE

The purpose of this guideline is to provide the actions necessary for maintaining core cooling and protecting the reactor core, containment, and spent fuel pit in the event of a complete loss of all AC power while on shutdown cooling.

TIME SENSITIVE ACTIONS

ENTRY CONDITIONS

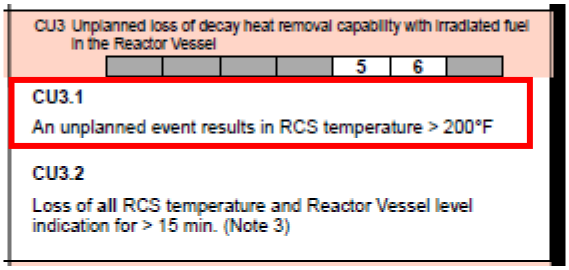
This procedure is applicable when RCS temperature is less than 350°F and RHR is in service. Using this procedure in any other plant condition requires a step by step evaluation to determine if a specified action is still applicable.

1) Transition from 1-AP-27.00, Loss of Decay Heat Removal Capability.

2) Indications exist that both emergency AC buses are de-energized.

3) RHR is aligned for service.

Reference: SEAL CU3.1 (P2 Correct answer).



CU3 Unplanned loss of decay heat removal capability with irradiated fuel in the Reactor Vessel

				5	6	

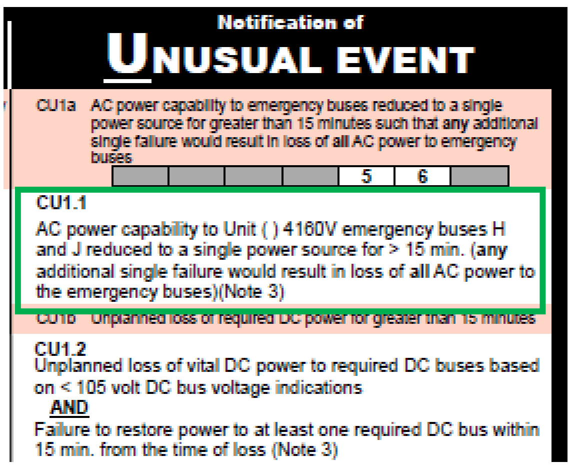
CU3.1

An unplanned event results in RCS temperature > 200°F

CU3.2

Loss of all RCS temperature and Reactor Vessel level indication for > 15 min. (Note 3)

Reference: SEAL CU1.1 (P2 distractor). Incorrect because 1H has EDG1, and 1J has D transfer bus.



Notification of

UNUSUAL EVENT

CU1a AC power capability to emergency buses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in loss of all AC power to emergency buses

				5	6	

CU1.1

AC power capability to Unit ( ) 4160V emergency buses H and J reduced to a single power source for > 15 min. (any additional single failure would result in loss of all AC power to the emergency buses)(Note 3)

CU1b Unplanned loss of required DC power for greater than 15 minutes

CU1.2

Unplanned loss of vital DC power to required DC buses based on < 105 volt DC bus voltage indications

AND

Failure to restore power to at least one required DC bus within 15 min. from the time of loss (Note 3)

ES 401-9 COMMENTS (WITH CORRECTIONS SHOWN IN RED), 3/28/16

Identification of the next action is not procedure selection.

Rewrote the question following discussion with Chief Examiner. Question entails a loss of power and requires candidate to classify EAL. Updated references section, removed Figure 2 [CFR55.43(b)(5)].

ES 401-9 COMMENTS, 3/30/16: The question does not test a knowledge of mitigation strategies. The SRO part can be classification but the applicant must also determine mitigation strategies. This may lend itself to procedure selection.

Changed Stem to 'A' RHR pump running to better support distractors since this pump will lose power. Changed Part 1 to question requiring selection of procedure used to mitigate loss of RHR. Kept Part 2 question requiring selection between two NOUEs.

K/A Number: WE08EG2.1.19, RCS Overcooling - PTS/4, Ability to use plant computer to evaluate system or component status.

Level: SRO Tier #: 1 Group #: 2 IR – RO: 3.9 IR-SRO: 3.8

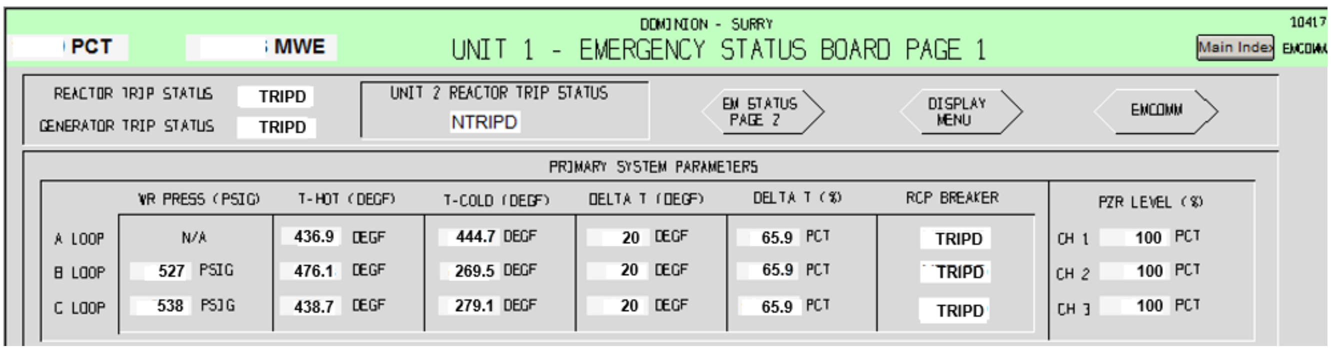
Proposed Question: 98

Initial Conditions:

- Unit 1 has experienced a Reactor Trip and SI.
- Security reports steam exiting the Unit 1 Safeguards Louvers.

Current Conditions:

- 1-E-2 steps that do not require Safeguards entry have been performed, and you have just directed transitioning to 1-E-1, Loss of Reactor or Secondary Coolant.
- Annunciator 1H-C8, AFW PP 3B LOCKOUT OR TRIP has just alarmed.
- The STA relays the following information from PCS.



Based on Current Conditions:

1) What procedure sequence should the SRO direct?
2) What EAL should be declared?

A. 1) Go to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition and perform the steps as directed.
   2) HA 2.1.

B. 1) Remain in 1-E-1, Loss of Reactor or Sec. Coolant and perform the steps as directed.
   2) HU 2.2.

C. 1) Go to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition and perform the steps as directed.
   2) HU 2.2.

D. 1) Remain in 1-E-1, Loss of Reactor or Sec. Coolant and perform the steps as directed.
   2) HA 2.1.

Proposed Answer: A.

Explanation: 1) With the conditions provided, the Candidate should select a transition to FR-P.1 due to RCS temperature and pressure. 2) EAL HA 2.1 because a steam explosion in a safe shutdown area has occurred and there is indications of damage to 1-FW-P-3B, AFW pump which is a safety related pump. HU 2.2 is also satisfied, but the highest EAL which is the EAL that should be declared is HA 2.1.

Technical Reference: FR-H.1, Rev. 37. F-4, Critical Safety Function Status Tree, Rev. 3.

Reference Provided to Applicant: Yes (Unit 1 – Emergency Status Board, Page 1).

Learning Objective: ND-95.3-LP-46, FR-P.1, Objective C, Given a copy of FR-P.1, Response To Imminent Pressurized Thermal Shock Condition, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: Modified (LEOP0765)

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 45.12)

Comments:

K/A Match Analysis: Question Matches K/A. Candidate must assess Plant Computer screen and determine, based on the information, that FR-P.1 to be implemented. Distractors based on choices available to the SRO based on FR-P.1 steps.

Distractor Analysis:

A. Correct.

B. Incorrect. 1) Incorrect. Based on Plant status, entry into FR-P.1 would be required based on “snapshot” on exit from FR-H.1. Plausible if Candidate bases transition on “A” Loop parameters vice lowest RCS Loop Tc indication. 2) Incorrect, because with the damage to 1-FW-P-3B HA 2.1 is met. Plausible because HU 2.2 is also satisfied, but the highest EAL which is the EAL that should be declared is HA 2.1.

C. Incorrect. 1) Correct. 2) Incorrect, because with the damage to 1-FW-P-3B HA 2.1 is met. Plausible because HU 2.2 is also satisfied, but the highest EAL which is the EAL that should be declared is HA 2.1.

D. Incorrect. 1) Incorrect. Based on Plant status, entry into FR-P.1 would be required based on “snapshot” on exit from FR-H.1. Plausible if Candidate bases transition on “A” Loop parameters vice lowest RCS Loop Tc indication. 2) Correct

Reference: Parent question LEOP0765

37	ID: LEOP0765	Points: 1.00

Initial Conditions:

- Unit 1 has experienced an ATWS and loss of heat sink.
- The team performed FR-S.1, Response to Nuclear Power Generation/ATWS, and transitioned to FR-H.1, Response to Loss of Secondary Heat Sink.
- RCS bleed and feed was initiated due to SG WR levels being less than required.
- The team re-established feedwater with "A" Main Feed Pump, 1-FW-P-1A, in service and is slowly feeding "C" SG.
- Hi CLS initiated and containment pressure got to 20 psia.

Current Conditions (one hour into the event):

- "C" SG NR level is 15%.
- "A" and "B" SG NR levels are off-scale low.
- The team has secured RCS bleed and feed.
- Containment pressure has decreased to 18 psia.
- Normal charging flow has been established.
- RVLIS Full Range is > 63%.
- "A" and "B" Loop Cold Leg Temperatures are 195 degrees F.
- "C" Loop Cold Leg Temperature is 385 degrees F.
- RCS pressure is approximately 1000 psig and stable.

Based on Current Conditions, which ONE of the following is the procedure that will be performed next by the team?

A. Go to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition.

B. Go to E-0, Reactor Trip or Safety Injection.

C. Go to ES-1.1, SI Termination, step 10.

D. Go to E-1, Loss of Reactor or Secondary Coolant.

Answer: A

Reference 0-F-4 Status Tree

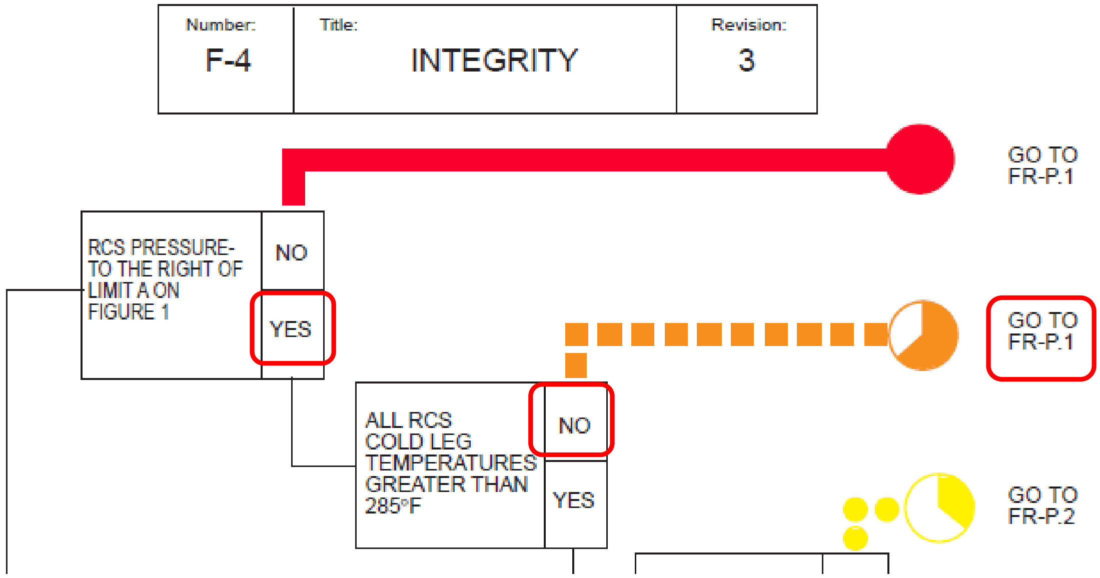
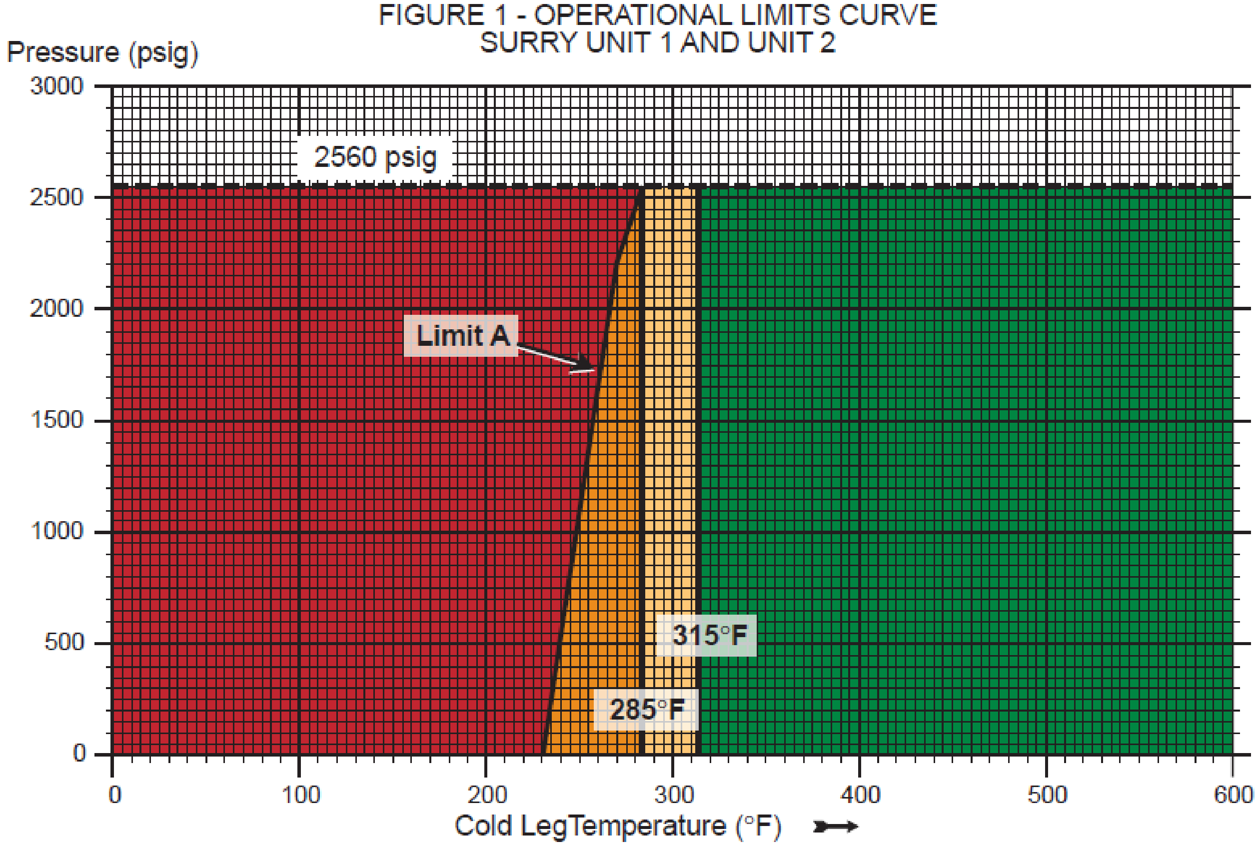


FIGURE 1 - OPERATIONAL LIMITS CURVE
SURRY UNIT 1 AND UNIT 2



Reference 1-FR-P.1 Step 1 RNO, Shows possible distractor based on LBLOCA.

NUMBER	PROCEDURE TITLE	REVISION
1-FR-P.1	RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION	25
		PAGE 2 of 20

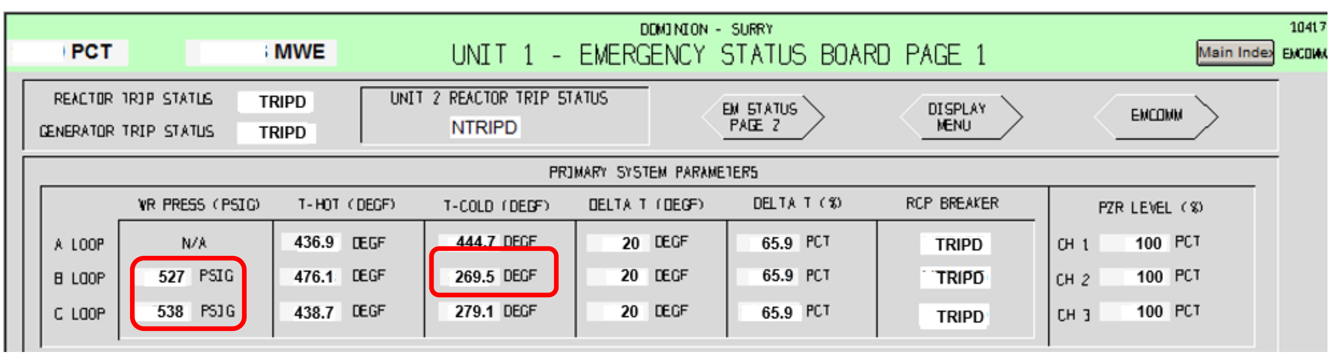
STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK RCS PRESSURE - GREATER THAN 185 PSIG [250 PSIG]	☐ IF LHSI pump flow greater than 1000 gpm, THEN RETURN TO procedure and step in effect.

* *

CAUTION:
- If the TD AFW pump is the only available source of feed flow, steam supply to the TD AFW pump must be maintained from at least one SG.
- Alternate AFW sources will be necessary if ECST level lowers to less than 20%. (Refer to Attachment 1)

* *

PCS Reference.



Reference EAL HA 2.1 (P2 Correct answer)

HA2 Fire or explosion affecting the operability of plant safety-related structures, systems or components required to establish or maintain safe shutdown

1	2	3	4	5	6	DEF

HA2.1

Fire or explosion in **any** Table H-1 area

AND EITHER:

Plant personnel report visible damage to **any** safety-related structure, system, or component within the area

OR

Affected system parameter indications show degraded performance

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Emergency Action Level Technical Bases Document
Attachment 1 – Emergency Action Level Technical Bases

HA2.1 (cont)

Mode Applicability:

All

Basis:

Safety-related structures, systems and components (as defined in 10 CFR 50.2) are those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary
(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

The listed areas contain functions and systems required for the safe shutdown of the plant. The SPS Appendix R report was consulted for equipment and plant areas required for the applicable mode (ref. 1).

The only explosions that should be considered are those of sufficient force to: damage permanent structures or equipment required for safe operation, or result in degraded performance of safety-related structures, systems and components within the identified plant areas. An explosion is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components. No attempt is made to assess the actual magnitude of the damage. The wording of this EAL does not imply that an assessment of safety-related structure, system and component performance should be performed; rather that safety-related structure, system and component parameter symptoms are degraded as a result of the event. The declaration of an Alert and the activation of the TSC provide the SEM with the resources needed to perform damage assessments. The SEM also needs to consider the security aspects of the explosions.

Reference EAL HU 2.2 (P2 Distractor)

HU2 Fire or explosion within Protected Area boundary

1	2	3	4	5	6	DEF

HU2.1

Fire in or restricting access to **any** Table H-1 area **not** extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm (Note 3)

HU2.2

Report by plant personnel of an unanticipated explosion within Protected Area boundary or at the Low Level Intake Structure resulting in visible damage to permanent structure or equipment

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Emergency Action Level Technical Bases Document
Attachment 1 – Emergency Action Level Technical Bases

HU2.2

Category: H – Hazards
Sub-category: 2 – Fire or Explosion
Initiating Condition: Fire or explosion within Protected Area boundary
EAL:

HU2.2	**Notification of Unusual Event**
Report by plant personnel of an unanticipated explosion within Protected Area boundary or at the Low Level Intake Structure resulting in visible damage to permanent structure or equipment	

Mode Applicability:

All

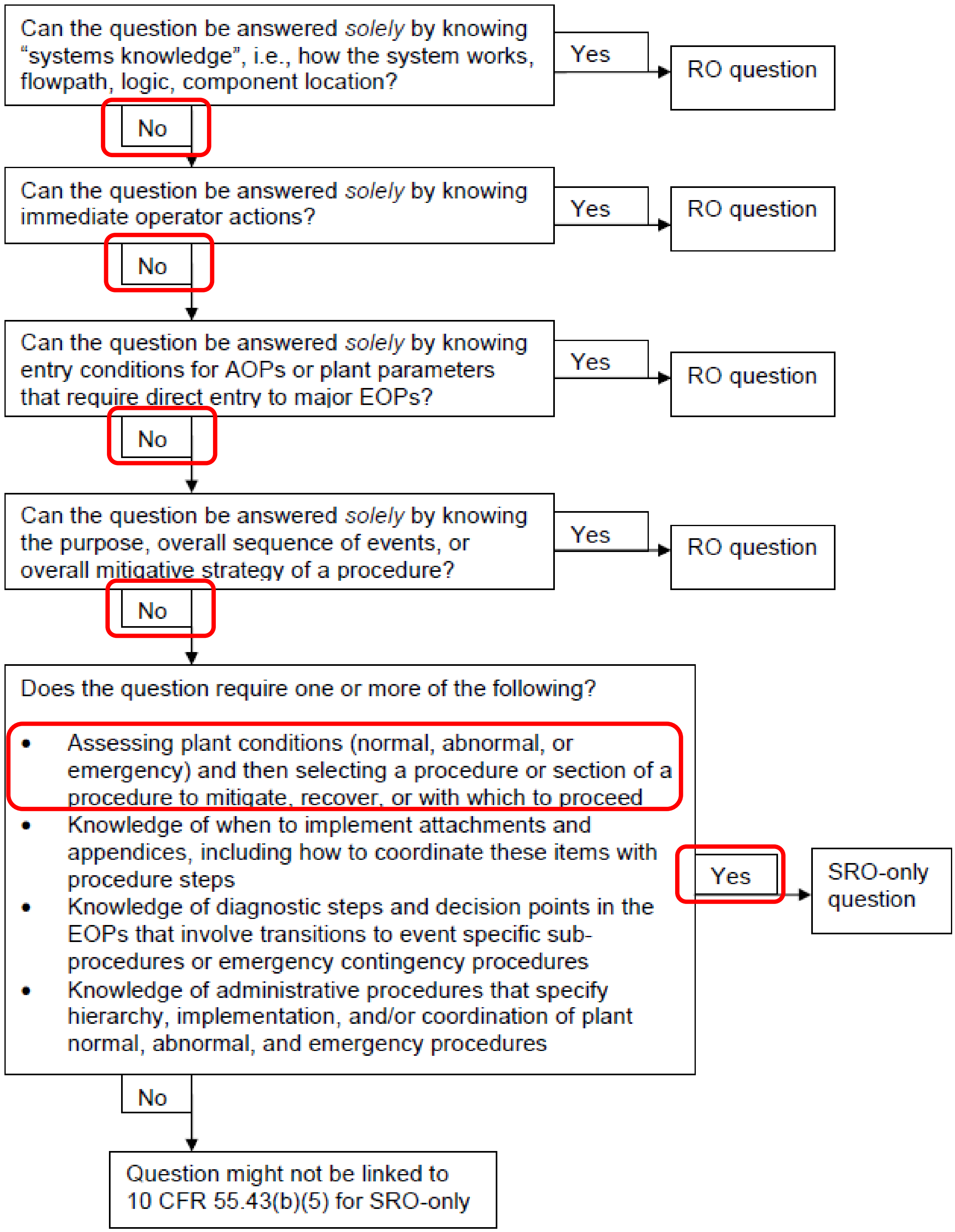
Basis:

For this EAL, only those unanticipated explosions within the Protected Area or at the Low Level Intake Structure should be considered. As used here, an explosion is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the explosion with reports of evidence of damage (e.g., deformation, scorching, etc.) is sufficient for declaration. The SEM also needs to consider any security aspects of the explosion.

A steam line break or steam explosion that damages surrounding permanent structures or equipment would be classified under this EAL. This does not mean the emergency is classified simply because the steam line break occurred. The method of damage is not as important as the degradation of plant structures or equipment.

Escalation of this emergency classification level, if appropriate, would be based on HA2.1.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



K/A Number: WE12EA2.2, Steam Line Rupture - Excessive Heat Transfer / 4, Ability to determine and interpret the following as they apply to the (Uncontrolled Depressurization of all Steam Generators), Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

Level: SRO Tier #: 1 Group #: 1 IR – RO: 3.4 IR-SRO: 3.9

Proposed Question: 99

Given the following conditions:

- Unit-1 tripped from 100% power due to a lightning strike.
- At least one safety valve is stuck open on each S/G.
- The crew is performing 1-ECA-2.1, Uncontrolled Depressurization of all SGs.
- The RO reports that RCS cooldown rate is 250°F/hr.
- The BOP has throttled AFW flow to 65 gpm to each SG.
- SG narrow-range levels are off-scale low.
- The STA reports that a red-path exists on the Heat Sink CSF status tree.
- The SRO transitions to 1-FR-H.1, Response to Loss of Secondary Heat Sink

Which ONE of the following describes:

(1) The **next action** the SRO directs is to _______.

(2) Following this action the SRO will ______.

A. (1) raise total AFW flow to at least 350 gpm
(2) remain in 1-FR-H.1

B. (1) maintain AFW flow throttled to 65 gpm to each SG
(2) remain in 1-FR-H.1

C. (1) raise total AFW flow to at least 350 gpm
(2) return to 1-ECA-2.1

D. (1) maintain AFW flow throttled to 65 gpm to each SG
(2) return to 1-ECA-2.1

Proposed Answer:

D. (1) maintain AFW flow throttled to 65 gpm to each SG;

(2) return to 1-ECA-2.1.

Explanation: With the uncontrolled depressurization of all S/Gs, the team is directed to throttle total AFW flow to 60 gpm [100 gpm] to minimize the cooldown stress on the plant. Per rules of usage, once the red-path FR-H.1 is identified, the team transitions to that procedure. Once there, a CAUTION states that if total feed flow has been reduced due to operator action, FR-H.1 should not be performed. Therefore, the team returns to the procedure and step in effect – 1-ECA-2.1.

Technical Reference: 1-ECA-2.1 – Uncontrolled Depressurization of All Steam Generators, rev. 41;

1-FR-H.1 – Response to Loss of Secondary Heat Sink, rev. 37

Reference Provided to Applicant: NO

Learning Objective: ND-95.2-LP-3 – Over Cooling Transients; Objective A. Break down the expected general plant responses to secondary break transients.

Question Source: Modified Bank (EOP0397 – Surry ILT Bank)

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43.5 / 45.13)

Comments:

<u>KA Match Analysis</u>: The question meets the KA in that the candidate must adhere to the appropriate procedure flowpath in response to an uncontrolled depressurization of all S/Gs.

<u>Distractor Analysis</u>:

A. Incorrect. 1) Incorrect. ECA-2.1 directs maintaining > 60 gpm to each S/G. Plausible because FR-H.1 will direct throttling AFW to 350 gpm. 2) Incorrect. Caution at beginning of FR-H.1 directs return to procedure and step in effect if AFW was intentionally throttled. Plausible because normally once A “Red path” procedure is entered, the procedure is normally performed to completion.
B. Incorrect. 1) Correct. 2) Incorrect. Caution at beginning of FR-H.1 directs return to procedure and step in effect if AFW was intentionally throttled. Plausible because normally once A “Red path” procedure is entered, the procedure is normally performed to completion.
C. Incorrect. 1) Incorrect. ECA-2.1 directs maintaining > 60 gpm to each S/G. Plausible because FR-H.1 will direct throttling AFW to 350 gpm. 2) Correct.
D. CORRECT.

(EOP0397 – Surry ILT Bank)

Original question:

Given the following conditions:

- *Unit-1 tripped from 100% power due to a lightning strike.*
- *At least one safety valve is stuck open on each S/G.*
- *Maintenance is attempting to gag the SG safety valves.*
- *The crew is performing 1-ECA-2.1, Uncontrolled Depressurization of all SGs.*
- *The BOP throttled AFW flow to 100 gpm to each SG.*
- *SG narrow-range levels are off-scale low.*
- *The STA reports that a red-path exists on the Heat Sink CSF status tree.*

*Which ONE of the following describes the **correct actions** and **implementation of procedures** for these conditions?*

A. *Transition to 1-FR-H.1, Response to Loss of Secondary Heat Sink, and increase AFW flow to at least 340 gpm total until one SG narrow-range level is at least 11%;*

Do NOT return to 1-ECA-2.1.

B. *Transition to 1-FR-H.1, Response to Loss of Secondary Heat Sink, but maintain AFW flow throttled to 100 gpm to each SG;*

Do NOT return to 1-ECA-2.1.

C. *Increase AFW flow to at least 340 gpm total to clear heat sink red-path and continue with 1-ECA-2.1;*

1-FR-H.1, Response to Loss of Secondary Heat Sink, should NOT be used.

D. *Transition to 1-FR-H.1, Response to Loss of Secondary Heat Sink, but maintain AFW flow throttled to 100 gpm to each SG;*

Read the CAUTION, then return to 1-ECA-2.1.

Answer: D

Reference: ECA-2.1 step 2

NUMBER	PROCEDURE TITLE	REVISION
1-ECA-2.1	UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	41
		PAGE 3 of 30

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: A minimum of 60 gpm [100 gpm] feed flow must be maintained to each SG with a narrow range level less than 12% [18%].

* *

NOTE: Shutdown Margin should be monitored during RCS cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. ___	CONTROL FEED FLOW TO MINIMIZE RCS COOLDOWN:	
	☐ a) Check cooldown rate in RCS cold legs - LESS THAN 100°F/hr	☐ a) Lower feed flow to 60 gpm [100 gpm] to each SG. GO TO Step 2c.

NUMBER	PROCEDURE TITLE	REVISION
1-FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	37
		PAGE 2 of 22

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- If total feed flow is less than 350 gpm [450 gpm] due to operator action, this procedure should NOT be performed.
- Feed flow should NOT be reestablished to any faulted SG if a non-faulted SG is available.

* *

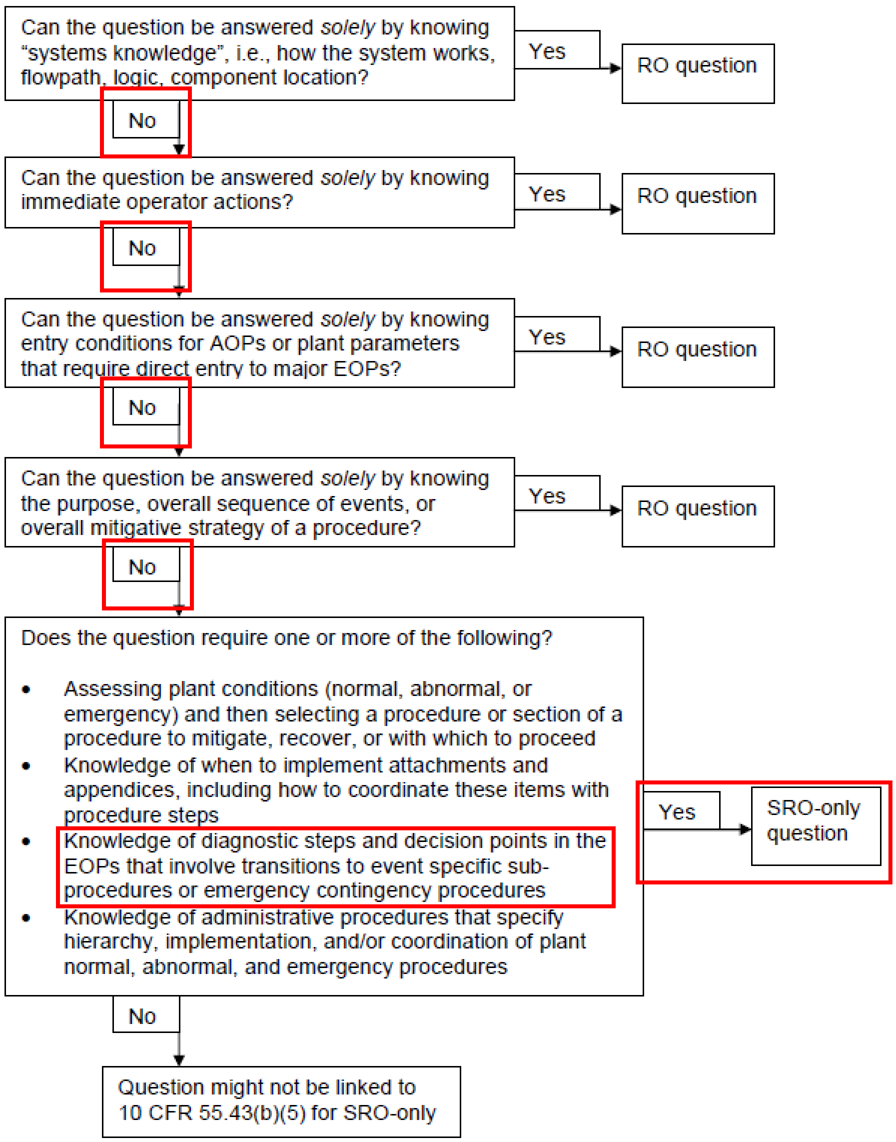
Reference FR-H.1 step 3.

NUMBER	PROCEDURE TITLE	REVISION
1-FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	37
		PAGE 4 of 22

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	TRY TO ESTABLISH AFW FLOW TO AT LEAST ONE SG: (Continued)	
	d) Check AFW - AVAILABLE	☐ d) IF minimum feed flow NOT established, THEN do the following:
	☐ 1) Start AFW Pumps	☐ 1) Stop ALL RCPs.
	☐ 2) Check total flow to SGs - GREATER THAN 350 GPM [450 GPM]	☐ 2) IF desired to transfer EDG 3 to Bus 2J to restore power to 2-FW-P-3B, THEN GO TO Attachment 2.

SRO Justification:

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



K/A Number: WE13EA2.1, Steam Generator Over-pressure/4, Ability to determine and interpret the following as they apply to the (Steam Generator Overpressure), Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Level: SRO Tier #: 1 Group #: 2 IR – RO: 2.9 IR-SRO: 3.4

Proposed Question: 100

Initial Conditions:

- Unit 2 operating at 100% power.
- Reactor trip caused by Loss of All AC.
- The Team has restored power to an Emergency bus, and has transitioned to 2-ES-0.2, Natural Circulation Cooldown.

Current Conditions:

- SG NR Levels: "A" - 68%, "B" - 65%, "C" – 63%.
- SG Pressures: "A" – 1150 psig, "B" – 1090 psig, "C" – 1050 psig.

Which ONE of the following identifies:

1) The Highest Priority procedure used for these conditions?
2) The expected Team action(s)?

A. 1) 2-FR-H.2, Response to Steam Generator Overpressure.
   2) Locally operate SG PORV to reduce pressure.

B. 1) 2-FR-H.2, Response to Steam Generator Overpressure.
   2) Align Blowdown to lower SG Level.

C. 1) 2-FR-H.3, Response to Steam Generator High Level.
   2) Locally operate SG PORV to reduce pressure.

D. 1) 2-FR-H.3, Response to Steam Generator High Level.
   2) Align Blowdown to lower SG Level.

Proposed Answer: A.

Explanation: Highest Priority procedure would be 2-FR-H.2; this procedure is identified first in the Heat Sink Status Tree. With SG level in 'A' SG less than 93%, 2-FR-H.2 would direct to try to dump steam via the SG PORV (preferred). Only if SG NR level is >93% is the implementer sent to 2-FR-H.3, where feed sources are isolated and blowdown is aligned to lower level.

Technical Reference: 2-FR-H.2, Rev. 8. 2-FR-H.3, Rev. 13. Heat Sink Status Tree, F-3, Rev. 5.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-41, FR-H.1, Objective A, Given a simulated plant condition requiring the use of the Critical Safety Function Status Trees, transition through the Heat Sink status tree denoting, in accordance with the rules of priority, any applicable Function Restoration Procedure needing implementation. ND-95.3-LP-42, FR-H.2 Response to Steam Generator Overpressure, Objective B, Given a copy of FR-H.2, Response to Steam Generator Overpressure, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition. ND-95.3-LP-43, Response to Steam Generator High Level, Objective B, Given a copy of FR-H.3, Response To Steam Generator High Level, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: Modified (Harris, 2009, Question 10, significantly modified).

Question History: Last NRC 2 Exams: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43.5 / 45.13)

Comments:

K/A Match Analysis: Question matches K/A, Candidate must determine based on conditions provided which Yellow Path FR procedure would be used for the established conditions, and what action would be taken based on knowledge of the procedure.

Distractor Analysis:

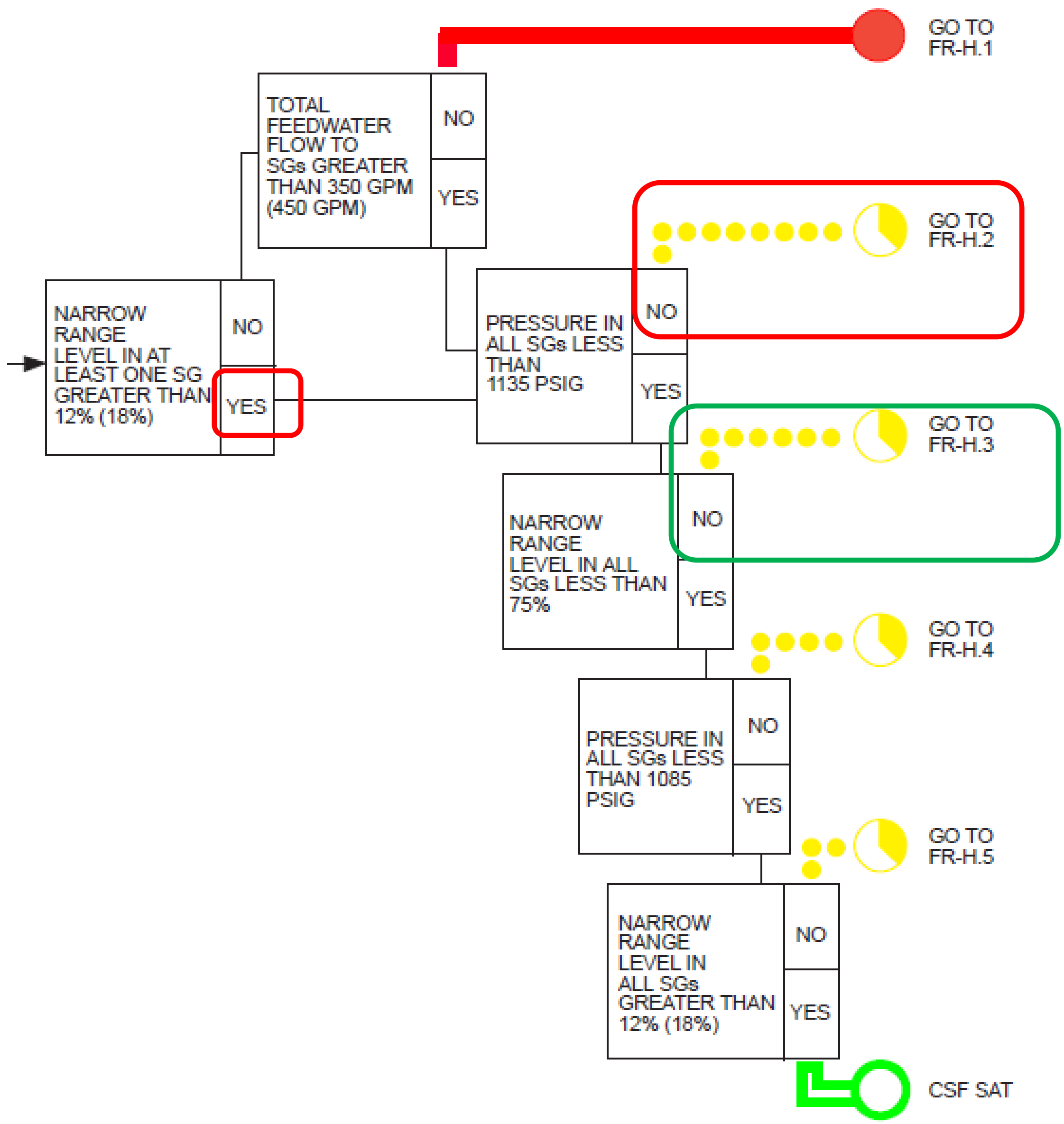
A. Correct. Both Parts 1) and 2) are correct.

B. Incorrect. Part 1) is correct, 2-FR-H.2 is the first Yellow Path identified on the CSFST. Part 2) is incorrect because level is < 93%. Plausible if candidate confuses threshold level for operation of SG PORVs. If affected SG level was > 93% SG PORV operation would be disallowed and blowdown would be used to lower SG level using 2-FR-H.3.

C. Incorrect. Part 1) is incorrect, 2-FR-H.2 is the higher priority procedure, and would send the implementer to the high SG level procedure (2-FR-H.3) if required. Plausible if the Candidate assumes the high level condition must be addressed before the high pressure condition. Part 2) is correct.

D. Incorrect. Part 1) is incorrect. 2-FR-H.2 is the higher priority procedure, and would send the implementer to the high SG level procedure (2-FR-H.3) if required. Plausible if the Candidate assumes the high level condition must be addressed before the high pressure condition. Part 2) is incorrect because level is < 93%. Plausible if candidate confuses threshold level for operation of SG PORVs. If affected SG level was > 93% SG PORV operation would be disallowed and blowdown would be used to lower SG level using 2-FR-H.3.

Number:	Title:	Revision:
F-3	HEAT SINK	5



SNSOC CHAIRMAN

DATE

Graphics No. CB381

NUMBER	PROCEDURE TITLE	REVISION
2-FR-H.2	RESPONSE TO STEAM GENERATOR OVERPRESSURE	8
		PAGE 2 of 3

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Throughout this procedure, AFFECTED refers to any SG in which pressure is greater than 1135 psig.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	IDENTIFY AFFECTED SG(s):	☐ RETURN TO procedure and step in effect.
	☐ • Any SG pressure - GREATER THAN 1135 PSIG	
2.___	VERIFY MFW ISOLATION TO AFFECTED SG(s):	☐ Manually close valves.
	☐ a) Feed REG valves - CLOSED	
	☐ b) SG FW bypass flow valves - CLOSED	
	☐ c) SG FW isolation MOVs - CLOSED	
	d) Locally close feed reg bypass valve manual isolation valve(s):	
	☐ • 2-FW-26 for SG A	
	☐ • 2-FW-57 for SG B	
	☐ • 2-FW-88 for SG C	
3.___	CHECK AFFECTED SG(s) NARROW RANGE LEVEL - LESS THAN 93% [82%]	☐ GO TO 2-FR-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.___ TRY TO DUMP STEAM FROM AFFECTED SG(s): ☐ • Manually use SG PORV(s) OR ☐ • Locally use SG PORV(s) IAW Attachment 1 OR ☐ • Condenser steam dump OR ☐ • Steam supply to TD AFW pump	☐ GO TO Step 6.

NUMBER	PROCEDURE TITLE	REVISION
2-FR-H.3	RESPONSE TO STEAM GENERATOR HIGH LEVEL	13
		PAGE 2 of 4

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: If SG narrow range level has increased to greater than 93% [82%], an evaluation should be made for SG overfill considerations. Steam should NOT be released from any SG with level greater than 93% [82%] before overfill evaluation.

* *

NOTE: Throughout this procedure, AFFECTED refers to any SG in which narrow range level is greater than 75%

9.___ ESTABLISH BLOWDOWN FROM AFFECTED SG(s) IAW 2-OP-BD-001, STEAM GENERATOR BLOWDOWN SYSTEM OPERATION

NUMBER	PROCEDURE TITLE	REVISION
2-FR-H.2	RESPONSE TO STEAM GENERATOR OVERPRESSURE	8
		PAGE 2 of 3

4.___ TRY TO DUMP STEAM FROM AFFECTED SG(s): ☐ • Manually use SG PORV(s) OR ☐ • Locally use SG PORV(s) IAW Attachment 1 OR ☐ • Condenser steam dump OR ☐ • Steam supply to TD AFW pump	☐ GO TO Step 6.

From Harris, 2009, Question 10

Given the following plant conditions:
- The USCO is evaluating FRPs for implementation.
- Containment pressure is 0.8 psig
- The following Steam Generator conditions exist:
  - 'A' SG Pressure = 1175 psig
  - 'A' SG Level = 79%
  - 'B' SG Pressure = 1235 psig
  - 'B' SG Level = 65%
  - 'C' SG Pressure = 1100 psig
  - 'C' SG Level = 23%

Which ONE of the following identifies the FRP that must be addressed first in accordance with EOP-CSFST, Critical Safety Function Status Trees?

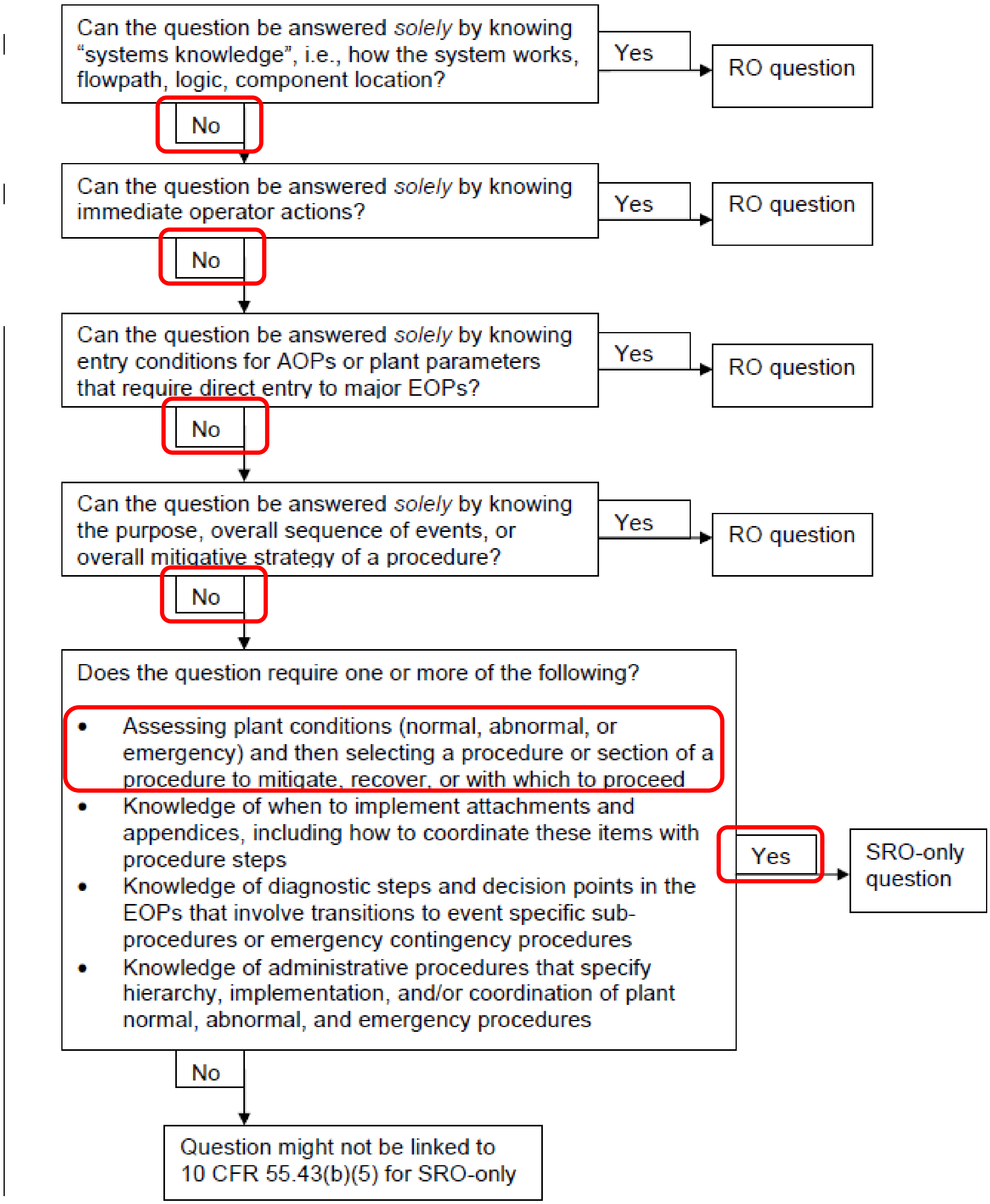
A✓ FRP-H.2, Response to Steam Generator Overpressure

B. FRP-H.3, Response to Steam Generator High Level

C. FRP-H.4, Response to Loss of Normal Steam Release Capability

D. FRP-H.5 Response to Steam Generator Low Level

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



Scenario #1

Facility: Surry Power Station   Scenario No.: 1   Op-Test No.: 2016-001
Examiners: ____________________   Operators: ____________________
____________________   ____________________
____________________   ____________________

Initial Conditions: Unit 1 and 2 at 100% power; MOL, 760 ppm boron.

- Controlling channels have been shifted to CH IV in reparation for Channel III testing.
- 1-SI-P-1B, "B" LHSI Pump, Tagged out for breaker PMs.
- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days

Turnover: The Team will pre-brief the shift of the running CH Pump SW pumps IAW 1-OP-51.5.3, Section 6.1 prior to Simulator Entry.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N(BOP/SRO)	Team will pre-brief and shift the running CH pump SW pumps IAW 1-OP-51.5.3, Section 6.1 ("B" to Hand, "A" to Auto)
2	MS0806, -1	I (BOP/SRO) TS (SRO)	SG C STM FLOW TRNSMTR MS-FT-495 Fails Low (AP-53.00) **(CT-1)**
3	NI1004, +1	I (RO/SRO) TS (SRO) N (BOP)	Power Range CHNL N44 Fails High (AP-53.00, AP-4.00)
4	CH2101, +1	I (RO/SRO)	VCT Level Transmitter, LT-1112 Fails High (AP-53.00)
5	SD0202, TRUE	C (BOP/SRO	Loss of 1-SD-P-1B, "B" HPD (AP-18.00) **(CT-2)**
6	N/A	R (RO/SRO) N (BOP)	Downpower Ramp to <75% power (AP-23.00)

7	MS0102, 20% MS22,23, 24 RC5601, 5602,5603 RD17 TRUE FW09 TRUE FW48 TRUE FW49 TRUE	M (All)	B SG Steam Line Break at Header in Turbine Building (E-0, E-2) Disable A,B,C MSTV Auto closure on HSF **(CT-4)** RCP Breakers trip. Reactor Trip Fail, (FR-S.1) **(CT-3)** FW-P-2 Trip/Throttle Valve Trips Closed Disable AFWP3A Auto Start Disable AFWP3B Auto Start
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor			

Event 1: Shift Running CH Pump SW Pumps IAW 1-OP-51.5.3, Section 6.1. (N- BOP/SRO)

Normal Evolution for shifting CH pump SW pumps; Team will pre-brief evolution prior to Simulator Entry and be given a copy of 1-OP-51.5.3 for use following Simulator Entry. 1-SW-P-10A, “A” CH Pump SW pump initially running in Hand, 1-SW-P-10B, “B” CH pump SW pump, secured in Auto. BOP will start “B” CH pump SW pump in hand, then secure the “A” CH pump SW pump and place in Auto. RO will perform peer check of component manipulations and assist in flow verification using the PCS. SRO will perform oversight of evolution.

Verifiable Action(s):

1) BOP – Operate CH pump SW pump control switches.

Technical Specifications, TRM Actions, Reg. Guide 1.97, VPAP-2802, Reportability, and EP-AA-303, Equipment Important to Emergency Response:

None.

Event 2: “C” SG CH IV Steam Flow Fails Low. (I – BOP/SRO, TS – SRO).

When the Evaluation Team is ready, the malfunction for the failure is implemented. The BOP will diagnose the failure based on alarms and indications received and perform the Immediate Action Steps of 0-AP-53.00, Loss of Vital Instrumentation/Controls. The “C” FRV will be placed in manual and adjusted to match steam and feed flow in “C” SG, and return “C” SG level to program. The RO will swap the controlling Steam Flow Channel for “C” FRV to Channel III. The BOP will return “C” FRV to auto.

Verifiable Actions(s):

1) BOP – Place the “C” FRV in manual and raise feed flow to match steam flow (**CT-1**).
2) RO – Swap SF for “C” FRV to CH III.
3) BOP – Return “C” FRV to Auto.

Critical Task CT-1: If the BOP fails to take timely action in response to the SF channel failure, an automatic reactor trip on SG NR low level will occur; an unanticipated reactor trip should be considered as failure criteria. An Operations representative will be available for consultation as necessary.

Technical Specifications:

1) TS Table 3.7-1, Item 17, Operator Action 6. Place the Inoperable Channel in trip within 72 hours; inoperable channel may be bypassed for 12 hours for surveillance testing; if conditions not satisfied, be in HOT SHUTDOWN in 6 hours.
2) TS Table 3.7-2, Item 1.e, Operator Action 20. Place the Inoperable Channel in trip within 72 hours; inoperable channel may be bypassed for 12 hours for surveillance testing; if conditions not satisfied, be in HSD in 6 hours and be less than 350°/450 psig in the following 12 hours.
3) TS Table 3.7-3, Item 2.a, See item #1.e Table 3/7-2 for operability requirements.

TRM Actions:

1) None.

Reg. Guide 1.97:

1) Main Steam Flow, 1-MS-Loop-F-1495, D-19 Variable, 1 CH per SG required. **(RG 1.97 check is an STA/SM function, recommend evaluation of RG requirements as follow-up question Post-Scenario, at Evaluator discretion.)**

VPAP-2802, Reportability

None.

Equipment Important to Emergency Response (EP-AA-303).

No Impact.

Event 3: Power Range Channel N44 Fails High. (I – RO/SRO, N – BOP)

When the Evaluating Team is ready, the malfunction is implemented. The failure will cause N44 indication to fail high and "D" bank control rods to step in at 72 steps per minute. The RO will diagnose the failure based on alarms and indications received and place rod control in manual to stop inward rod motion. The Team will respond by implementing 0-AP-53.00, Loss of Vital Instrumentation/Controls and transition to 1-AP-4.00, Nuclear Instrumentation Malfunction.

Verifiable Action(s):

1) RO – Place Rod Control in manual to stop inward rod motion.
2) BOP – Perform Control Room Actions to remove N-44 from service IAW 1-AP-4.00, Attachment 1.

Technical Specifications:

1) TS Table 3.7-1, Item 2, Operator Action 2. Place the inoperable channel in trip in 72 hours; channel may be bypassed for 12 hours for surveillance testing; either reduce reactor power to ≤ 75% and reduce neutron flux trip setpoint ≤ 85% of rated power in 78 hours, or monitor **Quadrant Power Tilt once per 12 hours** (more limiting than TS 3.12 D once per day requirement – see 4) below.).
2) TS Table 3.7-1, Item 5 (6), Operator Action 6. Place the inoperable channel in trip in 72 hours; channel may be bypassed for 12 hours for surveillance testing; if conditions not satisfied, be in HSD in 6 hours.
3) TS Table 3.7-1, item 20, Operator Action 13. Not Applicable, Minimum Number of Operable Channels requirement is met for P-7, P-8, and P-10.
4) TS 3.12.D. Quadrant Power Tilt. Item 1, if reactor operating above 75% with one excore channel out of service, QPTR shall be determined once per day or more than 30 inches of control rod motion. Item 2, QPTR is determined using movable detectors (2 per quadrant).

Technical Requirements Manual.
No requirement.
Reg. Guide 1.97:
No requirement.
VPAP-2802, Reportability:
No requirement.
Equipment Important to Emergency Response (EP-AA-303).
No required Action. Att. 6, Reactor Power Level Determination, adequate alternate indications available. **This is an STA/SM function, recommend evaluation of EP-AA-300 requirements as follow-up question Post-Scenario, at Evaluator discretion.**

Event #4: **VCT Level Transmitter, LT-1112 Fails High** (I – RO/SRO)
When the Evaluating Team is ready, the malfunction is implemented. VCT Level Transmitter LT-1112 Fails high, causing 1-CH-LCV-1115A to fail to Divert position. Team will respond IAW 0-AP-53.00, Loss of Vital Instrumentation/Controls, and manually position 1-LCV-1115A to VCT positon.
Verifiable Actions:
1) Manually place 1-CH-LCV-1115A to the VCT position.
2) Manually control 1-CH-LCV-1115A to control VCT level.

Technical Specifications/Technical Requirements Manual:
None.

Event 5: **Loss of 1-SD-P-1B, "B" HPD pump*.* (**C – BOP/SRO)
When the Evaluating Team is ready, the failure is implemented to trip the running HPD pump. The BOP will identify the failure using alarms and indications received. The Team will respond IAW 1-AP-18.00, Loss of HP Heater Drain Pump / Network 90 Failure. 1-AP-18.00 will require a power reduction to < 75% to allow return of the Polishing Building to service (Event 6).
Verifiable Action(s):
1) BOP – Start third Condensate Pump as required for plant conditions (SF/FF mismatch).
2) BOP – reduce Turbine load by 50 MW using Valve Position Limiter or Turbine Manual to control reactor power less than 100%. Further reductions may be required to maintain Loop ΔTs less than 100%.

Critical Task CT-2: Start an additional Condensate pump and verify feed flow returns to normal before a Steam Generator Level Reactor Trip. (Scenario specific CT)

Event #6: **Ramp of Unit to < 75% power.** (R – RO/SRO, N - BOP)
1-AP-18.00, Step 5, will require the Team to reduce reactor power to less than 75% IAW 0-AP-23.00, Rapid Load Reduction. The Team will perform a short brief and commence a ramp at 1% per minute.
Verifiable Actions:
1) RO - Move control rods as necessary to control Tave and Delta Flux during the power reduction.
2) RO – Emergency Borate for ~ 25 seconds to begin the Tave reduction, and then manipulate the CVCS Blender controls to begin a manual borate to add the required amount of Boric Acid for the power reduction.
3) BOP – manipulate Turbine Controls to begin the power reduction.

VPAP-2802, Reportability:
None.

Event #7: **Steam Break in Turbine Building, Reactor Trip Failure, FR-S.1 (ATWS)*.*** (M – All).
After the Evaluating Team is ready for the next event, the malfunction is implemented for a Steam Break on the "B" Steam line in the Turbine Building; the steam break will cause an automatic trip of the MFPs with the TDAFW pump trip throttle valve tripping on startup, and the failure of the MDAFW pumps to auto start; leading to a loss of Feed ATWS. The Team will respond based upon alarms and indications received, attempt to trip the reactor and transition to FR-S.1, Response to Nuclear Generation/ATWS. After the reactor is subcritical, the team will transition from FR-S.1 to E-0. In E-0 the Team will be directed to take action to isolate steam and feed to the faulted SG. Isolation of steam flow is complicated by the failure of all MSTVs to auto trip on the HSF SI. Steam flow is isolated by closing the A, B, and C Main Steam Trip valves

Verifiable Actions:

1) RO – Perform Immediate Actions of FR-S.1:
   a. Place Rod Control in Automatic on discovery that Reactor will not manually trip.
   b. Manually Trip the Main Turbine.
   c. Place Rod Control in Manual and insert control rods at 48 steps per minute when Rod Speed lowers shortly following the Turbine Trip.
   d. Close Main Steam Trip Valves; 1-MS-TV-101A, 101B, and 101C.
   e. Close 1-FW-MOV-151C/D to isolate AFW to the Faulted SG.
2) BOP – Manually start AFW pumps.

Critical Task:

CT-3: Manually start AFW pumps as needed before SG WR level decreases to <7% on both intact S/Gs.

Safety Significance: Following the steam break outside containment, a High Steam Flow SI will occur which will trip all Feed pumps. All S/G levels will lower rapidly and with no operator action all S/Gs will dry out (<7%) in approximately 5 minutes. The Turbine Drive Auxiliary Feed pump will auto start but will trip shortly afterwards due to a failure in its trip throttle valve. The Motor Driven Auxiliary Feed pumps will fail to automatically start which will require the operator to manually start them.

CT-4: Close Main Steam Trip valves prior to reaching an orange path on Integrity (FR-P.1). Critical Task **begins** when the Main Steamline rupture occurs and **ends** when the Main Steam Trip valves are closed.

The Scenario is terminated on Evaluating Team Cue, and Main Steam trip valves closed.

Initial Conditions: Initial Conditions: Unit 1 and 2 are at 100% power, MOL. The unit has been at 100% power for > 30 days.

Turnover: The Team will brief the shift of the running Ch Pump SW pumps IAW 1-OP-51.5.3, Section 6.1 prior to Simulator entry.

Pre-load malfunctions: (Trigger 30's)

- RD17, ATWS WITH MANUAL RX TRIP PB DEFEATED.
- FW48, DISABLE AFWP3A AUTO START.
- FW49, DISABLE AFWP3B AUTO START.
- FP0301, PACP07 ALARM HORN FAILURE
- FP0302, FPS PC SPEAKER FAILURE
- MS22, DISABLE A MSTV AUTO CLOSURE ON HSF.
- MS23, DISABLE B MSTV AUTO CLOSURE ON HSF.
- MS24, DISABLE C MSTV AUTO CLOSURE ON HSF.
- SIP1B_BKRPOS, SI-P-1B BKR 14J3 CUBICLE POSITION

Equipment Status/Procedures/Alignments/Data Sheets/etc.:

Unit 1 is at 100% power. All systems and crossties are operable with the following exception:

- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.
- 1-SI-P-1B, "B" LHSI pump tagged out for breaker PMs.
- Controlling channels have been shifted to CH IV in reparation for Channel III testing

Turnover:

- The Team will brief the shift of the running Ch Pump SW pumps IAW 1-OP-51.5.3, Section 6.1 prior to Simulator entry.

Turnover: The performance of this procedure has been analyzed based on the current plant configurations and the PSA indicates green.

Event	Malf. #'s	Severity	Instructor Notes and Required Feedback
1	NA	NA	Team will shift the running CH pump SW pumps IAW 1-OP-51.5.3, Section 6.1.
2	MS0803	-1	S/G "C" Steam Flow Transmitter, MS-FT-495 fails low. (AP-53.00) **(CT-1)**
3	NI1004	+1	Power Range channel N44 fails high. (AP-53.00, AP-4.00)
4	CH2101	+1	VCT Level transmitter, LT-1112 fails high. (AP-53.00)
5	SD0202	TRUE	Loss of "B" HPD pump, 1-SD-P-1B. (AP-18.00) **(CT-2)**
6	NA	NA	Ramp to 75% power. (AP-23.00)

7.	MS0102	20%	"B" Steam line break at the header in the Turbine Building. (E-0)
	MS22-24	TRUE	Main Steam Trip Valves fail to trip on HSF SI.
	RD17	TRUE	Auto and Manual Reactor trip fails from MCR (1-FR-S.1)
	FW48,49	TRUE	Both MDAFW pumps fail to auto start. **(CT-3)**
	RC5601-3	TRUE	RCPs spuriously trip. **(CT-4)**

SHIFT TURNOVER INFORMATION

OPERATING PLAN:

The initial conditions have Unit 1 at 100% power with RCS boron concentration of 760 ppm.

Unit conditions have been stable at approximately 100% power since the last refueling outage.

All systems and crossties are operable with the following exception:

- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.
- 1-SI-P-1B, "B" LHSI Pump, Tagged out for breaker PMs. Tech Spec 3.3.B.3 requires return to service within 72 hours. Expected completion of breaker PMs-later this shift.
- Controlling channels have been shifted to CH IV in preparation for Channel III testing

Unit #2 is at 100% power with all systems and crossties operable.

The last shift performed two 30 gallon dilutions, followed by manual makeups.

Shift orders are to maintain 100% power on Unit 1, and shift the running CH Pump SW pumps IAW 1-OP-51.5.3, Section 6.1.

PWR Scenario :

Scenario Objectives: List Each event as a single objective.

A. Given a SG "C" Steam flow transmitter failure, respond in accordance with 0-AP-53.00, Loss of Vital Instrumentation/Controls to place "C" FRV in manual, control "C" SG level in manual, and return "C" FRV to automatic.
B. Given a Power range N44 failure, respond in accordance with 0-AP-53.00, Loss of Vital Instrumentation/Controls, and 1-AP-4.00 to stop rod motion and remove N-44 from service.
C. Given a failure of VCT Level transmitter, LT-1112, respond in accordance with 0-AP-53.00, Loss of Vital Instrumentation/Controls and return VCT level to normal.
D. Given a loss of 1-SD-P-1B, "B" HPD pump, respond in accordance with 1-AP-18.00 and return feed flow to normal.
E. Demonstrate the ability to ramp the unit in accordance with 0-AP-23.00, Rapid Load Reduction.
F. Given a Main steamline break with Reactor protection system failures, respond in accordance with FR-S.1, ATWS; and 1-E-0, Reactor trip or SI to shutdown and stabilize the plant.

Scenario Sequence

Event One: Shift Running CH Pump SW Pumps IAW 1-OP-51.5.3, Section 6.1

Normal Evolution for shifting CH pump SW pumps; Team will pre-brief evolution prior to Simulator Entry and be given a copy of 1-OP-51.5.3 for use following Simulator Entry. 1-SW-P-10A, "A" CH Pump SW pump initially running in Hand, 1-SW-P-10B, "B" CH pump SW pump, secured in Auto. BOP will start "B" CH pump SW pump in hand, then secure the "A" CH pump SW pump and place in Auto. RO will perform peer check of component manipulations and assist in flow verification using the PCS. SRO will perform oversight of evolution.

Malfunctions required: None

Objectives: (BOP) Start "B" CH pump SW pump in hand, then secure the "A" CH pump SW pump and place in Auto.

Success Path: "B" CH pump SW pump, "A" CH pump SW pump secured and in AUTO.

Event Two: "C" SG CH IV Steam Flow Fails Low

When the Evaluation Team is ready, the malfunction for the failure is implemented. The BOP will diagnose the failure based on alarms and indications received and perform the Immediate Action Steps of 0-AP-53.00, Loss of Vital Instrumentation/Controls. The "C" FRV will be placed in manual and adjusted to match steam and feed flow in "C" SG, and return "C" SG level to program. The RO will swap the controlling Steam Flow Channel for "C" FRV to Channel III. The BOP will return "C" FRV to auto.

Malfunctions required: One MS0602

Objectives: (BOP) Place the “C” FRV in manual and raise feed flow to match steam flow.

(RO) Swap Steam flow channels for “C” FRV to CH III.

(SRO) Direct actions per 0-AP-53.00. Review Tech specs and identify 72 hour clock to place failed channel in TRIP. (Table 3.7-1, item 17; Table 3.7-2, item 1e)

Success Path: Perform Immediate Actions of 0-AP-53.00, assume manual control of “C” SG NR level, swap controlling channel of Steam Flow input to SGWLC system, and return “C” SG FRV to automatic.

Event Three: Power Range Channel N44 Fails High

When the Evaluating Team is ready, the malfunction is implemented. The failure will cause N44 indication to fail high and “D” bank control rods to step in at 72 steps per minute. The RO will diagnose the failure based on alarms and indications received and place rod control in manual to stop inward rod motion. The Team will respond by implementing 0-AP-53.00, Loss of Vital Instrumentation/Controls and transition to 1-AP-4.00, Nuclear Instrumentation Malfunction.

Malfunctions required: One NI1004

Objectives: (RO) Place rod control in MANUAL to stop inward rod motion.

(BOP) Perform Control Room Actions to remove N-44 from service IAW 1-AP-4.00, Attachment 1.

(SRO) Direct actions per 0-AP-53.00, and 1-AP-4.00. Review Tech Specs and identify 72 hour clock. (Table 3.7-1 item2, item5)

Success Path: Inward rod motion is stopped, and N-44 is removed from service.

Event Four: VCT Level Transmitter, LT-1112 Fails High

When the Evaluating Team is ready, the malfunction is implemented. VCT Level Transmitter LT-1112 Fails high, causing 1-CH-LCV-1115A to fail to Divert position. Team will respond IAW 0-AP-53.00, Loss of Vital Instrumentation/Controls, and manually position 1-LCV-1115A to VCT positon

Malfunctions required: One, CH2101

Objectives: (RO) Manually place 1-CH-LCV-1115A to the VCT position. Manually control 1-CH-LCV-1115A to control VCT level.

(SRO) Direct actions to recover VCT level per 0-AP-53.00.

Success Path: VCT level and control restored.

Event Five: Loss of 1-SD-P-1B, "B" HPD pump

Malfunctions required: One SD0202

Objectives: (BOP) Start third Condensate Pump as required for plant conditions (SF/FF mismatch). Reduce Turbine load by 50 MW using Valve Position Limiter or Turbine Manual to control reactor power less than 100%. Further reductions may be required to maintain Loop ΔTs less than 100%.

(SRO) Direct actions per 1-AP-18.00

Success Path: Additional Condensate pump started and S/G level returned to normal.

Event Six: Ramp Unit to < 75%.

The SRO will notify the Shift Manager of the requirement per AP-18.00 to ramp the unit below 75%. The SRO will conduct a Team brief to discuss the Reactivity Plan determined by the RO. The SRO will implement 0-AP-23.00, Rapid Load Reduction, to commence power reduction to be 70 – 74% power.

Malfunctions required: None.

Objectives: (RO) Manipulate the CVCS system to Emergency Borate and establish a continuous boration to control RCS Tave during the Turbine Ramp. Operate control rods to adjust delta flux and assist in RCS Tave control.

(BOP) Operate turbine controls to control the load ramp per AP-23.00.

(SRO) Direct actions to lower power in accordance with AP-23.00. Notify SEM of plant status and request Maintenance and Engineering support

Success Path: Maintain Tave/Tref mismatch within 5 ^{o}F of T_{ref}, and delta flux near target value

Event Seven: Steam Break in Turbine Building, Reactor Trip Failure, FR-S.1 (ATWS)

After the Evaluating Team is ready for the next event, the malfunction is implemented for a Steam Break on the "B" Steam line in the Turbine Building; the steam break will cause an automatic trip of the MFPs with the TDAFW pump trip throttle valve tripping on startup, and the failure of the MDAFW pumps to auto start; leading to a loss of Feed ATWS. The Team will respond based upon alarms and indications received, attempt to trip the reactor and transition to FR-S.1, Response to Nuclear Generation/ATWS. After the reactor is subcritical, the team will transition from FR-S.1 to E-0. In E-0 the Team will be directed to take action to isolate steam and feed to the faulted SG. Isolation of steam flow is complicated by the failure of all MSTVs to auto trip on the HSF SI. Steam flow is isolated by closing the A, B, and C Main Steam Trip valves

Malfunctions required: Eleven (11)

- MS0102-, B' Steam line rupture at header.
- MS22, MS23, MS24: MSTV auto closure failure on Hi Steamflow SI.

- FW09, FW-P-2 Trip throttle valve trips closed.
- FW48, FW49; AFW P-3A, and P-3B fail to auto start.
- RC5601, RC5602, RC5603; RCPs spuriously trip due to steam break.
- RD17, ATWS with failure of Rx trip pushbuttons

Objectives: (RO) Perform immediate actions of FR-S.1.

a. Place Rod Control in Automatic on discovery that Reactor will not manually trip.
b. Manually Trip the Main Turbine.
c. Place Rod Control in Manual and insert control rods at 48 steps per minute when Rod Speed lowers shortly following the Turbine Trip.
d. Close Main Steam Trip Valves; 1-MS-TV-101A, 101B, and 101C.
e. Close 1-FW-MOV-151C/D to isolate AFW to the Faulted SG.

(BOP) Manually start AFW pumps.

(SRO) Direct actions using the EOP Network.

Success Path: AFW started, and Main Steam trip valves are tripped. Actions per FR-S.1, and E-0 taken to stabilize the plant.

Scenario Recapitulation

Total Malfunctions:	7
Abnormal Events:	5 (0-AP-53.00, 1-AP-4.00, 0-AP-53.00, 1-AP-18.00, 0-AP-23.00)
Major Transients:	1 (Main Steam line break, ATWS, Failure of AFW and HSF isolation)
EOPs Entered:	1 (E-0,)
EOP Contingencies:	1 (FR-S.1)

Event Description: .Shift Charging Pump SW pumps.

Cue: When team ready.

Time	Position	Applicant's Action or Behavior
	SRO/BOP	1-OP-51.5.3 Team will pre-brief Initial Conditions, Precautions and Limitations, and procedure prior to entering simulator.
	BOP	1-OP-51.5.3, step 6.1.1 6.1.1 IF PCS available, THEN monitor Charging Pump Service Water flow for the expected response. *BOP operates PCS to monitor Charging Pump Service Water.*
	BOP	1-OP-51.5.3, step 6.1.2 6.1.2 Place the pump control mode selector switch for the standby charging pump service water pump in HAND. . *BOP places 1-SW-P-10B, CHG Pump SW pump in HAND.*
	BOP	1-OP-51.5.3, step 6.1.3 6.1.3 Place the pump control mode selector switch for the charging pump service water pump to be put in standby in AUTO *BOP places 1-SW-P-10A, CHG Pump SW pump in AUTO.*
	BOP	1-OP-51.5.3, step 6.1.4 6.1.4 Initiate 0-OP-SW-006, MER 3 and MER 4 Service Water System Chemical Injection Operation, Subsection 5.5, to perform capacity check and bleed line flow verification in order to check proper flow rates to service water header(s) with pump(s) in service. IF this is an evolution of short duration (i.e. testing) AND pumps will remain in their original configuration OR the as-left configuration will leave one pump running on each header, THEN enter N/A. *BOP enters N/A.*
	BOP	1-OP-51.5.3, step 6.1.5 Notify Chemistry of current alignment. *BOP notifies Chemistry of pump alignment.*
		END EVENT 1

Op-Test No.: Surry 2016-1 Scenario No.: 1 Event No.: 2

Event Description: "C" S/G Steam Flow Transmitter FT-495 fails low.

Cue: By Examiner.

Time	Position	Applicant's Action or Behavior
	Team	Steam Flow "C" SG Fail Low Diagnose this failure using the following alarms and indications: Annunciator 1F-C9, STM GEN 1C CH 3 FW < STM FLOW Annunciator 1H-E7, STM GEN 1C FW > < STM FLOW Annunciator 1H-G7, STM GEN 1C LVL Error Indicator 1-MS-FI-1495, failed low on Vertical Board "C" SG NR Level lowering on Vertical Board
	BOP	0-AP-53.00, Loss of Vital Instrumentation/Controls. Perform Immediate Actions of AP-53.00: [1] CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION – NORMAL. Identifies 1-MS-FI-1494 – NORMAL [2] PLACE AFFECTED CONTROL(S)/ COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION **Places "C" FRV in Manual and raises demand**. **Reports to SRO: Immediate Actions of AP-53.00 complete, "A" FRV in manual**. **Critical Task CT-1:** If the BOP fails to take timely action in response to the SF channel failure, an automatic reactor trip on SG NR low level will occur; an unanticipated reactor trip should be considered as failure criteria. *SRO Sets control band, monitoring frequency, and contingency actions.*
	SRO BOP SRO BOP SRO	0-AP-53.00 Reads AP-53.00 Immediate Action Steps: [1] CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION – NORMAL Yes, 1-MS-FI-1494 - Normal. [2] PLACE AFFECTED CONTROL(S)/ COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION. Reports "C" FRV in manual, "C" SG NR level trending to 44%. Establishes control band of 44% ± 5% for "C" SG NR level.

Event Description: “C” S/G Steam Flow Transmitter FT-495 fails low.

Cue: By Examiner.

	SRO	0-AP-53.00 The SRO will lead a transient brief. During the brief, the failure of “C” SG Steam Flow High will be discussed. The RO/BOP will report Annunciators received related to the event, and Critical Parameters affected. STA will have no input for the brief.
	SRO RO	0-AP-53.00 3. CHECK REACTOR POWER – LESS THAN OR EQUAL TO 100% Report reactor power is less than 100%, (provides current reactor power indication.)
	SRO	0-AP-53.00 Notes prior to Step 4. • Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud. • When the affected instrument/controller malfunction(s) has been addressed by this procedure, recovery actions should continue at Step 13. 4. DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE: • SG Steam Flow, Step 7.
	SRO BOP	0-AP-53.00 **CAUTION Prior to Step 7**: When CALCALC is based on Feedwater, changes in feed flow will affect calorimetric power. Reactor power must be monitored when adjusting feed flow. Acknowledges CAUTION.
	SRO BOP SRO	0-AP-53.00 7. CHECK STEAM GENERATOR LEVEL CONTROL INSTRUMENTS – NORMAL • Steam Flow Reports No, 1-MS-FI-1495 Failed Low. Goes to STEP 7 RNO

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Event Description: "C" S/G Steam Flow Transmitter FT-495 fails low.

Cue: By Examiner.

		0-AP-53.00
	SRO	7. RNO
		IF the selected steam flow, steam pressure, or feed flow input to the SG Water Level Control systemhas failed, THEN do the following: a) Place the associated Feed Reg Valve in MANUAL.
	BOP	Reports Yes, "C" FRV in manual.
	SRO	b) Control SG level at program level.
	BOP	Reports Yes.
	SRO	c) Select the redundant channel for affected SG(s)
	RO	**Selects Channel III SF input to "C" SG SGWLC system**.
	SRO	d) WHEN SG level returned to normal, THEN place the Feed Reg Valve in AUTOMATIC.
	BOP	**Returns "C" FRV to automatic when "C" SG NR level at program, and SF/FF are matched**.

Op-Test No.: Surry 2016-1 Scenario No.: 1 Event No.: 2

Event Description: “C” S/G Steam Flow Transmitter FT-495 fails low.

Cue: By Examiner.

	SRO	0-AP-53.00 7. RNO (Continued) Perform follow-up actions: a) Consult with Shift Manager on need to initiate ( )-OP-RP-001, ALIGNING CONTROL SYSTEM FOR PERFORMANCE OF CHANNEL I, II, III, AND IV PROCESS AND PROTECTION TESTING. Recommends to Shift Manager to wait until I&C ready to troubleshoot before shifting channels. b) Refer to the following Tech Spec 3.7 items: **Table 3.7-1**, 12 and **17** **Table 3.7-2**, 1.c, **1.e**, and 3.a **Table 3.7-3, 2.a**, and 3.a Table 3.7-6, 15 and 16 TS Table 3.7-1, Item 17, Low steam generator water level with steam/feedwater flow mismatch, OA 6, Place failed channel in trip within 72 hours, channel may be bypassed for surveillance testing for up to 12 hours, if requirements not met, place the Unit in HSD in 6 hours. TS Table 3.7-2.1.e.1), Operator Action 20, Place failed channel in trip in 72 hours, channel may be bypassed for surveillance testing for up to 12 hours, if requirements not met, place the Unit in HSD in 6 hours, and reduce RCS temperature and pressure <350°/450 psig in the following 12 hours. TS Table 3.7-3.2.a., See Item #1.e Table 3.7-2 for operability requirements.
	SRO	0-AP-53.00 7. RNO (Continued) c) Refer to Attachment 1. SRO hands out Attachment 1 for RO/BOP Review. d) IF no other instrumentation failure exists, THEN GO TO Step 13 SRO Goes to Step 13.
	SRO RO	0-AP-53.00 13. CHECK CALORIMETRIC - FUNCTIONAL IAW (1)-OPT-RX-001, ATTACHMENT 4 Directs RO to Review 1-OPT-RX-001, Attachment 4. Determines Calorimetric unaffected by SF failure

Event Description: “C” S/G Steam Flow Transmitter FT-495 fails low.

Cue: By Examiner.

		0-AP-53.00
	SRO	14. REVIEW THE FOLLOWING: Tech Spec 3.7
		Previously completed (Step 7, Previous Page)
		VPAP-2802, NOTIFICATIONS AND REPORTS
	SRO	Directs STA to review VPAP-2802
		TRM SECTION 3.3, INSTRUMENTATION
		Directs STA to review TRM Section 3.3
		Reg Guide 1.97
		Directs STA to review Reg Guide 1.97
		• EP-AA-303, Equipment Important to Emergency Response
		Directs STA to review EP-AA-303
	STA	Reports that VPAP-2802, TRM Section 3.3, **Reg Guide 1.97**, and EP-AA-303; Results of Review discussed with Shift Manager.
		Reg Guide 1.97: Main Steam Flow, D-19 Variable, 1 Channel per SG is required; review of this procedure is usually accomplished by STA, recommend verification by SRO as follow-up following scenario completion, at Evaluator discretion.
		0-AP-53.00
	SRO	15. CHECK ADDITIONAL INSTRUMENT / CONTROLLER MALFUNCTION – EXISTS
	BOP	Reports No, no additional controller malfunction exists.
	SRO	Goes to Step 17.

Event Description: “C” S/G Steam Flow Transmitter FT-495 fails low.

Cue: By Examiner.

		0-AP-53.00
	SRO	17. PROVIDE NOTIFICATIONS AS NECESSARY: Shift Supervision OMOC STA (PRA determination) I&C
	SRO	Consults with Shift Manager for OMOC notification; request I&C support to place the failed channel in trip; and update for Unit status, AP-53.00 completion, and Tech Spec clocks in effect.
		---END OF EVENT 2---

Event Description: Power Range Channel N44 Fails high.

Cue: When initiated by Team

Time	Position	Applicant's Action or Behavior
	Team	N44 Fail Low Diagnose this failure using the following alarms and indications: Annunciator 1G-E4, NIS PWR RNG CH AVG FLUX DEVIATION Annunciator 1G-G1, NIS PWR RNG HI FLUX ROD STOP Annunciator 1G-E4, NIS PWR RNG CH AVG FLUX DEVIATION Annunciator 1E-H5, NIS PWR RNG HI STPT CH 4 N44 indication on Benchboard and NI Drawer Fail HIGH. Rods Drive IN at 72 Steps/Minute.
	RO	0-AP-53.00, Loss of Vital Instrumentation/Controls. Perform Immediate Actions of AP-53.00: [1] CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION – NORMAL. **Places Rod Control switch in MANUAL to stop rod motion**. Identifies N41, N42, and N43 – NORMAL [2] PLACE AFFECTED CONTROL(S)/ COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION Checks Rod Motion Stopped. **Reports to SRO: Immediate Actions of AP-53.00 complete, N44 failed high, Rod control in Manual**.
	SRO	0-AP-53.00 Reads AP-53.00 Immediate Action Steps: [1] CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION – NORMAL
	RO	Reports N41, N42, N43 Normal.
	SRO	[2] PLACE AFFECTED CONTROL(S)/ COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION.
	RO	Reports Rod Control in Manual to stop inward rod motion.
	SRO	**Note:** *May direct BOP to Secure Turbine Ramp.*

Event Description: Power Range Channel N44 Fails high.

Cue: When initiated by Team

	SRO	0-AP-53.00 The SRO will lead a transient brief. During the brief, the failure of N44 will be discussed. The RO/BOP will report Annunciators received related to the event, and Critical Parameters affected. STA will have no input for the brief.
	SRO RO	0-AP-53.00 3. CHECK REACTOR POWER – LESS THAN OR EQUAL TO 100% Report reactor power is less than 100%, and provides current reactor power indication.
	SRO	0-AP-53.00 Notes prior to Step 4. • Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud. • When the affected instrument/controller malfunction(s) has been addressed by this procedure, recovery actions should continue at Step 13. 4. DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE: • NI Malfunction, 1-AP-4.00 ***SRO Transitions to AP-4.00***.
	SRO	1-AP-4.00, Nuclear Instrument Malfunction SRO will conduct focus brief, changes to parameters or Unit status will be discussed. RO/BOP will provide input for Unit Status change. STA will have no input for the brief. SRO will continue 1-AP-4.00
	SRO RO	1-AP-4.00, Nuclear Instrument Malfunction NOTE Prior to STEP 1: Attachments 6, 7, and 8 show one-line diagrams of Nuclear Instrumentation. Acknowledges Note.

Event Description: Power Range Channel N44 Fails high.

Cue: When initiated by Team

	SRO RO	1-AP-4.00, Nuclear Instrument Malfunction. 1 CHECK NI MALFUNCTION – POWER RANGE FAILURE. Reports Yes, N44 Failed
	SRO RO	1-AP-4.00, Nuclear Instrument Malfunction. 2. STABILIZE UNIT CONDITIONS Reports Yes, Conditions Stable
	SRO RO	1-AP-4.00, Nuclear Instrument Malfunction. 3. CHECK N-44 – FAILED Reports Yes, N44 Failed
	SRO RO	1-AP-4.00, Nuclear Instrument Malfunction. 4. CHECK ROD CONTROL - IN MANUAL Reports Yes, Rod Control in Manual
	SRO RO	1-AP-4.00, Nuclear Instrument Malfunction. CAUTION Prior to Step 5: To prevent operation with delta flux outside of target band, delta flux must be monitored and maintained within band if rods have moved. Acknowledges CAUTION
	SRO BOP	1-AP-4.00, Nuclear Instrument Malfunction. 5. PLACE 1-MS-43-N16, REACTOR POWER SOURCE, IN THE N43 POSITION (SWITCH LOCATED ON NI PROTECTION CHNL III CABINET) Reports Yes, 1-MS-43-N16 in N43 Position.
	SRO RO	1-AP-4.00, Nuclear Instrument Malfunction. 6. CHECK N-43 - FAILED Reports NO, N43 NOT failed SRO GOES to Step 8

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Event Description: Power Range Channel N44 Fails high.

Cue: When initiated by Team

	 SRO RO	1-AP-4.00, Nuclear Instrument Malfunction. 8. CHECK POWER RANGE CHANNELS - ONLY ONE FAILED Reports Yes, only N44 Failed.
	 SRO BOP	1-AP-4.00, Nuclear Instrument Malfunction. NOTE Prior to Step 9: Performance of Attachment 1 to place the failed Power Range Channel in trip requires I&C assistance for N-41, N-42, or N-43. Acknowledges NOTE.
	 SRO	1-AP-4.00, Nuclear Instrument Malfunction. 9. INITIATE ATTACHMENT 1 TO PLACE FAILED CHANNEL IN TRIP WITHIN 72 HOURS Directs BOP to perform 1-AP-4.00, Attachment 1, Part 1, 2, and 3. Attachment 1 actions begin on Page 26.
	 SRO RO	1-AP-4.00, Nuclear Instrument Malfunction. 10. CHECK NI MALFUNCTION – INTERMEDIATE RANGE FAILURE Reports No, Power Range Failure SRO GOES to Step 19
	 SRO RO	1-AP-4.00, Nuclear Instrument Malfunction. 19. CHECK NI MALFUNCTION – SOURCE RANGE FAILURE Reports No, Power Range Failure SRO Goes to Step 38
	 SRO	1-AP-4.00, Nuclear Instrument Malfunction. 38. NOTIFY THE FOLLOWING • Instrument Shop • OM on call

Event Description: Power Range Channel N44 Fails high.

Cue: When initiated by Team

	SRO	SRO Consults Tech Specs and identifies: 1) TS Table 3.7-1, Item 2, Operator Action 2, Place Failed Channel in trip within 72 hours, Inoperable channel may be bypassed up to 12 hours for surveillance testing, Either Thermal Power restricted to ≤ 75% of rated power and Neutron Flux trip setpoint reduced to ≤ 85% of Rated Power within 78 hours; OR the Quadrant Power Tilt monitored at least once per 12 hours. QPT shall be monitored using the in-core detectors. 2) TS-3.12.D, Quadrant Power Tilt. If the reactor is operating above 75% power with one ex-core nuclear channel out of service, the QPT shall be determined once per day, or a change in power level > 10%, or 30 inches of control rod motion.
	SRO	Notifies Shift Manager of Unit status, procedures used, and Tech Spec Requirements. Requests that the Shift Manager notify I&C and the OMOC. **NOTE:** *Rod Control may remain in Manual.*

Event Description: Power Range Channel N44 Fails high.

Cue: When initiated by Team

Time	Position	Applicant's Action or Behavior
	BOP	1-AP-4.00, Attachment 1: ONE POWER RANGE CHANNEL INOPERABLE 1. Perform the following at the NIS panel within 72 hours. • Comparator and Rate Drawer **a. Select the failed channel on the COMPARATOR CHANNEL DEFEAT switch. (N-44)** b. **Check annunciator 1G-E4, NIS PWR RANGE CH AVG FLUX DEVIATION - NOT LIT**. ***Annunciator will be NOT LIT***. • Miscellaneous Control and Indication Panel a. **Select the failed channel on the ROD STOP BYPASS switch. (N-44).** b. Check annunciator 1G-G1, NIS PWR RNG HI FLUX ROD STOP - NOT LIT. **Annunciator will be NOT LIT.** c. **Select the failed channel on the UPPER SECTION defeat switch. (N-44).** d. IF Reactor power greater than 50%, THEN check annunciator 1G-C4, UPPER ION CHAMBER DEVIATION OR AUTO DEFEAT < 50% - NOT LIT. (annunciator will remain LIT if any Power Range channel less than 50%) e. **Select the failed channel on the LOWER SECTION defeat switch. (N-44).** f. IF Reactor power greater than 50%, THEN check annunciator 1G-D4, LOWER ION CHAMBER DEVIATION OR AUTO DEFEAT < 50% - NOT LIT. (annunciator will remain LIT if any Power Range channel less than 50%)
	BOP	1-AP-4.00, Attachment 1: ONE POWER RANGE CHANNEL INOPERABLE NOTE Prior to Step 2: Annunciator NIS PWR RNG HI STPT (1E-E5, 1E-F5, 1E-G5, or 1E-H5) for the channel being placed in trip, NIS PWR RNG LOSS OF DET VOLT (1G-C3), and NIS DROPPED ROD FLUX DECREASE > 5% PER 2 SEC (1G-H1) will alarm when the instrument power fuses are pulled If Reactor power is less than 10%, annunciator NIS PWR RNG LO STPT HI FLUX (1E-D5) will alarm when the instrument power fuses are pulled. Acknowledges NOTE.

Op-Test No.: Surry 2016-1 Scenario No.: 1 Event No.: 3 Page 28 of 97

Event Description: Power Range Channel N44 Fails high.

Cue: When initiated by Team

	BOP	1-AP-4.00, Attachment 1: ONE POWER RANGE CHANNEL INOPERABLE BOP Notifies RO prior to Removing Instrument Fuses (a. Below) 2. Place the failed Power Range channel in trip IAW the following: a. **At the Power Range drawer, remove the INSTRUMENT POWER fuses. (N-44).** b. **At the Power Range drawer, put the POWER RANGE TEST switch in the TEST position. (N-44).** c. Check annunciator 1G-H1, NIS DROPPED ROD FLUX DECREASE > 5% PER 2 SEC - LIT. **Annunciator will be LIT.** d. Check annunciator 1G-C3, NIS PWR RNG LOSS OF DET VOLT - LIT. **Annunciator will be LIT.** e. IF Reactor power less than 10%, THEN check annunciator 1E-D5, NIS PWR RNG LO STPT HI FLUX - LIT. **Annunciator will not be NOT LIT.**
	BOP	1-AP-4.00, Attachment 1: ONE POWER RANGE CHANNEL INOPERABLE 3. Remove the following PCS points for the failed channel from scan: • N-41, N0041A and N0042A • N-42, N0043A and N0044A • N-43, N0045A and N0046A **• N-44, N0047A and N0048A** **The BOP will remove these points from scan at the Shift Manager PCS terminal**. ***Only N-44 points (in BOLD Above) will be taken off scan***.
		BOP will return Attachment to SRO and report Parts 1, 2, and 3 complete
		--- END OF EVENT 3 ---

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Event Description: VCT Level Transmitter, LT-1112 Fails High

Cue: By Examiner.

Time	Position	Applicant's Action or Behavior
	RO	Diagnoses failure 1-CH-LT-1112 with the following indications/alarms: Indications: • Upscale indication on VCT level on channel 1-CH-LI-1112. • Lowering indication on VCT level on channel 1-CH-LI-1115. • VCT Level Divert valve, 1-CH-LCV-1115A diverting to PDT. • 1-CH-LC-1112C, VCT LEVEL CNTRL showing 100% Demand.
	SRO	0-AP-53.00 Enters 0-AP-53.00 LOSS OF VITAL INSTRUMENTATION / CONTROLS
	RO	0-AP-53.00 [1] CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION - NORMAL RO identifies Channel LI-1115 VCT Level indication is NORMAL.
	RO	0-AP-53.00 [2] PLACE AFFECTED CONTROL(S)/COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION **RO places 1-CH-LCV-1115A to NORM (VCT), OR places 1-CH-LC-1112C to MAN and LOWER to 0% demand to shift 1-CH-LCV-1115A to VCT.** Monitors VCT level to confirm VCT is stabilizing.
	SRO	0-AP-53.00 Conducts a Brief summarizing the Event and Establish priorities. RO will provide Critical Parameters using Brief Placard. BOP will provide Critical Parameters using Brief Placard. *The STA will state "nothing to add".* Completes Brief and continues with AP-53.00.

Event Description: VCT Level Transmitter, LT-1112 Fails High

Cue: By Examiner.

	SRO RO	0-AP-53.00 *3 VERIFY REACTOR POWER – LESS THAN OR EQUAL TO 100% *Identify that reactor power is less than 100%.*
	SRO	0-AP-53.00 Notes Prior to Step 4: • Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud. • When the affected instrument/controller malfunction(s) has been addressed by this procedure, recovery actions should continue at Step 13.
	SRO RO	0-AP-53.00 *4 DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE: VCT Level, Step 12e.
	SRO RO SRO	0-AP-53.00 12 e) VCT level Instrumentation – NORMAL *States 1-CH-LT-1112, VCT Level Channel affected.* Refer to Attachment 6
	SRO	0-AP-53.00 Attachment 6, 0-AP-53.00 If 1-CH-LT-1112 fails high, 1-CH-LCV-1115A will open. Automatic swap-over of the CHG pump suction from the VCT to the RWST will NOT function. • 1-CH-LT-1112 provides input to the following: a. Controller 1-CH-LC1112C for modulating 1-CH-LCV-1115A open at the setpoint set on the controller. b. CHG pump suction swap over to the RWST at 13% (2/2) Discusses with RO control of VCT level and monitoring of channel 1-CH-LT-1115. The SRO will set a band for maintaining VCT level.

Event Description: VCT Level Transmitter, LT-1112 Fails High

Cue: By Examiner.

	SRO SRO/RO	0-AP-53.00 13 CHECK CALORIMETRIC – FUNCTIONAL IAW 1-OPT-RX-001, Attachment 4. SRO/RO determines that OPT-RX-001, Attachment 4, is NOT impacted and OPT-RX-007 will NOT need to be performed.
	SRO SRO STA	0-AP-53.00 14. Review the following: • TS 3.7 • VPAP-2802 • TRM Section 3.3, Instrumentation • Reg Guide 1.97 • EP-AA-303, Equipment Important to Emergency Response. SRO will consult Tech Specs and find no items affected. • Reports that VPAP-2802, TRM Section 3.3, **Reg Guide 1.97**, and EP-AA-303; Results of Review discussed with Shift Manager. • *Reg Guide 1.97 requires one channel of VCT level indication. Use an Alternate indication. Restore before next outage. There is no impact for VPAP-2802, and TRM section 3.3.*
	SRO RO	0-AP-53.00 15. Check additional Instrument Malfunction exists. Reports No goes to Step 17.
	SRO	0-AP-53.00 17. Provide Notifications as necessary. • Shift Supervision • OMOC • STA (PRA determination) • I&C
		END EVENT 4

Event Description: Loss of High Pressure Drain Pump (1-AP-18.00).

Cue: When initiated by Team.

Time	Position	Applicant's Action or Behavior
	BOP	Diagnose the trip of the High Pressure Drain Pump, 1-SD-P-1A based on the following indications: • 1K-D4 4 KV BKR AUTO TRIP • 1H-F3 CN POLISHING SYS TRBL • 1H-D3 CN POLISHING BYPASS AOV OPEN • 1J-B4 HP HTR DR RCVR TK HI LVL • 1H-G5/6/7 SG 1A/1B/1C LVL ERROR • 1J-E1 FW HTR 4A HI LVL • 1J-F1 FW HTR 4B HI LVL
	 SRO SRO BOP SRO BOP	Step 1, AP-18.00 Enter 1-AP-18.00 CHECK HP HEATER DRAIN PUMP STATUS: a) Check HP Heater Drain Pump – TRIPPED OR NOT PROVIDING SUFFICIENT FLOW *Reports that 1-SD-P-1B, HP Heater Drain pumps tripped.* b) Place pump control Switch in PTL **Places 1-SD-P-1B control switch in PTL**
	 SRO RO/BOP	Step 2, 1-AP-18.00 CHECK REACTOR POWER – GREATER THAN OR EQUAL TO 75% *Reports reactor power at 100%*
	 SRO BOP	Step 3, AP-18.00 START THIRD CONDENSATE PUMP AS REQUIRED BY PLANT CONDITIONS **Starts 1-CN-P-1A and verifies Proper Operation.**

Event Description: Loss of High Pressure Drain Pump (1-AP-18.00).

Cue: When initiated by Team.

	SRO BOP	Step 4, AP-18.00 NOTE: With unit at 100% power, Turbine load should be decreased approximately 50 MW. REDUCE TURBINE LOAD AS NECESSARY TO MAINTAIN LOOP ΔTs - LESS THAN 100%. • *Use Valve Position Limiter* *OR* • *Reduce Turbine load using Turbine Manual* **Using Guidance of NOTE preceding the Step, Turbine load will be reduced approximately 50 MW.**
	SRO	Step 5, 1-AP-18.00 NOTE: Ramping to 75% allows the Condensate Polishing Building to be placed fully in service. COMMENCE RAMP TO 75% POWER IAW 0-AP-23.00, RAPID LOAD REDUCTION
	SRO	Step 6, 1-AP-18.00 * USE CONTROL RODS AND CHEMICAL SHIM TO MAINTAIN ΔFLUX IN BAND
	SRO BOP	Step 7, 1-AP-18.00 MONITOR MAIN FEED REG VALVE RESPONSE - MAINTAINING SG LEVEL IN BAND Acknowledges Step direction.
	SRO BOP	Step 8, 1-AP-18.00 CHECK CP BUILDING BYPASSED. Reports that the CP Building is bypassed.
	SRO BOP	Step 9, 1-AP-18.00 CHECK HP HEATER DRAIN PUMP TRIP CAUSED BY NETWORK 90 FAILURE. Reports that No, the HP Heater Drain pump trip was not caused by Network 90 failure. Goes to Step 11

Event Description: Loss of High Pressure Drain Pump (1-AP-18.00).

Cue: When initiated by Team.

	SRO BOP	Step 11, 1-AP-18.00 NOTIFY THE FOLLOWING: ❑ CP Building ❑ Energy Supply (MOC) ❑ Chemistry ❑ STA Notifies required parties as directed.
	SRO	Initiate AP-23.00, Rapid Load Reduction
		END EVENT #5

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Event Description: Ramp Unit to < 75% per 0-AP-23.00.

Cue: When initiated by Team.

		Start of 0-AP-23.00
	SRO	Conducts a Brief summarizing the Event and Establish priorities.
		RO will provide Critical Parameters using Brief Placard.
		BOP will provide Critical Parameters using Brief Placard.
		The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.
	RO	Reactivity control during AP-23.00 Ramp: 154 gallons of Boric Acid needed to reduce power to 75%. 24 second Emergency Boration. Normal boration at an average rate of 6.2 gpm. Control Bank 'D' rod height at end of ramp 197 Steps. 1919 gallons of PG to stabilize at end of ramp.
	SRO	Completes Brief and continues with AP-23.00.
	SRO	Caution Prior to Step 1: • Conservative decision-making must be maintained during rapid load reductions. Refer to Attachment 1 for trip criteria. Notes Prior to Step 1: • Actions that can be completed independently of preceding steps may be performed out of sequence as directed by the SRO • When the Turbine is not being actively ramped, the REFERENCE and SETTER values must remain matched to prevent inadvertent ramp. • Pre-planned reactivity plans located in the Main Control Room will be used as guidance for ramping down to the desired power level. • The ramp rate in IMP OUT is nonlinear and therefore pre-planned reactivity plans based on IMP IN are not as accurate. However, total amounts of boration and dilution can be used as guidance. • For ramp rates greater than or equal to 1%/minute, Rod Control should remain in Automatic if available.
		0-AP-23.00
	RO	1. **TURN ON ALL PRZR HEATERS**

Op-Test No.: Surry 2016-1 Scenario No.: 1 Event No.: 6 Page 36 of 97

Event Description: Ramp Unit to < 75% per 0-AP-23.00.

Cue: When initiated by Team.

	BOP RO SRO/BOP BOP	2. INITIATE PLANT LOAD REDUCTION AT 2%/MINUTE OR LESS: a) Verify turbine valve position - NOT ON LIMITER The turbine is NOT on the limiter. Note: If Turbine MAN was used. If the Limiter was used to reduce power then the team will have to ramp off of the limiter. b) Insert control rods in AUTO or MANUAL as necessary to maintain Tave and Tref within 5°F. c) **Verify or place turbine in IMP IN or IMP OUT as determined by Shift Supervision** The SRO can choose IMP IN or IMP OUT. d) **Adjust SETTER to desired power level** e) **Adjust LOAD RATE %/MIN thumbwheel to desired ramp rate (1%/minute)** f) **Initiate Turbine load reduction using OPERATOR AUTO (pushes the GO button)** g) **Reduce Turbine Valve Position Limiter as load decreases** The BOP will periodically reduce the limiter setpoint during the ramp.
	SRO	3. CHECK EMERGENCY BORATION – REQUIRED The team may decide to emergency borate after the ramp has progressed to the point that Tave and Tref are matched (or close).
	SRO	Note Prior to Step 4: • Step 4 or Step 5 may be performed repeatedly to maintain Tave and Tref matched, ΔFlux in band, and control rod position above the LO-LO insertion limit.

Event Description: Ramp Unit to < 75% per 0-AP-23.00.

Cue: When initiated by Team.

		0-AP-23.00
	RO	4. PERFORM AN EMERGENCY BORATION IAW THE FOLLOWING: a) Verify or raise CHG flow to greater than 75 gpm b) **Transfer the in-service BATP to FAST** c) **Open 1-CH-MOV-1350** SRO will direct nominal opening of approximately 25 seconds. d) Monitor EMRG BORATE FLOW • 1-CH-FI-1110 e) After required emergency boration, perform the following: 1) **Close 1-CH-MOV-1350** 2) **Transfer the in-service BATP to AUTO** 3) Restore Charging flow control to normal SRO may direct rod motion to maintain Δ Flux within specified band.
	RO	5. ESTABLISH A NORMAL BORATION TO MAINTAIN CONTROL ROD POSITION ABOVE THE LO-LO INSERTION LIMITS IAW ATTACHMENT 4 Attachment 4 (Boration) and 5 (Manual Makeups) are at the end of this section. SRO may direct manual rod motion to maintain Δ flux within specified band.
	SRO	Notes Prior to Step 6: • If at any time plant conditions no longer require rapid load reduction, actions should continue at Step 35. • RCS Tave must be maintained less than or equal to 577°F and RCS pressure must be maintained greater than or equal to 2205 psig. Tech Spec 3.12.F.1 should be reviewed if either parameter is exceeded. • I & C should be contacted to provide assistance with adjusting IRPIs.
	RO	6. CONTROL RAMP RATE TO MAINTAIN RCS PRESSURE GREATER THAN 2205 PSIG

Event Description: Ramp Unit to < 75% per 0-AP-23.00.

Cue: When initiated by Team.

	RO	0-AP-23.00 *7. CHECK LETDOWN ORIFICES – TWO IN SERVICE *Evaluator note: two orifices will already be in service.*
	BOP	8. MONITOR STEAM DUMPS FOR PROPER OPERATION
	SRO	0-AP-23.00 9. NOTIFY THE FOLLOWING: • Energy Supply (MOC) • Polishing Building • Chemistry • OMOC
	SRO	10. EVALUATE THE FOLLOWING: • EPIP applicability *The Shift Manager will review EPIPs for applicability. They are not applicable.* • VPAP-2802, NOTIFICATIONS AND REPORTS, applicability *SRO directs STA to review VPAP-2802. The STA reports that he has completed his review of VPAP-2802 and no notifications are required.* *No further actions are required for this event.*
	SRO	11. CHECK RAMP WILL BE TO LESS THAN APPROXIMATELY 35% REACTOR POWER No, go to step 12.
	SRO	12. CHECK REACTOR POWER – HAS LOWERED MORE THAN 15% IN ONE HOUR. When reactor power has lowered >15%, then chemistry will be notified.
	SRO	13. NOTIFY CHEMISTRY OF POWER CHANGE > 15% IN ONE HOUR. Chemistry notified of power change > 15% in one hour.

Event Description: Ramp Unit to < 75% per 0-AP-23.00.

Cue: When initiated by Team.

	SRO	0-AP-23.00 CAUTION: Secondary plant evolutions affecting Feedwater Flow or temperature will affect RCS temperature and Reactor Power. This effect will be greater at beginning of core life due to a lower value for isothermal temperature coefficient. The operating team must be prepared to mitigate the effects of the secondary evolutions on the RCS. *RO acknowledges the Caution.*
	SRO	0-AP-23.00 14. AT APPROXIMATELY 70% REACTOR POWER CHECK AUXILIARY STEAM MAINTAINING BETWEEN 160 AND 180 PSIG. *RO Acknowledges the step.*
		END EVENT #6

Event Description: Ramp Unit to < 75% per 0-AP-23.00.

Cue: When initiated by Team.

		0-AP-23.00 Attachment 4 (NORMAL BORATION) Actions
	RO	1. **Place the MAKE-UP MODE CNTRL switch in the STOP position.**
	RO	2. **Adjust 1-CH-YIC-1113 to desired total gallons**
	RO	3. **Adjust 1-CH-FC-1113A to desired flow rate.**
	RO	4. **Place the MAKE-UP MOD SEL switch in the BORATE position**.
	RO	5. **Place the MAKE MODE CNTRL switch in the START position.**
	RO	6. Verify proper valve positions.
	RO	7. Adjust boration rate using 1-CH-FC-1113A, as necessary.
	RO	8. WHEN boration is complete, THEN perform the following. IF boric acid is to remain in the Blender to support ramping the Unit, THEN enter N/A. a) Manually blend approximately 20 gallons to flush the boration path IAW Attachment 5, Manual Makeups. b) Enter N/A for the remaining steps in this Attachment. *Attachment 5 is on the next page*
	RO	9. Verify controllers for Primary Grade water and Boric Acid are set correctly.
	RO	10. **Place the MAKE-UP MODE SEL switch in the AUTO position.**
	RO	11. **Place the MAKE-UP MODE CNTRL switch in the START position.**
	RO	12. Notify Shift Supervision of blender status.

Event Description: Ramp Unit to < 75% per 0-AP-23.00.

Cue: When initiated by Team.

		0-AP-23.00 Attachment 5 (Manual Makeups) Actions
		1. Place the MAKE-UP MODE CNTRL switch in the STOP position.
		2. Check controllers for the flow rate of Boric Acid and Primary Grade water are set correctly.
		3. Check integrators for the gallons of Boric Acid and Primary Grade water are set correctly.
		4. Place the MAKE-UP MODE SEL switch in the MANUAL position.
		5. Place the MAKE-UP MODE CNTRL switch in the START position.
		6. Open 1-CH-FCV-1113B, BLENDER TO CHG PUMP.
		7. Check proper valve positions.
		8. WHEN the Manual Makeup operation is complete, THEN place 1-CH-FCV-1 113B in the AUTO position
		9. Place the MAKE-UP MODE CNTRL switch in the STOP position.
		10. Check or place the control switches in the AUTO position.
		11. Check controllers for Primary Grade water and Boric Acid are set correctly.
		12. Place the MAKE-UP MODE SEL switch in the AUTO position.
		13. Place the MAKE-UP MODE CNTRL switch in the START position.
		14. Notify Shift Supervision of blender status.

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

Time	Position	Applicant's Action or Behavior
	Team	Diagnose the failure based upon the following alarms and indications: Annunciator 1H-A4, T AVG <> T REF DEVIATION Annunciator 1F-F4,(G4), STM GEN 1A CH3 (CH4) HI STM LINE FLOW Annunciator 1F-F5 (G5), STM GEN 1B CH3 (CH4) HI STM LINE FLOW Annunciator 1F-F6 (G6), STM GEN 1C CH3 (CH4) HI STM LINE FLOW Annunciator 1C-B8, PRZR LO PRESS All SG NR Level indications rising Report from Security that there is a steam rupture in U1 Turbine Hall Note: All RCPs will trip in approximately 4 minutes.
	SRO	Direct RO to trip the reactor and perform the Immediate Actions of 1-E-0.
	RO	1-E-0, Reactor Trip or Safety Injection [1] CHECK REACTOR TRIP: a) Manually trip reactor **Presses reactor trip buttons (both).** *RO determines Rx Trip breakers do NOT open after BOTH Reactor trip buttons are depressed.* *RO enters 1-FR-S.1.*
	RO	1-FR-S.1 Step 1 [1] VERIFY REACTOR TRIP. No perform step 1 RNO. *RO performs step 1 RNO, and verifies control rods are in AUTO*
	RO	1-FR-S.1 Step 2 [2] MANUALLY TRIP THE TURBINE. ❑ Verify all Turbine stop valves closed *RO Manually trips the turbine and verifies all turbine stop valves closed.*

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	RO BOP	1-FR-S.1 Step 3 [3] VERFIES CONTROL RODS – INSERTING IN AUTO AT GREATER THAN 48 STEPS/MINUTE *RO continuously monitors rod insertion, and places rods in MANUAL and inserts rods if rod insertion drops to < 48 steps/minute.* *Contacts Field Operator and directs locally opening of the Unit 1 Reactor Trip breakers.* NOTE: this direction can occur earlier or later. FR-S.1 will procedurally direct this at step 8
	SRO RO	1-FR-S.1 Step 4 4. VERIFY AFW PUMPS RUNNING ❑ MD AFW pumps – RUNNING ❑ TD AFW pump – RUNNING IF NECESSARY. *RO verifies no lockouts on MD AFW pumps, and starts 1-FW-P-3A, and 1-FW-P-3B.* **Critical Task** **CT-3:** Manually start AFW pumps as needed before SG WR level decreases to <7% on both intact S/Gs.
	SRO RO	1-FR-S.1 Step 5 5. INITIATE EMERGENCY BORATION OF RCS ❑ Verify CHG flow – GREATER THAN 75 GPM. ❑ Align boration path: o **Put BATP in FAST** o **Open 1-CH-MOV-1350** o Verify emergency borate flow. *RO initiates emergency boration.*

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	SRO RO	1-FR-S.1 Step 6 6. CHECK IF SI INITIATED ❑ LHSI pumps RUNNING ❑ SI annunciators LIT (AF-3, AF-4) *RO reports that SI is initiated.*
	SRO BOP	1-FR-S.1 Step 7 7. INITIATE ATTACHMENT 1. *BOP initiates Attachment 1, SRO continues in 1-FR-S.1 with RO.* Note: FR-S.1, Attachment 1 is included in next section.
	SRO RO	1-FR-S.1 Step 8 8. CHECK IF THE FOLLOWING HAVE OCCURRED: ❑ Reactor Trip ❑ Turbine Trip *RO reports that the reactor failed to auto trip or manual trip and dispatches operator to locally trip the reactor trip and bypass breakers.* Note: This action has probably already been taken.
	SRO RO	1-FR-S.1 Step 9 Note before Step 9. If adverse CTMT conditions have been exceeded, the Gamma-Metrics Excore Neutron Monitor system(Source and Wide Ranges) should be used to mnitor neutron flux for the duration of the event. 9. CHECK REACTOR SUBCRITICAL ❑ a. Check power range channels – LESS THAN 5% [Gamma-Metrics "Wide Range power – LESS THAN 5%] ❑ b. Check intermediate range channels –NEGATIVE STARTUP RATE [Gamma-Metrics Wide Range Power – DECREASING] ❑ c. GO TO Step 18 *RO reports that power range channels are < 5%.* SRO goes to Step 18.

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	SRO	1-FR-S.1 Step 18 CAUTION before Step 18. Boration should be continued to obtain adequate shutdown margin during subsequent actions. 18. RETURN TO PROCEDURE AND STEP IN EFFECT. SRO goes to 1-E-0, directs RO to perform Immediate actions of E-0.

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

		1-E-0 Step 1
	RO	[1] CHECK REACTOR TRIP: a) Manually trip reactor Presses reactor trip button. b) Check the following: All Rods On Bottom light – LIT Identifies All Rods on Bottom LIT on CERPI Screen. Reactor trip and bypass breakers – OPEN Identifies Reactor Trip and Bypass breakers Open on Benchboard Mimic. Neutron flux – LOWERING Identifies PR NI N41, N42, and N43 indications at ~0%; and IR indicators N35/N36 Lowering. Reports to SRO “Reactor Tripped”.

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Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	RO	1-E-0 Step 2 [2] CHECK TURBINE TRIP: a) Manually trip the turbine Presses both Turbine Trip pushbuttons – simultaneously. b) Check all turbine stop valves – CLOSED Identifies Turbine SVs closed using indication lights on Turbine Control section. c) Isolate reheaters by closing MSR steam supply SOV 1-MS-SOV-104 **Places 1-MS-SOV-104 control switch in close.** d) Check generator output breakers – OPEN (Time Delayed) Identifies Main generator output breakers open. Reports to SRO “Turbine is Tripped”.
	RO	1-E-0 Step 3 [3] CHECK BOTH AC EMERGENCY BUSES – ENERGIZED Identifies “H” and “J” buses are energized by checking Voltage indicated on #1 and #3 EDG control panels. Reports “Both AC Emergency Buses energized.”

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	RO	1-E-0 Step 4 [4] CHECK IF SI INITIATED: a) Check if SI is actuated: LHSI pumps – RUNNING Identifies A/B LHSI pumps running using breaker and amp indications. SI annunciators – LIT A-F-3 (SI Initiated Train A) A-F-4 (SI Initiated Train B) Identifies both Annunciators LIT. b) Manually initiate SI **Presses Manual SI buttons, Train "A" and Train "B".** Reports E-0 Immediate Actions are complete, Have SI flow to the core."
	SRO	1-E-0, Reactor Trip or Safety Injection Hands out Continuous Action Pages for E-0 to RO and BOP, provides Attachments 1, 2, and 3 to BOP. Leads a Transient Brief to describe the Plant Status, and asks RO/BOP if any items identified during the E-0 Immediate Actions would have higher priority than continuing with E-0. RO/BOP may identify MSTVs failing to close following safety injection. STA will have no input for the brief. SRO closes the Transient Brief and continues E-0 with the RO.

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	SRO	1-E-0 Step 5 5. INITIATE ATTACHMENT 1 Directs BOP to perform E-0 Attachment 1, 2, and 3. E-0 Attachments and components BOP will identify and reposition begin in next section on page 55. **Critical Task** **CT-4**: Close Main Steam Trip valves prior to reaching an orange path on Integrity (FR-P.1). Critical Task **begins** when the Main Steamline rupture occurs and **ends** when the Main Steam Trip valves are closed. Note: It is anticipated that this critical task will be accomplished during Attachment 1 by the BOP OR at Step 6 by the RO.
	SRO RO SRO RO SRO RO SRO RO	1-E-0 Step 6 *6. CHECK RCS AVERAGE TEMPERATURE STABLE AT 547°F OR TRENDING TO 547°F Report NO, RCS Temperature lowering (and provide current Tave value). Goes to Step 6 RNO <u>IF</u> temperature less than 547°F AND lowering, THEN do the following: a) Stop dumping steam. Reports Yes, Steam Dumps are closed. b) IF cooldown continues, THEN control total feed flow. Maintain total feed flow greater than 350 gpm [450 gpm] until narrow range level greater than 12% [18%] in at least one SG. Identify RCS Tave Lowering. Direct RO to throttle AFW to all SGs to ~120 gpm. Throttle AFW to the SGs to ~120 gpm per SG and report when complete. c) IF Cooldown continues, THEN close MSTVs. **RO CLOSES MSTVs.** **CT-4**: Close Main Steam Trip valves prior to reaching an orange path on Integrity (FR-P.1).

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

		1-E-0 Step 7
	SRO	7. CHECK PRZR PORVs AND SPRAY VALVES:
		a) PRZR PORVs – CLOSED
	RO	Reports Yes, PRZR PORVs closed.
	SRO	b) PRZR spray controls Demand at Zero OR Controlling pressure
	RO	Reports Yes, Demand at zero.
	SRO	c) PORV block valves - AT LEAST ONE OPEN
	RO	Reports Yes, one block valve open.
		1-E-0 Step 8
	SRO	NOTE Prior to Step 8: Seal injection flow should be maintained to all RCPs.
	RO	Acknowledges NOTE.
	SRO	*8. CHECK RCP TRIP AND MINIFLOW RECIRC CRITERIA:
		a) Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS
	RO	Reports Yes, 3 running and flowing to the RCS. May report 2 running depending upon BOP speed of progression through E-0, Attachment 1.
	SRO	b) RCS subcooling - LESS THAN 30°F [85°F]
	RO	Reports No, subcooling is (provides actual subcooling value.)
	SRO	Step 8 RNO: Goes to Step 9.

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	SRO RO	1-E-0 Step 9 9. CHECK IF SGs ARE NOT FAULTED: Check pressures in all SGs: STABLE OR RISING AND GREATER THAN 100 PSIG Reports Yes, SG pressures are (current value) and not lowering. Note: It is anticipated that Main Steam Trip valves are isolated at this point. If crew has not isolated Main Steam Trip valves then they may believe that all SGs are faulted and enter E-2. The first 3 steps of E-2 are included at the end of this section, page 54.
	SRO RO	1-E-0 Step 10 10. CHECK IF SG TUBES ARE NOT RUPTURED: Condenser air ejector radiation - NORMAL SG blowdown radiation – NORMAL SG MS radiation – NORMAL TD AFW pump exhaust radiation – NORMAL SG NR level – NOT RISING IN AN UNCONTROLLED MANNER Reports Yes, SG Tubes are not ruptured.
	SRO RO	1-E-0 Step 11 11. CHECK RCS - INTACT INSIDE CTMT CTMT radiation – NORMAL CTMT pressure – NORMAL CTMT RS sump level – NORMAL Reports Yes, the RCS is intact inside containment.

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Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	SRO RO	1-E-0 Step 12 12. CHECK RCS – HAS BEEN MAINTAINED INTACT OUTSIDE CTMT a) Radiation Monitors – Normal • MGPI vent-vent • Auxiliary Building Control Area b) Sump annunciators – NOT LIT • VSP-F4 • 1B-D1 • 1B-D2 • 1B-F3 Reports Yes, RCS has been maintained intact outside containment.
	SRO RO	1-E-0 Step 13 13. CHECK IF SI FLOW SHOULD BE REDUCED. a) RCS subcooling based on CETCs – GREATER THAN 30°F b) Secondary heat sink: • Total feed flow to SGs – GREATER THAN 350 GPM OR • Narrow range level in at least one SG – GREATER THAN 12% c) RCS pressure – STABLE OR RISING d) PRZR level – GREATER THAN 22% Reports to all Yes, SI flow should be reduced.
	SRO	1-E-0 Step 14 14. HAVE STA INITIATE MONITORING OF CRITICAL SAFETY FUNCTION STATUS TREES. Directs STA to monitor Critical Safety Function Status Trees. STA acknowledges direction.
	SRO RO	1-E-0 Step 15 15. RESET BOTH TRAINS OF SI. Resets both trains of SI.
	SRO RO	1-E-0 Step 16 16. RESET CLS: a) Check CTMT pressure – HAS EXCEEDED 17.7 psia. Reports No CTMT pressure has not exceeded 17.7 psia. Goes to step 17

Op-Test No.: Surry 2016-1 Scenario No.: 1 Event No.: 7 Page 53 of 97

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	SRO RO	1-E-0 Step 17 17. CHECK INSTRUMENT AIR AVAILABLE a) Check annunciator 1B-E6 – NOT LIT b) Check at least one CTMT IA compressor RUNNING • 1-IA-C-4A or 1-IA-C-4B c) Check 1-IA-TV-100 OPEN. Reports that Instrument Air is available.
	SRO RO	1-E-0 Step 18 18. STOP ALL BUT ONE CHG PUMP AND PUT IN AUTO. Stops one Charging pump (anticipate RO stopping 1-CH-P-1A).
	SRO RO	1-E-0 Step 19 19. CHECK RCS PRESSURE – STABLE OR RISING Reports that RCS pressure is __ (report current value), and rising.
	SRO RO	1-E-0 Step 20 20. ISOLATE HHSI TO COLD LEGS: a) Check CHG pump suctions from RWST OPEN: • 1-CH-MOV-1115B • 1-CH-MOV-1115D b) Check CHG pump miniflow recirc valves – OPEN. • 1-CH-MOV-1275A • 1-CH-MOV-1275B • 1-CH-MOV-1275C • 1-CH-MOV-1373 c) Close HHSI to Cold Leg: • **1-SI-MOV-1867C** • **1-SI-MOV-1867D** • **1-SI-MOV-1842** Performs above actions to isolate HHSI to Cold Legs.

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	SRO RO	1-E-0 Step 21 21. ESTABLISH CHG FLOW a) Close CHG flow control • **1-CH-MOV-1122** b) Check CHG line isolation -OPEN • 1-CH-HCV-1310A c) Open CHG line isolation MOVs • **1-CH-MOV-1289A** • **1-CH-MOV-1289B** Performs actions to establish CHG flow.
		1-E-0 Step 22 22. CONTROL CHG FLOW TO MAINTAIN PRZR LEVEL Reports, controlling CHG flow to maintain PRZR level.
		1-E-0 Step 23 23. GO TO 1-ES-1.1, SI TERMINATION, STEP 9
		END OF EVENT 7 **END OF SCENARIO**

Op-Test No.: Surry 2016-1 **Scenario No.: 1** **Event No.: 7** **Page 55 of 97**

Event Description: Main Steamline Rupture with ATWS.

Cue: By Evaluator.

	SRO	1-E-2, Faulted SG Isolation Hands out Continuous Action Pages for E-2 to RO and BOP. Leads a Transient Brief to describe the Plant Status, and asks RO/BOP there is a higher priority than continuing with E-2. RO/BOP may identify MSTVs failing to close following safety injection. STA will have no input for the brief. SRO closes the Transient Brief and continues E-2.
	SRO RO	1-E-2 Step 1 Caution before Step 1: • 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. 1. CHECK MSTV AND BYPASS VALVE ON AFFECTED SG(s) – CLOSED. RO reports that MSTVs are not closed and closes **1-MS-TV-101A, 1-MS-TV-101B, and 1-MS-TV-101C.**
	SRO RO	1-E-2 Step 2 2. CHECK IF ANY SG SECONDARY SIDE IS INTACT: • Check pressures in all SGs –ANY STABLE OR RISING Reports that pressures in all SGs are stable.
	SRO RO	1-E-2 Step 3 3. IDENTIFY FAULTED SG(S): Reports that no SGs are faulted.
		END OF EVENT 7 **END OF SCENARIO**

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-FR-S.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 26	VERIFYING APPLICABLE STEPS OF 1-E-0	PAGE 1 of 8

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	VERIFY REACTOR TRIP:	
	☐ a) Manually trip reactor	☐ Verify that an Operator has been sent to perform local trip.
	b) Check the following:	
	☐ • All Rods On Bottom light - LIT	
	☐ • Reactor trip and bypass breakers - OPEN	
	☐ • Neutron flux - DECREASING	
2.___	VERIFY TURBINE TRIP:	
	☐ a) Manually trip turbine	
	☐ b) Verify all turbine stop valves - CLOSED	b) Verify either of the following: ☐ • All Turbine Governor Valves - CLOSED OR ☐ • Turbine speed - DECREASING OR ☐ • Generator Motoring - INITIATED ☐ IF turbine will NOT trip, THEN close MSTVs.
	c) Isolate reheaters by closing MSR steam supply SOV ☐ • 1-MS-S0V-104	☐ c) IF reheater FCVs will NOT close, THEN close MSR steam supply MOVs.
	☐ d) Verify generator output breakers - OPEN (Time Delayed)	☐ d) IF Generator Output Breakers do NOT open within 30 seconds, THEN manually open output breakers AND place the EXCITATION control switch in OFF.

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-FR-S.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 26	VERIFYING APPLICABLE STEPS OF 1-E-0	PAGE 2 of 8

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.___	VERIFY BOTH AC EMERGENCY BUSES - ENERGIZED	Do the following: ☐ a) IF no AC Emergency Bus is energized, THEN GO TO 1-ECA-0.0, LOSS OF ALL AC POWER. ☐ b) Try to restore power to deenergized AC Emergency Bus. Initiate 1-AP-10.07, LOSS OF UNIT 1 POWER.
4.___	CHECK IF SI INITIATED:	
	a) Check if SI is actuated: ☐ • LHSI pumps - RUNNING • Safety Injection system Initiated annunciators - LIT ☐ • A-F-3 ☐ • A-F-4	a) Check if SI is required or imminent as indicated by any of the following: ☐ • Low PRZR pressure ☐ • High CTMT pressure ☐ • High steamline differential pressure ☐ • High steam flow with low Tave or low line pressure ☐ IF SI is required, THEN GO TO Step 4b. ☐ IF SI is NOT required, THEN GO TO 1-ES-0.1, REACTOR TRIP RESPONSE.
	☐ b) Manually initiate SI	

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-FR-S.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 26	VERIFYING APPLICABLE STEPS OF 1-E-0	PAGE 3 of 8

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.___	VERIFY FW ISOLATION: • Feed pump discharge MOVs - CLOSED ☐ • 1-FW-MOV-150A ☐ • 1-FW-MOV-150B ☐ • MFW pumps - TRIPPED ☐ • Feed REG valves - CLOSED ☐ • SG FW bypass flow valves - DEMAND AT ZERO ☐ • SG blowdown TVs - CLOSED	☐ Manually close valves and stop pumps.
6.___	VERIFY CTMT ISOLATION PHASE I: ☐ • Phase I TVs - CLOSED ☐ • 1-CH-MOV-1381 - CLOSED ☐ • 1-SV-TV-102A - CLOSED • PAM isolation valves - CLOSED ☐ • 1-DA-TV-103A ☐ • 1-DA-TV-103B	☐ Manually close valves.
7.___	VERIFY AFW PUMPS REQUIRED:	
	☐ a) MD AFW pumps - RUNNING (Time Delayed)	☐ a) Manually start pumps.
	☐ b) TD AFW pump - RUNNING IF NECESSARY	b) Manually open steam supply valves. ☐ • 1-MS-SOV-102A ☐ • 1-MS-SOV-102B

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-FR-S.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 26	VERIFYING APPLICABLE STEPS OF 1-E-0	PAGE 4 of 8

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. ___	VERIFY SI PUMPS RUNNING: ☐ • CHG pumps - RUNNING ☐ • LHSI pumps - RUNNING	☐ Manually start pumps.
9. ___	CHECK CHG PUMP AUXILIARIES: ☐ • CHG pump CC pump - RUNNING ☐ • CHG pump SW pump - RUNNING	☐ Manually start pumps.
10. ___	CHECK INTAKE CANAL: ☐ • Level - GREATER THAN 24 FT ☐ • Level - BEING MAINTAINED BY CIRC WATER PUMPS	☐ IF level is less than 24 FT OR decreasing in an uncontrolled manner, THEN initiate 0-AP-12.01, LOSS OF INTAKE CANAL LEVEL.

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-FR-S.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 26	VERIFYING APPLICABLE STEPS OF 1-E-0	PAGE 5 of 8

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.___	CHECK IF MAIN STEAMLINES SHOULD BE ISOLATED:	
	a) Check if ANY of the following annunciators - HAVE BEEN LIT ☐ • E-F-10 (High Steam Flow SI) ☐ • B-C-4 (Hi Hi CLS Train A) ☐ • B-C-5 (Hi Hi CLS Train B)	a) Do the following: ☐ IF annunciator E-H-10 (Hdr/Line SI) LIT, THEN GO TO Step 11d. ☐ IF annunciator E-H-10 NOT LIT, THEN GO TO Step 12.
	☐ b) Check MSTVs - CLOSED	☐ b) Manually close valves.
	c) Check either of the following - ACTUATED ☐ • Hi steam flow SI OR ☐ • Header to line SI	☐ c) GO TO Step 12.
	d) Verify RWST crosstie valves - OPEN ☐ • 1-SI-TV-102A ☐ • 1-SI-TV-102B ☐ • 2-SI-TV-202A ☐ • 2-SI-TV-202B	☐ d) Manually open valves.
	☐ e) Check RCS pressure - LESS THAN 185 PSIG	☐ e) Put BOTH RMT mode transfer switches in REFUEL.

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-FR-S.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 26	VERIFYING APPLICABLE STEPS OF 1-E-0	PAGE 6 of 8

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*12.___	CHECK IF CS REQUIRED:	
	☐ a) CTMT pressure - HAS EXCEEDED 23 PSIA	a) Do the following: 1) IF CTMT pressure has exceeded 17.7 psia, THEN verify or align the following valves: ☐ • 1-RM-TV-100A - CLOSED ☐ • 1-RM-TV-100B - CLOSED ☐ • 1-RM-TV-100C - CLOSED ☐ • 1-SV-TV-102 - CLOSED ☐ • 1-IA-TV-101A - CLOSED ☐ • 1-IA-TV-101B - CLOSED ☐ • 1-IA-AOV-103 - OPEN ☐ 2) GO TO Step 14.
	☐ b) Manually initiate HI HI CLS	
	☐ c) Trip all RCPs	
	☐ d) Verify CS pumps - RUNNING	☐ d) Manually start pump(s).
	☐ e) Initiate Attachment 2	

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-FR-S.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 26	VERIFYING APPLICABLE STEPS OF 1-E-0	PAGE 7 of 8

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*13.___	CHECK IF RS REQURIED:	
	☐ a) Check RWST level - LESS THAN OR EQUAL TO 60%	a) Do the following: ☐ 1) Continue to monitor RWST level. ☐ 2) GO TO Step 14. IF RWST level decreases to less than or equal to 60%, THEN perform Step 13b through Step 13d.
	☐ b) Check ISRS pumps - RUNNING	☐ b) Manually Start Pumps.
	☐ c) Check OSRS pumps - RUNNING (Time Delayed)	☐ c) Manually Start Pumps.
	☐ d) Check OSRS pumps - NOT CAVITATING	☐ d) Put affected OSRS pump in PTL.
*14.___	BLOCK LOW PRZR PRESS SI SIGNAL:	
	☐ a) Check PRZR pressure - LESS THAN 2000 psig	☐ a) GO TO Step 15. WHEN PRZR pressure less than 2000 psig, THEN perform Steps 14b and 14c.
	☐ b) Turn both LO PRZR PRESS & STM HDR/LINE ΔP switches to block	
	☐ c) Verify Permissive Status light C-2 - LIT	
*15.___	BLOCK LOW TAVE SI SIGNAL:	
	☐ a) Check RCS Tave - LESS THAN 543°F	☐ a) GO TO Step 16. WHEN Tave less than 543°F, THEN perform Steps 15b and 15c.
	☐ b) Turn both HI STM FLOW & LO TAVG OR LP switches to block	
	☐ c) Verify Permissive Status light F-1 - LIT	

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-FR-S.1	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 26	VERIFYING APPLICABLE STEPS OF 1-E-0	PAGE 8 of 8

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: • CHG pumps should be run in the following order of priority: C, B, A. • Subsequent SI signals may be reset by reperforming Step 16.	
16.___	VERIFY SI FLOW:	
	a) HHSI to cold legs - FLOW INDICATED ☐ • 1-SI-FI-1961 (NQ) ☐ • 1-SI-FI-1962 (NQ) ☐ • 1-SI-FI-1963 (NQ) ☐ • 1-SI-FI-1943 or 1-SI-FI-1943A	☐ a) Manually start pumps and align valves. <u>IF</u> flow <u>NOT</u> established, <u>THEN</u> consult with Shift Supervision to establish another high pressure injection flowpath while continuing with this procedure. ☐ • Alternate SI to cold legs ☐ • Hot leg injection
	☐ b) Check CHG pumps - THREE RUNNING	☐ b) GO TO Step 16e.
	☐ c) Reset SI	
	☐ d) Stop one CHG pump and put in AUTO	
	☐ e) RCS pressure - LESS THAN 185 PSIG	e) <u>IF</u> two LHSI pumps are running, <u>THEN</u> do the following: ☐ 1) Verify reset or reset SI. ☐ 2) Stop one LHSI pump and put in AUTO. ☐ 3) RETURN TO procedure step in effect. <u>IF</u> one LHSI pump running, <u>THEN</u> RETURN TO procedure step in effect.
	☐ f) LHSI flow - INDICATED	☐ f) Manually start pumps and align valves.

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 71	SYSTEM ALIGNMENT VERIFICATION	PAGE 1 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___	CHECK FW ISOLATION:	☐ Manually close valves and stop pumps.
	• Feed pump discharge MOVs - CLOSED	
☐	• 1-FW-MOV-150A	
☐	• 1-FW-MOV-150B	
☐	• MFW pumps - TRIPPED	
☐	• Feed REG valves - CLOSED	
☐	• SG FW bypass flow valves - DEMAND AT ZERO	
☐	• SG blowdown TVs - CLOSED	
2.___	CHECK CTMT ISOLATION PHASE I:	☐ Manually close valves.
☐	• Phase I TVs - CLOSED	
☐	• 1-CH-MOV-1381 - CLOSED	
☐	• 1-SV-TV-102A - CLOSED	
	• PAM isolation valves - CLOSED	
☐	• 1-DA-TV-103A	
☐	• 1-DA-TV-103B	
3.___	CHECK AFW PUMPS RUNNING:	
☐	a) MD AFW pumps - RUNNING (Time Delayed)	☐ a) Manually start pumps.
☐	b) TD AFW pump - RUNNING IF NECESSARY	b) Manually open steam supply valves.
		☐ • 1-MS-SOV-102A
		☐ • 1-MS-SOV-102B

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 71	SYSTEM ALIGNMENT VERIFICATION	PAGE 2 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.___	CHECK SI PUMPS RUNNING: ☐ • CHG pumps - RUNNING ☐ • LHSI pumps - RUNNING	☐ Manually start pumps.
5.___	CHECK CHG PUMP AUXILIARIES: ☐ • CHG pump CC pump - RUNNING ☐ • CHG pump SW pump - RUNNING	☐ Manually start pumps.
6.___	CHECK INTAKE CANAL: ☐ • Level - GREATER THAN 24 FT ☐ • Level - BEING MAINTAINED BY CIRC WATER PUMPS	☐ IF level is less than 24 ft OR lowering in an uncontrolled manner, THEN initiate 0-AP-12.01, LOSS OF INTAKE CANAL LEVEL.

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 71	SYSTEM ALIGNMENT VERIFICATION	PAGE 3 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.___	CHECK IF MAIN STEAMLINES SHOULD BE ISOLATED:	
	a) Check if ANY of the following annunciators - HAVE BEEN LIT ☐ • E-F-10 (High Steam Flow SI) ☐ • B-C-4 (Hi Hi CLS Train A) ☐ • B-C-5 (Hi Hi CLS Train B)	a) Do the following: ☐ IF annunciator E-H-10 (Hdr/Line SI) LIT, THEN GO TO Step 7d. ☐ IF annunciator E-H-10 NOT LIT, THEN GO TO Step 8.
	☐ b) Check MSTVs - CLOSED	☐ b) Manually close valves.
	c) Check either of the following - ACTUATED ☐ • Hi steam flow SI OR ☐ • Header to line SI	☐ c) GO TO Step 8.
	d) Check RWST crosstie valves - OPEN ☐ • 1-SI-TV-102A ☐ • 1-SI-TV-102B ☐ • 2-SI-TV-202A ☐ • 2-SI-TV-202B	☐ d) Manually open valves.
	☐ e) Check RCS pressure - LESS THAN 185 PSIG	☐ e) Put BOTH RMT mode transfer switches in REFUEL.

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 71	SYSTEM ALIGNMENT VERIFICATION	PAGE 4 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*8. ___	CHECK IF CS REQUIRED:	
	☐ a) CTMT pressure - HAS EXCEEDED 23 PSIA	a) Do the following: 1) IF CTMT pressure has exceeded 17.7 psia, THEN check or align the following valves: ☐ • 1-RM-TV-100A - CLOSED ☐ • 1-RM-TV-100B - CLOSED ☐ • 1-RM-TV-100C - CLOSED ☐ • 1-SV-TV-102 - CLOSED ☐ • 1-IA-TV-101A - CLOSED ☐ • 1-IA-TV-101B - CLOSED ☐ • 1-IA-AOV-103 - OPEN ☐ 2) GO TO Step 10.
	☐ b) Manually initiate HI HI CLS	
	☐ c) Trip all RCPs	
	☐ d) Check CS pumps - RUNNING	☐ d) Manually start pump(s).
	☐ e) Initiate Attachment 4	

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 71	SYSTEM ALIGNMENT VERIFICATION	PAGE 5 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*9.___	CHECK IF RS REQURIED:	
	☐ a) Check RWST level - LESS THAN OR EQUAL TO 60%	a) Do the following: ☐ 1) Continue to monitor RWST level. 2) GO TO Step 10. IF RWST level lowers to less than or equal to 60%, THEN perform Step 9b through Step 9d.
	☐ b) Check ISRS pumps - RUNNING	☐ b) Manually Start Pumps.
	☐ c) Check OSRS pumps - RUNNING (Time Delayed)	☐ c) Manually Start Pumps.
	☐ d) Check OSRS pumps - NOT CAVITATING	☐ d) Put affected OSRS pump in PTL.
*10.___	BLOCK LOW PRZR PRESS SI SIGNAL:	
	☐ a) Check PRZR pressure - LESS THAN 2000 psig	☐ a) GO TO Step 11. WHEN PRZR pressure less than 2000 psig, THEN perform Steps 10b and 10c.
	☐ b) Turn both LO PRZR PRESS & STM HDR/LINE ΔP switches to block	
	☐ c) Check Permissive Status light C-2 - LIT	
*11.___	BLOCK LOW TAVE SI SIGNAL:	
	☐ a) Check RCS Tave - LESS THAN 543°F	☐ a) GO TO Step 12. WHEN Tave less than 543°F, THEN perform Steps 11b and 11c.
	☐ b) Turn both HI STM FLOW & LO TAVG OR LP switches to block	
	☐ c) Check Permissive Status light F-1 - LIT	

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 71	SYSTEM ALIGNMENT VERIFICATION	PAGE 6 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: • CHG pumps should be run in the following order of priority: C, B, A. • Subsequent SI signals may be reset by reperforming Step 12.	
12.___	CHECK SI FLOW:	
	a) HHSI to cold legs - FLOW INDICATED ☐ • 1-SI-FI-1961 (NQ) ☐ • 1-SI-FI-1962 (NQ) ☐ • 1-SI-FI-1963 (NQ) ☐ • 1-SI-FI-1943 or 1-SI-FI-1943A	☐ a) Manually start pumps and align valves. IF flow NOT established, THEN consult with Shift Supervision to establish another high pressure injection flowpath while continuing with this procedure. ☐ • Alternate SI to cold legs ☐ • Hot leg injection
	☐ b) Check CHG pumps - THREE RUNNING	☐ b) GO TO Step 12e.
	☐ c) Reset SI	
	☐ d) Stop one CHG pump and put in AUTO	
	☐ e) RCS pressure - LESS THAN 185 PSIG	e) IF two LHSI pumps are running, THEN do the following: ☐ 1) Check reset or reset SI. ☐ 2) Stop one LHSI pump and put in AUTO. ☐ 3) GO TO Step 13. ☐ IF one LHSI pump running, THEN GO TO Step 13.
	☐ f) LHSI flow - INDICATED	☐ f) Manually start pumps and align valves.

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 71	SYSTEM ALIGNMENT VERIFICATION	PAGE 7 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.___	CHECK TOTAL AFW FLOW - GREATER THAN 350 GPM [450 GPM]	☐ IF SG narrow range level greater than 12% [18%] in any SG, THEN control feed flow to maintain narrow range level AND GO TO Step 14. ☐ IF SG narrow range level less than 12% [18%] in all SGs, THEN manually start pumps AND align valves as necessary. ☐ IF AFW flow greater than 350 GPM [450 GPM] can NOT be established, THEN GO TO 1-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.
14.___	CHECK AFW MOVs - OPEN	☐ Manually align valves as necessary.
15.___	INITIATE SI VALVE ALIGNMENT IAW ATTACHMENT 2	
16.___	INITIATE VENTILATION, AC POWER, AND SFP STATUS CHECKS IAW ATTACHMENT 3	

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE CHECKING SI VALVE ALIGNMENT	ATTACHMENT 2
REVISION 71		PAGE 1 of 2

NOTE: Components previously aligned by SI termination steps, must not be realigned by this Attachment.

1. ___ Check opened or open CHG pump suction from RWST MOVs.
   - ☐ • 1-CH-MOV-1115B
   - ☐ • 1-CH-MOV-1115D

2. ___ Check closed or close CHG pump suction from VCT MOVs.
   - ☐ • 1-CH-MOV-1115C
   - ☐ • 1-CH-MOV-1115E

3. ___ Check running or start at least two CHG pumps. (listed in preferred order)
   - ☐ • 1-CH-P-1C
   - ☐ • 1-CH-P-1B
   - ☐ • 1-CH-P-1A

4. ___ Check opened or open HHSI to cold legs MOVs.
   - ☐ • 1-SI-MOV-1867C
   - ☐ • 1-SI-MOV-1867D

5. ___ Check closed or close CHG line isolation MOVs.
   - ☐ • 1-CH-MOV-1289A
   - ☐ • 1-CH-MOV-1289B

6. ___ Check closed or close Letdown orifice isolation valves.
   - ☐ • 1-CH-HCV-1200A
   - ☐ • 1-CH-HCV-1200B
   - ☐ • 1-CH-HCV-1200C

7. ___ Check opened or open LHSI suction from RWST MOVs.
   - ☐ • 1-SI-MOV-1862A
   - ☐ • 1-SI-MOV-1862B

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 2
REVISION 71	CHECKING SI VALVE ALIGNMENT	PAGE 2 of 2

8. ___ Check opened or open LHSI to cold legs MOVs.

☐ • 1-SI-MOV-1864A
☐ • 1-SI-MOV-1864B

9. ___ Check running or start at least one LHSI pump.

☐ • 1-SI-P-1A
☐ • 1-SI-P-1B

10. ___ Check High Head SI flow to cold legs indicated.

☐ • 1-SI-FI-1961
☐ • 1-SI-FI-1962
☐ • 1-SI-FI-1963
☐ • 1-SI-FI-1943 or 1-SI-FI-1943A

11. ___ IF flow not indicated, THEN manually start pumps and align valves. IF flow NOT established, THEN consult with Shift Supervision to establish another high pressure injection flow path while continuing with this procedure.

☐ • Alternate SI to Cold legs
☐ • Hot leg injection

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 71	AUXILIARY VENTILATION, AC POWER, AND SFP STATUS CHECKS	PAGE 1 of 6

1. ___ Check or place REFUEL SFTY MODE switches in NORMAL.

2. ___ Check ventilation alignment IAW Tables 1 and 2.

TABLE 1
UNIT #1 VENTILATION PANEL

	MARK NUMBER	EQUIPMENT STATUS
☐	1-VS-F-4A & B	OFF
☐	1-VS-HV-1A & B	OFF
☐	1-VS-F-8A & B	OFF
☐	1-VS-F-9A & B	GREEN
☐	1-VS-F-59	GREEN
☐	1-VS-F-6	OFF
☐	1-VS-F-39	GREEN
☐	1-VS-F-7A & B	GREEN
☐	1-VS-HV-5	GREEN
☐	1-VS-F-56A & B	GREEN
☐	1-VS-F-40A & B	GREEN
☐	1-VS-HV-4	OFF
☐	2-VS-F-40A or B	RED
☐	2-VS-HV-4	OFF

Event Description: FOLDOUTS AND ATTACHMENTS

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TABLE 2
VNTX PANEL

	MARK NUMBER	EXPECTED EQUIPMENT STATUS		RESPONSE NOT OBTAINED
☐	a. AOD-VS-107A & B AOD-VS-108	RED GREEN	☐	a. Place AUX BLDG CENTRAL AREA MODE switch to FILTER.
☐	b. MOD-VS-100A & B AOD-VS-106	RED GREEN	☐	b. • Place MOD-VS-100A to FILTER. • Place MOD-VS-100B to FILTER.
☐	c. MOD-VS-200A & B AOD-VS-206	GREEN RED	☐	c. • Place MOD-VS-200A to UNFILTER. • Place MOD-VS-200B to UNFILTER.
☐	d. AOD-VS-103A & B AOD-VS-104	GREEN GREEN	☐	d. • Place AOD-VS-103A in UNFILTER. • Place AOD-VS-103B in UNFILTER. • Place AOD-VS-104 in FILTER.
☐	e. AOD-VS-101A & B AOD-VS-102	GREEN GREEN	☐	e. Place AOD-VS-101A and 101B in UNFILTER.
☐	f. AOD-VS-111A & B	GREEN	☐	f. Place COMBINE CONTAINMENT EXHAUST in ISOLATE.
☐	g. AOD-VS-110	GREEN	☐	g. Place AOD-VS-109A and 109B in FILTER.
☐	h. AOD-VS-112A & B	GREEN	☐	h. • Place AOD-VS-112A in CLOSE. • Place AOD-VS-112B in CLOSE.
☐	i. MOD-VS-58A & B 1-VS-F-58A & B	RED RED	☐	i. Start 1-VS-F-58A and 1-VS-F-58B.

3. ___ Check filtered exhaust flow: (as read on FI-VS-117A and FI-VS-117B)

- ☐ • Total flow - GREATER THAN 32400 cfm

AND

- ☐ • Flow through each filter bank - LESS THAN 39600 cfm

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 71	AUXILIARY VENTILATION, AC POWER, AND SFP STATUS CHECKS	PAGE 3 of 6

4. ___ Check all Station Service Buses - ENERGIZED. IF NOT, THEN initiate 1-AP-10.07, LOSS OF UNIT 1 POWER.

5. ___ Check annunciator VSP-J2 - LIT.

6. ___ Check Unit 1 RSST LTC time delay bypass light - LIT.

7. ___ Check stopped or stop 1-VS-AC-4.

8. ___ Place 1-VS-43-VS103X, MCR ISOLATION switch to the OFF position.

9. ___ Check closed or close MCR isolation dampers.

- ☐ • 1-VS-MOD-103A
- ☐ • 1-VS-MOD-103B
- ☐ • 1-VS-MOD-103C
- ☐ • 1-VS-MOD-103D

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
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* *

CAUTION:
- Only one Emergency Supply Fan must be started in the following step.
- Chilled Water flow to the in-service Unit 1 MCR AHU must be throttled to at least 15 gpm when the Emergency Supply fan is started.
- Chilled Water flow to the in-service Unit 2 MCR AHU must be throttled to at least 25 gpm when the Emergency Supply fan is started.
- An Emergency Supply Fan must not be started if the filter is wet.

* *

10. Immediately start ONE Emergency Supply Fan IAW the following: (1-VS-F-41 or 2-VS-F-41 preferred)

a. IF 1-VS-F-41, CONT RM EMERG SUP FAN, will be used, THEN perform the following substeps.

___ 1. Open 1-VS-MOD-104A, CONT RM EMERG SUP MOD.

___ 2. Start 1-VS-F-41.

b. IF 2-VS-F-41, CONT RM EMERG SUP FAN, will be used, THEN perform the following substeps.

___ 1. Open 2-VS-MOD-204A, CONT RM EMERG SUP MOD.

___ 2. Start 2-VS-F-41.

c. IF 1-VS-F-42, CONT RM EMERG SUP FAN, will be used, THEN perform the following substeps.

___ 1. Open 1-VS-MOD-104B, CONT RM EMERG SUP MOD.

___ 2. Start 1-VS-F-42.

d. IF 2-VS-F-42, CONT RM EMERG SUP FAN, will be used, THEN perform the following substeps.

___ 1. Open 2-VS-MOD-204B, CONT RM EMERG SUP MOD.

___ 2. Start 2-VS-F-42.

e. ___ Adjust Chilled Water flow to MCR AHUs IAW Step 10 Caution.

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
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11. ___ Check readings on the following Differential Pressure Indicators - POSITIVE PRESSURE INDICATED.
    - ☐ • PDI-VS-100, D.P.-U1CR/U1TB (Unit 2 Turbine Ventilation Panel)
    - ☐ • PDI-VS-101, D.P.-U1RR/U1TB (Unit 2 Turbine Ventilation Panel)
    - ☐ • PDI-VS-200, D.P.-U2CR/U2TB (Unit 2 Turbine Ventilation Panel)
    - ☐ • PDI-VS-201, D.P.-U2RR/U2TB (Unit 2 Turbine Ventilation Panel)
    - ☐ • 1-VS-PDI-118 (Unit 1 Computer Room)
    - ☐ • 1-VS-PDI-116 (Near Unit 1 Semi-Vital Bus)
    - ☐ • 2-VS-PDI-215 (Unit 2 AC Room)
    - ☐ • 2-VS-PDI-206 (Near Unit 2 Semi-Vital Bus)

12. ___ IF any reading NOT positive, THEN initiate Attachment 6 to secure MCR boundary fans.

13. ___ Check initiated or initiate 0-AP-50.00, OPPOSITE UNIT EMERGENCY.

14. ___ Check the following MCR and ESGR air conditioning equipment operating. IF NOT, THEN start equipment within 1 hour IAW the appropriate subsection of 0-OP-VS-006, CONTROL ROOM AND RELAY ROOM VENTILATION SYSTEM.
    - ☐ • One Control Room chiller
    - ☐ • One Unit 1 Control Room AHU
    - ☐ • One Unit 2 Control Room AHU
    - ☐ • One Unit 1 ESGR AHU
    - ☐ • One Unit 2 ESGR AHU

15. ___ IF both of the following conditions exist, THEN check that Load Shed is activated.
    - ☐ • Unit 2 - SUPPLIED BY RSST
    - ☐ • Unit 2 RCPs - RUNNING

16. ___ IF Load Shed is required and not activated, THEN initiate 0-AP-10.10, LOSS OF AUTO LOAD SHED.

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 71	AUXILIARY VENTILATION, AC POWER, AND SFP STATUS CHECKS	PAGE 6 of 6

NOTE:
- SFP checks should be initiated WITHIN ONE TO TWO HOURS of EOP entry.
- Loss of power may render SFP indications and alarms non-functional and require local checks. Power supplies are as follows:
  - TI-FC-103, Unit 1 Semi-Vital Bus
  - TI-FC-203, Unit 2 Semi-Vital Bus
  - 1-FC-LIS-104, Panel 1ABDA1
- Loss of AC Power to the SFP level indicator is indicated if both low and high level alarms are in simultaneously. (0-VSP-C4 and 0-VSP-D4)
- 1-DRP-003, CURVE BOOK, provides a graph for SFP time to 200°F if loss of SFP cooling occurs.

17. ___ Initiate monitoring SFP parameters:
    - ☐ • SFP level - Greater than Cooling Pump suction AND Stable
    - ☐ • SFP temperature - Stable or Lowering
    - ☐ • SFP Cooling Pumps - Either Running
    - ☐ • Component Cooling - Normal
    - ☐ • SFP Radiation - Normal

18. ___ Continue to monitor parameters every one to two hours or until authorized to terminate monitoring by the Station Emergency Manager and/or the Shift Manager.

19. ___ Notify the Station Emergency Manager and/or the Shift Manager of the status and trend of SFP parameters.

20. ___ IF any abnormality or adverse trend is identified, THEN initiate 0-AP-22.02, MALFUNCTION OF SPENT FUEL PIT SYSTEMS.

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 7
REVISION 71	TRANSIENT AFW FLOW CONTROL	PAGE 1 of 1

1. ___ Check SI is in progress. IF SI NOT in progress, THEN RETURN TO procedure step in effect.

2. ___ Check running or start AFW Pumps, as necessary.
   - ☐ • 1-FW-P-3A
   - ☐ • 1-FW-P-3B
   - ☐ • 1-FW-P-2

3. ___ Maintain at least 350 gpm [450 gpm] AFW flow untill one SG Narrow Range Level is greater than 12% [18%].

NOTE: AFW to idle loop(s) (RCP secured), should be throttled to prevent depressurization of the SG and subsequent Header / Line SI. AFW flow between approximately 60 gpm and 100 gpm should be adequate to prevent a Header / Line SI.

4. ___ When minimum heat sink has been verified, AFW MOVs should be controlled to maintain intact SG Narrow Range levels between 22% and 50% by throttling AFW Isolation MOVs:
   - ☐ • SG A, 1-FW-MOV-151E and 1-FW-MOV-151F
   - ☐ • SG B, 1-FW-MOV-151C and 1-FW-MOV-151D
   - ☐ • SG C, 1-FW-MOV-151A and 1-FW-MOV-151B

5. ___ Isolate AFW header with deenergized Emergency Bus MOVs by closing the following header isolation valves:
   - ☐ • Emergency Bus H deenergized: 1-FW-141, 1-FW-156, 1-FW-171
   - ☐ • Emergency Bus J deenergized: 1-FW-140, 1-FW-155, 1-FW-170

Event Description: FOLDOUTS AND ATTACHMENTS

.

NUMBER 1-E-0	CONTINUOUS ACTIONS PAGE	REVISION 71

1. RCP TRIP CRITERIA

   Trip all RCPs if BOTH conditions listed below occur:

   a. Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS
   b. RCS Subcooling - LESS THAN 30°F [85°F]

2. MINIFLOW RECIRC CRITERIA

   a. CLOSED - When RCS pressure is less than 1275 psig [1475 psig] AND RCP Trip Criteria are met (RCPs OFF).
   b. OPEN - When RCS pressure is greater than 2000 psig.

3. ADVERSE CONTAINMENT CRITERIA

   Use Adverse Containment setpoints if EITHER condition listed below occurs:

   - Containment Pressure - GREATER THAN 20 PSIA
   - Containment Radiation - GREATER THAN 1.0E5 R/HR

4. COLD LEG RECIRCULATION SWITCHOVER CRITERIA

   GO TO 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, if RWST level lowers to less than 20%.

---

1. AMSAC RESET CRITERIA

   AMSAC may be manually reset when level in all three SGs is greater than 13% or six minutes have elapsed since the Reactor trip. When AMSAC is reset, AMSAC ARMED annunciator H-D-1 should clear and affected components may be realigned as needed.

2. TD AFW PUMP SHUTDOWN CRITERIA

   The TD AFW pump may be secured when SG NR level is greater than 22% in at least 2 SGs, AMSAC is reset, and no auto-start signal exists. To secure the pump, the pump SOV control switches must be taken to OPEN-RESET and then to CLOSE.

3. MANUAL SI ALIGNMENT

   If SI fails to automatically align, Attachment 2 may be used for guidance on manual SI valve alignment.

4. * TRANSIENT AFW FLOW CONTROL (IF SI in progress)

   Attachment 7 may be used for guidance on transient AFW flow control.

5. * FAULTED SG ISOLATION AND AFW FLOW CONTROL (IF SI in progress)

   Attachment 8 may be used for guidance on faulted SG(s) isolation and AFW flow control.

6. * RUPTURED SG ISOLATION AND AFW FLOW CONTROL (IF SI in progress)

   Attachment 9 may be used for guidance on ruptured SG(s) isolation and AFW flow control.

7. * LOSS OF RCP SUPPORT CONDITIONS

   Trip RCPs if a loss of a support condition occurs. (for example, loss of CC)

* Preemptive Actions

Event Description: FOLDOUTS AND ATTACHMENTS

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NUMBER 1-E-0	CONTINUOUS ACTION STEPS	REVISION 71

1. Check RCS Average Temperature - STABLE AT OR TRENDING TO 547°F. (E-0, Step 6)
2. Monitor RCP Trip and Miniflow Recirc Criteria. (E-0, Step 8)
3. Check SG Narrow Range Level - ANY SG GREATER THAN 12%. (Control feed flow to maintain Narrow Range Level between 22% and 50%) (E-0, Step 25)
4. Monitor LHSI pumps and secure as necessary. (E-0, Step 30)

NOTE: Subsequent SI signals may be reset by reperforming Step 12 of Attachment 1.

5. Monitor CTMT pressure and check CLS initiation as necessary. (Attachment 1, Step 8)
6. Monitor RWST level and check RS initiation as necessary. (Attachment 1, Step 9)
7. Block Low PRZR Pressure SI signal when less than 2000 psig. (Attachment 1, Step 10)
8. Block Low Tave SI signal when less than 543°F. (Attachment 1, Step 11)

SIMULATOR OPERATOR'S GUIDE

Simulator Setup

Initial Conditions:

Recall IC -373 **and implement TRIGGER #30 to activate all passive malfunctions and verify Trigger #30 implemented.**

Open the monitor window and add the following points to it:

- asp_ao_off

Enter/Verify the following MALFUNCTIONS:

Malfunction	Delay	Ramp	Trigger	Value	Final	Trigger Type (Auto or Manual)
MS0806 SG C STM FLOW TRNSMTR MS-FT-595 FAILURE	10	60	1		1	MAN
NI1004 POWER RANGE CHNL N44 FAILURE	10	30	3		1	MAN
CH2101 VCT LEVEL TRANSMITTER LT-1112 FAILS	10	30	5		1	MAN
SD0202 HP HTR DRN PP SD-P-1B TRIPS:OVR-CURRENT	10		7		TRUE	MAN
FW09 FW-P-2 TRIP/THROTTLE VALVE TRIPS CLOSED			9		TRUE	AUTO
MS0102 'B' MAIN STM LINE RUPTURE AT HEADER	10	3:00	11		20	MAN
RC5601 RC-P-1A BKR SPURIOUS TRIP	4:00		11		TRUE	MAN
RC5602 RC-P-1B BKR SPURIOUS TRIP	4:15		11		TRUE	MAN
RC5603 RC-P-1C BKR SPURIOUS TRIP	4:30		11		TRUE	MAN
RD17 ATWS WITH MANUAL RX TRIP PB DEFEATED			30		TRUE	MAN
MS22 DISABLE A MSTV AUTO CLOSURE ON HSF			30		TRUE	MAN
MS23 DISABLE B MSTV AUTO CLOSURE ON HSF			30		TRUE	MAN
MS24 DISABLE C MSTV AUTO CLOSURE ON HSF			30		TRUE	MAN
FW48 DISABLE AFW P3A AUTO START			30		TRUE	MAN
FW49 DISABLE AFWP3B AUTO START			30		TRUE	MAN
FP0301 FPS FACP07 ALARM HORN FAILURE			30		TRUE	MAN
FP0302 FPS PC SPEAKER FAILURE			30		TRUE	MAN

SIMULATOR OPERATOR'S GUIDE

Enter the following Remote Functions

Description	Delay	Ramp	Trigger	Value	Final	Trigger Type (Auto or Manual)
RX_RTA_OPEN RX-RTA_OPEN REACTOR TRIP BREAKER	2:00		15		TRUE	M
RX_RTB_OPEN RX-RTB_OPEN REACTOR TRIP BREAKER	2:00		15		TRUE	M
SIP1B_BKRPOS SIP1B_BKRPOS SI-P-1B BKR 14J3 CUBICLE POSITION			30		RACK OPEN	M

Enter the following EVENT TRIGGERS:

Trigger#	EVENT	Command
9	fwp2_spd .gt. 0.5	Trigger 9 goes active

TRIGGER	TYPE	DESCRIPTION
1	Manual	SG "C" Steam Flow transmitter FT-495 fails high.
3	Manual	Power Range NI channel 44 fails high
5	Manual	VCT Level transmitter LT-1112 fails high
7	Manual	HP Heater Drain pump 1-SC-P-1B trips on overcurrent.
9	Auto	AFW pump P-2 Trip throttle valve fails closed.
11	Manual	"B" Main Steam line rupture at header (Turbine bldg..).
15	Manual	Locally open Reactor Trip Breaker RTA, and RTB.
30	Manual	1-SI-P-1B Breaker racked out.
30	Manual	FPS alarms fails.
30	Manual	Disable Main Steam Trip valves Auto closure on HSF
30	Manual	ATWS with failure of RX Trip PB.
30	Manual	Disable AFW Pump P-3A, and P-3B, auto starts.

SIMULATOR OPERATOR'S GUIDE

Verify the following control room setup:

- ❑ Place the simulator in RUN and verify normal 100% power operation indications.
- ❑ Reset the ROD CONTROL SYSTEM
- ❑ Verify Controlling channels selected to **IV.**
- ❑ Verify Red Magnets on the following components:

1-SI-P-1B

- ❑ Verify 1-RM-RI-112 aligned to A/B SG and 1-RM-RI-113 aligned to C SG (magnets).
- ❑ Verify Ovation System operating.
- ❑ Reset ICCMs.
- ❑ Verify Component Switch Flags.
- ❑ Verify Brass Caps properly placed.
- ❑ Verify SG PORVs set for 1035 psig.
- ❑ Verify Rod Control Group Step Counters indicate properly.
- ❑ Verify Ovation CRT display.
- ❑ Advance Charts
- ❑ Verify Turbine Thumb Wheel Settings @120 rpm/min and Position 6
- ❑ Verify Containment Instrument Air Compressors are on Inside Suction (all RMs reset)
- ❑ Verify all ARPs have been cleaned

❑ 1E-H5	❑ 1F-C9	❑ 1G-E4	❑ 1G-G1
❑ 1G-G4	❑ 1H-D3	❑ 1H-E7	❑ 1H-F3
❑ 1H-G5	❑ 1H-G6	❑ 1H-G7	❑ 1J-B4
❑ 1J-E1	❑ 1J-F1	❑ 1K-D4	❑

- ❑ Verify CLEAN copies of the following procedures are in place:

❑ 1-OP-51.5.3	❑ 0-AP-53.00 (3)	❑ 1-AP-4.00	❑ 1-AP-18.00
❑ 0-AP-23.00	❑ 1-E-0	❑ 1-FR-S.1	❑ 1-E-2

SIMULATOR OPERATOR'S GUIDE

- ❑ **Verify Reactivity Placard is current.**
- ❑ Verify ALL PINK MAGNETS are accounted for.
- ❑ Reset Blender Integrators for Boric Acid to 100 and PG 1000.

SIMULATOR OPERATOR'S GUIDE

Brief

This simulator performance scenario is performed in the EVALUATION MODE. You should not direct questions to the evaluators. Otherwise, you should perform as if you were in the MCR.

Your ability to maintain a log is not being graded, but maintaining a rough log is recommended to help during briefs.

If you need to communicate with the Unit 2 operator, verbally state, "Unit 2" and an instructor will locate to the Unit 2 area and respond to you as quickly as possible.

In the unlikely event that the simulator fails such that illogical indications result, the session will be terminated. In other words, respond to what you see. If there is a problem with the simulation, the session will be terminated or adjusted as appropriate based on the specific problem.

Assign operating positions.

Ask for and answer questions.

SIMULATOR OPERATOR'S GUIDE

Conduct shift turnover:

The initial conditions have Unit 1 at 100% power with RCS boron concentration of 760 ppm.

Unit conditions have been stable at approximately 100% power since the last refueling outage.

All systems and crossties are operable with the following exception:

- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.
- 1-SI-P-1B, "B" LHSI Pump, Tagged out for breaker PMs. Tech Spec 3.3.B.3 requires return to service within 72 hours. Expected completion of breaker PMs-later this shift.
- Controlling channels have been shifted to CH IV in preparation for Channel III testing

Unit #2 is at 100% power with all systems and crossties operable.

The last shift performed two 30 gallon dilutions, followed by manual makeups.

Shift orders are to maintain 100% power on Unit 1, and shift the running CH Pump SW pumps IAW 1-OP-51.5.3, Section 6.1.

When the team has accepted the shift, proceed to the Session Conduct Section.

SIMULATOR OPERATOR'S GUIDE

Session Conduct:

- Ensure conditions in Simulator Set-up are established.
- **Ensure Trigger 30 is active prior to team entering the simulator.**
- Verify Exam Security has been established and ASP_AO_OFF = True.

SIMULATOR OPERATOR'S GUIDE

EVENT 1 **Shifting Charging Pump SW pumps**

BOOTH: **When cued by the Evaluator place simulator in RUN if it isn't already in RUN**.

Field Operators: (*2 minute delay from request to answer*)

- **If contacted** regarding the Charging Pump SW pump shift, report that you have been briefed, and are standby by.
- **If contacted** following the shift of the Charging pump SW pumps, report that 1-SW-P-10B is running normally and 1-SW-P-10A has normal after stop indications.

Chemistry

- **If contacted** regarding the Charging pump SW pump alignment, acknowledge the pump alignment.

Role play as other individuals as needed.

SIMULATOR OPERATOR'S GUIDE

EVENT 2 "C" SG Steam Flow Channel fails low

When cued by examiner, implement Trigger #1.

Operations Supervisor/Management:

- **If contacted**, acknowledge the failure of the steam flow channel.
- **If contacted,** will notify I&C of the failure, will notify the OMOC.
- **When notified**: acknowledge but do not imply agreement with Tech Spec requirements as identified by the SRO.
- **If contacted**: will take responsibility for submitting CR.
- **If contacted**: Ask for recommendation concerning shifting channels IAW 1-OP-RP-001, agree to wait until I&C is ready to place the channel in trip if recommended.

STA:

- **If contacted**, acknowledge the failure of the steam flow channel.
- **If the team has a transient brief:** The STA will have no input for the brief.

I&C:

- **If requested**: will prepare for placing the channel in trip.

Maintenance/ Work Week Coordinator:

- **If contacted**, will the notify I&C of the channel failure, place the channel in trip, and initiate investigation of the failure,

Role play as other individuals as needed.

SIMULATOR OPERATOR'S GUIDE

EVENT 3 **PR Channel N-44 Fail High**

When cued by examiner, and rod control in automatic, implement **Trigger #3**.

Operations Supervisor/Management:

- **If contacted**, acknowledge N-44 failure.
- **If contacted,** will notify I&C of the failure, will notify the OMOC.
- **When notified**: acknowledge but do not imply agreement with Tech Spec requirements as identified by the SRO.
- **If contacted,** will take responsibility for writing the CR.
- **If asked:** will notify Reactor Engineering of need to perform flux map.
- **If asked:** SM will confer with the OMOC concerning continuing the ramp.

STA:

- **If contacted**, acknowledge Tech Spec requirements for the failure, but do not imply agreement with requirements identified by the SRO.
- **If the team has a transient brief:** The STA will have no input for the brief.
- **If asked:** will notify Reactor Engineering of need to perform flux map.

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge instrumentation failure and commence investigations and/or efforts to place the channel in trip.

Role play as other individuals as needed.

SIMULATOR OPERATOR'S GUIDE

EVENT 4 VCT Level Transmitter, LT-1112 Fails High.

When cued by examiner, implement **Trigger #5.**

Operations Supervisor/Management:

- **If contacted**, will acknowledge the failure of 1-CH-LT-1112 high.
- **If contacted,** will take responsibility for writing the CR.

STA:

- **If contacted**, will acknowledge the failure of 1-CH-LT-1112 high.
- **If contacted,** acknowledge the direction to review; Tech Specs, TRM, VPAP-2802, RG 1.97, EP-AA-303, and CEP 99-0029.
- **After directed,** the STA will report that all reviews have been completed and discussed with the Shift Manager.
- **If contacted,** will take responsibility for writing the CR.
- **If the team has a transient brief:** The STA will have no input for the brief.

Field Operators: (Wait 4 minutes from time of direction to report).

- **If contacted**, to check the status of 1-CH-LT-1112 locally, report no obvious abnormal condition with the transmitter.

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge the failure and commence investigations.

Unit 2 Operator:

- **If contacted**, will acknowledge the failure of 1-CH-LT-1112 high.

Role play as other individuals as needed.

SIMULATOR OPERATOR'S GUIDE

EVENT 5 Loss of HPD Pump, 1-SD-P-1B. (1-AP-18.00)

When the Evaluator indicates Ready, Activate Trigger #7.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the trip of 1-SD-P-1B, and the need to ramp at 1%/minute to 75% power.
- **If asked** for a recommended ramp rate, ask what the Unit Supervisor recommends. When authorized by the NRC, the Shift Manager will direct a 1%/minute ramp rate.

STA:

- **If contacted**, will acknowledge the trip of 1-SD P-1B and the need to ramp at 1%/minute to 75% power.
- **If asked,** the STA will acknowledge the need to borate and use rods (will acknowledge the team review of pre-planned reactivity plans and OP-RX-010, if performed). If asked to perform the OP-RX-010 review, the STA will state that he is not able to at this time.
- **After directed,** the STA will report that he has reviewed VPAP-2802 and no notifications were required.
- **If contacted,** will take responsibility for writing the CR.
- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.
- **When contacted:** EPIPs have been reviewed with the Shift Manager.

Maintenance/ Work Week Coordinator:

- **If contacted**, will acknowledge the failure and notify electrical maintenance to investigate.
- **If contacted,** will acknowledge the requirements to reduce reactor power.

SIMULATOR OPERATOR'S GUIDE

EVENTS 6 Ramp to < 75% power.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the ramp to 70% - 74%.
- **If contacted,** will take responsibility for writing the CR.
- **If asked** for a recommended ramp rate, ask what the Unit Supervisor recommends. When authorized by the NRC, the Shift Manager will direct a 1%/minute ramp rate.

Unit 2 Operator:

If notified, acknowledge the failure and impending ramp of Unit 1.

STA:

- **If contacted**, will acknowledge the Reactivity Plan reported by the RO.
- **If contacted,** will take responsibility for writing the WR and CR.
- **If the team has a transient brief:** The STA will, "nothing to add".
- If contacted, STA review of VPAP-2802 complete, reviewed with Shift Manager, no notifications required.

Maintenance/ Work Week Coordinator:

- **If contacted**, will acknowledge the requirements to reduce reactor power.

Chemistry

- **If contacted,** acknowledge the ramp.

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Field Operators:

- **If contacted**, as the Turbine Building Operator to walkdown the Turbine during the ramp, acknowledge the direction.
- **If contacted,** as the polishing building operator, acknowledge the direction to monitor polisher DP.

Role play as other individuals as needed.

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EVENT 7 Main Steamline Rupture with ATWS.-

When cued by examiner, implement **Trigger #11**.

Booth Note:

- Verify Trigger 9 automatically initiate following TD AFW pump start to cause Trip Throttle valves to close.
- Verify that RCPs trip following time delay (A-4:00, B-4:15, C-4:30)

Critical Tasks:

CT-3: Manually start AFW pumps as needed before SG WR level decreases to <7% on both intact S/G.

CT-4: Close Main Steam Trip valves prior to reaching an orange path on Integrity (FR-P.1).
Critical Task **begins** when the Main Steamline rupture occurs and **ends** when the Main Steam Trip valves are closed.

Security Roving Patrol

- **Call after 2 minutes** and report the following on **Channel 5**: "There is a lot of steam in Unit 1, there appears to be a steam break somewhere."

Operations Supervisor/Management:

- **If contacted:** Acknowledge MSL rupture and agree to notify the OMOC.
- **If contacted:** Acknowledge Failure to automatically or manually Trip the Reactor agree to notify the OMOC.

Unit 2 Operator:

- **If asked**, RWST cross-ties on Unit 2 are closed.
- **If asked,** Simulate manually opening Unit 2 RWST cross-tie valves.
- **If asked,** External MCR D/P indicators indicate the same as indicated pressure on Unit 2 Vent Panel.
- **If requested**, Chilled Water flows have been adjusted per caution prior to Step 10 of E-0, Attachment 3.
- **If contacted,** Unit Two has implemented AP-50.00, and all conditions on U2 are normal.
- **When:** BOP reaches Page 6 of E-0, Attachment 3, you will take responsibility for the Attachment at this point.
- **If asked:** Unit 1 main steam and condenser A/E radiation is normal.

Field Operators:

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- **If directed,** to locally open the Unit 1 Rx. Trip and Bypass breakers, **initiate Trigger 15** (this will open Rx Trip breakers following 2 min time delay). Report to MCR that Reactor trip and bypass breakers are open.

Maintenance/ Work Week Coordinator:

- **If contacted**, will acknowledge the failure(s), contact Maintenance to commence investigation

Role play as other individuals as needed.

The scenario will end upon entering 1-ES-1.1 or at the lead examiners discretion.

Scenario #2

Facility: Surry Power Station   Scenario No.: 2   Op-Test No.: 2016-002
Examiners: ______________________   Operators: ______________________

Initial Conditions: Unit 1 and 2 at 100% power; MOL 760 ppm Boron.

- C CH running on Alt, A CH in Auto, B CH pump Tagged out for breaker PMs.
- Controlling channels are aligned to CH IV for channel III testing.
- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.

Turnover: The Team will brief the swap of the BC pumps IAW 1-OP-BC-001, Section 5.3, prior to Simulator entry.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N (BOP/SRO)	Swap Running BC pumps IAW 1-OP-BC-001, Section 5.3
2	RC4901 +1	I (RO/SRO) TS (SRO)	PRZR Level CH 1 Fails High (AP-53.00)
3	FW1804 -1	I (BOP/SRO) TS (SRO)	B SG CH IV FF Fail Low (AP-53.00) **(CT-1)**
4	CN0102 TRUE CN1501 TRUE	C (BOP/SRO)	1-CN-P-1B Trip overcurrent 1-CN-P-1A Auto Start Fails **(CT-2)**
5	RD1244 TRUE	C (RO/SRO) TS (SRO)	Dropped RCCA, P-8, Control Bank D (AP-1.00)
6	N/A	R (RO/SRO) N (BOP)	Ramp Unit to < 75%, due to dropped rod
7	RC04 16.0%	M (ALL)	RCS Leak ~ 200 GPM (AP-16.00, E-0 with SI)
8	RC0103, 5.0%-30% SI14001, SI14002, CH4601, CH0504	M(ALL)	LBLOCA, E-1, FR-Z.1 Failure of SI and HI-HI-CLS **(CT-3)**

9	CH59 CH6401 RS1001 RS1002 CS0801	Team Failures	1-CH-MOV-1381 Auto Close Failure (BOP) 1-CH-P-1A Auto Start Failure (BOP) HI HI CLS FAIL to ACTUATE Train A **(CT-4)** HI HI CLS FAIL to ACTUATE Train B 1-CS-P-1A Trip on Overcurrent
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor			

Event 1: Swap Running BC Pumps IAW 1-OP-BC-001, Section 5.3). (N- BOP/SRO)

This evolution will be pre-briefed by the Team prior to Simulator Entry. 1-BC-P-1B, "B" BC Pump, initially running; 1-BC-P-1A, "A" BC Pump, secured. At end of evolution, the "A" BC Pump will be running in Auto, the "B" BC Pump secured. The RO will peer check the switch manipulations and monitor PCS for abnormal indications. The SRO will supervise the evolution.

Verifiable Action(s):

1) BOP – Manipulate BC Pump control switches to start "A' BC Pump and Secure "B" BC pump.

Technical Specifications/TRM/Reg Guide 1.97/VPAP-2802:

None.

Event 2: PRZR Level CH 1 Fail High (Selected Upper Channel). (I – RO, TS – SRO).

When the Evaluation Team is ready, the malfunction will be actuated. The malfunction will cause PRZR CH 1 to Fail High. The RO is expected to diagnose the failure based on alarms and indications received and take manual control of CH flow.

Verifiable Actions(s):

1) RO – Manual Control of CH flow.
2) RO – Change the position of the PRZR Level Channel Selector Switch to Position 3 (CH3/CH2).
3) RO – Restore CH flow to Auto.

Technical Specifications:

1) Tech Spec 3.7, Table 3.7-1, Item 9, Operator Action 7, With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied: The inoperable channel is placed in the tripped condition within 72 hours, the inoperable channel may be bypassed for up to 12 hours for surveillance testing, If the conditions are not satisfied in the time permitted, reduce power to less than the P-7 setpoint within the next 6 hours.
2) Tech Spec Table 3.7-6, Item 13, 2 Pressurizer level, 2 Channels required. Tech Spec is met.

TRM Actions:

- TRM 3.3.2.A.1, and A.2: Less than the minimum Primary Instruments or the minimum Alternate Instruments in Table 3.3.2-1 are functional. **RA A.1**: Implement a fire watch in the Cable Vault and Tunnel and the Emergency Switchgear room of the affected unit in accordance with TRM Section 5.2 within 14 days. **RA A.2**: Restore instrument to functional status within 60 days.

Reg. Guide 1.97:

- 1-RC-LOOP-1459, A-13 Variable, 2 Channels required
- 2 Channels required, Refer to TS Table 3.7-6.
- RG 1.97 requirements are a SM/STA function **(recommend evaluating item as a follow-up at Evaluator discretion, Post scenario)**.

VPAP-2802, Reportability

None.

Equipment Important to Emergency Response (EP-AA-303).

1-RC-LI-1459A, PRZR Level Chann1l 1, Category B, Verify alternate indication available, 2 Operable channels are available, requirement met. (Requirement is a SM/STA function (recommend evaluating item as a follow-up at Evaluator discretion, Post scenario)

***Event 3: B SG* Feed Flow Channel IV Fail Low.** (I-BOP/SRO, TS-SRO)

When the Evaluation Team is ready, the malfunction is implemented. This malfunction will cause the channel IV, selected cannel feed flow to fail low on the B SG. The BOP is expected to diagnose the failure based on alarms and indications received and perform the Immediate Actions of 0-AP-53.00, Loss of Vital Instrumentation/Controls; place the B FRV in manual, and adjust feed flow to control B SG NR level at program.

Verifiable Action(s):

1) BOP – place the B SG FRV in manual and control feed flow to the B SG.
2) RO – Swap controlling channels from channel IV to channel III.
3) BOP – When B SG NR level has been returned to program, return the B FRV to automatic.

Critical Task:

(CT-1): If the BOP fails to take timely action in response to the FF channel failure, an automatic reactor trip on SG NR high level will occur; an unanticipated reactor trip should be considered as failure criteria. An Operations representative will be available for consultation as necessary.

Technical Specifications:

Tech Spec Table 3.7-1, Item 17 Low SG level with SF/FF mismatch, Operator Action 6. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied: The inoperable channel is placed in the tripped condition within 72 hours, the inoperable channel may be bypassed for up to 12 hours for surveillance testing, If the conditions are not satisfied in the time permitted, be in at least Hot Shutdown within 6 hours.

Technical Requirements Manual.

None.

Reg. Guide 1.97:

1) Main Feed flow, 1-FW-LOOP-F-1486, D-20 variable, 1 channel per SG required, requirement is met.
2) RG 1.97 requirements are a SM/STA function (recommend evaluating item as a follow-up at Evaluator discretion, Post scenario)

VPAP-2802, Reportability:

None.

Equipment Important to Emergency Response (EP-AA-303).

None.

Event #4: 1-CN-P-1B Trip ("B" CN Pump), 1-CN-P-1A ("A" CN pump) fail to auto start (C – BOP/SRO)

When the Evaluating Team is ready, the malfunction is implemented. This malfunction will cause the trip of the "B" CN pump on overcurrent with the failure of the "A" CN pump to auto start. The BOP is expected to identify the failure based on alarms and indications received. The SRO is expected to direct the BOP to perform the Immediate Actions of 1-AP-21.00, Loss of Main Feedwater Flow. The BOP will start the "A" CN pump and verify Feed flow returns to normal. The Team will direct an operator to check the status of "B" and "A" CN pump, and dispatch an operator to check the condition of the "B" CN pump breaker

Verifiable Actions:

1) BOP – Start the "A" CN pump.

Critical Task:

(CT-2): Start an additional Condensate pump and verify feed flow returns to normal before a Steam Generator Level Reactor Trip. (Scenario specific CT)

Technical Specifications/ Technical Requirements Manual/ Reg. Guide 1.97/ VPAP-2802, Reportability/ Equipment Important to Emergency Response (EP-AA-303).

None.

Event 5:* Dropped Rod, P-8, Control Bank D*. (C – RO/SRO, TS-SRO)

When the Evaluating Team is ready, the malfunction is implemented. The malfunction will cause Control Rod P-8, Control Bank D, to drop into the core. The RO is expected to diagnose the dropped rod based on alarms and indications received. The Team will implement AP-1.00 in response to the dropped rod.

Verifiable Action(s):

1) RO – Place Rod Control in Manual to stop outward rod motion.

Technical Specifications:

1) Tech Spec 3.12.C.3; Startup and POWER OPERATION may continue with one control rod assembly inoperable provided that within one hour either:
   a. The control rod assembly is restored to OPERABLE status, as defined in Specification 3.12.C.1 and 2, or
   b. shutdown margin requirement of Specification 3.12.A.3.c is satisfied. POWER OPERATION may then continue provided that: either reactor power shall be reduced to <75% of Rated Power within one hour, and the High Neutron Flux trip setpoint shall be reduced to ≤ 85% within the next four hours; or the remaining rods in the group shall be aligned within 12 steps of the inoperable control rod within one hour.
2) Tech Spec 3.12.F.2 (potential DNB); When any of the parameters in Specification 3.12.F.1 (Tave, Press, Flow) has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce Thermal Power to less than 5% of rated power within the next 6 hours.

Technical Requirements Manual/ Reg. Guide 1.97/ Equipment Important to Emergency Response (EP-AA-303)/ VPAP-2802, Reportability.

None

***Event #6:* Ramp Unit to < 75%.** (R – RO/SRO, N - BOP)

The SRO will notify the Shift Manager the dropped rod and Tech Spec requirements. The SRO will conduct a Team brief to discuss the Reactivity Plan determined by the RO. The SRO will implement 0-AP-23.00, Rapid Load Reduction, to commence power reduction to be 70 – 74% power within one hour.

Verifiable Actions:

1) RO – manipulate control rods and CVCS blender to maintain RCS Tave and ∆flux in band.
2) BOP – Manipulate Turbine Controls to commence the down power ramp at 1% per minute.

Technical Requirements Manual/ Reg. Guide 1.97/ Equipment Important to Emergency Response (EP-AA-303)/ VPAP-2802, Reportability.

None.

Event #7: **RCS Leakage of approximately 200 gpm.** (M – ALL)

When the Evaluating Team is satisfied with the Teams ability to control the ramp, and the Evaluating Team is ready, the RCS leakage malfunction will be implemented. This malfunction causes RCS leakage of ~200 gpm. The RO will diagnose the RCS leakage based on indications and alarms received. The RO will perform the Immediate Actions of 1-AP-16.00, quantify RCS leakage in excess of the capability of a single CH pump, Trip the reactor, perform the Immediate Actions of E-0, and manually Safety Inject on Step 4 of E-0.

Verifiable Actions:

1) RO – Isolate letdown and place CH flow in manual to quantify RCS leakage.
2) RO - Trip the Reactor and manually initiate Safety Injection.
3) BOP – During performance of E-0 Attachments, the BOP will Identify the following: 1-CH-MOV-1381, Seal Return MOV, will fail to close on the SI and require manual closure. The A: HHSI pump will fail to auto start requiring a manual start of the pump. The “C” Charging pump will trip on overcurrent. The ‘A’ and ‘B’ LHSI pumps will fail to auto start.

Event #8: **Large Break LOCA/FR-Z.1** (M – ALL)

On Transition to 1-E-1, the RCS leakage will rise to a LBLOCA. The Teams response will be complicated by the failure of Hi-Hi CLS to automatically or manually actuate when CTMT pressure reaches 23 psia; requiring manual action by the RO/BOP to align Hi-Hi CLS components. With no CTMT cooling in service, the Team will transition to FR-Z.1 to start the CS pumps (“A” CS pump will trip on overcurrent when started), the “B” CS pump will start manually, manually align SW to the RSHXs, and when RWST level reaches 60% the RS pumps will require manual start. The Team will complete actions required in FR-Z.1, and return to 1-E-1.

Critical Tasks:

CT-3: Restore SI flow to the core prior to transitioning to FR-C.2, Response to Degraded Core Cooling.
Safety Significance: Failures to the HHSI and LHSI pumps will result in NO SI flow to the core until the operator manually starts HHSI and LHSI pumps.

Critical task **begins** when SI is manually initiated and **ends** when SI flow is restored to the core.

CT-4: Manually Actuate Containment Spray and Recirc Spray. Establish CS flow from at least one CS pump, AND at least one RS train before RMT.
Job Aid 17, Critical Task Development, CT-3 based on Pressurized Water Reactor Owners Group Westinghouse Emergency Response Guideline -Based Critical Tasks (PWROG-14043-NP) Appendix B Critical Task CT-3.

Safety Significance: Restoration of Containment Spray and Recirc Spray is necessary in order to protect the Containment and provide long term core cooling.

Critical Task **begins** when CTMT Pressure exceeds 23 psia; and **ends** when at least one Containment Spray pump and one Train of Recirc Spray is established.

The Scenario is terminated on Evaluating Team cue when CTs have been met and the Team has returned to 1-E-1.

Initial Conditions: Initial Conditions: Unit 1 and 2 are at 100% power, MOL. The unit has been at 100% power for > 30 days.

Turnover: The Team will brief the swap of the BC pumps IAW 1-OP-BC-001, Section 5.3, prior to Simulator entry.

Pre-load malfunctions: (Trigger 30's)

- CH 59, DISABLE CH-MOV-381 AUTO CLOSURE
- RS1001, TRN A HI HI CLS FAILS TO ACTIVATE
- RS1002 TRN B HI HI CLS FAILS TO ACTIVATE
- SI14001, DISABLE LHSI PUMP SI-P-1A AUTO START
- SI14002, DISABLE LHSI PUMP SI-P-1B AUTO START
- FP0301, FPS FACP07 ALARM HORN FAILURE
- FP0302, FPS PC SPEAKER FAILURE

Equipment Status/Procedures/Alignments/Data Sheets/etc.:

Unit 1 is at 100% power. All systems and crossties are operable with the following exception:

- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.
- C Charging pump running on Alt, A Charging in Auto, B Charging pump tagged out for breaker PMs.
- Controlling channels have been shifted to CH IV in reparation for Channel III testing.

Turnover:

- The Team will brief the swap of the BC pumps IAW 1-OP-BC-001, Section 5.3, prior to Simulator entry.

The performance of this procedure has been analyzed based on the current plant configurations and the PSA indicates green.

Event	Malf. #'s	Severity	Instructor Notes and Required Feedback
1	N/A	N/A	Swap Running BC pumps IAW 1-OP-BC-001, Section 5.3
2	RC4901	+1	PRZR Level CH 1 Fails High (AP-53.00)
3	FW1804	-1	B SG CH IV FF Fail Low (AP-53.00) **(CT-1)**
4	CN0102 CN1501	TRUE TRUE	1-CN-P-1B Trip overcurrent 1-CN-P-1A Auto Start Fails **(CT-2)**
5	RD1244	TRUE	Dropped RCCA, P-8, Control Bank D
6	N/A	N/A	Ramp Unit to < 75%, due to dropped rod
7.	RC04	16.0%	RCS Leak ~ 200 GPM (AP-16.00, E-0 with SI) **(CT-3)**

8	RC0103 SI14001,SI 14002, CH4601, CH0504	5.0% / 30% TRUE TRUE TRUE	LBLOCA, E-1, FR-Z.1 Failure of SI and HI-HI-CLS
9	CH59 CH6401 RS1001 RS1002 CS0801	TRUE	1-CH-MOV-1381 Auto Close Failure (BOP) 1-CH-P-1A Auto Start Failure (BOP) HI HI CLS FAIL to ACTUATE Train A **(CT-4)** HI HI CLS FAIL to ACTUATE Train B 1-CS-P-1A Trip on Overcurrent

SHIFT TURNOVER INFORMATION

OPERATING PLAN:

The initial conditions have Unit 1 is at 100% power with RCS boron concentration of 760 ppm.

Unit conditions have been stable at approximately 100% power since the last refueling outage.

All systems and crossties are operable with the following exception:

- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.
- C Charging pump running on Alt, A Charging in Auto, B Charging pump tagged out for breaker PMs.
- Controlling channels have been shifted to CH IV in reparation for Channel III test

Unit #2 is at 100% power with all systems and crossties operable.

Shift orders are to maintain 100% power on Unit #1 and upon relieving the watch, perform a swap of the BC pumps IAW 1-OP-BC-001, Section 5.3. PSA analyzed for current plant conditions.

The last shift performed two 30 gallon dilutions, followed by manual makeups.

PWR Scenario 2: Large Break LOCA with failure of HI HI CLS to actuate.

Scenario Objectives: List Each event as a single objective.

A. Given a Pressurizer Level channel failure, respond in accordance with 0-AP-53.00, Loss of Vital Instrumentation/Controls and restore Pressurizer Level control to Normal.
B. Given a SG "B" Feed flow transmitter failure, respond in accordance with 0-AP-53.00, Loss of Vital Instrumentation/Controls to place "B" FRV in manual, control "B" SG level in manual, and return "B" FRV to automatic.
C. Given a Condensate Pump trip, respond in accordance with 1-AP-21.00 and restore SG level to normal.
D. Given a Dropped rod, respond in accordance with 0-AP-1.00, Rod Control System Malfunctions.
E. Demonstrate the ability to ramp the unit in accordance with 0-AP-23.00, Rapid Load Reduction.
F. Given a Large Break LOCA demonstrate the ability to respond to the event and perform mitigative actions in accordance with E-0, E-1, and FR-Z.1.
G. Given multiple failures of auto plant functions, demonstrate the ability to identify and correct any failed components by utilizing the appropriate attachment.

Scenario Sequence

Event One: Swap Running BC Pumps IAW 1-OP-BC-001, Section 5.3

On Lead Evaluator cue, the crew will swap BC pumps by starting the "A" BC pump and securing the B BC pump.

Malfunctions required: None

Objectives: BOP-Start the "A" BC pump and secure the "B" BC pump.

Success Path: "B" BC pump running and the "A" BC pump in standby

Event Two: PRZR Level CH 1 Fail High (Selected Upper Channel)

On Lead Evaluator cue the Pressurizer level Ch 1 will fail high. This will cause charging flow to lower which causes actual Pressurizer level to lower. The team will perform AP-53.00, stabilize Pressurizer level and switch to an operable Pressurizer level.

Malfunctions required: One, RC4901.

Objectives: (RO) Identify the failure and place CH Flow in manual and restore Pressurizer level to normal band.

(RO) Selects Operable Channel combination of PRZR level input to level control system.

(SRO) Review Tech Specs; TS Table 3.7-1, Item 9 Pressurizer Hi Level, Operator Action 7. With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied: 1) The inoperable channel is placed in the tripped condition within 72 hours. 2) The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1. If the conditions are not satisfied in the time permitted, reduce power to less than the P-7 setpoint within the next 6 hours

Success Path: Crew restores Pressurizer level to normal band, shifts to new controlling Pressurizer level channel, and shifts charging flow control back to automatic.

Event Three: *B SG* Feed Flow Channel IV Fail Low

When actions associated with previous failure are complete and Lead Evaluator Cue, the malfunction is initiated. This failure will cause the "B" Feed flow transmitter to fail low. This will be sensed by SGWLC causing the "B" FRV to open. The BOP operator will diagnose this failure, take manual control of the "B" FRV and restore SG level to normal in accordance with 0-AP-53.00. The RO will shift the controlling FF input to the SGWLC system and when "B" SG level has been returned to normal the "B" FRV will be shifted back to automatic. The SRO will consult Tech Specs and identify TS Table 3.7-A, Item 17, Operator Action 6, is applicable; which requires placing channel in trip in 72 hours.

Malfunctions required: One. FW1804.

Objectives: (*BOP)* Identify failure based on alarms and indications received, and perform the Immediate Actions of 0-AP-53.00 to place the "B" FRV in manual and control "B" SG level near program.

(RO) Swap Feed Flow input to the "B" SGWLC system.

(BOP) Return the "B" FRV to automatic when SF/FF matched and SG NR level at program.

(SRO) Review Tech Specs and Identify one (1) 72 hour clocks in effect to place the affected channel in trip, and notify the Team and Shift Manager of the Tech Spec Requirements.

Success Path: Perform Immediate Actions of 0-AP-53.00, assume manual control of "B" SG NR level, swap controlling channel of Feed Flow input to SGWLC system, and return "B" SG FRV to automatic.

Event Four: Loss of one running Condensate pump with failure of Auto start of standby pump

Malfunctions required: Two: CN0102, and CN1501.

Objectives: *(BOP)* Identify trip of running and failure of standby CN pump to auto start based on alarms and indications received. Perform Immediate Actions on 1-AP-21.00, Loss of Main Feedwater Flow, to start the standby CN pump and restore feed flow to the SG.

(SRO) Monitor performance of 1-AP-21.00 Immediate Actions, conduct a Team brief, direct actions to perform local investigation of "B" CN pump and power supply breaker; and notify Shift Manager of failure, procedure entered, and Unit status.

Success Path: Perform Immediate Actions steps of 1-AP-21.00, start the standby CN pump to restore feedflow to normal.

Event Five: Dropped Rod, P-8, Control Bank D

When the Evaluation team is ready, the malfunction is implemented. The malfunction will cause Control Rod P-8, Control Bank D, to drop into the core. This in turn will cause power and temperature perturbations which will cause Rods to step out automatically.The RO is expected to diagnose the dropped rod based on alarms and indications received. The Team will implement AP-1.00 in response to the dropped rod.

Malfunctions required: One, RD1244.

Objectives: (RO) Identify the dropped rod from indications and perform immediate actions of 0-AP-1.00, Rod Control malfunctions. Stop auto rod withdrawal by placing Rod Control switch to MANUAL.

(SRO) Review Tech Specs and identify that per Tech Spec 3.13.C.3; Startup and POWER OPERATION may continue with one control rod assembly inoperable provided that within one hour either:

a. The control rod assembly is restored to OPERABLE status, as defined in Specification 3.12.C.1 and 2, or
b. Shutdown margin requirement of Specification 3.12.A.3.c is satisfied. POWER OPERATION may then continue provided that: either reactor power shall be reduced to <75% of Rated Power within one hour, and the High Neutron Flux trip setpoint shall be reduced to ≤ 85% within the next four hours; or the remaining rods in the group shall be aligned within 12 steps of the inoperable control rod within one hour.

(SRO) Determines that a ramp to < 75% power is required.

Success Path: Stabilizes the plant per 0-AP-1.00 and ensures that all control parameters, temperature and Delta flux are within their required bands.

Event Six: Ramp Unit to < 75%

The SRO will notify the Shift Manager the dropped rod and Tech Spec requirements. The SRO will conduct a Team brief to discuss the Reactivity Plan determined by the RO. The SRO will implement 0-AP-23.00, Rapid Load Reduction, to commence power reduction to be 70 – 74% power within one hour.

Malfunctions required: None.

Objectives: (RO) Manipulate the CVCS system to Emergency Borate and establish a continuous boration to control RCS Tave during the Turbine Ramp. Operate control rods to adjust delta flux and assist in RCS Tave control.

(BOP) Operate turbine controls to control the load ramp per AP-23.00.

(SRO) Direct actions to lower power in accordance with AP-23.00. Notify SEM of plant status and request Maintenance and Engineering support

Success Path: Maintain Tave/Tref mismatch within 5 ºF and delta flux near target value.

Event Seven: RCS Leakage of approximately 200 gpm

When the Evaluating Team is satisfied with the Teams ability to control the ramp, and the Evaluating Team is ready, the RCS leakage malfunction will be implemented. This malfunction causes RCS leakage of ~200 gpm. The RO will diagnose the RCS leakage based on indications and alarms received. The RO will perform the Immediate Actions of 1-AP-16.00, quantify RCS leakage in excess of the capability of a single CH pump, Trip the reactor, perform the Immediate Actions of E-0, and manually Safety Inject on Step 4 of E-0. Both LHSI and HHSI pumps will fail. The ‘A’ HHSI pump will fail to auto start, and the ‘C’ HHSI pump will trip on overcurrent. Both LHSI pumps will fail to auto start.

Malfunctions required: One, RC04 (200 gpm leak)

Objectives: (RO) Identify leakage and perform actions of 1-AP-16.00, RCS Leak. Determine that leakage is greater than 50 gpm, initiate a Reactor Trip and SI at step 4 of E-0.

(BOP) Manage alarms and peer check RO while performing immediate actions of E-0.

(SRO) Stabilize the plant per E-0, and direct actions per E-0, and E-1.

Success Path: Letdown is isolated per AP-16.00, Reactor Trip and SI performed and crew identifies leak is > 150 gpm.

Event Eight: Large Break LOCA/FR-Z.1

On Transition to 1-E-1, the RCS leakage will rise to a LBLOCA. The Teams response will be complicated by the failure of Hi-Hi CLS to automatically or manually actuate when CTMT pressure reaches 23 psia; requiring manual action by the RO/BOP to align Hi-Hi CLS components. With no CTMT cooling in service, the Team will transition to FR-Z.1 to start the CS pumps (“A” CS pump will trip on overcurrent when started), the “B” CS pump will start manually, manually align SW to the RSHXs, and when RWST level reaches 60% the RS pumps will require manual start. The Team will complete actions required in FR-Z.1, and return to 1-E-1.

Malfunctions required: Nine:

- SI14001, SI14002: Disable LHSI PUMP ‘A’, AND ‘B’ AUTO START.
- RC0103, RCS COLD LEG RUPTURE.
- CH6401, DISABLE CH-P-1A AUTO START.
- CH0504, OVER-CURRENT TRIP CHP PUMP CH-P-1C
- RS1001, RS1002: TRN A,B HI HI CLS FAILS TO ACTIVATE.
- CS0801, CS-P-1A OVERCURRENT TRIP.
- CH59, DISABLE CH-MOV-1381 AUTO CLOSURE

Objectives: (RO) Identify that there is no SI Flow to the core. Take actions per E-0 Attachments to restore SI flow to the core. Identify that HI HI CLS has not initiated.

(BOP) Take action per E-0 Attachments as directed. Identify that CH-MOV-1381 did not auto close and manually close CH-MOV-1381.

Success Path: Restore SI flow to the core prior to meeting FR-C.2 entry. Manually actuate Containment Spray and Recirc spray before RMT.

Scenario Recapitulation

Total Malfunctions:	6
Abnormal Events:	5 (0-AP-53.00, 0-AP-53.00, 1-AP-21.00, 0-AP-1.00, 1-AP-16.00)
Major Transients:	1 (LBLOCA, Failure of SI and HI HI CLS)
EOPs Entered:	2 (E-0, E-1)
EOP Contingencies:	1 (FR-Z.1)

Event Description: Swap running BC pumps IAW 1-OP-BC-001, Section 5.3.

Cue: When team ready.

Time	Position	Applicant's Action or Behavior
	SRO/BOP	1-OP-BC-001 Team will pre-brief Initial Conditions, Precautions and Limitations, and procedure prior to entering simulator.
	BOP	1-OP-BC-001, step 5.3.1 Record the equipment location of the Bearing Cooling Water pump to be started: *BOP records 1-BC-P-1A as the pump to be started.*
	BOP	1-OP-BC-001, step 5.3.2 Check the oil level in the inboard and outboard pump bearing of the pump to be started is approximately 50% level on the sight glass. *BOP contacts Turbine 1 who confirms that there is adequate oil in the sight glass.*
	BOP	1-OP-BC-001, step 5.3.3 Place the following pump bearing temperature points on the PCS for the pump to be started. (T2700A, T2701A, T2702A, T2703A) *BOP inserts bearing temps for the 'A' BC pump on PCS.*
	BOP	1-OP-BC-001, step 5.3.4 Place the standby pump to be started in PTL. *BOP places 1-BC-P-1A in PTL.*
	BOP	1-OP-BC-001, step 5.3.5 Close the discharge valve for the standby pump to be started. *BOP directs TB1 to close 1-BC-P-70, BC Pump 1A discharge valve.* *TB1 reports to BOP after 2 minutes that 1-BC-70 is closed.*
	BOP	1-OP-BC-001, step 5.36. Throttle open two turns of the discharge valve for the standby pump to be started. *BOP directs TB1 to throttle open two turns 1-BC-70. TB1 reports after 30 seconds that 1-BC-70 is throttled two turns open.*

Event Description: Swap running BC pumps IAW 1-OP-BC-001, Section 5.3.

Cue: When team ready.

	BOP	1-OP-BC-001, step 5.3.7 Direct the Turbine Building operator to monitor the BC pump start. *BOP directs TB1 To monitor the BC pump start. TB1 acknowledges direction.*
	BOP	1-OP-BC-001, step 5.3.8 Start the selected BC pump and monitor the pump and bearing temperatures. *BOP starts 1-BC-P-1A, and monitors bearing temperatures. TB1 reports that after pump start of 1-BC-P-1A is normal.*
	BOP	1-OP-BC-001, step 5.3.9 Slowly open the discharge valve of the BC pump started. *BOP directs TB1 to slowly open 1-BC-70. TB1 opens 1-BC-70 and reports that after 30 seconds.*
	BOP	1-OP-BC-001, step 5.3.10 Monitor BC Heat Exchanger BC outlet temperature and adjust SW flow through the in-service heat exchanger as required. *BOP directs TB1 to monitor BC outlet temperatures. TB1 acknowledges direction and reports after 30 seconds that BC outlet temperatures are normal.*
	BOP	1-OP-BC-001, step 5.3.11 When the BC Pump started in step 5.3.8 is running normally, THEN stop the BC pump to be secured and place in AUTO. *BOP stops 1-BC-P-1B and places it in AUTO.* **END EVENT 1**

Event Description: PRZR Level Ch1 Fails High (Selected Upper Channel)

Cue: By Examiner.

Time	Position	Applicant's Action or Behavior
	TEAM	PRZR Level Channel 1 Fails High. Diagnose the failure based on the following alarms and indications: • Annunciator 1E-F3, RX TRIP CH-1 PRZR HI LVL • Annunciator 1C-C8, PRZR HI LVL HTRS ON • Pressurizer level indication on 1-RC-LI-1459A fails high. • Charging line flow on 1-CH-FI-1122A is trending down. • Pressurizer level indication on 1-RC-LI-1460 is trending down.
	RO SRO	0-AP-53.00 Perform Immediate Actions of AP-53.00: [1] CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION – NORMAL Identifies 1-RC-LI-1460 and 1-RC-LI-1461 – NORMAL [2] PLACE AFFECTED CONTROL(S)/ COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION **Places CH Flow Control valve 1-CH-FCV-1122 in manual and opens the valve**. Reports Immediate Actions of AP-53.00 complete, CH flow in manual and set at xx gpm. Sets Control Band for PRZR level Control, program level ± 5%. **NOTE**: *Pressurizer level will lower until charging flow restored.*
	SRO BOP SRO RO	0-AP-53.00 Reads AP-53.00 Immediate Action Steps: [1] CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION – NORMAL Reports, 1-RC-LI-1460 and 1-RC-LI-1461 – NORMAL [2] PLACE AFFECTED CONTROL(S)/ COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION. Reports 1-CH-FCV-1122 in manual and opened to raise the rate of PRZR fill.
	SRO	0-AP-53.00 The SRO will lead a transient brief. During the brief, the failure of the PRZR Level Channel 1 High will be discussed.

Event Description: PRZR Level Ch1 Fails High (Selected Upper Channel)

Cue: By Examiner.

		The RO/BOP will report Annunciators received related to the event, and Critical Parameters affected. STA will have no input for the brief.
	  SRO RO	0-AP-53.00 3. Check Reactor Power – Less than or equal to 100%. Reports power is less than 100%.
	  SRO	0-AP-53.00 Notes prior to Step 4. • Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud. • When the affected instrument/controller malfunction(s) has been addressed by this procedure, recovery actions should continue at Step 13. 4. DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE: PRZR Level Control, Step 11 ***SRO Goes to Step 11***.
	  SRO RO SRO RO SRO RO	0-AP-53.00 11. CHECK PRZR LEVEL CONTROL CHANNELS – NORM a) Check PRZR LVL Instrumentation- NORMAL Reports No, Channel 3, 1-RC-LI-1459A, Failed high. Goes to Step 11 a) RNO a) Do the following, as necessary: 1) Place either of the following in MANUAL: (1)-CH-FC-(1)122C, CHG FLOW CNTRL OR (1)-CH-LC-(1)459G, PRZR LEVEL CNTRL Reports Yes, 1-CH-FC-1122C in Manual 2) Control PRZR level at program level. Reports Yes, PRZR level at program (states current PRZR level).
		0-AP-53.00 Step 11 a) RNO (Continued)

Op-Test No.: Surry 2016-1 **Scenario No.: 2** **Event No.: 2**

Event Description: PRZR Level Ch1 Fails High (Selected Upper Channel)

Cue: By Examiner.

	SRO BOP SRO BOP SRO SRO	3) Move PRZR LVL - CH SEL switch to defeat the failed channel. Directs BOP to place the selector switch in Position 3 (Channel 3 / 2). **Places selector switch in Position 3.** 4) Check or place recorder (1)-RC-LR-(1)459 on an operable channel. **Selects either Channel 2 or 3 on recorder**. 5) Refer to Tech Spec 3.1.A.5 (if Pressurizer heaters deenergized), Table 3.7-1, Item 9 and Table 3.7-6, Item 13. a) TS 3.1.A.5.b. Pressurizer, With the pressurizer inoperable due to inoperable pressurizer heaters, restore the inoperable heaters within 72 hours or be in at least HOT SHUTDOWN within 6 hours and the Reactor Coolant System temperature and pressure less than 350°F and 450 psig, respectively, within the following 12 hours. This TS is met. b) **TS Table 3.7-1, Item 9 Pressurizer Hi Level, Operator Action 7.** With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied: 1) The inoperable channel is placed in the tripped condition within 72 hours. 2) The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.1. If the conditions are not satisfied in the time permitted, reduce power to less than the P-7 setpoint within the next 6 hours. c) TS Table 3.7-6, item 13, Pressurizer Level. Required Channels 2. This TS is met. 6) Refer to Attachment 3 Hands out Attachment 3 to RO/BOP for review.
	SRO RO	0-AP-53.00 Step 11 b) Check Pressurizer Heaters - ENERGIZED Reports Yes, Pressurizer heaters are energized.
	SRO RO	0-AP-53.00 Step 11 c) Check letdown - IN SERVICE Reports Yes, Letdown is in service.

Event Description: PRZR Level Ch1 Fails High (Selected Upper Channel)

Cue: By Examiner.

	SRO RO SRO	0-AP-53.00 Step 11 d) Check PRZR level control – IN AUTOMATIC Step 11 d) RNO: d) Do the following as required: 1) Check PRZR level restored to program. 2) Unsaturate (1)-CH-LC-(1)459G, PRZR LEVEL CNTRL, as required. 3) Return (1)-CH-FCV-(1)122 to AUTOMATIC by checking or placing the following in AUTOMATIC: (1)-CH-FC-( )122C, CHG FLOW CNTRL (1)-CH-LC-( )459G, PRZR LEVEL CNTRL **Returns PRZR level control to Automatic when level at program.** Goes to Step 13 based on NOTE Prior to Step 4.
	SRO RO	0-AP-53.00 13. CHECK CALORIMETRIC – FUNCTIONAL IAW (1)-OPT-RX-001, ATTACHMENT 4 Reports Yes, Calorimetric is functional.
	SRO STA	0-AP-53.00 14. REVIEW THE FOLLOWING: Tech Spec 3.7 VPAP-2802, NOTIFICATIONS AND REPORTS-None. TRM SECTION 3.3, INSTRUMENTATION (Potential follow-up question) **TRM 3.3.2.A.1, and A.2:** Less than the minimum Primary Instruments or the minimum Alternate Instruments in Table 3.3.2-1 are functional. **RA A.1**: Implement a fire watch in the Cable Vault and Tunnel and the Emergency Switchgear room of the affected unit in accordance with TRM Section 5.2 within 14 days. **RA A.2**: Restore instrument to functional status within 60 days. Reg Guide 1.97 (Potential follow-up question) • 1-RC-LOOP-1459, A-13 Variable, 2 Channels required • 2 Channels required, Refer to TS Table 3.7-6. • RG 1.97 requirements are a SM/STA function EP-AA-303, Equipment Important to Emergency Response (Potential follow-up question) 1-RC-LI-1459A, PRZR Level Chann1l 1, Category B, Verify alternate indication available, 2 Operable channels are available, requirement met. (Requirement is a SM/STA function.

Event Description: PRZR Level Ch1 Fails High (Selected Upper Channel)

Cue: By Examiner.

		Directs STA to review items listed above. STA will report items have been reviewed and results have been discussed with the Shift Manager.
		0-AP-53.00
	SRO	15. CHECK ADDITIONAL INSTRUMENT / CONTROLLER MALFUNCTION – EXISTS
	BOP	Reports No, no additional failures exist.
	SRO	SRO Goes to Step 17.
		0-AP-53.00
	SRO	17. PROVIDE NOTIFICATIONS AS NECESSARY: Shift Supervision OMOC STA (PRA determination) I&C Confers with Shift Manager concerning required notifications. Confers with Shift Manager concerning Unit Status, procedure used and TS requirements.
	SRO	**END EVENT #2**

Event Description: “B” SG Feed Flow Channel IV fails low.

Cue: When initiated by Team

Time	Position	Applicant’s Action or Behavior
	BOP	Diagnoses failure FT-1497 with the following indications/alarms: Alarms: • 1H-E6 STM GEN 1B FW >< STM FLOW • 1F-D8 STM GEN 1B CH 4 FW < STM FLOW • 1H-G6 STM GEN 1B LVL ERROR. Indications: • Step decrease in 1B SG Feed Flow indication CH-4 • increase in 1B SG Feed Flow indication CH-3 • rising level on ‘B’ SG.
	SRO	Enters 0-AP-53.00 LOSS OF VITAL INSTRUMENTATION / CONTROLS
	BOP	Step 1, AP-53.00 [1] CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION - NORMAL BOP identifies Channel III indication for feed flow is NORMAL.
	BOP	Step 2, AP-53.00 [2] PLACE AFFECTED CONTROL(S)/COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION **BOP takes manual control of ‘B’ SG feed reg valve and lowers demand (FF < SF) to restore level to program using Channel IV indication.** **CT-1: Take manual control of feed flow; control feed flow to prevent a Steam Generator Level reactor trip. (Scenario specific CT)**
	SRO	Conduct a Brief using the Briefing Placard and obtains Critical Parameter information from the RO and BOP. The SRO will update the Shift Manager during AP-progression. SRO will provide a band for control of “B” SG level with “B” FRV in MANUAL. *The STA will state “nothing to add”.*

Event Description: "B" SG Feed Flow Channel IV fails low.

Cue: When initiated by Team

	SRO RO	Step 3, AP-53.00 * VERIFY REACTOR POWER – LESS THAN OR EQUAL TO 100% *Checks Reactor Power < 100% using PCS Calorimetric. Due to restoration of FF on 1B SG, power increase may be noted. As required, the SRO may direct the BOP to reduce turbine load to prevent exceeding 100% Calorimetric power.*
	SRO	Step 4, AP-53.00, Notes NOTE: • Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud. • When the affected instrument/controller malfunction has been addressed by this procedure, recovery actions may continue at Step 13.
	SRO RO/BOP SRO	Step 4, AP-53.00 DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE: *Identifies 1B SG Feed Flow affected.* Goes to Step 7.
	SRO SRO BOP SRO BOP SRO BOP	Step 7, AP-53.00 CAUTION: When CALCALC is based on Feedwater, changes in feed flow will affect calorimetric power. Reactor power must be monitored when adjusting feed flow. CHECK STEAM GENERATOR LEVEL CONTROL INSTRUMENTS – NORMAL • Steam Pressure • Steam Flow • **Feed Flow** • Steam Generator Level *Determines CH IV Feed flow instrumentation for 'B' SG is NOT normal.* Step 7. RNO, AP-53.00 IF the selected steam flow, steam pressure, or feed flow input to the SG Water Level Control system has failed, THEN do the following: **a) Place the associated Feed Reg Valve in MANUAL.** *Verifies 'B' SG MFRV controller, 1-FW-FCV-1488, in manual* b) Control SG level at program level (44%, a band will be given). *Verifies 'B' SG NR level is returning to program level.*

Event Description: "B" SG Feed Flow Channel IV fails low.

Cue: When initiated by Team

	SRO RO SRO BOP	c) Select the redundant channel for affected SG(s) **Selects Channel III Feed Flow for 'B' SG using two-position selector switch on Vertical Board 1-2 (applicant may also place the associated Steam Flow channel in Channel III)** d) WHEN SG level returned to normal, THEN place the Feed Reg Valve in AUTOMATIC. **Places 'B' SG FRV controller, 1-FW-FCV-1488, in automatic control.**
	  SRO SRO SRO	Step 7, RNO, AP-53.00 (Continued) Perform follow-up actions: a) Consult with Shift Manager on need to initiate ( )-OP-RP-001, ALIGNING CONTROL SYSTEM FOR PERFORMANCE OF CHANNEL I, II, III, AND IV PROCESS AND PROTECTION TESTING. *If asked the Shift Manager will recommend not performing 1-OP-RP-001 at this time.* b) Refer to the following Tech Spec 3.7 items: • Table 3.7-1, 12 and **<u>17</u>** • Table 3.7-2, 1.c, 1.e, and 3.a • Table 3.7-3, 2.a, and 3.a • Table 3.7-6, 15 and 16 **Determines Table 3.7-1 item 17, is applicable.** **1. Tech Specs Section 3.7, Table 3.7-1, Item 17**, Low steam generator water level with steam/feedwater flow mismatch, Operator Action 6. With the number of Operable channels one less than the total number of channels, Power Operation may proceed provided the channel is **placed in the trip condition in 72 hours**, the channel may be bypassed for 12 hours for surveillance; if these requirements are not met, the Unit must be placed in Hot Shutdown within 6 hours. c) Refer to Attachment 1. d) IF no other instrumentation failure exists, THEN GO TO Step 13.

Event Description: "B" SG Feed Flow Channel IV fails low.

Cue: When initiated by Team

	 SRO RO	Step 13, AP-53.00 13. Check Calorimetric – Functional IAW 1-OPT-RX-001, Attachment 4. *Determines that Calorimetric is functional.* Note: 1-OPT-RX-001, Attachment 4 is included on next page.
	 SRO STA	0-AP-53.00 14. REVIEW THE FOLLOWING • Tech Spec 3.7 *Determines Table 3.7-1 item 17, Operator action 6, is applicable (place channel in trip w/in 72 hours).* VPAP-2802, NOTIFICATIONS - None TRM SECTION 3.3, INSTRUMENTATION – None. Reg. Guide 1.97: (Potential follow up question) • Main Feed flow, 1-FW-LOOP-F-1486, D-20 variable, 1 channel per SG required, requirement is met. • RG 1.97 requirements are a SM/STA function EP-AA-303, EQ. IMPORTANT TO EMERGENCY RESPONSE - None *The STA reports he has completed review and has discussed the results with the Shift Manager.*
	 SRO BOP SRO	0-AP-53.00 15 CHECK ADDITIONAL INSTRUMENT / CONTROLLER MALFUNCTION – EXISTS Reports no additional failure exists GOES TO Step 17
	 SRO	0-AP-53.00 17. PROVIDE NOTIFICATIONS AS NECESSARY: • Shift Supervision • OMOC • STA (PRA determination) • I&C

Event Description: "B" SG Feed Flow Channel IV fails low.

Cue: When initiated by Team

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Attachment 4
CALORIMETRIC PROGRAM OPERABILITY

NOTE: TRM 3.3.5 shall be reviewed for required actions for non-functionality of the UFM Calorimetric, Normalized Feedwater Venturi, or PCS Calorimetric. Power reduction to less than 98.4% may be required.

1. To check the Primary Plant Performance Program (PP) operability perform the following:

_______ a. Open Programs - Operator Display / Engineering Display

_______ b. Open PP Output Summary - (Operator Display - Primary Plant Poke)

_______ c. Check short timed values for selected basis, Steam Flow (1-OPT-RX-002 box) or Feedflow (1-OPT-RX-003 box), are updating and either good or fair quality.

_______ d. IF selected basis NOT updating and either good or fair quality, THEN contact Reactor Engineering (if available), and then select another calorimetric basis IAW Step 6.3.1. Otherwise, enter N/A.

2. To check the Flow Corrections Program (FL) operability perform the following:

_______ a. Open Programs - Operator Display / Engineering Display

_______ b. Open FL Output Summary (Operator Display - Flow Corr Poke)

_______ c. Check FL Program Status is OK. IF NOT OK, THEN perform the following to check status of different bases.

_______ 1. Open FL0101 - Output Summary (FL Summary Poke)

_______ 2. Compare displayed values to the FL0101 Table below and check selected calorimetric values are updating and either good or fair quality.

_______ 3. IF selected basis NOT operable, THEN contact Reactor Engineering (if available), and then select another calorimetric basis IAW Step 6.3.1.

Event Description: "B" SG Feed Flow Channel IV fails low.

Cue: When initiated by Team

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Attachment 4
CALORIMETRIC PROGRAM OPERABILITY

FL0101 Table				
Flow Correction List	**Normalized Feedwater**	**UFM Feedwater**	**Venturi Feedwater**	**Steam Flow**
Charging Line Flow Corr	X	X	X	X
Letdown Line Flow Corr	X	X	X	X
SG A-1 FF CORR		X	X	
SG A-2 FF CORR		X	X	
SG B-1 FF CORR		X	X	
SG B-2 FF CORR		X	X	
SG C-1 FF CORR		X	X	
SG C-2 FF CORR		X	X	
SG A-1 SF CORR				X
SG A-2 SF CORR				X
SG B-1 SF CORR				X
SG B-2 SF CORR				X
SG C-1 SF CORR				X
SG C-2 SF CORR				X
SG A-1 FF CORR NORM	X			
SG A-2 FF CORR NORM	X			
SG B-1 FF CORR NORM	X			
SG B-2 FF CORR NORM	X			
SG C-1 FF CORR NORM	X			
SG C-2 FF CORR NORM	X			

		END EVENT #3

Event Description: "B" Condensate pump trip with failure of "A" Condensate pump to auto start.

Cue: By Examiner.

Time	Position	Applicant's Action or Behavior
	Team	Trip "B" CN Pump, "A" CN pump Fails to auto start Diagnose the failure based on the following alarms and indications: Annunciator 1K-D4, 4KV BKR AUTO TRIP. Feedflow less than Steam flow on all SGs. SG NR level lowering on all three SGs
	BOP	1-AP-21.00 Perform Immediate Actions of 1-AP-21.00. [1] CHECK MAIN FEED PUMP STATUS: a) Check Reactor Power – GREATER THAN 80% Identify Reactor power greater than 80% ***NOTE***: Step 1 a) RNO added for case where power not yet raised >80% at time of event initiation. a) RNO: IF at least one Main Feed Pump running, THEN GO TO Step 2. b) Check Main Feed Pumps – TWO RUNNING Identify two feed pumps running. ***NOTE***: Step 1b) RNO added for case where BOP fails to identify two (2) MFPs running. b) RNO: GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION. *SRO may have to direct BOP to perform AP-21.00 Immediate Actions*.
	BOP	1-AP-21.00 [2] START AN ADDITIONAL CONDENSATE PUMP Identify 1-CN-P-1A, "A" CN Pump, failed to auto start. **Start 1-CN-P-1A**. **(CT-2): Start an additional Condensate pump and verify feed flow returns to normal before a Steam Generator Level Reactor Trip. (Scenario specific CT)**
	BOP	1-AP-21.00 [3] REDUCE TURBINE LOAD TO MATCH STEAM FLOW WITH FEED FLOW Use Valve Position Limiter OR Reduce Turbine load using Turbine Manual Monitor Feed flow/Steam flow mismatch and determine load reduction is not necessary.

Event Description: "B" Condensate pump trip with failure of "A" Condensate pump to auto start.

Cue: By Examiner.

		Report Immediate Actions of AP-21.00 are complete add SG levels are trending to program level.
	  SRO BOP SRO BOP SRO BOP SRO BOP SRO BOP	AP-21.00 Read Immediate Actions steps of AP-21.00. [1] CHECK MAIN FEED PUMP STATUS: a) Check Reactor Power – GREATER THAN 80% Report Reactor power greater than 80% (Provide Actual Reactor Power) ***NOTE***: Step 1 a) RNO added for case where power not yet raised >80% at time of event initiation. a) RNO: IF at least one Main Feed Pump running, THEN GO TO Step 2. b) Check Main Feed Pumps – TWO RUNNING Report two feed pumps running ***NOTE***: Step 1b) RNO added for case where BOP fails to identify two (2) MFPs running. b) RNO: GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION. [2] START AN ADDITIONAL CONDENSATE PUMP Report 1-CN-P-1A failed to auto start, manually started "A" CN Pump. [3] REDUCE TURBINE LOAD TO MATCH STEAM FLOW WITH FEED FLOW Use Valve Position Limiter OR Reduce Turbine load using Turbine Manual Report Feed Flow matched without reducing turbine load.
	  SRO	AP-21.00 Conduct Transient Brief, describe event that occurred, procedure used, procedure used to continue further actions. RO/BOP will provide alarms received during the event and Critical Parameters. STA will provide no input. SRO will finalize the Transient Brief, direct the RO to contact the Unit 1 Turbine Building Operator and the Service Building Operator to perform local checks on the "A" CN pump (post start), "B" CN pump (indications of cause for tripping) and status of "A" CN pump breaker. SRO will then continue with AP-21.00.

Event Description: "B" Condensate pump trip with failure of "A" Condensate pump to auto start.

Cue: By Examiner.

		NOTE: Team may use 1K-D4 ARP to place the "B" CN pump in PTL; common alarm for a number of loads; allows alarm to be received if other loads subsequently trip.
	SRO BOP SRO	AP-21.00 4. CHECK CONDENSATE POLISHING BLDG BYPASS - REQUIRED Main Feed Pump Suction Pressure - LESS THAN 400 PSIG *If* No, goes to step 6. *If* Yes, goes to step 5. Note: 1-CP-MOV-100 may be stroking open at this time. Goes to Step 5 or 6
	SRO RO	AP-21.00 5. OPEN 1-CP-MOV-100 Opens or verifies 1-CP-MOV-100 OPEN.
	SRO RO	AP-21.00 6. ENERGIZE ALL PRZR HEATERS Reports all pressurizer heaters energized.
	SRO BOP	AP-21.00 7. CHECK STEAM DUMP OPERATION - REDUCING TAVE/TREF MISMATCH BASED ON DEMAND SIGNAL Reports Yes, steam dumps operating properly.
	SRO RO SRO RO	AP-21.00 **NOTE Prior to Step 8**: Depending on initial plant conditions, rod insertion or boration may be used to stabilize RCS temperature and maintain ΔFlux in band. Acknowledges NOTE. 8. CHECK CONTROL RODS - INSERTING IF NECESSARY Reports No, not necessary
	SRO	AP-21.00 9. CHECK ANNUNCIATOR 1E-E3, ΔFLUX DEVIATION - NOT LIT

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Event Description: “B” Condensate pump trip with failure of “A” Condensate pump to auto start.

Cue: By Examiner.

	RO	Reports Yes, Not Lit.

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Event Description: “B” Condensate pump trip with failure of “A” Condensate pump to auto start.

Cue: By Examiner.

	SRO BOP	AP-21.00 10. CHECK ALL SG FLOWS - STEAM FLOW IS LESS THAN OR EQUAL TO FEED FLOW Reports Yes, Steam Flow is equal to Feed Flow.
	SRO BOP	AP-21.00 11. CHECK ALL SG LEVELS - AT OR TRENDING TO PROGRAMMED LEVEL Reports Yes, all SGs are ~ 44%.
	SRO RO	AP-21.00 12. CHECK TAVE - MATCHED WITH TREF Reports Yes, (will provide actual Tave/Tref mismatch.)
	SRO BOP	AP-21.00 13. CHECK FEED HEADER TO STEAM HEADER ΔP - AT LEAST 50 PSID Yes, (will provide actual ΔP indicated.)
	SRO BOP	AP-21.00 14. CHECK AMPs ON EACH MOTOR OF THE RUNNING MAIN FEED PUMP(s) – LESS THAN 420 AMPS Reports Yes, (provides actual MFP amps indicated.)
	SRO BOP	AP-21.00 15. CHECK OPERATION OF MAIN FEED PUMP(s) Recirc valve position (**Closed**) Discharge MOV position (**Open**) Pump amps (Normal, may provide actual MFP Amp indication.)
	SRO RO	AP-21.00 16. CHECK REACTOR POWER CHANGE – LESS THAN 15% IN ONE HOUR Reports Yes, (will provide indicated reactor power.)

Event Description: “B” Condensate pump trip with failure of “A” Condensate pump to auto start.

Cue: By Examiner.

	SRO	AP-21.00 17. NOTIFY THE FOLLOWING: OMOC Maintenance Foreman SRO notifies Shift Manage of Plant Status, Completion of AP-21.00, Report results of local investigation of “B” CN pump and breaker, and requests OMOC and Maintenance Foreman be notified of the event.
		END EVENT #4

Event Description: DROPPED ROD, P-8, CONTROL BANK D

Cue: When initiated by Team.

Time	Position	Applicant's Action or Behavior
	Team RO	Diagnoses the dropped rod with the following indications/alarms: Alarms: • 1G-B5, COMPUTER PRINTOUT ROD CONT SYS. • 1G-C4, UPPER ION CHAMBER DEVIATION OR AUTO DEFEAT <50%. • 1G-D4, LOWER ION CHAMBER DEVIATION OR AUTO DEFEAT <50%. • 1E-E4, NIS PWR RNF CH AVG FLUX DEVIATION. • 1C-B8 – PRZR LO PRESS • 1G-H1 - NIS DROPPED ROD FLUX DECREASE $\geq$ 5% PER 2 SEC • 1G-H2 – RPI ROD BOTTOM <20 STEPS • 1G-F8, ROD BANK D WITHDRAWAL. Indications: • Outward Rod Motion. • RCS Temperature and Pressure decrease. • Flux Variations (radial). • CERPI indication that Control Rod P-8 is at 0 steps. In accordance with the immediate actions of 1-AP-1.00 (Rod Control System Malfunction) the **RO will place Control Rods in MANUAL to stop the outward rod motion.** **RCS pressure <2205 psig places the unit in a 2 hour clock iaw TS. 3.12.F.1 (and COLR) – it is expected that the SRO will track this clock (entry/exit times).**
	SRO	0-AP-1.00 Enters 0-AP-1.00 (Rod Control System Malfunction).
	SRO	CAUTION prior to Step 1: • If Tave decreases below 530 °F, 1-E-0, Reactor Trip or Safety Injection, must be implemented.
	RO	[1] CHECK FOR EITHER OF THE FOLLOWING: • Continuous rod withdrawal • Continuous rod insertion RO will note that there was a continuous rod withdrawal.

Event Description: DROPPED ROD, P-8, CONTROL BANK D

Cue: When initiated by Team.

	RO	[2] STOP ROD MOTION a. **Put ROD CONT MODE SEL switch in MANUAL** b. Verify rod motion – STOPPED RO will place control rods in MANUAL and note that Rods stopped moving.
	SRO	0-AP-1.00 The team will hold a transient brief. During the brief the dropped rod will be discussed. The RO and BOP will report out critical parameters, as per placard on Main Control Room Bench Board. *The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.*
	RO	0-AP-1.00 3. CHECK ROD MOTION - DUE TO INSTRUMENTATION FAILURE • First Stage Impulse Pressure • Tave/Tref • Nuclear Instrumentation RO will note that an instrumentation failure did not exist. The team will transition to the RNO for Step 3.
	RO	0-AP-1.00 3. RNO If rod motion due to a dropped rod, THEN GO TO Step 6. RO will note that dropped rod caused the rod motion.

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Event Description: DROPPED ROD, P-8, CONTROL BANK D

Cue: When initiated by Team.

	RO	0-AP-1.00 6. CHECK IF ANY ROD DROPPED: • **Annunciator ( )G-H2, RPI ROD BOTTOM ≤ 20 STEPS - LIT** OR • **Annunciator ( )G-H1, NIS DROPPED ROD FLUX DECREASE ≥ 5% PER 2 SEC - LIT** OR • **Rod Bottom Lights - ANY LIT** OR • **Any Rod On Bottom light - LIT** OR • Indication of a partially dropped rod in the core The RO will note that they have multiple indications of a dropped rod in the core.
	RO	0-AP-1.00 7. CHECK REACTOR STATUS PRIOR TO FAILURE – CRITICAL
	RO	0-AP-1.00 8. CHECK ONLY ONE ROD AFFECTED
	RO	0-AP-1.00 9. CHECK REACTOR POWER – GREATER THAN 25%
	BOP/RO	0-AP-1.00 10. CHECK UNIT CONDITIONS – STABLE
	RO	0-AP-1.00 11. **PLACE ROD CONTROL IN MANUAL**
	RO/BOP	0-AP-1.00 12. CHECK POWER RANGE NIs – ANY DROPPED ROD SIGNAL PRESENT
	BOP	0-AP-1.00 13. **RESET NIS DROPPED ROD SIGNAL IAW ATTACHMENT 4** Attachment 4 of AP-1.00 is located towards the end of this event.

Event Description: DROPPED ROD, P-8, CONTROL BANK D

Cue: When initiated by Team.

	RO	0-AP-1.00 14. CHECK ANNUNCIATOR 1G-F8, ROD BANK D WITHDRAWAL – NOT LIT If alarm is LIT, then the RNO will direct the team to initiate Attachment 5 – This attachment follows attachment 4 at the end of this event. This may not be lit if the RO quickly placed rods in MANUAL when withdrawing.
	SRO	0-AP-1.00 Note prior to Step 15: • Quadrant power tilt may cause unit ramp requirements to be more restrictive.
	SRO	0-AP-1.00 15. VERIFY QUADRANT POWER TILT RATIO • Initiate Attachment 6, QUADRANT POWER TILT CALCULATION • Review Technical Specification 3.12.B.6 SRO will direct the STA to perform the QPTR in accordance with Attachment 6 and review TS 3.12. *Summary of applicable Tech Specs are found following Step 18.*
	SRO	0-AP-1.00 Notes prior to Step 16: • Checking that the unit is operating between Hot Zero Power and Hot Full Power, that there is only one fully dropped (less than 10 steps) control rod, and that all other rods are above the rod insertion limit, checks that adequate SDM exists. • SDM must be re-verified using 1-OP-RX-001 every 12 hours after the control rod was dropped until the Reactor is shutdown or the rod is declared operable.
	SRO	0-AP-1.00 16. VERIFY SHUTDOWN MARGIN (WITHIN ONE HOUR) • The fully dropped rod (less than 10 steps) is the only inoperable rod AND • All other rods are greater than the Rod Insertion Limit
	SRO	0-AP-1.00 17. MAKE NARRATIVE LOG ENTRY THAT SDM IS MET

Event Description: DROPPED ROD, P-8, CONTROL BANK D

Cue: When initiated by Team.

	SRO	Caution prior to Step 18: • Loop ΔT is the most accurate measure of Reactor power with a mis-aligned rod, and must be monitored during the ramp and used as a basis for stabilizing power.
	RO	18. CHECK REACTOR POWER – LESS THAN OR EQUAL TO 75% Step 18 RNO actions: Do the following: a. Reduce Reactor Power to between 70% - 74% within one hour. b. Reduce NIS High Flux trip setpoints to less than or equal to 85% within the following four hours. c. WHEN Reactor Power has been reduced, THEN GO TO Step 19. The team will enter AP-23.00 to reduce reactor power. **GO TO EVENT 6**
	SRO	Applicable Tech Specs for Dropped Rod: 3.12.B.6: (Quadrant Power Tilt) If, except for operation at THERMAL POWER < 50% or for physics and control rod assembly surveillance testing, the QUADRANT POWER TILT exceeds 2%, then: a. Within 2 hours, either the hot channel factors shall be determined and the power level adjusted to meet the requirement of Specification 3.12.B.1, or b. The power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours. c. If the QUADRANT POWER TILT exceeds 10%, the power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT within the next 30 minutes. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

Event Description: DROPPED ROD, P-8, CONTROL BANK D

Cue: When initiated by Team.

		Applicable Tech Specs for Dropped Rod: (Continued) 3.12.C.3: (Inoperable rod) Startup and POWER OPERATION may continue with one control rod assembly inoperable provided that within one hour either: a. The control rod assembly is restored to OPERABLE status, as defined in Specification 3.12.C.1 and 2, or b. the shutdown margin requirement of Specification 3.12.A.3.c is satisfied. c. POWER OPERATION may then continue provided that: 1) either: a) power shall be reduced to less than 75% of RATED POWER within one (1) hour, and the High Neutron Flux trip setpoint shall be reduced to less than or equal to 85% of RATED POWER within the next four (4) hours, or b) the remainder of the control rod assemblies in the group with the inoperable control rod assembly are aligned to within 12 steps of the inoperable rod within one (1) hour while maintaining the control rod assembly sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT; the THERMAL POWER level shall be restricted pursuant to Specification 3.12.A during subsequent operation. 2) the shutdown margin requirement of Specification 3.12.A.3.c is determined to be met within one hour and at least once per 12 hours thereafter. 3) the hot channel factors are shown to be with the design limits of TS 3.12.B.1 within 72 hours. 4) a reevaluation of each accident analysis of Table 3.12-1 is performed within 5 days.

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Event Description: DROPPED ROD, P-8, CONTROL BANK D

Cue: When initiated by Team.

	BOP	ATTACHMENT 4 of AP-1.00 – RESETTING NIS DROPPED ROD SIGNALS Perform the following for each NI with a dropped rod status light lit. 1. N41 RESET a) Place the Power Range Test Switch in RESET. b) Check the Dropped Rod Status Light on N41 - NOT LIT. c) Return the Power Range Test Switch to NORMAL. 2. N42 RESET a) Place the Power Range Test Switch in RESET. b) Check the Dropped Rod Status Light on N42 - NOT LIT. c) Return the Power Range Test Switch to NORMAL. 3 N43 RESET a) Place the Power Range Test Switch in RESET. b) Check the Dropped Rod Status Light on N43 - NOT LIT. c) Return the Power Range Test Switch to NORMAL. 4. N44 RESET a) Place the Power Range Test Switch in RESET. b) Check the Dropped Rod Status Light on N44- NOT LIT. c) Return the Power Range Test Switch to NORMAL. 5. Check annunciator ( )G-H1, NIS DROPPED ROD FLUX DECREASE >5% PER SEC - NOT LIT 6. Check annunciator ( )G-F8, ROD BANK D WITHDRAWAL - NOT LIT 7. IF ( )G-F8, ROD BANK D WITHDRAWAL is LIT, THEN initiate Attachment 5.

Event Description: DROPPED ROD, P-8, CONTROL BANK D

Cue: When initiated by Team.

	BOP	ATTACHMENT 5 of AP-1.00 – RESPONSE TO ROD BANK D WITHDRAWAL (Rev. 27)

ACTION / EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.___ CHECK ALARM - ROD DEMAND POSITION GREATER THAN OR EQUAL TO 228 STEPS ON CONTROL BANK D - GP 1 DEMAND POSTN	
2.___ CHECK ROD CONTROL MODE SEL SWITCH IN MANUAL	
3.___ CHECK BOTH CONTROL BANK D STEP COUNTERS - LESS THAN OR EQUAL TO 230 STEPS	Do the following: ☐ a) Step rods out to make both Bank D Step Counters equal. (Group step counters must be equalized by stepping out only, to prevent Rod Group Sequence Error) ☐ b) Reset Bank D Step Counters to 230 steps. (Refer to Attachment 3)

NOTE: Rods may have to be stepped in as many as two steps below the fully withdrawn position to clear Annunciator ( )G-F8.

4.___ STEP CONTROL BANK D RODS IN AS NECESSARY TO CLEAR ( )G-F8, ROD BANK D WITHDRAWAL	
5.___ POSITION CONTROL BANK D RODS AS DETERMINED BY SHIFT SUPERVISION	

NOTE: Both group step counters for Control Bank D should be the same before performing Step 6.

6.___ LOCALLY CHECK THE BANK OVERLAP COUNTER - AT 384 PLUS CURRENT CONTROL BANK D DEMAND POSITION	☐ Notify I&C to reset the Bank Overlap Counter.
7.___ NOTIFY UNIT SRO THAT ROD OVERSTEP HAS BEEN RESET	

- END -

Op-Test No.: Surry 2016-1 Scenario No.: 2 Event No.: 6

Event Description: 0-AP-23.00, RAMP UNIT TO < 75% power.

Cue: When initiated by Team.

Time	Position	Applicant's Action or Behavior
	SRO RO SRO	**Start of 0-AP-23.00** Conducts a Brief summarizing the Event and Establish priorities. RO will provide Critical Parameters using Brief Placard. BOP will provide Critical Parameters using Brief Placard. *The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.* Reactivity control during AP-23.00 Ramp: 154 gallons of Boric Acid needed to reduce power to 75%. Rod drop resulted in 4 degree drop in Tave, therefore boration can subtract 38 gal (9.5 gal/°F) for total of 116 gal. boration. If 30 gallons added during Emergency Boration (estimate), that this leaving 86 gallons to be added using normal boration at an average rate of 5.8 gpm. Control Bank 'D' rod height at end of ramp 143 Steps. Completes Brief and continues with AP-23.00.
	SRO	Caution Prior to Step 1: • Conservative decision-making must be maintained during rapid load reductions. Refer to Attachment 1 for trip criteria. Notes Prior to Step 1: • Actions that can be completed independently of preceding steps may be performed out of sequence as directed by the SRO • When the Turbine is not being actively ramped, the REFERENCE and SETTER values must remain matched to prevent inadvertent ramp. • Pre-planned reactivity plans located in the Main Control Room will be used as guidance for ramping down to the desired power level. • The ramp rate in IMP OUT is nonlinear and therefore pre-planned reactivity plans based on IMP IN are not as accurate. However, total amounts of boration and dilution can be used as guidance. • For ramp rates greater than or equal to 1%/minute, Rod Control should remain in Automatic if available.
	RO	0-AP-23.00 1. **TURN ON ALL PRZR HEATERS**

Event Description: 0-AP-23.00, RAMP UNIT TO < 75% power.

Cue: When initiated by Team.

	BOP	2. INITIATE PLANT LOAD REDUCTION AT 2%/MINUTE OR LESS: a) Verify turbine valve position - NOT ON LIMITER The turbine is NOT on the limiter.
	RO	b) Insert control rods in AUTO or MANUAL as necessary to maintain Tave and Tref within 5°F.
	SRO/BOP	c) **Verify or place turbine in IMP IN or IMP OUT as determined by Shift Supervision** The SRO can choose IMP IN or IMP OUT.
	BOP	d) **Adjust SETTER to desired power level** e) **Adjust LOAD RATE %/MIN thumbwheel to desired ramp rate (1%/minute)** f) **Initiate Turbine load reduction using OPERATOR AUTO (pushes the GO button)** g) **Reduce Turbine Valve Position Limiter as load decreases** The BOP will periodically reduce the limiter setpoint during the ramp.
	SRO	3. CHECK EMERGENCY BORATION – REQUIRED The team may decide to emergency borate after the ramp has progressed to the point that Tave and Tref are matched (or close).
	SRO	Note Prior to Step 4: • Step 4 or Step 5 may be performed repeatedly to maintain Tave and Tref matched, ΔFlux in band, and control rod position above the LO-LO insertion limit.

Event Description: 0-AP-23.00, RAMP UNIT TO < 75% power.

Cue: When initiated by Team.

	RO	0-AP-23.00 4. PERFORM AN EMERGENCY BORATION IAW THE FOLLOWING: a) Verify or raise CHG flow to greater than 75 gpm b) **Transfer the in-service BATP to FAST** c) **Open 1-CH-MOV-1350** SRO will direct nominal opening of approximately 25 seconds. d) Monitor EMRG BORATE FLOW • ( )-CH-FI-( )110 e) After required emergency boration, perform the following: 1) **Close ( )-CH-MOV-( )350** 2) **Transfer the in-service BATP to AUTO** 3) Restore Charging flow control to normal SRO may direct rod motion to maintain Δ Flux within specified band.
	RO	5. ESTABLISH A NORMAL BORATION TO MAINTAIN CONTROL ROD POSITION ABOVE THE LO-LO INSERTION LIMITS IAW ATTACHMENT 4 Attachment 4 (Boration) and 5 (Manual Makeups) are at the end of this section. SRO may direct manual rod motion to maintain Δ flux within specified band.
	SRO	Notes Prior to Step 6: • If at any time plant conditions no longer require rapid load reduction, actions should continue at Step 35. • RCS Tave must be maintained less than or equal to 577°F and RCS pressure must be maintained greater than or equal to 2205 psig. Tech Spec 3.12.F.1 should be reviewed if either parameter is exceeded. • I & C should be contacted to provide assistance with adjusting IRPIs.
	RO	6. CONTROL RAMP RATE TO MAINTAIN RCS PRESSURE GREATER THAN 2205 PSIG

Event Description: 0-AP-23.00, RAMP UNIT TO < 75% power.

Cue: When initiated by Team.

	RO	0-AP-23.00 *7. CHECK LETDOWN ORIFICES – TWO IN SERVICE *Evaluator note: two orifices will already be in service.*
	BOP	8. MONITOR STEAM DUMPS FOR PROPER OPERATION
	SRO	0-AP-23.00 9. NOTIFY THE FOLLOWING: • Energy Supply (MOC) • Polishing Building • Chemistry • OMOC
	SRO	10. EVALUATE THE FOLLOWING: • EPIP applicability *The Shift Manager will review EPIPs for applicability. They are not applicable.* • VPAP-2802, NOTIFICATIONS AND REPORTS, applicability *SRO directs STA to review VPAP-2802. The STA reports that he has completed his review of VPAP-2802 and no notifications are required.* *No further actions are required for this event.*
	SRO	11. CHECK RAMP WILL BE TO LESS THAN APPROXIMATELY 35% REACTOR POWER No, go to step 12.
	SRO	12. CHECK REACTOR POWER – HAS LOWERED MORE THAN 15% IN ONE HOUR. When reactor power has lowered >15%, then chemistry will be notified.
	SRO	13. NOTIFY CHEMISTRY OF POWER CHANGE > 15% IN ONE HOUR. Chemistry notified of power change > 15% in one hour.

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Event Description: 0-AP-23.00, RAMP UNIT TO < 75% power.

Cue: When initiated by Team.

	SRO	0-AP-23.00 CAUTION: Secondary plant evolutions affecting Feedwater Flow or temperature will affect RCS temperature and Reactor Power. This effect will be greater at beginning of core life due to a lower value for isothermal temperature coefficient. The operating team must be prepared to mitigate the effects of the secondary evolutions on the RCS. *RO acknowledges the Caution.*
	SRO	0-AP-23.00 14. AT APPROXIMATELY 70% REACTOR POWER CHECK AUXILIARY STEAM MAINTAINING BETWEEN 160 AND 180 PSIG. *RO Acknowledges the step.*
		END EVENT #6

Event Description: 0-AP-23.00, RAMP UNIT TO < 75% power.

Cue: When initiated by Team.

		0-AP-23.00 Attachment 4 (NORMAL BORATION) Actions
	RO	1. **Place the MAKE-UP MODE CNTRL switch in the STOP position.**
	RO	2. **Adjust 1-CH-YIC-1113 to desired total gallons**
	RO	3. **Adjust 1-CH-FC-1113A to desired flow rate.**
	RO	4. **Place the MAKE-UP MOD SEL switch in the BORATE position**.
	RO	5. **Place the MAKE MODE CNTRL switch in the START position.**
	RO	6. Verify proper valve positions.
	RO	7. Adjust boration rate using 1-CH-FC-1113A, as necessary.
	RO	8. WHEN boration is complete, THEN perform the following. IF boric acid is to remain in the Blender to support ramping the Unit, THEN enter N/A. a) Manually blend approximately 20 gallons to flush the boration path IAW Attachment 5, Manual Makeups. b) Enter N/A for the remaining steps in this Attachment. *Attachment 5 is on the next page*
	RO	9. Verify controllers for Primary Grade water and Boric Acid are set correctly.
	RO	10. **Place the MAKE-UP MODE SEL switch in the AUTO position.**
	RO	11. **Place the MAKE-UP MODE CNTRL switch in the START position.**
	RO	12. Notify Shift Supervision of blender status.

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Event Description: 0-AP-23.00, RAMP UNIT TO < 75% power.

Cue: When initiated by Team.

		0-AP-23.00 Attachment 5 (Manual Makeups) Actions
		1. Place the MAKE-UP MODE CNTRL switch in the STOP position.
		2. Check controllers for the flow rate of Boric Acid and Primary Grade water are set correctly.
		3. Check integrators for the gallons of Boric Acid and Primary Grade water are set correctly.
		4. Place the MAKE-UP MODE SEL switch in the MANUAL position.
		5. Place the MAKE-UP MODE CNTRL switch in the START position.
		6. Open 1-CH-FCV-1113B, BLENDER TO CHG PUMP.
		7. Check proper valve positions.
		8. WHEN the Manual Makeup operation is complete, THEN place 1-CH-FCV-1 113B in the AUTO position
		9. Place the MAKE-UP MODE CNTRL switch in the STOP position.
		10. Check or place the control switches in the AUTO position.
		11. Check controllers for Primary Grade water and Boric Acid are set correctly.
		12. Place the MAKE-UP MODE SEL switch in the AUTO position.
		13. Place the MAKE-UP MODE CNTRL switch in the START position.
		14. Notify Shift Supervision of blender status.

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

Time	Position	Applicant's Action or Behavior
	Team	Diagnoses the failure with the following indications: Alarms: • RM-Q7 – CTMT PART ALERT/FAILURE • RM-Q8 – CTMT GAS ALERT/FAILURE • RM-R8 – CTMT GAS RM-RI-160 HIGH **Indications:** • CTMT Sump Level increasing • 1-DA-P-4A running continuously *Unit 2 will perform RM ARPs, but will be unable to perform Auto Action verification.*
	SRO	1-AP-16.00 Direct initiation of 1-AP-16.00, EXCESSIVE RCS LEAKAGE
	SRO	Notes Prior to Step 1: • If SI Accumulators are isolated, 1-AP-16.01, Shutdown LOCA, should be used for guidance. • RCS average temperature has a direct impact on pressurizer level.
	RO	[1] MAINTAIN PRZR LEVEL: • **Isolate Letdown** • **Control Charging flow** **RO closes 1-CH-LCV-1460A and 1-CH-LCV-1460B to isolate letdown. The RO will take manual control of 1-CH-FCV-1122 to stabilize pressurizer level.**
	RO	1-AP-16.00 2. CHECK THE FOLLOWING PARAMETERS – STABLE OR INCREASING • PRZR Level • PRZR Pressure • RCS Subcooling Identifies that not all parameters are stable under the control of the operator. Team should identify RCS leak rate greater than 150 gpm. **Team goes to 1-E-0.**

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

	SRO SRO	GO TO 1-E-0, REACTOR TRIP OR SAFETY INJECTION **Directs RO to trip the reactor, perform E-0, and Manually Safety Inject on Step 4.**
	  RO	**<u>1-E-0 – Reactor Trip or Safety Injection</u>** [1] CHECK REACTOR TRIP: a) Manually trip reactor **Pushes the reactor trip push buttons.** b) Check the following: • All Rods On Bottom light – LIT • Reactor trip and bypass breakers – OPEN • Neutron flux - DECREASING
	RO RO	[2] CHECK TURBINE TRIP: a) **Manually trip the turbine** **Pushes the turbine trip push buttons.** b) Verify all turbine stop valves - CLOSED c) Isolate reheaters by closing MSR steam supply SOV • 1-MS-SOV-104 d) Verify generator output breakers – OPEN (Time Delayed)

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

	RO	[3] CHECK BOTH AC EMERGENCY BUSES – ENERGIZED RO confirms that both AC emergency buses are energized.
	RO RO	[4] CHECK IF SI INITIATED: a) Check if SI is actuated: • LHSI pumps – RUNNING • SI annunciators – LIT • A-F-3 SI INITIATED – TRAIN A • A-F-4 SI INITIATED – TRAIN B 4b) **Manually initiate SI** **The RO will manually initiate SI at step 4 by pushing both SI pushbuttons.** After the immediate actions of 1-E-0 are reported as complete, the SRO will check off immediate action steps in his copy of 1-E-0. After the immediate actions are verified, the team will conduct a brief. RO should identify that there is NO SI flow to the core. *The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal, with the exception of the alarms already received. He will also state that containment conditions and the electrical conditions are as you see them (or as reported by the RO/BOP).*

Op-Test No.: Surry 2016-1 **Scenario No.: 2** **Event No.: 7**

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

	SRO	The team will hold a transition brief. During the brief it will be identified that SI was initiated and the RCS leak has degraded into a LOCA.
	SRO/BOP	5. Initiate Attachment 1 (***Attachment 1, 2, and 8 actions contained at the end of this section***). **CT-3: Restore SI flow to the core prior to transitioning to FR-C.2, Response to Degraded Core Cooling. Critical task begins when SI is manually initiated and ends when SI flow is restored to the core.**
	SRO/RO	SRO may direct the RO to perform Attachment 8 of 1-E-0 for Transient AFW Control, or opt to throttle AFW as part of E-0, Step 6.
	RO	*6. CHECK RCS AVERAGE TEMPERATURE • STABLE AT 547°F OR • TRENDING TO 547°F The team will identify that RCS temperature is decreasing. The team should attribute this to the injection of SI into the RCS and AFW flow to the SGs. It is acceptable for the team to enter the RNO portion of this step and perform the applicable steps (summarized below): Stop dumping steam **Reduce AFW flow to the SGs** Close MSTVs if cooldown continues
	RO	7. CHECK PRZR PORVs AND SPRAY VALVES: a) PRZR PORVs – CLOSED b) PRZR spray controls • Demand at Zero (or) • Controlling Pressure c) PORV block valves - AT LEAST ONE OPEN

Op-Test No.: Surry 2016-1 Scenario No.: 2 Event No.: 7 Page 55 of 93

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

	SRO RO	**NOTE:** Seal injection flow should be maintained to all RCPs. *8. CHECK RCP TRIP AND MINIFLOW RECIRC CRITERIA: a) Charging Pumps – AT LEAST ONE RUNNING AND FLOWING TO RCS Two or three Charging pumps will be running and flowing to the RCS. b) RCS subcooling - LESS THAN 30°F [85°F] RCS subcooling may or may not be less than 30 °F at this time. RNO for the step is to go to step 9. **NOTE**: If RCP trip Criteria are met prior to reaching this Step, the Continuous Actions Page (CAP) contains these criteria. After this Step is read, this step or the CAP can be used for criteria as long as the Team is performing E-0.
	RO	9. CHECK IF SGs ARE NOT FAULTED: • Check pressures in all SGs a) STABLE OR INCREASING AND b) GREATER THAN 100 PSIG RO will observe a slightly decreasing trend on SG pressures. This will be attributed to the RCS cooldown. The team will not transition to 1-E-2.

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

	SRO Unit 2 Unit 2 RO	10. CHECK IF SG TUBES ARE NOT RUPTURED: • Condenser air ejector radiation – NORMAL • SG blowdown radiation – NORMAL • SG MS radiation – NORMAL • TD AFW pump exhaust radiation – NORMAL • SG NR Level - NOT INCREASING IN AN UNCONTROLLED MANNER RO will use PCS indication for MS and TDAFW Exhaust radiation. Unit 2 will be used to report Condenser A/E and SG Blowdown radiation trend.
	SRO RO SRO	11 CHECK RCS - INTACT INSIDE CTMT • CTMT radiation - NORMAL • CTMT pressure - NORMAL • CTMT RS sump level – NORMAL Identify CTMT conditions as abnormal GO TO 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.
	SRO	The team will hold a transition brief. During the brief the SRO will set priorities. The RO and BOP will state the parameters associated with the brief placard. *The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal with the exception of the previously identified alarms. He will also state that containment conditions and the electrical conditions are as you see them.* The SRO will complete the Brief and continue with E-1.
		END OF EVENT 7

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

Time	Position	Applicant's Action or Behavior
		Attachment 1 of E-0 1 CHECK FW ISOLATION: • Feed pump discharge MOVs - CLOSED • 1-FW-MOV-150A • 1-FW-MOV-150B • MFW pumps - TRIPPED • Feed REG valves - CLOSED • SG FW bypass flow valves - DEMAND AT ZERO • SG blowdown TVs – CLOSED
		Attachment 1 of E-0 2 CHECK CTMT ISOLATION PHASE I: • Phase I TVs - CLOSED • 1-CH-MOV-1381 – CLOSED *Identifies that 1-CH-MOV-1381 did not close and manually closes MOV.* • 1-SV-TV-102A - CLOSED • PAM isolation valves - CLOSED • 1-DA-TV-103A • 1-DA-TV-103B
		Attachment 1 of E-0 3 CHECK AFW PUMPS RUNNING: a) MD AFW pumps - RUNNING (Time Delayed) b) TD AFW pump - RUNNING IF NECESSARY

Op-Test No.: Surry 2016-1 **Scenario No.: 2** **Event No.: 7**

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

	BOP	**Attachment 1 of E-0** 4. CHECK SI PUMPS RUNNING: • CHG pumps – NOT RUNNING • LHSI pumps – NOT RUNNING Manually starts LHSI and HHSI pumps. **CT-3: Restore SI flow to the core prior to transitioning to FR-C.2, Response to Degraded Core Cooling.**
	BOP	**Attachment 1 of E-0** 5. CHECK CHG PUMP AUXILIARIES: • CHG pump CC pump – RUNNING • CHG pump SW pump - RUNNING
	BOP	**Attachment 1 of E-0** 6. CHECK INTAKE CANAL: • Level - GREATER THAN 24 FT • Level - BEING MAINTAINED BY CIRC WATER PUMPS
	BOP	**Attachment 1 of E-0** 7. CHECK IF MAIN STEAMLINES SHOULD BE ISOLATED: a) Check if ANY of the following annunciators - HAVE BEEN LIT • E-F-10 (High Steam Flow SI) • B-C-4 (Hi Hi CLS Train A) • B-C-5 (Hi Hi CLS Train B) Identifies annunciators not lit and goes to step 8.
	BOP	**Attachment 1 of E-0** *8. CHECK IF CS REQUIRED: a) CTMT pressure – HAS EXCEEDED 23 PSIA Identifies pressure has not exceeded 23 or 17.7 psia and goes to step 10. Note: Continuous action step, when press > 23 psia, should re-perform, if in E-0.

Op-Test No.: Surry 2016-1 Scenario No.: 2 Event No.: 7 Page 59 of 93

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

	BOP	**Attachment 1 of E-0** *10. BLOCK LOW PRZR PRESS SI SIGNAL: a) Check PRZR pressure – LESS THAN 2000 psig b) Turn both LO PRZR PRESS & STM HDR/LINE ΔP switches to block c) Check Permissive Status light C-2 - LIT BOP may block the low pressurizer pressure SI signal depending on current RCS pressure.
	BOP	**Attachment 1 of E-0** *11. BLOCK LOW TAVE SI SIGNAL: Step may not be performed at this time (if Tave is greater than 543°F). a) Check RCS Tave - LESS THAN 543°F b) Turn both HI STM FLOW & LO TAVG OR LP switches to block c) Check Permissive Status light F-1 - LIT
	BOP	**Attachment 1 of E-0** **NOTE:** • CHG pumps should be run in the following order of priority: C, B, A. • Subsequent SI signals may be reset by re-performing Step 12. 12. CHECK SI FLOW: a) HHSI to cold legs - FLOW INDICATED • 1-SI-FI-1961 (NQ) • 1-SI-FI-1962 (NQ) • 1-SI-FI-1963 (NQ) • 1-SI-FI-1943 or 1-SI-FI-1943A b) Check CHG pumps - THREE RUNNING *No only one Chg pump running. Go to step 12e.*

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

		Step 12, Attachment 1 of E-0, Continued e) RCS pressure - LESS THAN 185 PSIG RNO: e) IF two LHSI pumps are running, THEN do the following: 1) Verify reset or reset SI. 2) Stop one LHSI pump and put in AUTO. 3) GO TO Step 13.
	BOP	**Attachment 1 of E-0** 13. CHECK TOTAL AFW FLOW - GREATER THAN 350 GPM [450 GPM]
	BOP	**Attachment 1 of E-0** 14. CHECK AFW MOVs - OPEN BOP will identify that all AFW MOVS are not open and will read the RNO portion of this step and manually align valves as necessary.
	BOP	**Attachment 1 of E-0** 15. INITIATE SI VALVE ALIGNMENT IAW ATTACHMENT 2 See attached copy of Attachment 2. (Next Page of this guide) Depending on timing, this attachment may have already been completed.
	BOP	**Attachment 1 of E-0** 16. INITIATE VENTILATION, AC POWER, AND SFP STATUS CHECKS IAW ATTACHMENT 3 *Attachment 3 follows Attachment 2 on next page* *Unit 2 Operator will state that Unit 2 is at 100% power (if asked)* *Unit 2 will also accept responsibility to complete Attachment 3 if it is given to Unit 2 at the point where differential pressure indications are requested.*

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

Time	Position	Applicant's Action or Behavior
	SRO	ATTACHMENT 2 of 1-E-0 **NOTE:** Components previously aligned by SI termination steps, must not be realigned by this Attachment.
	RO/BOP	ATTACHMENT 2 of 1-E-0 1. Check opened or open CHG pump suction from RWST MOVs. • 1-CH-MOV-1115B • 1-CH-MOV-1115D
	RO/BOP	ATTACHMENT 2 of 1-E-0 2. Check closed or close CHG pump suction from VCT MOVs. • 1-CH-MOV-1115C • 1-CH-MOV-1115E
	RO/BOP	ATTACHMENT 2 of 1-E-0 3. Check running or start at least two CHG pumps. (listed in preferred order) • 1-CH-P-1C • 1-CH-P-1B • 1-CH-P-1A
	RO/BOP	ATTACHMENT 2 of 1-E-0 4. Check opened or open HHSI to cold legs MOVs. • 1-SI-MOV-1867C • 1-SI-MOV-1867D
	RO/BOP	ATTACHMENT 2 of 1-E-0 5. Check closed or close CHG line isolation MOVs. • 1-CH-MOV-1289A • 1-CH-MOV-1289B

Event Description: RCS LEAK OF 200 GPM. 1-AP-16.00.

Cue: Pre-Event Failures.

	RO/BOP	ATTACHMENT 2 of 1-E-0 6. Check closed or close Letdown orifice isolation valves. • 1-CH-HCV-1200A • 1-CH-HCV-1200B • 1-CH-HCV-1200C
	RO/BOP	ATTACHMENT 2 of 1-E-0 7. Check opened or open LHSI suction from RWST MOVs. • 1-SI-MOV-1862A • 1-SI-MOV-1862B
	RO/BOP	ATTACHMENT 2 of 1-E-0 8. Check opened or open LHSI to cold legs MOVs. • 1-SI-MOV-1864A • 1-SI-MOV-1864B
	RO/BOP	ATTACHMENT 2 of 1-E-0 9. Check running or start at least one LHSI pump. • 1-SI-P-1A • 1-SI-P-1B
	RO/BOP	ATTACHMENT 2 of 1-E-0 10. Check High Head SI flow to cold legs indicated. • 1-SI-FI-1961 • 1-SI-FI-1962 • 1-SI-FI-1963 • 1-SI-FI-1943 or 1-SI-FI-1943A

Op-Test No.: Surry 2016-1 Scenario No.: 2 Event No.: 8 Page 63 of 93

Event Description: LARGE BREAK LOCA FR-Z.1

Cue: When crew enters 1-E-1 or Evaluator cue.

Time	Position	Applicant's Action or Behavior
	SRO	1-FR-P.1 IF 1-FR-P.1, PTS is entered; SRO determines at step 1 that RCS pressure is < 185 psig, and LHSI flow is > 1000 gpm and exits FR-P.1
	SRO RO RO RO	1-FR-Z.1 Actions **CAUTION**: If 1-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, containment spray systems should be operated as directed by 1-ECA-1.1, instead of Step 1 below. 1 CHECK IF CS REQUIRED: a) Check CTMT pressure - HAS INCREASED TO GREATER THAN 23 PSIA b) Check CS pumps - RUNNING — b) IF RWST level greater than 3%, THEN start CS pumps. IF any CS pump can NOT be started, THEN monitor OSRS pumps for cavitation. **Starts 1-CS-P-1A / 1B** — IF cavitation is indicated, THEN put affected OSRS pump in PTL 1-CS-P-1A will auto trip on overcurrent. **1-CS-P-1B will manually start.** c) Check CS system valves - OPEN — c) Manually align CS valves. • 1-CS-MOV-100A • 1-CS-MOV-100B **• 1-CS-MOV-101A and B** **• 1-CS-MOV-101C and D** **• 1-CS-MOV-102A and B** **Opens Valves Listed Above (BOLDED).** **Critical Task:** **CT-4:** Manually Actuate Containment Spray and Recirc Spray. **Establish CS flow from at least one CS pump**, AND at least one RS train before RMT. d) Stop all RCPs *Stops RCPS and acknowledges Loop Low Flow Alarms.* **CRITICAL TASK STOP TIME:** ________________

Event Description: LARGE BREAK LOCA FR-Z.1

Cue: When crew enters 1-E-1 or Evaluator cue.

	SRO RO RO	1-FR-Z.1 Actions 2 CHECK SW FLOW TO RS HXs - GREATER THAN 4750 GPM — Align valves as necessary. *Checks flow > 4750 GPM following the Opening of the SW Valves Listed below.* a) Check the following valves – OPEN • 1-SW-MOV-103A, B, C, and D • 1-SW-MOV-104A, B, C, and D • 1-SW-MOV-105A, B, C, and D **Open the SW MOVs listed above.**
	SRO	1-FR-Z.1 Actions 3 CHECK RS SYSTEMS: a) Check RWST level -LESS THAN 60% — a) Do the following: 1) Monitor RWST level. 2) WHEN RWST level is less than 60%, THEN perform Steps 3b and 3c.
	SRO RO	1-FR-Z.1 Actions 4 CHECK INTAKE CANAL LEVEL – GREATER THAN 24 FT *Reports Intake Canal Level 29 feet and Stable.*
	SRO BOP	1-FR-Z.1 Actions 5 CHECK CTMT ISOLATION VALVES - CLOSED IAW ATTACHMENT 1 — IF flow path NOT necessary, THEN close valves Aligns Valves as necessary. **NOTE**: BOP and SRO may confer on leaving 1-IA-TV-100 open. **NOTE**: FR-Z.1 Actions located at the end of this section.
	SRO RO/BOP	1-FR-Z.1 Actions 6. ___ CHECK MSTVs - CLOSED — Manually close MSTVs. **Closes MSTVs if not already performed.**

Op-Test No.: Surry 2016-1 Scenario No.: 2 Event No.: 8 Page 65 of 93

Event Description: LARGE BREAK LOCA FR-Z.1

Cue: When crew enters 1-E-1 or Evaluator cue.

		1-FR-Z.1 Actions **CAUTION**: • At least one SG must be maintained available for RCS cooldown. • If all SGs are faulted, at least 60 gpm [100 gpm] feed flow should be maintained to each SG.
	SRO	7. ___ CHECK IF FEED FLOW SHOULD BE ISOLATED TO ANY SG(s): a) Check pressures in all SGs: • ANY SG PRESSURE DECREASING IN AN UNCONTROLLED MANNER OR • ANY SG COMPLETELY DEPRESSURIZED — a) GO TO Step 8
	RO/BOP	*Reports SGs NOT Faulted.*
		1-FR-Z.1 Actions
	SRO	8 CHECK SERVICE WATER AVAILABLE: a) Check Intake Canal level – BEING MAINTAINED BY CIRC WATER PUMPS
	RO/BOP	Reports Intake Canal Level 29 feet and Stable. Being maintained by CW pumps. b) RETURN TO procedure and step in effect
	SRO	**Returns to 1-E-1, Step In Effect.**

Event Description: LARGE BREAK LOCA FR-Z.1

Cue: When crew enters 1-E-1 or Evaluator cue.

NUMBER 1-FR-Z.1	ATTACHMENT TITLE CONTAINMENT ISOLATION VALVES	ATTACHMENT 1
REVISION 20		PAGE 1 of 1

LOCATION: Vertical Board LIGHTS: GREEN

___ 1-BD-TV-100A	___ 1-SS-TV-106A	___ 1-SS-TV-100A	___ 1-VG-TV-109A	___ 1-SI-TV-101A
___ 1-BD-TV-100B	___ 1-SS-TV-106B	___ 1-SS-TV-100B	___ 1-VG-TV-109B	___ 1-SI-TV-101B
___ 1-BD-TV-100C	___ 1-SS-TV-102A	___ 1-SS-TV-101A	___ 1-DG-TV-108A	___ 1-RC-TV-1519A
___ 1-BD-TV-100D	___ 1-SS-TV-102B	___ 1-SS-TV-101B	___ 1-DG-TV-108B	___ 1-SI-TV-100
___ 1-BD-TV-100E	___ 1-SS-TV-104A	___ 1-SS-TV-103A	___ 1-CC-TV-109A	___ 1-CH-TV-1204A
___ 1-BD-TV-100F	___ 1-SS-TV-104B	___ 1-SS-TV-103B	___ 1-CC-TV-109B	___ 1-CH-TV-1204B
___ 1-RM-TV-100C	___ 1-LM-TV-100A	___ 1-LM-TV-100E		___ 1-CV-TV-150A
___ 1-RM-TV-100B	___ 1-LM-TV-100B	___ 1-LM-TV-100F		___ 1-CV-TV-150B
___ 1-RM-TV-100A	___ 1-LM-TV-100C	___ 1-LM-TV-100G	___ 1-DA-TV-100A	___ 1-CV-TV-150C
___ 1-CC-TV-105A	___ 1-LM-TV-100D	___ 1-LM-TV-100H	___ 1-DA-TV-100B	___ 1-CV-TV-150D
___ 1-CC-TV-105B	___ 1-CC-TV-140A	___ 1-CC-TV-110A	___ 1-MS-TV-110	___ 1-MS-TV-109
___ 1-CC-TV-105C	___ 1-CC-TV-140B	___ 1-CC-TV-110B	___ 1-CC-TV-110C	___ 1-IA-TV-100
___ 1-SV-TV-102A	___ 1-SV-TV-102	___ 1-IA-TV-101A		___ 1-IA-TV-101B
				___ 1-CH-MOV-1381

Valves Operated for Hi CLS Failure.

Valves Operated for Hi Hi CLS Failure.

Valve BOP/SRO may confer and leave open.

Event Description: LARGE BREAK LOCA FR-Z.1

Cue: When crew enters 1-E-1 or Evaluator cue.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 4
REVISION 69	CLS COMPONENT VERIFICATION	PAGE 1 of 2

LOCATION: Vertical Board

VALVE POSITION: CLOSED
LIGHTS: GREEN

Valves Operated for Hi CLS Failure.

Valves Operated for Hi Hi CLS Failure.

Valve BOP/SRO may confer and leave open.

__ 1-RM-TV-100C
__ 1-RM-TV-100B
__ 1-RM-TV-100A

__ 1-CC-TV-105A

__ 1-CC-TV-105B	__ 1-CC-TV-140A	__ 1-CC-TV-110A		
__ 1-CC-TV-105C	__ 1-CC-TV-140B	__ 1-CC-TV-110B	__ 1-CC-TV-110C	__ 1-IA-TV-100
	__ 1-SV-TV-102	__ 1-IA-TV-101A		__ 1-IA-TV-101B

LOCATION: Unit 1 Vent Panel

RECIRC FAN STATUS: OFF
LIGHTS: AMBER

__ 1-VS-F-1A __ 1-VS-F-1B

LOCATION: Bench Board

VALVE POSITION: OPEN
LIGHTS: RED

__ 1-SW-MOV-105A	__ 1-SW-MOV-105B	__ 1-SW-MOV-105C	__ 1-SW-MOV-105D
__ 1-SW-MOV-104A	__ 1-SW-MOV-104B	__ 1-SW-MOV-104C	__ 1-SW-MOV-104D
__ 1-SW-MOV-103A	__ 1-SW-MOV-103B	__ 1-SW-MOV-103C	__ 1-SW-MOV-103D

Check SW Outlet flow from RS HXs between 6,000 gpm and 12,500 gpm:

- 1-SW-FI-106A, RS HX A
- 1-SW-FI-106B, RS HX B
- 1-SW-FI-106C, RS HX C
- 1-SW-FI-106D, RS HX D

LOCATION: Radiation Monitoring Panel

PUMPS: RUNNING (Time delayed)

__ 1-SW-P-5A __ 1-SW-P-5B __ 1-SW-P-5C __ 1-SW-P-5D

Event Description: LARGE BREAK LOCA FR-Z.1

Cue: When crew enters 1-E-1 or Evaluator cue.

NUMBER 1-E-0	ATTACHMENT TITLE CLS COMPONENT VERIFICATION	ATTACHMENT 4
REVISION 69		PAGE 2 of 2

LOCATION: Annunciator Panel A ALARMS: CLEAR

__ A-D-6 RS HX 1A RAD MON PP NO FLOW

__ A-E-6 RS HX 1B RAD MON PP NO FLOW

__ A-F-6 RS HX 1C RAD MON PP NO FLOW

__ A-G-6 RS HX 1D RAD MON PP NO FLOW

☐ IF alarm is LIT, THEN stop associated rad monitor pump AND monitor SW activity using RI-SW-120.

LOCATION: Bench Board VALVE POSITION: OPEN LIGHTS: RED

__ 1-CS-MOV-102A __ 1-CS-MOV-102B

__ 1-RS-MOV-156A __ 1-RS-MOV-156B

__ 1-CS-MOV-101B __ 1-CS-MOV-101D

__ 1-CS-MOV-101A __ 1-CS-MOV-101C

__ 1-RS-MOV-155A __ 1-RS-MOV-155B __ 1-CS-MOV-100A __ 1-CS-MOV-100B

----------- IF EVENT - CLS HI HI AND LOSS OF RSS -----------

LOCATION: Bench Board VALVE POSITION: CLOSED LIGHTS: GREEN

__ 1-CW-MOV-100A	__ 1-CW-MOV-100B	__ 1-CW-MOV-100C	__ 1-CW-MOV-100D
__ 1-CW-MOV-106A	__ 1-CW-MOV-106B	__ 1-CW-MOV-106C	__ 1-CW-MOV-106D
__ 1-SW-MOV-101A	__ 1-SW-MOV-101B	__ 1-SW-MOV-102A	__ 1-SW-MOV-102B

Section Not Performed for this Scenario.

Op-Test No.: Surry 2016-1 **Scenario No.: 2** **Event No.: 8**

Event Description: LARGE BREAK LOCA FR-Z.1

Cue: When crew enters 1-E-1 or Evaluator cue.

		End of Event 8. **End of Scenario 2**

FOLDOUT PAGES FOR REFERENCED PROCEDURES

NUMBER 1-E-0	CONTINUOUS ACTIONS PAGE	REVISION 71

1. RCP TRIP CRITERIA
   Trip all RCPs if BOTH conditions listed below occur:
   a. Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS
   b. RCS Subcooling - LESS THAN 30°F [85°F]
2. MINIFLOW RECIRC CRITERIA
   a. CLOSED - When RCS pressure is less than 1275 psig [1475 psig] AND RCP Trip Criteria are met (RCPs OFF).
   b. OPEN - When RCS pressure is greater than 2000 psig.
3. ADVERSE CONTAINMENT CRITERIA
   Use Adverse Containment setpoints if EITHER condition listed below occurs:
   - Containment Pressure - GREATER THAN 20 PSIA
   - Containment Radiation - GREATER THAN 1.0E5 R/HR
4. COLD LEG RECIRCULATION SWITCHOVER CRITERIA
   GO TO 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, if RWST level lowers to less than 20%.

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1. AMSAC RESET CRITERIA
   AMSAC may be manually reset when level in all three SGs is greater than 13% or six minutes have elapsed since the Reactor trip. When AMSAC is reset, AMSAC ARMED annunciator H-D-1 should clear and affected components may be realigned as needed.
2. TD AFW PUMP SHUTDOWN CRITERIA
   The TD AFW pump may be secured when SG NR level is greater than 22% in at least 2 SGs, AMSAC is reset, and no auto-start signal exists. To secure the pump, the pump SOV control switches must be taken to OPEN-RESET and then to CLOSE.
3. MANUAL SI ALIGNMENT
   If SI fails to automatically align, Attachment 2 may be used for guidance on manual SI valve alignment.
4. * TRANSIENT AFW FLOW CONTROL (IF SI in progress)
   Attachment 7 may be used for guidance on transient AFW flow control.
5. * FAULTED SG ISOLATION AND AFW FLOW CONTROL (IF SI in progress)
   Attachment 8 may be used for guidance on faulted SG(s) isolation and AFW flow control.
6. * RUPTURED SG ISOLATION AND AFW FLOW CONTROL (IF SI in progress)
   Attachment 9 may be used for guidance on ruptured SG(s) isolation and AFW flow control.
7. * LOSS OF RCP SUPPORT CONDITIONS
   Trip RCPs if a loss of a support condition occurs. (for example, loss of CC)

* Preemptive Actions

FOLDOUT PAGES FOR REFERENCED PROCEDURES

NUMBER 1-E-0	CONTINUOUS ACTION STEPS	REVISION 71

1. Check RCS Average Temperature - STABLE AT OR TRENDING TO 547°F. (E-0, Step 6)
2. Monitor RCP Trip and Miniflow Recirc Criteria. (E-0, Step 8)
3. Check SG Narrow Range Level - ANY SG GREATER THAN 12%. (Control feed flow to maintain Narrow Range Level between 22% and 50%) (E-0, Step 25)
4. Monitor LHSI pumps and secure as necessary. (E-0, Step 30)

NOTE: Subsequent SI signals may be reset by reperforming Step 12 of Attachment 1.

5. Monitor CTMT pressure and check CLS initiation as necessary. (Attachment 1, Step 8)
6. Monitor RWST level and check RS initiation as necessary. (Attachment 1, Step 9)
7. Block Low PRZR Pressure SI signal when less than 2000 psig. (Attachment 1, Step 10)
8. Block Low Tave SI signal when less than 543°F. (Attachment 1, Step 11)

FOLDOUT PAGES FOR REFERENCED PROCEDURES

CONTINUOUS ACTIONS PAGE FOR 1-E-1

1. RCP TRIP CRITERIA

   Trip all RCPs if BOTH conditions listed below occur:

   a. Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS

   b. RCS Subcooling - LESS THAN 30°F [85°F]

2. SI REINITIATION CRITERIA

   Following SI termination or SI flow reduction, manually start SI pumps and align valves as necessary if EITHER condition listed below occurs:

   - RCS subcooling based on CETCs - LESS THAN 30°F [85°F]
   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 22% [50%]

3. MINIFLOW RECIRC CRITERIA

   a. CLOSED - When RCS pressure is less than 1275 psig [1475 psig] AND RCP Trip Criteria are met (RCPs OFF).

   b. OPEN - When RCS pressure is greater than 2000 psig.

4. ADVERSE CONTAINMENT CRITERIA

   Use Adverse Containment setpoints if EITHER condition listed below occurs:

   - Containment Pressure - GREATER THAN 20 PSIA
   - Containment Radiation - GREATER THAN 1.0E5 R/HR

5. SECONDARY INTEGRITY CRITERIA

   Manually start SI pumps as necessary and GO TO 1-E-2, FAULTED STEAM GENERATOR ISOLATION, if any SG pressure is lowering in an uncontrolled manner or has completely depressurized, and has not been isolated.

6. E-3, TRANSITION CRITERIA

   Manually start SI pumps as necessary and GO TO 1-E-3, STEAM GENERATOR TUBE RUPTURE, if any SG level rises in an uncontrolled manner or any SG has abnormal radiation.

7. COLD LEG RECIRCULATION SWITCHOVER CRITERIA

   GO TO 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, if RWST level lowers to less than 20%.

8. AFW SUPPLY SWITCHOVER CRITERIA (Refer to Attachment 5)

   Transfer to one of the following alternate AFW water supplies if ECST level lowers to less than 20%.

   a. 1-CN-TK-2, using 1-CN-150.

   b. 1-CN-TK-3, using AFW Booster Pumps.

   c. AFW Crosstie.

   d. Firemain.

9. RCP SEAL INJECTION CRITERIA

   Seal Injection flow should be maintained to all RCPs.

10. LOSS OF RCP SUPPORT CONDITIONS

    Trip RCPs if a loss of a support condition occurs. (for example, loss of CC)

SIMULATOR OPERATOR'S GUIDE

Simulator Setup

Initial Conditions:

Recall IC -374 **and implement TRIGGER #30 to activate all passive malfunctions and verify Trigger #30 implemented.**

Open the monitor window and add the following points to it:

- asp_ao_off

Enter the following MALFUNCTIONS:

Malfunction	Delay	Ramp	Trigger	Value	Final	Trigger Type (Auto or Manual)
RS1001, TRN A HI HI CLS FAILS TO ACTIVATE			30		TRUE	
RS1002, TRN B HI HI CLS FAILS TO ACTIVATE			30		TRUE	
SI14001, DISABLE LHSI PUMP SI-P-1A AUTO START			30		TRUE	
SI14002, DISABLE LHSI PUMP SI-P-1B AUTO START			30		TRUE	
FP0301, FPS FACP07 ALARM HORN FAILURE			30		TRUE	
FP0302, FPS PC SPEAKER FAILURE			30		TRUE	
CH59, DOSAB;E CJ-MOV-381 AUTO CLOSURE			30		TRUE	
RC4901, PRZR LEVEL XMTR CH1 FAILURE	10		1		1	MAN
FW1804, B S/G MN FD FLOW XMTR FT-1486 FAILS	10	2:00	3		-1	MAN
CN0102, MAIN CN PUMP CN-P-1B TRIPS:OVER-CURRENT	10		5		TRUE	MAN
CN1501, DISABLE CN-P-1A AUTO START			5		TRUE	MAN
RD1244, DROPPED RCCA P-8 CONTROL BANK D	10		7		TRUE	MAN
RC04, RCS LEAK NONISOLABLE (0-1200 GPM)	10	1:00	9		16	MAN
RC0103, RCS COLD LEG C PIPE RUPTURE	5:00	25:00	9		5	MAN
CH6401, DISABLE CH-P-1A AUTO START			9		TRUE	MAN
CS0801, CS-P-1A BKR 14H5 OVERCURRENT TRIP	30		13		TRUE	AUTO
CH0504, OVER-CURRENT TRIP CHG PUMP CH-P-1C (ALT)			15		TRUE	AUTO

SIMULATOR OPERATOR'S GUIDE

Enter the following Remote Functions

Description	Delay	Ramp	Trigger	Value	Final	Trigger Type (Auto or Manual)
BC70, BEARING COOLING WATER PUMP BC-P-1A OUTLET ISOLATION VALVE			30		10%	

SIMULATOR OPERATOR'S GUIDE

Enter the following EVENT TRIGGERS:

Trigger#	EVENT	Command
13	csp1a_spd .gt. 0.5	
15	SI4: SI Train 'A' actuated	

Trigger Summary:

TRIGGER	TYPE	DESCRIPTION
1	MAN	Initiates RC4901, Pressurizer Level Xmtr 459 fails high.
3	MAN	Initiates FW1804, B S/G Feed Flow Xmtr 1486 fails low.
5	MAN	Initiates CN0102, 1-CN-P-1A over current trip, and CN1501, Disable CN-P-1A auto start.
7	MAN	Initiates RD1244, P-8 dropped rod.
9	MAN	Initiates RC04, RCS Leak, and RC0103, RCS Cold leg C rupture.
13	AUTO	Initiates 1-CS-P-1A over current trip.
15	AUTO	Initiates 1-CH-P-1C over current trip.
30	PRE-SCENARIO	Initiates: RS1001, RS1002, SI14001, SI14002, CH59, FP0301, FP0302, BC_70.

SIMULATOR OPERATOR'S GUIDE

Verify the following control room setup:

- ❑ Place the simulator in RUN and verify normal 100% power operation indications.
- ❑ Reset the ROD CONTROL SYSTEM
- ❑ Verify Controlling channels selected to **IV**
- ❑ Verify PRZR Level Recorder, **1-RC-LR-1459 is selected to CH I**
- ❑ Verify Red Magnets on the following components:

1-CH-P-1B, 'B' Chg pump

- ❑ Verify 1-RM-RI-112 aligned to A/B SG and 1-RM-RI-113 aligned to C SG (magnets).
- ❑ Verify Ovation System operating.
- ❑ Reset ICCMs.
- ❑ Verify Component Switch Flags.
- ❑ Verify Brass Caps properly placed.
- ❑ Verify SG PORVs set for 1035 psig. ('A'-1038, 'B'-1036, 'C'-1034)
- ❑ Verify Rod Control Group Step Counters indicate properly.
- ❑ Verify Ovation CRT display.
- ❑ Advance Charts
- ❑ Verify Turbine Thumb Wheel Settings @120 rpm/min and Position 6
- ❑ Verify Containment Instrument Air Compressors are on Inside Suction (all RMs reset)
- ❑ Verify all ARPs have been cleaned

❑ 1C-B8	❑ 1C-C8	❑ 1E-C8	❑ 1E-F3
❑ 1H-B5	❑ 1H-C4	❑ 1H-D3	❑ 1H-D4
❑ 1H-E4	❑ 1H-E6	❑ 1H-F8	❑ 1H-G6
❑ 1H-H2	❑ 1J-E1	❑ 1J-F1	❑ 1K-D4

- ❑ Verify CLEAN copies of the following procedures are in place:

❑ 1-OP-BC-001	❑ 0-AP-53.00	❑ 1-AP-21.00	❑ 0-AP-1.00

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❑ 0-AP-23.00	❑ 1-AP-16.00	❑ 1-E-0	❑ 1-E-1
❑ 1-FR-Z.1	❑	❑	❑

❑ **Verify Reactivity Placard is current.**

❑ Verify ALL PINK MAGNETS are accounted for.

❑ Reset Blender Integrators for Boric Acid to 100 and PG 1000.

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Brief

This simulator performance scenario is performed in the EVALUATION MODE. You should not direct questions to the evaluators. Otherwise, you should perform as if you were in the MCR.

Your ability to maintain a log is not being graded, but maintaining a rough log is recommended to help during briefs.

If you need to communicate with the Unit 2 operator, verbally state, "Unit 2" and an instructor will locate to the Unit 2 area and respond to you as quickly as possible.

In the unlikely event that the simulator fails such that illogical indications result, the session will be terminated. In other words, respond to what you see. If there is a problem with the simulation, the session will be terminated or adjusted as appropriate based on the specific problem.

Assign operating positions.

Ask for and answer questions.

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Conduct shift turnover:

The initial conditions have Unit 1 at 100% power with RCS boron concentration of 760 ppm.

Unit conditions have been stable at approximately 100% power since the last refueling outage.

All systems and crossties are operable with the following exception:

- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.
- C CH running on Alt, A CH in Auto, B CH pump Tagged out for breaker PMs.
- Controlling channels have been shifted to CH IV in reparation for Channel III testing

Unit #2 is at 100% power with all systems and crossties operable.

The last shift performed one 30 gallon dilution. Previous to this, shifts had been performing two 30 gallon dilutions.

Shift orders are to maintain 100% power on Unit 1, and swap BC pumps IAW 1-OP-BC-001, Section 5.3.

When the team has accepted the shift, proceed to the Session Conduct Section.

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Session Conduct:

- Ensure conditions in Simulator Set-up are established.
- **Ensure Trigger 30 is active prior to team entering the simulator.**
- Verify Exam Security has been established and ASP_AO_OFF = True.

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EVENT 1 Swap running BC pumps IAW 1-OP-BC-001, Section 5.3.

BOOTH: **When cued by the Evaluator place simulator in RUN if it isn't already in RUN.**

Field Operators: (*2 minute delay from request to answer, unless otherwise indicated*)

- If contacted regarding the BC pump swap, report that you have been briefed, and are standing by the BC pump to be started.
- When contacted to check the oil level for 1-BC-P-1A, report that you have already checked it and the oil level is approximately 50% on the sight glas
- When contacted to close 1-BC-70, acknowledge direction, wait 2 minutes then, **Report to operator,** that 1-BC-70 is CLOSED. Booth Note: 1-BC-70 was closed to 10% OPEN by T 30 during Sim setup in preparation for this task.
- When contacted to open two turns on 1-BC-70. **Report to operator after approx. 30 sec.** that 1-BC-70 is open two turns.
- When directed to monitor the pump start **report to the operator** that you are standing by.
- When directed to slowly open 1-BC-70. **Open 1-BC-70 (100%, 60 sec ramp). After 1-BC-70 is full open, Report to the operator** that 1-BC-70 is full open, and 1-BC-P-1A is running normally.
- When directed to monitor the BC Heat Exchanger outlet temperatures. **Report to the operator** that BC Heat exchanger outlet temperatures are normal.

Role play as other individuals as needed.

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EVENT 2 PRZR Level CH 1 Fails High (Selected upper channel)

When cued by examiner, implement Trigger #1.

Operations Supervisor/Management:

- **If contacted**, acknowledge failure of PRZR level channel.
- **If asked,** will notify I&C of the failure.
- **If asked,** will notify the OMOC.
- **When notified,** acknowledge Tech Spec requirements for the channel failure, but do not imply concurrence with SRO Tech Spec determination.

STA:

- **If contacted**, acknowledge the failure, acknowledge the Tech Spec requirements for the failed channel, but do not imply concurrence with the Tech Spec determination by the SRO.
- **When notified,** VPAP-2802, Reg Guide 1.97, TRM Section 3.3, and EP-AA-303 have been reviewed and results discussed with the Shift Manager.
- **If the team has a transient brief:** The STA will have no input.

I&C:

- **If notified**, acknowledge the failure and the need to place the channel in trip.

Maintenance/Work Week Coordinator:

- **If contacted**, will notify I&C of the failed channel, have I&C prepare to place the channel in trip, and prepare to troubleshoot the cause of the failure.

Role play as other individuals as needed.

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EVENT 3 FEED FLOW CHANNEL IV "B" SG FAILS LOW.

When cued by examiner, implement Trigger #3.

BOOTH NOTE: **Critical Task: (CT-1):** If the BOP fails to take timely action in response to the FF channel failure, an automatic reactor trip on SG NR high level will occur; an unanticipated reactor trip should be considered as failure criteria.

Operations Supervisor/Management:

- **If contacted**, acknowledge feed flow channel failure. The individual(s) contacted will also acknowledge any TS LCOs.
- **If contacted,** will take responsibility for writing the CR.
- **If contacted,** will acknowledge entry into AP-53.00.
- **If contacted,** will recommend to the team that channels remain as they are for now (i.e., do not perform 1-OP-RP-001 at this time).

STA:

- **If contacted**, will acknowledge feed flow channel failure. The individual(s) contacted will also acknowledge any TS LCOs. The STA will not confirm or deny any TS decisions.
- **If contacted,** will take responsibility for writing the CR.
- **IF contacted:** regarding Regulatory procedure review. Acknowledge request. If asked for status, report that all reviews have been completed and discussed with the Shift Manager.
- **If the team has a transient brief:** The STA will state "nothing to add".

Field Operators:

- Will perform actions as directed.

Maintenance/ Work Week Coordinator:

- **If contacted**, will acknowledge the feed flow channel failure and contact I&C to commence preparation to place the failed channel in trip.

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Unit 2:

- **If contacted**, will acknowledge the failure of the feed flow channel.

Role-play as other individuals as needed.

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EVENT 4 **1-CN-P-1B trip/Failure of 1-CN-P-1A to auto start-.**

When cued by examiner, implement Trigger #5.

BOOTH NOTE: **Critical Task: (**CT-2): Start an additional Condensate pump and verify feed flow returns to normal before a Steam. Generator Level Reactor Trip.

Operations Supervisor/Management:

- **If contacted**, Acknowledge failure.
- **If contacted:** Take responsibility for submitting CR.
- **If contacted:** Will notify Maintenance and OMOC of the failure.

Maintenance/ Work Week Coordinator:

If contacted, will acknowledge the failure, contact Maintenance to commence investigation.

STA:

- **If contacted**, Acknowledge the failure
- **If the team has a transient brief:** Will have no input for a transient brief.

Field Operators:

When contacted to check status of CN Pumps: Wait three (3) minutes and report 1-CN-P-1A conditions normal after start; 1-CN-P-1B exhibits no obvious cause for the trip.

When contacted to check status of 1-EP-BKR-15B4: Wait 3 minutes and report breaker 15B4 has an instantaneous overcurrent drop on "B" phase.

Role play as other individuals as needed.

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EVENT 5 DROPPED ROD, P-8 CBD.

When cued by examiner, implement Trigger #7.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the dropped rod and the need to ramp at 1%/minute to 70 - 74% power.
- **If asked** for a recommended ramp rate, ask what the Unit Supervisor recommends. When authorized by the NRC, the Shift Manager will direct a 1%/minute ramp rate.
- **If contacted,** will take responsibility for writing the WR and CR.

STA:

- **If asked,** the STA will perform a shutdown margin calculation.
- **If asked,** the STA will perform a QPTR. See AP-1.00, Attachment 6, attached.
- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal with the exception of the previously identified radiation monitor alarms. He will also state that containment conditions and the electrical conditions are as you see them.
- **If asked,** the STA will acknowledge the need to borate and use rods (will acknowledge the team review of pre-planned reactivity plans and OP-RX-010, if performed). If asked to perform the OP-RX-010 review, the STA will state that he is not able to at this time.
- **When contacted,** SRO directs STA to review VPAP-2802. The STA reports that he has completed his review of VPAP-2802 and no notifications are required.

Unit Two:

- **If contacted,** all conditions on U2 are normal.

Field Operators:

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- **If contacted**, Will perform actions as directed (i.e., may be asked to read bank overlap counter- RD 2).

I&C:

- If requested by MCR to reset the bank overlap – use RD2 drawing to set as requested.

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge the failure and commence investigations.

Role play as other individuals as needed.

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EVENTS 6 Ramp to < 75% power.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the ramp to 70% - 74%.
- **If contacted,** will take responsibility for writing the CR.
- **If asked** for a recommended ramp rate, ask what the Unit Supervisor recommends. When authorized by the NRC, the Shift Manager will direct a 1%/minute ramp rate.

Unit 2 Operator:

If notified, acknowledge the failure and impending ramp of Unit 1.

STA:

- **If contacted**, will acknowledge the Reactivity Plan reported by the RO.
- **If contacted,** will take responsibility for writing the WR and CR.
- **If the team has a transient brief:** The STA will, "nothing to add".
- If contacted, STA review of VPAP-2802 complete, reviewed with Shift Manager, no notifications required.

Maintenance/ Work Week Coordinator:

- **If contacted**, will acknowledge the requirements to reduce reactor power.

Chemistry

- **If contacted,** acknowledge the ramp.

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Field Operators:

- **If contacted**, as the Turbine Building Operator to walkdown the Turbine during the ramp, acknowledge the direction.
- **If contacted,** as the polishing building operator, acknowledge the direction to monitor polisher DP.

Role play as other individuals as needed.

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EVENT 7 RCS leak of 200 gpm-

When cued by examiner, implement Trigger #9.

Booth Notes: After Trigger 9 is entered the following will occur:

- RC04 (0-1200 GPM) will ramp in over 60 sec to approx. 200 gpm.
- After 5 min.: RC0103, Cold Leg Rupture, leak rate will ramp to 5% over 25 min ramp.
- **Verify Trigger Trigger 15 goes active after SI Train A is actuated.**

Operations Supervisor/Management:

- **If contacted,** will take responsibility for writing the WR and CR.
- **If contacted,** will acknowledge entry into AP-16.00.
- **If contacted,** will acknowledge Unit shutdown imminent due to excessive RCS leakage.
- **If contacted,** will acknowledge EPIPs require evaluation.

STA:

- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal with the exception of the previously identified radiation monitor alarms. He will also state that containment conditions and the electrical conditions are as you see them.
- **When Notified,** acknowledge that EPIPs require evaluation.

Unit Two:

- **When** radiation alarms sound on the radiation alarm panel, silence the alarms when directed and report the alarm to the Unit 1 SRO.
- **If directed** perform the associated RM ARP without leaving the confines of the Unit 2 control area. If actions or verifications are required on the Unit 1 side, inform the Unit 1 SRO/BOP.
- **If asked**, blowdown and air ejector RM readings are [*as indicated at the time*].
- **When required**, assist Unit by acknowledging Fire Panel Alarms.

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- **If contacted,** Unit Two has implemented AP-50.00, and all conditions on U2 are normal.

Field Operators:

- **If contacted**, field operators will perform valve manipulations as required:
- **If contacted,** regarding CH-P-1C overcurrent trip. Acknowledge and call back after 2 minutes and report that 1-CH-P-1C breaker has an overcurrent trip flag on 'C' phase.

Health Physics:

- **If contacted**, will acknowledge the leak of RCS coolant into containment.

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge the failures to HHSI, and LHSI pumps failing to auto start..

STA:

- **If asked**, will acknowledge the increase in RCS leakage.
- **If contacted**, will enter the control room and and prepare for the transient brief (items are reported "as you see them or previously reported").
- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. Radiological conditions are as indicated. He will also state that containment conditions and the electrical conditions are as you see them.

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge the failure.

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EVENT 8: LARGE BREAK LOCA-

When cued by examiner, OR WHEN crew has entered 1-E-1. **Raise severity level of RC0103 to 30% (no time delay or ramp).** This will cause Containment pressure to rise rapidly to > 30 psia.

Booth Note: **Verify Trigger 13 goes active after 1-CS-P-1A is started and speed is > 50%.**

BOOTH NOTE: **Critical Tasks: CT-3:** Restore SI flow to the core prior to transitioning to FR-C.2, Response to Degraded Core Cooling. **CT-4:** Manually Actuate Containment Spray and Recirc Spray. Establish CS flow from at least one CS pump, AND at least one RS train before RMT.

Operations Supervisor/Management:

- **If contacted,** will acknowledge increase in LOCA severity.
- **If contacted,** will acknowledge failure of HI HI CLS to actuate.
- **If contacted,** will acknowledge EPIPs require evaluation.

STA:

- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal with the exception of the previously identified radiation monitor alarms. He will also state that containment conditions and the electrical conditions are as you see them.
- **When Notified,** acknowledge that EPIPs require evaluation.

Unit Two:

- **When** radiation alarms sound on the radiation alarm panel, silence the alarms when directed and report the alarm to the Unit 1 SRO.
- **If directed** perform the associated RM ARP without leaving the confines of the Unit 2 control area. If actions or verifications are required on the Unit 1 side, inform the Unit 1 SRO/BOP.
- **If asked to assist with EPIP actions,** perform actions as directed.

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Field Operators:

- **If contacted**, field operators will perform valve manipulations as required:
- **If contacted,** regarding CH-P-1C overcurrent trip. Acknowledge and call back after 2 minutes and report that 1-CH-P-1C breaker has an overcurrent trip flag on 'C' phase.

Health Physics:

- **If contacted**, will acknowledge the leak of RCS coolant into containment.

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge the failures to HHSI, and LHSI pumps failing to auto start..

STA:

- **If asked**, will acknowledge the increase in RCS LOCA.
- **When conditions are met for FR-Z.1.** Inform the SRO that conditions are met for FR-Z.1 (orange path).
- **If contacted**, will enter the control room and and prepare for the transient brief (items are reported "as you see them or previously reported").
- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. Radiological conditions are as indicated. He will also state that containment conditions and the electrical conditions are as you see them.

Role play as other individuals as needed.

The scenario will end when actions have been performed for FR-Z.1 and the team has returned to 1-E-1. OR When the Lead NRC Evaluator is satisfied that all actions have been performed.

Scenario #3

Facility: Surry Power Station Scenario No.: 3 Op-Test No.: 2016-003

Examiners: ____________________ Operators: ____________________

Initial Conditions: Unit 1 and 2 at 100% power; MOL, 760 ppm Boron.

- C CH running on Alt, A CH in Auto, B CH pump Tagged out for breaker PMs.
- Controlling channels are aligned to CH IV for channel III testing.
- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days

Turnover: The Team will pre-brief the swap of the running EHC pumps IAW 1-OP-EH-001, Section 5.6.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP/SRO	Swap running EHC pumps IAW 1-OP-EH-001, Section 5.6
2	RM0201, +0.5 CA03 CA06	I – BOP/SRO	Process Rad Monitor RI-RM-160 (CTMT Gas) fails high, Auto Close Failure 1-IA-TV-101A/B (ARP RM – Q8/R8)
3	RC4802, +1	I-RO/SRO TS - SRO	PRZR Press Cont XMTR Failure (445) fails high 0-AP-53.00/1-AP-31.00
4	FW1303, +1	I – BOP/SRO TS - SRO	A S/G Nar Rng Lvl XMTR LT-476 fails high (AP-53.00) **(CT-1)**
5	SW0401 SW1202	C – RO/SRO TS-SRO	Overload Trip of Pump SW-P-10A (AP-12.00) Disable SW-P-10B Auto Start
6	N/A	R – RO/SRO N - BOP	Ramp to Hot Shutdown (AP-23.00)
7	SI1502, 20% SI1601 SI1604 SI34 SI35 CH50-53 CH0901	M - ALL	LOCA outside of CTMT (Isolable) **(CT-2, CT-3)** - SI-MOV-1867C, and 1867D not open - CH-MOV-1115B, and 1115D fail to AUTO open - CH-MOV-11152 and 1115E fail to AUTO close - Boric Acid Transfer pump Thermal overload
	BD01 BD02 VS0101	Team Failures	1-BD-TV-100A (BD "A" not closed on AFW start) 1-BD-TV-100B (BD "B" not closed on AFW Start) 1-VS-F-58A not auto start on SI

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Event 1: **Swap of running EHC pumps**. (N- BOP/SRO)

The swap of the running EHC pumps will be pre-briefed prior to the Team entering the Simulator.

Verifiable Action(s):

BOP – Manipulate EH pump control switches to start MP-2 and secure MP-1.

Technical Specifications/ TRM Actions/ Reg. Guide 1.97/ VPAP-2802, Reportability/ Equipment Important to Emergency Response (EP-AA-303).
None.

Event 2: **CTMT Gas RM failed high 1-RM-RI-160 with auto action failed.** (I – BOP/SRO)
When the Evaluating Team is ready, the malfunction for CTMT Gas RM failure is implemented. This failure will cause the RM to go into High Alarm. The BOP is expected to silence the RM alarm and report to the SRO. The SRO will direct the BOP to perform the Annunciator Response Procedure for the alarm received.

Verifiable Action(s):

1) BOP – Close 1-IA-TV-101A and 1-IA-TV-101B, CTMT IA Compressor inside TVs, and check 1-IA-AOV-103, CTMT IA Compressor Outside Suction, open.

Technical Specifications:
None

Technical Requirements Manual.
None.

Reg. Guide 1.97:
None.

VPAP-2802, Reportability:
None.

Equipment Important to Emergency Response (EP-AA-303).
1-RM-RMS-160, Category B, verify alternate indications available within the group, no actions required. Identification is usually performed by SM/STA. **Recommended to evaluate SRO identification of this item post scenario as a follow-up at the Evaluator's discretion.**

Event 3: **Przr Pressure Control Channel Fails High (1-RC-PI-1445).** (I – RO/SRO, TS – SRO).
When the Evaluation Team is ready, the malfunction is implemented. This failure causes Przr PORV 1-RC-PCV-1456 to open fully resulting in a lowering RCS pressure. The RO is expected to diagnose the failure based on the alarms and indications received, perform the Immediate Actions of 0-AP-53.00, Loss of Vital Instrumentation/Controls, or 1-AP-31.00, Increasing or Decreasing RCS Pressure, and place the control switch for 1-RC-PCV-1456 in close to stop the RCS pressure decrease. The Team will implement 0-AP-53.00/1-AP-31.00 in response to the failure.

Verifiable Actions(s):

1) RO – Place the control switch for 1-RC-PCV-1456 in close.
2) RO/BOP – May place rod control switch in manual to stop rods stepping out due to the negative reactivity inserted by the decrease in RCS pressure.
3) RO – Close 1-RC-MOV-1535, Przr PORV 1-RC-PCV-1456 block valve.

Technical Specifications:

1) Tech Spec 3.12.F.2, DNB Parameters, RCS pressure < 2205 psig, Return RCS to >2205 psig within 2 hours or reduce Thermal Power to less than 5% of Rated Power within the next 6 hours.

2) Tech Spec 3.1.A.6.a, Relief Valves: With one or both power operated relief valves (PORVs) inoperable but capable of being manually cycled, within 1 hour either restore the PORV to Operable status or close the associated block valve and maintain power to the block valve.

TRM Actions / Reg. Guide 1.97/ VPAP-2802, Reportability/ Equipment Important to Emergency Response (EP-AA-303):

None.

Event 4: **A SG NR Level Channel III Fail High.** (I – RO/SRO, TS - SRO)

When the Evaluating Team is ready, the next malfunction is implemented. This failure will cause the channel III NR level on A SG to fail high. The BOP is expected to diagnose the failure based on alarms and indications received, perform the Immediate Actions of 0-AP-53.00, Loss of Vital Instrumentation/Controls, place the A FRV in manual, and control the A SG level at program. A SG FRV will remain in manual control until Major event.

Critical Task CT-1:

Should the BOP fail to take action to control A SG NR level, an automatic reactor trip will occur. This would be considered failure criteria. An Operations representative will be available for consultation during the Scenario.

Verifiable Action(s):

1) BOP – Place the A SG FRV in manual and control level at program.

Technical Specifications:

1) Tech Spec Table 3.7-1, Item 12, Operator Action 6, With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied: The inoperable channel is placed in the tripped condition within 72 hours, The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1, If the conditions are not satisfied in the time permitted, be in at least HOT SHUTDOWN within 6 hours.
2) Tech Spec Table 3.7-2, Item 3.a.1) and 2), Operator Action 20; With the number of OPERABLE channels less than the Total Number of Channels, REACTOR RITICAL and/or POWER OPERATION may proceed provided the following conditions are satisfied: The inoperable channel is placed in the tripped condition within 72 hours, The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1, If the conditions are not satisfied in the time permitted, be in HOT SHUTDOWN within the next 6 hours and reduce RCS temperature & pressure to less than 350°F/450 psig, respectively in the following 12 hours.
3) Tech Spec Table 3.7-3, Item 3a, SG Water Level Hi-Hi, Operator Action 20; With the number of OPERABLE channels less than the Total Number of Channels, REACTOR RITICAL and/or POWER OPERATION may proceed provided the following conditions are satisfied: The inoperable channel is placed in the tripped condition within 72 hours, The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1, If the conditions are not satisfied in the time permitted, be in HOT SHUTDOWN within the next 6 hours and

reduce RCS temperature & pressure to less than 350°F/450 psig, respectively in the following 12 hours.

4) Tech Spec Table 3.7-6, Item 15, SG water level NR; 2 channels required. This requirement is met.

Technical Requirements Manual.

None.

Reg. Guide 1.97:

1) SG NR level, A-01 Variable, 2 Channel per SG required, see TS Table 3.7-6. Identification is usually performed by SM/STA. **Recommended to evaluate SRO identification of this item post scenario as a follow-up at the Evaluator's discretion.**

VPAP-2802, Reportability:

None.

Equipment Important to Emergency Response (EP-AA-303).

Heat Sink Capability, A SG NR Level CH III, Att. 6 Page 6 of 7, Category B variable, Alternate indications available within grouping, no action required. Identification is usually performed by SM/STA. **Recommended to evaluate SRO identification of this item post scenario as a follow-up at the Evaluator's discretion.**

Event #5: **Trip of the running CH pump SW pump, with failure of the redundant pump to auto start.**

(C – RO/SRO, TS - SRO)

When the Evaluation Team is ready, the malfunction is implemented. This failure causes the running CH pump SW pump to trip, and the redundant pump fails to auto start on low pressure. The Team should respond by implementing ARP 1D-G5, SW OR CC PPS DISCH TO CHG PUMPS LO PRESS, start 1-SW-P-10B, and verify CH Pump SW flow restored.

Verifiable Actions:

RO - Start 1-SW-P-10B.

Technical Specifications:

1) Tech Spec 3.01, Place the Unit in HSD in 6 hours, CSD in the following 30 hours, in effect.
2) Tech Spec 3.2.C.1; No Boron injection subsystems OPERABLE, all CH pumps Inoperable due to the Inoperable CH PP SW pumps being Inoperable.
3) Tech Spec 3.3.B.3; No safety injection subsystem Operable, all HHSI pumps are also Inoperable due to the Inoperable CH PP SW pumps.

Technical Requirements Manual.

TRM Table 3.7.9-1, Sheet 16 of 17, 1-SW-P-10A, App R component, App R Fire Watch in Unit 1 ESR, Unit 1 and 2 Turbine Building North Wall, MER 3. Implement RMAs within 72 hours, and implement App R fire watch within 14 days. TRM requirements identified by SM/STA. **Recommended to evaluate SRO identification of this item post scenario as a follow-up at the Evaluator's discretion.**

Reg. Guide 1.97:
None.
VPAP-2802, Reportability:
None.
Equipment Important to Emergency Response (EP-AA-303).
None.

Technical Requirements Manual/ Reg. Guide 1.97/ Equipment Important to Emergency Response (EP-AA-303):
None

Event #6: **Ramp Unit to HSD. (**R – RO/SRO, N – BOP)
The SRO will notify the Shift Manager, and conduct a Team brief to discuss the Reactivity Plan determined by the RO. The SRO will implement 0-AP-23.00, Rapid Load Reduction, to commence power reduction to Hot Shutdown.

VPAP-2802, Reportability:
A 4 hour report is required per VPAP-2802, 6.3.4.a.1, Initiation of plant shutdown (reduction of power or temperature) required by Technical Specifications. Identification is usually performed by SM/STA**. Recommended to evaluate SRO identification of this item post scenario as a follow-up at the Evaluator's discretion.**

Event #7: **LOCA outside CTMT***.* (M – ALL)
When the Evaluation Team is ready, the malfunction is implemented. The malfunction initiates a LOCA outside of CTMT. The RO will diagnose the LOCA based on alarms and indications received, perform the Immediate Actions of 1-AP-16.00, Excessive RCS Leakage, isolate letdown, and place CH in manual to quantify the leakage. When the RO determines that leakage is beyond the capability of a single CH pump, the SRO will direct the RO to trip the reactor and manually safety inject. Safety Injection will actuate, but failures to CH-MOV-1115B, 1115C, 1115D, 1115E, SI-MOV-1867C, and 1867D will result in a failure of HHSI to inject into the core and both charging pumps will start and take a suction from the VCT with no Letdown. Additionally the in-service Boric Acid Transfer pump will trip preventing any makeup to the VCT. This will result in the VCT drawing down at approximately 4%/minute. The team will initiate attachments 1-3 to restore SI flow to the core. The Team will perform E-0, transition to ECA-1.2. LOCA Outside CTMT where the leak is isolated, and then transition to 1-E-1 to assess SI termination Criteria.

Verifiable Actions:

1) RO - Isolate letdown.
2) RO - Place CH Flow in manual to quantify RCS leakrate.
3) RO - Manually trip the reactor and manually actuate Safety Injection.
4) BOP – Using E-0 Attachments, identify and close A SG BD TVs, Start 1-VS-F-58A filtered exhaust fan, and open 1-SI-MOV-1867D.

Critical Task:

1) **CT-2:** Restore HHSI flow from the RWST to the core prior to air-binding of the HHSI pumps.

   Safety Significance: Failure of 1867C and D to open, failure of 1289A and B to close, and failure of the RWST and VCT suction valves to align result in the Charging pumps running with suction from the VCT with flow maintained to RCS. If candidate does not properly align the suction source prior to the VCT emptying the

Charging pumps will be airbound and will start to cavitate. Furthermore because of the size of the LOCA, there will be NO SI flow to the core until the operator manually aligns the HHSI Cold leg discharge MOV s.

..

2) **CT-3:** Isolate LOCA outside of Containment before power is lost to 1-SI-MOV-1890C.

Safety Significance, delay in Team response will cause SFGDS sump to fill to a level where 1-SI-MOV-1890C (isolation source) power is lost, thereby preventing isolation of the LOCA outside CTMT. Approximately 30 minutes elapse from time of automatic reactor trip on low RCS pressure to 1-SI-MOV-1890C power loss. Critical Task **begins** on initiation of LOCA outside CTMT; and **ends** when Team has closed 1-SI-MOV-1890C and has verified LOCA isolated using RCS pressure rise.

The Scenario is terminated based on Evaluator Cue, and the Team has transitioned to 1-E-1.

Initial Conditions: Initial Conditions: Unit 1 and 2 are at 100% power, MOL. The unit has been at 100% power for > 30 days.

Turnover: The Team will brief the swap of the running EH pumps IAW 1-OP-EH-001, Section 5.6, prior to Simulator entry.

Pre-load malfunctions: (Trigger 30's)

- BD01, DISABLE BDTV100A AUTO CLOSURE
- BD02, DISABLE BDTV100B AUTO CLOSURE
- VS0101, PRESSURE SWITCH VS-P-127A STUCK AS IS
- SI34, DISABLE SI-MOV-867C AUTO OPEN
- SI35, DISABLE SI-MOV-867D AUTO OPEN
- CA03, DISABLE IA-TV-101A AUTO CLOSURE
- CA06, DISABLE IA-TV-101B AUTO CLOSURE
- CH50, DISABLE CH-MOV-115B AUTO OPEN
- CH51, DISABLE CH-MOV-115C AUTO CLOSE
- CH52, DISABLE CH-MOV-115D AUTO OPEN
- CH53, DISABLE CH-MOV-115E AUTO CLOSE
- FP0301, FPS FACP07 ALARM HORN FAILURE
- FP0302, FPS PC SPEAKER FAILURE

Equipment Status/Procedures/Alignments/Data Sheets/etc.:

Unit 1 is at 100% power. All systems and crossties are operable with the following exception:

- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.
- C Charging pump running on Alt, A Charging in Auto, B Charging pump tagged out for breaker PMs.
- Controlling channels have been shifted to CH IV in reparation for Channel III testing.

Turnover:

The Team will pre-brief the swap of the running EHC pumps IAW 1-OP-EH-001, Section 5.6.

The performance of this procedure has been analyzed based on the current plant configurations and the PSA indicates green.

Turnover: The performance of this procedure has been analyzed based on the current plant configurations and the PSA indicates green.

Event	Malf. #'s	Severity	Instructor Notes and Required Feedback
1	N/A	N/A	Swap running EHC pumps IAW 1-OP-EH-001, Section 5.6.
2	RM0201	1.0	Process Rad Monitor RI-RM-160 (CTMT Gas) fails high, Auto Close Failure 1-IA-TV-101A/B (ARP RM – Q8/R8)
3	RC4802	+1	PRZR Press Cont XMTR Failure (445) fails high. (AP-53.00, AP-31.00)
4	FW1303	+1	A S/G Nar Rng Lvl XMTR LT-476 fails high (AP-53.00). **(CT-1)**
5	SW0401 SW1202	TRUE	Overload Trip of Pump SW-P-10A (AP-12.00) Disable SW-P-10B Auto Start
6	N/A	N/A	Ramp to Hot Shutdown (AP-23.00)
7.	SI1502 SI1601 SI1604 SI34,35 CH50-53 CH0901	20% TRUE TRUE TRUE TRUE TRUE	LOCA outside CTMT (ISOLABLE). (E-0, ECA-1.2) **(CT-2, CT-3).** SI Cold leg check valves fail SI-MOV-1867C, and 1867D fail to auto open. CH-MOV-115B, and D fail to open. CH-MOV-115C, and E fail to close BORON XFER PP Thrml overload
8	BD01 BD02 VS0101	TRUE TRUE TRUE	Team failures: BD-TV-100A, and BD-TV-100B fail to auto close. 1-VS-F-58A fail to auto start.

SHIFT TURNOVER INFORMATION

OPERATING PLAN:

The initial conditions have Unit 1 is at 100% power with RCS boron concentration of 760 ppm.

Unit conditions have been stable at approximately 100% power since the last refueling outage.

All systems and crossties are operable with the following exception:

- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.
- C Charging pump running on Alt, A Charging in Auto, B Charging pump tagged out for breaker PMs.
- Controlling channels have been shifted to CH IV in reparation for Channel III test

Unit #2 is at 100% power with all systems and crossties operable.

Shift orders are to maintain 100% power on Unit #1 and upon relieving the watch, perform a swap of the eh pumps IAW 1-OP-EH-001, Section 5.6. PSA analyzed for current plant conditions.

The last shift performed two 30 gallon dilutions, followed by manual makeups.

PWR Scenario 3: LOCA Outside Containment with failure of HHSI.

Scenario Objectives: List Each event as a single objective

A. Given a failure of CTMG Gas RM, 1-RM-RI-160, respond in accordance with ARP RM-Q8 or RM-R8 and verify auto actions have occurred.
B. Given a failure of Pressurizer press transmitter 1-RC-RT-1445, respond in accordance with 0-AP-53.00, Loss of Vital Instrumentation/Controls, and restore Pressurizer pressure to normal.
C. Given a SG "A" Level transmitter failure, respond in accordance with 0-AP-53.00, Loss of Vital Instrumentation/Controls to place "A" FRV in manual, control "A" SG level in manual, and return "A" FRV to automatic.
D. Given a Condensate Service Water Pump trip, respond in accordance with 1-AP-12.00 and restore Condensate SW to normal.
E. Demonstrate the ability to ramp the unit in accordance with 0-AP-23.00, Rapid Load Reduction.
F. Given a Large Break LOCA outside Containment, demonstrate the ability to respond to the event and perform mitigative actions in accordance with E-0, and ECA-1.2.
G. Given multiple failures of auto plant functions, demonstrate the ability to identify and correct any failed components by utilizing the appropriate attachment.

Scenario Sequence

Event One: Swap of running EHC pumps

The team will swap running EH pumps by starting MP-2 and securing MP-1.

Malfunctions required: None

Objectives: (BOP) Manipulate EH pump control switches to start MP-2 and secure MP-1.

Success Path: MP-2 running, and MP-1 secured.

Event Two: CTMT Gas RM failed high 1-RM-RI-160 with auto action failed.

When the Evaluating Team is ready, the malfunction for CTMT Gas RM failure is implemented. This failure will cause the RM to go into High Alarm. The BOP is expected to silence the RM alarm and report to the SRO. The SRO will direct the BOP to perform the Annunciator Response Procedure for the alarm received

Malfunctions required: Three; RM0201 (1-RM-RI-160) fails high, CA03 (1-IA-TV-101A fails to auto close), CA06, (1-IA-TV-101B fails to auto close).

Objectives: (BOP) Identify the failure and take actions per the ARP to close 1-IA-TV-101A, B.

(SRO) Direct actions to perform the ARP.

Success Path: Close 1-IA-TV-101A, and B.

Event Three: Przr Pressure Control Channel Fails High (1-RC-PI-1445).

When the Evaluation Team is ready, the malfunction is implemented. This failure causes Przr PORV 1-RC-PCV-1456 to open fully resulting in a lowering RCS pressure. The RO is expected to diagnose the failure based on the alarms and indications received, perform the Immediate Actions of 0-AP-53.00, Loss of Vital Instrumentation/Controls, or 1-AP-31.00, Increasing or Decreasing RCS Pressure, and place the control switch for 1-RC-PCV-1456 in close to stop the RCS pressure decrease. The Team will implement 0-AP-53.00/1-AP-31.00 in response to the failure

Malfunctions required: One, RC4802.

Objectives: (RO) Identify the failure of PT-1145, and perform immediate actions of AP-53.00, or AP-31.00 to close PORV 1-RC-PCV-1456, and restore Pressurizer pressure.

(SRO) Direct actions per AP-53.00, AP-31.00 to close PORV 1-RC-PCV-1456 and restore Pressurizer pressure. Identify Tech Specs 3.1.A.6.a not met and direct required actions.

Success Path: Pressurizer PORV 1-RC-PCV-1456 is closed, Pressurizer pressure is restored to normal band, and Pressurizer block valve 1-RC-MOV-1535 is closed with power available.

Event Four: A SG NR Level Channel III Fail High.

When the Evaluating Team is ready, the next malfunction is implemented. This failure will cause the channel III NR level on A SG to fail high. The BOP is expected to diagnose the failure based on alarms and indications received, perform the Immediate Actions of 0-AP-53.00, Loss of Vital Instrumentation/Controls, place the A FRV in manual, and control the A SG level at program. A SG FRV will remain in manual control until Major event

Malfunctions required: One FW1804

Objectives: (BOP) Place the A SG FRV in manual and control level at program.

(SRO) Direct actions per 0-AP-53.00. Identify Tech Specs that are not met:

- TS Table 3.7-1, Item 12, Operator Action 6.
- TS Table 3.7-2, Item 3.a.1) and 2), Operator Action 20.
- TS Table 3.7-3, Item 3a, Operator Action 20.

Success Path: ‘A’ FRV in MANUAL, and SG ‘A’ level restored to normal band.

Event Five: Trip of the running CH pump SW pump, with failure of the redundant pump to auto start.

When the Evaluation Team is ready, the malfunction is implemented. This failure causes the running CH pump SW pump to trip, and the redundant pump fails to auto start on low pressure. The Team should respond by implementing ARP 1D-G5, SW OR CC PPS DISCH TO CHG PUMPS LO PRESS, start 1-SW-P-10B, and verify CH Pump SW flow restored

Malfunctions required: Two; SW0401 (Trip of SW-P-10A), SW1202 (SW-P-10B fails to auto start)

Objectives: (RO) Identify trip of SW-P-10A, and failure of SW-P-10B to auto start. Start SW-P-10B per ARP, OR 0-AP-12.00, OR as directed per Surry Standards (failure of auto action).

(SRO) Direct actions per ARP or 0-AP-12.00. Identify TS 3.3.B.3, and 3.2.C.1 not met. This will require placing the unit in HSD within 6 hours per TS 3.01.

Success Path: Start 1-SW-P-10B to restore cooling flow to Charging pumps.

Event Six: Ramp Unit to HSD.

The SRO will notify the Shift Manager, and conduct a Team brief to discuss the Reactivity Plan determined by the RO. The SRO will implement 0-AP-23.00, Rapid Load Reduction, to commence power reduction to Hot Shutdown

Malfunctions required: None.

Objectives: (RO) Manipulate the CVCS system to Emergency Borate and establish a continuous boration to control RCS Tave during the Turbine Ramp. Operate control rods to adjust delta flux and assist in RCS Tave control.

(BOP) Operate turbine controls to control the load ramp per AP-23.00.

(SRO) Direct actions to lower power in accordance with AP-23.00. Notify SEM of plant status and request Maintenance and Engineering support Success Path: Maintain Tave/Tref mismatch within 5 ^{o}F and delta flux near target value.

Success Path: Maintain Tave/Tref mismatch within 1.5 oF and delta flux near target value

Event Seven: LOCA outside CTMT.

When the Evaluation Team is ready, the malfunction is implemented. The malfunction initiates an LOCA outside of CTMT. The RO will diagnose the LOCA based on alarms and indications received, perform the Immediate Actions of 1-AP-16.00, Excessive RCS Leakage, isolate letdown, and place CH in manual to quantify the leakage. When the RO determines that leakage is beyond the capability of a single CH pump, the SRO will direct the RO to trip the reactor and manually safety inject. Safety Injection will actuate, but failures to CH-MOV-1115B, 1115C, 1115D, 1115E, SI-MOV-1867C, and 1867D will result in a failure of HHSI to inject into the core and all three charging pumps will start and take a suction from the VCT with no Letdown. Additionally the in-service Boric Acid Transfer pump will trip preventing any makeup to the VCT. This will result in the VCT drawing down at approximately 4%/minute. The team will initiate attachments 1-3 to restore SI flow to the core. The Team will perform E-0, transition to ECA-1.2. LOCA Outside CTMT where the leak is isolated, and then transition to 1-E-1 to assess SI termination Criteria

Malfunctions required for MAJOR: Nine

- SI 1502, Cold leg rupture upstream of MOV-SI-1890C.
- SI 1601, Fail Check valve SI-79.
- SI1604, Fail Check valve SI-241.
- SI 34, Disable SI-MOV-1867C auto open.
- SI 35, Disable SI-MOV-1867D auto open.
- CH50, Disable CH-MOV-115B auto open.
- CH51, Disable CH-MOV-115C auto close.
- CH52, Disable CH-MOV-115D auto open.
- CH53, Disable CH-MOV-115E auto close.

Additional BOP malfunctions: Three

- BD01, 1-BD-TV-100A fails to auto close.
- BD02, 1-BD-TV-100B fails to auto close.
- VS0101, 1-VS-F-58A fails to auto start.

Objectives: (RO) Isolate letdown per AP-16.00. Quantify leakage as > 150 gpm, and perform immediate actions of tripping the reactor and manually initiating SI at step 4. Identify failure of HHSI to inject to the core. Restore HHSI as directed prior to loss of suction to the HHSI pumps.

(BOP) Using E-0 Attachments, identify and close A SG BD TVs, Start 1-VS-F-58A filtered exhaust fan, and restore HHSI as directed.

(SRO) Direct actions per E-0, ECA-1.2, and E-1 to isolate the leak and restore HHSI to the core before power is lost, or suction is lost to HHSI pumps.

Success Path: Leak is isolated, RCS pressure is stable or rising, and HHSI restored to the core.

Scenario Recapitulation

Total Malfunctions:	5
Abnormal Events:	5 (ARP RM-R8, 1-AP-31.00, 0-AP-53.00, 0-AP-12.00, 1-AP-16.00)
Major Transients:	1 (LOCA outside Containment)
EOPs Entered:	2 (E-0, E-1)
EOP Contingencies:	1 (ECA-1.2)

Event Description: Swap Running EH Pumps per 1-OP-EH-001, section 5.6.

Cue: When team ready.

Time	Position	Applicant's Action or Behavior
	SRO/BOP	1-OP-EH-001, Section 5.6 Team will pre-brief Initial Conditions, Precautions and Limitations, and procedure prior to entering simulator.
	SRO/BOP	1-OP-EH-001, Precautions and Limitations **Note**: The following will be completed by the Team prior to Simulator entry. Precautions and Limitations of 1-OP-EH-001, Electro-Hydraulic Fluid System (EHC): 4.1 The EHC fluid (Fyrquel) has caustic properties. Adequate precautions must be used to avoid direct contact with Fyrquel EHC fluid. If contact with Fyrquel EHC fluid occurs, then the affected area should be immediately flushed with copious amounts of water. 4.2 The Fyrquel fluid removes paint and varnish, reacts with rubber and on most wire insulating material. Caution should always be used when handling the fluid, and spills must be cleaned up immediately. 4.3 Operation of the EHC Fluid System below 50°F is prohibited under any circumstances. External heaters must be used if the EHC Fluid supply system is operated in low ambient temperatures. 4.4 Prolonged operation with fluid temperature between 50°F and 70°F is NOT recommended. A Westinghouse vendor-supplied emergency startup procedure should be implemented in the EHC fluid temperature range of 50°F to 70°F.

Event Description: Swap Running EH Pumps per 1-OP-EH-001, section 5.6.

Cue: When team ready.

		1-OP-EH-001, Precautions and Limitations 4.5 All EHC fluid (Fyrquel) must be filtered before the fluid is added to the EHC Reservoir. 4.6 At an EHC Fluid high temperature alarm (1TS-C3, E-H FLUID RES HI TEMP) setpoint of 140°F, the Bearing Cooling Water to coolers, 1-EH-E-1 and 1-EH-E-2, should be checked for adequate bearing cooling water flow. 4.7 The EHC Reservoir should not be filled above the high alarm (1TS-C1, E-H FLUID RES HI LEVEL) setpoint of 22 inches (21.5 inches to 22.5 inches) from the bottom of the EHC Reservoir. This precaution is applicable when the Turbine is in the latched or unlatched condition. 4.8 The normal operating level in the EHC Reservoir is 5/8 to 11/16 full as indicated on local level gauge 1-EH-LI-100. The local level gauge should read approximately ¾ FULL at the alarm setpoint for high level alarm 1TS-C1, E-H FLUID RES HI LEVEL.
	SRO/BOP	1-OP-EH-001, Precautions and Limitations, Continued 4.9 The polishing filter system should be left in service continuously during normal operation. 4.10 The blocking device for 1-EH-14 alignment to left side cooler is stored in the Ops M&TE Locker. The device requires a 3/16 inch allen wrench to remove or install. Blocking device is not required when aligned to both sides or right side.
	SRO/BOP	Step 5.6.1, 1-OP-EH-001 NOTE: High differential press across either of the EHC Pump, 1-EH-P-MP-1 or 1-EH-P-MP-2, discharge filters will cause alarm 1TS-C4, E-H STRAINERS HI DIFF PRESS, to actuate in the MCR at approximately 100 psi (90 psig to 110 psig). 5.6.1 Swap EHC Pumps IAW the following. Enter N/A if not required. a. Start the standby EHC Pump. (✓) 1-EH-P-MP-1 1-EH-P-MP-2 **BOP Starts 1-EH-MP-2** b. Stop the pump that was initially running. (✓) 1-EH-P-MP-1 1-EH-P-MP-2 **BOP Stops 1-EH-MP-1**

Event Description: Swap Running EH Pumps per 1-OP-EH-001, section 5.6.

Cue: When team ready.

	SRO/BOP	Step 5.6.2, 1-OP-EH-001 NOTE: Alarm 1TS-D4, E-H FLUID DRN RET HI PRESS, will actuate in the MCR at approximately 30 psig (28.5 psig to 31.5 psig), indicating a dirty or clogged filter. 5.6.2 IF blocking device is installed on 1-EH-14, THEN remove blocking device. Otherwise, enter N/A. **BOP Enters N/A for this step.**
	SRO/BOP	Step 5.6.3, 1-OP-EH-001 5.6.3 Swap Drain Return Filters by repositioning 1-EH-14, EH DRAIN RET 3-WAY XFR VALVE TO EH CLRS, to allow flow through the alternate filter. Enter N/A if not required. **BOP Enters N/A for this Step.**
	SRO/BOP	Step 5.6.2, 1-OP-EH-001 5.6.4 IF 1-EH-14 is positioned to the left side cooler/filter, THEN install blocking device on 1-EH-14. Otherwise, enter N/A. **BOP Enters N/A for this Step**.
	SRO/BOP	1-OP-EH-001 BOP Signs and Dates Procedure Section. Reports completion of EH pump swap to SRO.
		END EVENT 1

Event Description: Containment gas RM fails, no Auto Actions.

Cue: By Examiner.

Time	Position	Applicant's Action or Behavior
	BOP	**1-RM-R8 Actions** NOTE: BOP will be handed Steps 3 and 4 of ARP to verify auto actions of CTMT GAS High alarm.
	BOP	3 VERIFY CTMT PURGE SYS - ISOLATED a) CTMT PURGE SUP - CLOSED • 1-VS-MOV-100A • 1-VS-MOV-100B • 1-VS-MOV-100C • 1-VS-MOV-100D *BOP will verify Purge valves closed on the Unit 1 Ventilation Panel by observing RED and GREEN lights extinguished, and GREEN magnet over Shut position indicating light signifying Valve Closed.* b) CTMT PURGE SUP fans - STOPPED • 1-VS-F-4A • 1-VS-F-4B *BOP will identify Purge Supply Fans secured by observing RED and GREEN indicating lights extinguished.* c) CTMT PURGE BYP valve - CLOSED • 1-VS-MOV-101
	BOP	4 VERIFY CTMT IA SYSTEM ALIGNMENT: • 1-IA-TV-101A - CLOSED • 1-IA-TV-101B – CLOSED *Identifies 1-IA-TV-101A / B Open by observing position indicating lights on Unit 1 Vertical board RED Light LIT.* Step 4 RNO Manually Align Valves **BOP will close the valves by depressing and holding the GREEN indicating light until only the GREEN indicating light is lit.** **NOTE:** *When either 1-IA-TV-101A / B are closed, 1-IA-AOV-103 will open (checked in Step 5 below).*

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Event Description: Containment gas RM fails, no Auto Actions.

Cue: By Examiner.

	BOP	5 CHECK CTMT IA COMPR OUTSIDE SUCT - OPEN • 1-IA-AOV-103 BOP will observe 1-IA-AOV-103 OPEN by observing RED position indicating light LIT.
		END EVENT 2

Event Description: Pressurizer Pressure control transmitter 1-RC-PT-1445 fails HIGH.

Cue: When initiated by Team

Time	Position	Applicant's Action or Behavior
	RO	Diagnoses the event by the following indications: Alarms: • 1C-F8, PRZR HI PRESS • 1D-H4, PRZR SFTY VV PWR RELIEF VV OPEN Indications: • 1-RC-PCV-1456, PRZR PORV, RED Open Lights LIT. • RCS Pressure Decreasing
	SRO	Enters 0-AP-53.00 LOSS OF VITAL INSTRUMENTATION / CONTROLS **Note: It is entirely possible that crew may enter 1-AP-31.00 directly before 0-AP-53.00. Either way is correct. Actions per 1-AP-31.00 included after step 6.**
	RO	[1] Check REDUNDANT INSTRUMENT CHANNEL(S) INDICATION - NORMAL Identifies RCS Pressure decreasing using 1-RC-PI-1444 or RCS Wide Range Pressure Indicated on A / B ICCM Display.
	RO RO/BOP	[2] PLACE AFFECTED CONTROL(S)/COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION **Places 1-RC-PV-1456 control switch in Close**. *Verifies 1-RC-PCV-1456 closes by GREEN Light LIT, RED Light off.* *During pressure transient, the Rod Control Switch may be placed in MANUAL due to Rods stepping OUT due to pressure coefficient.*
	SRO	Conducts a Brief summarizing the Event and Establish priorities. RO will provide Critical Parameters using Brief Placard. BOP will provide Critical Parameters using Brief Placard. *The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.* Completes Brief and continues with AP-53.00

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Event Description: Pressurizer Pressure control transmitter 1-RC-PT-1445 fails HIGH.

Cue: When initiated by Team

	SRO RO	*3 VERIFY REACTOR POWER – LESS THAN OR EQUAL TO 100% *Identify that reactor power is less than 100%.*
	SRO	Notes Prior to Step 4: • Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud. • When the affected instrument/controller malfunction(s) has been addressed by this procedure, recovery actions should continue at Step 13.
	SRO RO	*4 DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE: • PRZR Pressure Control, Step 5 *The RO will identify that 1-RC-PI-1445 failed high.*
	  SRO RO	NOTE: RCS pressure decrease will cause a slight decrease in RCS Tave due to negative reactivity from the moderator pressure coefficient. 5 CHECK PRZR SPRAY VALVE CONTROLLERS - NORMAL *RO identifies PRZR Spray Valves controllers normal.*
	SRO	6 GO TO ( )-AP-31.00, INCREASING OR DECREASING RCS PRESSURE **SRO announces transition to AP-31.00**.

Op-Test No.: Surry 2016-1 Scenario No.: 3 Event No.: 3

Event Description: Pressurizer Pressure control transmitter 1-RC-PT-1445 fails HIGH.

Cue: When initiated by Team

	SRO RO	1-AP-31.00 [1] ___ CHECK PRZR PORVS – CLOSED *Identifies 1-RC-PCV-1456 closed.*
		1-AP-31.00 Conducts a Brief summarizing the Event and Establish priorities. RO will provide Critical Parameters using Brief Placard. BOP will provide Critical Parameters using Brief Placard. *The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.* Completes Brief and continues with AP-31.00
	SRO RO	1-AP-31.00 **CAUTION**: A Safety Injection may occur if the unit is not tripped prior to RCS pressure decreasing below 2100 psig. 2 CHECK RCS PRESSURE – DECREASING *Identifies RCS Pressure rising following PORV Closure.* 2 RNO IF RCS pressure is stable or increasing following PORV closure, THEN GO TO Step 10.
	SRO RO	1-AP-31.00 10 CHECK RCS PRESSURE - STABILIZING AT OR TRENDING TO 2235 PSIG *Identifies RCS pressure trending to 2235 psig.*
	SRO	1-AP-31.00 11. GO TO STEP 17

Event Description: Pressurizer Pressure control transmitter 1-RC-PT-1445 fails HIGH.

Cue: When initiated by Team

	SRO RO	1-AP-31.00 17 CHECK MASTER CONTROLLER - IN MANUAL GO TO Step 19 *Identifies Master Pressure Controller in Auto*
	SRO RO	1-AP-31.00 19 CHECK PRZR PORVS – EITHER INOPERABLE • 1-RC-PCV-1455C • 1-RC-PCV-1456 *Identifies 1-RC-PCV-1456 Inoperable.*
	SRO RO	1-AP-31.00 20 CLOSE BLOCK VALVE FOR INOPERABLE PORV • 1-RC-MOV-1535 if 1-RC-PCV-1456 inoperable **Places 1-RC-MOV 1535 control switch in close. Monitors MOV until GREEN indicating light LIT and RED Light out**.
	SRO RO	1-AP-31.00 21 CHECK PRZR PORVS – EITHER INCAPABLE OF BEING MANUALLY CYCLED GO TO Step 23 • 1-RC-PCV-1455C • 1-RC-PCV-1456 States Both PORVs are capable of being manually cycled.
	SRO	1-AP-31.00 23 NOTIFY THE FOLLOWING: • OM on call • STA • I&C Notifies Shift Manager of the Event and requests notifications be made.
	SRO RO	1-AP-31.00 17 CHECK MASTER CONTROLLER - IN MANUAL GO TO Step 19 *Identifies Master Pressure Controller in Auto*

Event Description: Pressurizer Pressure control transmitter 1-RC-PT-1445 fails HIGH.

Cue: When initiated by Team

	SRO	1-AP-31.00 24 REFER TO TECH SPECS: • 3.1.A.5 • 3.1.A.6 • 3.1.C • 3.12.F 1) The SRO will consult Tech Specs, TS Section 3.1.A.6, and determine that 1-RC-PCV-1456 is Inoperable but capable of being manually cycled, and enter a one hour clock to close the block valve for the PORV. The block valve will remain energized. 2) Further consultation will find Tech Spec 3.12.F.1 to restore RCS pressure >2205 psig in 2 hours or reduce Thermal Power to less than 5% in the next 6 hours. The Core Operating Limits Report (COLR, DRP-21) provides the reference for 2205 psig RCS pressure. **NOTE**: Following closure of block valve the 1 hour Tech Spec clock clears. RCS pressure decreases to < 2205 psig during the transient; once the PORV has been closed, pressure recovers quickly and goes above 2205 psig. *SRO updates the Team concerning the identified items in Tech Specs and updates the Shift manager.*
	SRO	1-AP-31.00 25 REVIEW APPLICABILITY: • VPAP-2802 • EAL Matrix SU6.1 *Notifies Shift Manager of need to review VPAP-2802 and EALs for this Event.*
	SRO	1-AP-31.00 26 RESTORE PRESSURE CONTROL SYSTEM(S) TO NORMAL
		END EVENT #3

Event Description: A SG NR Level Channel III fails HIGH.

Cue: By Examiner.

Time	Position	Applicant's Action or Behavior
	BOP	Diagnoses channel failure with the following indications/alarms: Alarms: • 1H-A5 – STM GEN 1A HI LVL • 1F-C10 – STM GEN 1A HI LVL CH 3 • 1H-G5 – STM GEN 1A LVL ERROR • 1H-E5 – STM GEN 1A FW>< STM FLOW • 1F-C7 – STM GEN 1A CH 3 FW< STM FLOW • 1F-D7 – STM GEN 1A CH 4 FW< STM FLOW Indications: • 1-FW-FCV-1478 respond to level channel failure by opening in automatic
	SRO	Enters 0-AP-53.00, Loss of Vital Instrumentation / Controls.
	RO	0-AP-53.00 [1] CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION - NORMAL Verifies Channel I and II are NORMAL.
	BOP	0-AP-53.00 [2] PLACE AFFECTED CONTROL(S)/COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION **BOP takes manual control of 'A' SG feed reg valve and raises demand (FF > SF) to restore level to program**. **Critical Task CT-1:** Should the BOP fail to take action to control A SG NR level, an automatic reactor trip will occur. This would be considered failure criteria.
	SRO	0-AP-53.00 The team will hold a transient brief. During the brief the failure of 1-FW-LI-1476 will be discussed. *The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.*

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Event Description: A SG NR Level Channel III fails HIGH.

Cue: By Examiner.

	RO	*3. VERIFY REACTOR POWER – LESS THAN OR EQUAL TO 100%. RO will identify that reactor power, by ΔT, is less than 100%.
	SRO RO	0-AP-53.00, Loss of Vital Instrumentation / Controls **NOTE:** • Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud. • When the affected instrument/controller malfunction(s) has been addressed by this procedure, recovery actions should continue at Step 13. *4 DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE:. • SG NR Level, Step 7 The RO will identify that 1-FW-LT-1476 has failed <u>HIGH.</u>
	BOP	0-AP-53.00 7. CHECK STEAM GENERATOR LEVEL CONTROL INSTRUMENTS – NORMAL • Steam Pressure • Steam Flow • Feed Flow • **Steam Generator Level** *Determines CH III Steam Generator Level instrumentation for 'A' SG is NOT normal*

Event Description: A SG NR Level Channel III fails HIGH.

Cue: By Examiner.

		X0-AP-53.00 Step 7. RNO
	SRO	IF the selected steam flow, steam pressure, or feed flow input to the SG Water Level Control system has failed, THEN do the following: *Determines that none of the listed instruments are affected and continues in the RNO column.*
	SRO	IF SG Level Channel III has failed, THEN do the following:
	BOP	a) Place the associated Feed Reg Valve in MANUAL. b) IF manual control of Feedwater is inoperable, THEN do the following: 1) Control SG level with Feedwater Isolation MOVs. 2) Consult with the Shift Manager concerning the need to place the MFRV on the jack.
	BOP	c) Control SG level at program level.
	BOP	d) Main Feed Reg Valve Bypass Valves may be used for fine control of SG level. Perform follow-up actions:
	SRO	a) Consult with Shift Manager on need to initiate ( )-OP-RP-001, ALIGNING CONTROL SYSTEM FOR PERFORMANCE OF CHANNEL I, II, III, AND IV PROCESS AND PROTECTION TESTING. b) Refer to the following Tech Spec 3.7 items:
	SRO	• Table 3.7-1, **12** and 17 • Table 3.7-2, 1.c, 1.e, and **3.a** • Table 3.7-3, 2.a, and **3.a** • Table 3.7-6 items 15 & 16

Event Description: A SG NR Level Channel III fails HIGH.

Cue: By Examiner.

		0-AP-53.00
		Step 7. RNO (Continued)
	SRO	SRO should determine the follow Technical Specification implications for this failure:
		Technical Specification:
	SRO SRO	• Tech Spec Table 3.7-1, Item 12, Operator Action 6, With the number of OPERABLE channels less than the Total Number of Channels, REACTOR CRITICAL and POWER OPERATION may proceed provided the following conditions are satisfied: The inoperable channel is placed in the tripped condition within 72 hours, The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1, If the conditions are not satisfied in the time permitted, be in at least HOT SHUTDOWN within 6 hours
	SRO	• Tech Spec Table 3.7-2, Item 3.a, Operator Action 20; With the number of OPERABLE channels less than the Total Number of Channels, REACTOR RITICAL and/or POWER OPERATION may proceed provided the following conditions are satisfied: The inoperable channel is placed in the tripped condition within 72 hours, The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1, If the conditions are not satisfied in the time permitted, be in HOT SHUTDOWN within the next 6 hours and reduce RCS temperature & pressure to less than 350°F/450 psig, respectively in the following 12 hours.
		• Tech Spec Table 3.7-3, Item 3a, SG Water Level Hi-Hi, Operator Action 20; With the number of OPERABLE channels less than the Total Number of Channels, REACTOR RITICAL and/or POWER OPERATION may proceed provided the following conditions are satisfied: The inoperable channel is placed in the tripped condition within 72 hours, The Minimum OPERABLE Channels requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.1, If the conditions are not satisfied in the time permitted, be in HOT SHUTDOWN within the next 6 hours and reduce RCS temperature & pressure to less than 350°F/450 psig, respectively in the following 12 hours.
		c) Refer to Attachment 1.
		d) IF no other instrumentation failure exists, THEN GO TO Step 13

Event Description: A SG NR Level Channel III fails HIGH.

Cue: By Examiner.

	SRO	0-AP-53.00 13. CHECK CALORIMETRIC – FUNCTIONAL IAW ( )-OPT-RX-001 If the FRV bypass valves are manipulated, the SRO/RO determines that OPT-RX-001 is impacted and OPT-RX-007 will need to be performed. *If it appears that the SRO/RO will take action to perform 1-OPT-RX-007, the Unit 2 Operator will state that he will have the fourth RO perform 1-OPT-RX-007.* *A copy of 1-OPT-RX-001 is included following step 17.*
	SRO	0-AP-53.00 14. REVIEW THE FOLLOWING • Tech Spec 3.7 *Determines Table 3.7-1 item 17, Operator action 6, is applicable (place channel in trip w/in 72 hours).* VPAP-2802, NOTIFICATIONS - None TRM SECTION 3.3, INSTRUMENTATION – None. Reg. Guide 1.97: (Potential follow up question) • SG NR level, A-01 Variable, 2 Channel per SG required, see TS Table 3.7-6. Identification is usually performed by SM/STA. EP-AA-303, EQ. IMPORTANT TO EMERGENCY RESPONSE (Potential follow up question). Heat Sink Capability, A SG NR Level CH III, Att. 6 Page 6 of 7, Category B variable, Alternate indications available within grouping, no action required. Identification is usually performed by SM/STA. *The STA reports he has completed review and has discussed the results with the Shift Manager.*
	SRO BOP SRO	0-AP-53.00 15 CHECK ADDITIONAL INSTRUMENT / CONTROLLER MALFUNCTION – EXISTS Reports no additional failure exists GOES TO Step 17

Op-Test No.: Surry 2016-1 **Scenario No.: 3** **Event No.: 4**

Event Description: A SG NR Level Channel III fails HIGH.

Cue: By Examiner.

	SRO	0-AP-53.00 17. PROVIDE NOTIFICATIONS AS NECESSARY: • Shift Supervision • OMOC • STA (PRA determination) • I&C

Event Description: A SG NR Level Channel III fails HIGH.

Cue: By Examiner.

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Attachment 4

CALORIMETRIC PROGRAM OPERABILITY

NOTE: TRM 3.3.5 shall be reviewed for required actions for non-functionality of the UFM Calorimetric, Normalized Feedwater Venturi, or PCS Calorimetric. Power reduction to less than 98.4% may be required.

1. To check the Primary Plant Performance Program (PP) operability perform the following:

_______ a. Open Programs - Operator Display / Engineering Display

_______ b. Open PP Output Summary - (Operator Display - Primary Plant Poke)

_______ c. Check short timed values for selected basis, Steam Flow (1-OPT-RX-002 box) or Feedflow (1-OPT-RX-003 box), are updating and either good or fair quality.

_______ d. IF selected basis NOT updating and either good or fair quality, THEN contact Reactor Engineering (if available), and then select another calorimetric basis IAW Step 6.3.1. Otherwise, enter N/A.

2. To check the Flow Corrections Program (FL) operability perform the following:

_______ a. Open Programs - Operator Display / Engineering Display

_______ b. Open FL Output Summary (Operator Display - Flow Corr Poke)

_______ c. Check FL Program Status is OK. IF NOT OK, THEN perform the following to check status of different bases.

_______ 1. Open FL0101 - Output Summary (FL Summary Poke)

_______ 2. Compare displayed values to the FL0101 Table below and check selected calorimetric values are updating and either good or fair quality.

_______ 3. IF selected basis NOT operable, THEN contact Reactor Engineering (if available), and then select another calorimetric basis IAW Step 6.3.1.

Event Description: A SG NR Level Channel III fails HIGH.

Cue: By Examiner.

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Attachment 4

CALORIMETRIC PROGRAM OPERABILITY

FL0101 Table				
Flow Correction List	**Normalized Feedwater**	**UFM Feedwater**	**Venturi Feedwater**	**Steam Flow**
Charging Line Flow Corr	X	X	X	X
Letdown Line Flow Corr	X	X	X	X
SG A-1 FF CORR		X	X	
SG A-2 FF CORR		X	X	
SG B-1 FF CORR		X	X	
SG B-2 FF CORR		X	X	
SG C-1 FF CORR		X	X	
SG C-2 FF CORR		X	X	
SG A-1 SF CORR				X
SG A-2 SF CORR				X
SG B-1 SF CORR				X
SG B-2 SF CORR				X
SG C-1 SF CORR				X
SG C-2 SF CORR				X
SG A-1 FF CORR NORM	X			
SG A-2 FF CORR NORM	X			
SG B-1 FF CORR NORM	X			
SG B-2 FF CORR NORM	X			
SG C-1 FF CORR NORM	X			
SG C-2 FF CORR NORM	X			

		END EVENT #4

Op-Test No.: Surry 2016-1 **Scenario No.: 3** **Event No.: 5** **Page 33 of 81**

Event Description: Trip of the running CH pump SW pump with failure of standby pump auto start.

Cue: When initiated by Team.

Time	Position	Applicant's Action or Behavior
	RO	1-SW-P-10A overload trip / 1-SW-P-10B fail to auto start Diagnose the failure based on the following alarms and indications: Annunciator 1D-G5, SW of CC PPS DISCH TO CHG PPS LO PRESS. 1-SW-P-10B NOT running. **Note**: SRO may conduct a Team brief and direct the start of 1-SW-P-10B.
	BOP RO	1D-G5 Annunciator Response Procedure 1. CHECK CHG PUMP CC OR SW PP(S) - TESTING IN PROGRESS Reports No, testing not in progress. Step 1 RNO: Go TO Step 3.
	BOP RO BOP RO BOP RO/BOP BOP RO BOP BOP RO	1D-G5 Annunciator Response Procedure **Note prior to Step 3**: The standby CHG Pump SW Pump will auto-start at 8 psig. **NOTE**: SRO may hold a focus brief for starting 1-SW-P-10B; RO starts 1-SW-P-10B following closure of the brief. 3. CHECK STANDBY CHG PUMP CC PP OR SW PP - AUTO STARTED Report No, 1-SW-P-10B not running. Step 3 RNO DO the following: a) Locally check CHG Pump CC and SW PPs. Dispatch an Operator to check the status of the CH Pump CC and SW pumps. b) Monitor CHG Pump CC and SW flows on PCS (ERFCS if not removed): 1-CC-P-2A, F1CC003A 1-CC-P-2B, F1CC004A 1-SW-P-10A, F1SW007A 1-SW-P-10B, F1SW008A RO/BOP Monitors parameters using the PCS c) IF CHG Pump CC and SW PPs are operating normally, THEN do the following: Reports No, Pumps are not operating properly. d) IF CHG Pump CC or SW PP NOT operating normally, THEN do the following: 1) Swap CHG CC or SW PPs. **Starts 1-SW-P-10B**

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Event Description: Trip of the running CH pump SW pump with failure of standby pump auto start.

Cue: When initiated by Team.

		1D-G5 Annunciator Response Procedure, Step 3 RNO (Continued).
	BOP	2) IF standby CHG Pump CC or SW PP unavailable, THEN return lead pump to service.
		Reports Yes standby pump is running.
	BOP	3) Submit Plant Issue and Work Request
	BOP	Notifies SRO that a CR and work request is required.
	BOP	4) GO TO Step 13.
		1D-G5 Annunciator Response Procedure
	BOP	13 PROVIDE NOTIFICATIONS: OMOC STA SYSTEM ENGINEERING Notifies SRO of required notifications.
	SRO	Either Tech Spec 3.3.B.3 or 3.2.C.1 may be used by SRO to determine Required Action which is Tech Spec 3.01. Tech Specs: TS 3.3. B.3, allows ONE Train of HHSI pumps to be Inoperable. With Both Trains Inoperable **TS 3.01** is in effect, requiring the Unit be placed in HSD in 6 hours, and CSD in the next 30 hours. **Note**: Change of TS clock to 72 hours would be a Licensing/Station Management decision. Note: Tech Spec 3.2.C.1 is also applicable: No Boron injection subsystems OPERABLE, all CH pumps Inoperable due to the Inoperable CH PP SW pumps being Inoperable. With Both Trains Inoperable **TS 3.01** is in effect, requiring the Unit be placed in HSD in 6 hours, and CSD in the next 30 hours. TRM: 1-SW-P-10A, MRule – No, App 'R' – yes. TRM 3.7.9.A.2, Implement App R fire watch in the area(s) associated with the nonfunctional equipment in Table 3.7.9-1 in accordance with TRM Section 5.2 within 14 days and restore the equipment to functional status in 60 days. Hourly Fire Watch required in Unit 1 ESR, Unit 1 and 2 Turbine Buildings Basement North Wall, MER 3&4. **Recommend TRM referral post-scenario since this function is normally performed by the STA with consult with the Shift Manager**.
		---END OF EVENT 5---

Op-Test No.: Surry 2016-1 Scenario No.: 3 Event No.: 6

Event Description: Ramp Unit to HSD

Cue: When initiated by Team.

Time	Position	Applicant's Action or Behavior
	 SRO RO SRO	**Start of 0-AP-23.00** Conducts a Brief summarizing the Event and Establish priorities. RO will provide Critical Parameters using Brief Placard. BOP will provide Critical Parameters using Brief Placard. *The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.* Reactivity control during AP-23.00 Ramp: 453 gallons of Boric Acid needed to reduce power to HSD. 50 gallons added during Emergency Boration (estimate), leaving 403 gallons to be added using normal boration at an average rate of 5.0 gpm. Control Bank 'D' rod height at end of ramp 165 Steps. Completes Brief and continues with AP-23.00.
	SRO	Caution Prior to Step 1: • Conservative decision-making must be maintained during rapid load reductions. Refer to Attachment 1 for trip criteria. Notes Prior to Step 1: • Actions that can be completed independently of preceding steps may be performed out of sequence as directed by the SRO • When the Turbine is not being actively ramped, the REFERENCE and SETTER values must remain matched to prevent inadvertent ramp. • Pre-planned reactivity plans located in the Main Control Room will be used as guidance for ramping down to the desired power level. • The ramp rate in IMP OUT is nonlinear and therefore pre-planned reactivity plans based on IMP IN are not as accurate. However, total amounts of boration and dilution can be used as guidance. • For ramp rates greater than or equal to 1%/minute, Rod Control should remain in Automatic if available.
	  RO	0-AP-23.00 1. **TURN ON ALL PRZR HEATERS**

Event Description: Ramp Unit to HSD

Cue: When initiated by Team.

	BOP	2. INITIATE PLANT LOAD REDUCTION AT 2%/MINUTE OR LESS: a) Verify turbine valve position - NOT ON LIMITER The turbine is NOT on the limiter.
	RO	b) Insert control rods in AUTO or MANUAL as necessary to maintain Tave and Tref within 5°F.
	SRO/BOP	c) **Verify or place turbine in IMP IN or IMP OUT as determined by Shift Supervision** The SRO can choose IMP IN or IMP OUT.
	BOP	d) **Adjust SETTER to desired power level** e) **Adjust LOAD RATE %/MIN thumbwheel to desired ramp rate (1%/minute)** f) **Initiate Turbine load reduction using OPERATOR AUTO (pushes the GO button)** g) **Reduce Turbine Valve Position Limiter as load decreases** The BOP will periodically reduce the limiter setpoint during the ramp.
	SRO	3. CHECK EMERGENCY BORATION – REQUIRED The team may decide to emergency borate after the ramp has progressed to the point that Tave and Tref are matched (or close).
	SRO	Note Prior to Step 4: • Step 4 or Step 5 may be performed repeatedly to maintain Tave and Tref matched, ΔFlux in band, and control rod position above the LO-LO insertion limit.

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Event Description: Ramp Unit to HSD

Cue: When initiated by Team.

	RO	0-AP-23.00 4. PERFORM AN EMERGENCY BORATION IAW THE FOLLOWING: a) Verify or raise CHG flow to greater than 75 gpm b) **Transfer the in-service BATP to FAST** c) **Open 1-CH-MOV-1350** SRO will direct nominal opening of approximately 25 seconds. d) Monitor EMRG BORATE FLOW • ( )-CH-FI-( )110 e) After required emergency boration, perform the following: 1) **Close ( )-CH-MOV-( )350** 2) **Transfer the in-service BATP to AUTO** 3) Restore Charging flow control to normal SRO may direct rod motion to maintain Δ Flux within specified band.
	RO	5. ESTABLISH A NORMAL BORATION TO MAINTAIN CONTROL ROD POSITION ABOVE THE LO-LO INSERTION LIMITS IAW ATTACHMENT 4 Attachment 4 (Boration) and 5 (Manual Makeups) are at the end of this section. SRO may direct manual rod motion to maintain Δ flux within specified band.
	SRO	Notes Prior to Step 6: • If at any time plant conditions no longer require rapid load reduction, actions should continue at Step 35. • RCS Tave must be maintained less than or equal to 577°F and RCS pressure must be maintained greater than or equal to 2205 psig. Tech Spec 3.12.F.1 should be reviewed if either parameter is exceeded. • I & C should be contacted to provide assistance with adjusting IRPIs.
	RO	6. CONTROL RAMP RATE TO MAINTAIN RCS PRESSURE GREATER THAN 2205 PSIG

Op-Test No.: Surry 2016-1 Scenario No.: 3 Event No.: 6 Page 38 of 81

Event Description: Ramp Unit to HSD

Cue: When initiated by Team.

	RO	0-AP-23.00 *7. CHECK LETDOWN ORIFICES – TWO IN SERVICE *Evaluator note: two orifices will already be in service.*
	BOP	8. MONITOR STEAM DUMPS FOR PROPER OPERATION
	SRO	0-AP-23.00 9. NOTIFY THE FOLLOWING: • Energy Supply (MOC) • Polishing Building • Chemistry • OMOC
	SRO	10. EVALUATE THE FOLLOWING: • EPIP applicability *The Shift Manager will review EPIPs for applicability. They are not applicable.* • VPAP-2802, NOTIFICATIONS AND REPORTS, applicability *SRO directs STA to review VPAP-2802. The STA reports that he has completed his review of VPAP-2802 and no notifications are required.* *No further actions are required for this event.*
	SRO	11. CHECK RAMP WILL BE TO LESS THAN APPROXIMATELY 35% REACTOR POWER No, go to step 12.
	SRO	12. CHECK REACTOR POWER – HAS LOWERED MORE THAN 15% IN ONE HOUR. When reactor power has lowered >15%, then chemistry will be notified.
	SRO	13. NOTIFY CHEMISTRY OF POWER CHANGE > 15% IN ONE HOUR. Chemistry notified of power change > 15% in one hour.

Event Description: Ramp Unit to HSD

Cue: When initiated by Team.

	SRO	0-AP-23.00 CAUTION: Secondary plant evolutions affecting Feedwater Flow or temperature will affect RCS temperature and Reactor Power. This effect will be greater at beginning of core life due to a lower value for isothermal temperature coefficient. The operating team must be prepared to mitigate the effects of the secondary evolutions on the RCS. *RO acknowledges the Caution.*
	SRO	0-AP-23.00 14. AT APPROXIMATELY 70% REACTOR POWER CHECK AUXILIARY STEAM MAINTAINING BETWEEN 160 AND 180 PSIG. *RO Acknowledges the step.*
		END EVENT #6

Event Description: Ramp Unit to HSD

Cue: When initiated by Team.

		0-AP-23.00 Attachment 4 (NORMAL BORATION) Actions
	RO	1. **Place the MAKE-UP MODE CNTRL switch in the STOP position.**
	RO	2. **Adjust 1-CH-YIC-1113 to desired total gallons**
	RO	3. **Adjust 1-CH-FC-1113A to desired flow rate.**
	RO	4. **Place the MAKE-UP MOD SEL switch in the BORATE position**.
	RO	5. **Place the MAKE MODE CNTRL switch in the START position.**
	RO	6. Verify proper valve positions.
	RO	7. Adjust boration rate using 1-CH-FC-1113A, as necessary.
	RO	8. WHEN boration is complete, THEN perform the following. IF boric acid is to remain in the Blender to support ramping the Unit, THEN enter N/A. a) Manually blend approximately 20 gallons to flush the boration path IAW Attachment 5, Manual Makeups. b) Enter N/A for the remaining steps in this Attachment. *Attachment 5 is on the next page*
	RO	9. Verify controllers for Primary Grade water and Boric Acid are set correctly.
	RO	10. **Place the MAKE-UP MODE SEL switch in the AUTO position.**
	RO	11. **Place the MAKE-UP MODE CNTRL switch in the START position.**
	RO	12. Notify Shift Supervision of blender status.

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Event Description: Ramp Unit to HSD

Cue: When initiated by Team.

		0-AP-23.00 Attachment 5 (Manual Makeups) Actions
		1. Place the MAKE-UP MODE CNTRL switch in the STOP position.
		2. Check controllers for the flow rate of Boric Acid and Primary Grade water are set correctly.
		3. Check integrators for the gallons of Boric Acid and Primary Grade water are set correctly.
		4. Place the MAKE-UP MODE SEL switch in the MANUAL position.
		5. Place the MAKE-UP MODE CNTRL switch in the START position.
		6. Open 1-CH-FCV-1113B, BLENDER TO CHG PUMP.
		7. Check proper valve positions.
		8. WHEN the Manual Makeup operation is complete, THEN place 1-CH-FCV-1 113B in the AUTO position
		9. Place the MAKE-UP MODE CNTRL switch in the STOP position.
		10. Check or place the control switches in the AUTO position.
		11. Check controllers for Primary Grade water and Boric Acid are set correctly.
		12. Place the MAKE-UP MODE SEL switch in the AUTO position.
		13. Place the MAKE-UP MODE CNTRL switch in the START position.
		14. Notify Shift Supervision of blender status.

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Event Description: LOCA Outside Containment

Cue: Evaluator Cue.

Time	Position	Applicant's Action or Behavior
	Team	Identify Failure based on the following: Alarms: 1B-F3, SFGDS AREA SUMP HI LEVEL 1C-B-8, PRZR LO PRESS 1C-D8, PRZR LO LEVEL 1D-E5, CHG PP TO REGEN HX HI-LO FLOW RMA-D6, VENT STACK #2PART ALERT/HI RMA-D7, VENT STACK #2 NORM RNG GAS ALERT/HI Indications: Lowering PRZR Level Rising CH flow
	SRO	1-AP-16.00 Direct RO to perform the Immediate Actions of 1-AP-16.00.
	RO RO SRO	1-AP-16.00 NOTE: • If SI Accumulators are isolated, 1-AP-16.01, SHUTDOWN LOCA, should be used for guidance. • RCS average temperature has a direct impact on pressurizer level. [1] MAINTAIN PRZR LEVEL: • Isolate Letdown Closes 1-CH-LCV-1460 A / B • Control Charging flow Takes Manual control of 1-CH-FCV-1122 and raises CH Flow. Reports that CH flow at maximum value and PZR level decreasing. **Direct RO to perform the Immediate Action Steps of E-0, and Safety Inject on Step 4**.

Op-Test No.: Surry 2016-1 Scenario No.: 3 Event No.: 7

Event Description: LOCA Outside Containment

Cue: Evaluator Cue.

	RO	**<u>1-E-0 – Reactor Trip or Safety Injection</u>** [1] CHECK REACTOR TRIP: a) Manually trip reactor **Pushes the reactor trip push buttons.** b) Check the following: • All Rods On Bottom light – LIT • Reactor trip and bypass breakers – OPEN • Neutron flux - DECREASING
	RO RO	[2] CHECK TURBINE TRIP: a) **Manually trip the turbine** **Pushes the turbine trip push buttons.** b) Verify all turbine stop valves - CLOSED c) Isolate reheaters by closing MSR steam supply SOV • 1-MS-SOV-104 d) Verify generator output breakers – OPEN (Time Delayed)
	RO	[3] CHECK BOTH AC EMERGENCY BUSES – ENERGIZED RO confirms that both AC emergency buses are energized.

Event Description: LOCA Outside Containment

Cue: Evaluator Cue.

	RO
	
	RO
		[4] CHECK IF SI INITIATED:
		
		a) Check if SI is actuated:
		• LHSI pumps – RUNNING
		• SI annunciators – LIT
		  • A-F-3 SI INITIATED – TRAIN A
		  • A-F-4 SI INITIATED – TRAIN B
		
		4b) **Manually initiate SI**
		
		The RO will manually initiate SI at step 4 by pushing both SI pushbuttons.
		
		After the immediate actions of 1-E-0 are reported as complete, the SRO will check off immediate action steps in his copy of 1-E-0. After the immediate actions are verified, the team will conduct a brief.
		
		RO should identify that there is NO SI flow to the core.
		
		The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal, with the exception of the alarms already received. He will also state that containment conditions and the electrical conditions are as you see them (or as reported by the RO/BOP).

	SRO	1-E-0 Will Check off the Immediate Action Steps of 1-E-0. SRO will conduct a commensurate Brief. Ensure the Team agrees that a LOCA outside of containment is in progress. Poll the Team as to any higher priority than continuing actions for a LOCA outside of containment. SRO closes the brief. **Directs BOP to perform Attachments 1, and 3 of 1-E-0**. BOP failures included in next section. **Directs RO to perform Attachment 2** *Attachment 1, 2, and 3 located in next Section.* Continues with 1-E-0 with RO. **Critical Task** **CT-2:** Restore HHSI flow from the RWST to the core prior to air-binding of the HHSI pumps.

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Event Description: LOCA Outside Containment

Cue: Evaluator Cue.

		Note: Approx. 5 min after start of LOCA the Boric Xfer pump will trip. This will result in a rapid VCT level draw-down of 4%/min.
	RO SRO RO SRO RO SRO	1-E-0 Identify RCS Subcooling < 30 °F and SI flow indicated to all three loops. Using 1-E-0 Continuous Actions Page: 1. RCP TRIP CRITERIA Trip all RCPs if BOTH conditions listed below occur: a. Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS b. RCS Subcooling - LESS THAN 30°F [85°F] Checks RCP Trip Criteria exist. Directs RO to Trip all RCPs. 2. MINIFLOW RECIRC CRITERIA a. CLOSED - When RCS pressure is less than 1275 psig [1475 psig] AND RCP Trip Criteria are met (RCPs OFF). Identifies Criteria are met. Direct RO to close mini-flow recirc valves.
	SRO	*6. CHECK RCS AVERAGE TEMPERATURE • STABLE AT 547°F OR • TRENDING TO 547°F

Event Description: LOCA Outside Containment

Cue: Evaluator Cue.

	SRO	7. CHECK PRZR PORVs AND SPRAY VALVES: a) PRZR PORVs – CLOSED b) PRZR spray controls • Demand at Zero (or) • Controlling Pressure c) PORV block valves - AT LEAST ONE OPEN
	SRO	**NOTE:** Seal injection flow should be maintained to all RCPs. *8. CHECK RCP TRIP AND MINIFLOW RECIRC CRITERIA: a) Charging Pumps – AT LEAST ONE RUNNING AND FLOWING TO RCS
	RO	Two or three Charging pumps will be running and flowing to the RCS. b) RCS subcooling - LESS THAN 30°F [85°F] RCS subcooling will be less than 30 °F c) Stop all RCPs d) RCS pressure - LESS THAN 1275 psig [1475 PSIG] e) Close CHG pump miniflow recirc valves: • 1-CH-MOV-1275A • 1-CH-MOV-1275B • 1-CH-MOV-1275C *Mini-flow recirc valves closed.*
	SRO	9. CHECK IF SGs ARE NOT FAULTED: • Check pressures in all SGs a) STABLE OR INCREASING AND b) GREATER THAN 100 PSIG RO will observe a slightly decreasing trend on SG pressures. This will be attributed to the RCS cooldown. The team will not transition to 1-E-2.

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Event Description: LOCA Outside Containment

Cue: Evaluator Cue.

	SRO SRO RO	10. CHECK IF SG TUBES ARE NOT RUPTURED: • Condenser air ejector radiation – NORMAL • SG blowdown radiation – NORMAL • SG MS radiation – NORMAL • TD AFW pump exhaust radiation – NORMAL • SG NR Level - NOT INCREASING IN AN UNCONTROLLED MANNER 11 CHECK RCS - INTACT INSIDE CTMT • CTMT radiation - NORMAL • CTMT pressure - NORMAL • CTMT RS sump level – NORMAL *Reports RCS intact in side CTMT.*
	SRO RO RO SRO	12 CHECK RCS - HAS BEEN MAINTAINED INTACT OUTSIDE CTMT a) Radiation Monitors - NORMAL • MGPI vent-vent • Auxiliary Building Control Area Reports MGPI Vent-Vent in Alarm b) Sump annunciators - NOT LIT • VSP-F-4 • B-D-1 • B-D-2 • B-F-3 Step 12 RNO Determine cause of abnormal conditions. IF the cause is a loss of RCS inventory outside CTMT, THEN GO TO 1-ECA-1.2, LOCA OUTSIDE CONTAINMENT. Reports conditions caused by LOCA outside CTMT **Transitions to ECA-1.2**.

Event Description: LOCA Outside Containment

Cue: Evaluator Cue.

		1-ECA-1.2 CAUTION: Depending on break location, higher than normal dose levels should be expected in the Auxiliary Building and the Safeguards after a LOCA outside CTMT.
	SRO	1 VERIFY PROPER VALVE ALIGNMENT: a) Locally unlock and close the following breakers: • 1H1-2N 8A for 1-SI-MOV-1890A • 1J1-2E 8B for 1-SI-MOV-1890B • 1H1-2N 9A for 1-SI-MOV-1890C **Directs the BOP to contact the Shift Manager/Desk SRO and have an operator briefed and dispatched to remove tags and close the breakers for 1-SI-MOV-1890 A / B / C.** **NOTE**: *Booth will close breakers after 3 min. delay.* b) LHSI to hot legs - CLOSED • 1-SI-MOV-1890A • 1-SI-MOV-1890B *When Valves Energized, the valves indicate closed.* c) SI accumulator test valves - CLOSED • HCV-SI-1850A • HCV-SI-1850B • HCV-SI-1850C • HCV-SI-1850D • HCV-SI-1850E • HCV-SI-1850F *When Checked, Valves Indicate closed.*

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Event Description: LOCA Outside Containment

Cue: Evaluator Cue.

	SRO	1-ECA-1.2 NOTE: The SRO is expected to have a Focus Brief prior to this step. The SRO will bring the BOP into the discussion concerning the expected response of the RCS when 1-SI-MOV-1890C is closed. If closure of this valve isolates the RCS break, RCS pressure and LHSI flow is expected to increase. If the break remains isolated, RCS pressure and LHSI flow will remain the same or decrease. The SRO will complete the brief and continue ECA-1.2.
	RO	2 TRY TO IDENTIFY AND ISOLATE BREAK: a) Close LHSI to cold legs • **1-SI-MOV-1890C**
	Team	**Shortly after closure of 1-SI-MOV-1890C, the Team should identify that RCS pressure and LHSI flow is increasing.**
		Critical Task **CT-3: Isolate LOCA outside of Containment before power is lost to 1-SI-MOV-1890C.**
	SRO	b) Check RCS pressure – INCREASING
	RO	*RO will identify RCS increasing.*
		c) Place LHSI pumps in PTL Places LHSI pumps in PTL.
	RO	d) Close LHSI pump suctions from RWST • 1-SI-MOV-1862A • 1-SI-MOV-1862B
	SRO	**Closes 1-SI-MOV-1862A / B**. e) GO TO 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT **SRO Transitions to 1-E-1**.
		END EVENT 7 END of Scenario 3

Op-Test No.: Surry 2016-1 Scenario No.: 3 Event No.: 7 Page 50 of 81

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

Time	Position	Applicant's Action or Behavior
	BOP	**ATTACHMENT 1 OF E-0** 1. CHECK FW ISOLATION: • Feed pump discharge MOVs – CLOSED • 1-FW-MOV-150A • 1-FW-MOV-150B • MFW pumps – TRIPPED • Feed REG valves – CLOSED • SG FW bypass flow valves – DEMAND AT ZERO • SG blowdown TVs – CLOSED

	BOP	2. CHECK CTMT ISOLATION PHASE I:
		• Phase I TVs – CLOSED
		Identifies 1-BD-TV-100A, 1- BD-TV-100B OPEN, Closes valves
		• 1-CH-MOV-1381 – CLOSED
		• 1-SV-TV-102A – CLOSED
		• PAM isolation valves – CLOSED
		  • 1-DA-TV-103A
		  • 1-DA-TV-103B

	BOP	3. CHECK AFW PUMPS RUNNING: a) MD AFW pumps – RUNNING (Time Delayed) b) TD AFW pump - RUNNING IF NECESSARY

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Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

	BOP	**Attachment 1 of E-0** 4. CHECK SI PUMPS RUNNING: • CHG pumps – RUNNING • LHSI pumps – RUNNING **Manually Starts 1-SI-P-1A.**
	BOP	5. CHECK CHG PUMP AUXILIARIES: • CHG pump CC pump – RUNNING • CHG pump SW pump - RUNNING
	BOP	6. CHECK INTAKE CANAL: • Level - GREATER THAN 24 FT • Level - BEING MAINTAINED BY CIRC WATER PUMPS
	BOP	7. CHECK IF MAIN STEAMLINES SHOULD BE ISOLATED: a) Check if ANY of the following annunciators - HAVE BEEN LIT • E-F-10 (High Steam Flow SI) • B-C-4 (Hi Hi CLS Train A) • B-C-5 (Hi Hi CLS Train B) Identifies annunciators not lit and goes to step 8.

Op-Test No.: Surry 2016-1 Scenario No.: 3 Event No.: 7 Page 52 of 81

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

	BOP	*8 CHECK IF CS REQUIRED: a) CTMT pressure – HAS EXCEEDED 23 PSIA 8, a) RNO Do the following: 1) IF CTMT pressure has exceeded 17.7 psia, THEN check or align the following valves: Identifies CTMT pressure remains at normal pressure 2) GO TO Step 10.
	BOP	**Attachment 1 of E-0** *10. BLOCK LOW PRZR PRESS SI SIGNAL: a) Check PRZR pressure – LESS THAN 2000 psig b) Turn both LO PRZR PRESS & STM HDR/LINE ΔP switches to block c) Verify Permissive Status light C-2 - LIT BOP may block the low pressurizer pressure SI signal depending on current RCS pressure.
	BOP	*11. BLOCK LOW TAVE SI SIGNAL: Step may not be performed at this time (if Tave is greater than 543°F). a) Check RCS Tave - LESS THAN 543°F b) Turn both HI STM FLOW & LO TAVG OR LP switches to block c) Verify Permissive Status light F-1 - LIT

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

	BOP	**NOTE:** • CHG pumps should be run in the following order of priority: C, B, A. • Subsequent SI signals may be reset by re-performing Step 12. 12. CHECK SI FLOW: a) HHSI to cold legs - FLOW INDICATED • 1-SI-FI-1961 (NQ) • 1-SI-FI-1962 (NQ) • 1-SI-FI-1963 (NQ) • 1-SI-FI-1943 or 1-SI-FI-1943A b) Check CHG pumps - THREE RUNNING c) Reset SI. d) Stop one CHG pump and out in AUTO
		Attachment 1 of E-0 e) RCS pressure - LESS THAN 185 PSIG RNO: e) IF two LHSI pumps are running, THEN do the following: 1) Verify reset or reset SI. 2) Stop one LHSI pump and put in AUTO.
	BOP	13. CHECK TOTAL AFW FLOW - GREATER THAN 350 GPM [450 GPM]
	BOP	14. CHECK AFW MOVs - OPEN BOP will identify that all AFW MOVS are not open and will read the RNO portion of this step and manually align valves as necessary.
	BOP	15. INITIATE SI VALVE ALIGNMENT IAW ATTACHMENT 2 See attached copy of Attachment 2. (following this attachment) Depending on timing, this attachment should be completed by The RO.

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Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

	BOP	16. INITIATE VENTILATION, AC POWER, AND SFP STATUS CHECKS IAW ATTACHMENT 3 *Attachment 3 follows Attachment 2.* *Unit 2 Operator will state that Unit 2 is at 100% power (if asked)* *Unit 2 will also accept responsibility to complete Attachment 3 if it is given to Unit 2 at the point where differential pressure indications are requested.*

Op-Test No.: Surry 2016-1 Scenario No.: 3 Event No.: 7

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

Time	Position	Applicant's Action or Behavior
	RO	ATTACHMENT 2 of 1-E-0 **NOTE:** Components previously aligned by SI termination steps, must not be realigned by this Attachment. **CT-2: Restore HHSI flow from the RWST to the core prior to air-binding of the HHSI pumps.**
	RO	ATTACHMENT 2 of 1-E-0 1. Check opened or open CHG pump suction from RWST MOVs. • **1-CH-MOV-1115B** • **1-CH-MOV-1115D**
	RO	ATTACHMENT 2 of 1-E-0 2. Check closed or close CHG pump suction from VCT MOVs. • **1-CH-MOV-1115C** • **1-CH-MOV-1115E**
	RO	ATTACHMENT 2 of 1-E-0 3. Check running or start at least two CHG pumps. (listed in preferred order) • 1-CH-P-1C • 1-CH-P-1B • 1-CH-P-1A
	RO	ATTACHMENT 2 of 1-E-0 4. Check opened or open HHSI to cold legs MOVs. • **1-SI-MOV-1867C** • **1-SI-MOV-1867D**
	RO	ATTACHMENT 2 of 1-E-0 5 Check closed or close CHG line isolation MOVs. • 1-CH-MOV-1289A • 1-CH-MOV-1289B

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

	RO	ATTACHMENT 2 of 1-E-0 6. Check closed or close Letdown orifice isolation valves. • 1-CH-HCV-1200A • 1-CH-HCV-1200B • 1-CH-HCV-1200C
	RO	ATTACHMENT 2 of 1-E-0 7. Check opened or open LHSI suction from RWST MOVs. • 1-SI-MOV-1862A • 1-SI-MOV-1862B
	RO	ATTACHMENT 2 of 1-E-0 8. Check opened or open LHSI to cold legs MOVs. • 1-SI-MOV-1864A • 1-SI-MOV-1864B
	RO	ATTACHMENT 2 of 1-E-0 9. Check running or start at least one LHSI pump. • 1-SI-P-1A • 1-SI-P-1B
	RO	ATTACHMENT 2 of 1-E-0 10. Check High Head SI flow to cold legs indicated. • 1-SI-FI-1961 • 1-SI-FI-1962 • 1-SI-FI-1963 • 1-SI-FI-1943 or 1-SI-FI-1943A
	RO	ATTACHMENT 2 of 1-E-0 11. IF flow not indicated, THEN manually start pumps and align valves. IF flow NOT established, THEN consult with Shift Supervision to establish another high pressure injection flow path while continuing with this procedure. • Alternate SI to Cold legs • Hot leg injection

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 71	AUXILIARY VENTILATION, AC POWER, AND SFP STATUS CHECKS	PAGE 1 of 6

1. ___ Check or place REFUEL SFTY MODE switches in NORMAL.

2. ___ Check ventilation alignment IAW Tables 1 and 2.

TABLE 1
UNIT #1 VENTILATION PANEL

	MARK NUMBER	EQUIPMENT STATUS
☐	1-VS-F-4A & B	OFF
☐	1-VS-HV-1A & B	OFF
☐	1-VS-F-8A & B	OFF
☐	1-VS-F-9A & B	GREEN
☐	1-VS-F-59	GREEN
☐	1-VS-F-6	OFF
☐	1-VS-F-39	GREEN
☐	1-VS-F-7A & B	GREEN
☐	1-VS-HV-5	GREEN
☐	1-VS-F-56A & B	GREEN
☐	1-VS-F-40A & B	GREEN
☐	1-VS-HV-4	OFF
☐	2-VS-F-40A or B	RED
☐	2-VS-HV-4	OFF

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

NUMBER 1-E-0	ATTACHMENT TITLE AUXILIARY VENTILATION, AC POWER, AND SFP STATUS CHECKS	ATTACHMENT 3
REVISION 71		PAGE 2 of 6

TABLE 2
VNTX PANEL

MARK NUMBER	EXPECTED EQUIPMENT STATUS	RESPONSE NOT OBTAINED
☐ a. AOD-VS-107A & B AOD-VS-108	RED GREEN	☐ a. Place AUX BLDG CENTRAL AREA MODE switch to FILTER.
☐ b. MOD-VS-100A & B AOD-VS-106	RED GREEN	☐ b. • Place MOD-VS-100A to FILTER. • Place MOD-VS-100B to FILTER.
☐ c. MOD-VS-200A & B AOD-VS-206	GREEN RED	☐ c. • Place MOD-VS-200A to UNFILTER. • Place MOD-VS-200B to UNFILTER.
☐ d. AOD-VS-103A & B AOD-VS-104	GREEN GREEN	☐ d. • Place AOD-VS-103A in UNFILTER. • Place AOD-VS-103B in UNFILTER. • Place AOD-VS-104 in FILTER.
☐ e. AOD-VS-101A & B AOD-VS-102	GREEN GREEN	☐ e. Place AOD-VS-101A and 101B in UNFILTER.
☐ f. AOD-VS-111A & B	GREEN	☐ f. Place COMBINE CONTAINMENT EXHAUST in ISOLATE.
☐ g. AOD-VS-110	GREEN	☐ g. Place AOD-VS-109A and 109B in FILTER.
☐ h. AOD-VS-112A & B	GREEN	☐ h. • Place AOD-VS-112A in CLOSE. • Place AOD-VS-112B in CLOSE.
☐ i. MOD-VS-58A & B 1-VS-F-58A & B	RED RED	☐ i. Start 1-VS-F-58A and 1-VS-F-58B.

3. ___ Check filtered exhaust flow: (as read on FI-VS-117A and FI-VS-117B)

- ☐ • Total flow - GREATER THAN 32400 cfm

AND

- ☐ • Flow through each filter bank - LESS THAN 39600 cfm

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 71	AUXILIARY VENTILATION, AC POWER, AND SFP STATUS CHECKS	PAGE 3 of 6

4. ___ Check all Station Service Buses - ENERGIZED. IF NOT, THEN initiate 1-AP-10.07, LOSS OF UNIT 1 POWER.

5. ___ Check annunciator VSP-J2 - LIT.

6. ___ Check Unit 1 RSST LTC time delay bypass light - LIT.

7. ___ Check stopped or stop 1-VS-AC-4.

8. ___ Place 1-VS-43-VS103X, MCR ISOLATION switch to the OFF position.

9. ___ Check closed or close MCR isolation dampers.

   - ☐ • 1-VS-MOD-103A
   - ☐ • 1-VS-MOD-103B
   - ☐ • 1-VS-MOD-103C
   - ☐ • 1-VS-MOD-103D

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 71	AUXILIARY VENTILATION, AC POWER, AND SFP STATUS CHECKS	PAGE 4 of 6

* *

CAUTION:
- Only one Emergency Supply Fan must be started in the following step.
- Chilled Water flow to the in-service Unit 1 MCR AHU must be throttled to at least 15 gpm when the Emergency Supply fan is started.
- Chilled Water flow to the in-service Unit 2 MCR AHU must be throttled to at least 25 gpm when the Emergency Supply fan is started.
- An Emergency Supply Fan must not be started if the filter is wet.

* *

10. Immediately start ONE Emergency Supply Fan IAW the following: (1-VS-F-41 or 2-VS-F-41 preferred)

a. IF 1-VS-F-41, CONT RM EMERG SUP FAN, will be used, THEN perform the following substeps.

___ 1. Open 1-VS-MOD-104A, CONT RM EMERG SUP MOD.

___ 2. Start 1-VS-F-41.

b. IF 2-VS-F-41, CONT RM EMERG SUP FAN, will be used, THEN perform the following substeps.

___ 1. Open 2-VS-MOD-204A, CONT RM EMERG SUP MOD.

___ 2. Start 2-VS-F-41.

c. IF 1-VS-F-42, CONT RM EMERG SUP FAN, will be used, THEN perform the following substeps.

___ 1. Open 1-VS-MOD-104B, CONT RM EMERG SUP MOD.

___ 2. Start 1-VS-F-42.

d. IF 2-VS-F-42, CONT RM EMERG SUP FAN, will be used, THEN perform the following substeps.

___ 1. Open 2-VS-MOD-204B, CONT RM EMERG SUP MOD.

___ 2. Start 2-VS-F-42.

e. ___ Adjust Chilled Water flow to MCR AHUs IAW Step 10 Caution.

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 71	AUXILIARY VENTILATION, AC POWER, AND SFP STATUS CHECKS	PAGE 5 of 6

11. ___ Check readings on the following Differential Pressure Indicators - POSITIVE PRESSURE INDICATED.

- ☐ • PDI-VS-100, D.P.-U1CR/U1TB (Unit 2 Turbine Ventilation Panel)
- ☐ • PDI-VS-101, D.P.-U1RR/U1TB (Unit 2 Turbine Ventilation Panel)
- ☐ • PDI-VS-200, D.P.-U2CR/U2TB (Unit 2 Turbine Ventilation Panel)
- ☐ • PDI-VS-201, D.P.-U2RR/U2TB (Unit 2 Turbine Ventilation Panel)
- ☐ • 1-VS-PDI-118 (Unit 1 Computer Room)
- ☐ • 1-VS-PDI-116 (Near Unit 1 Semi-Vital Bus)
- ☐ • 2-VS-PDI-215 (Unit 2 AC Room)
- ☐ • 2-VS-PDI-206 (Near Unit 2 Semi-Vital Bus)

12. ___ IF any reading NOT positive, THEN initiate Attachment 6 to secure MCR boundary fans.

13. ___ Check initiated or initiate 0-AP-50.00, OPPOSITE UNIT EMERGENCY.

14. ___ Check the following MCR and ESGR air conditioning equipment operating. IF NOT, THEN start equipment within 1 hour IAW the appropriate subsection of 0-OP-VS-006, CONTROL ROOM AND RELAY ROOM VENTILATION SYSTEM.

- ☐ • One Control Room chiller
- ☐ • One Unit 1 Control Room AHU
- ☐ • One Unit 2 Control Room AHU
- ☐ • One Unit 1 ESGR AHU
- ☐ • One Unit 2 ESGR AHU

15. ___ IF both of the following conditions exist, THEN check that Load Shed is activated.

- ☐ • Unit 2 - SUPPLIED BY RSST
- ☐ • Unit 2 RCPs - RUNNING

16. ___ IF Load Shed is required and not activated, THEN initiate 0-AP-10.10, LOSS OF AUTO LOAD SHED.

Event Description: EOP Attachments 1-3.

Cue: Pre-event failures.

NUMBER 1-E-0	ATTACHMENT TITLE	ATTACHMENT 3
REVISION 71	AUXILIARY VENTILATION, AC POWER, AND SFP STATUS CHECKS	PAGE 6 of 6

NOTE:
- SFP checks should be initiated WITHIN ONE TO TWO HOURS of EOP entry.
- Loss of power may render SFP indications and alarms non-functional and require local checks. Power supplies are as follows:
  - TI-FC-103, Unit 1 Semi-Vital Bus
  - TI-FC-203, Unit 2 Semi-Vital Bus
  - 1-FC-LIS-104, Panel 1ABDA1
- Loss of AC Power to the SFP level indicator is indicated if both low and high level alarms are in simultaneously. (0-VSP-C4 and 0-VSP-D4)
- 1-DRP-003, CURVE BOOK, provides a graph for SFP time to 200°F if loss of SFP cooling occurs.

17. ___ Initiate monitoring SFP parameters:

- ☐ • SFP level - Greater than Cooling Pump suction AND Stable
- ☐ • SFP temperature - Stable or Lowering
- ☐ • SFP Cooling Pumps - Either Running
- ☐ • Component Cooling - Normal
- ☐ • SFP Radiation - Normal

18. ___ Continue to monitor parameters every one to two hours or until authorized to terminate monitoring by the Station Emergency Manager and/or the Shift Manager.

19. ___ Notify the Station Emergency Manager and/or the Shift Manager of the status and trend of SFP parameters.

20. ___ IF any abnormality or adverse trend is identified, THEN initiate 0-AP-22.02, MALFUNCTION OF SPENT FUEL PIT SYSTEMS.

FOLDOUT PAGES FOR REFERENCED PROCEDURES

NUMBER 1-E-0	CONTINUOUS ACTIONS PAGE	REVISION 71

1. RCP TRIP CRITERIA
   Trip all RCPs if BOTH conditions listed below occur:
   a. Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS
   b. RCS Subcooling - LESS THAN 30°F [85°F]
2. MINIFLOW RECIRC CRITERIA
   a. CLOSED - When RCS pressure is less than 1275 psig [1475 psig] AND RCP Trip Criteria are met (RCPs OFF).
   b. OPEN - When RCS pressure is greater than 2000 psig.
3. ADVERSE CONTAINMENT CRITERIA
   Use Adverse Containment setpoints if EITHER condition listed below occurs:
   - Containment Pressure - GREATER THAN 20 PSIA
   - Containment Radiation - GREATER THAN 1.0E5 R/HR
4. COLD LEG RECIRCULATION SWITCHOVER CRITERIA
   GO TO 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, if RWST level lowers to less than 20%.

---

1. AMSAC RESET CRITERIA
   AMSAC may be manually reset when level in all three SGs is greater than 13% or six minutes have elapsed since the Reactor trip. When AMSAC is reset, AMSAC ARMED annunciator H-D-1 should clear and affected components may be realigned as needed.
2. TD AFW PUMP SHUTDOWN CRITERIA
   The TD AFW pump may be secured when SG NR level is greater than 22% in at least 2 SGs, AMSAC is reset, and no auto-start signal exists. To secure the pump, the pump SOV control switches must be taken to OPEN-RESET and then to CLOSE.
3. MANUAL SI ALIGNMENT
   If SI fails to automatically align, Attachment 2 may be used for guidance on manual SI valve alignment.
4. * TRANSIENT AFW FLOW CONTROL (IF SI in progress)
   Attachment 7 may be used for guidance on transient AFW flow control.
5. * FAULTED SG ISOLATION AND AFW FLOW CONTROL (IF SI in progress)
   Attachment 8 may be used for guidance on faulted SG(s) isolation and AFW flow control.
6. * RUPTURED SG ISOLATION AND AFW FLOW CONTROL (IF SI in progress)
   Attachment 9 may be used for guidance on ruptured SG(s) isolation and AFW flow control.
7. * LOSS OF RCP SUPPORT CONDITIONS
   Trip RCPs if a loss of a support condition occurs. (for example, loss of CC)

* Preemptive Actions

FOLDOUT PAGES FOR REFERENCED PROCEDURES

NUMBER 1-E-0	CONTINUOUS ACTION STEPS	REVISION 71

1. Check RCS Average Temperature - STABLE AT OR TRENDING TO 547°F. (E-0, Step 6)
2. Monitor RCP Trip and Miniflow Recirc Criteria. (E-0, Step 8)
3. Check SG Narrow Range Level - ANY SG GREATER THAN 12%. (Control feed flow to maintain Narrow Range Level between 22% and 50%) (E-0, Step 25)
4. Monitor LHSI pumps and secure as necessary. (E-0, Step 30)

NOTE: Subsequent SI signals may be reset by reperforming Step 12 of Attachment 1.

5. Monitor CTMT pressure and check CLS initiation as necessary. (Attachment 1, Step 8)
6. Monitor RWST level and check RS initiation as necessary. (Attachment 1, Step 9)
7. Block Low PRZR Pressure SI signal when less than 2000 psig. (Attachment 1, Step 10)
8. Block Low Tave SI signal when less than 543°F. (Attachment 1, Step 11)

FOLDOUT PAGES FOR REFERENCED PROCEDURES

CONTINUOUS ACTIONS PAGE FOR 1-E-1

1. RCP TRIP CRITERIA

   Trip all RCPs if BOTH conditions listed below occur:

   a. Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS

   b. RCS Subcooling - LESS THAN 30°F [85°F]

2. SI REINITIATION CRITERIA

   Following SI termination or SI flow reduction, manually start SI pumps and align valves as necessary if EITHER condition listed below occurs:

   - RCS subcooling based on CETCs - LESS THAN 30°F [85°F]
   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 22% [50%]

3. MINIFLOW RECIRC CRITERIA

   a. CLOSED - When RCS pressure is less than 1275 psig [1475 psig] AND RCP Trip Criteria are met (RCPs OFF).

   b. OPEN - When RCS pressure is greater than 2000 psig.

4. ADVERSE CONTAINMENT CRITERIA

   Use Adverse Containment setpoints if EITHER condition listed below occurs:

   - Containment Pressure - GREATER THAN 20 PSIA
   - Containment Radiation - GREATER THAN 1.0E5 R/HR

5. SECONDARY INTEGRITY CRITERIA

   Manually start SI pumps as necessary and GO TO 1-E-2, FAULTED STEAM GENERATOR ISOLATION, if any SG pressure is lowering in an uncontrolled manner or has completely depressurized, and has not been isolated.

6. E-3, TRANSITION CRITERIA

   Manually start SI pumps as necessary and GO TO 1-E-3, STEAM GENERATOR TUBE RUPTURE, if any SG level rises in an uncontrolled manner or any SG has abnormal radiation.

7. COLD LEG RECIRCULATION SWITCHOVER CRITERIA

   GO TO 1-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, if RWST level lowers to less than 20%.

8. AFW SUPPLY SWITCHOVER CRITERIA (Refer to Attachment 5)

   Transfer to one of the following alternate AFW water supplies if ECST level lowers to less than 20%.

   a. 1-CN-TK-2, using 1-CN-150.

   b. 1-CN-TK-3, using AFW Booster Pumps.

   c. AFW Crosstie.

   d. Firemain.

9. RCP SEAL INJECTION CRITERIA

   Seal Injection flow should be maintained to all RCPs.

10. LOSS OF RCP SUPPORT CONDITIONS

    Trip RCPs if a loss of a support condition occurs. (for example, loss of CC)

SIMULATOR OPERATOR'S GUIDE

Simulator Setup

Initial Conditions:

Recall IC -375 **and implement TRIGGER #30 to activate all passive malfunctions and verify Trigger #30 implemented.**

Open the monitor window and add the following points to it:

- asp_ao_off

Enter the following MALFUNCTIONS:

Malfunction	Delay	Ramp	Trigger	Value	Final	Trigger Type (Auto or Manual)
BD01, DISABLE BDTV100A AUTO CLOSURE			30		TRUE	
BD02, DISABLE BDTV100B AUTO CLOSURE			30		TRUE	
VS0101, PRESSURE SWITCH VS-P-127A STUCK AS IS			30		TRUE	
SI34, DISABLE SI-MOV-1867C AUTO OPEN			30		TRUE	
SI35, DISABLE SI-MOV-1867D AUTO OPEN			30		TRUE	
CA03, DISABLE IA-TV-101A AUTO CLOSURE			30		TRUE	
CA06, DISABLE IA-TV-101B AUTO CLOSURE			30		TRUE	
CH50, DISABLE CH-MOV-1115B AUTO OPEN			30		TRUE	
CH51, DISABLE CH-MOV-1115C AUTO CLOSED			30		TRUE	
CH52, DISABLE CH-MOV-1115D AUTO OPEN			30		TRUE	
CH53, DISABLE CH-MOV-1115E AUTO CLOSED			30		TRUE	
FP0301, FPS FACP07 ALARM HORN FAILURE			30		TRUE	
FP0302, FPS PC SPEAKER FAILURE			30		TRUE	
RM0201, PROCESS RAD MONITOR RI-RMS-160 FAILURE	10	60	1		1	MAN
RC4802, PRZR PRESS CONT XMTR FAILURE (445)	10		3		1	MAN
FW1303, A S/G NAR RNG LVL XMTR LT-476 CH-3 FAILS	10	30	5		1	MAN

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SW0401, OVERLOAD TRIP OF PUMP SW-P-10A	10		7		TRUE	MAN
SW0402, DISABLE SW-P-10B AUTO START			7		TRUE	MAN
SI1601, FAIL CHECK VALVES SI-79			11		TRUE	MAN
SI1604, FAIL CHECK VALVES SI-241			11		TRUE	MAN
SI1502, SI COLD LEG HDR LEAK UPSTRM MOV-SI-1890C	10	60	11		20	MAN
CH0901, BORON XFER PP CH-P-2A THRML OVRLOAD TRIP	5:00		11		TRUE	MAN

Enter the following Remote Functions:

Override	Set Condition	Trigger
SIMOV890A, BKR SI-MOV-1890A	CLOSED	15
SIMOV890B, BKR SI-MOV-1890B	CLOSED	15
SIMOV890C, BKR SI-MOV-1890C	CLOSED	15

TRIGGER	TYPE	DESCRIPTION
1	MAN	Initiates RM0201, RI-160 failure
3	MAN	Initiates RC4802, Pressurizer press transmitter 455 failure
5	MAN	Initiates FW1303, A S/G LT-476 failure
7	MAN	Initiates SW0401, SW1202; SW-P-10A failure with auto start failure
11	MAN	Initiates LOCA outside Ctmt. with failure of Boron Xfer pump
30	PRE SCENARIO	Initiates failures to; CH-MOV-1115B-E, SI-MOV-1867C/D, BDTV100A/B, 1-VS-F-58A, IA-TV-101A/B, and Fire protection.

SIMULATOR OPERATOR'S GUIDE

Verify the following control room setup:

- ❑ Place the simulator in RUN and verify normal 100% power operation indications.
- ❑ Reset the ROD CONTROL SYSTEM
- ❑ Verify Controlling channels selected to **IV**
- ❑ Verify Red Magnets on the following components:

1-CH-P-1B, 'B' Chg pump

- ❑ Verify 1-RM-RI-112 aligned to A/B SG and 1-RM-RI-113 aligned to C SG (magnets).
- ❑ Verify Ovation System operating.
- ❑ Reset ICCMs.
- ❑ Verify Component Switch Flags.
- ❑ Verify Brass Caps properly placed.
- ❑ Verify SG PORVs set for 1035 psig.
- ❑ Verify Rod Control Group Step Counters indicate properly.
- ❑ Verify Ovation CRT display.
- ❑ Advance Charts
- ❑ Verify Turbine Thumb Wheel Settings @120 rpm/min and Position 6
- ❑ Verify Containment Instrument Air Compressors are on Inside Suction (all RMs reset)
- ❑ Verify all ARPs have been cleaned

❑ 1B-F3	❑ 1C-B8	❑ 1C-D8	❑ 1C-F7
❑ 1C-F8	❑ 1D-C5	❑ 1D-E5	❑ 1D-G5
❑ 1D-H4	❑ 1F-C7	❑ 1F-C10	❑ 1F-D7
❑ 1H-A5	❑ 1H-E5	❑ 1H-G5	❑ RMA-D7
❑ RMA-Q8	❑ RMA-R8	❑	❑

- ❑ Verify CLEAN copies of the following procedures are in place:

❑ 1-OP-EH-001	❑ 0-AP-53.00	❑ 1-AP-31.00	❑ 0-AP-12.00

SIMULATOR OPERATOR'S GUIDE

❑ 0-AP-23.00	❑ 1-AP-16.00	❑ 1-E-0	❑ 1-E-1
❑ 1-ECA-1.2	❑	❑	❑

❑ **Verify Reactivity Placard is current.**

❑ Verify ALL PINK MAGNETS are accounted for.

❑ Reset Blender Integrators for Boric Acid to 100 and PG 1000.

SIMULATOR OPERATOR'S GUIDE

Brief

This simulator performance scenario is performed in the EVALUATION MODE. You should not direct questions to the evaluators. Otherwise, you should perform as if you were in the MCR.

Your ability to maintain a log is not being graded, but maintaining a rough log is recommended to help during briefs.

If you need to communicate with the Unit 2 operator, verbally state, "Unit 2" and an instructor will locate to the Unit 2 area and respond to you as quickly as possible.

In the unlikely event that the simulator fails such that illogical indications result, the session will be terminated. In other words, respond to what you see. If there is a problem with the simulation, the session will be terminated or adjusted as appropriate based on the specific problem.

Assign operating positions.

Ask for and answer questions.

SIMULATOR OPERATOR'S GUIDE

Conduct shift turnover:

The initial conditions have Unit 1 at 100% power with RCS boron concentration of 760 ppm.

Unit conditions have been stable at approximately 100% power since the last refueling outage.

All systems and crossties are operable with the following exception:

- Containment Smoke and heat detectors are non-functional due to local fire panel failure. TRM Section 3.3.1, Fire Detection Instrumentation, Condition B, Smoke Detectors, and Condition C, Heat Detectors is in effect. Containment air temperatures monitored once/hour, and restore to Functional status in 14 days.
- 'C' Charging pump running on Alt, 'A' Charging pump is in Auto, 'B' Charging pump is tagged out for breaker PMs.
- Controlling channels have been shifted to CH IV in reparation for Channel III testing

Unit #2 is at 100% power with all systems and crossties operable.

The last shift performed two 30 gallon dilutions, followed by manual makeups.

Shift orders are to maintain 100% power on Unit 1, and swap EH pumps IAW 1-OP-EH-001, Section 5.6.

When the team has accepted the shift, proceed to the Session Conduct Section.

SIMULATOR OPERATOR'S GUIDE

Session Conduct:

- Ensure conditions in Simulator Set-up are established.
- **Ensure Trigger 30 is active prior to team entering the simulator.**
- Verify Exam Security has been established and ASP_AO_OFF = True.

EVENT 1 Swap EH Pumps, section 5.6, 1-OP-EH-001.

BOOTH:
30 minutes prior to the beginning of the scenario, provide the team with a copy of 1-OP-EH-001, ELECTRO-HYDRAULIC FLUID SYSTEM (EHC). The team will pre-brief the OP prior to entering the simulator.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the completion of the evolution.

Field Operator: (*2 minute delay from request to answer*)

- **If Contacted**, EHC skid is clear of personnel.
- **If Contacted,** MP-2 is running normally.
- **If Contacted,** MP-1 is normal after stop.

Role play as other individuals as needed.

SIMULATOR OPERATOR'S GUIDE

EVENT 2 **Particulate and gas RM fail, No Auto Actions.**

When cued by examiner, implement **Trigger #1**.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the failure of the automatic actions on RM-160 High alarm.
- **If contacted,** will take responsibility for writing the WR and CR.

Unit 2 Operator:

- **When** radiation alarms sound on the radiation alarm panel, silence the alarms and report the alarm to the Unit 1 SRO.
- **If directed,** perform the associated RM ARP without leaving the confines of the Unit 2 control area. If actions or verifications are required on the Unit 1 side, inform the Unit 1 SRO of the need for an operator to complete the ARP.

Field Operators:

- **If directed,** field operators will perform local manipulations as required.

Role play as other individuals as needed.

SIMULATOR OPERATOR'S GUIDE

EVENT 3 1-RC-PT-1145 Fails High, 1-AP-31.00/0-AP-53.00

When cued by examiner, implement **Trigger #3**.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the failure of 1-RC-PT-1445. The individual(s) contacted will also acknowledge any TS LCOs and entry into AP-53.00/AP-31.00.
- **If contacted,** will take responsibility for writing the CR.

STA:

- **If contacted**, will acknowledge the failure of 1-RC-PT-1445. The individual(s) contacted will also acknowledge (but not confirm/deny) any TS LCOs.
- **If asked,** the STA will review VPAP-2802 and TRM Section 3.3 and report that he has completed his review and this failure does not impact these documents.
- **If contacted,** will take responsibility for writing the CR.
- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.

Field Operators:

No feedback required for this failure.

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge instrumentation failure and commence investigations.

Unit 2 Operator:

If Contacted, acknowledge the failure of 1-RC-PT-1445.

Role play as other individuals as needed.

SIMULATOR OPERATOR'S GUIDE

EVENT 4 A SG NR Level Channel III fails HIGH.

When cued by examiner, implement **Trigger #5**.

Booth Note: **Critical Task CT-1:**
Should the BOP fail to take action to control A SG NR level, an automatic reactor trip will occur. This would be considered failure criteria.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the failure of 1-FW-LT-1476. The individual(s) contacted will also acknowledge any TS LCOs and entry into AP-53.00.
- **If contacted,** will take responsibility for writing the CR.
- **If contacted,** will recommend waiting for I&C to be ready to perform 0-OP-RP-001

STA:

- **If contacted**, will acknowledge the failure of 1-FW-LT-1476. The individual(s) contacted will also acknowledge (but not confirm/deny) any TS LCOs.
- **If asked,** the STA will report that 1-FW-LT-1476 is a Reg. Guide 1.97 component. The STA will also report that upon review of CEP 99-0029 that only one channel of SG level indication is required per SG, so no actions for Reg. Guide 1.97 are required.
- **If asked,** the STA will review VPAP-2802 and TRM Section 3.3 and report that he has completed his review and this failure does not impact these documents.
- **If contacted,** will take responsibility for writing the CR.
- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal. He will also state that containment conditions and the electrical conditions are as you see them.

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Maintenance:

- **If contacted**, will acknowledge instrumentation failure and commence investigations and/or repair efforts.

Field Operators:

- **If contacted**, field operators will report no issues at the MFRVs.

Unit 2:

- If team directs performance of 1-OPT-RX-007, Shift Average Power Calculation, state that you will have the 4th RO perform the procedure.

Role play as other individuals as needed.

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EVENT 5 Trip of running 1-SW-P-10A with failure of 1-SW-P-10B auto start.

When cued by examiner, implement **Trigger #7**.

Operations Supervisor/Management:

- **If contacted**, acknowledge trip of 1-SW-P-10A and the failure of 1-SW-P-10B to auto start.
- **If asked,** will notify the OMOC.
- **When notified,** acknowledge Tech Spec requirements for the 1-SW-P-10A tripping, but do not imply concurrence with SRO Tech Spec determination.

STA:

- **If contacted**, acknowledge the trip of 1-SW-P-10A, acknowledge the Tech Spec requirements for the pump trip, but do not imply concurrence with the Tech Spec determination by the SRO.
- **If the team has a transient brief:** The STA will have no input.

Field Operators: (Wait three minutes between direction to perform local action/status check and report.)

- **If contacted**, 1-SW-P-10A will have no local indications for cause of the trip.
- **When contacted,** as Service Building Operator, MCC 1H1-1 1D (1-SW-P-10A) thermal overload will be found tripped.
- **When contacted:** 1-SW-P-10B post start checks are normal.

Maintenance/ Work Week Coordinator:

- **If contacted**, will notify Electricians to troubleshoot the cause of 1-SW-10A (MCC 1H1-1 1D breaker) overload trip.

Role play as other individuals as needed.

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EVENTS 6 **Ramp Unit to HSD.**

Operations Supervisor/Management:

- **If contacted**, will acknowledge the ramp to HSD.
- **If contacted,** will take responsibility for writing the CR.
- **If asked** for a recommended ramp rate, ask what the Unit Supervisor recommends. When authorized by the NRC, the Shift Manager will direct a 1%/minute ramp rate.

Unit 2 Operator:

If notified, acknowledge the failure and impending ramp of Unit 1.

STA:

- **If contacted**, will acknowledge the Reactivity Plan reported by the RO.
- **If contacted,** will take responsibility for writing the WR and CR.
- **If the team has a transient brief:** The STA will, "nothing to add".
- If contacted, STA review of VPAP-2802 complete, reviewed with Shift Manager, no notifications required.

Maintenance/ Work Week Coordinator:

- **If contacted**, will acknowledge the requirements to reduce reactor power.

Chemistry

- **If contacted,** acknowledge the ramp.

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Field Operators:

- **If contacted**, as the Turbine Building Operator to walkdown the Turbine during the ramp, acknowledge the direction.
- **If contacted,** as the polishing building operator, acknowledge the direction to monitor polisher DP.

Role play as other individuals as needed.

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EVENT 8/9 LOCA Outside Containment, ECA-1.2

When cued by examiner, implement **Trigger #11**.

This will cause a LOCA to occur upstream of 1-SI-MOV-1890C. The break size will result in loss of power to 1-SI-MOV-1890C in approximately 30 minutes. **Monitor VD5** (Safeguards Pit) until 1-SI-MOV-1890C is isolated.

Five minutes after the LOCA starts the Boron Xfer pump 1-CH-P-2A will thermal overload. The HHSI pumps will be taking a suction on the VCT due to failures to 1-CH-MOV-1115B-E. **Monitor VCT level** because the HHSI pumps will be drawing down VCT at approximately 4%/minute.

CT-2: Restore HHSI flow from the RWST to the core prior to air-binding of the HHSI pumps.

CT-3: Isolate LOCA outside of Containment before power is lost to 1-SI-MOV-1890C.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the reactor trip failure, completion of FR-S.1, completion of E-0 Immediate Actions, and transition to ES-0.1.
- **If contacted**, will acknowledge the subsequent fault on the previously identified ruptured SG. Will also acknowledge any TS information (time permitting) and information related to radiation monitors alarming.
- **If contacted,** will take responsibility for writing the WR and CR.

Unit 2 Operator:

- **If directed** to respond to radiation alarms on the radiation alarm panel, silence the alarms when and report the alarm to the Unit 1 SRO.
- **If directed** perform the associated RM ARP without leaving the confines of the Unit 2 control area. If actions or verifications are required on the Unit 1 side, inform the Unit 1 SRO of the need for an operator to complete the ARP.
- **When contacted by BOP,** Chilled Water has been throttled IAW the Caution prior to Step 10 of Attachment 3. MCR boundary D/P indicates the same value as Unit 2 ventilation panel D/P gauges. Unit 2 will assume responsibility for Attachment 3 when last page is reached (SFP monitoring).

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Field Operators:

- **If directed,** field operators will remove tags and close breakers for 1-SI-MOV-1890A / B / C. **Insert Trigger 15.** This will result in power restored to the breakers for 1-SI-MOV-1890 A, B, and C in 3 minutes.

Role play as other individuals as needed.

The scenario will end upon entering 1-ECA-3.1 or at the lead examiners discretion.

Scenario #4

Facility: Surry Power Station   Scenario No.: 4   Op-Test No.: 2016-004

Examiners: ____________________   Operators: ____________________

Initial Conditions: Unit 1 at ~5%, BOL, 1415 ppm Boron. Unit 2 at 100% power

- Unit startup in progress, 1-GOP-1.8 and 1-OP-TM-001.

Turnover: The Team will pre-brief placing the generator on-line and ramping up in power prior to entering the Simulator.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R –RO/SRO N - BOP	Place Unit on-line and begin ramp up in power
2	RM0209, +.5 CC07	I- BOP/SRO TS-SRO	CC RM Fail upscale with failure auto close HCV-CC-100
3	PG0101 PG0202	C-RO/SRO	1-PG-P-1A trips on overcurrent, 1-PG-P-1B fails to auto start
4	RC1503, +1	C-RO/SRO TS-SRO	PRZR Spray fails open (AP-53.00) **(CT-1)**
5	EL0801	C-BOP/SRO	Loss of 1G Screenwell Transformer (AP-12.01)
6	EL01 ED0201 ED0602F FW47	M - ALL	Loss of Offsite Power with Failure of EDG #1 and EDG #2 (ECA-0.0) and TDAFW pump auto start failure. **(CT-2, CT-3)**

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Event 1: **Place Unit On-Line and Ramp Up in Power**. (R – RO/SRO, N- BOP)

The Team will pre-brief 1-GOP-1.8, Unit Startup, HSD to Max Allowable Power (step 5.6.13), and 1-OP-TM-001 (step 5.7), Turbine-Generator Startup to 20% - 25% Turbine Power prior to Simulator entry. A reactivity plan will be provided for the Team use during the pre-brief and in the Simulator for the Ramp up in power. The Team will place the Unit on-line and commence a ramp up in power. A surrogate operator will be available to the Team until the FRVs are in AUTO and the Bypass valves are closed; the BOP will then assume SG level control.

Verifiable Action(s):

1) RO – Manipulate rod control and CVCS Blender to control Tave and ΔFlux during the power escalation.
2) BOP – manipulate Generator output breakers.
3) BOP – Manually control SG NR level, following relief of surrogate.

Technical Specifications/ TRM Actions/ Reg. Guide 1.97/ VPAP-2802, Reportability/ Equipment Important to Emergency Response (EP-AA-303).

Event 2: **1-CC-RI-105, CC RM, Fail high without associated auto action**. (I – BOP/SRO, TS – SRO).
After the Team has raised power, stable control of SG NR level with FRVs in auto has been achieved, and the Evaluation Team is ready, the malfunction is initiated. This failure causes 1-CC-RI-105 Alert and High alarms to actuate with the failure of HCV-CC-100, CC Surge TK VNT Isol VV, to auto close. The BOP will respond to the RM alarm and take action IAW with RM Annunciator Response Procedure.

Verifiable Actions(s):

1) BOP – Close HCV-CC-100, CC Surge Tank vent valve.

Technical Specifications:

1) Tech Spec table 3.7-5, Item 1, Component Cooling water radiation monitors close HCV-CC-100, See Specification 3.13.
2) Tech Spec 3.13.C, Whenever the component cooling water radiation monitor is inoperable, the surge tank vent valve shall remain closed.

TRM Actions/ Reg. Guide 1.97/ VPAP-2802, Reportability/ Equipment Important to Emergency Response (EP-AA-303).

None.

Event 3: **Running PG pump trips, backup PG pump fails to start.** (C – RO/SRO)
When the Evaluating Team is ready, the malfunction is implemented. This failure causes the running PG pump to trip with the backup pump failing to auto trip. The RO will diagnose the failure based on alarms and indications received, and start 1-PG-P-1B.

Verifiable Action(s):

1) RO – Start 1-PG-P-1B.

Technical Specifications/ Technical Requirements Manual/ Reg. Guide 1.97/ VPAP-2802, Reportability/ Equipment Important to Emergency Response (EP-AA-303).

None.

Event #4: **1-RC-PCV-1455B, "B" Spray valve fails Open** (C – RO/SRO, TS - SRO)
When the Evaluating Team is ready, the malfunction is initiated. The controller for the "B" Spray Valve, 1-RC-PCV-1455B, fails causing the spray valve to open fully. The RO will diagnose he failure based on alarms and indications received, place the controller in manual and reduce output to close the spray valve to allow pressure to recover. The Team will implement 0-AP-53.00, Loss of Vital Instrumentation/Controls for this failure.

Verifiable Actions:

1) RO – Place the failed controller in manual and reduce output.

Critical Task 1: If the RO fails to take timely action in response to the Spray Valve opening, an automatic reactor trip on RCS Low Pressure will occur; an unanticipated reactor trip should be considered as failure criteria. An Operations representative will be available for consultation as necessary.

Technical Specifications:

1) Tech Spec 3.12.F.2, DNB Parameters, RCS pressure < 2205 psig, Return RCS to >2205 psig within 2 hours or reduce Thermal Power to less than 5% of Rated Power within the next 6 hours.

Technical Requirements Manual/ Reg. Guide 1.97/ VPAP-2802, Reportability/ Equipment Important to Emergency Response (EP-AA-303).

None.

Event 5: **Loss of the 1G Screenwell Bus. (**C – BOP/SRO)

When the Evaluating Team is ready the malfunction is initiated. This failure causes the loss of the 1G screenwell bus which causes the loss of the running Unit 1 CW pumps. The Team will respond to the alarms received and implement 0-AP-12.01, Loss of Intake Canal Level.

Verifiable Action(s):

1) RO – Throttle the Waterbox outlet MOVs to control canal level.
2) BOP – Use PCS interface to restart Unit 1 CW pumps using Attachment 3 of 0-AP-12.01.

Technical Specifications/ Technical Requirements Manual/ Reg. Guide 1.97/ VPAP-2802, Reportability/ Equipment Important to Emergency Response (EP-AA-303).

None.

Event #6: **Loss of Offsite Power, #1/#2 EDGs fail, ECA-0.0.** (M – ALL)

When the Evaluation Team is ready, the malfunction is initiated. These failure cause a Loss of Off-site power, failure of the Air Start System for EDG #1, #2 EDG trip on overspeed on startup, and lead the Team to implement ECA-0.0. The TDAFW pump will also fail to auto start leading to a condition where no AFW is available to the SGs following the reactor trip. The Team is expected to start the TDAFW pump to control SGs level, initiate 0-AP-17.06, AAC Diesel Generator - Emergency Operations, to strip the 1J Emergency Bus and load the AAC on the 1J bus. When the AAC DG is loaded on the 1J bus, ECA-0.0 will send the Team to Step 33, where operating equipment is checked. The Team is expected to transition to ECA-0.1, Loss of All AC Power Recovery Without SI Required.

Verifiable Actions:

1) RO – Close MSTVs on Step 2 of ECA-0.0.
2) BOP – Start TDAFW pump.
3) BOP – Place components supplied by the 1J bus in PTL.
4) BOP – Load the AAC diesel on the 1J bus.

Critical Task:

1) **CT-2:** Establish AFW Flow during SBO; Establish greater than 350 gpm AFW flow before wide range level in any 2 steam generators is less than 7%.
Job Aid 17, Critical Task Development, CT-23, based on Pressurized Water Reactor Owners Group Westinghouse Emergency Response Guideline -Based

Critical Tasks (PWROG-14043-NP) Appendix B Critical Task CT-23. Old CT - ECA-0.0 – B.

2) **CT-3:** Time Critical Operator Actions, E-11, Align AAC Diesel to respective emergency prior to declaration of an ELAP.

Job Aid 17, Critical Task Development, SPS-CT-1 Re-energize an Emergency Bus on the AAC DG within 10 minutes.

Critical Task start time **begins** when crew enters ECA-0.0, and Critical Task stop time **ends** when AAC DG is loaded on the 1J bus.

The Scenario is terminated on Evaluation Team cue after power is restored to the 1J bus.

Initial Conditions: Unit 1 at ~5% power with a startup in progress; BOL. Unit 2 is operating at 100% power with all system and cross-ties operable.
Pre-load malfunctions: (Trigger 30's) ○ FW47, Disable FW-P-2 Auto Start ○ CC07, Disable CC-HCV-100 Auto closure
Equipment Status/Procedures/Alignments/Data Sheets/etc.: ○ 1-GOP-1.8, Unit Startup, 2% Reactor Power to Max Allowable Power. ○ 1-OP-TM-001, Turbine-Generator Startup to 20% - 25% Turbine Power. ○ Ramp Plan for Power Ascension.
Turnover: Unit 1 at ~5% reactor Power, with a startup in progress. The Team will pre-brief 1-GOP-1.8, Unit Startup, 2% Reactor Power to Max Allowable Power, and 1-OP-TM-001, Turbine-Generator Startup to 20% - 25% Turbine Power. Once in the Simulator, the Team will place Unit 1 on-line and begin ramping the Unit to 100% power. The performance of this procedure has been analyzed based on the current plant configurations and the PSA indicates green

Event	Malf. #'s	Severity	Instructor Notes and Required Feedback
1	N/A	N/A	Place unit on line and begin ramp up in power. (1-GOP-2.8, 1-OP-TM-001)
2	RM0209	+0.5	CC RM Fail upscale with failure of HCV-CC-100 to auto close.
3	PG0101 PG0202	TRUE	1-PG-P-1A trips on overcurrent, 1-PG-P-1B fails to auto start.
4	RC1503	+1	Przr spray valve fails open (0-AP-53.00)
5	EL0801	TRUE	Loss of Screenwell Transformer (0-AP-12.01)
6	EL01 ED0201 ED0602 FW47	TRUE	Loss of Offsite power with failure of EDG #1, and EDG #2. TDAFW pump failure. (ECA-0.0)

SHIFT TURNOVER INFORMATION

OPERATING PLAN:

Unit 1 is stable at ~5% reactor power with RCS boron concentration of 1415 ppm.

Unit startup is progress in accordance with 1-GOP-1.8, Unit Startup, 2% Reactor Power to Max Allowable Power, and 1-OP-TM-001, Turbine-Generator Startup to 20% - 25% Turbine Power.

All systems and crossties are operable.

Unit #2 is at 100% power with all systems and crossties operable.

Shift orders are to place the Unit on-line and commence ramp to 100% power in accordance with 1-GOP-1.8, Unit Startup, 2% Reactor Power to Max Allowable Power, starting at Step 5.6.13; and 1-OP-TM-001, Turbine-Generator Startup to 20% - 25% Turbine Power, Section 5.7. Performance of this evolution has been authorized and has been PSA analyzed for current plant conditions.

PWR Scenario :

Scenario Objectives: List Each event as a single objective.

A. Given the Unit at ~5% reactor power, place the Unit on-line and commence a ramp up in power in accordance with 1-GOP-1.8, Unit Startup, 2% Reactor Power to Max Allowable Power, and 1-OP-TM-001, Turbine-Generator Startup to 20% - 25% Turbine Power.
B. Given a failure of RI-CC-105, respond in accordance with the ARP 0-RM-M5, and perform actions to isolate the CC Surge Tank.
C. Given a failure of the running PG pump, respond in accordance with ARP 1B-D4 and restore PG flow.
D. Given a failure of the Pressurizer spray valve, respond in accordance with 0-AP-53.00, and restore pressure to normal.
E. Given a failure of the 1G Screenwell Transformer, respond in accordance with 0-AP-12.01 and restore Canal level to normal.
F. Given a Station blackout with loss of AFW, respond in accordance with 1-ECA-0.0, and 0-AP-17.06 to restore power to 1J bus.

Scenario Sequence

Event One: Place Unit On-Line and Ramp Up in Power.

The Team will pre-brief 1-GOP-1.8, Unit Startup, HSD to Max Allowable Power, and 1-OP-TM-001, Turbine-Generator Startup to 20% - 25% Turbine Power prior to Simulator entry. A reactivity plan will be provided for the Team use during the pre-brief and in the Simulator for the Ramp up in power. The Team will place the Unit on-line and commence a ramp up in power. A surrogate operator will be available to the Team until the FRVs are in AUTO and the Bypass valves are closed; the BOP will then assume SG level control.

Malfunctions required: None

Objectives: (RO) Control reactor power per 1-GOP-1.8.

(BOP) Place the Unit on-line IAW 1-OP-TM-001

(BOP) Commence ramp up in power using Turbine controls.

Success Path: (RO) control RCS Tave using control rods and dilution. (BOP) Place Unit –on-line and begin ramp up in power using turbine controls. (Team) When conditions established, place FRVs in automatic.

Event Two: 1-CC-RI-105, CC RM, Fail high without associated auto action.

After the Team has raised power, stable control of SG NR level with FRVs in auto has been achieved, and the Evaluation Team is ready, the malfunction is initiated. This failure causes 1-CC-RI-105 Alert and High alarms to actuate with the failure of HCV-CC-100, CC Surge TK VNT Isol VV, to auto close. The BOP will respond to the RM alarm and take action IAW with RM Annunciator Response Procedure

Malfunctions required: One, RM0209

Objectives: (BOP) Close HCV-CC-100, CC Surge Tank vent valve.

(SRO) Identifies Tech Spec 3.13.C is not met until the surge tank vent valve HCV-CC-100 is closed.

Success Path: HCV-CC-100, CC Surge Tank vent valve is closed.

Event Three: Running PG pump trips, backup PG pump fails to start.

When the Evaluating Team is ready, the malfunction is implemented. This failure causes the running PG pump to trip with the backup pump failing to auto trip. The RO will diagnose the failure based on alarms and indications received, and start 1-PG-P-1B.

Malfunctions required: Two PG0101, PG0202

Objectives: (RO) Identifies loss of running PG pump and failure of the standby pump to auto start. RO manually starts 1-PG-P-1B.

Success Path: Standby PG pump is started.

Event Four **1-RC-PCV-1455B, "B" Spray valve fails Open** (C – RO/SRO, TS - SRO)

When the Evaluating Team is ready, the malfunction is initiated. The controller for the "B" Spray Valve, 1-RC-PCV-1455B, fails causing the spray valve to open fully. The RO will diagnose the failure based on alarms and indications received, place the controller in manual and reduce output to close the spray valve to allow pressure to recover. The Team will implement 0-AP-53.00, Loss of Vital Instrumentation/Controls for this failure.

Malfunctions required: One, RC1503

Objectives: (RO) Identify failure of Spray controller, takes spray controller to MAN and restores pressure to normal band.

(SRO) Direct actions of 0-AP-53.00. Identify Tech Spec 3.12.F.2, DNB Parameters, RCS pressure < 2205 psig, Return RCS to >2205 psig within 2 hours or reduce Thermal Power to less than 5% of Rated Power within the next 6 hours.

Success Path: Stop lowering RCS pressure by closing spray valve, and restore Pressurizer pressure to normal.

Event Five: **Loss of the 1G Screenwell Bus. (**C – BOP/SRO)

When the Evaluating Team is ready the malfunction is initiated. This failure causes the loss of the 1G screenwell bus which causes the loss of the running Unit 1 CW pumps. The Team will respond to the alarms received and implement 0-AP-12.01, Loss of Intake Canal Level.

Malfunctions required: One, EL0801

Objectives: (BOP) Identify loss of Screenwell transformer. Start Unit 1 CW pumps per 0-AP-12.01.

(RO) Throttle Waterbox outlet MOVs to control canal level.

(SRO) Direct actions per 0-AP-12.01.

Success Path:

Event Six: **Loss of Offsite Power, #1/#2 EDGs fail, ECA-0.0.**

When the Evaluation Team is ready, the malfunction is initiated. These failure cause a Loss of Off-site power, failure of the Air Start System for EDG #1, #2 EDG trip on overspeed on startup, and lead the Team to implement ECA-0.0. The TDAFW pump will also trip on startup leading to a condition where no

SHIFT TURNOVER INFORMATION

AFW is available to the SGs following the reactor trip. The Team is expected to start the TDAFW pump to control SGs level, initiate 0-AP-17.06, AAC Diesel Generator - Emergency Operations, to strip the 1J Emergency Bus and load the AAC on the 1J bus. When the AAC DG is loaded on the 1J bus, ECA-0.0 will send the Team to Step 33, where operating equipment is checked. The Team is expected to transition to ECA-0.1, Loss of All AC Power Recovery Without SI Required

Malfunctions required: Four; EL01, ED0201, ED0602, and FW47

Objectives: (RO) Perform immediate actions of ECA-0.0. Close MSTVs.

(BOP) Start TDAFW pump. Strip the 1J bus. Load the AAC diesel on the 1J bus per AP-17.06.

(SRO) Direct actions per ECA-0.0 to restore power to 1J bus.

Success Path: AFW is restored to the SGs. Power is restored to the 1J bus.

The Scenario is terminated once power is restored to 1J bus as determined by NRC Lead evaluator.

Scenario Recapitulation

Total Malfunctions: 10
Abnormal Events: 5, (ARP RM-M5, ARP 1D-B4, 0-AP-53.00, 0-AP-12.01, 0-AP-17.06)
Major Transients: 1
EOPs Entered: 0
EOP Contingencies: 1 (ECA-0.0)

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

Time	Position	Applicant's Action or Behavior
	Team	**1-GOP-1.8** Team will pre-brief Initial Conditions, Precautions and Limitations, and procedure prior to entering simulator.
	RO BOP BOP BOP	**1-GOP-1.8** ***1-OP-CH-021 (*Alternate Dilution Using Blender*) procedure steps are contained in this guide starting at page 20*.** 5.6.13 IF the Steam Dumps are in Auto in Steam Pressure Mode, THEN do the following. Otherwise, enter N/A. a. **Increase Reactor power to approximately 6% -10% by withdrawing the Control Rods and/or using chemical shim.** b. Check that the Steam Dumps come open to maintain Steam Header pressure at approximately 1005 psig. 5.6.14 Check that condenser pressure will be equal to or less than 3.5 inches of Hg pressure (or greater than 26.5 inches of Hg vacuum) before synchronization. 5.6.15 Notify the System Operator and Energy Supply (MOC) that the unit is coming on line. 5.6.16 Check that at least five Polishing beds are in service. *CP operator will report 6 polishers in service.*
	SRO	**1-GOP-1.8** Note prior to Step 5.6.17: • Hotwell temperature should be greater than 70°F before synchronization. This recommended temperature is based on a North Anna Reactor trip caused by low feedwater temperature.
	SRO	**1-GOP-1.8** 5.6.17 Synchronize the Generator with the bus in accordance with 1-OP-TM-001, Subsection 5.7, Synchronizing and Loading the Turbine to 5 percent Rated Load in the OPER AUTO Mode. The team will now go to 1-OP-TM-001 (Subsection 5.7). All previous subsections will be completed. 1-OP-TM-001 actions start on page 15 of this guide.

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Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

	RO SRO SRO	**1-GOP-1.8** 5.6.18 AFTER the generator breakers are closed, THEN verify annunciator 1K-B1, GEN BKR AUX REL FAIL TURB TRIP CKT, is NOT LIT. 5.6.19 Notify the System Operator and Energy Supply (MOC) that the unit is on the line and log the on-line time in the Unit 1 Narrative Log. 5.6.20 Check that the VOLTAGE REGULATOR is in automatic control. IF the VOLTAGE REGULATOR is NOT in automatic control, THEN notify Supervisor - System Operations at 8-730-3345 (Innsbrook).
	SRO SRO SRO	**1-GOP-1.8** **CAUTION** prior to Step 5.6.21: • To provide for a positive channel check indication, steam flow must be verified on all six channels of SG STEAM FLOW PROTECT before 23 percent reactor power is exceeded. **NOTES** prior to Step 5.6.21: • Power level increases should be monitored closely and rods adjusted to maintain Tave close to Tref. Ramp rate will be a function of Steam Generator Level Control. • Chemistry should be notified when power level changes are equal to or greater than 15 percent/hr. • The Turbine should be operated in IMP IN while ramping is in progress. If desired, the turbine may be placed in IMP OUT at approximately 90 to 91 percent power. If the power increase is stopped during the ramp to 100%, IMP OUT may be used to assist in stabilizing the Turbine. 5.6.21 Continue in 1-OP-TM-001, Subsection 5.8, Power Escalation to 20% - 25% Turbine Power, while continuing to perform this procedure.

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

		1-GOP-1.8
	SRO	**CAUTION** prior to Step 5.6.22: • To prevent a Reactor Trip, Step 5.3.24 must be repeated if Reactor Power has decreased below 10 percent and PERM STATUS LIGHTs B1 and C1 are NOT LIT.
		1-GOP-1.8
	RO	5.6.22 WHEN reactor power increases above 10 percent power, THEN perform the following.
	RO	a. Check that the following Trip Status Lights are LIT. _______ 1. Trip Status Light E1, NIS PWR RGE P-10 CH-1 _______ 2. Trip Status Light F1, NIS PWR RGE P-10 CH-2 _______ 3. Trip Status Light G1, NIS PWR RGE P-10 CH-3 _______ 4. Trip Status Light H1, NIS PWR RGE P-10 CH-4
	RO	b. Check that the Perm Status Light A3, P-10 NIS PWR RGE > 10%, is LIT.
	RO	c. Check that the Perm Status Light B2, P-7 NIS PWR RGE AND TURB PWR < 10%, is NOT LIT.
		d. Block the Intermediate Range Trip by performing the following.
	RO	**1. Depress 1/N 38A TRA, INT RNG TRIP - BLOCK, pushbutton.** **2. Depress 1/N 38B TRB, INT RNG TRIP - BLOCK, pushbutton.** 3. Check Perm Status Light B1, NIS INT RNG RX TRIP AND ROD STOP BLOCKED, is LIT.
	RO	e. Block the Power Range Low Trip by performing the following. **1. Depress 1/N 47A TRA, PWR RNG (LO SETPT) TRIP - BLOCK, pushbutton.** **2. Depress 1/N 47B TRB, PWR RNG (LO SETPT) TRIP - BLOCK, pushbutton.** 3. Check Perm Status Light C1, NIS PWR RNG LO SP TRIP - BLOCKED, is LIT
	BOP	5.6.23 Perform the following substeps at the described Turbine Power.
		a. WHEN turbine power increases through 10 percent, THEN check that the following Trip Status Lights are LIT. _______ 1. Trip Status Light E3, TURB PWR > 10% CH-3 _______ 2. Trip Status Light F3, TURB PWR > 10% CH-4
		b. WHEN turbine power increases through 15 percent, THEN check Perm Status Light K1, P-2 AUTO ROD CONTROL BLOCKED TURB PWR < 15%, is NOT LIT.

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

		1-GOP-1.8
	SRO	**NOTE** prior to Step 5.6.24: • When Steam Dumps close, a decrease in RCS temperature should be anticipated and compensatory actions taken.
	RO/BOP	5.6.24 IF Steam Dumps in Auto, THEN verify the Steam Dumps modulate closed as Turbine Power is increased. 5.6.25 IF the Steam Header Pressure controller in Manual, THEN as Turbine power level continues to increase, reduce the STM DUMP VVS DEMAND signal to zero while maintaining Reactor power constant. Enter N/A if controller in Auto.
	BOP	5.6.26 Maintain Turbine Valve Position Limiter approximately 5% above Governor Valve demand. **NOTE:** Steam Flow / Feed Flow indications do not have to be matched to be considered stable. All three MFRVs should be placed in Auto at the same time to ease the transition to Auto feed control.
	TEAM SURROGATE	5.6.27 WHEN Feedwater temperature is greater than 260°F (PCS points T0418A, T0438A, T0458A) with stable Steam Flow / Feed Flow, THEN perform the following: a. Check that the MFRVs are closed. b. Place the MFRVs in Auto. c. WHEN MFRV demand exceeds approximately 9%, THEN slowly close the MFRV Bypass HCVs as the MFRVs come open. **NOTE:** When the Steam Dumps are fully closed, Tave will decrease as Turbine power is increased.
	RO/BOP	5.6.28 IF the Steam Header Pressure controller is in Auto, THEN as Turbine power level is increased, perform the following. Enter N/A if controller in Manual. a. Check that the Steam Dumps modulate closed. b. WHEN the Steam Dumps are closed, THEN place the Steam Header Pressure controller in Manual.
	SRO	5.6.29 IF the Steam Header Pressure controller is in Manual, THEN as Turbine power level continues to increase, reduce the STM DUMP VVS DEMAND signal to zero while maintaining Reactor power constant. Enter N/A if controller was operated in Auto.

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

		1-GOP-1.8
	RO/BOP	5.6.30 Place the STM DUMP MODE SEL switch in the TAVG position as follows. a. Check STM HDR pressure controller demand at zero. b. Place STM DUMP CNTRL switch to OFF/RESET. c. Place STM DUMP MODE SEL switch to RESET and spring return to TAVG. d. Check annunciator 1H-D7, STM DUMP PERM, is NOT LIT. e. Place STM DUMP CNTRL switch to ON. **END OF GOP ACTIONS – 1-OP-TM-001 ACTIONS BEGIN NEXT PAGE.**

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

		1-OP-TM-001
		5.7 Synchronizing and Loading the Turbine to 5% Rated Load in the OPER AUTO Mode
	BOP	**NOTES** prior to Step 5.7.1: • Shift Supervision may authorize entry or exit from this subsection at any step or substep based upon existing plant conditions. N/A must be entered for the specific steps or substeps in the subsection that were not performed as a result of the authorized exit or entry. • Hotwell temperature should be greater than 70°F before synchronization. This recommended temperature is based on a North Anna Reactor trip caused by low feedwater temperature.
	BOP	5.7.1 Check that the Hotwell temperature is greater than 70°F. IF Hotwell temperature is NOT greater than 70°F, THEN evaluate the effects of synchronization with temperature less than 70°F
		1-OP-TM-001
	SRO	**CAUTION** prior to Step 5.7.2: • During Turbine startup and operation at less than 10% electrical load, Condenser vacuum, as read on MCR Condenser Vacuum Recorders CN-PR-101A and CN-PR-101B, should be maintained as high as possible and greater than 26.5 in. Hg to prevent Turbine blade flutter. During shutdown, Condenser vacuum should be maintained as high as possible, and greater than 26.5 in. Hg until the Turbine rotor is on the Turning Gear.
	BOP	5.7.2 Check that the Turbine vacuum indicated on MCR Condenser Vacuum Recorders CN-PR-101A and CN-PR-101B is greater than 26.5 inches of Hg Vacuum.
	BOP	5.7.3 Check that the pumps and fans for the three Main Transformers are in operation. *Field operator will report pumps and fans in service*
		1-OP-TM-001
	BOP	5.7.4 Check that UNIT NO. 1 LOAD MEGAWATTS chart recorder is ON.
	BOP	5.7.5 Check or depress the VV POSTN LIMITER raise button until the VV POSTN LIMIT indicator registers 10% VALVE POSITION.
	SRO	5.7.6 Check that the applicable GOP has been completed up to synchronization, and that the Startup Team is ready to synchronize the generator with the bus.

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

	BOP BOP	**1-OP-TM-001** **NOTE** prior to Step 5.7.7: Shift Supervision may adjust the ramp rate to aid in unit stabilization. 5.7.7 Verify or place the LOAD RATE % PER MIN thumbwheel to position 1. (1%/MIN)
	SRO	**CAUTIONS** prior to step 5.7.8: • The Sync Switch should not be turned to the AUTO position as the AUTO SYNC function is inoperative. • To prevent breaker disagreement, the Generator output breaker control switch should be held in CLOSE until the red light is LIT or the breaker indicates tripped.
	BOP	5.7.8 Synchronize the Generator with the bus using OCB-G102, GEN OUTPUT BKR, by performing the following substeps. IF the Generator will be synchronized using OCB-G1T240, THEN enter N/A AND GO TO Step 5.7.9. a. **Insert the Sync Key into CS-G102, GEN OUTPUT BKR SYNC SWITCH**. b. **Turn CS-G102 to MAN.** c. **Raise the SETTER to 1805 rpm and press the GO button.** d. Check that voltage is indicated on the INCOMING and RUNNING voltmeters. **NOTE**: Slow in the fast direction is one clockwise rotation in 20 or more seconds. e. **Check a slow rotation of the synchroscope in the fast direction. (clockwise) IF NOT, THEN raise or lower the SETTER as required and press the GO Button.**

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

	BOP	**1-OP-TM-001** *5.7.8 (Continued)* **NOTE**: INCOMING and RUNNING voltages should be within 2 volts. f. **Equalize the INCOMING voltage with the RUNNING voltage using the EXCITATION LEVEL control switch.** **CAUTION**: If Generator output is not indicated at the time of synchronization and no operator action is taken, an anti-motoring trip will occur **NOTE**: With the Synchroscope running as close to a 20-second cycle as possible, very little load will be placed on the generator. **NOTE**: Reflexes should be mentally checked with respect to the Synchroscope needle speed so that the Generator Breaker is closed at 12:00 o'clock. g. **WHEN the Synchroscope is at (approximately) 2 minutes to 12:00 o'clock, THEN close OCB-G102, Generator Output Breaker.** **NOTE**: Approximately 15 to 20 seconds may elapse before the Setter indication increases above zero. h. Check that the following indications are NOT LIT. • Permissive Status Light E-3 GEN NO. 1 MOTORING INITIATED • Annunciator 1J-D7, GEN MOTORING TURB LO ΔP i. IF the Generator is motoring, THEN immediately raise the setter to 5% and depress the GO pushbutton. Otherwise, enter N/A. (The ramp rate may be raised as necessary to clear the motoring alarms. When the alarms are clear, the Turbine ramp rate may be lowered or halted as desired.) j. **Turn CS-G102, GEN OUTPUT BKR SYNC SWITCH, to OFF.** k. **Insert the Sync Key into CS-G1T240, GEN OUTPUT BKR SYNC SWITCH.** l. **Turn CS-G1T240 to MAN.** m. **Check that the synchroscope needle stopped at approximately the 12:00 o'clock position.** n. Check that the INCOMING and RUNNING voltages are within 2 volts. o. **Close OCB-G1T240.** p. **Turn CS-G1T240 to OFF and remove the Sync Key.**

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

	SRO/BOP	**1-OP-TM-001** 5.7.9 Synchronize the Generator with the bus using OCB-G1T240, GEN OUTPUT BKR, by performing the following substeps. IF the Generator was synchronized using OCB-G102, THEN enter N/A AND GO TO Step 5.7.10
	SRO/BOP	Notes prior to Step 5.7.10: • The VV POSTN LIMIT setpoint should be raised proportionally as the Turbine load is raised. The Turbine Governor Valves should not be run up against the Limiter. • The Turbine should not be continuously operated on the VV POSTN LIMIT. • During Power Escalation, the VV POSTN LIMIT should be maintained as close as reasonably possible just above the actual governor valve position for the desired power level. This method of operation will prevent a Turbine Governor Valve(s) from failing to an open position due to an electronic or hydraulic failure thereby causing an excessive load on the Unit or causing the Unit to exceed licensed power limits.
	BOP/SRO BOP BOP	5.7.10 IF the VALVE POS LIMIT light is LIT, THEN slowly raise the VV POS LIMIT setpoint until the light is NOT LIT, OR lower Unit load until the VALVE POS LIMIT light is NOT LIT AND adjust the VV POS LIMIT setpoint as required. 5.7.11 Verify that the SPEED light is <u>NOT</u> LIT and the LOAD light is LIT. 5.7.12 RETURN TO appropriate startup GOP to continue the Unit Startup. **THE team will return to GOP-1.8 (momentarily).**
	SRO	**CAUTIONS** prior to Step 5.8.1: • Constant communication between the Reactor Operators on the S/G Level Controls, the Control Rods, Steam Dumps, and the Turbine must be maintained to prevent temperature or level transients. • Rapid Loading of the Turbine - Generator may cause a Steam Generator High Level Trip. **(Reference 2.4.1)**

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

	SRO/BOP	**1-OP-TM-001** **NOTES** prior to Step 5.8: • Shift Supervision may authorize entry or exit from this subsection at any step or substep based upon existing plant conditions. N/A must be entered for the specific steps or substeps in the subsection that were not performed as a result of the authorized exit or entry. • Ramping the Turbine at 1%/min until the Steam Dumps are closed will aid in the transition to auto feed control. Once the Steam Dumps are closed the normal ramp rate is Position 6. • In the OPER AUTO mode, Turbine loading may be stopped by depressing the HOLD pushbutton and may be restarted by depressing the GO pushbutton. 5.8.1 With the OPER AUTO mode selected, set the desired load in the SETTER and depress the GO pushbutton. 5.8.2 Maintain the System Voltage on the 230 KV BUS VOLT meter as requested by the System Operator. 5.8.3 WHEN Turbine power increases above 10%, THEN check PCS alarm Y2060D, Exh Hood Sprays OFF, is received. 5.8.4 WHEN IMPULSE CHAMBER PRESSURE (Turbine Power) passes through 30 percent OR when the startup has stabilized, THEN check or depress the IMP IN pushbutton AND check that the IMP IN light is LIT and the IMP OUT light is NOT LIT. Enter N/A if Turbine control will remain in IMP OUT. *Evaluator's Note: No further actions are expected for this event.*
	BOP	**END EVENT 1**

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Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

Time	Position	Applicant's Action or Behavior
	RO	1-OP-CH-021, Alternate Dilution Using Blender 3.0 Initial Conditions 3.1 Check Primary Grade water is available to the Blender. 4.0 Precautions and Limitations 4.1 Control rod position, Tavg, and/or power level should be observed when making up to the RCS. 4.2 Operation of pressurizer heaters and spray valves should be used to equalize Boron concentration (CB) when changing CB. 4.3 The blender must be frequently monitored for proper operation during the entire duration of the makeup. **(Reference 2.4.1)** 4.4 The Reactor Operator shall notify Shift Supervision before performing any Blender evolution. **(Reference 2.4.1)** 4.5 Rapidly changing VCT level and pressure may affect RCP Seal leakoff, which should be monitored for normal response. 4.6 Operation of the Blender must be Peer checked. 4.7 Due to system configuration, PG flow will continue after reaching the integrator endpoint. Depending on the total flow rate, 0-5 gallons of additional flow should be anticipated. At 100 gpm, four gallons of additional flow is expected. At 60 gpm, two gallons of additional flow is expected. 4.8 The Blender may lock up if the RATE function is in use at the end of make up. (Integrator lock up does NOT affect the Blender AUTO function). **(Ref. 2.4.2)**
	RO	1-OP-CH-021, Alternate Dilution Using Blender 4.0 Precautions and Limitations (Continued) 4.9 If the ENT button on the BA or PG integrator is pressed twice in succession, the integrator will flash the Grand Total on the digital display. The ENT button should be pressed an additional time to exit the Grand Total display. The integrator setpoint will NOT be affected. 4.10 Dilutions of greater than 2000 gallons can result in RCS H2 concentration going low out of band. Dilutions of this amount should be made to the top of the VCT using 1-OP-CH-007, Blender Operations.

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Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

		1-OP-CH-021, Alternate Dilution Using Blender
	RO	5.1 Alternate Dilution **NOTE:** This subsection will be used for the first alternate dilution of the shift. Attachment 3 will be used as a guide for further alternate dilutions for the remainder of the shift, unless Excess Letdown is in service. 5.1.1 Notify Shift Supervision of impending Alternate Dilution. **(Reference 2.4.1)** 5.1.2 Notify STA of impending Alternate Dilution. 5.1.3 Place the MAKE-UP MODE CNTRL switch in the STOP position. 5.1.4 Adjust both of the following controllers for the flow rate and total gallons of Primary Grade water for the dilution. IF the **PG FLOW CNTRL** controller setpoint has previously been set, THEN enter N/A for that substep. a. 1-CH-FC-1114A, PG FLOW CNTRL ______ GPM (IAW Attachment 10) b. Determine the required integrator setpoint by performing the following: ________________ gal (-) _____________=______ Integrator setpoint (Desired Dilution) (anticipated additional flow, dependent on flowrate)

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

		1-OP-CH-021, Alternate Dilution Using Blender
	RO	5.1.4 (Continued) c. Record number of gallons of PG to be added from Step 5.4.4.b and enter into 1-CH-YIC-1114A, PRI WATER SUP BATCH INTEGRATOR (GAL) as follows: ______ GAL 1. Depress PRESET A Button (Controller will read the last value entered into the controller; reads in gallons.) 2. To clear PRESET A, depress the CLR Button. Enter N/A if not required. 3. Enter desired PRESET A value. Enter N/A if not required. 4. Depress ENT Button. 5.1.5 Place the MAKE-UP MODE SEL switch in the ALT DIL position. 5.1.6 IF it is desired to direct the dilution water to the charging pump suction only, THEN place 1-CH-FCV-1114B, BLENDER TO VCT, in the CLOSE position. Otherwise, enter N/A. 5.1.7 Place the MAKE-UP MODE CNTRL switch in the START position. 5.1.8 Check all of the following conditions. a. 1-CH-FCV-1113A, BORIC ACID TO BLENDER, is closed. b. 1-CH-FCV-1113B, BLENDER TO CHG PUMP, is open. c. 1-CH-FCV-1114A, PGW TO BLENDER, is controlling in AUTO. d. 1-CH-1114B, BLENDER TO VCT, is OPEN – **N/A** 5.1.9 IF it is desired to stop the Dilution before the selected amount, THEN place the MAKE-UP MODE CNTRL switch in the STOP position. IF the PRI WATER SUP BATCH INTEGRATOR (GAL) is used to stop the flow, THEN enter N/A for this step. 5.1.10 WHEN the desired amount of makeup has been reached, THEN check both of the following. • 1-CH-FCV-1113B closes. • IF Step 5.4.6 was NOT performed, THEN check that 1-CH-FCV-1114B is closed. Otherwise, enter N/A.

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

		1-OP-CH-021, Alternate Dilution Using Blender
	RO	5.1.11 IF Step 5.4.6 was performed, THEN place 1-CH-FCV-1114B in AUTO. Otherwise, enter N/A.
		5.1.12 IF a manual makeup is desired, THEN enter N/A for Steps 5.4.14 through 5.4.16 AND GO TO Subsection 5.5.
		5.1.12 Place the MAKE-UP MODE SEL switch in the AUTO position.
		5.1.13 Place the MAKE-UP MODE CNTRL switch in the START position.
		5.1.14 Notify Shift Supervision of Blender status. **(Reference 2.4.1)**
		Additional Alternate Dilutions will be performed using 1-OP-CH-021, Attachment 1 (Next Page).

Event Description: .Place Unit on line and ramp up in power.

Cue: When team ready.

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Attachment 1

REPEATED ALTERNATE DILUTIONS

NOTE: This attachment will be used for repeated Dilutions after the initial Subsection 5.1 has been filled out for the shift.

Procedure Steps:	Initial (1)		Initial (2)		Initial (3)	
	Perf.	IV	Perf.	IV	Perf.	IV
1.1 Notify Shift Supervision of impending Alternate Dilution. **(Reference 2.4.1)**						
1.2 Notify STA of impending Alternate Dilution.						
1.3 Place the MAKE-UP MODE CNTRL switch in the STOP position.						
1.4 Check set or set controller and integrator for the flow rate and total gallons of Primary Grade water for the dilution.						
1.5 Place the MAKE-UP MODE SEL switch in the ALT DIL position.						
1.6 IF it is desired to direct the dilution water to the charging pump suction only, THEN place 1-CH-FCV-1114B, BLENDER TO VCT, in the CLOSE position. Otherwise, enter N/A.						
1.7 Place the MAKE-UP MODE CNTRL switch in the START position.						
1.8 Check proper valve positions.						
1.9 WHEN the desired amount of makeup has been reached, THEN check proper valve positions.						
1.10 Check or place 1-CH-FCV-1114B in AUTO.						
1.11 Place the MAKE-UP MODE SEL switch in the AUTO position.						
1.12 Place the MAKE-UP MODE CNTRL switch in the START position.						
1.13 Notify Shift Supervision of Blender status. **(Reference 2.4.1)**						

Event Description: CC RM Fail upscale with failure of HCV-CC-100 to auto close.

Cue: By Examiner.

Time	Position	Applicant's Action or Behavior
	BOP	Diagnoses failure of 1-RM-CC-105 with the following indications/alarms: Alarms: • 0-RM-L5 CC HX A/B OUT ALERT/FAILURE • 0-RM-M5 1-CC-RI-105 HIGH Indications: 1-CC-RI-105 indicates all "EEEEEEs." 1-CC-RI-105, HIGH, WARN and RANGE lights LIT. **NOTE**: RM-M5 Guidance located at end of this section.
	BOP SRO	0-RM-L5 Notes Prior to Step 1: • If a monitor fails, the automatic functions associated with that monitor must be verified or performed. • When HP has surveyed the area and declared radiation levels normal, the components that were realigned due to monitor failure may be returned to normal and activities in the affected area may continue. • Tech Spec 3.13.C requires that HCV-CC-100 remain closed if either CC radiation monitor is inoperable.
	BOP	0-RM-L5 1. CHECK ALARM - READING ON MONITOR GREATER THAN OR EQUAL TO ALERT SETPOINT <u>OR</u> RADIATION LEVEL HAS TRENDED UP • 1-CC-RI-105, HDR A • 1-RM-RR-150C, Pen 1 BOP will identify that the meter indicates "EEEEEEEs" and a steeply rising trend.
	BOP	0-RM-L5, Step 1 RNO a) IF all EEEEEs indicated on display, THEN GO TO 0-OPT-RM-001, Radiation Monitoring Equipment Check.

Event Description: CC RM Fail upscale with failure of HCV-CC-100 to auto close.

Cue: By Examiner.

		0-OPT-RM-001 Precautions and Limitations 4.1 Each process radiation monitor paper advance uses about 5 hours worth of paper. Unnecessary paper advances will cause the roll to run out prematurely. 4.2 Check Sources for the Victoreen digital radiation monitors operate as follows. • For digital PROCESS monitors - The Check Source is exposed to the detector by depressing and holding the CHECK SOURCE pushbutton. • For digital AREA monitors - A Check Source signal is inserted into the detector circuit by depressing and releasing the CHECK SOURCE pushbutton. The Check Source signal is removed when the CHECK SOURCE pushbutton is depressed and released a second time or after approximately three minutes. The digital AREA monitors do not have a radioactive Check Source.
	BOP	0-OPT-RM-001 6.1 Work Preparation 6.1.1 IF a radiation monitor is out of service, THEN enter OOS in applicable spaces of Attachments. NOTE: • A failed Digital Rate Meter is indicated by "EEEEEs" in the digital display window, and the FAIL Alarm light LIT. • If the Radiation Monitor has associated automatic actions, those actions will occur when the monitor fails. 6.1.2 IF this procedure is being performed due to failure of a Digital Radiation Monitor with all EEEEEs displayed, THEN perform the following. Otherwise, enter N/A. a. IF Radiation Monitor has associated automatic actions, THEN check or perform actions as necessary. Otherwise, enter N/A. *BOP may use RM-L5or RM-M5 guidance for completion of the verification of Auto Actions.* **Places 1-HCV-CC-100 in close**. *Directs Unit 2 to place SOV-CC-200 in close.*

Event Description: CC RM Fail upscale with failure of HCV-CC-100 to auto close.

Cue: By Examiner.

	BOP	0-OPT-RM-001 b. On the front panel of rate meter, depress the ON/OFF push button, and check the meter is OFF. **BOP places On/Off switch in Off**. c. WHEN 30 seconds have elapsed, THEN perform Step 6.1.2.d. d. On the front panel of rate meter, depress the ON/OFF push button, and check the meter is ON. BOP Places RM in On, Meter immediately goes to all “EEEEE’s” with the HIGH, WARN and RANGE lights Lit. BOP Notifies SRO RM has failed and I&C assistance is required.
	SRO	SRO will review Technical Specifications 3.13.C and identify that whenever the component cooling water radiation monitor is inoperable, the surge tank vent valve shall remain closed. The SRO may review Technical Specification Table 3.7-5, which will refer the SRO to Technical Specification 3.13.
	SRO	The team will hold a transient brief. During the brief the failure of the CC RM and Vent Valve will be discussed. The RO and BOP will report out critical parameters, as per placard on Main Control Room Bench Board. *The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal, with the exception of the failure of the CC RM. He will also state that containment conditions and the electrical conditions are as you see them.*
	SRO	SRO will notify the Shift Manager of the failure and request I&C assistance.

Event Description: CC RM Fail upscale with failure of HCV-CC-100 to auto close.

Cue: By Examiner.

	BOP	RM-M5 **Note**: Candidate may refer to this ARP initially in response to the High Alarm. **NOTE**: • If a monitor fails, the automatic functions associated with that monitor should be verified or performed. • When HP has surveyed the area and declared radiation levels normal, the components that were realigned due to monitor failure may be returned to normal and activities in the affected area may continue. • Tech Spec 3.13.C requires that HCV-CC-100 remain closed if either CC radiation monitor is inoperable. 1 VERIFY ALARM - READING ON MONITOR GREATER THAN OR EQUAL TO HIGH SETPOINT • 1-CC-RI-105, HDR A • 1-RM-RI-150C, Pen 1 *Identifies Monitor reading is greater than High Alarm.*
	BOP	RM-M5 2 VERIFY CC HEAD TANK VENT VALVE - CLOSED a) Place HCV-CC-100 in OFF (Unit 1) **Places HCV-CC-100 in Close.** b) Place SOV-CC-200 in CLOSE (Unit 2) *Directs Unit 2 operator to place SOV-CC-200 in close.*

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Event Description: CC RM Fail upscale with failure of HCV-CC-100 to auto close.

Cue: By Examiner.

		RM-M5 **NOTE**: The following components are the most likely sources of inleakage to the CC System: • RCP Thermal Barrier • NRHX • Primary Sample coolers • Excess Letdown HX • HRSS coolers • RHR HX • SFP coolers • RHR Pump Seal coolers
	BOP	3 MONITOR CC HEAD TANK LEVEL AND CC TEMP FOR INCREASING LEAKAGE TO CC SYSTEM
	BOP	RM-M5 4 NOTIFY HP TO DO THE FOLLOWING: • Verify area evacuated as necessary • Control access as necessary • Investigate cause • Determine need for setpoint change
	BOP	*Notifies HP.*
	BOP	RM-M5 5 PERFORM ( )-OPT-RC-10.0, REACTOR COOLANT LEAKAGE OR ( )-AP-16.00, EXCESSIVE RCS LEAKAGE, AS NECESSARY *Notifies RO/SRO to perform 1-OPT-RC-10.0, ass necessary.*
	BOP	RM-M5 6 DETERMINE LEAKAGE SOURCE BY SAMPLING AS NECESSARY *Notifies SRO concerning Step.*
	BOP	RM-M5 7 ISOLATE LEAKAGE *Notifies SRO of need to isolate leakage if discovered by sampling.*

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Event Description: CC RM Fail upscale with failure of HCV-CC-100 to auto close.

Cue: By Examiner.

	BOP	RM-M5 8 PROVIDE NOTIFICATIONS AS NECESSARY: • Shift Supervision • OMOC • STA • Health Physics • Instrumentation Department *Informs SRO of required notifications.*
		END EVENT 2

Event Description: 1-PG-P-1A trips on overcurrent, 1-PG-P-1B fails to auto start

Cue: When initiated by Team

Time	Position	Applicant's Action or Behavior
	Team	Diagnose the failure using the following: Alarms: BR-D10, PRI GRADE WTR LO HDR PRESS 1D-B4, PRI WTR TO BLEND LO PRESS Indications: 1-PG-P-1A RED indication light OFF. 1-PG-P-1B does not Auto Start as expected. **NOTE**: Actions for 1D-B4, at the end of this section.
	SRO	Directs the BOP to implement the ARP for BR-D10 Annunciator SRO notifies Unit 2 that PG water flow has been lost.
	BOP	NOTE: The standby PG Pump should automatically start when systempressure decreases to less than or equal to 50 psig. 1 VERIFY PG SYSTEM PRESSURE LESS THAN OR EQUAL TO 75 PSIG • PI-BR-121, Pri Wtr Sup Press *BOP checks 1-BR-PI-121, PRI WTR SUP PRESS meter on BR Panel and notes pressure has decreased to 0 psig.*
	BOP	2 VERIFY PG PUMPS - RUNNING AS NECESSARY • One in Hand • One in Auto Do the following a) Align switches as necessary b) IF PG Pumps off for planned evolution, THEN return to procedure in effect
	BOP	**Starts 1-PG-P-1B by placing control switch in HAND**.
	BOP	*Verifies PG pressure returns to Normal pressure*. (1D-B4 annunciator clears).
	SRO	*Informs Unit 2 of PG system status.* *Directs RO to dispatch operator to check status of 1-PG-P-1A breaker.*

Event Description: 1-PG-P-1A trips on overcurrent, 1-PG-P-1B fails to auto start

Cue: When initiated by Team

	BOP	3 CHECK EXCESSIVE USE OF PG WATER AS INDICATED ON FI-BR-126, PRI WTR SUP PPS FLOW Identifies 0 indicated flow on FR-BR-126, PRI WTR SUP PPS FLOW. Step 3 RNO Do the following: a) Locally check PG Pump for proper operation. *BOP directs Service Building North Yard to check status of 1-PG –P-1A and 1B.* b) Locally check for system integrity *Directs Auxiliary Building operator/SB North Yard to check system integrity.* c) IF system leakage or rupture exists, THEN isolate as necessary AND GO TO Step 6. d) GO TO **Step 5**.
	BOP	5 VERIFY PROPER OPERATION OF PUMP RECIRCULATION VALVE • 1-BR-PCV-114, PRI SUP WATER PUMPS 1A/B RECIRC HDR *Identifies PG header pressure normal.*
	BOP	6 SUBMIT CONDITION REPORT AS NECESSARY
	BOP	7 PROVIDE NOTIFICATION AS NECESSARY • Shift Supervision Notifies SRO that ARP actions complete.
	BOP	Reviews ARP for 1D-B4, PRI WTR TO BLEND LO PRESS.
	BOP	1D-B4 1 CHECK PG HDR PRESS - LESS THAN OR EQUAL TO 65 PSIG ON 1-BR- PI-121 AT THE BORON RECOVERY PANEL Identifies 0 psig indicated.

Op-Test No.: Surry 2016-1 Scenario No.: 4 Event No.: 3

Event Description: 1-PG-P-1A trips on overcurrent, 1-PG-P-1B fails to auto start

Cue: When initiated by Team

	BOP RO RO	1D-B4 2 CHECK PRIMARY GRADE WATER PUMPs - ONE RUNNING • 1-PG-P-1A • 1-PG-P-1B *Identifies no PG pump running.* Step 2 RNO: Start a pump. **Starts 1-PG-P-1B**
	BOP BOP	1D-B4 3 CHECK 1-BR-PCV-114 - THROTTLING TO RAISE HDR PRESS *Identifies PG Pressure returns to normal.*
	BOP	1D-B4 4 CHECK VALVE LINEUP – CORRECT *Identifies PG pump alignment correct.* *BOP may direct Auxiliary Building Operator to walk down PG system for abnormalities.*
	BOP	1D-B4 5 CHECK PG HDR PRESS – GREATER THAN 65 PSIG *Identifies PG pressure normal and stable.*
	BOP	1D-B4 6 NOTIFY SHIFT SUPERVISION *Notifies SRO that ARP review complete.*

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Event Description: 1-PG-P-1A trips on overcurrent, 1-PG-P-1B fails to auto start

Cue: When initiated by Team

	SRO	Update Shift Manager on status of PG system and request electrical maintenance to determine cause of 1-PG-P-1A trip and 1-PG-P-1B failure to AUTO Start.
		END EVENT #3

Event Description: Pressurizer Spray valve fails open.

Cue: By Examiner.

Time	Position	Applicant's Action or Behavior
	RO	Diagnoses Failure based on the following indications: 1-RC-PCV-1455B, Spray from Loop C, fails to 100%. 1-RC-PCV-1444J, Master Cntrl, starts lowering in response to press. Annunciator 1C-B8, PRZR LO PRESS. Pressurizer Heater Banks A / B/ E energize.
	RO	Performs the Immediate Actions of AP-53.00 [1] **Checks redundant indications of pressurizer pressure – NORMAL** [2] **Places the PRZR SPRAY LOOP C Controller in MANUAL closes spray valve.** **Critical Task 1: Close the Spray valve before Reactor trip on low pressure.** If the RO fails to take timely action in response to the Spray Valve opening, an automatic reactor trip on RCS Low Pressure will occur; an unanticipated reactor trip should be considered as failure criteria. Announces completion of Immediate Actions of AP-53.00.
	SRO RO	Step 1, 0-AP-53.00 CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION – NORMAL. Reports Redundant indications of Pressurizer pressure normal.
	SRO RO	Step 2, 0-AP-53.00 PLACE AFFECTED CONTROL(S)/ COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION Reports 1-RC-PCV-1455B, Spray from Loop C in manual and closed.
	SRO	*Conducts brief using Brief Placard. RO Will report Critical parameters. BOP will report Critical Parameters. STA will state Primary integrity intact parameters as RO reported, Secondary integrity intact parameters as the BOP reported, Containment, Radiation Monitoring and Electrical Distribution are normal.*
	SRO RO	Step 3, 0-AP-53.00 * VERIFY REACTOR POWER – LESS THAN OR EQUAL TO 100% Reports reactor power approximately 100% using PCS indication.

Event Description: Pressurizer Spray valve fails open.

Cue: By Examiner.

	SRO	Step 4, 0-AP-53.00 NOTE: • Step 4 failures are listed in order of performance priority. Only the failed instrument/control and associated step number should be read aloud. • When the affected instrument/controller malfunction(s) has been addressed by this procedure, recovery actions should continue at Step 13.
	SRO RO	Step 4, 0-AP-53.00, Continued * DETERMINE THE FAILED INSTRUMENT / CONTROL AND GO TO APPROPRIATE STEP OR PROCEDURE: • PRZR Pressure Control, Step 5
	SRO SRO RO	Step 5, 0-AP-53.00 **NOTE**: RCS pressure decrease will cause a slight decrease in RCS Tave due to negative reactivity from the moderator pressure coefficient. CHECK PRZR SPRAY VALVE CONTROLLERS – NORMAL Reports PRZR Spray Vale Controller is NOT Normal.
	SRO RO	Step 5 RNO, 0-AP-53.00 a. Place failed controller in Manual. b. Restore pressure to normal and stabilize. c. Check or stabilize Turbine load. d. If manual control is ineffective or unavailable, <u>THEN</u> GO TO 1-AP-31.00. e. If RCS pressure returns to normal <u>THEN</u> do the following: 1) Maintain stable Turbine load until pressure control system is returned to normal. 2) If no other instrumentation failure exists, <u>THEN</u> GO TO Step 13. Otherwise, GO TO Step 7. Reports that Spray valve controller is in Manual. Pressure is stable or returning to normal. GO TO step 13.
	SRO RO	Step 13, AP-53.00 13. Check Calorimetric – Functional IAW 1-OPT-RX-001, Attachment 4. *Determines that Calorimetric is functional.* Note: 1-OPT-RX-001, Attachment 4 is included after step 17..

Event Description: Pressurizer Spray valve fails open.

Cue: By Examiner.

	SRO STA	0-AP-53.00 14. REVIEW THE FOLLOWING • Tech Spec 3.7 *Determines Tech Spec 3.7 is met.* *NOTE: If pressure lowered to 2205 psig, then SRO declares Tech Spec 3.12.F.2 is applicable. This requires pressure be raised above 2205 psig in 2 hours.* • VPAP-2802, NOTIFICATIONS - None • TRM SECTION 3.3, INSTRUMENTATION – None. • Reg. Guide 1.97: None • EP-AA-303, EQ. IMPORTANT TO EMERGENCY RESPONSE - None *The STA reports he has completed review and has discussed the results with the Shift Manager.*
	SRO BOP SRO	0-AP-53.00 15 CHECK ADDITIONAL INSTRUMENT / CONTROLLER MALFUNCTION – EXISTS Reports no additional failure exists GOES TO Step 17
	SRO	0-AP-53.00 17. PROVIDE NOTIFICATIONS AS NECESSARY: • Shift Supervision • OMOC • STA (PRA determination) • I&C

Event Description: Pressurizer Spray valve fails open.

Cue: By Examiner.

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Attachment 4

CALORIMETRIC PROGRAM OPERABILITY

NOTE: TRM 3.3.5 shall be reviewed for required actions for non-functionality of the UFM Calorimetric, Normalized Feedwater Venturi, or PCS Calorimetric. Power reduction to less than 98.4% may be required.

1. To check the Primary Plant Performance Program (PP) operability perform the following:

_______ a. Open Programs - Operator Display / Engineering Display

_______ b. Open PP Output Summary - (Operator Display - Primary Plant Poke)

_______ c. Check short timed values for selected basis, Steam Flow (1-OPT-RX-002 box) or Feedflow (1-OPT-RX-003 box), are updating and either good or fair quality.

_______ d. IF selected basis NOT updating and either good or fair quality, THEN contact Reactor Engineering (if available), and then select another calorimetric basis IAW Step 6.3.1. Otherwise, enter N/A.

2. To check the Flow Corrections Program (FL) operability perform the following:

_______ a. Open Programs - Operator Display / Engineering Display

_______ b. Open FL Output Summary (Operator Display - Flow Corr Poke)

_______ c. Check FL Program Status is OK. IF NOT OK, THEN perform the following to check status of different bases.

_______ 1. Open FL0101 - Output Summary (FL Summary Poke)

_______ 2. Compare displayed values to the FL0101 Table below and check selected calorimetric values are updating and either good or fair quality.

_______ 3. IF selected basis NOT operable, THEN contact Reactor Engineering (if available), and then select another calorimetric basis IAW Step 6.3.1.

Event Description: Pressurizer Spray valve fails open.

Cue: By Examiner.

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Attachment 4

CALORIMETRIC PROGRAM OPERABILITY

FL0101 Table				
Flow Correction List	Normalized Feedwater	UFM Feedwater	Venturi Feedwater	Steam Flow
Charging Line Flow Corr	X	X	X	X
Letdown Line Flow Corr	X	X	X	X
SG A-1 FF CORR		X	X	
SG A-2 FF CORR		X	X	
SG B-1 FF CORR		X	X	
SG B-2 FF CORR		X	X	
SG C-1 FF CORR		X	X	
SG C-2 FF CORR		X	X	
SG A-1 SF CORR				X
SG A-2 SF CORR				X
SG B-1 SF CORR				X
SG B-2 SF CORR				X
SG C-1 SF CORR				X
SG C-2 SF CORR				X
SG A-1 FF CORR NORM	X			
SG A-2 FF CORR NORM	X			
SG B-1 FF CORR NORM	X			
SG B-2 FF CORR NORM	X			
SG C-1 FF CORR NORM	X			
SG C-2 FF CORR NORM	X			

		END EVENT #4

Event Description: Loss of 1G Screenwell Transformer.

Cue: When initiated by Team.

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Action or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>Diagnoses failure using the following indications/alarms:

• VSP-J6 – 230 KV HSE TRBL
• VSP L7 - LLIS TROUBLE
  • PCS Alarms:
    • IS-C8 – S/W NO.1 OR NO. 2 XFMR TRBL
    • IS-D9 - XFMR NO. 1 DIFF OPER
  • PCS screens indicating a loss of Unit 1 CW pumps

ARP VSP L-7 directs you to PCS, IS-C8 dispatches personnel to inspect transformer at low levels, and IS-D9 verifies proper crosstie of the 1G & 2G busses then initiates 0-AP-12.01.</td></tr>
<tr><td></td><td>SRO

RO</td><td>Enters 0-AP-12.01, LOSS OF INTAKE CANAL LEVEL

NOTE: EPIPs may be applicable.

1. CHECK ENTRY INTO THIS PROCEDURE-FROM AN EOP OR AN AP TO RESTORE SW FLOW TO THE CCHXs

Step 1 RNO actions: GO TO Step 3.

Determines entry is not from an EOP or an AP to restore SW flow to the CCHXs and goes to step 3</td></tr>
<tr><td></td><td>SRO

RO
BOP</td><td>Caution and Note prior to step 3

CAUTION:
• To prevent turbine damage from turbine stall flutter, Main Condenser vacuum must be maintained greater than 26.5 in-Hg when turbine power is less than or equal to 10%.
• Abnormal Procedure ( )-AP-14.00, LOSS OF CONDENSER VACUUM, should be reviewed if turbine vacuum can NOT be maintained.

NOTE: If both units are at power, it may be necessary to trip one unit to reduce the rate of Intake Canal inventory loss.

*3. TRY TO MAINTAIN INTAKE CANAL LEVEL:
• Throttle Waterboxes
• Reduce Unit load as necessary to maintain Condenser vacuum</td></tr>
</table>

Event Description: Loss of 1G Screenwell Transformer.

Cue: When initiated by Team.

	SRO RO/BOP	**AP-12.01** 4. CHECK INTAKE CANAL LEVEL – LOWERING DUE TO SUSPECTED BREACH. *RO/BOP reports canal level trending* down *but not due to suspected breach.* Goes to step 8.
	BOP SRO BOP	8. START CIRC WATER PUMPS AS REQUIRED FROM THE MCR IAW ATTACHMENT 3 Directs BOP to perform attachment 3. 1. <u>STARTING CIRCULATING WATER PUMPS FROM THE MCR OR LLIS</u> **NOTE:** If starting circ water pumps that have just tripped, 5 minutes must be allowed for coastdown before starting. a. Check 4160 VAC G bus energized by Normal Supply or X-Tie. b. Check the LOCAL CONTROL indication for the pump to be started is NOT LIT. IF remote start is inoperable, THEN verify CW pumps to be started are in LOCAL. c. **Select Soft Control for the pump to be started. N/A if local start to occur.** d. **Enable Soft Control and verify red border**. N/A if local start to occur. e. **Start the selected Circulating Water Pump by pushing the START button**. IF local start to occur due to inoperable PCS, THEN start CW pumps locally. f. Check amps indicated for pump started. g. Direct Outside Operator to perform local operational checks IAW OP-48.1.1. BOP will wait 5 minutes before REMOTELY starting all CW pumps that had previously tripped.

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Event Description: Loss of 1G Screenwell Transformer.

Cue: When initiated by Team.

	SRO BOP	**AP-12.01** Note prior to step 9 **NOTE:** • A PIN Number and a Key are required for entry into the ESW Pump house. • A Key is required for entry into the ESW Pump house and the Low Level Switchgear Room. • There are no restrictions on the number of CW pumps which may be started with the 1G and 2G buses crosstied. 9. SEND OPERATOR TO LOW LEVEL INTAKE TO PERFORM THE FOLLOWING PROCEDURES: • Attachment 2, LOW LEVEL INTAKE RESPONSES • 0-OP-SW-002, EMERGENCY SERVICE WATER PUMP OPERATION • OP-48.1.1, STARTING ANY CW PUMP
	SRO/RO	10. CHECK INTAKE CANAL LEVEL – GREATER THAN TRIP SETPOINT • Annunciator ( )F-G1, INTK CANAL LO LVL TRIP - NOT LIT • Intake Canal level – GREATER THAN 23.5 FEET
	SRO/RO	11. CHECK CW LOSS - SUSTAINED WITH NO EXPECTATION OF RECOVERY Step 11 RNO actions: GO TO Step 13.
	SRO	13. CONTINUE TO REDUCE UNIT LOAD AS NECESSARY TO MAINTAIN VACUUM Team will determine not necessary to reduce turbine load. 14. CHECK GRAVEL NECK MAKEUP FROM INTAKE CANAL – IN SERVICE. GO TO STEP 16 16. CHECK INTAKE CANAL LEVEL – STABLE OR INCREASING 17. GO TO STEP 32

Event Description: Loss of 1G Screenwell Transformer.

Cue: When initiated by Team.

	SRO	**AP-12.01** 32. NOTIFY THE FOLLOWING: • OM on call • Manager Nuclear Operations • STA
	SRO RO RO	Note prior to step 33 **NOTE:** Intake canal level must be maintained above 17.2 FT for adequate RS HX inventory. 33. CHECK INTAKE CANAL LEVEL - RISING 34. CHECK INTAKE CANAL LEVEL - GREATER THAN 17.2 FT
	SRO RO	Note prior to step 35 **NOTE:** If SW flow to the CC HXs is increased, intake canal level should be monitored. The valves must be returned to their original position if canal level decreases. 35. THROTTLE OPEN CC HX SW OUTLET VALVES AS NECESSARY TO LOWER CC TEMPERATURE Team will determine that this is not necessary. 36. VERIFY CAUSE OF LEVEL DECREASE - CORRECTED 37. RESTORE CW AND SW COMPONENTS (INCLUDING CW CHEMICAL INJECTION BY OPENING 1-SA-285 AND 2-SA-274) TO SUPPORT PLANT CONDITIONS - END AP-12.01 actions
		End Event #5

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Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

	Team	Diagnoses loss of power to both Unit 1 Emergency AC busses. Crew will probably direct enter 1-ECA-0.0. Crew may perform1-E-0 and enter 1-ECA-0.0 at step 3 of E-0. Either way is acceptable.
	       RO SRO	ECA-0.0, Loss of All AC Power Performs ECA-0.0 Immediate Actions, OR SRO Prompts ECA-0.0 IAs. **NOTE Prior to Step 1**: CSF Status Trees should be monitored for information only. FRs should NOT be implemented. [1] CHECK REACTOR TRIP: a) Manually trip reactor b) Check the following: Reactor trip and bypass breakers - OPEN Neutron Flux – LOWER Reports “Reactor Tripped” Acknowledges “Reactor Tripped”
	  RO SRO	ECA-0.0, Loss of All AC Power [2] CHECK TURBINE TRIP: a) Manually trip the turbine b) Close MSTVs **Closes MSTVs** Reports “Turbine Tripped”, Immediate Actions of ECA-0.0 complete. Acknowledges “Turbine Tripped” and completion of ECA-0.0 Immediate Actions. **Note**: MSTVs may have been closed on E-0, Step 2.

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Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

		ECA-0.0
	SRO RO	3. CHECK RCS ISOLATION: a) Letdown isolation valves - CLOSED 1-CH-LCV-1460A – **Manually Closes Valve** 1-CH-LCV-1460B – **Manually Closes Valve** 1-RH-HCV-1142 – Checks Valve Closed
	SRO	b) PRZR PORVs – CLOSED
	RO	Reports Yes, PRZR PORVs Closed.
	SRO	c) Loop drain valves - CLOSED 1-RC-HCV-1557A 1-RC-HCV-1557B 1-RC-HCV-1557C
	RO	Reports Yes, Loop drain valves closed.
	SRO	d) RX Head vent valves - CLOSED 1-RC-SOV-100A-1 1-RC-SOV-100A-2 1-RC-SOV-100B-1 1-RC-SOV-100B-2
	RO	Reports Yes, RX Head vent valves closed.
	SRO	e) PRZR vent valves - CLOSED 1-RC-SOV-101A-1 1-RC-SOV-101A-2 1-RC-SOV-101B-1 1-RC-SOV-101B-2
	RO	Reports Yes, PRZR vent valves closed

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

		ECA-0.0
		NOTES Prior to Step 4: Local actions performed in the field may require use of the following: Sound Powered Headphones to communicate and coordinate with Control Room or other locations. Appropriate Vital Area Key to access locked areas.
	SRO	4. CHECK AFW FLOW – GREATER THAN 350 GPM [450 GPM]
	BOP	Reports No, 0 gpm AFW flow indicated.
		Step 4 RNO
	SRO	Do the following: Check TD AFW pump running. IF NOT, THEN manually open steam supply valves.
	BOP	Reports No, TDAFW pump not running, opening steam supply valves. Monitors TDAFW start, and reports AFW Indicated at (actual AFW indicated).
		CT-2: **Establish AFW Flow during SBO**. Establish greater than 350 gpm AFW flow before wide range level in any 2 steam generators is less than 7%.
		TS 3.7, Table 3.7-2, Item 3b, RCP undervoltage starts turbine driven pump, Operator Action 20, With current plant conditions, RCS Temperature and pressure must be reduced to less than 350 °F/450 psig in the next 12 hours. (**Recommend consideration of TS determination concerning TDAFW as a follow-up question, post scenario**).

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

		ECA-0.0
	SRO	CAUTIONS prior to Step 5: If EDG 3 is loaded on 2J emergency bus, an evaluation should be made prior to taking any actions affecting EDG 3. Instrument Air must be restored in a timely manner using 1-IA-C-1, 2-IA-C-1, or the Temporary Diesel Driven Air Compressor. Emergency Bus restoration must consider operation of an emergency bus supplied air compressor. Breaker 15J3 will not close automatically with an Undervoltage or Degraded Voltage Signal on Unit 2 J Bus, except with a valid Unit 1 SI signal.
	RO/BOP	Acknowledge CAUTIONS.
	SRO	5. TRY TO RESTORE POWER TO ANY AC EMERGENCY BUS
		a) Check EDG – RUNNING May query Unit 2 concerning status of #2 and #3 EDG. Unit 2 will respond that #3 EDG is supplying 2J Emergency Bus; #2 EDG started but tripped on overspeed. Report No #1 EDG not running; there appears to be a problem with the air start motors.
	SRO	Step 5. RNO: Do the following: 1) Put EDG in EXERCISE.
	BOP	**Places #1 EDG in EXERCISE.**
	SRO	2) Start EDG.
	BOP	**Presses Engine Start pushbutton, monitors #1 EDG speed rising and stabilizing at 900 RPM.** Reports #1 EDG is NOT running.
	SRO	3) Check established or establish generator voltage by depressing Field Flash pushbutton.
	SRO	5) GO TO Step 5c.

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

		ECA-0.0, Step 5 (Continued)
	SRO	c) Check AC emergency buses - AT LEAST ONE ENERGIZED
	RO/BOP	Reports No, 1H and 1J emergency busses are NOT energized.
	SRO	1) IF both EDGs <u>NOT</u> running THEN initiate 0-AP-17.06, AAC DIESEL GENERATOR – EMERGENCY OPERATIONS <u>AND</u> GO TO Step 6.
		ECA-0.0, Step 6
	SRO	6. CHECK LOSS OF EMERGENCY BUSES - DUE TO APPENDIX R FIRE
	RO	Reports No, Loss due to a loss of off-site power.
	SRO	Goes to Step 8.

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

		ECA-0.0, Step 8
	SRO	**Caution Prior to Step 8**: If an SI signal exists or an SI signal is actuated during this procedure, it should be reset to permit manual loading of equipment on an AC emergency bus.
	RO	Acknowledges Caution.
	SRO	8. DEFEAT AUTO START OF EQUIPMENT: a) Put the following switches in Pull to Lock: **Air recirc fans** ("A" & "C" CTMT Recirc Fans) **LHSI pumps** **OSRS pumps** **ISRS pumps** **CS pumps** **BC pumps** **PRZR heaters** ("A" PRZR heater) **CHG pumps** **MD AFW pumps** **CN pumps** **Filter exhaust fans** (1-VS-F-58A) **CC pumps**
	RO	Places component switches in PTL/Lockout.
	SRO	b) Check breakers open by checking breaker position indicating lights - RED LIGHTS NOT LIT CS pumps ISRS pumps
	RO	Reports Yes, Red lights not lit.
	SRO	c) Check breakers open by checking breaker position indicating lights - RED LIGHTS NOT LIT. MD AFW pumps
	RO	Reports Yes, Red Light not lit.

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

	SRO RO	ECA-0.0, Step 9 9. LOCALLY ISOLATE RECP SEALS. • RCP Seal Return, 1-CH-MOV-1381 • RCP Seal injection, 1-CH-MOV-1370 • Seal Injection needle valves, 1-CH-294, 1-CH-297, 1-CH-300. • Thermal barrier CC, 1-CC-96. *RO closes 1-CH-MOV-1381, and 1-CH-MOV-1370.* *RO directes Field operator to close Seal Injection valves 1-CH-294, 1-CH-297, and 1-CH-300; and 1-CC-96.*
	SRO RO	ECA-0.0 Step 10 10. CHECK COMMUNICATIONS CAPABILITY. • Gaitronics or • **Station Radios**
	SRO RO	ECA-0.0 Step 11 11. TAKE ACTION TO CONSERVE INTAKE CANAL INVENTORY. a. Check Unit 2J bus – ENERGIZED. *RO reports 2J IS ENERGIZED.* *b. Check closed or close condenser circ water isolation valves:* • ***1-CW-MOV-100A*** • ***1-CW-MOV-106B*** • ***1-CW-MOV-100C*** • ***1-CW-MOV-106D*** c. Initiate SW isolation IAW Attachment 4. *RO closes Circ Water isolation valves.* *RO directs field operator to initiate SW isolation IAW Attachment 4*
	SRO	ECA-0.0 Step 12 Caution before Step 12 • When power is restored to either AC emergency bus from an offsite source or the associated EDG recovery actions should continue, starting with Step 33. • If the AAC Diesel Generator is supplying only Bus 1J and is not required by Unit 2, recovery actions should continue, starting with Step 33. 12. TRY TO LOCALLY RESTORE AC POWER. a. Initiate AP-17 series procedures to restore EDGs. b. Initiate 0-AP-10.08, STATION POWER RESTORATION, to resote power to transfer buses. c. Initiate backfeed alignment. Note: At this time power should be restored to 1J bus from the AAC Diesel Generator

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

	SRO RO/BOP	ECA-0.0 Step 33. 33. CHECK IF 4160V AC EMERGENCY POWER IS RESTORED. a. AC emergency buses – AT LEAST ONE ENERGIZED FROM THE FOLLOWING. • Off Site Source • Associated EDG • **AAC Diesel Generator** b. Check FSGs – Any implemented. *No, go to Step 33.d* c. Perform 1-FSG-13. *Not needed goes to next step.* d. Check Bus 1H or 1J ENERGIZED FROM ANY OF THE FOLLOWING. • Off Site Source • Associated EDG • **AAC Diesel Generator** Goes to Step 34.
	SRO RO	ECA-0.0 Step 34 34. MANUALLY OR LOCALLY CONTROL SG PORVS TO STABILIZE SG PRESSURES. *RO checks SG PORVs and adjusts if necessary. (Should not be necessary at this time)*
	SRO RO	ECA-0.0 Step 35 35.CHECK FOLLOWING EQUIPMENT LOADED ON AC EMERGENCY BUS • CHG pump SW pump • CHG pump CC pump • 480 VAC MCCs • Vital bus UPS • Gaitronics • One control room chiller(Two Control Room Chillers if any MCR or ESGR temperature indicator greater than 84°F, monitor approximately every hour) • One Unit 1 Control Room AHU and one Unit 2 Control Room AHU. • One Unit 1 ESGR AHU and one Unit 2 ESGR AHU • Aux Vent filtered exhaust fan • Turning Gear oil pump • Turbine Turning Gear • Common Radiation monitoring cabinets • Semi Vital Bus • IA Compressor. *RO checks all equipment available or directs field operator to start equipment as necessary.*

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

		ECA-0.0 Step 36 Note before Step 36. NOTE: If RCP seal cooling was previously isolated, further cooling of the RCP seals will be established by natural circulation cooldown as directed in subsequent guidelines. 36. IDENTIFY RECOVERY PROCEDURE. a. Check RCS subcooling based on CETCs – GREATER THAN 30°F [85 °F]. b. Check PRZR level – GREATER THAN 22% [50%]. c. Check SI equipment satus: • SI equipment HAS REMAINED SECURED UPON AC POWER RESTORATION. • SI flow – isolated. d. GO TO 1-ECA-1.1, LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED. The Scenario may be terminated when Lead Evaluator determines.

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

Time	Position	Applicant's Action or Behavior
	BOP	0-AP-17.06, AAC DG Generator – Emergency Operations NOTES Prior to Step 1: A one-line diagram showing the AAC Electrical distribution is provided in Attachment 1. The AAC Diesel Generator should automatically start when Transfer Buses D and F OR E and F are deenergized. 1. CHECK EMERGENCY BUSES 1J AND 2H - EITHER OR BOTH DEENERGIZED Yes, both de-energized. **CT-3:** Align AAC Diesel to respective emergency bus 1J prior to declaration of an ELAP. Critical Task start time **begins** when crew enters ECA-0.0, and Critical Task stop time **ends** when AAC DG is loaded on the 1J bus.
	BOP	0-AP-17.06 CAUTION Prior to Step 2: Loading of the AAC Diesel should consider availability of Instrument Air from 1-IA-C-1 or the Temporary Diesel Air Compressor. 2. GO TO THE APPROPRIATE STEP BASED ON DESIRED USE OF THE AAC DIESEL GENERATOR Step 3, Only Bus 1J to be energized Goes to Step 3.
	BOP	0-AP-17.06 3. CHECK AAC DIESEL GENERATOR - AVAILABLE AND RUNNING Annunciator 0-WD-C2, AAC SYSTEM AVAILABLE BUS 1D - LIT AND Annunciator 0-WD-D1, AAC GENERATOR TRIP - NOT LIT Locates to the Waste Disposal Board and verifies annunciator status as above.

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

	BOP	0-AP-17.06 **CAUTION Prior to Step 4**: An overcurrent fault on 15D1 will prevent 0-AAC-BKR-05L3 from closing. NOTE Prior to Step 4: Annunciator 0-WD-C2, AAC SYSTEM AVAILABLE BUS 1D, should go out when 0-AAC-BKR-05L3 closes. Acknowledge CAUTION and NOTE. 4. ENERGIZE TRANSFER BUS D BY CLOSING 0-AAC-BKR-05L3 a) At Unit 1 EDG 3 Control Panel, place Transfer Switch NORMAL/AAC, 0-AAC-43-15J8, in AAC position **Locates to the #3 EDG Panel and places Switch in AAC Position**. b) Check Annunciator 1K-D3, BUS 1D UNDERVOLT - NOT LIT Verifies annunciator 1K-D3 – Not LIT. :
	BOP	0-AP-17.06 5. CHECK OR PLACE THE FOLLOWING LOADS IN PTL a) Put the following switches in PTL / LOCKOUT: **1-VS-F-1B (14J7)** **1-SI-P-1B (14J3)** **1-RS-P-2B (14J8)** **1-RS-P-1B (14J4)** **1-CS-P-1B (14J5)** **PRZR Heater Group A (14J9)** **1-CH-P-1B (15J5)** **1-CH-P-1C (15J2, ALT)** **1-FW-P-3B (15J4)** **1-CC-P-1B (15J10)** 1-VS-F-58B, if powered from Alternate source, 14J13 Places control switches in PTL / Lockout (bolded above). b) Check breakers open by checking breaker position indicating lights - RED LIGHTS NOT LIT 1-CS-P-1B (14J-5) 1-RS-P-1B (14J-4) Verifies indicating lights OUT.

Op-Test No.: Surry 2016-1 **Scenario No.: 1** **Event No.: 6**

Event Description: Loss of Offsite Power, ECA-0.0

Cue: When initiated by Team.

	BOP	0-AP-17.06, Step 5 c) c) Check breaker open by checking breaker position indicating lights - RED LIGHTS NOT LIT 1-FW-P-3B (15J4) Identifies Red Light NOT LIT; Goes to Step 5.
	BOP	0-AP-17.06 Step 6 6. ENERGIZE EMERGENCY BUS 1J a) **Place the Sync switch for 15J8 in ON** b) Check breaker 15J3 is OPEN **Note**: May reset amber light by taking 15J8 control switch to Trip and return to Auto-After-Trip position. Notifies the Team that 1J Emergency Bus will be energized. c) **Close breaker 15J8 by holding control switch in the Closed position for at least five seconds** d) **Place the Sync switch for 15J8 in OFF** Performs actions bolded above. Reports that 1J Bus has been Energized.
		End EVENT #6 **End SCENARIO**

FOLDOUT PAGES FOR REFERENCED PROCEDURES

CONTINUOUS ACTIONS PAGE FOR 1-ECA-0.0

1. ALTERNATE LOW PRESSURE FEEDWATER
   Perform Attachment 10 if TD AFW flow is lost and is NOT immediately recoverable, after Step 5 has been performed.

2. LOSS OF VITAL INSTRUMENTATION OR CONTROL POWER
   Perform 1-FSG-7, LOSS OF VITAL INSTRUMENTATION OR CONTROL POWER, if ELAP is in progress and EITHER condition listed below occurs:
   - ALL DC bus voltages – LESS THAN 105 VDC

     OR
   - Required vital instruments – CAN NOT BE MAINTAINED ENERGIZED

3. LOW DECAY HEAT TEMPERATURE CONTROL
   Perform 1-FSG-9, LOW DECAY HEAT TEMPERATURE CONTROL, if SG pressure can NOT be maintained at or above the target pressure and EITHER condition below:
   - SG pressure 300 psig psig with accumulators NOT isolated/vented

     OR
   - SG pressure 175 psig with accumulators isolated/vented

4. ALTERNATE ECST MAKEUP
   Perform 1-FSG-6, ALTERNATE ECST MAKEUP, if ECST level – LESS THAN 20% and ALL conditions listed below occurs:
   - ELAP is in progress

     AND
   - ECST is available

     AND
   - Step 4 has been performed

5. LONG TERM RCS INVENTORY CONTROL
   Perform 1-FSG-1, LONG TERM RCS INVENTORY CONTROL, if ELAP is in progress and ANY condition listed below occur:
   - PRZR level - LESS THAN 35% [63%]

     AND
   - Time and personnel - AVAILABLE

     OR
   - RVLIS - LESS THAN 78%

     AND
   - RCS pressure - LESS THAN 400 psig

     OR
   - Prior to 16 hours since Loss Of All AC

SIMULATOR OPERATOR'S GUIDE

Simulator Setup

Initial Conditions:

Recall IC -376 **and implement TRIGGER #30 to activate all passive malfunctions and verify Trigger #30 implemented.**

Open the monitor window and add the following points to it:

- asp_ao_off

Enter the following MALFUNCTIONS:

Malfunction	Delay	Ramp	Trigger	Value	Final	Trigger Type (Auto or Manual)
FW47 DISABLE FW-P-2 AUTO START			30		TRUE	
CC07 DISABLE CC-HCV-100 AUTO CLOSURE			30		TRUE	
RM0209 PROCESS RAD MONITOR RI-CC-105 FAILURE	10		1		0.5	MAN
PG0202 DISABLE PG-P-1B AUTO START			3		TRUE	MAN
PG0101 THERMAL OVERLOAD PG-P-1B	10		3		TRUE	MAN
RC1503 PRZR PRESS CONTROLLER FAILURE (1-RC-PC-1444H)	10		5		1	MAN
EL0801 LOSS OF SCREENWELL TRANSFORMER 1	10		7		TRUE	MAN
EL01 LOSS OF OFFSITE POWER	10		9		TRUE	MAN
ED0201 EDG 1 AIR START SYSTEM FAILURE			9		TRUE	MAN
ED0602 EMERG DIESEL 2 OVERSPEED			9		TRUE	MAN

SIMULATOR OPERATOR'S GUIDE

Enter the following Remote Functions

Description	Delay	Ramp	Trigger	Value	Final	Trigger Type (Auto or Manual)
SA_223 SERVICE AIR COMPRESSOR SA-C-2C DISCHARGE ISO VALVE	2:00	1:00	11		100	MAN
CH_294 RC PMP A SEAL ISOL VLV	2:00		13		0	MAN
CH_297 RC PMP B SEAL ISOL VLV	2:00		13		0	MAN
CH_300 RC PMP C SEAL ISOL VLV	2:00		13		0	MAN
SW_39 OUTLET SW ISOLATION VALVE FM 1-CC-E-1A	2:00		15		0	MAN
SW_35 OUTLET SW ISOLATION VALVE FM 1-CC-E-1B	2:00		15		0	MAN
SW_52 OUTLET SW ISOLATION VALVE FM 1-BC-E-1A	2:00		15		0	MAN
SW_48 OUTLET SW ISOLATION VALVE FM 1-BC-E-1B	2:00		15		0	MAN
SW_43 OUTLET SW ISOLATION VALVE FM 1-BC-E-1C	2:00		15		0	MAN
FW_141 AUX FW PUMP FW-P-2 discharge flow to hdr A isol vlv	2:00		17		0	MAN
FW_156 AUX FW PUMP FW-P-3A discharge flow to hdr A isol vlv	2:00		17		0	MAN
FW_171 AUX FW PUMP FW-P-3B discharge flow to hdr A isol vlv	2:00		17		0	MAN

SIMULATOR OPERATOR'S GUIDE

Enter the following EVENT TRIGGERS:

TRIGGER	TYPE	DESCRIPTION
1	MAN	PROCESS RAD MON FAILURE, RI-CC-105
3	MAN	THERMAL OL PG-P-1B, WITH AUTO START FLR
5	MAN	PRZR PRESS CONTROLLER, 1-RC-PC-1444H, FAILURE
7	MAN	LOSS OF SCREENWELL XFMR 1
9	MAN	LOSS OF OFFSITE, WITH EDG 1AND EDG 2 FAILURE
11	MAN	STARTS TEMP DIESEL AIR COMPRESSOR
13	MAN	RCP SEAL INJECTION ISOLATION
15	MAN	SW ISOLATION
17	MAN	AUX FW ISOLATION OF H HEADER
30	MAN	DISABLE FW-P-2 AUTO START
30	MAN	DISABLE CC-HCV-100 AUTO CLOSURE

SIMULATOR OPERATOR'S GUIDE

Verify the following control room setup:

- ❑ Place the simulator in RUN and verify normal 5% power operation indications.
- ❑ Reset the ROD CONTROL SYSTEM
- ❑ Verify SF/FF temp recorder is installed.
- ❑ Verify Red Magnets on the following components: NONE
- ❑ Verify 1-RM-RI-112 aligned to A/B SG and 1-RM-RI-113 aligned to C SG (magnets).
- ❑ Verify Ovation System operating.
- ❑ Reset ICCMs.
- ❑ Verify Component Switch Flags.
- ❑ Verify Brass Caps properly placed.
- ❑ Verify SG PORVs set for 1035 psig.
- ❑ Verify Rod Control Group Step Counters indicate properly.
- ❑ Verify Ovation CRT display.
- ❑ Advance Charts
- ❑ Verify Turbine Thumb Wheel Settings @120 rpm/min and Position 6
- ❑ Verify Containment Instrument Air Compressors are on Inside Suction (all RMs reset)
- ❑ Verify all ARPs have been cleaned

❑ 1C-B8	❑ 1D-B4	❑ VSP-J6	❑ VSP-L7
❑ 0-RM-L5	❑ 0-RM-M5	❑ 0-BR-10	❑

- ❑ Verify CLEAN copies of the following procedures are in place:

❑ 1-GOP-1.8	❑ 1-OP-TM-001	❑ 1-OP-CH-021	❑ 0-OPT-RM-001
❑ 1-AP-31.0	❑ 0-AP-12.01	❑ 0-AP-17.06	❑ 1-ECA-0.0
❑ 0-AP-53.00	❑	❑	❑

- ❑ **Verify Reactivity Placard is current.**
- ❑ Verify ALL PINK MAGNETS are accounted for.

SIMULATOR OPERATOR'S GUIDE

- ❑ Reset Blender Integrators for Boric Acid to 100 and PG 1000.

SIMULATOR OPERATOR'S GUIDE

Brief

This simulator performance scenario is performed in the EVALUATION MODE. You should not direct questions to the evaluators. Otherwise, you should perform as if you were in the MCR.

Your ability to maintain a log is not being graded, but maintaining a rough log is recommended to help during briefs.

If you need to communicate with the Unit 2 operator, verbally state, "Unit 2" and an instructor will locate to the Unit 2 area and respond to you as quickly as possible.

In the unlikely event that the simulator fails such that illogical indications result, the session will be terminated. In other words, respond to what you see. If there is a problem with the simulation, the session will be terminated or adjusted as appropriate based on the specific problem.

Assign operating positions.

Ask for and answer questions.

SIMULATOR OPERATOR'S GUIDE

Conduct shift turnover:

The initial conditions.

Unit 1 currently operating at ~5% reactor Power.

All systems and crossties are operable.

Unit #2 is at 100% power with all systems and crossties operable.

Shift orders are to place the Unit on-line and commence a ramp to 30% in accordance with 1-GOP-1.8, Unit Startup, HSD to Max Allowable Power, Step 5.6.13; and 1-OP-TM-001, Turbine – Generator Startup to 20% - 25% Turbine Power, Section 5.7. Station Management has given permission to continue the startup.

Current Boron concentration is 1415 ppm. 8.0 weight percent in the "A" BAST.

When the team has accepted the shift, proceed to the Session Conduct Section.

SIMULATOR OPERATOR'S GUIDE

Session Conduct:

- Ensure conditions in Simulator Set-up are established.
- **Ensure Trigger 30 is active prior to team entering the simulator.**
- Verify Exam Security has been established and ASP_AO_OFF = True.

EVENT 1: Place Unit on line, commence ramp up

Operations Supervisor/Management:

- **If contacted,** acknowledge placing Unit on-line.

System Operator (SOC)/MOC:

- **When contacted**, acknowledge Unit 1 on-line.

Field Operators:

- **If contacted**, Turbine Building Operator is monitoring Lube Oil temperatures and will adjust cooling flow as necessary.
- **If contacted:** Operator reports main transformer pumps and fans operating.

Polishing Building:

- **If contacted**: 6 polishing beds are in service.

STA:

Will monitor ramp.

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EVENT 2 1-CC-RI-105 Fails high with no Auto Action

When cued by examiner, implement **Trigger #1**.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the failure 1-CC-RI-105.
- **If contacted,** will acknowledge TS 3.13 requirement to maintain the CC Surge Tank Vent Valve closed.
- **If contacted,** will contact I&C.
- **If contacted,** will take responsibility for writing the CR.

Unit 2 Operator:

- Will provide copy of 0-OPT-RM-001

STA:

- **If contacted**, will acknowledge the failure of 1-CC-RI-105. The individual(s) contacted will also acknowledge (but not confirm/deny) any TS LCOs.
- **If contacted,** will take responsibility for writing the CR.
- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal, with the exception of 1-CC-RI-105. He will also state that containment conditions and the electrical conditions are as you see them.

Field Operators:

- **If contacted**, field operators will report no issues at the RM detector.

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge instrumentation failure and Notify I&C.

Role play as other individuals as needed.

SIMULATOR OPERATOR'S GUIDE

EVENT 3 **1-PG-P-1A trips, no Auto Start 1-PG-P-1B.**

When cued by examiner, implement **Trigger #3**.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the trip of 1-PG-P-1A, and failure of 1-PG-P-1B to auto start.
- **If contacted,** will take responsibility for writing the CR.
- **If requested**. Will notify Electrical Maintenance to investigate.

STA:

- **If contacted**, will acknowledge the trip of 1-PG-P-1A, and failure of 1-PG-P-1B to auto start.
- **If contacted,** will take responsibility for writing the CR.
- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal, with the exception of the failed CC RM. He will also state that containment conditions and the electrical conditions are as you see them.

Field Operators: (three minutes elapse from dispatch to report).

- Will check local status of 1-PG-P-1A, no abnormalities noted.
- Will check status of 1-PG-P-1A breaker, MCC 1B1-1A 1C. Will report that breaker has thermalled.

Work Week Coordinator:

- **If contacted**, will acknowledge thermal trip of 1-PG-P-1A and take responsibility for contact of Electrical Maintenance.

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Unit 2:

- **If contacted**, will acknowledge the loss of PG flow and restoration of PG when 1-PG-P-1B started.

Role-play as other individuals as needed.

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EVENT 4 **PRZR Spray valve fails open.**

When cued by examiner, implement **Trigger #5**.

Critical Task 1: Close the Spray valve before Reactor trip on low pressure. If the RO fails to take timely action in response to the Spray Valve opening, an automatic reactor trip on RCS Low Pressure will occur; an unanticipated reactor trip should be considered as failure criteria.

Operations Supervisor/Management:

- **If contacted**, acknowledge failure of PRZR Spray valve controller.
- **If asked,** will notify I&C of the failure.
- **If asked,** will notify the OMOC.
- **When notified,** acknowledge any Tech Spec requirements (DNB) due to the failure, but do not imply concurrence with SRO Tech Spec determination.

STA:

- **If contacted**, acknowledge the failure, acknowledge the Tech Spec requirements for the failed channel, but do not imply concurrence with the Tech Spec determination by the SRO.
- **When notified,** VPAP-2802, Reg Guide 1.97, TRM Section 3.3, and EP-AA-303 have been reviewed and results discussed with the Shift Manager.
- **If the team has a transient brief:** The STA will have no input.

I&C:

- **If notified**, acknowledge the failure and the need to place the channel in trip.

Maintenance/Work Week Coordinator:

- **If contacted**, will notify I&C of the failed channel, have I&C prepare to place the channel in trip, and prepare to troubleshoot the cause of the failure.

Role play as other individuals as needed.

SIMULATOR OPERATOR'S GUIDE

EVENT 5 Loss of 1G Screenwell Transformer-.

When cued by examiner, implement **Trigger #7**.

Operations Supervisor/Management:

- **If contacted**, will acknowledge the loss of 1G transformer and entry into AP-12.01.
- **If contacted,** will take responsibility for writing the CRwi.

Unit 2 Operator:

- If directed, acknowledge that they need to implement AP-12.01.
- If directed acknowledge they need to throttle waterboxes.

STA:

- **If contacted**, will acknowledge the loss of 1G transformer.
- **If contacted,** will take responsibility for writing the CR.
- **If the team has a transient brief:** The STA will state that primary integrity is as the RO reported and that secondary integrity is as the BOP reported. The STA will state that radiological conditions are normal with the exception of the previously identified radiation monitor alarms. He will also state that containment conditions and the electrical conditions are as you see them.

Field Operators:

- **If contacted,** will report to the low levels and/or switchyard as directed. The outside operator will report a large bird, dead on top of 1G transformer.

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge the loss of 1G transformer and will commence investigations and/or efforts to repair.

SIMULATOR OPERATOR'S GUIDE

EVENTS 6 Loss of Offsite power, ECA-0.0.

When cued by examiner, implement **Trigger #9**.

Critical Tasks
CT-2: Establish AFW Flow during SBO; Establish greater than 350 gpm AFW flow before wide range level in any 2 steam generators is less than 7%.
CT-3: Align AAC Diesel to respective emergency bus 1J prior to declaration of an ELAP.

Operations Supervisor/Management:

- **If contacted**, acknowledge LOOP, and Reactor Trip of Both Units.

Unit Two:

- **If contacted**, Event initiated, acknowledge WD, BR, and RMA alarms.
- **If asked**, will report that #2 EDG is NOT running (started but tripped immediately), and #3 EDG is loaded on 2J Emergency Bus.

Field Operators:

- **If asked**, to start and align temporary diesel air compressor, **Insert Trigger 11**. After 3 minutes inform MCR that Temporary Diesel air compressor is running.
- **If asked,** to close seal injection valves, **Insert Trigger 13**, and inform operator after 2 minutes that Seal injection valves are isolated.
- **If asked,** to close thermal barrier isolation, 1-CC-96, **Insert Trigger 13**, and inform operator after 2 minutes that 1-CC-96 IS isolated.
- **If asked,** to isolate SW IAW Attachment 4, **Insert Trigger 15,** and inform operator after 2 minutes that Attachment 4 is complete.
- **If asked,** to isolate AFW H Train, 1-FW-141/156/171, **Insert Trigger 17,** wait 2 minutes then inform operator that H Train of AFW is isolated.

SIMULATOR OPERATOR'S GUIDE

Maintenance/Work Week Coordinator:

- **If contacted**, will acknowledge the failures and commence investigations.

STA:

- **If contacted**, acknowledge LOOP and go to floor 10 minutes following call.
- **If the team has a transient brief:** The STA will have no input for the brief.

Role play as other individuals as needed.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Simulator Job Performance Measure 001AA2.05
[Alternate Path]

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

Adjust the PRNIs in accordance with 1-OPT-RX-001

K/A: 001AA2.05 Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Uncontrolled rod withdrawal, from available indications. (4.4 / 4.7)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)	10 Minutes	____ Minutes

Conditions

- Task is to be PERFORMED in the simulator.
- Unit 1 operating at 90% power. 1-OPT-RX-001 has been completed up to Section 6.2.

Standards

- Adjust N42 and N44 IAW 1-OPT-RX-001, Section 6.2 and Attachment 1.

Initiating Cues

- Unit 1 operating at 90% power.
- The Unit 1 RO has completed 1-OPT-RX-001, Section 6.1.
- I am the Nuclear Shift Manager. You are the Unit 1 BOP. You are to perform 1-OPT-RX-001, Section 6.2.
- When you have completed the actions associated with this task, please inform me.

Terminating Cues

- Rods placed in Manual IAW 0-AP-1.00.

Procedures

- 1-OPT-RX-001, Rev 51

Tools and Equipment
- None

Safety Considerations
- None

Simulator Setup

- Call up 90% power IC and initialize. Place simulator in RUN.
- Adjust N41 and N43 to 90% indication using drawer gain control.
- Adjust N42 to an indication of 87% power using the drawer gain control.
- Adjust N44 to an indication of 88% power using the drawer gain control.
- **Trigger 1 -** Enter malfunction RD0102, Continuous Rod Withdrawal All Modes.

Directions to the Applicant

- I am the Nuclear Shift Manager. You are the Unit 1 BOP.
- You are to perform Section 6.2, Adjusting NI Channels, of 1-OPT-RX-001.
- When you finish the actions necessary to accomplish this task, please inform me.

Notes

- The Applicant is given the marked-up copy of 1-OPT-RX-001. This evolution may be pre-briefed.

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- *An additional instructor may be needed to silence alarms for the examinee.*
- **START TIME:**

STEP 1: Reviews Purpose, Initial Conditions, and Precautions and Limitations of 1-OPT-RX-001. **STANDARD:** a) Reviews Purpose 1.1, 1.2, and 1.3. b) Reviews Initial Conditions 3.1 and 3.2. c) Reviews Precautions and Limitations 4.1 through 4.24; noting 4.3, and 4.6. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2: Compare each NI channel percent power indication with the Calcalc Total Thermal Pwr (UFM, Venturi or Normalized Feedwater) or Calcalc 10-Min Avg Pwr (Steam Flow), whichever is the standard. (Each NI should be within + 2% and - 0% of the Calorimetric value if Reactor power is greater than or equal to 90%, OR within + 4% and - 0% of the Calorimetric value if Reactor power is less than 90%.). *(Step 6.2.1)* **STANDARD:** a) Reads and Initials Step 6.2.1. b) Refers to Step 6.1.12 to determine Calcalc Total Thermal Power: 89.25%. c) Locates to PRNI drawers and observes N41 indicating 90%, N42 indicating 87%, N43 indicating 90%, and N44 indicating 88%. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 3: **NOTE**: Gain potentiometer adjustment can cause average flux deviation alarms as well as high flux rod stop alarms. This should be anticipated when adjusting gain potentiometers. (Reference 2.4.6). **STANDARD:** Reviews NOTE prior to Step 6.2.2. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 4: IF the NI Channel is within tolerance but adjustment will better align it with the calorimetric, THEN obtain Shift Supervision concurrence AND adjust NI Channel IAW Attachment 1 to the value recorded in Step 6.1.12 or Step 6.1.13. Record initials on Attachment 1. IF no NI adjustment is made, OR NI is NOT within tolerance, THEN enter N/A. *(Step 6.2.2)* **STANDARD:** Enters N/A and Initials Step 6.2.2. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 5: IF NI channel is NOT within tolerance, THEN obtain Shift Supervision concurrence AND adjust the gain potentiometer on the front panel of each NI Channel IAW Attachment 1 to the value recorded in Step 6.1.12 or Step 6.1.13. Record initials on Attachment 1. IF all NI channels are within tolerance, THEN enter N/A. *(Step 6.2.3)* **STANDARD:** a) Initials Step 6.2.3. b) Reports to Shift Manager (Evaluator) that N42 and N44 require adjustment, and requests authorization to make these adjustments. c) Initiates Attachment 1. **EVALUATOR'S NOTE:** **When asked:** Initial Step 6.2.3 to authorize adjustment of PRNIs. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 6: Attachment 1, 1-OPT-RX-001, NI Calibration. **CAUTION**: To prevent introducing non-conservative High Flux Trip and High Flux Rod Stop setpoints, setpoint changes required by the following step must be completed before any associated Gain Potentiometer adjustments are performed. **STANDARD:** Reviews CAUTION Prior to Step 1 of Attachment 1. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 7: IF Reactor power is less than 90% AND the Gain Potentiometer on any NI will be decreased, THEN before adjusting NIs, have I & C lower the High Flux Trip and High Flux Rod Stop setpoints on all NIs based on current Reactor power level. Otherwise, enter N/A. (Reference 2.4.5). *(Attachment 1, Step 1)* **STANDARD:** Enters N/A and Initials Step 1 of Attachment 1. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 8: N41. (*Attachment 1 Table*) **STANDARD:** a) Enters N/A in Item 3) block, N41 column of the Table. b) Enters N/A in item 4) block, N41 Column of the Table. c) Enters N/A in Item 5) block, N41 column of the Table. **EVALUATOR'S NOTE:** A **KEY** is provided on Page 9 of 11, depicting the completed Table on Page 26 of 1-OPT-RX-001. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 9: N42. (*Attachment 1 Table*) **STANDARD:** a) Enters 87% in Item 3) block, N42 column of the Table. b) Checks alternate indications of reactor Power (i.e., N41, N43, Turbine Impulse Pressure, Calorimetric power) prior to adjustment of N42 IAW P&L 4.6. **c) Adjusts gain control on N42 Drawer to 90% indication. (Band: 90 – 92%)** d) Enters Initials in item 4) block, N42 Column of the Table. e) Records 90% in Item 5) block, N42 column of the Table. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 10: N43. (*Attachment 1 Table*) **STANDARD:** a) Enters N/A in Item 3) block, N43 column of the Table. b) Enters N/A in item 4) block, N43 Column of the Table. c) Enters N/A in Item 5) block, N43 column of the Table. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 11: N44. (*Attachment 1 Table*) **STANDARD:** **a) Places Rod control in manual, and initials item 2) block, N44 column of the Table.** a) Enters 88% in Item 3) block, N44 column of the Table. b) Checks alternate indications of reactor Power (i.e., N41, N43, Turbine Impulse Pressure, and Calorimetric power) prior to adjustment of N44 IAW P&L 4.6. **c) Adjusts gain control on N44 Drawer to 90% indication. (Band: 90 - 92%)** d) Enters Initials in item 4) block, N44 Column of the Table. e) Records 90% in Item 5) block, N44 column of the Table. f) Allows at least one (1) minute to pass before placing rod control in automatic following gain control manipulation. **g) Places Rod control in Automatic.** h) Identifies Rod outward rod motion with no Tave/Tref deviation. **i) Returns rod control to manual.** j) Notes Rod Motion NOT stopped. **EVALUATOR'S NOTE:** **Booth Operator:** When rod control placed in Automatic for item g) above, actuate **Trigger 1**. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 12: Performs Immediate Action Steps of AP-1.00, Rod Control System Malfunction. a) Check for continuous rod withdrawal – YES b) Stop rod motion • Put ROD CONT MODE SEL switch in MANUAL • Check rod motion stopped – NO **c) Trip Reactor and GO TO 1-E-0, Reactor Trip or Safety Injection** d) Performs 1-E-0 Immediate Actions **EVALUATOR'S NOTE:** **Evaluator:** JPM may be stopped any time after the reactor is tripped. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 13: NOTIFY NUCLEAR SHIFT MANAGER (EVALUATOR) STATUS OF TASK. When report of completion of AP-1.00 Immediate Actions made, Candidate should report completion of task. **COMMENTS:** ** **JPM COMPLETE** **	_______ **SAT** _______ **UNSAT**

STOP TIME:

____

____

____

____

KEY

	NI-41	NI-42	NI-43	NI-44
2) Place rod control to MANUAL. Enter N/A if NI-44 will <u>NOT</u> be adjusted.				Candidate Initials
3) Record As Found NI power level for each channel to be adjusted. Enter N/A for channel(s) not being adjusted.	N/A	87%	N/A	88%
4) Adjust the Gain Potentiometer on the front panel of each NI channel to the new Reactor Power value and initial appropriate block(s). Enter N/A for channel(s) not being adjusted.	N/A	Candidate Initials	N/A	Candidate Initials
5) Record As Left NI power level for each channel adjusted. Enter N/A for channel(s) not adjusted.	N/A	90%	N/A	90%
6) Allow at least one minute to pass before placing the rod control back to AUTO. Enter N/A if NI-44 was <u>NOT</u> adjusted.				---------

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- Unit 1 is operating at 90%.
- The Unit 1 RO has completed 1-OPT-RX-001, Section 6.1, Calculating Reactor Power, Using Primary Performance Program, and recorded CALCALC Total Thermal Power on Step 6.1.12.

Initiating Cues

- I am the Nuclear Shift Manager. You are the Unit 1 BOP.
- You are to perform Section 6.2, Adjusting NI Channels, of 1-OPT-RX-001.
- When you finish the actions necessary to accomplish this task, please inform me.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- Unit 1 is operating at 90%.
- The Unit 1 RO has completed 1-OPT-RX-001, Section 6.1, Calculating Reactor Power, Using Primary Performance Program, and recorded CALCALC Total Thermal Power on Step 6.1.12.

Initiating Cues

- I am the Nuclear Shift Manager. You are the Unit 1 BOP.
- You are to perform Section 6.2, Adjusting NI Channels, of 1-OPT-RX-001.
- When you finish the actions necessary to accomplish this task, please inform me.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Simulator Job Performance Measure 011EA1.11

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

Isolate Leaking RSHX (“D”)

K/A: 011EA1.11 Ability to operate and monitor the following as they apply to a Large Break LOCA: Long-term cooling of core. (4.2 / 4.2)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)/SRO(U)	8 Minutes	____ Minutes

Conditions

- Task is to be PERFORMED in the simulator.
- A LBLOCA has occurred on Unit 1. Following completion of 1-E-0, Attachment 1, Step 9d (Check OSRS pumps – NOT CAVITATING), Annunciator RM-C8, RS/SW HX D ALERT/FAILURE was received followed by Annunciator RM-D8, 1-SW-RI-117 High.

Standards

- Completion of Annunciator Response Procedure 1-RM-D8, 1-SW-RI-117 HIGH, Step 12.

Initiating Cues

- A LBLOCA has occurred on Unit 1. Unit 2 is operating at stable power level of 100%.
- The Operating Team is currently performing 1-E-1, Loss of Reactor or Secondary Coolant.
- 1-E-0, Attachment 1, Step 9d, Check OSRS pump –NOT CAVITATING has just been completed.
- Annunciator 1-RM-C8, RS/SW HX D ALERT/FAILURE, and 1-RM-D8, 1-SW-RI-117 HIGH have just been received.
- I am the Nuclear Shift Manager. You are the Unit 1 BOP. You are to perform Annunciator Response Procedure 1-RM-D8, 1-SW-RI-117 HIGH.
- When you have completed the actions to accomplish this Task, please inform me.

Terminating Cues

- Report of completion of Annunciator Response Procedure 1-RM-D8, 1-SW-RI-117 HIGH, Step 12.

Procedures

- 1-RM-D8, 1-SW-RI-117 HIGH (Rev 4)

Tools and Equipment

- None

Safety Considerations

- None

Simulator Setup

- Call up 100% power IC and initialize. Place simulator in RUN.
- Enter Malfunctions:
  - RC0101, RCS Cold Leg A Pipe Rupture; 50% severity
  - RS0504, Malf Leak Flow from RS-E-1D to SW System; 100% severity
- Place Simulator in Run. Perform Actions of E-0 until ORS Pumps start.
- Ensure RM-D8, 1-SW-RI-117 HI goes into alarm.
- Reset CLS signal when Containment pressure lowers to < 14.2 psia.
- Place Simulator in Freeze until JPM performance.

Directions to the Applicant

- I am the Nuclear Shift Manager. You are the Unit 1 BOP. You are to perform Annunciator Response Procedure 1-RM-D8, 1-SW-RI-117 HIGH.
- When you have completed the actions to accomplish this Task, please inform me.

Notes

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- *An additional instructor may be needed to silence alarms for the examinee.*
- **START TIME:**

STEP 1: VERIFY ALARM - READINGS ON MONITOR OR CHART RECORDER GREATER THAN OR EQUAL TO HIGH SETPOINT. *(Step 1)* **STANDARD:** a) Reviews CAUTION prior to Step 1: Operation of the RSHX SW radiation monitors is not required to directly support RS functional requirements. b) Verifies 1-SW-RI-117 greater than the High alarm setpoint by pressing the drawer High pushbutton and observing indication less than current reading. c) May also verify that the red HIGH light is lit and bar graph LEDs are red. d) Verifies 1-RM-RR-150A, Pen 4 is trending with drawer indication. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2: CHECK UNIT 2 - OPERATIONS NORMAL. *(Step 2)* **STANDARD:** Asks Unit 2 if Unit 2 Operation Normal or remembers Unit 2 status from initial briefing. **EVALUATOR'S NOTE:** **If asked**: Unit 2 is stable at 100% power. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 3: CHECK AFFECTED HX IN SERVICE. *(Step 3)* **STANDARD:** a) Checks SW Flow to 1-RS-E-1D by observing 1-SW-MOV-103A or 103B OPEN, GREEN lights Out, RED Lights Lit. b) Check SW flow to RS-E-1D by observing 1-SW-MOV-104D and 1-SW-MOV-105D OPEN, GREEN Lights Out, RED Lights Lit. c) Check SW flow by observing SW flow through 1-SW-FI-106D, RS HX D SW OUTLET FLOW. d) Checks 1-RS-P-2B running by observing breaker closed indications, RED Light Lit and GREEN Light Out. e) Check amps indicated for 1-RS-P-2B and discharge pressure indication on 1-RS-PI-156B, DISCH PRESS PUMP B. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 4: PLACE ADDITIONAL RS HX(s) IN SERVICE AS REQUIRED. *(Step 4)* **STANDARD:** a) Identifies RS-E-1A, RS-E-1B, and 1-RS-E-1C in service by observing SW valves Open, SW Flow through the HXs, and 1-RS-P-1A/1B/2A running. b) Determines sufficient HXs in service to maintain heat sink for Reactor Core following shift to RMT. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 5: CONSULT WITH SHIFT SUPERVISION AND OMOC TO DETERMINE IF THE AFFECTED RSHX SHOULD BE REMOVED FROM SERVICE. *(Step 5)* **STANDARD:** a) Reviews CAUTION prior to Step 5: CAUTION: An operating RS train should NOT be secured unless sufficient redundant trains are available for containment heat removal. b) Consults with Shift Manager (Evaluator) to determine if affected RSHX should be removed from service. **EVALUATOR'S NOTE:** **When asked:** State that the OMOC has been consulted and will concur with Candidates decision. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 6: CHECK AFFECTED RSHX TO BE REMOVED FROM SERVICE. *(Step 6)* **STANDARD:** Identifies 1-RS-E-1D to be removed from service. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 7: STOP ASSOCIATED RS PUMP AND PLACE IN PTL: *(Step 7)* **STANDARD:** **a) Stops 1-RS-2B.** b) Checks 1-RS-P-2B secured by observing zero (0) amps indicated, GREEN Light Lit and RED Light Out on breaker control switch, and pump discharge pressure on 1-RS-PI-156B, DISCH PRESS PUMP B. c) Acknowledges Annunciator 1A-H8, RS PP 2B LOCKOUT OR OL TRIP. **EVALUATOR'S NOTE:** Per 1-RM-D8, Step 7, the pump control switch should be placed in PTL. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 8: STOP ASSOCIATED RSHX SW PUMP: *(Step 8)* **STANDARD:** a) Reviews NOTE prior to Step 8: NOTE: CLS must be reset to allow securing RSHX SW pump from MCR. b) Verifies CLS is reset by verifying the following annunciators are NOT LIT: • 1B-A4 – CLS SYS COIL FAILURE • 1B-B4 – CLS HI TR A • 1B-B5 – CLS HI TR B • 1B-C4 – CLS HI-HI TR A • 1B-C5 – CLS HI-HI-TR B **c) Places 1-SW-P-5D control switch to Stop.** b) Verifies 1-SW-P-5D stopped by observing GREEN Light On, RED Light Off. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 9: ISOLATE SW TO 1-RS-E-1D BY CLOSING THE ASSOCIATED MOVS: *(Step 9)* **STANDARD:** a) Reviews NOTE prior to Step 9: NOTE : If an undervoltage condition occurs before CLS reset, the SW MOVs will reopen when voltage is returned to normal. b**) Closes 1-SW-MOV-104D by placing control switch in close position.** c) Verifies 1-SW-MOV-104D closed by observing GREEN Light Lit, and RED Light Off. d) **Closes 1-SW-MOV-105D by placing control switch in close position.** e) Verifies 1-SW-MOV-105D closed by observing GREEN Light Lit, and RED Light Off. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 10: INCREASE SURVEILLANCE ON INSERVICE MONITORS. *(Step 10)* **STANDARD:** a) Locates to RM Panel and observes videographic recorder 1-RM-RR-150A and 1-SW-RI-114, 1-SW-RI-115, and 1-SW-RI-116. b) Observes trend stable on 1-RM-RR-150A, pen 1 through 3. c) Observes trend stable on 1-RM-RI-114, 115, and 116. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 11: INITIATE A CONDITION REPORT. *(Step 11)* **STANDARD:** Candidate states need to initiate a Condition Report (CR) to Shift Manager (Evaluator). **EVALUATOR'S NOTE:** Inform Candidate that the STA will initiate the CR. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 12: PROVIDE NOTIFICATIONS AS NECESSARY. *(Step 12)* **STANDARD:** Candidate notifies Shift Manager (Evaluator) that the following Notifications are required: • Shift Supervision • OMOC • STA • Health Physics • Instrumentation Department **EVALUATOR'S NOTE:** Acknowledge Notifications are required. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 13: NOTIFY NUCLEAR SHIFT MANAGER (EVALUATOR) STATUS OF TASK. Candidate should report completion of task. **COMMENTS:** ** **JPM COMPLETE** **	_______ **SAT** _______ **UNSAT**

STOP TIME:

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Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- A LBLOCA has occurred on Unit 1. Unit 2 is operating at stable power level of 100%.
- The Operating Team is currently performing 1-E-1, Loss of Reactor or Secondary Coolant.
- 1-E-0, Attachment 1, Step 9d, Check OSRS pump –NOT CAVITATING has just been completed.
- Annunciator 1-RM-C8, RS/SW HX D ALERT/FAILURE, and 1-RM-D8, 1-SW-RI-117 HIGH have just been received.

Initiating Cues

- I am the Nuclear Shift Manager. You are the Unit 1 BOP. You are to perform Annunciator Response Procedure 1-RM-D8, 1-SW-RI-117 HIGH.
- When you have completed the actions to accomplish this Task, please inform me.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- A LBLOCA has occurred on Unit 1. Unit 2 is operating at stable power level of 100%.
- The Operating Team is currently performing 1-E-1, Loss of Reactor or Secondary Coolant.
- 1-E-0, Attachment 1, Step 9d, Check OSRS pump –NOT CAVITATING has just been completed.
- Annunciator 1-RM-C8, RS/SW HX D ALERT/FAILURE, and 1-RM-D8, 1-SW-RI-117 HIGH have just been received.

Initiating Cues

- I am the Nuclear Shift Manager. You are the Unit 1 BOP. You are to perform Annunciator Response Procedure 1-RM-D8, 1-SW-RI-117 HIGH.
- When you have completed the actions to accomplish this Task, please inform me.

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U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Simulator Job Performance Measure WE14EA1.3

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

Perform 1-E-0, Attachment 4

K/A: WE14EA1.3 Ability to operate and/ or monitor the following as they apply to the (High Containment Pressure): Desired operating results during abnormal and emergency situations. (3.3 / 3.8)

Applicability	Estimated Time	Actual Time
RO/SRO(I)/SRO(U)	5 Minutes	_____ Minutes

Conditions

- Task is to be PERFORMED in the simulator.
- A LBLOCA has occurred on Unit 1 coincident with a Loss of Offsite Power.

Standards

- Completion of 1-E-0, Attachment 4 – CLS Component Verification.

Initiating Cues

- A LBLOCA has occurred on Unit 1 coincident with a Loss of Offsite Power.
- 1-E-0, Attachment 1, Step 8e, is directing the initiation of Attachment 4 to perform CLS component verification.
- I am the Nuclear Shift Manager. You are the Unit 1 BOP. You are to perform 1-E-0, Attachment 4 – CLS Component Verification.
- When you have completed the actions to accomplish this Task, please inform me.

Terminating Cues

- Report of completion of 1-E-0, Attachment 4 – CLS Component Verification.

Procedures

- 1-E-0, Attachment 4 – CLS Component Verification (Rev 71)

Tools and Equipment

- None

Safety Considerations

- None

Simulator Setup

- Call up 100% power IC and initialize.
- Enter Malfunctions:
    - EL01, Loss of Offsite Power, **Trigger 1**
    - RC0101, RCS Cold Leg A Pipe Rupture; final value = 50, **Trigger 1**
    - CA03, Disable IA-TV-101A Auto Closure; Active
    - CC15, Disable CC-TV-105C Auto Closure; Active
    - CW23, Disable CW-MOV-106C Auto Closed; Active
    - RM1002, Disable RM-TV-100B Auto Close; Active
    - SW1304, Disable SW-P-5D Auto Start; Active
- Place Simulator in Run. Insert **Trigger 1**.
- Perform 1-E-0 actions up to Attachment 1, Step 8e.
- Place Simulator in Freeze until JPM performance.

Directions to the Applicant

- I am the Nuclear Shift Manager. You are the Unit 1 BOP.
- A LBLOCA has occurred on Unit 1 coincident with a Loss of Offsite Power.
- You are to perform 1-E-0, Attachment 4 – CLS Component Verification.
- When you have completed the actions necessary to accomplish this task, please inform me.

Notes

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- *An additional instructor may be needed to silence alarms for the examinee.*
- **START TIME:**

STEP 1: Check Phase II and Phase III Containment Isolation Valves are closed. **STANDARD:** a) Locates to the Vertical Board. b) Checks Phase II and Phase III Containment Isolation Valves Closed / Green lights lit. **c) For valves out of position, closes the valves:** • **1-RM-TV-100B** • **1-CC-TV-105C** • **1-IA-TV-101A** d) Applicant annotates Attachment. Applicant may also place “pink magnets” on valves out of position. **EVALUATOR’S NOTE:** **If asked**: Acknowledge valves out of position. Tell Applicant to continue performing attachment. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2: Checks Containment Air Recirculation Fans tripped. **STANDARD:** a) Locates to the Unit 1 Ventilation Panel. b) Checks Containment Air Recirculation Fans OFF (green & amber lights lit): • 1-VS-F-1A • 1-VS-F-1B **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 3: Checks Recirculation Spray Service Water in operation. **STANDARD:** a) Locates to the Bench Board. b) Checks SW MOVs for all RSHXs Open / Red Lights lit: • 1-SW-MOV-105A through -105D • 1-SW-MOV-104A through -104D • 1-SW-MOV-103A through -103D c) Checks SW flow by observing SW flow through 1-SW-FI-106A through-106D between 6,000and 12,500 gpm. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 4: Checks RSHX SW RM Sample Pumps running. **STANDARD:** a) Locates to Radiation Monitoring Panel. b) Checks RSHX SW RM Sample Pumps running (*time delayed – 1 minute*). Red lights lit. **c) For non-running pump (1-SW-P-5D), starts pump.** d) Applicant annotates pump start on Attachment. Applicant may also place "pink magnet" on 1-SW-P-5D control switch. **EVALUATOR'S NOTE:** **If asked**: Acknowledge pump not running as expected. Tell Applicant to continue performing attachment. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 5: Checks RSHX RM Pump No-Flow annunciators clear. **STANDARD:** a) Locates to Vertical Board. b) Verifies all RSHX SW RM Pump alarms clear. c) Acknowledges NOTE that CLS must be reset to allow RM pumps to be secured from MCR. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 6: Checks Containment Spray and Recirc Spray Systems valve positions. **STANDARD:** a) Locates to Bench Board. b) Checks CS and RS System Valves Open / Red lights lit. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 7: Checks Circulating and Service Water Systems isolation due to Hi-Hi CLS with LOOP. **STANDARD:** a) Recalls from Initial Conditions that a Loss of Offsite Power has also occurred. b) Checks CW isolation valves for Main Condenser Closed / Green lights lit. **c) For valve out of position, closes the valve:** • **1-CW-MOV-106C** d) Applicant annotates Attachment. Applicant may also place “pink magnet” on valve out of position. e) Checks SW isolation valves for BC and CC Heat Exchangers Closed / Green lights lit. **EVALUATOR’S NOTE:** **If asked**: Acknowledge valve out of position. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 8: Notify Nuclear Shift Manager (Evaluator) Status of Task. Applicant should report completion of task. Applicant should also notify the SM (Evaluator) of components found out of position and actions taken. **COMMENTS:** ** **JPM COMPLETE** **	_______ **SAT** _______ **UNSAT**

STOP TIME:

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- A LBLOCA has occurred on Unit 1 coincident with a Loss of Offsite Power.
- The Operating Team is currently performing 1-E-1, Loss of Reactor or Secondary Coolant.
- 1-E-0, Attachment 1, Step 8e, is directing the initiation of Attachment 4 to perform CLS component verification.

Initiating Cues

- I am the Nuclear Shift Manager. You are the Unit 1 BOP.
- A LBLOCA has occurred on Unit 1 coincident with a Loss of Offsite Power.
- You are to perform 1-E-0, Attachment 4 – CLS Component Verification.
- When you have completed the actions necessary to accomplish this task, please inform me.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- A LBLOCA has occurred on Unit 1 coincident with a Loss of Offsite Power.
- The Operating Team is currently performing 1-E-1, Loss of Reactor or Secondary Coolant.
- 1-E-0, Attachment 1, Step 8e, is directing the initiation of Attachment 4 to perform CLS component verification.

Initiating Cues

- I am the Nuclear Shift Manager. You are the Unit 1 BOP.
- A LBLOCA has occurred on Unit 1 coincident with a Loss of Offsite Power.
- You are to perform 1-E-0, Attachment 4 – CLS Component Verification.
- When you have completed the actions necessary to accomplish this task, please inform me.

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U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Simulator Job Performance Measure 013A4.01
[Alternate Path]

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

Transfer the SI System to the Cold Leg Recirculation Mode

K/A: 013A4.01 Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate. (4.5/4.8)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)/SRO(U)	10 Minutes	______ minutes

Conditions

- Task is to be PERFORMED in the simulator.
- A LBLOCA has occurred on Unit 1. The RWST is near the RMT swapover point.

Standards

- 1-ES-1.3, Transfer to Cold Leg Recirculation, Steps 1-5.

Initiating Cues

- 1-E-1, Loss of Reactor or Secondary Coolant, Step 21.
- Nuclear Shift Manager direction.

Terminating Cues

- Completion of 1-ES-1.3, Transfer to Cold Leg Recirculation, Steps 1-5.

Procedures

- 1-ES-1.3, Transfer to Cold Leg Recirculation. (Rev. 20)

Tools and Equipment	**Safety Considerations**
• None	• None

Simulator Setup

- Call up 100% power IC and initialize. Place simulator in RUN.
- Initiate the following annunciator overrides:
  V1AE2 (RMT in test mode) to OFF.
- Initiate the following switch overrides:
  CSRMTA_REFUEL to REFUEL,
  CSRMTB_REFUEL to REFUEL.
- Initiate malfunction for "A" loop cold leg rupture (RC0101).
- Perform E-0 & transition to 1-E-1. Perform 1-E-1 through Step 21, which checks for transition to ES-1.3.
- Two (2) LHSI pumps, two (2) HHSI pumps & all ISRS pumps, OSRS pumps and CS pumps should be running. HHSI pumps should be on redundant flowpath alignment and charging pump mini-flow recirc valves should be closed.
- When RWST level is 21% insert malfunction EL1201 – Loss of 480v Emergency Switchgear 1H. This will result in loss of power to 1-RS-P-1A (ISRS), 1-RS-P-2A (OSRS), and 1-SI-P-1A (LHSI). The red lights should remain lit, but there will be no amps indicated for these pumps.
- When RWST level is 20% (RWST LOW LEVEL alarm is LIT), freeze simulator for performance.

Directions to the Applicant

- I am the Nuclear Shift Manager. A Large-Break LOCA has occurred on Unit 1 and the 1H 480v emergency bus has been lost.
- RWST level is less than 20%.
- Here's a copy of 1-ES-1.3, Transfer to Cold Leg Recirculation. I need you to perform Steps 1-5.
- When you finish the actions necessary to accomplish this, please inform me.

Notes

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- *An additional instructor may be needed to silence and acknowledge alarms for the examinee.*
- **START TIME:**

STEP 1: CAUTIONS and NOTES Prior to Step 1. **STANDARD:** a) Reads caution that SI recirc flow to RCS must be maintained at all times. b) Reads caution that transfer to recirculation may cause high radiation in the Auxiliary Building. c) Notes that Steps 1 through 5 should be performed without delay and FRs should not be implemented before completion of these steps. d) Notes that if sump blockage or a complete loss of sump suction capability occurs, FRs should NOT be implemented until directed in Attachment 1, or in 1-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2: CHECK OR PLACE BOTH RMT MODE TRANSFER SWITCHES IN RMT. *(Step 1)* **STANDARD:** Verifies BOTH RMT Transfer Switches in RMT position. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 3: RESET SI. *(Step 2)* **STANDARD:** Depresses both SI Reset Pushbuttons on Benchboard 1-1. **EVALUATOR'S NOTE:** If SI previously reset, verification of SI reset by noting annunciators A-F-3 & 4 (SI Initiated) clear will suffice. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 4: CHECK SI RECIRC PHASE HEAT SINK. *(Step 3)* Check SW flow established to at least two RSHXs. *(Step 3a)* **STANDARD:** Checks the following flow indications for SW flow to at least two RS HXs: a) 1-SW-FI-106A (SW flow to "A" RSHX). b) 1-SW-FI-106B (SW flow to "B" RSHX). c) 1-SW-FI-106C (SW flow to "C" RSHX). d) 1-SW-FI-106D (SW flow to "D" RSHX). **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 5: Check AC emergency buses - ENERGIZED BY OFF-SITE POWER. *(Step 3b)* **STANDARD:** a) Checks "H" Bus voltage indicated (between 4000 and 4400 volts). b) Checks "H" Bus normal supply breaker, 15H8, closed (red light on & green off). c) Checks "J" Bus voltage indicated (between 4000 and 4400 volts). d) Checks "J" Bus normal supply breaker, 15J8, closed (red light on & green off). **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 6: Check RS pumps associated with RSHXs supplied by SW - AT LEAST TWO RUNNING. *(Step 3c)* **STANDARD:** Checks if the following pumps to determine if at least two are running: a) 1-RS-P-1A (associated w/ A RSHX) – breaker closed BUT NO AMPS INDICATED. b) 1-RS-P-1B (associated w/ B RSHX). c) 1-RS-P-2A (associated w/ C RSHX) – breaker closed BUT NO AMPS INDICATED. d) 1-RS-P-2B (associated w/ D RSHX). **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 7: CHECK LHSI PUMPS. *(Step 4)* LHSI pumps – BOTH RUNNING. *(Step 4a)* **STANDARD:** a) Checks 1-SI-P-1A breaker indication red light on BUT NO amps indicated. b) Checks 1-SI-P-1B breaker indication red light on and amps indicated. c) **Determines only one LHSI pump in service and secures 1-CH-P-1B and places in PTL. Verifies 1-CH-P-1A in PTL.** **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 8: ALIGNS SI SYSTEM FOR RECIRC. *(Step 5a & b)* **STANDARD:** a) Notes CAUTION that if suction source is lost to any SI or CS pump, the pump should be stopped. b) Checks 1-CH-MOV-1275A closed by observing green light on and red light off. c) Checks 1-CH-MOV-1275B closed by observing green light on and red light off. d) Checks 1-CH-MOV-1275C closed by observing green light on and red light off. e) Checks RWST level less than 13% on the following indicators: • 1-CS-LI-100A • 1-CS-LI-100B • 1-CS-LI-100C • 1-CS-LI-100D f) If RWST level not less than 13%, waits for level to reach 13%. **EVALUATOR'S NOTE:** If annunciators 1A-A2, 1A-B2, 1A-C2, and 1A-D2 are lit, the operator should identify auto RMT failure and proceed to Step 5c. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 9: Check Phase 1 - INITIATED. *(Step 5c(1) and RNO)* **STANDARD:** a) Checks Phase 1 White Status light NOT lit. (Goes to RNO) b) Pushes both RMT actuation pushbuttons for Train A. c) Pushes both RMT actuation pushbuttons for Train B. d) Verifies RMT not actuated and that valves must be manually aligned. **EVALUATOR'S NOTE:** • Phase 1 White Status light is not lit. • RMT actuation pushbuttons will not function when pushed. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 10: LHSI discharge to HHSI - OPEN. *(Step 5c(2))* **STANDARD:** a) Opens 1-SI-MOV-1863A by placing control switch to OPEN. b) Checks valve open by observing red light on & green off. **c) Opens 1-SI-MOV-1863B by placing control switch to OPEN.** d) Checks valve open by observing red light on & green off. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 11: LHSI Recirc valves - CLOSED. *(Step 5c(3))* **STANDARD:** a) Closes 1-SI-MOV-1885A by placing control switch to CLOSE. b) Checks 1-SI-MOV-1885A closed by observing green light on & red off. **c) Closes 1-SI-MOV-1885B by placing control switch to CLOSE.** d) Checks 1-SI-MOV-1885B closed by observing green light on & red off. **e) Closes 1-SI-MOV-1885C by placing control switch to CLOSE.** f) Checks 1-SI-MOV-1885C closed by observing green light on & red off. g) Closes 1-SI-MOV-1885D by placing control switch to CLOSE. h) Checks 1-SI-MOV-1885D closed by observing green light on & red off. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 12: Check Phase 2 - INITIATED. *(Step 5d(1) and RNO)* **STANDARD:** a) Checks Phase 2 Amber Status light NOT LIT. (Goes to RNO). b) Ensures 1 minute elapsed since RMT *should have actuated* prior to continuing. **EVALUATOR'S NOTE:** **If asked**: 1 minute has elapsed since RMT alarms came in. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 13: LHSI suction from sump - OPEN. *(Step 5d(2))* **STANDARD:** a) Opens 1-SI-MOV-1860A by placing control switch to OPEN. b) Checks 1-SI-MOV-1860A open by observing red light on & green off. **c) Opens 1-SI-MOV-1860B by placing control switch to OPEN.** d) Checks 1-SI-MOV-1860B open by observing red light on & green off. **EVALUATOR'S NOTE:** These valves take approximately 1 minute to open. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 14: LHSI suction from RWST - CLOSED. *(Step 5d(3))* **STANDARD:** a) Closes 1-SI-MOV-1862A by placing control switch to CLOSE. b) Checks 1-SI-MOV-1862A closed by observing green light on & red off. **c) Closes 1-SI-MOV-1862B by placing control switch to CLOSE.** d) Checks 1-SI-MOV-1862B closed by observing green light on & red off. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 15: CHG pump suction from RWST valves - CLOSED. *(Step 5d(4))* **STANDARD:** **a) Closes 1-CH-MOV-1115B by placing control switch to CLOSE.** b) Checks 1-CH-MOV-1115B closed by observing green light on & red off. **c) Closes 1-CH-MOV-1115D by placing control switch to CLOSE.** d) Checks 1-CH-MOV-1115D closed by observing green light on & red off. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 16: Check recirculation flow - ESTABLISHED. *(Step 5e)* **STANDARD:** Verifies SI flow to the Rx core via cold leg flowpath by checking the following flow instrumentation: • 1-SI-FI-1945 (A LHSI FT) – 0 gpm, • 1-SI-FI-1946 (B LHSI FT) - ~3400 gpm, • 1-SI-FI-1961 (A Loop FT) - ~ 150 gpm, • 1-SI-FI-1962 (B Loop FT) - ~ 150 gpm, • 1-SI-FI-1963 (C Loop FT) - ~ 150 gpm, • 1-SI-FI-1943 (Total flow normal hdr) - ~440 gpm, • 1-SI-FI-1943A (Total flow normal hdr) - ~ 440 gpm, • 1-SI-FI-1940 (Total flow alt hdr) – 0 gpm, • 1-SI-FI-1940A (Total flow alt hdr) – 0 gpm. **EVALUATOR'S NOTE:** No flow should be noted on the "A" LHSI and Alternate Header flow indicators since 1-SI-P-1A has no power and only the Charging Pump aligned to the Normal Header is running. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 17: REPORTS TO NUCLEAR SHIFT MANAGER (EVALUATOR). **STANDARD:** Verbal status report made that cold leg recirculation established. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STOP TIME:

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initiating Cues

- A Large-Break LOCA has occurred on Unit 1. RWST level is less than 20%.
- The team has just transitioned to 1-ES-1.3.

Directions to the Applicant

- I am the Nuclear Shift Manager. A Large-Break LOCA has occurred on Unit 1 and the 1H 480v emergency bus has been lost.
- RWST level is less than 20%.
- Here's a copy of 1-ES-1.3, Transfer to Cold Leg Recirculation. I need you to perform Steps 1-5.
- When you finish the actions necessary to accomplish this, please inform me.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initiating Cues

- A Large-Break LOCA has occurred on Unit 1. RWST level is less than 20%.
- The team has just transitioned to 1-ES-1.3.

Directions to the Applicant

- I am the Nuclear Shift Manager. A Large-Break LOCA has occurred on Unit 1 and the 1H 480v emergency bus has been lost.
- RWST level is less than 20%.
- Here's a copy of 1-ES-1.3, Transfer to Cold Leg Recirculation. I need you to perform Steps 1-5.
- When you finish the actions necessary to accomplish this, please inform me.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Simulator Job Performance Measure 005A2.03

Applicant_______________________________ Start Time__________________

Examiner______________________________

Date __________________________________ Stop Time__________________

Title

RESPOND TO A LOSS OF DECAY HEAT REMOVAL

K/A: 005A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR pump / motor malfunction. (2.9 / 3.1)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)	15 minutes	______minutes

Conditions

- Task is to be PERFORMED in the simulator.
- 1-RH-P-1A was in service on "A" RHR HX. We just received annunciator 1B-G6, RHR HX LO FLOW, and 1-RH-P-1A has tripped.

Standards

- 1-RH-P-1B started, RHR flow restored, and heat sink restored to the in service RHR HX in accordance with 1-AP-27.00, Loss of Decay Heat Removal Capability.

Initiating Cues

- 1-RH-P-1A was in service on "A" RHR HX. We just received annunciator 1B-G6, RHR HX LO FLOW, and 1-RH-P-1A has tripped.

Terminating Cues

- 1-AP-27.00, Loss of Decay Heat Removal Capability, step 14 completed.

Procedures

- 1-AP-27.00, Loss of Decay Heat Removal Capability (Rev. 26).

Tools and Equipment
- None

Safety Considerations
- None

Simulator Setup

- Call up RHR IC and initialize. Place simulator in RUN.
- Verify "A" RHR pump running and "B" in AUTO.
- Verify 1-RH-FCV-1605 in auto with flow rate set at 3400 gpm.
- Implement malfunction for over-current trip of 1-RH-P-1A (RH0501) and allow annunciators 1B-G6 & 1B-G7 to alarm.
- Close 1-CC-TV-109A.
- Place simulator in FREEZE until ready to perform JPM.
- **Simulator Operator Note**: Place RED magnets on 1-RH-MOV-1700, 1701 and 1720A red bulbs. Place green magnets on SI accumulator green bulbs and verify magnets are correct for SI system for CSD. Place a white magnet and green arrow on the Pressurizer level cold cal channel 1-RC-LI-1460 and make sure the trend recorder is set for this channel.

Directions to the Applicant

- I am the Nuclear Shift Manager and you are the Unit RO. The unit has been operating on RHR.
- 1-RH-P-1A, “A” RHR pump was in service on "A" RHR heat exchanger at 3400 gpm.
- We just received annunciator 1B-G6, RHR HX LO FLOW, and 1-RH-P-1A has tripped following a lightning strike near the station.
- Here is a copy of 1-AP-27.00, Loss of Decay Heat Removal Capability. I need you to perform the necessary steps to address this condition and restore decay heat removal.
- No RCS draindown evolutions are in progress and inventory is stable.
- When you complete 1-AP-27.00, please inform me.

Notes

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- *An additional instructor may be needed to silence and acknowledge alarms for the examinee.*
- **START TIME:**

STEP 1: ***CAUTIONS and NOTE PRIOR TO STEP 1*** **CAUTION:** • Loss of RHR due to a total loss of IA is addressed by 0-AP-40.00, NON-RECOVERABLE LOSS OF IA. • Loss of RHR due to a total loss of AC Power is addressed by 1-AP-10.27, LOSS OF ALL AC POWER WHILE ON RHR. • Loss of RHR may cause CTMT radiological and heat stress conditions to degrade. Local actions in CTMT should be coordinated with HP. • During solid plant operation, inadvertent actuation of the OPMS may occur if letdown is isolated. • If RCS boiling occurs, non-essential personnel should be evacuated from CTMT. **NOTE:** EPIPs may be applicable. **STANDARD:** • Acknowledges note and acknowledges cautions and recognizes that a total loss of IA or AC Power is not occurring. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 2: CHECK RCS INVENTORY - LOWERING. (*Step 1)* • PRZR level - LOWERING • Standpipe level - LOWERING • Reactor cavity level - LOWERING • RCS Narrow Range level - LOWERING • CTMT sump level - RISING • Makeup rate - RISING • PRT level, pressure, or temperature - RISING • PDTT level - RISING • RWST level - RISING **STANDARD:** • Notes that there are no draindown evolutions in progress and inventory is stable based on the directions given. • Checks Containment Sump level (1-DA-LI-100) is stable and not rising. • Checks PRT conditions (level, LI-1-470; pressure, PI-1-472; and temperature, TI-1-471) are stable and not rising. • Checks PDTT level (1-DG-LI-107) is stable and not rising. • Checks RWST level stable. • Determines that RCS inventory is NOT lowering and performs RNO to transition to procedure STEP 4. **EVALUATOR'S NOTE:** • **If asked**: All indications are as you see them. • **If asked**: No personnel are in Containment. • **If asked**: Cavity is not flooded up. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 3: CHECK RHR PUMP - ONE RUNNING. *(Step 4)* **STANDARD:** • Checks that no RHR pumps are running by observing zero amps indicated and 1-RH-P-1A has red and amber lights lit, 1-RH-P-1B has green light lit. Goes to RNO. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

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STEP 4:

IF Emergency Bus power is available, THEN do the following: *(Step 4 RNO)*

a) Manually close RH control valves:
- 1-RH-FCV-1605
- 1-RH-HCV-1758
- 1-RH-HCV-1142

b) Start one RHR pump.

c) Adjust RH control valves to return flow to pre-event rate:
- 1-RH-FCV-1605
- 1-RH-HCV-1758
- 1-RH-HCV-1142

d) IF an RHR pump can NOT be started, THEN GO TO Step 16.

IF RHR pump NOT running due to loss of Emergency Bus power, THEN do the following:
a) Check initiated or initiate 1-AP-10.07, LOSS OF UNIT 1 POWER.
b) IF ELAP in progress, THEN GO TO 1-AP-10.27, Loss of All AC Power While on RHR.
c) GO TO Step 16.

STANDARD:

- **Places 1-RH-FCV-1605 in manual and closes valve.**
- Notes setpoint on ten turn pot for 1-RH-HCV-1758 (9.8) and then **closes 1-RH-HCV-1758 using ten turn pot**.
- **Closes 1-RH-HCV-1142 using ten turn pot.**
- **Starts 1-RH-P-1B by taking control switch to the start position and verifying amps are indicated.**
- **Manually opens 1-RH-FCV-1605 using controller pushbuttons until flow is approximately 3400 gpm.**
- **Places ten turn pot for 1-RH-FCV-1758 at pre-event setpoint of 9.8**.
- **Fully opens 1-RH-HCV-1142.**

COMMENTS:

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_______ **SAT**

_______ **UNSAT**

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STEP 5: CHECK RHR FLOW - INDICATED ON RHR SYS FLOW. *(Step 5)* • 1-RH-FI-1605 **STANDARD:** • Checks RHR flow at approximately 3400 gpm on 1-RH-FI-1605. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 6: CHECK RHR PUMP – VORTEXING *(Step 6)* • Flow indication on 1-RH-FI-1605 - OSCILLATING • Amperage indication - OSCILLATING **STANDARD:** • Checks flow steady on 1-RH-FI-1605 and amps steady for 1-RH-P-1B. • Goes to step 6 RNO and transitions to Step 12. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 7: CHECK RHR HEAT SINK: *(Step 12)* a) Flow on 1-RH-FI-1605 - NORMAL b) CC to RHR HX 1) In-Service RHR HX CC Outlet HDR Flow - NORMAL • 1-CC-FI-110A OR • 1-CC-FI-110B **STANDARD:** • Checks flow on 1-RH-FI-1605 indicating normal about 3400 gpm. • Checks CC to RHR HX on 1-CC-FI-110A NOT normal at zero gpm. Goes to RNO. **EVALUATOR'S NOTE:** • **If asked**, pre-event flow on 1-CC-FI-110A was 4700 gpm and temperature on 1-CC-TI-109A was approximately 88 degrees. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

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STEP 8:

1) Check opened or open 1-CC-TV-109A or 1-CC-TV-109B. *(Step 12b.1 RNO)*

IF TV can NOT be opened due to a localized loss of IA, THEN locally open IAW 0-FCA-16.00, LOCAL OPERATION OF AIR OPERATED VALVES.

IF the in-service RHR HX TV can NOT be opened, THEN place the other RHR HX in service IAW Attachment 11.

IF CC flow can NOT be established to either RHR HX, THEN do the following:
a. Evaluate initiating 1-AP-15.00, LOSS OF COMPONENT COOLING.
b. GO TO Step 16.

2) RHR HX CC Outlet HDR TEMP - NORMAL
• 1-CC-TI-109A
OR
• 1-CC-TI-109B

STANDARD:

- **Opens 1-CC-TV-109A to restore CC flow thru "A" RHR HX.**
- Monitors "A" RHR HX CC outlet temperature to verify temperatures are normal.

EVALUATOR'S NOTE:

- **If asked**: Pre-event flow on 1-CC-FI-110A was 4700 gpm and temperature on 1-CC-TI-109A was approximately 88 degrees.

COMMENTS:

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_______ **SAT**

_______ **UNSAT**

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STEP 9:

CHECK RCS TEMPERATURE - STABLE OR LOWERING. *(Step 13)*

STANDARD:

- Checks RCS temperature and/or RHR temperature stable or lowering.

EVALUATOR'S NOTE: Candidate may adjust 1-RH-HCV-1758 to reduce temperature.

COMMENTS:

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_______ **SAT**

_______ **UNSAT**

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STEP 10:

RETURN TO PROCEDURE IN EFFECT. *(Step 14)*

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_______ **SAT**

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STANDARD: • Operator Exits 1-AP-27.00. **COMMENTS:**	_______ **UNSAT**
STEP 11: REPORT TO NUCLEAR SHIFT MANAGER (EVALUATOR). **STANDARD:** Verbal status report that 1-AP-27.00, Loss of Decay Heat Removal Capability is complete and Decay Heat Removal has been restored. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STOP TIME:

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- 1-RH-P-1A was in service on "A" RHR HX. We just received annunciator 1B-G6, RHR HX LO FLOW, and 1-RH-P-1A has tripped.

Initiating Cues

- I am the Nuclear Shift Manager and you are the Unit RO. The unit has been operating on RHR.
- 1-RH-P-1A, “A” RHR pump was in service on "A" RHR heat exchanger at 3400 gpm.
- We just received annunciator 1B-G6, RHR HX LO FLOW, and 1-RH-P-1A has tripped following a lightning strike near the station.
- Here is a copy of 1-AP-27.00, Loss of Decay Heat Removal Capability. I need you to perform the necessary steps to address this condition and restore decay heat removal.
- No RCS draindown evolutions are in progress and inventory is stable.
- When you complete 1-AP-27.00, please inform me.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- 1-RH-P-1A was in service on "A" RHR HX. We just received annunciator 1B-G6, RHR HX LO FLOW, and 1-RH-P-1A has tripped.

Initiating Cues

- I am the Nuclear Shift Manager and you are the Unit RO. The unit has been operating on RHR.
- 1-RH-P-1A, “A” RHR pump was in service on "A" RHR heat exchanger at 3400 gpm.
- We just received annunciator 1B-G6, RHR HX LO FLOW, and 1-RH-P-1A has tripped following a lightning strike near the station.
- Here is a copy of 1-AP-27.00, Loss of Decay Heat Removal Capability. I need you to perform the necessary steps to address this condition and restore decay heat removal.
- No RCS draindown evolutions are in progress and inventory is stable.
- When you complete 1-AP-27.00, please inform me.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Simulator Job Performance Measure 086A4.02
[Alternate Path]

Applicant________________________________ Start Time________________________________

Examiner ________________________________

Date ________________________________ Stop Time________________________________

Task

Bypass Containment Detection on 0-FP-MON-IMS-1.

K/A: 086A4.02 Ability to manually operate and / or monitor in the control room: Fire detection panels. (3.5 / 3.5)

Applicability	**Est. Completion Time**	**Actual Time**
RO/SRO(I)	5 Minutes	_____ Minutes

Conditions

- Task is to be performed in the simulator.
- A small break LOCA is in progress on Unit 1.

Standards

- The operator will bypass the Unit 1 Containment Fire Detection alarms on fire protection panel 0-FP-MON-IMS-1 IAW posted placard.

Initiating Cues

- A small break LOCA is in progress on Unit 1.
- Shift Manager direction to address the fire panel alarms.

Terminating Cues

- Fire panel alarms in containment bypassed for 10 minutes.

Procedures

- None

Tools and Equipment	**Safety Considerations**
• None	• None

Simulator Setup

- Recall IC-1 (100% power).
- Place simulator in RUN.
- Insert RC04 RCS LEAK NONISOLABLE (1200GPM) at 70% Severity and let simulator run until first *FIRE* alarm is received.
- Place simulator in freeze.

Directions to the Applicant

- I am the Nuclear Shift Manager. You are the Unit 1 BOP.
- There is a small break LOCA in progress on Unit 1.
- You are to bypass the Unit 1 Containment Fire Detection alarms on fire protection panel 0-FP-MON-IMS-1.
- When you finish the actions necessary to accomplish this, please inform me.

Notes

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- *An additional instructor may be needed to silence alarms for the examinee.*
- **START TIME: ________**

Step 1. Proceeds to Fire Protection touch screen panel 0-FP-MON-IMS-1. **STANDARD:** (a) Verifies the only fire detectors in alarm are in the Unit 1 containment building. (b) Returns to main screen. **(c) Touches CONTAINMENT DETECTION button on lower right portion of screen.** **(d) Selects UNIT 1 CONTAINMENT DETECTION ENABLED button.** **(e) Selects DISABLE button.** **(f) Acknowledges TROUBLE alarms received due to bypassing the Unit 1 Containment fire detectors.** **(g) Acknowledges any additional fire alarms that were delayed due to computer polling time lag.** (h) Returns the panel to the home screen. **EVALUATOR'S NOTE:** A placard is posted above 0-FP-MON-IMS-1 giving instructions on how to bypass the Containment Fire Detection alarms. The Applicant should use this as reference. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
Step 2. Reports to Shift Manager (Evaluator) that alarms in Containment have been bypassed for 10 minutes. **STOP TIME: ___________** **Evaluator's Note** **Note:** The fire protection system uses a polling method where the computer polls one detector at a time in a predetermined order. It is possible that a detector is in alarm but there is no indication on the IMS because that point has not yet been polled. If during this period, the operator bypasses the containment alarms and acknowledges all current fire alarms, more alarms can come in because they have not been polled yet. For this reason it is expected that the operator will have to acknowledge the alarm panel more than once during response to a HELB in containment. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- There is a small break LOCA in progress on Unit 1.

Initiating Cues

- I am the Nuclear Shift Manager. You are the Unit 1 BOP.
- There is a small break LOCA in progress on Unit 1.
- You are to bypass the Unit 1 Containment Fire Detection alarms on fire protection panel 0-FP-MON-IMS-1.
- When you finish the actions necessary to accomplish this, please inform me.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

<u>Initial Conditions</u>

- There is a small break LOCA in progress on Unit 1.

<u>Initiating Cues</u>

- I am the Nuclear Shift Manager. You are the Unit 1 BOP.
- There is a small break LOCA in progress on Unit 1.
- You are to bypass the Unit 1 Containment Fire Detection alarms on fire protection panel 0-FP-MON-IMS-1.
- When you finish the actions necessary to accomplish this, please inform me.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Simulator Job Performance Measure 016A2.01
[Alternate Path]

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

RESPOND TO A SECONDARY SYSTEM TRANSIENT.

K/A: 016A2.01, Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector Failure. (3.0/3.1)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)	5 Minutes	_____ Minutes

Conditions

- Task is to be PERFORMED in the simulator.
- Unit is operating at 90% power with all systems normal and in automatic.

Standards

- 0-AP-53.00, Loss of Vital Instrumentation / Controls, Revision 21.

Initiating Cues

- Nuclear Shift Manager direction.

Terminating Cues

- Report received that unit is stabilized and 0-AP-53.00 is completed.

Procedures

- 0-AP-53.00, Loss of Vital Instrumentation / Controls, Revision 21.

Tools and Equipment

- None

Safety Considerations

- None

Simulator Setup

- Recall 90% power IC and initialize. Place simulator in RUN.
- Enter Malfunction, MS1308 – SG Pressure Transmitter MS-PT-1495 Failure ("C" S/G Ch. 3 PT), Final Value = 1, 10 Sec Ramp, **Trigger 1**.
- Enter Controller Override, FW, FWFC498F_MANUAL, OFF, Active.
- Enter Controller Override, FW, FWFC498F_AUTO, ON, Active.
- Freeze Simulator until ready.

Initial Conditions

- You are the Unit 1 RO and I am the Nuclear Shift Manager. The unit is operating at 90% power with all systems in automatic.

Initiating Cues

- You are to respond to a plant transient.
- When you finish the actions necessary to stabilize the unit, please inform me.

Notes

This failure will cause "C" SG Channel III Main Steam Pressure Transmitter to fail high, causing the selected feed flow channel to go low. "C" FRV will open, causing "C" SG NR level to rise. The "C" FRV will fail to shift to manual control, requiring the operator to throttle flow with the Feedwater isolation MOV to prevent a turbine trip/reactor trip on SG NR high-high level.

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- *An additional instructor may be needed to silence alarms for the examinee.*
- **START TIME_______**

STEP 1 CHECK REDUNDANT INSTRUMENT CHANNEL(S) INDICATION – NORMAL. *(0-AP-53.00, Step 1)* **STANDARD:** a) Identifies the affected SG using Steam Flow/Feed Flow mismatch annunciators, “C” SG Level trend, and/or “C” FRV demand. b) Identifies Channel II and IV “C” SG Pressure Normal; Channel III failed high. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2 PLACE AFFECTED CONTROL(S)/COMPONENT(S) IN MANUAL CONTROL AND STABILIZE PARAMETER USING REDUNDANT INDICATION. *(0-AP-53.00, Step 2)* **STANDARD:** a) Attempts to place “C” FRV in manual. Controller will not shift to manual. **b) Places 1-FW-MOV-154C, SG C FW ISOL, in Close.** c) Continues to hold control switch in close until “C” SG Feed flow is less than Steam flow, and “C” SG Level begins to trend towards 44%. **EVALUATOR’S NOTE:** **JPM Failure Criteria**. An automatic Turbine/Reactor trip due to untimely Applicant response to the event is considered a JPM Failure. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 3 Reports completion of Immediate Steps of 0-AP-53.00 to Shift manager (Evaluator). **STANDARD:** Reports “C” SG Feed flow is less than Steam Flow and NR level trending to Normal. **COMMENTS:** ****END OF JPM****	_______ **SAT** _______ **UNSAT**

STOP TIME:

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Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- You are the Unit 1 RO and I am the Nuclear Shift Manager. The unit is operating at 90% power with all systems in automatic.

Initiating Cues

- You are to respond to a plant transient.
- When you finish the actions necessary to stabilize the unit, please inform me.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- You are the Unit 1 RO and I am the Nuclear Shift Manager. The unit is operating at 90% power with all systems in automatic.

Initiating Cues

- You are to respond to a plant transient.
- When you finish the actions necessary to stabilize the unit, please inform me.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Simulator Job Performance Measure 062.A4.01

Applicant________________________________ Start Time_________________________________

Examiner _______________________________

Date ___________________________________ Stop Time_________________________________

Task

SYNCHRONIZE AND TRANSFER ELECTRICAL POWER SYSTEMS.

K/A: 062A4.01 Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard). (3.3 / 3.1)

Applicability	Est. Completion Time	Actual Time
RO	4 Minutes	_____ Minutes

Conditions

- Task is to be PERFORMED in the Simulator.
- Unit 1 is stable at 25% - 30% power with feed control in automatic.

Standards

- 1-GOP-1.5, Unit Startup, 2% Reactor Power to Max Allowable Power, Step 5.3.43.

Initiating Cues

- 1-GOP-1.5, Unit Startup, 2% Reactor Power to Max Allowable Power, Step 5.3.43.
- Shift Manager direction.

Terminating Cues

- Completion of 1-GOP-1.5, Attachment 5.

Procedures

- 1-GOP-1.5, Unit Startup, 2% Reactor Power to Max Allowable Power, Attachment 5.

Tools and Equipment	Safety Considerations
• None	• None

Simulator Setup

- Call up 25% power IC and initialize. Place simulator in RUN.
- Sync and close breakers ACB-15A1, ACB-15B1 & ACB-15C1. Then open breakers ACB-15A2, ACB-15B2 & ACB-15C2.

<u>Directions to the Applicant</u>:

- You are the Unit RO and I am the Shift Manager.
- Unit 1 startup is on standby for the 30% power chemistry hold.
- Feed control is in automatic.
- The next action is to transfer Station Electrical Service from Reserve to Normal in accordance with 1-GOP-1.5, Attachment 5.
- Here is 1-GOP-1.5, Attachment 5. I want you to place the Station Service buses on normal supply.
- An Alternate Load Shed alignment is NOT in effect.
- When you finish the actions necessary to accomplish this, please inform me.

<u>Notes</u>

- This JPM may be pre-briefed at the discretion of the NRC Chief Examiner.

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- Critical step sequencing requirements: Buses (A, B & C) may be done in any order; steps should be sequenced 1-2-4-5 (bus A), 6-7-9-10 (bus B), 11-12-15-16 (bus C).
- **START TIME:______________**

STEP 1. Transfer Bus 1A to Station Service. *(Att. 5, Step 1)* **STANDARD** (a) Acknowledges NOTE that sync-volts incoming and running should be within ± 5 volts and how both can be adjusted. **(b) Inserts sync key into sync switch, CS-25-15A2.**(*Step 1a*) **(c) Turns sync switch to the ON position.**(*Step 1b*) (d) Checks sync meter response indicates incoming and running voltages are within ±5 volts.(*Step 1c*) **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2. Transfer Bus 1A to Station Service. *(Att. 5, Step 1; continued)* **STANDARDS** **(a) Places control switch for ACB-15A2 to the CLOSE position.**(*Step 1e*) (b) Verifies closed indication (red light on - green light off) received. **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENTS:**	_______ **SAT** _______ **UNSAT**

Step 3. Transfer Bus 1A to Station Service. *(Att. 5, Step 1; continued)* **STANDARD** Verifies amps indicated on Bus 1A amperage meter.(*Step 1f*) **COMMENTS:**	_______ **SAT** _______ **UNSAT**
Step 4. Transfer Bus 1A to Station Service. *(Att. 5, Step 1; continued)* **STANDARD** **(a) Places control switch for ACB-15A1 to the OPEN position.**(*Step 1h*) (b) Verifies open indication (green light on - red light off) received. **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENTS:**	_______ **SAT** _______ **UNSAT**
Step 5. Transfer Bus 1A to Station Service. *(Att. 5, Step 1; continued)* **STANDARD** **(a) Turns off the sync switch (CS-25-15A2)** (*Step 1i*) **(b) Removes sync key.**(*Step 1j*) **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENTS:**	_______ **SAT** _______ **UNSAT**

Step 6. Transfer Bus 1B to Station Service. *(Att. 5, Step 2)* **STANDARD** (a) Acknowledges NOTE that sync-volts incoming and running should be within ± 5 volts and how both can be adjusted. **(b) Inserts sync key into sync switch, CS-25-15B2.**(*Step 2a*) **(c) Turns sync switch to the ON position.**(*Step 2b*) (d) Verifies sync meter response indicates incoming and running voltages are within ± 5 volts.(*Step 2c*) **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENTS:**	_______ **SAT** _______ **UNSAT**
Step 7. Transfer Bus 1B to Station Service. *(Att. 5, Step 2; continued)* **STANDARD** **(a) Places control switch for ACB-15B2 to the CLOSE position.**(*Step 2e*) (b) Verifies closed indication (red light on - green light off) received. **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENTS:**	_______ **SAT** _______ **UNSAT**
Step 8. Transfer Bus 1B to Station Service. *(Att. 5, Step 2; continued)* **STANDARD** Verifies amps indicated on Bus 1B amperage meter.(*Step 2f*) **COMMENTS:**	_______ **SAT** _______ **UNSAT**

Step 9. Transfer Bus 1B to Station Service. *(Att. 5, Step 2; continued)* **STANDARD** **(a) Places control switch for ACB-15B1 to the OPEN position.**(*Step 2h*) (b) Verifies open indication (green light on - red light off) received. **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENTS:**	_______ **SAT** _______ **UNSAT**
Step 10. Transfer Bus 1B to Station Service. *(Att. 5, Step 2; continued)* **STANDARD** **(a) Turns off the sync switch (CS-25-15B2)** (*Step 2i*) **(b) Removes sync key.**(*Step 2j*) **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENT:**	_______ **SAT** _______ **UNSAT**
Step 11. Transfer Bus 1C to Station Service. *(Att. 5, Step 3)* **STANDARD** (a) Acknowledges NOTE that sync-volts incoming and running should be within ± 5 volts and how both can be adjusted. **(b) Places sync key into sync switch, CS-25-15C2.**(*Step 3a*) **(c) Turns sync switch to the ON position.**(*Step 3b*) (d) Verifies sync meter response indicates incoming and running voltages are within ± 5 volts.(*Step 3c*) **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENT:**	_______ **SAT** _______ **UNSAT**

Step 12. Transfer Bus 1C to Station Service. *(Att. 5, Step 3; continued)* **STANDARD** **(a) Places control switch for ACB-15C2 to the CLOSE position.**(*Step 3e*) (b) Verifies closed indication (red light on - green light off) received. **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENT:**	_______ **SAT** _______ **UNSAT**
Step 13. Transfer Bus 1C to Station Service. *(Att. 5, Step 3; continued)* **STANDARD** Verifies annunciator 1K-F6 has cleared (not lit).(*Step 3f*) **COMMENT:**	_______ **SAT** _______ **UNSAT**
Step 14. Transfer Bus 1C to Station Service. *(Att. 5, Step 3; continued)* **STANDARD** Verifies amps indicated on Bus 1C amperage meter.(*Step 3g*) **COMMENTS:**	_______ **SAT** _______ **UNSAT**

Step 15. Transfer Bus 1C to Station Service. *(Att. 5, Step 3; continued)* **STANDARD** **(a) Places control switch for ACB-15C1 to the OPEN position.**(*Step 3i*) (b) Verifies open indication (green light on - red light off) received. **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENTS:**	_______ **SAT** _______ **UNSAT**
Step 16. Transfer Bus 1C to Station Service. *(Att. 5, Step 3; continued)* **STANDARD** **(a) Turns off the sync switch (CS-25-15A2)** (*Step 3j)* (b) Removes sync key.(*Step 3k*) **EVALUATOR'S NOTE:** ***This step is sequence critical.*** **COMMENTS:**	_______ **SAT** _______ **UNSAT**

Step 17. IF 0-OP-EP-004 is open for configuration control of Alternate Load Shed alignment, THEN using the open 0-OP-EP-004, restore Load Shed to normal alignment, while continuing in this procedure. Otherwise, enter N/A. *(Att. 5, Step 4)* **STANDARD** (a) Acknowledges NOTE before Step 4 that if an Alternate Load Shed alignment is in effect, 0-OP-EP-004, Load Shed, will be in the possession of Shift Supervision for the purpose of configuration control. (b) Determines that an Alternate Load Shed alignment is NOT in effect and enters "N/A" for Step 4. **EVALUATOR'S NOTE:** **If asked**: An Alternate Load Shed alignment is not in effect. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
Step 18. REPORTS TO SHIFT MANAGER (EVALUATOR). **STANDARD** Verbal status report made that action completed. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STOP TIME: ___________

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Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- Start up in progress on Unit 1.
- 1-GOP-1.5, Unit Startup, 2% Reactor Power to Max Allowable Power is complete up Step 5.3.43.

Initiating Cues

- You are the Unit RO and I am the Shift Manager.
- Unit 1 startup is on standby for the 30% power chemistry hold.
- Feed control is in automatic.
- The next action is to transfer Station Electrical Service from Reserve to Normal in accordance with 1-GOP-1.5, Attachment 5.
- Here is 1-GOP-1.5, Attachment 5. I want you to place the Station Service buses on normal supply.
- An Alternate Load Shed alignment is NOT in effect.
- When you finish the actions necessary to accomplish this, please inform me.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- Start up in progress on Unit 1.
- 1-GOP-1.5, Unit Startup, 2% Reactor Power to Max Allowable Power is complete up Step 5.3.43.

Initiating Cues

- You are the Unit RO and I am the Shift Manager.
- Unit 1 startup is on standby for the 30% power chemistry hold.
- Feed control is in automatic.
- The next action is to transfer Station Electrical Service from Reserve to Normal in accordance with 1-GOP-1.5, Attachment 5.
- Here is 1-GOP-1.5, Attachment 5. I want you to place the Station Service buses on normal supply.
- An Alternate Load Shed alignment is NOT in effect.
- When you finish the actions necessary to accomplish this, please inform me.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Job Performance Measure 076A2.01
[ALTERNATE PATH]

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

LOCALLY ISOLATE SERVICE WATER TO #3 MER DURING FLOODING.

K/A: 076A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS. (3.5/3.7)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)	15 Minutes	__________

Conditions

- Task is to be SIMULATED in the plant.
- Any plant mode/condition where a large SW leak in #3 MER has occurred.

Standards

- Flooding isolated to #3 MER as indicated by lowering water level in accordance with 0-AP-13.00 steps 38 and 39.

Initiating Cues

- 0-AP-13.00, Turbine Building or #3 MER Flooding, Step 38.
- Shift Manager direction.

Terminating Cues

- 0-AP-13.00, Turbine Building or #3 MER Flooding, Step 39 completed.

Procedures

- 0-AP-13.00, Turbine Building or #3 MER Flooding, Revision 29.

Tools and Equipment

- None

Safety Considerations

- Standard Personal Safety Equipment

Initial Conditions

- This task is to be SIMULATED. Do NOT turn switches, manipulate controls or reposition valves.
- I am the Shift Manager. There is a major Service Water leak in #3 MER.

Initiating Cues

- I need you to isolate Service Water to #3 MER in accordance with Steps 38 and 39 of 0-AP-13.00, Turbine Building or #3 MER Flooding.
- 1-SW-P-10B and 2-SW-P-10B have been secured.
- 1-VS-E-4A, 4B, and 4C have been secured.
- MER 3 Watertight Door has been checked closed.
- When you finish the actions necessary to accomplish this, please inform me.

Notes to the Evaluator.

- **Task briefing should occur in the pre-determined location.**
- This task is to be SIMULATED. Do NOT allow the Applicant to manipulate controls, operate switches or reposition valves.
- Critical step sequencing requirements: None.

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- **START TIME:**

STEP 1 ISOLATE SW TO MER 3: *(Step 38.a)* a) Fail 1-SW-263 closed by opening Circuit 8 on Lighting Panel 2T3 (located north of 2-FW-E-2A) **STANDARD:** a) Locates lighting panel 2T3 (located north of 2-FW-E-2A). **b) Opens circuit 8 on lighting panel 2-EP-LP-2T3.** **COMMENTS:**	**_______ SAT** **_______ UNSAT**
STEP 2 Check open or open 1-SW-500, SW Header Crosstie (MER 4) *(Step 38.b)* **STANDARD:** a) Proceeds to #4 MER. b) Locates manual valve 1-SW-500 (halfway across room under smoke detector). **c) Pulls pin from handwheel actuator.** **d) Opens or verifies open 1-SW-500 by rotating valve handwheel in the clockwise direction (verifying open) or counter-clockwise direction (opening).** **EVALUATOR'S NOTE:** **Tell Applicant:** Actual valve position in plant may be open or closed. Provide appropriate cue based on initial valve position. **COMMENTS:**	**_______ SAT** **_______ UNSAT**

STEP 3 Close 2-SW-476, Water Box 2C Isol (MER 4). *(Step 38c)* **STANDARD:** a) Locates 2-SW-476 (in #4 MER 2/3 of the way across the room on the right hand side). b) **Pulls pin from handwheel actuator**. c) **Closes 2-SW-476 by rotating the handwheel clockwise until position indicator points to CLOSED**. **<u>EVALUATOR'S NOTE:</u>** **Tell Applicant:** As the handwheel is rotated in the clockwise direction, the position indication pointer rotates from the OPEN to the CLOSED position. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 4 Close 2-SW-478, SW Header Crosstie. (MER 4) *(Step 38.d)* **STANDARD:** a) Locates 2-SW-478 (in #4 MER 2/3 of the way across the room on the right hand side). b) **Pulls pin from handwheel actuator**. **c) Closes or verifies closed 1-SW-478 by rotating valve handwheel in the clockwise direction.** **<u>EVALUATOR'S NOTE:</u>** **Tell Applicant:** Provide appropriate cue for valve operation based on actual in plant valve position. **Safety concern: Trainee does not have to crawl across pipes to check the valve labels at the east end of the #4 MER. He can identify which valve label he is looking at, and the evaluator can state the label reads "2-SW-478" if the correct label is identified. The trainee can then describe the required actions to complete valve manipulation from the west end of #4 MER.** . **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 5 Secure Chemical Injection to MER 3 SW Supply Header IAW 0-OP-SW-006, MER 3 and MER 4 Service Water Chemical Injection Operation. *(Step 38.e)* **STANDARD:** Notifies Shift Manager (Evaluator) that Chemical Injection to MER 3 SW Supply Header needs to be secured IAW 0-OP-SW-006. **EVALUATOR'S NOTE:** **Tell Applicant:** Unit 1 Turbine Building Operator will secure chemical injection. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 6 CHECK WATER LEVEL IN MER 3 ON MER 4 GAUGE 2-PL-LI-201 – LOWERING *(Step 39)* **STANDARD:** a) Locates MER 3 level gauge 2-PL-LI-201 in MER 4. b) Checks that level in MER 3 is lowering. **EVALUATOR'S NOTE:** **Tell Applicant:** After the Applicant locates MER 3 level gauge, tell him the level is 50" H_2O and slowly rising. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 7 Secure CHG Pump SW Pumps: *(Step 39.a RNO)* • 1-SW-P-10A • 2-SW-P-10A **STANDARD:** a) Calls Unit 1 RO and directs securing of 1-SW-P-10A. b) Calls Unit 2 RO and directs securing of 2-SW-P-10A. **EVALUATOR'S NOTE:** **Tell Applicant:** When the Applicant has given the directions to secure 1/2-SW-P-10A, inform the Applicant that the pumps are secured. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 8 Close 1-SW-499, SW Header Crosstie. *(Step 39.b RNO)* **STANDARD:** a) Locates manual valve 1-SW-499 in MER 4. **b) Pulls pin from handwheel actuator.** c) **Closes or verifies closed 1-SW-499 by rotating valve handwheel in the clockwise direction until the needle points to CLOSED.** **EVALUATOR'S NOTE:** **Tell Applicant:** As the handwheel is rotated in the clockwise direction, the position indication pointer rotates from the OPEN to the CLOSED position **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 9 Close 1-SW-500, Water Box 1D Isol. *(Step 39.c RNO)* **STANDARD:** a) Locates manual valve 1-SW-500 (halfway across room under smoke detector). **b) Pulls pin from handwheel actuator.** **c) Closes 1-SW-500 by rotating valve handwheel in the clockwise direction.** **EVALUATOR'S NOTE:** **Tell Applicant:** As the handwheel is rotated in the clockwise direction, the position indication pointer rotates from the OPEN to the CLOSED position **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 10 REPORT TO SHIFT SUPERVISOR (EVALUATOR). **STANDARD:** Verbal status report that steps 38 and 39 of 1-AP-13.00 are complete. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STOP TIME: ______________________

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Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- This task is to be SIMULATED. Do NOT turn switches, manipulate controls or reposition valves.
- I am the Shift Manager. There is a major Service Water leak in #3 MER.

Initiating Cues

- I need you to isolate Service Water to #3 MER in accordance with Steps 38 and 39 of 0-AP-13.00, Turbine Building or #3 MER Flooding.
- 1-SW-P-10B and 2-SW-P-10B have been secured.
- 1-VS-E-4A, 4B, and 4C have been secured.
- MER 3 Watertight Door has been checked closed.
- When you finish the actions necessary to accomplish this, please inform me.

Notes to the Evaluator.

- **Task briefing should occur in the pre-determined location.**
- This task is to be SIMULATED. Do NOT allow the Applicant to manipulate controls, operate switches or reposition valves.
- Critical step sequencing requirements: None.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- This task is to be SIMULATED. Do NOT turn switches, manipulate controls or reposition valves.
- I am the Shift Manager. There is a major Service Water leak in #3 MER.

Initiating Cues

- I need you to isolate Service Water to #3 MER in accordance with Steps 38 and 39 of 0-AP-13.00, Turbine Building or #3 MER Flooding.
- 1-SW-P-10B and 2-SW-P-10B have been secured.
- 1-VS-E-4A, 4B, and 4C have been secured.
- MER 3 Watertight Door has been checked closed.
- When you finish the actions necessary to accomplish this, please inform me.

TO BE GIVEN TO APPLICANT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
38. ___	ISOLATE SW TO MER 3:	
	☐ a) Fail 1-SW-263 closed by opening Circuit 8 on Lighting Panel 2T3 (located north of 2-FW-E-2A)	
	☐ b) Check open or open 1-SW-500, SW Header Crosstie (MER 4)	
	☐ c) Close 2-SW-476, Water Box 2C Isol (MER 4)	c) Do the following: ☐ 1) Check open or open 2-SW-533, located in MER 5. ☐ 2) Close 2-SW-474, located in Unit 2 BC HX SW MOV pit.
	☐ d) Close 2-SW-478, SW Header Crosstie. (MER 4)	
	☐ e) Secure Chemical Injection to MER 3 SW Supply Header IAW 0-OP-SW-006, MER 3 and MER 4 Service Water Chemical Injection Operation	
39. ___	CHECK WATER LEVEL IN MER 3 ON MER 4 GAUGE 2-PL-LI-201 - LOWERING	Do the following: a) Secure CHG Pump SW Pumps: ☐ • 1-SW-P-10A ☐ • 2-SW-P-10A ☐ b) Close 1-SW-499, Water Box 1D Isol. ☐ c) Close 1-SW-500, SW Header Crosstie.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
In Plant Job Performance Measure KA Number 068AA1.12

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

LOCALLY ESTABLISH UNIT 2 RCS & S/G HI/LO INTERFACE INTEGRITY.

K/A: 068AA1.12, Ability to operate and / or monitor the following as they apply to the Control Room Evacuation: Auxiliary shutdown panel controls and indicators. (4.4/4.4)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)	4 Minutes	____ Minutes

Conditions

- Task is to be SIMULATED in the Plant.
- A limiting MCR fire has occurred and forced MCR evacuation. 0-FCA-1.00 Attachment 1 is in progress at step 6.

Standards

- FCA-1.00, Limiting MCR Fire (Rev. 49), Attachment 1, ESGR/Cable Vault Operations, step 6-9.

Initiating Cues

- FCA-1.00, Limiting MCR Fire, Attachment 1, ESGR/Cable Vault Operations
- Shift Manager direction.

Terminating Cues

- Report received that Unit 2 HI/LO Interface Integrity established (steps 6-9 completed).

Procedures

- 0-FCA-1.00, Limiting MCR Fire (Rev. 49), Attachment 1, ESGR Cable Vault Operations.

Tools and Equipment	**Safety Considerations**
• None	• Standard Personal Safety Equipment

Initial Conditions

- This task is to be SIMULATED. Do NOT turn switches, manipulate controls or reposition valves.
- I am the Shift Manager. A fire has forced evacuation of the control room.

Initiating Cues

- Here is Attachment 1 of FCA-1.00, steps 1 through 9.
- I need you to locally establish RCS & SG HI/LO interface integrity for Unit 2 only. You have already obtained the required keys.
- Steps 1 through 5 have already been completed.
- When you finish the actions necessary to accomplish this, please inform me.

Notes to the Evaluator

- This task is to be SIMULATED. Do NOT allow the operator to manipulate controls, operate switches or reposition valves.
- Critical step sequencing requirements: None.

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- **START TIME:**

STEP 1 Open Unit 2 Appendix R Isolation Panel door (Key required). *(Step 6)* **STANDARD:** a) Proceeds the Unit 2 Appendix R panel. b) Simulates unlocking and opening panel. **EVALUATOR'S NOTE:** **If asked:** You have already obtained the required keys. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2 On Unit 2 Appendix R Isolation Panel, place MSTVs in EMERG CLOSE position using FIRE EMERG CLOSE switch. *(Step 7)* **STANDARD:** **Disables the MSTVs by placing the MSTV Fire Emergency Closure switch to EMERG CLOSE.** **EVALUATOR'S NOTE:** **If asked:** Point to EMERG CLOSE position on FIRE EMERG CLOSE switch. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 3 Open Unit 2 Auxiliary Shutdown Panel door (Key required). *(Step 8)* **STANDARD:** a) Proceeds the Unit 2 Auxiliary Shutdown Panel. b) Simulates unlocking and opening panel. **EVALUATOR'S NOTE:** **If asked:** You have already obtained the required keys. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 4 Establish RCS and SG High/Low Interface integrity on Unit 2: *(Step 9)* a) On Auxiliary Shutdown Panel, place the following switches in DISABLE: • RX HEAD/PRZR VENT VVS TRAIN A • 2-RC-SOV-200A-1 - 2-RC-SOV-201A-1 • PRZR PORV, 2-RC-PCV-2455C • LETDOWN LINE ISOL, 2-CH-LCV-2460A • RX HEAD/PRZR VENT VVS TRAIN B • 2-RC-SOV-200B-1 - 2-RC-SOV-201B-1 • PRZR PORV, 2-RC-PCV-2456 • EXCESS LETDOWN FLOW, 2-CH-HCV-2137 **STANDARD:** a) Locates each switch on Unit 2 ASDP. **b) Simulates placing each switch in the DISABLE position.** **EVALUATOR'S NOTE:** **If asked:** Point to DISABLE position on each switch as Applicant simulates correct manipulation. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 5 Reports to Shift Manager (Evaluator) that task is complete. **STANDARD:** Informs Evaluator task has been completed. **COMMENTS:** ****END OF JPM****	_______ **SAT** _______ **UNSAT**

STOP TIME:

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Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- This task is to be SIMULATED. Do NOT turn switches, manipulate controls or reposition valves.
- I am the Shift Manager. A fire has forced evacuation of the control room.

Initiating Cues

- Here is Attachment 1 of FCA-1.00, steps 1 through 9.
- I need you to locally establish RCS & SG HI/LO interface integrity for Unit 2 only. You have already obtained the required keys.
- Steps 1 through 5 have already been completed.
- When you finish the actions necessary to accomplish this, please inform me.

Notes to the Evaluator

- **Task briefing should occur in #2 ESGR in the open area inside the double doors.**
- This task is to be SIMULATED. Do NOT allow the operator to manipulate controls, operate switches or reposition valves.
- Critical step sequencing requirements: None.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- This task is to be SIMULATED. Do NOT turn switches, manipulate controls or reposition valves.
- I am the Shift Manager. A fire has forced evacuation of the control room.

Initiating Cues

- Here is Attachment 1 of FCA-1.00, steps 1 through 9.
- I need you to locally establish RCS & SG HI/LO interface integrity for Unit 2 only. You have already obtained the required keys.
- Steps 1 through 5 have already been completed.
- When you finish the actions necessary to accomplish this, please inform me.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
In Plant Job Performance Measure KA Number 024AA1.04
[ALTERNATE PATH]

Applicant______________________________ Start Time_________________

Examiner_____________________________

Date _________________________________ Stop Time_________________

Title

Locally Emergency Borate Unit 1 (1-CH-228).

K/A: 024AA1.04, Ability to operate and / or monitor the following as they apply to Emergency Boration: Manual boration valve. (3.6/3.7)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)/SRO(U)	5 Minutes	____ Minutes

Conditions

- Task is to be SIMULATED in the Plant.
- A simulated unit startup is in progress when a condition requiring emergency boration occurred.

Standards

- 1-AP-3.00, Emergency Boration, Rev. 5.

Initiating Cues

- Shift Manager direction.

Terminating Cues

- Report received 1-CH-228 locally opened.

Procedures

- 1-AP-3.00, Emergency Boration, Rev 5.

Tools and Equipment

- None

Safety Considerations

- Standard Personal Safety Equipment

Initial Conditions

- This task is to be SIMULATED. Do NOT turn switches, manipulate controls or reposition valves.
- I am the Shift Manager. There is a challenge to Unit 1's Shutdown Margin in progress. We started to emergency borate but 1-CH-MOV-1350 thermalled before it opened.

Initiating Cues

- I need you to locally initiate emergency boration by opening 1-CH-MOV-1350 IAW 1-AP-3.00, Step 2.b RNO.
- When you finish the actions necessary to accomplish this, please inform me.

Notes to the Evaluator:

- A copy of 1-AP-3.00, Emergency Boration, Page 2 of 4 is attached to this JPM.
- This task is to be SIMULATED. Do NOT allow the operator to manipulate controls, operate switches or reposition valves. DO NOT allow the operator to enter a contaminated area or break the vertical plane of a contaminated area for the simulation of this JPM.
- Critical step sequencing requirements: None.

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- **START TIME:**

STEP 1 Locates 1-CH-MOV-1350. **STANDARD:** a) Proceeds the Auxiliary Building, 13 ft. level, Boric Acid Flats. b) Locates 1-CH-MOV-1350 on WEST end of BA flats. **EVALUATOR'S NOTE:** **If asked:** The breaker for 1-CH-MOV-1350 has been opened. If valve is in contaminated area, pointing out w/ flashlight is acceptable. **Provide to Applicant**: A copy of 1-AP-3.00, Emergency Boration, Page 2 of 4 (attached). **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2 Locally open 1-CH-MOV-1350. *(Step 2.b RNO)* **STANDARD:** a) Attempts to engage 1-CH-MOV-1350 manual operator by depressing clutch mechanism. b) Attempts to open valve by rotating MOV handwheel in the counter-clockwise direction. **EVALUATOR'S NOTE:** **Tell Applicant:** The MOV handwheel will not turn or move in the counter-clockwise direction (regardless of how much opening pressure is applied). **If asked:** Valve stem rod does not move. **If asked:** No flow noise heard through valve. **If asked:** No change in either the audible BATP operating noise or BATP discharge pressure. **If asked:** The MOV handwheel will move in the clockwise direction approximately one-quarter turn. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

<table>
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STEP 3

Reports to Shift Manager (Evaluator) that 1-CH-MOV-1350 is not operating.

STANDARD:

a) Contacts SM (Evaluator).
b) Informs SM that 1-CH-MOV-1350 will not locally operate.

EVALUATOR'S NOTE:

Tell Applicant: You are directed to continue with Step 2.b RNO.

COMMENTS:

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_______ **SAT**

_______ **UNSAT**

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STEP 4

Manually open 1-CH-FCV-1113A *(Step 2.b.1 RNO)*

STANDARD:

a) Simulates contacting Unit 1 RO. *May also contact SM (Evaluator).*
b) Directs RO to manually open 1-CH-FCV-1113A.

EVALUATOR'S NOTE:

When asked: 1-CH-FCV-1113A has been manually opened.

COMMENTS:

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_______ **SAT**

_______ **UNSAT**

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</table>

STEP 5 Locally open 1-CH-228. *(Step 2.b.2 RNO)* **STANDARD:** a) Locates 1-CH-228 WEST of 1-CH-MOV-1350. **b) Opens 1-CH-228 by turning valve handwheel in the counter-clockwise direction.** **EVALUATOR'S NOTE:** **Tell Applicant:** When Applicant attempts to open 1-CH-228, simulate rotation of the valve handwheel in the counter-clockwise direction. **If asked:** Valve stem rises as valve handwheel turned in the counter-clockwise direction. 1-CH-228 is a small Grinnell valve in a small alcove WEST of 1-CH-MOV-1350 in the BA piping. If valve is in Contaminated area, pointing out with flashlight is acceptable **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 6 Reports to Shift Manager (Evaluator) that task is complete. **STANDARD:** Informs Evaluator task has been completed. **COMMENTS:** **END OF JPM**	_______ **SAT** _______ **UNSAT**

STOP TIME:

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Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- This task is to be SIMULATED. Do NOT turn switches, manipulate controls or reposition valves.
- I am the Shift Manager. There is a challenge to Unit 1's Shutdown Margin in progress. We started to emergency borate but 1-CH-MOV-1350 thermalled before it opened.

Initiating Cues

- I need you to locally initiate emergency boration by opening 1-CH-MOV-1350 IAW 1-AP-3.00, Step 2.b RNO.
- When you finish the actions necessary to accomplish this, please inform me.

Notes to the Evaluator:

- A copy of 1-AP-3.00, Emergency Boration, Page 2 of 4 is attached to this JPM.
- This task is to be SIMULATED. Do NOT allow the operator to manipulate controls, operate switches or reposition valves. DO NOT allow the operator to enter a contaminated area or break the vertical plane of a contaminated area for the simulation of this JPM.
- Critical step sequencing requirements: None.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- This task is to be SIMULATED. Do NOT turn switches, manipulate controls or reposition valves.
- I am the Shift Manager. There is a challenge to Unit 1's Shutdown Margin in progress. We started to emergency borate but 1-CH-MOV-1350 thermalled before it opened.

Initiating Cues

- I need you to locally initiate emergency boration by opening 1-CH-MOV-1350 IAW 1-AP-3.00, Step 2.b RNO.
- When you finish the actions necessary to accomplish this, please inform me.

TO BE GIVEN TO APPLICANT

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE: • If a Reactor Trip occurs or is required, 1-E-0, REACTOR TRIP OR SAFETY INJECTION, should be implemented. • When the Reactor is shutdown with the Shutdown Banks withdrawn, tripping the Shutdown Banks may eliminate the need for emergency boration.	
1. ___	CHECK CHARGING FLOW - GREATER THAN 75 GPM	☐ Manually adjust charging flow to greater than 75 gpm.
2. ___	START EMERGENCY BORATION	
	☐ a) Transfer the in-service BATP to FAST	a) Manually align CHG pump suction to the RWST: ☐ 1) Open 1-CH-MOV-1115B and D. ☐ 2) Close 1-CH-MOV-1115C and E. ☐ 3) GO TO Step 5.
	☐ b) Open 1-CH-MOV-1350	☐ b) Locally open 1-CH-MOV-1350. IF 1-CH-MOV-1350 can NOT be opened, THEN do the following: ☐ 1) Manually open 1-CH-FCV-1113A. ☐ 2) Locally open 1-CH-228. ☐ 3) Monitor Boric Acid flow on FR-1-113 (red trace). ☐ 4) GO TO Step 3. ☐ IF neither valve can be opened, THEN manually align CHG pump suction to the RWST AND GO TO Step 5.
	c) Monitor EMRG BORATE FLOW ☐ • 1-CH-FI-1110	

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Administrative Job Performance Measure G2.1.5

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

Evaluate Overtime Eligibility

K/A: G.2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (2.9/3.9)

Applicability	**Estimated Time**	**Actual Time**
SRO(I)/SRO(U)	20 Minutes	

Conditions

- Task is to be PERFORMED in the classroom.

Standards

- Correctly determines which operator(s) is/are able to stay over for two hours without prior overtime approval IAW LI-AA-700 – Fatigue Management and Work Hour Limits for Covered Workers (Rev. 12).

Initiating Cues

- Evaluate the work history for all six (6) operators.
- Determine which operator(s), if any, can be held over for two hours without prior overtime approval, and determine which operators CANNOT be held over for two hours without prior overtime approval.
- A copy of LI-AA-700, section 3.1 is available for use during this JPM.
- When you are finished responding to the questions, inform an examiner.

Terminating Cues

- Applicant has evaluated all operators and determined their overtime eligibility.

Tools and Equipment

- Calculator
- Overtime History

Safety Considerations

- None

Initial Conditions:

- A start-up, following a mid-cycle reactor trip, is planned for the following shift. One Reactor Operator must be held over two hours for the start-up.
- Attached is the work history (excluding shift turnover time) of the available Reactor Operators on shift.
- A break of at least 10 hours occurred between all work periods.
- All operators began their shift at the same time each day. (Day Shift)
- The computer program for calculating overtime eligibility is not available.
- Calculation of shift cycle rolling average is not required.

Initiating Cues

You have been directed to:

- Evaluate the work history for all six (6) operators.
- Determine which operator(s), if any, can be held over for an additional two hours without prior overtime approval and determine which operator(s), if any, CANNOT be held over for two hours without prior overtime approval.
- State the reason the operators can or cannot be held over for the additional two hours.
- All six operators have tomorrow (Day 9) off.

Notes

- The applicable procedure, LI-AA-700 – Fatigue Management and Work Hour Limits for Covered Workers, should be distributed to the applicant with the directions.
- The evaluations of the individual operators may be performed in any order.

Work History

Day	1	2	3	4	5	6	7	8 (Today)
Operator #1	12	0	12	12	12	12	8	14
Operator #2	0	12	12	12	12	12	0	12
Operator #3	0	12	12	12	0	0	12	12
Operator #4	12	0	12	12	12	12	0	13
Operator #5	12	12	12	12	0	0	0	15
Operator #6	0	12	12	12	12	8	8	8

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- **START TIME:**

STEP 1 OBTAINS A CURRENT COPY OF LI-AA-700 – FATIGUE MANAGEMENT AND WORK HOUR LIMITS FOR COVERED WORKERS. **STANDARD:** Obtains a copy of LI-AA-700. This can be accomplished by either requesting a copy of the procedure from the evaluator or by logging on to a network computer and obtaining an electronic version of the procedure. **EVALUATOR NOTES:** **Provide to Applicant**: A copy of LI-AA-700, section 3.3.1 through 3.3.4 is attached to this JPM. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2 DETERMINES THE ELIGIBILITY OF OPERATOR #1. **STANDARD:** **Identifies that Operator #1 would not exceed any overtime limits.** **EVALUATOR NOTES:** *Days 2-8 will reach 72 hours, but not exceed it. Day 8 will reach 16 hours, but not exceed it.* *Reference: LI-AA-700, Section 3.3.1.* **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 3 DETERMINES THE ELIGIBILITY OF OPERATOR #2. **STANDARD:** **Identifies that Operator #2 would exceed hours limitations.** **EVALUATOR NOTES:** *Days 2 - 8 already have 72 hours in a 7-day period. The additional two hours would yield 74 hours in a 7-day period, which would require management approval.* *Reference: LI-AA-700, Section 3.3.1.* **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 4 DETERMINES THE ELIGIBILITY OF OPERATOR #3. **STANDARD:** **Identifies that Operator #3 would not exceed hours limitations.** **EVALUATOR NOTES:** *Day 7 and 8 have 24 hours in a 48 hour period. The additional two hours would yield 26 hours in a 48 hour period, which is the maximum.* *Reference: LI-AA-700, Section 3.3.1.* **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 5 DETERMINES THE ELIGIBILITY OF OPERATOR #4. **STANDARD:** **Identifies that Operator #4 would not exceed any overtime limits.** **EVALUATOR NOTES:** *No limits are exceeded.* *Reference: LI-AA-700, Section 3.3.1.* **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 6 DETERMINES THE ELIGIBILITY OF OPERATOR #5. **STANDARD:** **Identifies that Operator #5 would exceed hours limitations.** **EVALUATOR NOTES:** *Day 8 already has 15 hours in a 24 hour period. The additional two hours would yield 17 hours consecutively and in a 24 hour period, which would require management approval.* *Reference: LI-AA-700, Section 3.3.1.* **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 7 DETERMINES THE ELIGIBILITY OF OPERATOR #6. **STANDARD:** **Identifies that Operator #6 would exceed hours limitations.** **EVALUATOR NOTES:** *Days 2 - 8 already have 72 hours in a 7-day period. The additional two hours would yield 74 hours in a 7-day period, which would require management approval.* *Reference: LI-AA-700, Section 3.3.1.* **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 8 REPORTS ELIGIBILITY OF ALL OPERATORS. **STANDARD:** Applicant reports that Operators 1, 3, and 4 are eligible to receive the overtime without prior management approval and the others are not. Table correctly filled out. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STOP TIME:

ANSWER KEY

Work History

Day	1	2	3	4	5	6	7	8 (Today)
Operator #1	12	0	12	12	12	12	8	14
Operator #2	0	12	12	12	12	12	0	12
Operator #3	0	12	12	12	0	0	12	12
Operator #4	12	0	12	12	12	12	0	13
Operator #5	12	12	12	12	0	0	0	15
Operator #6	0	12	12	12	12	8	8	8

***ALL BLOCKS OF THIS TABLE ARE CRITICAL STEPS**

	Eligible for 2 hours of additional overtime without management approval?	Reason, if any, for **NOT** being eligible for overtime without prior management approval.
Operator #1	**Yes**	
Operator #2	**No**	**Exceeds 72 hours in 7 day period**
Operator #3	**Yes**	
Operator #4	**Yes**	
Operator #5	**No**	**Exceeds 16 consecutive hours** **Exceeds 16 hours in 24 hour period**
Operator #6	**No**	**Exceeds 72 hours in 7 day period**

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions:

- A start-up, following a mid-cycle reactor trip, is planned for the following shift. One Reactor Operator must be held over two hours for the start-up.
- Attached is the work history (excluding shift turnover time) of the available Reactor Operators on shift.
- A break of at least 10 hours occurred between all work periods.
- All operators began their shift at the same time each day. (Day Shift)
- The computer program for calculating overtime eligibility is not available.
- Calculation of shift cycle rolling average is not required.

Initiating Cues

You have been directed to:

- Evaluate the work history for all six (6) operators.
- Determine which operator(s), if any, can be held over for an additional two hours without prior overtime approval and determine which operator(s), if any, CANNOT be held over for two hours without prior overtime approval.
- State the reason the operators can or cannot be held over for the additional two hours.
- All six operators have tomorrow (Day 9) off.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions:

- A start-up, following a mid-cycle reactor trip, is planned for the following shift. One Reactor Operator must be held over two hours for the start-up.
- Attached is the work history (excluding shift turnover time) of the available Reactor Operators on shift.
- A break of at least 10 hours occurred between all work periods.
- All operators began their shift at the same time each day. (Day Shift)
- The computer program for calculating overtime eligibility is not available.
- Calculation of shift cycle rolling average is not required.

Initiating Cues

You have been directed to:

- Evaluate the work history for all six (6) operators.
- Determine which operator(s), if any, can be held over for an additional two hours without prior overtime approval and determine which operator(s), if any, CANNOT be held over for two hours without prior overtime approval.
- State the reason the operators can or cannot be held over for the additional two hours.
- All six operators have tomorrow (Day 9) off.

(TO BE GIVEN TO APPLICANT)

Work History

Day	1	2	3	4	5	6	7	8 (Today)
Operator #1	12	0	12	12	12	12	8	14
Operator #2	0	12	12	12	12	12	0	12
Operator #3	0	12	12	12	0	0	12	12
Operator #4	12	0	12	12	12	12	0	13
Operator #5	12	12	12	12	0	0	0	15
Operator #6	0	12	12	12	12	8	8	8

	Eligible for 2 hours of additional overtime without management approval?	Reason, if any, for **NOT** being eligible for overtime without prior management approval.
Operator #1		
Operator #2		
Operator #3		
Operator #4		
Operator #5		
Operator #6		

(TO BE GIVEN TO APPLICANT)

3.3 Work Hours, Breaks, and Days Off

Cognizant Supervisor

3.3.1 **APPLY** the following limits to individuals regardless of unit status:

- No more than 16 consecutive hours
- No more than 16 work hours in any rolling 24-hour period
- No more than 26 work hours in any rolling 48-hour period
- No more than 72 work hours in any rolling 168-hour (7-day) period
- At least a 10-hour break between successive work periods or an 8-hour break when a break of less than 10 hours is necessary to accommodate a crew's or individual's scheduled transition between work schedules or shifts
- A 34-hour uninterrupted, continuous break in any 216-hour (9-day) period

3.3.2 **REFER** to Section 3.2, Calculating Work Hours, and **CALCULATE** the weekly average work hours over the averaging period (not to exceed six weeks) by dividing the total hours worked by the number of full weeks (Sunday through Saturday) in the rolling averaging period.

NOTE: During the first 60 days of an unplanned security system outage or increased threat condition, security personnel need **NOT** meet the minimum days off requirements specified in Table 2 or the maximum average work hour requirements specified in step 3.3.3.

NOTE: The non-outage maximum average work hour limit could allow individuals to work every day and remain less than the limit. It remains Dominion's responsibility to schedule hours consistent with the objective of preventing impairment from fatigue consistent with the requirements of 10CFR26.205(c) and Section 3.7 of this procedure.

3.3.3 During online operations and without issuance of a waiver, **ENSURE** an individual does **NOT** work more than a weekly average of 54 hours, calculated using an averaging period of up to six weeks, which advances by seven consecutive calendar days at the finish of every averaging period.

3.3.4 For the purposes of compliance with maximum average work hours requirements, **EXCLUDE** hours worked over the weekly 54-hour limit by Security personnel during the actual conduct of force-on-force exercises evaluated by the NRC.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Administrative Job Performance Measure G2.1.19

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

Perform calculation of reactor power using 1-OPT-RX-003, Reactor Power Calorimetric using Feed Flow and PCS Computer Points (Manual).

K/A: G2.1.19, Ability to use plant computers to evaluate system or component status. (3.9 / 3.8)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)/SRO(U)	45 Minutes	

Conditions

- Task is to be PERFORMED in the classroom.
- Unit 1 is at 90% power.
- Feedwater Ultrasonic Flow Measurement (UFM) is non-functional.

Standards

- 1-OPT-RX-003, Sections 6.1 and 6.2 completed satisfactorily.

Initiating Cues

- Nuclear Shift Manager direction.

Terminating Cues

- 1-OPT-RX-003, Sections 6.1 and 6.2 completed.

Procedures

- 1-OPT-RX-003， Reactor Power Calorimetric using Feed Flow and PCS Computer Points (Manual), Rev. 26
- 1-DRP-003, Curve Book (Unit 1), Rev. 125

Tools and Equipment	**Safety Considerations**
• Calculator	• None

Initial Conditions

- Unit 1 is at 90% power and has been stable for 2 hours. No periodic tests or calibration evolutions are in progress.
- Feed Water Ultrasonic Flow Measurement (UFM) is non-functional.
- The PCS Calorimetric program is otherwise functional.
- The following unit conditions exist:
  - The Manual Calorimetric Spreadsheet will NOT be used.
  - Feed Water Regulating Valve bypass valves are closed.

Initiating Cues

- I am the Nuclear Shift Manager and you are the Assistant RO.
- Using the attached PP Output Summary sheet, I need you to perform Sections 6.1 and 6.2 of 1-OPT-RX-003, Reactor Power Calorimetric using Feed Flow and PCS Computer Points (Manual).
- Please inform me when you have completed the assigned task.

Notes

- A marked-up copy of 1-OPT-RX-003 should be given to the Applicant.
- A copy of 1-DRP-003, Unit 1 Curve Book, shall be made available.

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- **START TIME:**

STEP 1 Applicant reviews sections of procedure that are already signed. **STANDARD:** a) Reviews Section 1.0, Purpose. b) Reviews Section 2.0, References. c) Reviews Section 3.0, Initial Conditions. d) Reviews Section 4.0, Precautions and Limitations. e) Reviews Section 5.0, Special Tools and Equipment. **EVALUATOR'S NOTE:** **If asked**: The Manual Calorimetric Spreadsheet will not be used. **If asked**: Data points from the UFM cabinet will not be used. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 2 Applicant commences Section 6.1. **NOTE**: This calorimetric uses the corrected Steam Generator Feedwater flow as calculated by the Flow Corrections program to calculate reactor power according to the following equation. Reactor Power = (h_{steam} h_{feed}) x $Flow_{feed}$ - Added Pump Heat – Added Pressurizer Heat – Blowdown Heat Loss + Insulation Losses + Letdown, Charging, and Seal Injection Heat Contributions Where: • Pump Heat equals 40.96 x 106 BTU/hr. • Blowdown Flow is recorded from PCS points (F2551A) FPP0001K , (F2552A) FPP0002K, (F2553A) FPP0003K, (preferred) or from Control Room indications. • Insulation losses equal 1.5 MWth = 5.12 x 106 BTU/hr. • Charging, letdown, and seal water injection heat contributions equals 5.0 MWth = 17.06 x 106 BTU/hr **STANDARD:** Acknowledges NOTE regarding data collection. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 3 IF this procedure is being performed because the Primary Plant Performance Program is not operational, THEN initiate 1-OPT-RX-007 to determine the shift average power. Otherwise, enter N/A. *(Step 6.1.1)* **STANDARD:** a) Applicant notes that Shift Average Power indications listed on provided sheet indicate "BAD". b) Informs SM (Evaluator) that 1-OPT-RX-007 should be initiated. **EVALUATOR'S NOTE:** **If asked**: Another Operator will perform 1-OPT-RX-007 as necessary. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 4 IF the Manual Calorimetric Spreadsheet will be used, THEN line through Attachment 3 and mark with "See Attached." Otherwise, enter N/A. *(Step 6.1.2)* **STANDARD:** a) Recalls from Initial Conditions that the Manual Calorimetric Spreadsheet will not be used. b) Enters N/A for the step. **EVALUATOR'S NOTE:** **If asked:** The Manual Calorimetric Spreadsheet will not be used. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 5 **NOTE:** Venturi Feed flow transmitter data will be invalid if feed flow transmitters are bypassed. *(NOTE before Step 6.1.3)* **STANDARD:** Acknowledges NOTE. **EVALUATOR'S NOTE:** **If asked**: The feed flow venturi transmitters are in service. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 6 IF Primary Plant Performance is based on Venturi feedwater flow AND the Feed Reg Bypass HCVs are NOT closed AND feedwater flow is NOT aligned through the feed flow transmitters, THEN close the Feed Reg Bypass HCVs OR align bypass flow to the feed flow transmitters to obtain Flow Corrections data. Otherwise, enter N/A for this step. *(Step 6.1.3)* **STANDARD:** a) Recalls from Initial Conditions that the Feed Regulating Valve bypasses are closed. b) Enters N/A for step. **EVALUATOR'S NOTE:** **If asked**: Feed Regulating Valve bypasses are closed. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 7 IF Step 6.1.3 was performed, THEN wait approximately 5 minutes AND GO TO Subsection 6.2. *(Step 6.1.4)* **STANDARD:** Enters N/A for step since the previous step was not performed. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 8 Applicant commences Section 6.2. **NOTE**: When using feedwater flow as the basis for the calorimetric the Filtered Average Feed Flow should be used. **STANDARD:** Acknowledges NOTE. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 9 Obtain the values for SG Pressure, FW Temperature, and Main Feedwater Flow for each loop from the PCS computer. Record the computer point values in the appropriate boxes below and on Attachment 3, Page 1. *(Step 6.2.1)* • U9171 SG A Corrected Stm Press __*829.98*__ psia • U9172 SG B Corrected Stm Press __*828.62*__ psia • U9173 SG C Corrected Stm Press __*827.66*__ psia • T0418A SG A FW Temp (RTD-111A) __*431.16*__ °F • T0438A SG B FW Temp (RTD-111B) __*431.16*__ °F • T0458A SG C FW Temp (RTD-111C) __*431.16*__ °F U9174 SG A Filtered Average Feed Flow __*3391.24*__ x 10^3 lbm/hr U9175 SG B Filtered Average Feed Flow __*3392.67*__ x 10^3 lbm/hr U9176 SG C Filtered Average Feed Flow __*3392.99*__ x 10^3 lbm/hr **STANDARD:** Using JPM Attachment 1 (PP Output Summary), Applicant fills in numbers for this step and on procedure Attachment 3. **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 10 IF Feedwater temperature for any loop is greater than 443°F, THEN notify Reactor Engineering. Otherwise, enter N/A. *(Step 6.2.2)* **STANDARD:** a) Applicant notes that Feedwater temperature is 431.16°F. b) Applicant places N/A in initial block. **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 11 IF Step 6.1.3 was performed, THEN return the Feed Reg Bypass HCVs to desired position. Otherwise, enter N/A. *(Step 6.2.3)* **STANDARD:** a) Applicant recalls that the Feed Reg Bypass HCVs were not manipulated. b) Enters N/A in initial block. **EVALUATOR'S NOTE:** **If asked**: Feed Reg Valve bypasses were not manipulated. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 12 Obtain pressurizer heater input by using the computer point listed below. Record this value in the appropriate box on Attachment 3, Page 2. (Enter 0 KW if computer point inoperable) *(Step6.2.4)* • Q0400A Pressurizer Heater Power __*850.7*__ KW **STANDARD:** a) Applicant references attached PP Output Summary from PCS. b) Notes that PZR Heater Power is 850.7 KW. c) Records 850.7 KW in the step block and on page 2 of Attachment 3. **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 13 **NOTE:** • Blowdown flow must be maintained as constant as possible. The most accurate data will be obtained by isolating blowdown, but isolation is not required. • PCS points for automatic Blowdown flow are the preferred inputs for the following step. *(NOTE prior to Step 6.2.5)* **STANDARD:** Applicant acknowledges NOTE. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 14 Obtain loop blowdown flow by using the PCS points or indicators listed below. Circle PCS point (preferred) or indicator used. Record these values in the appropriate boxes on Attachment 3, Page 1. *(Step 6.2.5)* • (F2551A) FPP0001K, FI-BD-103A or FI-BD-104A SG A BD Flow ___*57.540*__ gpm • (F2552A) FPP0002K, FI-BD-103B or FI-BD-104B SG B BD Flow ___*62.593*__ gpm • (F2553A) FPP0003K, FI-BD-103C or FI-BD-104C SG C BD Flow ___*58.400*__ gpm **STANDARD:** a) Applicant refers to attached PP Output Summary from PCS for blowdown flows. b) Circles the PCS point (F2551A, etc.) and records value in step and on Attachment 3. **EVALUATOR'S NOTE:** **If asked**: Blowdown is in AUTO mode for PCS. A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

<table>
<tr><td>

STEP 15

Find the enthalpy of steam, h_s, for each loop using Corrected Steam Pressure from Attachment 3 and the Enthalpy Steam Table (100% Quality) in 1-DRP-003. Record values in the appropriate boxes on Attachment 3, Page 1. *(Step 6.2.6)*

STANDARD:

a) Applicant locates Enthalpy Steam Table (100% Quality) in 1-DRP-003 (Attachment 72).
b) **Determines h_s for each loop.** Applicant may interpolate exact values or round to the nearest psia.
- **Loop A – 1198.54 BTU/lbm** *(band 1198.57 – 1198.51 BTU/lbm)*
- **Loop B – 1198.57 BTU/lbm** *(band 1198.60 – 1198.54 BTU/lbm)*
- **Loop C – 1198.60 BTU/lbm** *(band 1198.63 – 1198.57 BTU/lbm)*

c) Records values on Attachment 3.

EVALUATOR'S NOTE:

The listed band was developed by rounding steam pressure to the nearest psia, then taking the enthalpy value for ±1 psia.

A completed table is attached at the end of this JPM showing all data.

COMMENTS:

</td><td>

_______ **SAT**

_______ **UNSAT**

</td></tr>
<tr><td>

STEP 16

NOTE: Using a FW pressure of 800 psia in the next step will be conservative for all Reactor Power levels.

Find the enthalpy of feedwater, h_f, for each loop, using Feedwater Temperature from Attachment 3 and the Enthalpy Compressed Liquid Table (800 psia) in 1-DRP-003. Record values in the appropriate boxes on Attachment 3, Page 1. *(Step 6.2.7)*

STANDARD:

a) Acknowledges NOTE prior to step that using FW pressure of 800 psia is conservative.
b) Applicant locates Enthalpy Compressed Liquid Table (800 psia) in 1-DRP-003 (Attachment 74).
c) **Determines h_f for each loop.** Applicant may interpolate exact values or round to the nearest tenth of a degree.
- **Loop A – 409.61 BTU/lbm** *(band 409.50 – 409.72 BTU/lbm)*
- **Loop B – 409.61 BTU/lbm** *(band 409.50 – 409.72 BTU/lbm)*
- **Loop C – 409.61 BTU/lbm** *(band 409.50 – 409.72 BTU/lbm)*

d) Records values on Attachment 3.

EVALUATOR'S NOTE:

The listed band was developed by rounding feedwater temperature to the nearest tenth of a degree, then taking the enthalpy value for ±1/10th of a degree.

A completed table is attached at the end of this JPM showing all data.

COMMENTS:

</td><td>

_______ **SAT**

_______ **UNSAT**

</td></tr>
</table>

STEP 17 Calculate $\Delta h_1 = h_s - h_f$ for each loop and record results in appropriate boxes on Attachment 3, Page 1. *(Step 6.2.8)* **STANDARD:** **Applicant calculates Δh_1 and records values.** • **Loop A – 788.93 BTU/lbm** *(band 789.07 – 788.79 BTU/lbm)* • **Loop B – 788.96 BTU/lbm** *(band 789.10 – 788.82 BTU/lbm)* • **Loop C – 788.99 BTU/lbm** *(band 789.13 – 788.85 BTU/lbm)* **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	**_______ SAT** **_______ UNSAT**
STEP 18 Calculate Blowdown Flow M_{bd} (lbm/hr) = BD (gpm) x 496.6563 $\frac{\text{lbm/hr}}{\text{gpm}}$. Record values in the appropriate boxes on Attachment 3, Page 1. *(Step 6.2.9)* **STANDARD:** **Applicant calculates M_{bd} and records values.** • **Loop A – 28577.60350 lbm/hr** *(band 28577 – 28578 lbm/hr)* • **Loop B – 31087.20779 lbm/hr** *(band 31087 – 31088 lbm/hr)* • **Loop C – 29004.72792 lbm/hr** *(band 29004 – 29005 lbm/hr)* **EVALUATOR'S NOTE:** The listed band was developed by rounding up or down to the nearest whole number. A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	**_______ SAT** **_______ UNSAT**

STEP 19 Find the enthalpy of the blowdown, h_{bd}, for each loop, using the Corrected Steam Pressure from Attachment 3 and the Enthalpy Saturated Liquid Table in 1-DRP-003. Record values in the appropriate boxes on Attachment 3, Page 1. *(Step 6.2.10)* **STANDARD:** a) Applicant locates Enthalpy Saturated Liquid Table in 1-DRP-003 (Attachment 73). b) **Determines h_{bd} for each loop.** Applicant may interpolate exact values or round to the nearest psia. • **Loop A – 515.00 BTU/lbm** *(band 514.83 – 515.17 BTU/lbm)* • **Loop B – 514.83 BTU/lbm** *(band 514.66 – 515.00 BTU/lbm)* • **Loop C – 514.66 BTU/lbm** *(band 514.49 – 514.83 BTU/lbm)* c) Records values on Attachment 3. **EVALUATOR'S NOTE:** The listed band was developed by rounding steam pressure to the nearest psia, then taking the enthalpy value for ±1 psia. A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 20 Calculate $\Delta h_2 = h_s - h_{bd}$ for each loop and record results in appropriate boxes on Attachment 3, Page 1. *(Step 6.2.11)* **STANDARD:** **Applicant calculates Δh_2 and records values**. • **Loop A – 683.54 BTU/lbm** *(band 683.74 – 683.34 BTU/lbm)* • **Loop B – 683.74 BTU/lbm** *(band 683.94 – 683.54 BTU/lbm)* • **Loop C – 683.94 BTU/lbm** *(band 684.14 – 683.74 BTU/lbm)* **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 21 Perform the following for each loop: *(Step 6.2.12)* a. Calculate (M_f x Δh_1) and (M_{bd} x Δh_2) for each loop and record results in appropriate boxes on Attachment 3, Page 1 b. Calculate Q_{loop} = (M_f x Δh_1) - (M_{bd} x Δh_2) for each loop and record results in appropriate boxes on Attachment 3, Page 1. **STANDARD:** **a) Applicant calculates (M_f x Δh_1) and (M_{bd} x Δh_2) for each loop and record results in appropriate boxes on Attachment 3.** **(M_{bd} x Δh_2)** • **Loop A – 19,533,935.10 BTU/hr** *(band 19,539,237.98 – 19,528,490.52 BTU/hr)* • **Loop B – 21,255,567.45 BTU/hr** *(band 21,261,642.78 – 21,249,891.52 BTU/hr)* • **Loop C – 19,837,493.61 BTU/hr** *(band 19,842,796.56 – 19,831,878.70 BTU/hr)* **(M_f x Δh_1)** • **Loop A – 2,675,450,973 BTU/hr** *(band 2,675,925,747 – 2,674,976,200 BTU/hr)* • **Loop B – 2,676,680,923 BTU/hr** *(band 2,677,155,897 – 2,676,205,949 BTU/hr)* • **Loop C – 2,677,035,180 BTU/hr** *(band 2,677,510,199 – 2,676,560,162 BTU/hr)* **b) Applicant calculates Q_{loop} = (M_f x Δh_1) - (M_{bd} x Δh_2) for each loop and record results in appropriate boxes on Attachment 3.** • **Loop A – 2,655,917,038 BTU/hr** *(band 2,656,386,509 – 2,655,447,709 BTU/hr)* • **Loop B – 2,655,425,356 BTU/hr** *(band 2,655,894,254 – 2,654,956,057 BTU/hr)* • **Loop C – 2,657,197,686 BTU/hr** *(band 2,657,667,402 – 2,656,728,283 BTU/hr)* **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 22 Convert Pressurizer Heat Input from KW to BTU/hr by multiplying by 3413.0 BTU/hr/KW, and record results in appropriate boxes on Attachment 3, Page 2. *(Step 6.2.13)* **STANDARD:** a) Applicant multiplies PZR Heat Input (*850.7* KW) by 3413.0 BTU/hr/KW. **b) Records 2,903,439.1 BTU/hr in appropriate block.** **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 23 Calculate total heat from Reactor by using $Q_{Total} = Q_{loop\ A} + Q_{loop\ B} + Q_{loop\ C}$ (BTU/hr) - PRZR HTR Input (BTU/hr) - RCP Heat Input (BTU/hr) + Letdown, Seal Injection, and Charging Heat Loss (BTU/hr) + Insulation Loss (BTU/hr). Record results in appropriate box on Attachment 3, Page 2 *(Step 6.2.14)* **STANDARD:** Applicant calculates Q_{Total}. • **$Q_{loop\ A} + Q_{loop\ B} + Q_{loop\ C}$ = 7,968,540,080 BTU/hr** *(band 7,969,948,165 – 7,958,132,049 BTU/hr)* • -RCP Heat Input + Letdown, Seal Injection, and Charging Heat Loss + Insulation Loss = – 18.78E6 BTU/hr • **-PZR Heat Input = 2,903,439.1 BTU/hr** • **Q_T = 7,946,856,641 BTU/hr** *(band 7,948,264,726 – 7,936,448,610 BTU/hr)* **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 24 Divide Q_T by 3.413 x 10^6 to find Reactor output in MW_{th}. Record results in appropriate box on Attachment 3, Page 2. *(Step 6.2.15)* **STANDARD:** Applicant calculates Reactor output in MW_{th}. • **MW_{th} = 7,946,856,641 BTU/hr ÷ 3413000** **= 2,328.0804** *(band 2328.8206 – 2325.3585)* **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 25 Find the percent power level by using % Power = (MW_{th}/2587) x 100. Record results in appropriate box on Attachment 3, Page 2. *(Step 6.2.16)* **STANDARD:** Applicant calculates % Reactor Power. • **% Power = (2,328.0804 ÷ 2587) MWth x 100** **= 90.00 %** *(band 90.02% - 89.89%)* **EVALUATOR'S NOTE:** A completed table is attached at the end of this JPM showing all data. The Applicant may sign and date the Attachment at this time and report the JPM completed. It is at the Evaluator's discretion to continue the procedure. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 26 IF the Manual Calorimetric Spreadsheet was used, THEN sign and date the computer generated printouts (performer and independent reviewer) and attach the printouts to this procedure. *(Step 6.2.17)* **STANDARD:** Applicant recalls that the Manual Calorimetric Spreadsheet was not used and enters N/A for the step. **EVALUATOR'S NOTE:** **If asked**: The Manual Calorimetric Spreadsheet was not used. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 27 IF Attachment 3 was used, THEN sign and date Attachment 3 (performer and independent reviewer). *(Step 6.2.18)* **STANDARD:** a) Applicant signs and dates Attachment 3. b) Requests an Independent Verifier to check work. **EVALUATOR'S NOTE:** The Applicant may report the JPM complete at this time. It is at the Evaluator's discretion to continue the procedure **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 28 **CAUTION** To prevent exceeding maximum rated Reactor thermal power, Reactor power must not be increased based on the result of this calorimetric. NOTE: Due to differences in the uncertainty calculations for Primary Plant Performance and the manual calorimetric, indicated power between the two may vary by 0.4%. IF Reactor Power as calculated is greater than 98.4%, THEN perform the following: *(Step 6.2.19)* a. Immediately reduce Reactor Power to less than 98.4% power IAW Attachment 4. b. Terminate this procedure and reperform calorimetric. **STANDARD:** Applicant notes that Reactor Power is 90%. Enters N/A in both blocks. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 29 Report to Shift Manager (Evaluator) completion of Task. **COMMENTS:** ** **JPM COMPLETE** **	_______ **SAT** _______ **UNSAT**

STOP TIME: ___________________________

Comments: ____

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	LOOP A	LOOP B	LOOP C
Corrected Steam Pressure (psia)	U9171 *829.98*	U9172 *828.62*	U9173 *827.66*
Enthalpy Steam h_s (BTU/lbm)	***1198.54***	***1198.57***	***1198.60***
Feedwater Temp (°F)	T0418A *431.16*	T0438A *431.16*	T0458A *431.16*
Enthalpy FW h_f (BTU/lbm)	***409.61***	***409.61***	***109.61***
$\Delta h_1 = (h_s - h_f)$ BTU/lbm	***788.93***	***788.96***	***788.99***
Blowdown Flow (gpm)	(SG A) *57.540*	(SG B) *62.593*	(SG C) *58.400*
x Conversion gpm to lbm/hr	x 496.6563	x 496.6563	x 496.6563
Blowdown Flow M_{bd} (lbm/hr)	***= 28577.6035***	***= 31087.20779***	***= 29004.72792***
Enthalpy h_{bd} (BTU/lbm)	***515.00***	***514.83***	***514.66***
$\Delta h_2 = (h_s - h_{bd})$ BTU/lbm	683.54	683.74	683.94
M_{bd} x Δh_2 (BTU/hr)	***= 19533935.1***	***= 21255567.45***	***= 19837493.61***
Feedwater Flow M_{fw} (lbm/hr)	SG A Feed Flow *3391.24E3*	SG B Feed Flow *3392.67E3*	SG C Feed Flow *3392.99E3*
M_{fw} x Δh_1 (BTU/hr)	***2,675,450,973***	***2,676,680,923***	***2,677,035,180***
Q_{loop} = (M_{fw} x Δh_1) – (M_{bd} x Δh_2) BTU/hr	$Q_{loop\ A}$ = ***2,655,917,038***	$Q_{loop\ B}$ = ***2,655,425,356***	$Q_{loop\ C}$ = ***2,657,197,686***

Pressurizer Heater Input (KW)	*850.7* (Q0400A)
x Conversion KW to BTU/hr	x 3413
Pressurizer Heater Input	= ***2,903,439.1*** *(2.9E6)*

$Q_{loop\ A}$ + $Q_{loop\ B}$ + Q_{loop} C (BTU/hr)	= ***7,968,540,080***
- RCP Input + Letdown, Charging, and Seal Injection Losses + Insulation Losses	- 18.78 x 106 BTU/hr
- Pressurizer Heater Input (BTU/hr)	- *2903439.1*
QT (BTU/hr)	= ***7946856641***
MW_{th} = QT / 3413000	***= 2,328.40804 MWth***
% POWER = (MW/th / 2587) x 100	***= 90.00417627% POWER***

- Instructor-calculated values are in ***BOLD Italics***.

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

SRO Only Question

- The Feedwater Ultrasonic Flow Measurement System (UFM) has been determined to be NON-FUNCTIONAL.
- The PCS Calorimetric program is otherwise functional.
- You are to evaluate the Surry Technical Requirements Manual regarding the UFM.
- When you have determined the Applicable TRM item (if any), please inform me.

STANDARD:

- Applicant evaluates TRM-3.3.5, Feedwater Ultrasonic Flow Meter Calorimetric:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Feedwater UFM System not FUNCTIONAL.	A.1 Change the calorimetric program from the Feedwater UFM System to the Normalized Feedwater Venturi System. AND A.2 Restore Feedwater UFM System to FUNCTIONAL status.	1 hour 48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to $\leq$ 2546 MWt (98.4% rated power). AND B.2 Change the calorimetric program from the Normalized Feedwater Venturi System to the Feed or Steam Venturi System.	1 hour 1 hour

EVALUATOR'S NOTE:

- **Provide Candidate with a copy of Surry Technical Requirements Manual.**

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

SRO Only Question

- The Feedwater Ultrasonic Flow Measurement System (UFM) has been determined to be NON-FUNCTIONAL.
- The PCS Calorimetric program is otherwise FUNCTIONAL.
- You are to evaluate the Surry Technical Requirements Manual regarding the UFM.
- When you have determined the Applicable TRM item (if any), please inform me.

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- Unit 1 is at 90% power and has been stable for 2 hours. No periodic tests or calibration evolutions are in progress.
- Feed Water Ultrasonic Flow Measurement (UFM) is non-functional.
- The PCS Calorimetric program is otherwise functional.
- The following unit conditions exist:
  - The Manual Calorimetric Spreadsheet will NOT be used.
  - Feed Water Regulating Valve bypass valves are closed.

Initiating Cues

- I am the Nuclear Shift Manager and you are the Assistant RO.
- Using the attached PP Output Summary sheet, I need you to perform Sections 6.1 and 6.2 of 1-OPT-RX-003, Reactor Power Calorimetric using Feed Flow and PCS Computer Points (Manual).
- Please inform me when you have completed the assigned task.

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- Unit 1 is at 90% power and has been stable for 2 hours. No periodic tests or calibration evolutions are in progress.

- Feed Water Ultrasonic Flow Measurement (UFM) is non-functional.

- The PCS Calorimetric program is otherwise functional.

- The following unit conditions exist:

  - The Manual Calorimetric Spreadsheet will NOT be used.
  - Feed Water Regulating Valve bypass valves are closed.

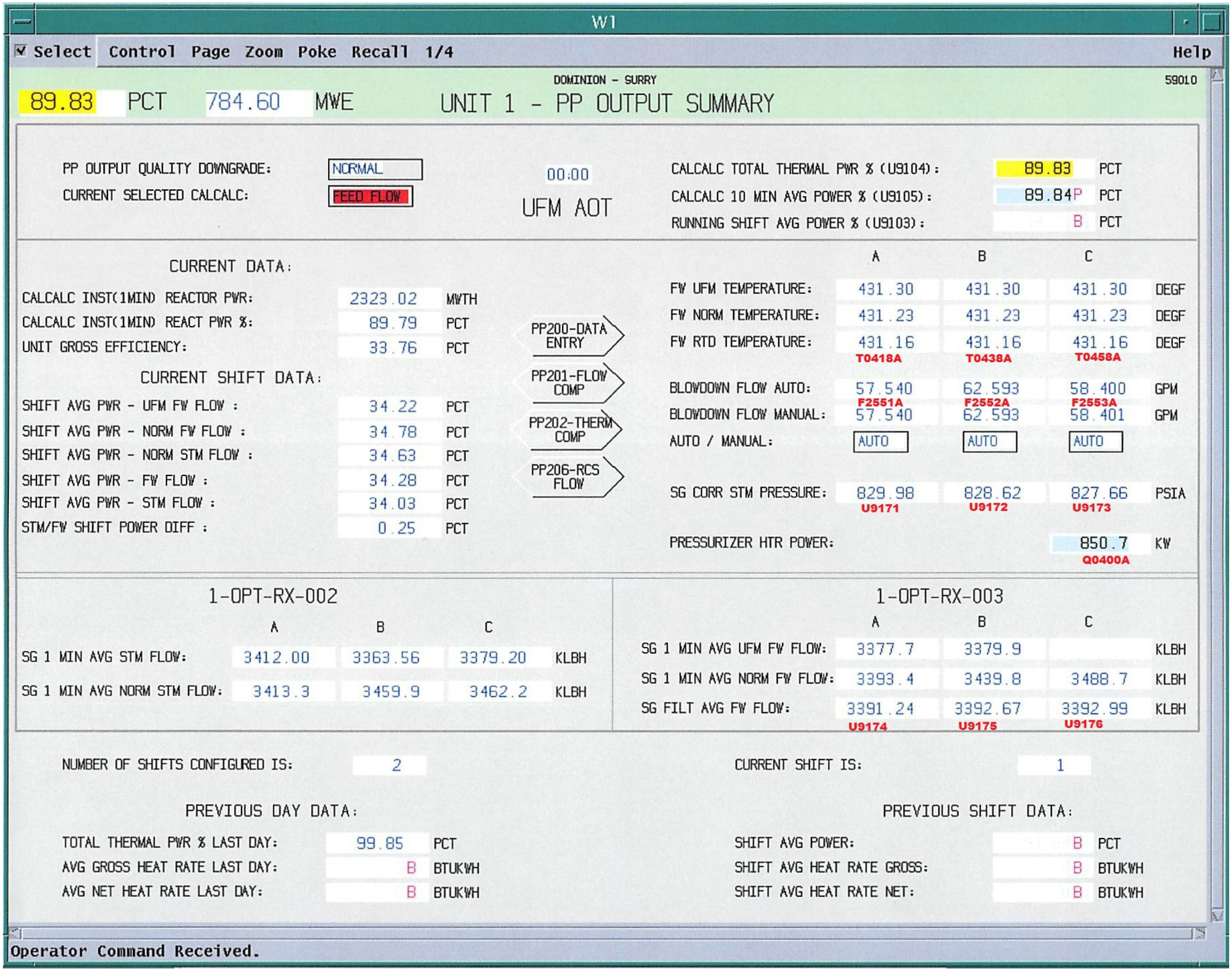
Initiating Cues

- I am the Nuclear Shift Manager and you are the Assistant RO.

- Using the attached PP Output Summary sheet, I need you to perform Sections 6.1 and 6.2 of 1-OPT-RX-003, Reactor Power Calorimetric using Feed Flow and PCS Computer Points (Manual).

- Please inform me when you have completed the assigned task.

Attachment 1



U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Administrative Job Performance Measure G2.3.2

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

Calculate the radiation exposure when locally operating 2-SI-MOV-2865B.

K/A: G2.3.4 Knowledge of radiation exposure limits under normal and emergency conditions. (3.2/3.7)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO(I)/SRO(U)	25 Minutes	

Conditions

- Task is to be PERFORMED in the classroom.

Standards

- Correctly calculates radiation exposure.

Initiating Cues

- You are to answer the radiation control question, placing your response on the paper containing the question.
- When you are finished responding to the questions, inform an examiner.

Terminating Cues

- Applicant informs the examiner that the answer to the question has been completed.

Tools and Equipment

- Calculator
- Survey Data

Safety Considerations

- None

Initial Conditions:

- Unit 2 is in a refueling outage and leakage out of the 'B' SI Accumulator has been noted.
- The operating team has attempted to close 2-SI-MOV-2865B; the valve cannot be closed from the MCR and needs to be locally closed.
- You have been tasked with entering containment and locally closing 2-SI-MOV-2865B and then exiting containment.
- HP has reported that there is airborne contamination in the area of 2-SI-MOV-2865B and have asked that the use of SCBAs be considered, although they have not required it.
- HP has stated that if respirators are **NOT** worn, the internal dose rate while manipulating 2-SI-MOV-2865B is **60 mRem/hr**; this dose rate is in addition to the radiation field (external dose rate = 70 mRem/hr) where the valve is located.
- HP has also stated that the average dose rate from the containment entry point to the valve is 20 mRem/hr and there is no airborne contamination while traveling to and from 2-SI-MOV-2865B.
- A second operator is available to aid you in performance of this task, if you deem it necessary.
- HP personnel are currently unavailable to provide assistance for dose determination.

Initiating Cues

Given the travel times, times to perform the task and dose rates on the attached table, calculate which one of the following will result in the lowest total dose for this task (Travel to and from the valve and close the valve)?

a. One operator wearing an SCBA

b. Two operators not wearing an SCBA

Notes

Times to perform the task:

Two way travel time to 2-SI-MOV-2865B without a SCBA	12 minutes
Two way travel time to 2-SI-MOV-2865B with a SCBA	16 minutes
Time for two people without SCBAs to close 2-SI-MOV-2865B	5 minutes
Time for one person with a SCBA to close 2-SI-MOV-2865B	16 minutes

Dose rates for the Areas

Average Dose Rate while traveling to and from 2-SI-MOV-2865B	20 mRem/hr
Average External Dose Rate while closing 2-SI-MOV-2865B	70 mRem/hr
Average Internal Dose Rate while closing 2-SI-MOV-2865B (if not wearing SCBA)	60 mRem/hr

PERFORMANCE CHECKLIST

Notes to the Evaluator

- Task critical elements are **bolded**.
- **START TIME:**

Steps can be performed in any order **STEP 1** APPLICANT CALCULATES THE TOTAL EXPOSURE FROM TRAVELING TO AND FROM 2-SI-MOV-2865B. **STANDARD:** **a) One operator, <u>with</u> SCBA** **Total travel time is 16 minutes.** **Average Dose Rate during travel is 20 mRem/hr.** **(20 mRem/hr) x (1 hr / 60 minutes) x (16 minutes) = 5.33 mRem [5 – 6 mRem]** **Total Travel Time Dose: <u>5.33 mRem</u>** b) Two operators, **NO** SCBAs Total travel time is 12 minutes. Average Dose Rate during travel is 20 mRem/hr. (20 mRem/hr) x (1 hr / 60 minutes) x (12 minutes) = 4 mRem (2 operators) x (4 mRem) = 8 mRem Total Travel Time Dose: <u>8 mRem</u> **COMMENTS:**	_______ **SAT** _______ **UNSAT**

Steps can be performed in any order **STEP 2** APPLICANT CALCULATES THE TOTAL EXPOSURE FROM CLOSING 2-SI-MOV-2865B. **STANDARD:** **a) One operator, <u>with</u> SCBA** **Time to close the valve is 16 minutes.** **Average External Dose Rate 70 mRem/hr.** **(70 mRem/hr) x (1 hr / 60 minutes) x (16 minutes) = 18.67 mRem external.** **Total dose to close the valve: <u>18.67 mRem</u> [18 – 19 mRem]** b) Two operators, **NO** SCBAs Time to close the valve is 5 minutes. Average External Dose Rate 70 mRem/hr. Average Internal Dose Rate 60 mRem/hr. (70 mRem/hr) x (1 hr / 60 minutes) x (5 minutes) = 5.83 mRem external. (60 mRem/hr) x (1 hr / 60 minutes) x (5 minutes) = 5 mRem internal. (2 operators) x (5.83 mRem + 5 mRem) = 21.66 mRem Total dose to close the valve: <u>21.66 mRem</u> [21 – 22 mRem] **COMMENTS:**	_______ **SAT** _______ **UNSAT**

Steps can be performed in any order **STEP 3** APPLICANT DETERMINES TOTAL JOB DOSE. **STANDARD:** **a) One operator, <u>with</u> SCBA** **Total dose to close the valve: 18.67 mRem [18 – 19 mRem]** **Total Travel Time Dose: 5.33 mRem [5 – 6 mRem]** **Total Job Dose: <u>24 mRem</u> [23 – 25 mRem]** b) Two operators, **NO** SCBAs Total dose to close the valve: 21.66 mRem [21 –22 mRem] Total Travel Time Dose: 8 mRem Total Job Dose: <u>29.66 mRem</u> [29 – 30 mRem] **COMMENTS:**	_______ **SAT** _______ **UNSAT**
Steps can be performed in any order **STEP 4** TASK TEAM DETERMINATION. **STANDARD:** Applicant reports that one operator with a respirator is the team that will perform the task with the minimum total dose. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STOP TIME:

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions:

- Unit 2 is in a refueling outage and leakage out of the 'B' SI Accumulator has been noted.
- The operating team has attempted to close 2-SI-MOV-2865B; the valve cannot be closed from the MCR and needs to be locally closed.
- You have been tasked with entering containment and locally closing 2-SI-MOV-2865B and then exiting containment.
- HP has reported that there is airborne contamination in the area of 2-SI-MOV-2865B and have asked that the use of SCBAs be considered, although they have not required it.
- HP has stated that if respirators are **NOT** worn, the internal dose rate while manipulating 2-SI-MOV-2865B is **60 mRem/hr**; this dose rate is in addition to the radiation field (external dose rate = 70 mRem/hr) where the valve is located.
- HP has also stated that the average dose rate from the containment entry point to the valve is 20 mRem/hr and there is no airborne contamination while traveling to and from 2-SI-MOV-2865B.
- A second operator is available to aid you in performance of this task, if you deem it necessary.
- HP personnel are currently unavailable to provide assistance for dose determination.

Initiating Cues

Given the travel times, times to perform the task and dose rates on the attached table, calculate which one of the following will result in the lowest total dose for this task (Travel to and from the valve and close the valve)?

a. One operator wearing an SCBA

b. Two operators not wearing an SCBA

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)
(Page 1 of 2)

Initial Conditions:

- Unit 2 is in a refueling outage and leakage out of the 'B' SI Accumulator has been noted.
- The operating team has attempted to close 2-SI-MOV-2865B; the valve cannot be closed from the MCR and needs to be locally closed.
- You have been tasked with entering containment and locally closing 2-SI-MOV-2865B and then exiting containment.
- HP has reported that there is airborne contamination in the area of 2-SI-MOV-2865B and have asked that the use of SCBAs be considered, although they have not required it.
- HP has stated that if respirators are **NOT** worn, the internal dose rate while manipulating 2-SI-MOV-2865B is **60 mRem/hr**; this dose rate is in addition to the radiation field (external dose rate = 70 mRem/hr) where the valve is located.
- HP has also stated that the average dose rate from the containment entry point to the valve is 20 mRem/hr and there is no airborne contamination while traveling to and from 2-SI-MOV-2865B.
- A second operator is available to aid you in performance of this task, if you deem it necessary.
- HP personnel are currently unavailable to provide assistance for dose determination.

Initiating Cues

Given the travel times, times to perform the task and dose rates on the attached table, calculate which one of the following will result in the lowest total dose for this task (Travel to and from the valve and close the valve)?

a. One operator wearing an SCBA

b. Two operators not wearing an SCBA

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Times to perform the task:

Two way travel time to 2-SI-MOV-2865B without a SCBA	12 minutes
Two way travel time to 2-SI-MOV-2865B with a SCBA	16 minutes
Time for two people without SCBAs to close 2-SI-MOV-2865B	5 minutes
Time for one person with a SCBA to close 2-SI-MOV-2865B	16 minutes

Dose rates for the Areas

Average Dose Rate from while traveling to and from 2-SI-MOV-2865B	20 mRem/hr
Average External Dose Rate while closing 2-SI-MOV-2865B	70 mRem/hr
Average Internal Dose Rate while closing 2-SI-MOV-2865B (if not wearing SCBA)	60 mRem/hr

U.S. Nuclear Regulatory Commission
Surry Power Station

SR2014301
Administrative Job Performance Measure G2.4.39
TIME CRITICAL

Applicant______________________________ Start Time__________________

Examiner______________________________

Date __________________________________ Stop Time__________________

Title

Complete Report of Emergency to State and Local Governments Form for SEM approval.

K/A: G2.4.39 – Knowledge of RO responsibilities in emergency plan implementation. (3.8/3.7)

Applicability	**Estimated Time**	**Actual Time**
RO/SRO	12 Minutes **(Time Critical)**	

Conditions

- Task may be PERFORMED in classroom or Simulator.
- A simulated ALERT is in progress.

Standards

- Completes EPIP-2.01, Notification of State and Local Governments steps 1-5 and Report of Emergency to State and Local Governments Form for SEM approval.

Initiating Cues

- Nuclear Shift Manager direction.

Terminating Cues

- EPIP-2.01 steps 1-5 and Report of Emergency to State and Local Governments Form complete.

Procedures

- EPIP-2.01, Revision 44
- Report of Emergency to State and Local Governments Form (730860)

Tools and Equipment

- None

Safety Considerations

- None

Initial Conditions

- An Alert was just declared due to a SBLOCA on Unit 1. (*Use current time & date*)
- The EAL identifier is FA1.1.
- PCS is inoperable.

Initiating Cues

- **This JPM is Time Critical.**
- I am the Station Emergency Manager. An Alert was just declared due to a SBLOCA on Unit 1 at *(use current time & date*). There are no radiological releases and plant access is available.
- You are the State and Local Emergency Communicator. You are to perform EPIP-2.01, steps 1 through 5 and fill out a Report of Emergency to State and Local Governments.
- When you finish filling out the Report of Emergency to State and Local Governments Form, give it to me for approval.

Notes

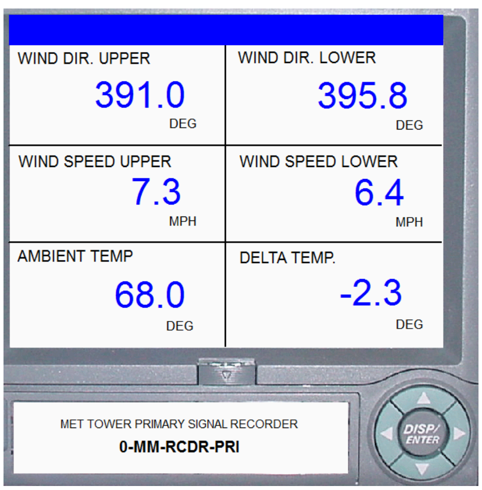
PERFORMANCE CHECKLIST

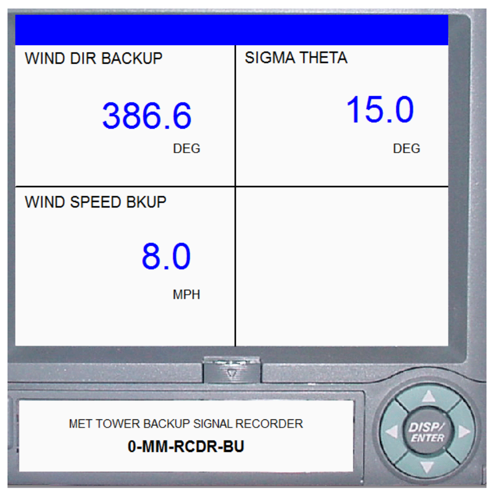
Simulator Set-up

- Enter the following Meter Overrides for MET PANEL

Recorder Name	Variable Name	Initial Value	Final Value	Recorder Reading
Wind Dir Upper	MET_WIND_DIR_UPR	0.5185185	.724	391.0°
Wind Dir Lower	MET_WIND_DIR_LWR	0.5185185	.733	395.8°
Wind Speed Upper	MET_WIND_SPD_UPR	0.1	0.073	7.3 mph
Wind Speed Lower	MET_WIND_SPD_LWR	0.1	0.064	6.4 mph
Ambient Temp.	MET_TEMP	0.651852	0.6	68°F
Delta Temp	MET_DELTAT	0.2222222	0.185	-2.3°F
Wind Dir Backup	MET_WIND_DIR_BKP	0.5185185	0.716	386.6°
Wind Spd Backup	MET_WIND_SPD_BKP	0.1	0.08	8 mph
Sigma Theta	MET_WIND– MET_SIGMA_THETA	0.46	0.3	15.0°

When complete the Recorders should look as shown below.





Notes to the Evaluator.

- Task critical elements are **bolded**.
- **START TIME: __________.**

STEP 1 Initiates Procedure. (*EPIP-2.01*, *Step 1*) **STANDARD:** Applicant fills out Name, Date, Time, and Location. **EVALUATOR'S NOTE:** **If asked**: Have Applicant use current date and time. Location should be Surry Power Station (or similar). JPM Start time is EAL declaration time. **COMMENTS:**	**_______ SAT** **_______ UNSAT**
STEP 2 CHECK FIRST REPORT OF EMERGENCY FOR EVENT – REQUIRED (*EPIP-2.01, Step 2*) **STANDARD:** Determines that this is the initial report of the event. **EVALUATOR'S NOTE:** **If asked**: Inform Applicant that this is the first report for the event. **COMMENTS:**	**_______ SAT** **_______ UNSAT**

STEP 3 Reviews NOTES before *Step 3*. **STANDARD:** a) Identifies that the state and local EOCs must be notified within 15 minutes of declaration. b) Identifies that Attachment 1 may be referenced as necessary to help fill out the report. c) Identifies that steps 7 through 11 are optional for the initial report, any changes in the emergency class, or PAR changes. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 4 CHECK EMERGENCY – REMAINS IN EFFECT. (*EPIP 2.01, Step 3*) **STANDARD:** Applicant determines that the emergency is still in effect. **EVALUATOR'S NOTE:** **If asked**: The emergency is still on-going. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 5 RECORD INFORMATION ON REPORT OF EMERGENCY TO STATE AND LOCAL GOVERNMENTS (*EPIP-2.01, Step 4*) **STANDARD:** a) Obtains Report of Emergency to State and Local Governments Form. b) Applicant may use EPIP-2.01, Attachment 1 to assist in filling out report. **EVALUATOR'S NOTE:** Report of Emergency to State and Local Governments Form is provided. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 6

Fill out Report of Emergency to State and Local Governments Form.

_______ **SAT**

_______ **UNSAT**

STANDARD:

The applicant will:

a) Place a "1" in the ROE Message blank.
b) Check "Drill Message" block.
c) For Item 1 (classification):
- **Check "Alert" block.**
- **Record FA1.1 in EAL.**
- **Record JPM start time/date as "Declared" time/date.**
- **Record Name.**

d) For Item 2 (meteorological data):
- **Check the "on-site measurements block.**
- **Records time data was obtained**
- **Records AVE wind speed and direction from available MET instruments.**
- **AVE Wind Speed (Lower) = 6.4 mph**
- **AVE Wind Direction (Lower) = 35.8° *(395.8° - 360°)***

e) For Item 3 (releases), check the "A" block for no radiological releases.
f) For Item 4 (site access), check the "Available" block.
g) For Item 5 (PAR), check the "is NOT required" block.
h) For Item 6 (updates), check the "60 minute" block.
i) Reviews and acknowledges NOTE that Items 7 through 11 may be excluded from message.
j) Checks "Excluded from message" blocks on Items 7 through 11.
k) Hands form to SEM (Evaluator) for approval.

EVALUATOR'S NOTE:

- **If asked**: This is a Drill Message. Applicant may check "Emergency Message" block if the evaluator is not asked. This is not a critical task.
- From EPIP-2.01, Attachment 1:

  NOTE: WHEN wind direction indicates GREATER THAN 360°, THEN subtract 360° from the indication to obtain net wind direction.

  Multiple indications of wind direction and wind speed are available. The priority using these indications is:

  1 Main Tower Lower Level

  2 Backup Tower

  3 Main Tower Upper Level

- **If asked**: There are no radiological releases in progress.
- **If asked**: Site access is available.
- Applicant may fill out "date" blanks pre-emptively for the notification time/date.

COMMENTS:

STEP 7 HAVE SEM/RM APPROVE REPORT. *(EPIP-2.01, Step 5)* **STANDARD:** Applicant will hand Report form to SEM (Evaluator) for approval. **COMMENTS:** **STOP CRITICAL TASK TIME:__________** ** **JPM COMPLETE** **	_______ **SAT** _______ **UNSAT**

STOP TIME: ___________________________

Comments: ____

____

____

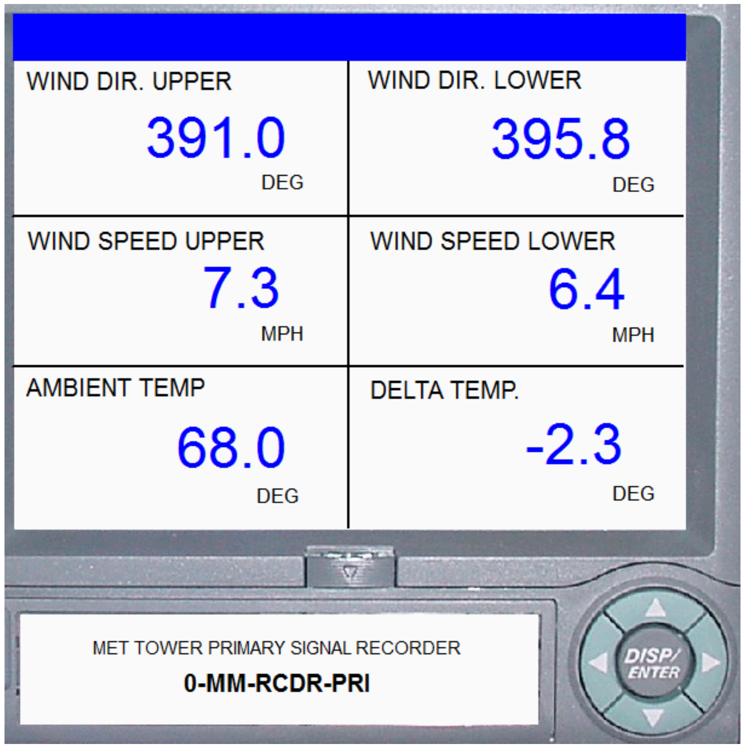
____

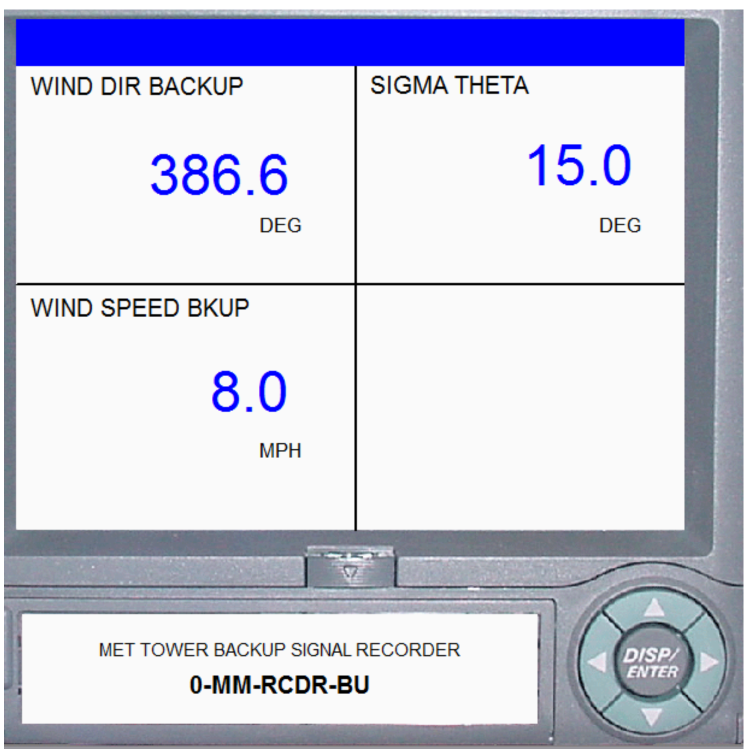
____

____

____

IF JPM PERFORMED IN CLASSROOM, PROVIDE THIS SHEET TO THE APPLICANT





Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- An Alert was just declared due to a SBLOCA on Unit 1. (*Use current time & date*)
- The EAL identifier is FA1.1.
- PCS is inoperable.

Initiating Cues

- **This JPM is Time Critical.**

- I am the Station Emergency Manager. An Alert was just declared due to a SBLOCA on Unit 1 at (*use current time & date*). There are no radiological releases and plant access is available.

- You are the State and Local Emergency Communicator. You are to perform EPIP-2.01, steps 1 through 5 and fill out a Report of Emergency to State and Local Governments.

- When you finish filling out the Report of Emergency to State and Local Governments Form, give it to me for approval.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- An Alert was just declared due to a SBLOCA on Unit 1. (*Use current time & date*)
- The EAL identifier is FA1.1.
- PCS is inoperable.

Initiating Cues

- **This JPM is Time Critical.**

- I am the Station Emergency Manager. An Alert was just declared due to a SBLOCA on Unit 1 at (*use current time & date*). There are no radiological releases and plant access is available.

- You are the State and Local Emergency Communicator. You are to perform EPIP-2.01, steps 1 through 5 and fill out a Report of Emergency to State and Local Governments.

- When you finish filling out the Report of Emergency to State and Local Governments Form, give it to me for approval.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Administrative Job Performance Measure G2.4.44

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

DETERMINE REQUIRED PAR ACTIONS

K/A: G2.4.44 – Knowledge of emergency plan protective action recommendations. (2.4/4.4)

Applicability	**Est Completion Time**	**Actual Time**
SRO ONLY	7 Minutes **(Time Critical)**	___________

Conditions

- Task is to be PERFORMED in the SIMULATOR.
- A simulated GENERAL EMERGENCY is in progress.

Standards

- EPIP-1.06, Protective Action Recommendations.

Initiating Cues

- EPIP-1.05, Response to General Emergency, Step 5.

Terminating Cues

- EPIP-1.06, Step 6 Completed.

Procedures

- EPIP-1.06, Protective Action Recommendations, Revision 10.

Tools and Equipment

- None

Safety Considerations

- None

Performance Checklist

Directions to the Operator.

- **This JPM is TIME CRITICAL.**
- You are the Nuclear Shift Manager.
- A General Emergency based on EAL SG1.1 (Loss of all offsite and onsite power to Unit 1 Emergency Busses H and J) has just been declared (*use current date/time*).
- Here's a copy of EPIP-1.06, Protective Action Recommendations. You are to complete EPIP-1.06, Protective Action Recommendations in its entirety.
- All radiation monitors indicate pre-event radiation levels.
- The State and Local Communicator has determined wind direction to be from 180° and wind speed to be 17 mph.
- When you finish the actions necessary to accomplish this, please inform me.

Notes to the Evaluator.

- Task critical elements are **bolded**.
- **TIME CRITICAL REQUIREMENT:**
  This PAR must be identified and relayed to Emergency Communicators within 15 minutes.
- **START TIME: ________________**(Also used for GE declaration time)

STEP 1 Initiates EPIP-1.06. (*Step 1*) **STANDARD:** a) Acknowledges NOTE before Step 1 that Attachments 4 and 5 may be used for reference. b) Initiates procedure by filling out name, date, and time. **EVALUATOR'S NOTE:** **If asked**: Have Applicant use current date and time. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 2 USE ATTACHMENT 1, PROTECTIVE ACTION RECOMMENDATION FLOWCHART SPS, TO DETERMINE INITIAL PAR. (*Step 2*) **STANDARD:** a) Applicant uses Attachment 1 (flowchart) to determine the initial PAR. b) *Known impediments make evacuation dangerous* = NO c) *Radiological release in progress or has occurred related to event* = NO d) **PAR = Evacuate: 2 mile 360° radius and 5 miles downwind.** **EVALUATOR'S NOTE:** **If asked:** There are no known impediments to site evacuation. **If asked**: There are no radiological releases in progress. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 3 IMPLEMENT ATTACHMENT 2, AFFECTED SECTOR(S) MAP. (*Step 3*) **STANDARD:** a) Acknowledges NOTE before Step 3 that Attachment 2 is used for EPIP-1.06 PARs only, not EPIP-4.07 PARs. **b) Applicant implements Attachment 2 to determine affected sector(s):** - **Acknowledges NOTE regarding rounding of wind direction up or down.** - **Records time data acquired.** - **Records wind direction from 180°.** - **Uses table to determine that the affected sectors are R, A, B and records on attachment.** - Marks the affected sectors on map using pen, pencil, highlighter, etc. c) Goes to Step 6. **EVALUATOR'S NOTE:** **If asked:** Meteorological data was acquired at START TIME of JPM. **If asked**: Wind direction is from 180° and wind speed is 17 mph (*provided in directions*). **COMMENTS**	_______ **SAT** _______ **UNSAT**

STEP 4 COMPLETE ATACHMENT 3, REPORT OF PROTECTIVE ACTION RECOMMENDATION. (*Step 6 and Attachment 3*) **STANDARD:** **Applicant completes Attachment 3:** − Records #1 in PAR MESSAGE space − Acknowledges NOTES to transmit PAR to Virginia EOC only using ARD or direct dial. Only use Insta-Phone if all other methods of contacting VEOC are non-functional. − Places check mark in “Drill Message” box**.** − **Places check mark in “EVACUATE” box. Fills in <u>2</u> Mile radius 360° and <u>5</u> miles downwind in the following sectors: <u>R, A, B</u>.** − Acknowledges NOTE in REMARKS block regarding Shelter-in-Place recommendations. − Signs for approval to transmit. Uses current date and time. **EVALUATOR’S NOTE:** **If asked:** Virginia EOC ARD is functional. **If asked**: This is a Drill Message. Failure to sign/date for approval would require a follow-up question. This step must be complete within 15 minutes of start of task. **Record Time: ______** **COMMENTS**	_______ **SAT** _______ **UNSAT**

STEP 5 HAVE EMERGENCY COMMUNICATORS NOTIFY OFF-SITE AUTHORITIES OF PAR. (*Step 7*) **STANDARD:** **Applicant directs Emergency Communicators:** - **State and Local Communicator to notify Virginia EOC.** - **NRC Communicator to notify NRC IAW EPIP-2.02.** **EVALUATOR'S NOTE:** **Tell SRO**: State and Local EC will transmit PAR. **Tell SRO**: NRC EC will notify NRC of PAR. **If asked**: HPN has not been requested yet. This step must be complete within 15 minutes of start of task. **Record Time: ______** **COMMENTS**	_______ **SAT** _______ **UNSAT**
STEP 6 CHECK IF CURRENT PROTECTIVE ACTION RECOMMENDATION - INITIAL PAR. (*Step 8*) **STANDARD:** Applicant determines that the current PAR is the initial PAR (no revised PAR from HP). **EVALUATOR'S NOTE:** **If asked**: There is no revised PAR. **COMMENTS**	_______ **SAT** _______ **UNSAT**

STEP 7 HAVE RADIOLOGICAL ASSESSMENT DIRECTOR (RAD) IMPLEMENT EPIP-4.07, PROTECTIVEMEASURES [RADIOLOGICAL ASSESSMENT COORDINATOR (RAC) IF IN LEOF/CEOF]. (*Step 9*) **STANDARD:** Applicant directs RAD/RAC to implement EPIP-4.07, Protective Measures. **COMMENTS**	_______ **SAT** _______ **UNSAT**
STEP 8 CHECKS FOR REVISED PAR OR WIND SHIFT. (*Step 10*) **STANDARD:** d) Acknowledges CAUTION before Step 10 stating that previously issued PARs should not be reduced until threat fully under control and VEAC has been consulted. e) Acknowledges NOTE before Step 10 that a “Shelter-in-Place” recommendation may supersede a “Radiological Evacuation” recommendation based on known impediments. f) Applicant checks the following: - Revised PAR provided by RAD/RAC = NO - Wind shift = NO - PAR in effect – UNCHANGED = YES g) Goes to Step 11. **EVALUATOR’S NOTE:** **If asked**: The RAD/RAC does not recommend a revised PAR. **If asked**: Wind direction remains the same. **COMMENTS**	_______ **SAT** _______ **UNSAT**

STEP 9 CHECK EMERGENCY - TERMINATED. (*Step 11*) **STANDARD:** a) Applicant checks emergency terminated. − Asks if emergency terminated. − Asks if RAD/RAC have any further recommendations. b) Determines that procedure is complete until conditions change. **EVALUATOR'S NOTE:** **If asked**: Emergency is still in effect. **If asked:** RAD/RAC does not recommend a PAR change at this time. **COMMENTS** ****END JPM****	_______ **SAT** _______ **UNSAT**

Stop Time:__________

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- Task is to be PERFORMED in the Simulator.

Initiating Cues

- **This JPM is TIME CRITICAL.**
- You are the Nuclear Shift Manager.
- A General Emergency based on EAL SG1.1 (Loss of all offsite and onsite power to Unit 1 Emergency Busses H and J) has just been declared (*use current date/time*).
- Here's a copy of EPIP-1.06, Protective Action Recommendations. You are to complete EPIP-1.06, Protective Action Recommendations in its entirety.
- All radiation monitors indicate pre-event radiation levels.
- The State and Local Communicator has determined wind direction to be from 180° and wind speed to be 17 mph.
- When you finish the actions necessary to accomplish this, please inform me.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

Initial Conditions

- Task is to be PERFORMED in the Simulator.
- A simulated GENERAL EMERGENCY is in progress.

Initiating Cues

- **This JPM is TIME CRITICAL.**
- You are the Nuclear Shift Manager.
- A General Emergency based on EAL SG1.1 (Loss of all offsite and onsite power to Unit 1 Emergency Busses H and J) has just been declared (*use current date/time*).
- Here's a copy of EPIP-1.06, Protective Action Recommendations. You are to complete EPIP-1.06, Protective Action Recommendations in its entirety.
- All radiation monitors indicate pre-event radiation levels.
- The State and Local Communicator has determined wind direction to be from 180° and wind speed to be 17 mph.
- When you finish the actions necessary to accomplish this, please inform me.

U.S. Nuclear Regulatory Commission
Surry Power Station

SR16301
Administrative Job Performance Measure G2.4.47

Applicant________________________________ Start Time___________________

Examiner_______________________________

Date ___________________________________ Stop Time___________________

Title

DETERMINE RCS LEAK RATE.

K/A: G2.4.47 – Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (4.2/4.2)

Applicability	**Est Completion Time**	**Actual Time**
RO/SRO(I)/SRO(U)	30 Minutes	___________

Conditions

- Task is to be PERFORMED in the CLASSROOM.

Standards

- Determine RCS leak rate based on PCS trace.

Initiating Cues

- Shift Manager direction.

Terminating Cues

- The Applicant states that the calculation of RCS leak rate is complete.

Procedures

- 1-OPT-RC-10.01, Reactor Coolant Leakage – Manually Calculated, Revision 19. Attachments 3 and 4.
- Surry Technical Specifications

Tools and Equipment

- Calculator

Safety Considerations

- None

Performance Checklist

Directions to the Operator.

- I am the Nuclear Shift Manager and you are the Unit 1 BOP Operator.
- A few minutes ago, the Unit 1 RO noted a change in the VCT level trend as indicated on the PCS.
- Here is copy of the VCT level trace from the PCS (Attachment 1).
  - At 0515, VCT Level was 34.1%
  - At 0520, VCT Level was 27.1%
- All other Unit parameters are stable and normal for 100% power operation.
- Here is a copy of 1-OPT-RC-10.01, Reactor Coolant Leakage – Manually Calculated, Attachments 3 and 4.
- You are to calculate the RCS leak rate based on the change in VCT level. Only those portions of the Attachments that specifically address the change in VCT level need to be completed.
- When you have completed your determination of the leak rate, please inform me of the value in gallons per minute.

FOR SRO CANDIDATES ONLY:

- You are to evaluate Technical Specifications regarding calculated RCS leakage.
- When you have determined the applicable Limiting Condition for Operation (if any), please inform me.

Notes to the Evaluator.

- Task critical elements are **bolded**.
- **START TIME: _______________**

STEP 1: DETERMINES CHANGE IN VCT LEVEL. **STANDARD:** • Records final and initial values for VCT level on Attachment 3. **COMMENTS:**	_______ **SAT** _______ **UNSAT**

STEP 2: CONVERTS % CHANGE IN LEVEL OVER TIME TO LBM/MINUTE. **STANDARD:** • **Multiplies the percent change in VCT level by the conversion factor listed on Attachment 4.** – **7% x 116.6 lbm/% = 816.2 lbm.** (allow for rounding) • **Divides the calculated value by the elapsed time for the change.** – **816.2 lbm / 5 minutes = 163.24 lbm/min** (allow for rounding) **EVALUATOR'S NOTE:** **If asked:** Identified leakage is 0 gpm. **COMMENTS:**	_______ **SAT** _______ **UNSAT**
STEP 3. CONVERTS THE LEAK RATE TO GPM BY MULTIPLYING THE CONVERSION FACTOR LISTED IN TABLE 5 OF ATTACHMENT 4. **STANDARD:** • **Multiplies the lbm/min value obtained in previous step by the conversion factor listed in Table 5.** – **(7.48052 gal/ft^3 ÷ 61.856 lbm/ft^3) x 163.24 lbm/min = 19.7 gpm** (allow for rounding) **EVALUATOR'S NOTE:** **If asked:** Identified leakage is 0 gpm. **COMMENTS**	_______ **SAT** _______ **UNSAT**

STEP 4. INFORMS SHIFT MANAGER OF THE CALCULATED LEAK RATE. **STANDARD:** • Informs Shift Manager that the calculated RCS leak rate based on the change in VCT level is 19.7 gpm **EVALUATOR'S NOTE:** An answer in the band of 19.0 to 20.0 gpm is acceptable based on rounding tolerances. **COMMENTS** ****END OF JPM****	_______ **SAT** _______ **UNSAT**

Stop Time:__________

Comments

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Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

SRO Only Question

- The leakage has been identified as coming from 1-RC-PCV-1455B, "C" Loop Spray Valve packing.
- You are to evaluate Technical Specifications regarding calculated RCS leakage.
- When you have determined the applicable Limiting Condition for Operation (if any), please inform me.

STANDARD:

- Applicant evaluates TS-3.1.C, RCS Operational LEAKAGE:
  - **Applicant notes that the calculated 19.7 gpm exceeds the allowable 10 gpm Identified leakage criteria per TS-3.1.C.1.c.**
  - **Since it has been determined to NOT be pressure boundary leakage, TS-3.1.C.2a is applicable -** If RCS operational LEAKAGE is not within the limits of 3.1.C.1 for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within the specified limits within 4 hours.
  - **Applicant determines that TS-3.1.C.2b is applicable -** If the LEAKAGE is not reduced to within the specified limits within 4 hours, the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

EVALUATOR'S NOTE:

- **Provide Candidate with a copy of Surry Technical Specifications.**

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

SRO Only Question

- The leakage has been identified as coming from 1-RC-PCV-1455B, “C” Loop Spray Valve packing.
- You are to evaluate Technical Specifications regarding calculated RCS leakage.
- When you have determined the applicable Limiting Condition for Operation (if any), please inform me.

Operator Directions Handout
(TO BE READ TO APPLICANT BY EXAMINER)

Initial Conditions

- Unit 1 is at 100% power.

Initiating Cues

- Shift Manager direction.

Directions to the Operator

- I am the Nuclear Shift Manager and you are the Unit 1 BOP Operator.
- A few minutes ago, the Unit 1 RO noted a change in the VCT level trend as indicated on the PCS.
- Here is copy of the VCT level trace from the PCS (Attachment 1).
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- You are to calculate the RCS leak rate based on the change in VCT level. Only those portions of the Attachments that specifically address the change in VCT level need to be completed.
- When you have completed your determination of the leak rate, please inform me of the value in gallons per minute.

Operator Directions Handout
(TO BE GIVEN TO APPLICANT)

<u>Initial Conditions</u>

- Unit 1 is at 100% power.

<u>Directions to the Operator</u>

- I am the Nuclear Shift Manager and you are the Unit 1 BOP Operator.
- A few minutes ago, the Unit 1 RO noted a change in the VCT level trend as indicated on the PCS.
- Here is copy of the VCT level trace from the PCS (Attachment 1).
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- When you have completed your determination of the leak rate, please inform me of the value in gallons per minute.

Attachment 1